DOE/EIS-0542
December 2020

Draft Versatile Test Reactor Environmental Impact Statement

Volume 2

Appendices A-H

U.S. Department of Energy
Office of Nuclear Energy

COVER SHEET

Lead Agency: U.S. Department of Energy (DOE)

Cooperating Agencies: None

Title: *Draft Versatile Test Reactor Environmental Impact Statement* (VTR EIS) (DOE/EIS-0542)

Location: Idaho, South Carolina, Tennessee

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This document is available on the DOE NEPA website (http://energy.gov/nepa/nepa-documents) for viewing and downloading.

Abstract: This *Versatile Test Reactor Environmental Impact Statement* (VTR EIS) evaluates the potential environmental impacts of proposed alternatives for the construction and operation of a new test reactor, as well as associated facilities that are needed for performing post-irradiation evaluation of test articles and managing spent nuclear fuel (SNF). In accordance with the Nuclear Energy Innovation Capabilities Act of 2017 (NEICA) (Pub. L. 115–248), DOE assessed the mission need for a versatile reactor-based fast-neutron source (or Versatile Test Reactor) to serve as a national user facility. DOE determined that there is a need for a fast-neutron spectrum VTR to enable testing and evaluating nuclear fuels, materials, sensors, and instrumentation for use in advanced reactors and other purposes. In accordance with NEICA, DOE is pursuing construction and operation of the 300 megawatt (thermal) VTR. The reactor would be a pool-type, sodium-cooled reactor that uses a uranium-plutonium-zirconium metal fuel. The analysis also includes the potential impacts from post-irradiation examination of test articles, management of spent fuel, and activities necessary for VTR driver fuel production.

The Idaho National Laboratory (INL) VTR Alternative would include the construction of the VTR adjacent to the Materials and Fuels Complex (MFC) at the INL Site. Existing MFC facilities, some requiring new equipment, would be used for post-irradiation examination and conditioning SNF. The Oak Ridge National Laboratory (ORNL) VTR Alternative would include the construction of a VTR and a hot cell building at ORNL. The hot cell building would provide post-irradiation examination and SNF conditioning capabilities. Both alternatives would require construction of a concrete pad for dry storage of SNF pending shipment

to an offsite storage or disposal facility. DOE does not intend to separate, purify, or recover fissile material from VTR driver fuel.

DOE also evaluates options for preparing the uranium/plutonium/zirconium feedstock for use in the reactor driver fuel (fuel needed to run the reactor) and for fabricating the driver fuel. Feedstock preparation would be performed using new capabilities installed in an existing building at the INL Site or the Savannah River Site (SRS). Fuel fabrication would be performed using existing or newly installed equipment in existing buildings at the INL Site or SRS.

Preferred Alternative: DOE's Preferred Alternative is the INL VTR Alternative. DOE would construct and operate the VTR at the INL Site adjacent to the MFC. Existing facilities within the MFC would be modified and used for post-irradiation examination of test assemblies. SNF would be treated to remove the sodium and converted into a form that would meet the acceptance criteria for a future permanent repository. The treated SNF would be temporarily stored at a new storage pad near the VTR.

DOE has no preferred option at this time for where it would perform reactor fuel production (feedstock preparation or driver fuel fabrication) for the VTR. This EIS evaluates options for both processes at the INL Site and at SRS. DOE will state its preferred options for feedstock preparation and driver fuel fabrication in the Final VTR EIS, if preferred options are identified before issuance.

Public Involvement: DOE issued a Notice of Intent to Prepare an environmental impact statement for a Versatile Test Reactor in the *Federal Register* (84 FR 38021) on August 5, 2019, to solicit public input on the scope and environmental issues to be addressed in this VTR EIS. Comments received during the August 5 through September 4, 2019, scoping period were considered in the preparation of this Draft EIS. Comments on this Draft EIS will be accepted following publication of the U.S. Environmental Protection Agency Notice of Availability. Comments can be submitted to the address provided above or emailed to VTR.EIS@nuclear.energy.gov. Opportunities to provide oral comments will be announced in news media near the DOE sites at a later date. Comments received during the comment period will be considered during the preparation of the Final EIS. Comments received after the close of the comment period will be considered to the extent practicable.

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ACRONYMS, ABBREVIATIONS, AND CONVERSION CHARTS

ACRONYMS, ABBREVIATIONS, AND CONVERSION CHARTS

°C	degrees Celsius
°F	degrees Fahrenheit
AEG	Acute Exposure Guideline Level
ALARA	as low as reasonably achievable
ANS	Advanced Neutron Source
AoA	Analysis of Alternatives
APC	Air Pollution Control
AQD	Air Quality Division
APE	area of potential effects
ARF	airborne release fraction
ATR	Advanced Test Reactor
B_4C	boron carbide
BCCs	Birds of Conservation Concern
BCR	Bird Conservation Region
BEIR	Biological Effects of Ionizing Radiation
BGEPA	Bald and Golden Eagle Protection Act
BJWSA	Beaufort-Jasper Water and Sewer Authority
BLM	Bureau of Land Management
CAA	Clean Air Act
CAIRS	Computerized Accident Incident Reporting System
CBCG	Columbia Basin Consulting Group
CCOs	criticality control overpacks
CD	Critical Decision
CEM	continuous emission monitoring
CEQ	Council on Environmental Quality
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFA	Central Facilities Area
CFR	*Code of Federal Regulations*
CO	carbon monoxide
CO_2	carbon dioxide
CO_2e	carbon dioxide equivalent
CRBR	Clinch River Breeder Reactor
CSWTF	Central Sanitary Wastewater Treatment Facility
CWA	Clean Water Act
D&D	decontamination and decommissioning
D&R	dismantlement and removal
DART	Days Away, Restricted or on-the-job Transfer
dB	decibels
dBA	A-Weighted Decibel Scale
DHS	U.S. Department of Homeland Security
DNFSB	Defense Nuclear Facilities Safety Board
DOD	U.S. Department of Defense

DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DR	damage ratio
DSAs	documented safety analyses
EA	environmental assessment
EBR	Experimental Breeder Reactor
EDGs	emergency diesel generators
EFF	Experimental Fuels Facility
EIS	environmental impact statement
EM	electromagnetic
EPA	U.S. Environmental Protection Agency
EPHA	Emergency Planning Hazard Assessment
ERDA	U.S. Energy Research and Development Administration
ERO	Emergency Response Organization
ERPG	Emergency Response Planning Guideline
ESA	Endangered Species Act
ESER	Environmental Surveillance, Education, and Research
ESRP	Eastern Snake River Plain
ETTP	East Tennessee Technology Park
FC	frequency-consequence
FCF	Fuel Conditioning Facility
FFCA	Federal Facilities Compliance Act
FFTF	Fast Flux Test Facility
FMF	Fuel Manufacturing Facility
FONSI	Finding of No Significant Impact
FOTAs	Fuels Open Test Assemblies
FR	*Federal Register*
FRR	Foreign Research Reactor
FTC	fuel transfer cask
FTE	full-time equivalent (person)
GEH	GE Hitachi Nuclear Energy
GHG	greenhouse gas
gpm	gallons per minute
GWP	global warming potential
HACs	hazardous air contaminants
HALEU	high assay, low-enriched uranium
HAPs	hazardous air pollutants
HAZMAT	hazardous materials
HEPA	high-efficiency particulate air
HEU	highly enriched uranium
HFEF	Hot Fuel Examination Facility
HFIR	High Flux Isotope Reactor
HLW	high-level radioactive waste
HRS	heat removal system
HT-9	a stainless-steel alloy of iron, chromium, molybdenum, tungsten, nickel, and carbon
HVAC	heating, ventilation, and air conditioning

IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IDA	International Dark-Sky Association
IDAPA	Idaho Administrative Procedures Act
IDEQ	Idaho Department of Environmental Quality
IDFG	Idaho Fish and Game
IHXs	internal heat exchangers
IMCL	Irradiated Materials Characterization Laboratory
INF	irradiated nuclear fuel
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
IPaC	Information for Planning and Consultation
IPDES	Idaho Pollutant Discharge Elimination System
ISA	Idaho Settlement Agreement
ISCORS	Interagency Steering Committee on Radiation Standards
ISO	International Organization for Standardization
IVTM	In-Vessel Transfer Machine
JFD	joint frequency distribution
KIS	K-Area Interim Surveillance
kV	kilovolt
L_{dn}	Day-Night Average Sound Level
L_{eg}	Equivalent Sound Level
LAMDA	Low Activation Materials Design and Analysis Laboratory
LANL	Los Alamos National Laboratory
LCF	latent cancer fatality
LEU	low-enriched uranium
LFR	lead-cooled fast reactor
LFTR	lead/lead-bismuth-cooled fast test reactor
LOOP	loss of offsite power
LOS	level of service
LLW	low-level radioactive waste
LPF	leak path factor
MACCS	MELCOR Accident Consequence Code System
Magnox	magnesium alloy
MAP	mitigation action plan
MAR	material at risk
MBTA	Migratory Bird Treaty Act
MCEP	Motor Carrier Evaluation Program
MCL	Maximum Contaminant Level
MDOR	Multicycle Direct Oxide Reduction
MEI	maximally exposed individual
MeV	million electron volts
MFC	Materials and Fuels Complex
MFFF	MOX Fuel Fabrication Facility
MHR	Multipurpose Haul Road
MLLW	mixed low-level radioactive waste

MOTAs	Materials Open Test Assemblies
MOX	mixed oxide
MSA	Material Storage Area
MSE	Molten Salt Extraction
MSFTR	molten-salt-cooled fast test reactor
MSR	molten salt reactors
MVa	megavolt-amperes
MW	megawatt
MWh	megawatt-hour
MWth	megawatts thermal
NAAQS	National Ambient Air Quality Standards
NASA	National Aeronautics and Space Administration
NCRP	National Council on Radiation Protection and Measurements
NDC	Natural Phenomena Hazards Design Category
NEAC	Nuclear Energy Advisory Committee
NEPA	National Environmental Policy Act
NEICA	Nuclear Energy Innovation Capabilities Act of 2017
NERP	National Environmental Research Park
NESHAP	National Emission Standards for Hazardous Air Pollutants
NFS	Nuclear Fuel Services, Inc.
NHPA	National Historic Preservation Act
NIW	noninvolved worker
NNSA	National Nuclear Security Administration
NNSS	Nevada National Security Site
NO_2	nitrogen dioxide
NOA	Notice of Availability
NOI	Notice of Intent
NOx	nitrogen oxides
NPDES	National Pollutant Discharge Elimination System
NPH	natural phenomena hazard
NPS	National Park Service
NRAD	neutron radiography
NRC	U.S. Nuclear Regulatory Commission
NRHP	National Register of Historic Places
NSUF	Nuclear Science User Facilities
O_3	ozone
ORR	Oak Ridge Reservation
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
PAC	protective action criteria
PC	Performance Category
PCBs	polychlorinated biphenyls
pCi/L	picocuries per liter
PCS	plant control systems
PDCF	Pit Disassembly and Conversion Facility
PEIS	Programmatic Environmental Impact Statement

PF-4	Plutonium Facility
PGA	peak ground acceleration
PHTS	Primary Heat Transport System
PIDAS	Perimeter Intrusion Detection and Assessment System
PLOHS	protected loss of heat sink
$PM_{2.5}$	particulate matter less than or equal to 2.5 microns in diameter
PM_{10}	particulate matter less than or equal to 10 microns in diameter
PMDA	Plutonium Management and Disposition Agreement
PNNL	Pacific Northwest National Laboratory
POCs	pipe overpack containers
ppm	parts per million
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module
PSBO	protected station blackout
PSD	Prevention of Significant Deterioration
psig	pounds per square in gauge
PTC	permit to construct
PTOP	protected transient overpower
R&D	research and development
RADTRAN	Radioactive Material Transportation Risk Assessment
RCRA	Resource Conservation and Recovery Act
REDC	Radiochemical Engineering Development Center
rem	roentgen equivalent man
RESL	Radiological and Environmental Sciences Laboratory
RF	respirable fraction
RH-TRU	remote-handled transuranic waste
RISKIND	Risks and Consequences of Radioactive Material Transport
ROD	Record of Decision
ROI	region of influence
RPS	Reactor Protection System
RVACS	Reactor Vessel Auxiliary Cooling System
RWMC	Radioactive Waste Management Complex
SA	Supplemental Analysis
SAHX	sodium-to-air heat exchanger
SBO	station blackout
SCDHEC	South Carolina Department of Health and Environmental Control
SCDNR	South Carolina Department of Natural Resources
SDA	Subsurface Disposal Area
SDC	seismic design category
SDWA	Safe Drinking Water Act
SGCN	Species of Greatest Conservation Need
SHPO	State Historic Preservation Officer
SIP	State Implementation Plan
SNF	spent nuclear fuel
SO_2	sulfur dioxide
SPD	Surplus Plutonium Disposition

SR	State Route
SRNL	Savannah River National Laboratory
SRNS	Savannah River Nuclear Solutions
SRPA	Snake River Plain Aquifer
SRR	Savannah River Remediation
SRS	Savannah River Site
SRTE	Savannah River Tritium Enterprise
STA	Secure Transportation Asset
SWAP	State Wildlife Action Plan
SWPPP	stormwater pollution prevention plan
SWPT	sanitary wastewater treatment plant
T&E	threatened and endangered
TAPs	toxic air pollutants
TDEC	Tennessee Department of Environment and Conservation
TEEL	Temporary Emergency Exposure Limit
TRC	Total Reportable Cases
TREAT	Transient Reactor Test
TRU	transuranic
TRUPACT-II	Transuranic Package Transporter Model 2
TSCA	Toxic Substance Control Act
TSP	total suspended particulates
TTHM	total trihalomethanes
TVA	Tennessee Valley Authority
TWPC	Transuranic Waste Processing Center
TWRA	Tennessee Wildlife Resources Agency
μg/L	micrograms per liter
U/Pu/Zr	uranium/plutonium/zirconium alloy
UK	United Kingdom
ULOF	unprotected (without scram) loss-of flow
ULOHS	unprotected loss-of-heat-sink
U.S.C.	*United States Code*
USDA	U.S. Department of Agriculture
USFS	U.S. Forest Service
USFWS	U.S. Fish and Wildlife Service
USGCRP	U.S. Global Change Research Program
USGS	U.S. Geological Survey
UTOP	unprotected transient overpower
VOC	volatile organic compound
VRM	Visual Resource Management
VTR	Versatile Test Reactor
WAG	Waste Area Group
WCS	Waste Control Specialists
WebTRAGIS	Web Transportation Routing Analysis Geographic Information System
WIPP	Waste Isolation Pilot Plant
Y-12	Y-12 National Security Complex
ZPPR	Zero Power Physics Reactor

CONVERSIONS

METRIC TO ENGLISH			ENGLISH TO METRIC		
Multiply	**by**	**To get**	**Multiply**	**by**	**To get**
Area					
Square meters	10.764	Square feet	Square feet	0.092903	Square meters
Square kilometers	247.1	Acres	Acres	0.0040469	Square kilometers
Square kilometers	0.3861	Square miles	Square miles	2.59	Square kilometers
Hectares	2.471	Acres	Acres	0.40469	Hectares
Concentration					
Kilograms/square meter	4.46	Tons/acre	Tons/acre	0.224	Kilograms/square meter
Milligrams/liter	1 [a]	Parts/million	Parts/million	1 [a]	Milligrams/liter
Micrograms/liter	1 [a]	Parts/billion	Parts/billion	1 [a]	Micrograms/liter
Micrograms/cubic meter	1 [a]	Parts/trillion	Parts/trillion	1 [a]	Micrograms/cubic meter
Density					
Grams/cubic centimeter	62.428	Pounds/cubic feet	Pounds/cubic feet	0.016018	Grams/cubic centimeter
Grams/cubic meter	0.0000624	Pounds/cubic feet	Pounds/cubic feet	16,018.5	Grams/cubic meter
Length					
Centimeters	0.3937	Inches	Inches	2.54	Centimeters
Meters	3.2808	Feet	Feet	0.3048	Meters
Kilometers	0.62137	Miles	Miles	1.6093	Kilometers
Radiation					
Sieverts	100	Rem	Rem	0.01	Sieverts
Temperature					
Absolute					
Degrees C + 17.78	1.8	Degrees F	Degrees F - 32	0.55556	Degrees C
Relative					
Degrees C	1.8	Degrees F	Degrees F	0.55556	Degrees C
Velocity/Rate					
Cubic meters/second	2118.9	Cubic feet/minute	Cubic feet/minute	0.00047195	Cubic meters/second
Grams/second	7.9366	Pounds/hour	Pounds/hour	0.126	Grams/second
Meters/second	2.237	Miles/hour	Miles/hour	0.44704	Meters/second
Volume					
Liters	0.26418	Gallons	Gallons	3.7854	Liters
Liters	0.035316	Cubic feet	Cubic feet	28.316	Liters
Liters	0.001308	Cubic yards	Cubic yards	764.54	Liters
Cubic meters	264.17	Gallons	Gallons	0.0037854	Cubic meters
Cubic meters	35.314	Cubic feet	Cubic feet	0.028317	Cubic meters
Cubic meters	1.3079	Cubic yards	Cubic yards	0.76456	Cubic meters
Cubic meters	0.0008107	Acre-feet	Acre-feet	1233.49	Cubic meters
Weight/Mass					
Grams	0.035274	Ounces	Ounces	28.35	Grams
Kilograms	2.2046	Pounds	Pounds	0.45359	Kilograms
Kilograms	0.0011023	Tons (short)	Tons (short)	907.18	Kilograms
Metric tons	1.1023	Tons (short)	Tons (short)	0.90718	Metric tons
ENGLISH TO ENGLISH					
Acre-feet	325,850.7	Gallons	Gallons	0.000003069	Acre-feet
Acres	43,560	Square feet	Square feet	0.000022957	Acres
Square miles	640	Acres	Acres	0.0015625	Square miles

a. This conversion is only valid for concentrations of contaminants (or other materials) in water.

METRIC PREFIXES

Prefix	Symbol	Multiplication factor
exa-	E	1,000,000,000,000,000,000 = 10^{18}
peta-	P	1,000,000,000,000,000 = 10^{15}
tera-	T	1,000,000,000,000 = 10^{12}
giga-	G	1,000,000,000 = 10^{9}
mega-	M	1,000,000 = 10^{6}
kilo-	k	1,000 = 10^{3}
deca-	D	10 = 10^{1}
deci-	d	0.1 = 10^{-1}
centi-	c	0.01 = 10^{-2}
milli-	m	0.001 = 10^{-3}
micro-	µ	0.000 001 = 10^{-6}
nano-	n	0.000 000 001 = 10^{-9}
pico-	p	0.000 000 000 001 = 10^{-12}

Appendix A
Federal Register Notices

APPENDIX A
FEDERAL REGISTER NOTICES

A.1 Notice of Intent – August 5, 2019

for hearing may be made through the Commission's web-based comment system, a link to which is provided at *www.drbc.gov.* Use of the web-based system ensures that all submissions are captured in a single location and their receipt is acknowledged. Exceptions to the use of this system are available based on need, by writing to the attention of the Commission Secretary, DRBC, P.O. Box 7360, 25 Cosey Road, West Trenton, NJ 08628–0360. For assistance, please contact Paula Schmitt at *paula.schmitt@drbc.gov.*

Accommodations for Special Needs. Individuals in need of an accommodation as provided for in the Americans with Disabilities Act who wish to attend the meeting or hearing should contact the Commission Secretary directly at 609–883–9500 ext. 203 or through the Telecommunications Relay Services (TRS) at 711, to discuss how we can accommodate your needs.

Additional Information, Contacts. Additional public records relating to hearing items may be examined at the Commission's offices by appointment by contacting Denise McHugh, 609–883–9500, ext. 240. For other questions concerning hearing items, please contact David Kovach, Project Review Section Manager at 609–883–9500, ext. 264.

Dated: July 29, 2019.

Pamela M. Bush,

Commission Secretary and Assistant General Counsel.

[FR Doc. 2019–16610 Filed 8–2–19; 8:45 am]

BILLING CODE 6360–01–P

DEPARTMENT OF EDUCATION

[Docket No.: ED–2019–ICCD–0094]

Agency Information Collection Activities; Comment Request; HEAL Program: Physician's Certification of Borrower's Total and Permanent Disability

AGENCY: Federal Student Aid (FSA), Department of Education (ED).

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995, ED is proposing an extension of an existing information collection.

DATES: Interested persons are invited to submit comments on or before October 4, 2019.

ADDRESSES: To access and review all the documents related to the information collection listed in this notice, please use *http://www.regulations.gov* by searching the Docket ID number ED–2019–ICCD–0094. Comments submitted in response to this notice should be submitted electronically through the Federal eRulemaking Portal at *http://www.regulations.gov* by selecting the Docket ID number or via postal mail, commercial delivery, or hand delivery. If the *regulations.gov* site is not available to the public for any reason, ED will temporarily accept comments at *ICDocketMgr@ed.gov.* Please include the docket ID number and the title of the information collection request when requesting documents or submitting comments. *Please note that comments submitted by fax or email and those submitted after the comment period will not be accepted.* Written requests for information or comments submitted by postal mail or delivery should be addressed to the Director of the Information Collection Clearance Division, U.S. Department of Education, 550 12th Street SW, PCP, Room 9086, Washington, DC 20202–0023.

FOR FURTHER INFORMATION CONTACT: For specific questions related to collection activities, please contact Beth Grebeldinger, 202–377–4018.

SUPPLEMENTARY INFORMATION: The Department of Education (ED), in accordance with the Paperwork Reduction Act of 1995 (PRA) (44 U.S.C. 3506(c)(2)(A)), provides the general public and Federal agencies with an opportunity to comment on proposed, revised, and continuing collections of information. This helps the Department assess the impact of its information collection requirements and minimize the public's reporting burden. It also helps the public understand the Department's information collection requirements and provide the requested data in the desired format. ED is soliciting comments on the proposed information collection request (ICR) that is described below. The Department of Education is especially interested in public comment addressing the following issues: (1) Is this collection necessary to the proper functions of the Department; (2) will this information be processed and used in a timely manner; (3) is the estimate of burden accurate; (4) how might the Department enhance the quality, utility, and clarity of the information to be collected; and (5) how might the Department minimize the burden of this collection on the respondents, including through the use of information technology. Please note that written comments received in response to this notice will be considered public records.

Title of Collection: HEAL Program: Physician's Certification of Borrower's Total and Permanent Disability.

OMB Control Number: 1845–0124.

Type of Review: An extension of an existing information collection.

Respondents/Affected Public: Individuals or Households; State, Local, and Tribal Governments.

Total Estimated Number of Annual Responses: 78.

Total Estimated Number of Annual Burden Hours: 20.

Abstract: This is a request for an extension of OMB approval of information collection requirements associated with the form for the Health Education Assistance Loan (HEAL) Program, Physician's Certification of Borrower's Total and Permanent Disability currently approved under OMB No. 1845–0124. The form is HEAL Form 539. A borrower and the borrower's physician must complete this form. The borrower then submits the form and additional information to the lending institution (or current holder of the loan) who in turn forwards the form and additional information to the Secretary for consideration of discharge of the borrower's HEAL loans. The form provides a uniform format for borrowers and lenders to use when submitting a disability claim.

Dated: July 31, 2019.

Kate Mullan,

PRA Coordinator, Information Collection Clearance Program, Information Management Branch, Office of the Chief Information Officer.

[FR Doc. 2019–16620 Filed 8–2–19; 8:45 am]

BILLING CODE 4000–01–P

DEPARTMENT OF ENERGY

Notice of Intent To Prepare an Environmental Impact Statement for a Versatile Test Reactor

AGENCY: Office of Nuclear Energy, Department of Energy.

ACTION: Notice of intent.

SUMMARY: As required by the ''Nuclear Energy Innovation Capabilities Act of 2017'' the Department of Energy (DOE) assessed the mission need for a versatile reactor-based fast-neutron source. Having identified the need for such a fast-neutron source, the Act directs DOE to complete construction and approve the start of facility operations, to the maximum extent practicable, by December 31, 2025. To this end, the Department intends to prepare an environmental impact statement (EIS) in accordance with the National Environmental Policy Act (NEPA) and its implementing regulations. This EIS will evaluate alternatives for a versatile reactor-based fast-neutron source facility and associated facilities for the

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preparation, irradiation and post-irradiation examination of test/experimental fuels and materials.

DATES: DOE invites public comment on the scope of this EIS during a 30-day public scoping period commencing August 5, 2019, and ending on September 4, 2019. DOE will hold webcast scoping meetings on August 27, 2019 at 6:00 p.m. ET/4:00 p.m. MT and on August 28, 2019 at 8:00 p.m. ET/6:00 p.m. MT.

In defining the scope of the EIS, DOE will consider all comments received or postmarked by the end of the scoping period. Comments received or postmarked after the scoping period end date will be considered to the extent practicable.

ADDRESSES: Written comments regarding the scope of this EIS should be sent to Mr. Gordon McClellan, Document Manager, by mail at: U.S. Department of Energy, Idaho Operations Office, 1955 Fremont Avenue, MS 1235, Idaho Falls, Idaho 83415; or by email to *VTR.EIS@nuclear.energy.gov*. To request further information about the EIS or to be placed on the EIS distribution list, you may use any of the methods listed in this section. In requesting to be added to the distribution list, please specify whether you would like to receive a copy of the Summary and Draft EIS on a compact disk (CD); a printed copy of the Summary and a CD with the Draft EIS; a full printed copy of the Summary and Draft EIS; or if you prefer to access the document via the internet. The Draft EIS and Summary will be available at: *https://www.energy.gov/nepa*.

FOR FURTHER INFORMATION CONTACT: For information regarding the Versatile Test Reactor (VTR) Project or the EIS, contact Mr. Gordon McClellan at the address given above; or email *VTR.EIS@nuclear.energy.gov*; or call (208) 526–6805. For general information on DOE's NEPA process, contact Mr. Jason Sturm at the address given above; or email *VTR.EIS@nuclear.energy.gov*; or call (208) 526–6805.

SUPPLEMENTARY INFORMATION:

Background

Part of the mission of DOE is to advance the energy, environmental, and nuclear security of the United States and promote scientific and technological innovation in support of that mission. DOE's 2014–2018 Strategic Plan states that DOE will "support a more economically competitive, environmentally responsible, secure and resilient U.S. energy infrastructure." Specifically, "DOE will continue to explore advanced concepts in nuclear energy that may lead to new types of reactors with further safety improvements and reduced environmental and nonproliferation concerns."

Many commercial organizations and universities are pursuing advanced nuclear energy fuels, materials, and reactor designs that complement the efforts of DOE and its laboratories in achieving DOE's goal of advancing nuclear energy. These designs include thermal and fast-spectrum [1] reactors targeting improved fuel resource utilization and waste management and utilizing materials other than water for cooling. Their development requires an adequate infrastructure for experimentation, testing, design evolution, and component qualification. Existing irradiation test capabilities are aging, and some are over 50 years old. The existing capabilities are focused on testing of materials, fuels, and components in the thermal neutron spectrum and do not have the ability to support the needs for fast reactors. Only limited fast-neutron-spectrum-testing capabilities, with restricted availability, exist outside the United States.

Recognizing that the United States does not have a dedicated fast-neutron-spectrum testing capability, DOE performed a mission needs assessment to assess current testing capabilities (domestic and foreign) against the required testing capabilities to support the development of advanced nuclear technologies. This needs assessment was consistent with the Nuclear Energy Innovation Capabilities Act of 2017, or NEICA, (Pub. L. 115–248) to assess the mission need for, and cost of, a versatile reactor-based fast-neutron source with a high neutron flux, irradiation flexibility, multiple experimental environment (*e.g.,* coolant) capabilities, and volume for many concurrent users. This assessment identified a gap between required testing needs and existing capabilities. That is, there currently is an inability to effectively test advanced nuclear fuels and materials in a fast-neutron spectrum irradiation environment at high neutron fluxes. Specifically, the DOE Office of Nuclear Energy (NE), Nuclear Energy Advisory Committee (NEAC) report, *Assessment of Missions and Requirements for a New U.S. Test Reactor,* confirmed that there was a need in the U.S. for fast-neutron testing capabilities, but that there is no facility that is readily available domestically or internationally. The NEAC study confirmed the conclusions of an earlier study, *Advanced Demonstration and Test Reactor Options Study*. That study established the strategic objective that DOE "provide an irradiation test reactor to support development and qualification of fuels, materials, and other important components/items (*e.g.,* control rods, instrumentation) of both thermal and fast neutron-based advanced reactor systems." To meet its obligation to support advanced reactor technology development, DOE needs to develop the capability for large-scale testing, accelerated testing, and qualification of advanced nuclear fuels, materials, instrumentation, and sensors. This testing capability is essential for the United States to modernize its nuclear energy infrastructure and for developing transformational nuclear energy technologies that re-establish the U.S. as a world leader in nuclear technology commercialization.

The key recommendation of the NEAC report was that "DOE–NE proceed immediately with pre-conceptual design planning activities to support a new test reactor" to fill the domestic need for a fast-neutron test capability. The considerations for such a capability include:

- An intense, neutron-irradiation environment with prototypic spectrum to determine irradiation tolerance and chemical compatibility with other reactor materials, particularly the coolant.
- Testing that provides a fundamental understanding of materials performance, validation of models for more rapid future development, and engineering-scale validation of materials performance in support of licensing efforts.
- A versatile testing capability to address diverse technology options and, sustained and adaptable testing environments.
- Focused irradiations, either long- or short-term, with heavily instrumented experimental devices, and the possibility to do in-situ measurements and quick extraction of samples.
- An accelerated schedule to regain and sustain U.S. technology leadership and to enable the competiveness of U.S-based industry entities in the advanced reactor markets. This can be achieved through use of mature technologies for the reactor design (*e.g.,* sodium coolant

[1] Fast neutrons are highly energetic neutrons (ranging from 0.1 to 5 million electron volts [MeV] and travelling at speeds of thousands to tens of thousands kilometers per second) emitted during fission. The fast-neutron spectrum refers to the range of energies associated with fast neutrons. Thermal neutrons are neutrons that are less energetic than fast neutrons (more than a million times less energetic [about 0.025eV] and travelling at speeds of less than 5 kilometers per second), having been slowed by collisions with other materials such as water. The thermal neutron spectrum refers to the range of energies associated with thermal neutrons.

in a pool-type, metallic-alloy-fueled fast reactor) while enabling innovative experimentation.

A summary of preliminary requirements that meet these considerations include:

• Provide a high peak neutron flux (neutron energy greater than 0.1 MeV) with a prototypic fast-reactor-neutron-energy spectrum; the target flux is 4×10^{15} neutrons per square centimeter per second (neutrons/cm^2-sec) or greater.

• Provide high neutron dose rate for materials testing [quantified as displacements per atom]; the target is 30 displacements per atom per year or greater.

• Provide an irradiation length that is appropriate for fast reactor fuel testing; the target is 0.6 to 1 meter.

• Provide a large irradiation volume within the core region; the target is 7 liters.

• Provide innovative testing capabilities through flexibility in testing configuration and testing environment (coolants) in closed loops.

• Provide the ability to test advanced sensors and instrumentation for the core and test positions.

• Expedite experiment life cycle by enabling easy access to support facilities for experiments fabrication and post-irradiation examination.

• Provide life-cycle management (spent nuclear fuel storage pending ultimate disposal) for the reactor driver fuel (fuel needed to run the reactor) while minimizing cost and schedule impacts.

• Make the facility available for testing as soon as possible by using proven technologies with a high technology readiness level.

Having identified the need for the VTR, NEICA directs DOE ''to the maximum extent practicable, complete construction of, and approve the start of operations for, the user facility by not later than December 31, 2025.''

Secretary of Energy Rick Perry announced the launch of the Versatile Test Reactor Project on February 28, 2019 as a part of modernizing the nuclear research and development (R&D) user facility infrastructure in the United States.

An initial evaluation of alternatives during the pre-conceptual design planning activity recommends the development of a well-instrumented sodium-cooled, fast-neutron-spectrum test reactor in the 300 megawatt-thermal power level range. This design would provide a flexible, reconfigurable testing environment for known and anticipated testing. It is the most practical and cost-effective strategy to meet the mission need and address constraints and considerations identified above. The evaluation of alternatives is consistent with the conclusions of the test reactor options study and the NEAC recommendation.

DOE expects that the VTR, coupled with the existing supporting R&D infrastructure, would provide the basic and applied physics, materials science, nuclear fuels, and advanced sensor communities with a unique research capability. This capability would enable a comprehensive understanding of the multi-scale and multi-physics performance of nuclear fuels and structural materials to support the development and deployment of advanced nuclear energy systems. To this end, DOE is collaborating with universities, commercial industry, and national laboratories to identify needed experimental capabilities.

Purpose and Need for Agency Action

The purpose of this DOE action is to provide a domestic versatile reactor-based fast-neutron source and associated facilities that meet identified user needs (*e.g.,* providing a high neutron flux of at least 4×10^{15} neutrons/cm^2-sec and related testing capabilities). Associated facilities include those for the preparation of driver fuel and test/experimental fuels and materials and those for the ensuing examination of the test/experimental fuels and materials; existing facilities would be used to the extent possible. The United States has not had a viable domestic fast-neutron-spectrum testing capability for over two decades. DOE needs to develop this capability to establish the United States' testing capability for next-generation nuclear reactors—many of which require a fast-neutron spectrum for operation—thus enabling the United States to regain technology leadership for the next generation nuclear fuels, material, and reactors. The lack of a versatile fast-neutron-spectrum testing capability is a significant national strategic risk affecting the ability of DOE to fulfill its mission to advance the energy, environmental, and nuclear security of the United States and promote scientific and technological innovation. This testing capability is essential for the United States to modernize its nuclear energy industry. Further, DOE needs to develop this capability on an accelerated schedule to avoid further delay in the United States' ability to develop and deploy advanced nuclear energy technologies. If this capability is not available to U.S. innovators as soon as possible, the ongoing shift of nuclear technology dominance to other international states (*e.g.,* China, the Russian Federation) will accelerate, to the detriment of the U.S. nuclear industrial sector.

Proposed Action

The Proposed Action is for DOE to construct and operate the VTR at a suitable DOE site. DOE would utilize existing or expanded, collocated, post-irradiation examination capabilities as necessary to accomplish the mission. DOE would use or expand existing facility capabilities to fabricate VTR driver fuel and test items and to manage radioactive wastes and spent nuclear fuel.

Versatile Test Reactor

The Nuclear Energy Innovation Capabilities Act of 2017 (Pub. L. 115–248) directed DOE, to the maximum extent practicable, to approve the start of operations for the user facility by not later than December 31, 2025. DOE recognized that a near-term deadline would require the technology selected for the user facility to be a mature technology, one not requiring significant testing or experimental efforts to qualify the technology needed to provide the capability.

The generation of a high flux of high-energy or fast neutrons requires a departure from the light-water-moderated technology of current U.S. power reactors and use of other reactor moderating and cooling technologies. The most mature technology that could provide the high-energy neutron flux is a sodium-cooled reactor, for which experience with a pool-type configuration and qualification of metallic alloy fuels affords the desired level of technology maturity and safety approach. Sodium-cooled reactor technology has been successfully used in Idaho at the Experimental Breeder Reactor (EBR)-II, in Washington at the Fast Flux Test Facility, and in Michigan at the Fermi 1 Nuclear Generating Station.

The current VTR concept would make use of the proven, existing technologies incorporated in the small, modular GE Hitachi Power Reactor Innovative Small Module (PRISM) design. The PRISM design [2] meets the need to use a sodium-cooled, pool-type reactor of proven (mature) technology. The VTR would be a smaller (approximately 300 megawatt thermal) version of the GE Hitachi

[2] The PRISM design is based on the EBR–II reactor, which operated for over 30 years. PRISM received a review by the Nuclear Regulatory Commission as contained in NUREG–1368, *Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor,* which concluded that ''no obvious impediments to licensing the PRISM design had been identified.''

APPENDIX B
DETAILED PROJECT INFORMATION

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DETAILED PROJECT INFORMATION

B.1 Introduction

A conceptual design for the Versatile Test Reactor (VTR) has been developed to meet user-identified needs for a fast neutron flux test facility. The VTR would provide an environment in which test specimens, such as new types of reactor fuels and materials, could be exposed to high levels of neutron flux, enabling the simulation of years of neutron exposure in a power reactor in significantly less time. After irradiation in the VTR, test specimens would be examined in post-irradiation examination facilities. Test assembly examination would be performed in facilities specifically designed to safely handle radioactive materials. VTR fuel would be fabricated at existing U.S. Department of Energy (DOE) facilities where upgrades involving removal of existing equipment and installation of new equipment would be required. DOE would put in place the facilities and processes for the treatment and disposition of spent VTR driver fuel. VTR driver fuel would not be reprocessed for the recovery of special nuclear material (plutonium or enriched uranium), but instead the entire driver assembly (including upper and lower reflectors, caps, etc.) would be melted for ultimate disposal.

This appendix provides information about the design of these facilities: the VTR, test assembly post-irradiation examination facilities, feedstock preparation facilities, driver fuel fabrication facilities, and spent fuel treatment and storage facilities. It also provides information about how the activities at these facilities would be implemented at the proposed DOE sites. The VTR would be a new facility, but other activities could be performed in new facilities or at existing facilities (with or without modification).

B.2 Versatile Test Reactor

B.2.1 Introduction

The current VTR concept is a sodium-cooled, pool-type fast reactor that provides a fast neutron spectrum environment for testing advanced nuclear fuels and materials. It generates approximately 300 megawatts thermal (MWth) and would make use of the technologies incorporated into the GE Hitachi Power Reactor Innovative Small Module (PRISM) design.[1] The VTR would meet the test reactor requirements identified in the *Mission Need Statement for the Versatile Test Reactor (VTR), A Major Acquisition Project*, as shown in **Table B–1** (DOE 2018b). In addition to these reactor parameters, the selection of the reactor type and fuel type would meet the requirement for the test facility program to provide management of the reactor fuel.

Unlike the PRISM reactor, which is designed as an electrical power plant, the VTR would be used solely as a test reactor for advancing the understanding of materials and fuels that could be used in current or future reactor designs. This results in several differences in the design and operation of the VTR from the PRISM.

The VTR, like the PRISM, would be a fast reactor. A fast reactor is a category of nuclear reactor in which the fission chain reaction is sustained by fast neutrons (carrying energies above 0.1 million electron volts (MeV) to about 10 MeV and travelling at speeds of thousands to tens of thousands of kilometers per second), as opposed to thermal neutrons used to sustain the fission chain reaction in thermal-neutron reactors. A fast reactor needs no neutron moderator, but requires fuel that is relatively rich in fissile

[1] The PRISM design is an evolutionary design based on the Experimental Breeder Reactor (EBR)-II, which operated for over 30 years. PRISM received a review by the Nuclear Regulatory Commission as contained in NUREG-1368, *Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor*, which concluded that "no obvious impediments to licensing the PRISM design had been identified."

material when compared to that required for a thermal-neutron reactor.[2] Since the VTR would be designed to test fuels and other materials in a fast flux environment, the design has been selected to maximize the number of fast neutrons present in the reactor core. The core design incorporates a reflector. The reflector would consist of assemblies of material surrounding the core that reflect neutrons that travel out of the fueled (active) region of the core back into the core, without significantly slowing them down. Also, there are no materials within the reactor specifically intended to moderate (slow down) the neutrons as there are in water-cooled nuclear power reactors; moderated neutrons are effectively lost fast neutrons.

Table B–1. Versatile Test Reactor Test Requirements

Key Performance Parameter	***Target Objective***	***VTR Conceptual Design*** [a]
Provide a high-peak neutron flux (neutron energy > 0.1 million electron volts) with a prototypic fast reactor neutron energy spectrum	$\geq 4{\times}10^{15}$ neutrons per square centimeter/second	$\geq 4{\times}10^{15}$ neutrons per square centimeter/second
Provide high neutron dose rate for materials testing, quantified as displacements per atom	> 30 displacements per atom/year	51 displacements per atom/year for HT-9 and other structural materials with irradiation over three 100-day cycles (17 displacements per atom/cycle).
Provide an irradiation length that is typical of fast reactor designs	0.6 meters ≤ irradiation length ≤ 1.0 meter	0.8 meter active core height
Provide a large irradiation test volume within the core region	≥ 7 liters	Individual test volumes of greater than 7 liters, in multiple test locations
Provide experiment hardware such as casks and storage locations to support experimental mission	Provide capability for open-core, closed loops, and rabbit facility for testing sodium, lead, lead-bismuth, helium, and molten salt loops	Incorporates six positions for highly instrumented test assemblies that allow testing under different coolants, and including a rabbit facility for rapid insertion/removal of a test specimen, plus additional positions for non-instrumented assemblies

HT-9 = a stainless-steel alloy of iron, chromium, molybdenum, tungsten, nickel, and carbon; VTR = Versatile Test Reactor.

[a] The VTR test requirement parameters are as identified in the VTR Conceptual Design Report (INL 2019b). As the design evolves, these parameters are subject to change. But, none would be allowed to be changed to the extent that any Key Performance Parameter Target Objective would not be met.

A sodium-cooled reactor is a type of liquid metal reactor that uses liquid sodium as the primary coolant for the reactor. Because of the physical and thermal properties of sodium, the reactor operates slightly above atmospheric pressure and with coolant temperatures of up to 1,100 degrees Fahrenheit (°F). The primary heat removal system (HRS) operating pressure is significantly lower than that of a typical commercial light water reactor, and the operating (coolant) temperature of the fuel is higher than a typical commercial light water reactor. The reactor, primary HRS, and safety systems would be similar to those of the PRISM design. However, since the VTR is a test reactor and would not be used for electrical power generation, the secondary systems would be much simpler. The heat generated during operation would be transferred from the primary HRS to a secondary coolant system. Both coolant systems would use liquid sodium as coolant. Heat would ultimately be rejected to the atmosphere through a set of sodium-to-air heat exchangers within the secondary coolant system.

The VTR would be a pool-type reactor with both a reactor vessel and a guard vessel. This designation reflects the configuration of the primary HRS. In a pool-type reactor, the components of the primary HRS

[2] In contrast, most operating commercial nuclear power plants are thermal reactors, and the fission chain reaction is sustained by thermal neutrons. Thermal neutrons are less energetic than fast neutrons (more than a million times less energetic [about 0.025 MeV] and travelling at speeds of about 2.2 kilometers per second), having been slowed by collisions with other materials such as water. The thermal neutron spectrum refers to the range of energies associated with thermal neutrons.

are physically located within the reactor vessel. In the case of the VTR, this includes the primary electromagnetic[3] (EM) pumps and the intermediate heat exchangers. There are no penetrations in the sides of a pool-type reactor vessel or the guard vessel. The secondary cooling system pipes exit the reactor through the reactor vessel head. In contrast, a loop-type reactor has vessel penetrations for primary coolant, and the major pieces of equipment for the primary HRS are located outside of the reactor vessel. The major advantages of the pool-type reactor are a reduction in the number of penetrations in the reactor vessel and an overall reduction in size of the primary cooling systems. With the use of a guard vessel, which would maintain the sodium level within the core high enough to ensure core cooling, there is a significantly reduced likelihood of a loss of cooling accident.

The VTR, like the PRISM, would use metallic alloy fuels. The conceptual design for the first fuel core of the VTR proposes to utilize a uranium-plutonium-zirconium alloy fuel. Such an alloy fuel was tested previously in the Experimental Breeder Reactor (EBR)-II, the Fast Flux Test Facility (FFTF), and the INL Transient Reactor Test Facility. Later reactor fuel could consist of other mixtures and varying enrichments of uranium and plutonium and could use other alloying metals in place of zirconium.

The VTR is being designed for an operational lifetime of 60 years.

Unless otherwise identified, the following information is taken from the Idaho National Laboratory (INL) VTR Conceptual Design Report (INL 2019b).

B.2.2 Versatile Test Reactor General Arrangement

Regardless of the location of the VTR, the physical layout of the facility is expected to be similar (see **Figure B–1)**. The design can be developed independent of the final siting of the facility. There would be four major structures associated with the VTR: the reactor building (called the Reactor Facility), the secondary heat rejection system sodium-to-air heat exchangers (SAHXs), a plant electrical switchyard, and an Operational Support Facility. Additional structures[4] would include a Perimeter Intrusion Detection and Assessment System (PIDAS) with a double fence and guard posts/access ports. The Operational Support Facility would be located outside of the PIDAS. The VTR complex would cover approximately 25 acres (INL 2020c).

The Reactor Facility would contain most of the systems and components required for operation of the reactor. At grade level, the facility would house reactor systems equipment, experiment support area, operating floor crane (bridge crane), receiving and shipping area (truck bay), access to below-grade storage for fuel casks and experiments, and the Reactor Vessel Auxiliary Cooling System (RVACS) stacks. The reactor vessel, temporary storage locations for fresh fuel[5] and irradiated test assemblies, and most of the RVACS would be located below grade (see **Figure B–2)**. Among the other areas that would be located within the Reactor Building are the control room, electrical and battery rooms, staging and storage areas, radiological waste storage, reactor auxiliary systems areas, and secondary cooling equipment areas. The Reactor Facility would have a single operating crane, capable of transferring core assemblies, fuel casks, test assemblies, and equipment throughout the facility.

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[3] EM pumps use the interaction between magnetic fields generated by magnets and electric currents to induce flow in an electrically conductive liquid such as molten sodium. EM pumps can be designed with no moving parts.

[4] This set of additional structures is not all inclusive. Other smaller structures are included in the VTR conceptual design. Additionally, as the VTR design evolves the need for additional structures may be identified. It is anticipated that any such structures would fit within the VTR complex and not materially affect construction or operation.

[5] Spent fuel would be temporarily stored within the reactor vessel. Once sufficiently cool, this fuel would be placed in transfer casks and moved to the fuel storage pad pending transfer to the spent fuel treatment facility.

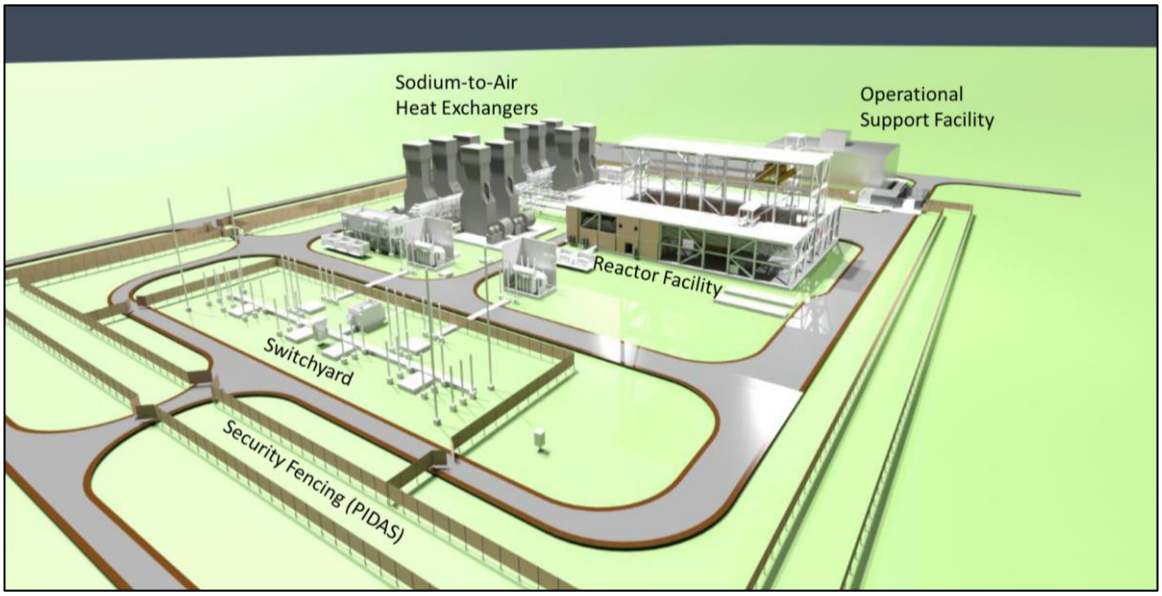


Figure B–1. Site Arrangement

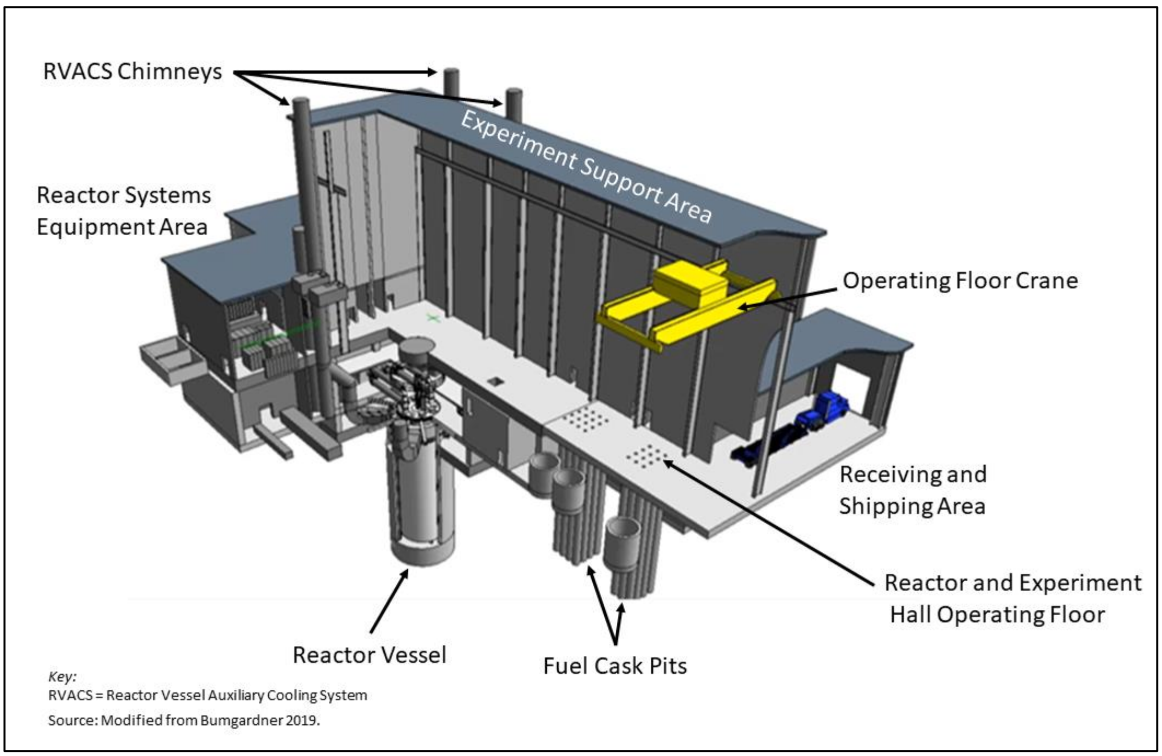


Figure B–2. Conceptual Design for the Versatile Test Reactor Facility

Most VTR activities would be performed at grade level, primarily on the Reactor and Experiment Hall operating floor. Material going into and out of the facility would pass through the shipping and receiving area. Most of the activity associated with fuel movement, spent fuel cleaning, and test assembly movement and final assembly would occur within the Reactor and Experiment Hall operating floor area. (The Reactor operating floor would be located above the reactor vessel; the Experiment Hall extends from this area to and connects with the receiving and shipping area.) The experiments support area includes locations for experiment control systems for experiments and capsule insertion and receipt areas for rabbit capsules (test capsules that can be rapidly inserted and removed from the reactor core during operation). Temporary storage areas, pits, for fresh fuel and unirradiated and irradiated test assemblies would be located beneath the operating floor; the tops of these pits would be at the floor level of the operating floor.

Approximate physical dimensions of the Reactor Facility and a listing of the equipment located at each level of the facility are provided in **Table B–2**.

Table B–2. Versatile Test Reactor Facility Physical Dimensions

VTR Facility Level	*Dimensions Length by Width (in feet)/ Area (in square feet)*	*Equipment*
Footprint		
16 to 88.5 feet above grade	280 × 180	HVAC equipment, secondary cooling system equipment rooms, RVACS stacks, operating floor crane, gaseous radwaste equipment and stack, stairs and elevators
At grade to 16 feet above grade	280 × 180 [a]/ 42,000	Main operating floor, shipping and receiving, experiment support areas, control room, secondary cooling system equipment, electrical and battery rooms, HVAC equipment, RVACS stacks, stairs and elevators, solid radwaste storage
0 feet to 29 feet below grade	280 × 160 [a]/ 39,000	Reactor head access area, fuel cask and temporary irradiated test assembly storage areas, radiological waste storage areas, secondary cooling system equipment rooms, experiment support areas, electrical and battery rooms, building HVAC equipment, RVACS stacks and ductwork, stairs and elevators
Below grade from 29 to 41 feet	250 × 60 [b]/ 15,000	Reactor vessel, fuel cask and temporary irradiated test assembly storage areas, secondary coolant system equipment (coolant drain tanks) area, RVACS cold air plenum and ductwork, ladders
Below grade from 41 to 93 feet	31 diameter/ 750 area	The reactor vessel and enclosure (enclosure floor is at -93 feet), RVACS collector cylinder, sodium fire suppression collection tanks, sump pumps
Height (feet)		
Main building	88.5	
Annex	36	
RVACS chimneys	98	Height of the 4 chimneys (hot air exhaust elevation)
	56	Cold air intake elevation

HVAC = heating, ventilation, and air conditioning; RVACS = Reactor Vessel Auxiliary Cooling System; VTR = Versatile Test Reactor.

[a] Structure is not rectangular. Dimensions are for the longest and widest portions of the structure.

[b] The below-grade building structure would be approximately 150 × 60 feet. Fuel and test assembly storage pits comprise the remainder of the area.

The secondary HRS structures would consist of approximately 10 individual SAHXs and auxiliary equipment (e.g., SAHX fans). These SAHXs would be similar to those used for the FFTF. Heat generated by the reactor core during operations would be transferred to the HRS from the primary sodium coolant system within the reactor vessel. Pumps located within the Reactor Facility would circulate the secondary coolant (sodium) from the reactor vessel to the SAHXs. SAHX fans would dissipate heat to the atmosphere.

The Operational Support Facility would contain three floors. During construction of the VTR facility, construction workers would use the facility for office space, and a high-bay area would be used as a fabrication facility and serve as a warehousing area. Following construction completion, all three floors would be refinished with drywall, ceilings, office cubicles, and office furniture for approximately 200 full-time staff. The building heating, ventilation, and air conditioning (HVAC) would be housed above the third floor. A reactor plant simulator would be installed to support initial commissioning and operations on the second floor of this facility. The high-bay facility area would be used to support maintenance activities and serve as a clean parts storage area. A parking lot located nearby would accommodate approximately 200 parking spaces.

B.2.3 Versatile Test Reactor Core and Fuel Design

The VTR core would consist of three regions: the fuel, reflector assemblies, and shield assemblies (see **Figure B–3**). Within the fuel region, the active part of the core, there would be driver fuel assemblies, control and safety assemblies, and test assembly locations. (Test assemblies are discussed in Section B.2.4.) The reactor core achieves peak fast neutron fluxes greater than 4×10^{15} neutrons per square centimeter per second for neutron energies greater than 0.1 MeV inside of multiple core locations for experiment items. Experiments (i.e., test specimens) would be placed in test locations in the active reactor core and in test pins located in driver fuel. Additionally, non-instrumented test locations could be located in the first row of reflector assemblies.

Core Components

Driver (fuel) assembly located in the active region of the core contains the fuel needed to power the reactor and produces the fast neutron flux necessary for irradiation of test assemblies or specimens.

Reflector assembly surrounds the active central region of the core that contains driver assemblies and test assemblies and contains material to reflect neutrons back into the central part of the core.

Shield assembly is positioned outside of the reflector assemblies within the core and contains material to absorb neutrons that pass through the reflector to reduce neutron damage to the reactor structural components.

Test assembly contains the test specimen and any equipment needed to support the experiment. Instrumented test assemblies could be as long as 65 feet and are located in the active region of the core. Non-instrumented assemblies would be the same length as driver assemblies (less than 13 feet) and may be located in either the active region of the core or in the first row of reflector assemblies.

Test specimen is the material being exposed to a fast neutron flux to determine the effects of the exposure and includes any capsule necessary to support the test. The test specimen can be no more than about 31 inches long.

Control assembly provides the core startup control, power control, burnup compensation, and absorber run-in in response to demands from the plant control system. In conjunction with safety assemblies, provide a rapid shutdown capability.

Safety assembly provides redundant rapid shutdown capability.

Core

The conceptual design for the VTR core contains 66 driver fuel assemblies within the active core. Each assembly would contain 39.9 kilograms of uranium and plutonium for a total core fuel loading of approximately 2.6 metric tons (INL 2019a). The nine safety and control assemblies would contain fuel poisons (neutron absorbers). There would be six instrumented test locations within the core. These test locations could contain instrumented fuel or material test assemblies, rabbit facility (a rapid transport system for insertion and extraction of specimens or samples during a VTR irradiation cycle), or instrumented cartridge loop assemblies. Non-instrumented experiments (i.e., test specimens) could be placed in multiple locations in the reactor core or in the reflector region. **Table B–3** summarizes these core design features.

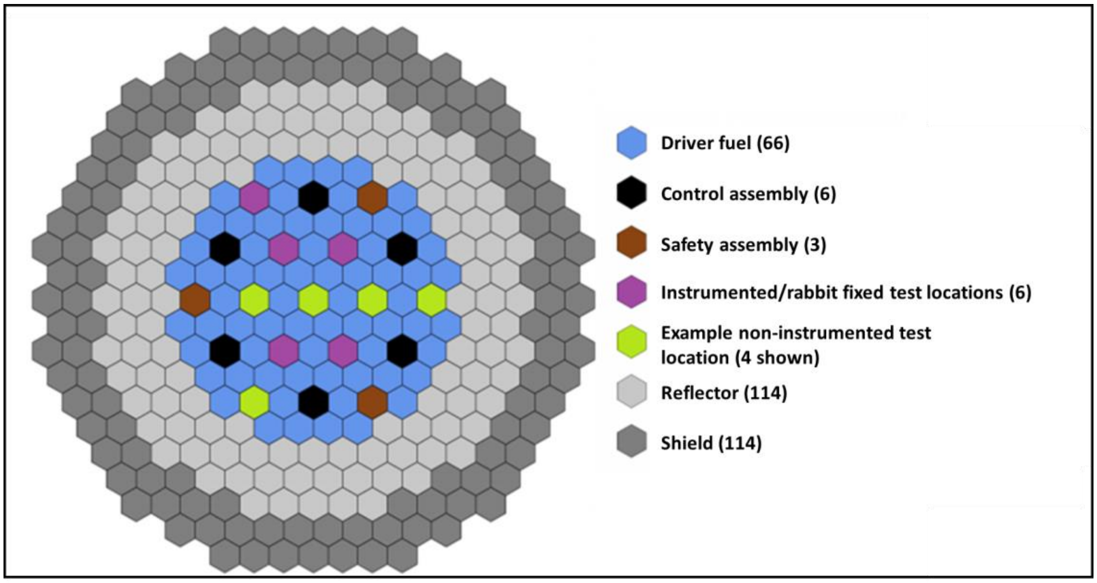


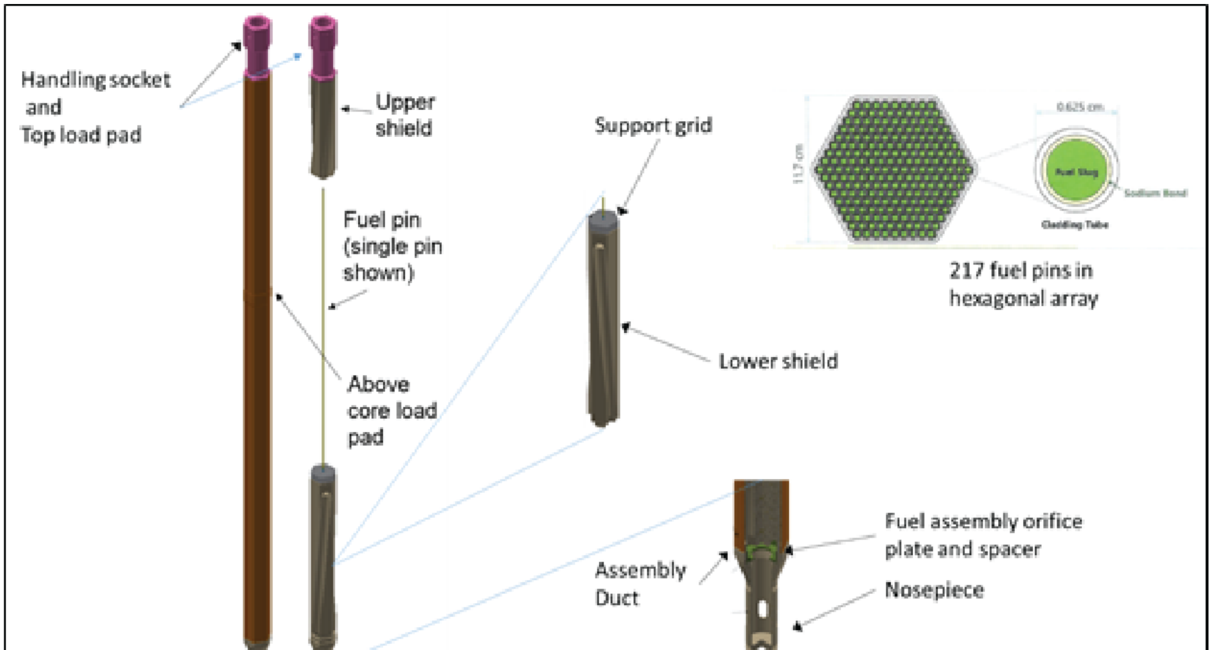
Figure B–3. Versatile Test Reactor Core Configuration

Table B–3. Key Design Characteristics of Versatile Test Reactor Core

Core Design Parameter	*Value*
General Conditions	
Pins per assembly	217
Number of driver fuel assemblies	66
Number of test assembly locations	Six fixed instrumented test locations and multiple options for non-instrumented locations in the core and reflector.
Available test volume	greater than 7 liters per test assembly location
Number of control and safety assemblies	9 (6 control and 3 safety)
Total number of fuel pins in core	14,322
Core diameter [a]	2.35 meters
Core heavy metal mass [b]	2.6 metric tons
Number of reflector assemblies [c]	114
Number of shield assemblies [d]	114
Pin Conditions	
Fuel pin length	165 centimeters
Fuel length	80 centimeters
Sodium height (above fuel)	2 centimeters
Argon height (above sodium)	80 centimeters
Pin diameter	0.625 centimeters
Fuel slug diameter	0.455 centimeters
Assembly Conditions	
Inter-assembly gap	0.3 centimeters
Duct width outside (flat to flat)	11.7 centimeters
Fuel assembly length	3.85 meters

[a] The core diameter includes fuel/test assemblies, reflector assemblies, and shield assemblies. The active core diameter (fuel and test assemblies only) would be between 132 to 144 centimeters (INL 2019a).
[b] Total uranium and plutonium mass for the initial core load.
[c] Some assembles within the inner ring of reflector assemblies could be replaced with non-instrumented test assemblies.
[d] The outer ring of shield assemblies could be replaced with spent fuel assemblies. This would provide up to 60 spent fuel storage locations.

The driver fuel would consist of hexagonal assemblies, with each assembly containing 217 HT-9 stainless-steel clad, uranium-plutonium-zirconium alloy fuel pins (see **Figure B–4**). From the bottom to the top, the driver fuel assembly is composed of the nosepiece/inlet nozzle module, the lower shield, the fuel pin bundle, the upper shield and the upper handling socket module. An assembly duct extends from the inlet to outlet modules and contains the two shields and the pin bundle. The assembly duct, support grid, and upper and lower shields would be constructed of HT-9 stainless steel. Overall, the driver fuel assembly would be about 3.85 meters long and would measure 11.7 centimeters from one flat side to the opposite flat side.



Source: Crawford 2019.

Figure B–4. Driver Fuel Assembly

The VTR core design would include six control assemblies and three safety assemblies (see **Figure B–5**). The control assemblies adjust for changes in reactivity and control the power level of the core. The safety assemblies are fully withdrawn from the fuel region during normal operation and are fully inserted into the core during reactor shutdown to provide additional shutdown margins. Each control and safety assembly is connected to a control driveline connected to a control drive mechanism, located atop the reactor upper head through penetrations in the reactor top assembly rotatable plug. All nine assemblies are configured to form a double-ducted assembly, with the inner duct containing an array of 37 wire-wrapped absorber pins. The pins are made of an HT-9 stainless-steel cladding and boron carbide (B_4C) pellets. **Table B–4** summarizes the characteristics of the control and safety assemblies.

There would be 114 radial reflector assemblies and 114 radial shield assemblies. Reflector assemblies improve neutron efficiencies (more of the neutrons generated during fission remain within the core for a longer time) by reflecting some leaked neutrons back into the core. The shield assemblies protect surrounding structures (e.g., the reactor vessel and guard vessel) from the effects of neutron radiation. Both sets of assemblies would be made with a hexagonal HT-9 stainless-steel duct.

The volume inside the reflector assembly duct would consist of HT-9 stainless-steel rods. These rods would be tightly packed (there would be no wire wrap around the rods as there would be in the driver fuel assemblies) to achieve a high steel volume. Within the reflector assembly, the HT-9 and coolant volume fractions would be 0.80 and 0.20, respectively.

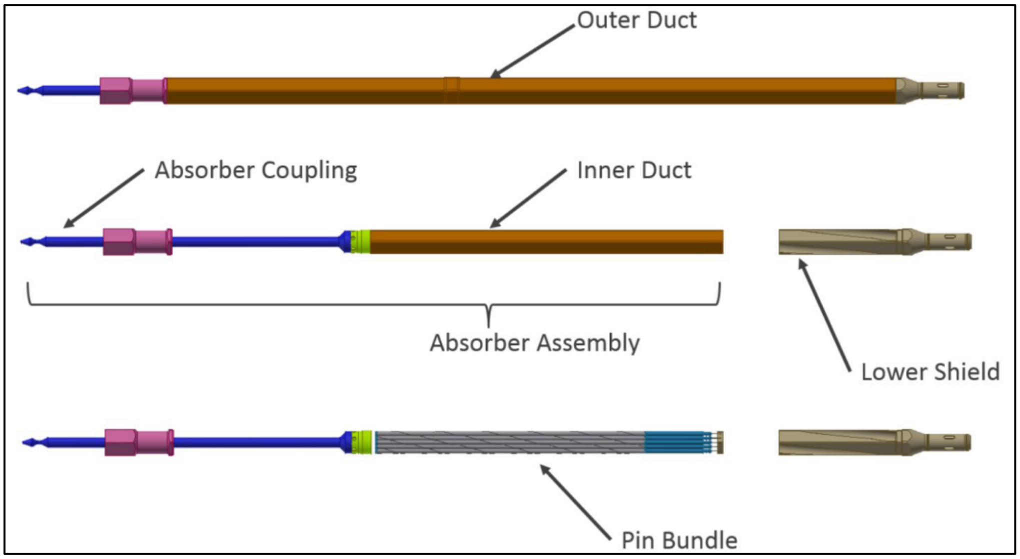


Figure B–5. Control or Safety Assembly

Table B–4. Control and Safety Rod Assembly Dimensions

Conditions	***Value***
Inter-assembly gap	3.0 millimeters
Outer hexagonal duct inside flat-to-flat distance	11.1 centimeters
Inner hexagonal duct inside flat-to-flat distance	9.9 centimeters
Number of absorber pins	37
Absorber pin outer diameter	1.54 centimeters

The shield assembly ducts would contain a bundle of wire-wrapped absorber pins made of an HT-9 stainless-steel cladding and B_4C pellets. Within the shield assembly, the B_4C absorber, HT-9, coolant, and bond gas volume fractions would be 0.40, 0.28, 0.24, and 0.08, respectively.

Driver Fuel

Both metallic and mixed oxide fuel were considered for the VTR. Metallic fuels provide several advantages over oxide fuel and were identified as the preferred fuel option. Advantages of metallic fuels over oxide fuels include:

- A smaller core at the same neutron flux level due to the higher density of fissionable metals (uranium and plutonium),
- Better performance under accident conditions,
  - Lower likelihood of energetic events that could threaten the reactor vessel and containment boundaries during core meltdown
  - Better response during a transient without scram
- Consistent performance over a wide range of fuel enrichments and alloy compositions, and
- Greater experience base with metallic fuels for fast reactors (EBR-II, Fermi-1) providing support for the licensing basis for the fuel and reactor (TerraPower 2019).

DOE considered several fuel compositions of plutonium and uranium to fuel the VTR. DOE determined that for a 300-MWth VTR, a U-20Pu-10Zr fuel with the uranium enriched to 5 percent provides the highest combination of peak neutron flux (about 4.5×10^{15} neutrons per centimeter squared per second) and technical readiness. It is the most likely fuel combination to be used in the initial fuel loading for the VTR. By weight, this fuel is 70 percent uranium enriched to 5 percent uranium-235, 20 percent plutonium, and

10 percent zirconium. The total amount of heavy metal (uranium and plutonium) required annually, as shown in **Table B–5**, for the VTR would be about 1.8 metric tons.[6] The initial fuel loading for the VTR would require about 2.6 metric tons of heavy metal (uranium and plutonium) (INL 2019a).

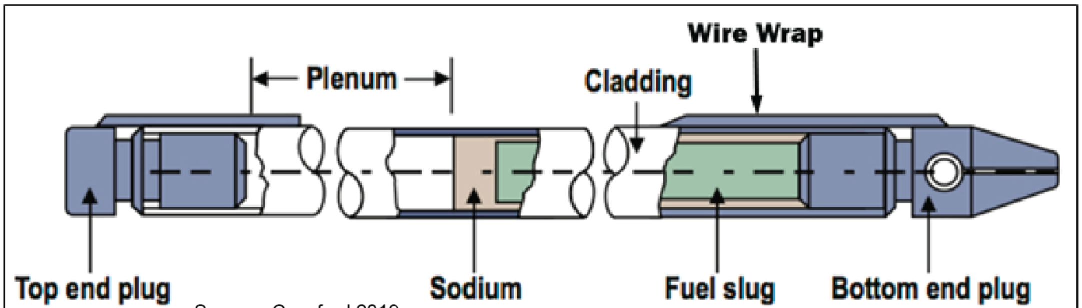
Table B–5. Versatile Test Reactor Fuel Requirements

Fuel Component	*Initial Core (kilograms)*	*Annual Requirement (kilograms)*	*Lifetime – 60 Years (metric tons)*
Plutonium	590	400	24
Uranium	2,000	1,400	85
Zirconium	290	200	12
Total Heavy Metal	2,600	1,800	110

Source: Derived from INL 2019a.

Several factors could impact the selection of future VTR fuel. For example, a desire to increase the fast neutron flux with an improvement in the readiness level (more mature fabrication and use) of higher content plutonium fuels could result in a decision to use higher plutonium content fuel. Other factors could result in the need to use lower plutonium content, but higher uranium enrichment fuels. For this environmental impact statement (EIS), it has been assumed that future fuel requirements for the VTR would be met using the U-20Pu-10Zr fuel anticipated to be used in the initial core.

Each fuel pin (see **Figure B–6**) would be 165 centimeters long and have an outer diameter of 0.625 centimeters. Only about 80 centimeters of the fuel pin would contain metallic fuel, approximately 184 grams of heavy metal (INL 2019a). Each fuel pin would contain fuel slugs, with a diameter of 0.455 centimeters. There would be an approximately equal length of a gas plenum, filled with argon in the proposed VTR design, above the fuel. This gas space provides a mechanism to limit pressure increases within the fuel pin. (When fuel is irradiated in a fast reactor, the metallic fuel swells as fission products are generated. Pores form throughout the fuel as it swells due to irradiation and pressure from the gaseous fission products. The fission product gases escape through these pores to this plenum in the fuel pin.) Between the fuel and the gas plenum, there would be a short length (2 centimeters) of sodium created during the VTR driver fuel production process (see Section B.5). The fuel, sodium, and gas plenums would be enclosed within HT-9 stainless-steel (a stainless-steel alloy of iron, chromium, molybdenum, tungsten, nickel, and carbon) cladding, about 0.05 centimeters thick. The space between the fuel and the cladding would be filled with metallic sodium to improve the heat transfer from the fuel to the reactor coolant through the stainless-steel cladding. The small amount of sodium initially above the fuel ensures that there would be sodium between the fuel and the cladding at all times. The wire wrap shown in Figure B–6 maintains spacing between fuel pins within the driver fuel assembly and is also made of HT-9 stainless steel. Top and bottom end plugs complete the structure of the fuel pin.



Source: Crawford 2019.

Figure B–6. Fuel Pin

[6] Based on the replacement of up to 45 fuel assemblies each year (INL 2020c).

B.2.4 Test Assemblies

Non-instrumented experiments (i.e., test specimens) could be placed in multiple locations in the reactor core or in the reflector regions, by replacing a fuel or reflector assembly. Instrumented experiments, which can provide real-time information while the reactor is operating, require a penetration in the reactor cover for the instrumentation stalk and can only be placed in any of six fixed locations. Any of these positions could be used for instrumented test vehicles; a rabbit test facility, and cartridge closed loops;[7] which can provide real-time information while the reactor is operating. At any one time only one of these six locations can accommodate a "rabbit" test facility, where samples can be inserted/removed while the reactor is in operation. The six instrumented test positions are served by six penetrations for the instrumentation stalk and have a direct connection through the reactor vessel head to monitors in the experiment support area with transfers on the rotatable plug, similar to the penetrations for the control assemblies (see Section B.2.5). In addition to the test assemblies, test pins could be located within the driver fuel assemblies. The number of instrumented test locations, plus the flexibility in the number and location of non-instrumented tests would strengthen the versatility of the reactor as a test facility.

Instrumented test vehicle designs have not been developed specifically for the VTR, but they would be developed based on test vehicle designs developed for the EBR-II and FFTF (**Figure B–7** provides a representative design). Based upon previous experience, instrumented test assemblies can incorporate many (e.g., greater than 50) instruments, including those to measure local temperatures, flowrates, pressures (including pressures inside fuel pin fission gas plena), and neutron fluxes. The three test assembly types currently envisioned for use in the VTR are:

- Normal Test Assembly (NTA)
  - NTAs would be the standard non-instrumented or passively instrumented open test assemblies that would be the same size, flat-to-flat, as the driver fuel assemblies.
  - The NTAs would use the same path and equipment as driver fuel for insertion and removal from the reactor.
  - These experiments would be fuels (NTA-F) or materials (NTA-M).
- Extended Length Test Assembly (ELTA)
  - All ELTAs would extend through the reactor head, and typically would have various instrumentation leads, etc., that run to the Non-Radiation and/or Radiation Experiment Rooms adjacent to the Head Access Area.
  - The ELTAs would have specialized casks capable of preheating using downward flowing argon; providing power, as necessary, to the ELTA (e.g., for cartridge loops); and the required lifting fixtures.
  - ELTAs would include fuels (ELTA-F) or materials (ELTA-M) or can be cartridge loops (ELTA CL) that could contain coolants separate from the primary sodium. Figure B–7 provides a representative design of an ELTA-M.
  - The rabbit thimble that would go into the primary coolant would be handled by the same pathway as the ELTAs, although the rabbit thimble is not considered to be an ELTA, but would use the same infrastructure for insertion and removal.
- Rabbit Test Assembly (RTA)
  - The RTA would use a capsule that contains the experiment specimens, which would be propelled down the rabbit tube into the rabbit thimble, irradiated, and recovered during or between test cycles.

[7] Non-instrumented test assemblies could also be placed within an instrumented location.

- The RTA capsules would be loaded and removed from a shielded transfer station in the Radioactive Experiment Room adjacent to the Head Access Area.
- The RTA capsule would be very specialized with tight tolerances to ensure compatibility with the rabbit thimble, fins for heat rejection if needed, and would be qualified as an experiment containment boundary. The capsule typically would contain very small samples which would nearly always be materials due to the extremely rapid insertion, which could result in a significant short reactor power disturbance for fueled specimens/tests.

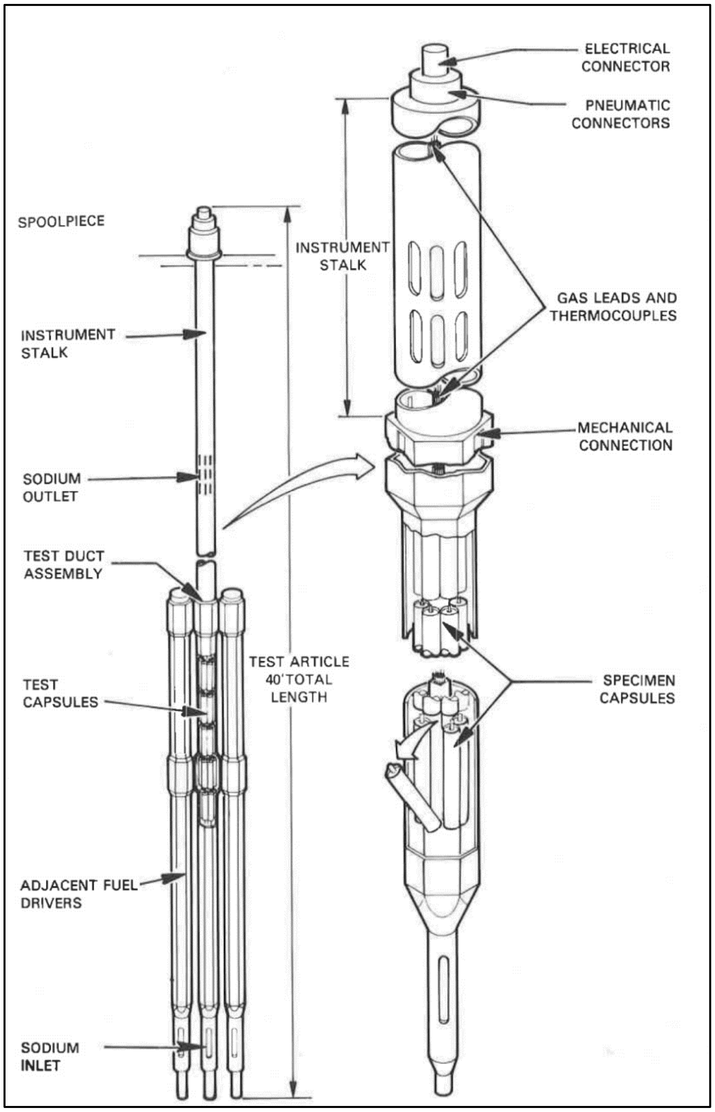


Figure B–7. Representative Instrumented Test Assembly

An important capability for the VTR would be the capability to irradiate cartridge closed loops (see **Figure B–8**) with different closed-loop coolants such as molten lead, molten salt, helium, or even sodium at different conditions than the VTR primary sodium. Thus, the VTR can directly support the development of lead- and lead-bismuth eutectic-cooled fast reactor, molten salt reactor, fluoride high-temperature reactor, high-temperature gas-cooled reactor, and advanced sodium-cooled fast reactor designs. Cartridge loop experiments have been successfully used in other test reactors; designs for VTR-specific closed loop test assemblies able to handle different coolants are under development.

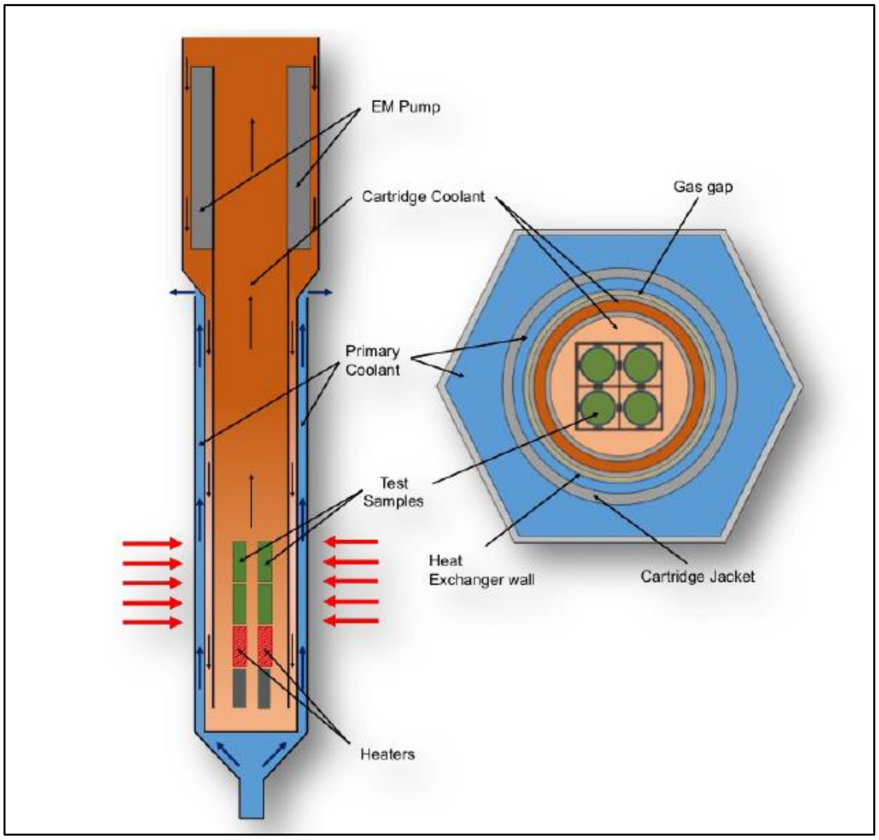


Figure B–8. Closed-Loop Cartridge Test Assembly

In the VTR, the closed-loop coolant would flow upward through a closed-loop fuel region and downward through a surrounding downcomer, where heat would be rejected through a double-wall pressure boundary to upward-flowing VTR primary sodium. Thus, the VTR primary sodium would be the heat sink for the cartridge closed loop. The cartridge closed loop may be similar in height to a driver fuel assembly and coupled to an overlying stalk with instrument leads (including leads for instrumentation to monitor the coolant purity), gas lines (some providing the ability to alter the coolant chemistry to reduce or eliminate corrosion of cladding and structures), and power cables. Each cartridge closed-loop design could incorporate an EM pump or a mechanical pump or gas circulator coupled to a motor atop the stalk through a magnetic coupling.

The remaining test locations within the core and reflector would be used for non-instrumented test assemblies.[8] Non-instrumented experiments (i.e., test specimens) could be placed in multiple locations in the reactor core or in the reflector regions, by replacing a driver fuel assembly, instrumented assembly, or reflector assembly. The non-instrumented test vehicles would be fuel assemblies used to test alternative fuel concepts (possibly a lead test assembly), cladding, and structural materials that may differ from the fuel assemblies. These test assemblies would maintain the same outer dimensions as any fuel assembly. The non-instrumented test vehicle may contain passive instrumentation (e.g., melt wires). Closed-loop cartridges would be used only in instrumented locations; all non-instrumented assemblies would be open and the VTR primary sodium would be the coolant.

[8] Generally, the number of non-instrumented test locations are 4 in the core and an additional 10 in the reflector. However, the number of non-instrumented test locations relies upon the specific cycle-dependent physics and safety calculations. In any given test cycle the number of non-instrumented test assemblies could be more or less than these estimates.

B.2.5 Reactor Vessel and Primary Heat Transport System

The VTR would be a pool-type reactor (see **Figure B–9**), so there are no primary coolant loops external to the reactor vessel. Sufficient space would be provided within the reactor vessel for the reactor core, components of the Primary Heat Transport System (PHTS) and spent fuel storage. The stainless-steel reactor vessel would be cylindrical, approximately 55.8 feet tall with a diameter of approximately 18.7 feet. The reactor vessel would be enveloped by a steel guard vessel, which envelopes the primary vessel and collects sodium in case of a leakage of the primary vessel. The guard vessel surrounds the reactor vessel and extends from beneath the reactor vessel to the upper head/top plate assembly. The space between the two vessels would be filled with argon. Attached to the top of the reactor and guard vessels would be the upper head/top plate assembly. (This assembly would connect with both the reactor vessel and the guard vessel.) The vessels would be supported by horizontal beams arranged like radial spokes and partly supported by vertical beams surrounding the guard vessel. The core is supported from the bottom on a core support structure welded to supports on the inside of the reactor vessel. The reactor vessel would be located below grade within the Reactor Building (from approximately -29 feet to -90 feet) within a concrete enclosure (see Figure B–2). Additional physical parameters are provided in **Table B–6**.

The reactor vessel contains all of the liquid sodium primary coolant. Additionally, an argon cover gas plenum would fill the top of the reactor vessel. The cover gas provides a barrier between the sodium coolant and the reactor closure assembly and serves two functions. The gas plenum provides an additional barrier to atmospheric oxygen, especially during refueling. Fission gases, air, and moisture either generated in the reactor core or present in the sodium migrate to the cover gas and would be removed by a Cover Gas Cleanup System.

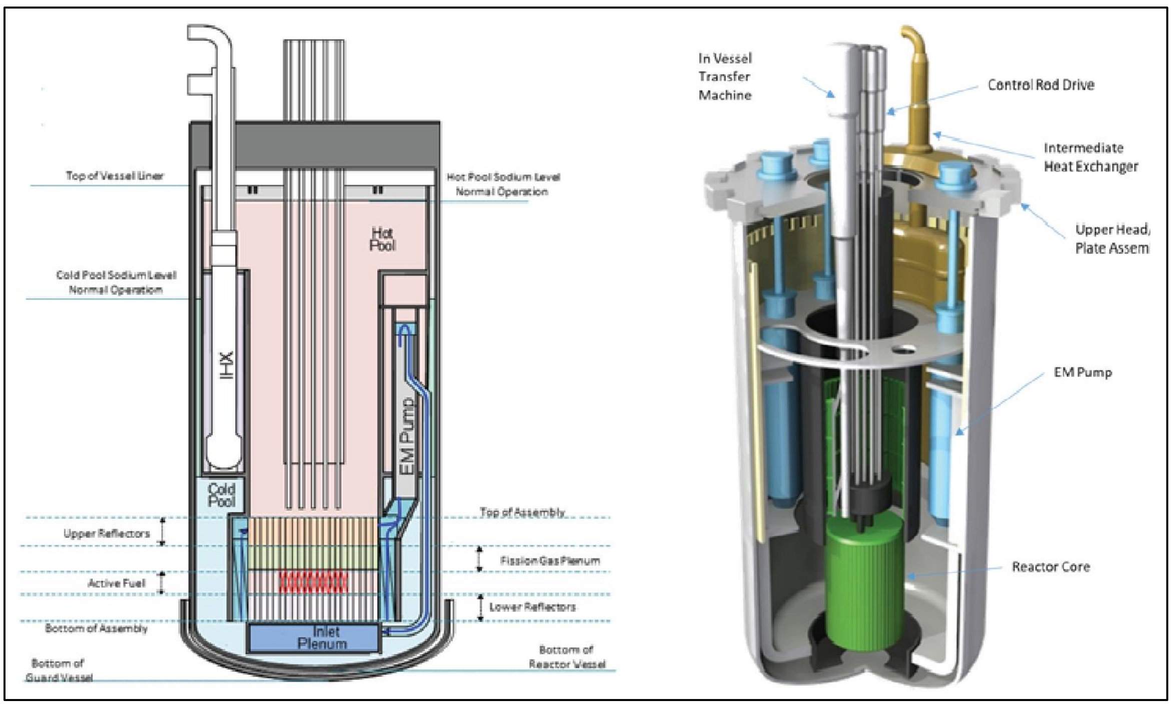


Figure B–9. Versatile Test Reactor Vessel

Table B–6. Conditions and Dimensions for the Versatile Test Reactor Primary Heat Transport System and Reactor Vessel Conceptual Design

Condition or Dimension	Value
Core thermal power	300 megawatts (thermal)
PHTS inlet/outlet temperatures	350/500 °C
Reactor vessel height	17.1 meters
Reactor vessel outer diameter	5.74 meters
Reactor vessel lower head outside height	1.34 meters
Guard vessel height	17.3 meters
Guard vessel outer diameter	6.04 meters
Reactor operating pressure	Slightly above atmospheric
Spent fuel storage capacity [a]	110 assemblies

°C = degrees Celsius; PHTS = Primary Heat Transport System.

[a] Spent fuel capacity includes 60 locations in the outer ring of shield assemblies and 50 locations above but outside the core diameter (at the height of the intermediate heat exchangers).

Source: INL 2019b.

There are no penetrations in the sides or bottom of the reactor vessel or the guard vessel. All penetrations are through the reactor upper head/top plate assembly which consists primarily of a reactor top plate (with a rotatable plug) and a layer of thermal insulation. Penetrations would be provided for intermediate heat exchangers (inlet and outlet flow), the primary EM pumps, the fuel handling In-Vessel Transfer Machine (IVTM), control and safety assembly drive mechanisms, experiments, core instrumentation, a maintenance access port, a transfer port, and a sodium cleanup port. The following penetrations are located in the head outside of the rotating plug; the EM pumps, intermediate heat exchangers, a transfer port, and a sodium cleanup port, **Figure B–10**.

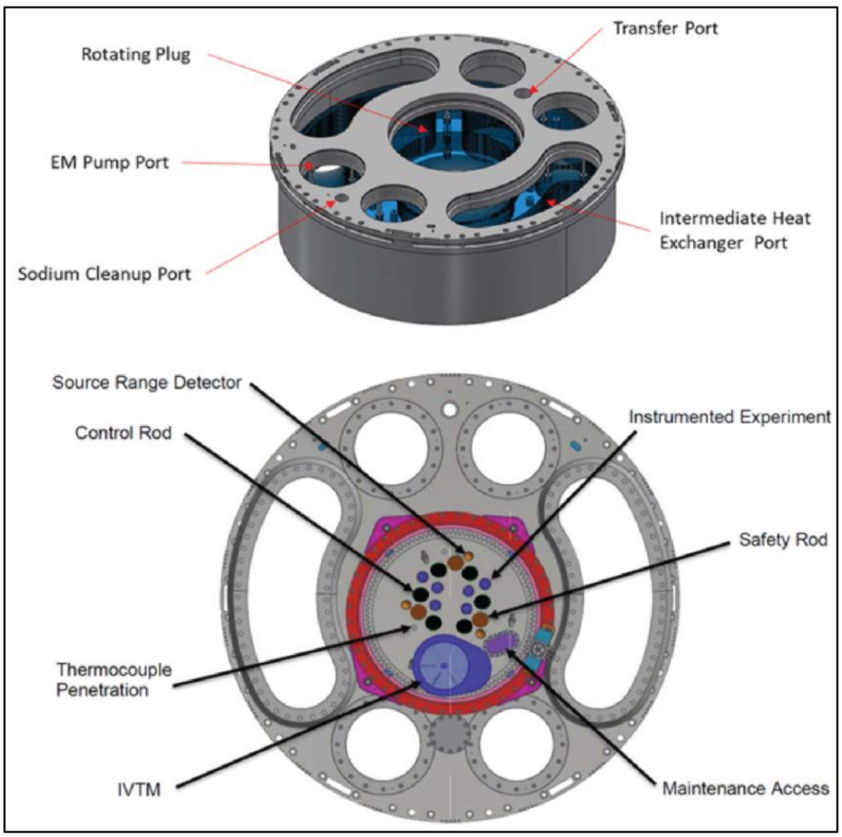


Figure B–10. Versatile Test Reactor Upper Head/Top Plate Assembly

All PHTS components would be within the reactor vessel. Major components would consist of four EM pumps and two intermediate heat exchangers, one heat exchanger for each of the two HRS secondary sodium loops. As shown in Figure B–9, the EM pumps draw sodium from the area surrounding the core within the cold pool and injects the sodium coolant through vertical piping into the inlet plenum (a space filled, in this case, with sodium) beneath the core. Coolant flows through the core to the hot pool region of the reactor vessel where it enters the intermediate heat exchangers. Heat is transferred to the secondary HRS and the primary sodium coolant returns to the cold pool portion within the reactor vessel. Primary sodium coolant pressure and temperature parameters are provided in Table B–6. The PHTS would be sized so that when the EM pumps are operating the system would be able to remove the heat generated within the reactor vessel. This includes the thermal energy of the core; energy generated by the driver fuel assemblies and test assemblies, and other heat sources including spent fuel and the thermal power deposition in the primary sodium from the EM pumps. The EM pumps and intermediate heat exchangers are mounted above the core and supported from the reactor top plate. The PHTS contains no rotating machinery such as a motor, flywheel, or generator.

The VTR reactor vessel design would allow for the storage of spent fuel within the reactor vessel. Storage of spent fuel within the reactor vessel eliminates the need for an external spent fuel storage tank. The fuel would be stored in the reactor vessel until it had cooled sufficiently to be removed from the reactor vessel and transferred to a spent fuel storage cask. Locations for the spent fuel within the reactor vessel include the outer ring of the core shielding assemblies (a spent driver fuel assembly could replace a core shielding assembly) and above and outside of the core at the level of the intermediate heat exchangers. Storage capacity for up to 110 assemblies can be obtained in this manner.

B.2.6 Heat Removal System (Secondary)

The Secondary HRS transfers heat from the PHTS to the environment. This system interfaces with the PHTS in the intermediate heat exchangers located within the reactor vessel (see Figure B–9). The system (see **Figure B–11**), would consist of two identical trains; each containing one full capacity or possibly two 50 percent capacity EM pump, a sodium expansion tank, a sodium drain tank, drain valves, a sodium purification system, and five SAHXs. The sodium drain tank, EM pumps, sodium expansion tank, and sodium purification system would have interconnecting piping located inside the rooms in the Reactor facility, and outside the building, connecting these components to the SAHXs. The design of the SAHXs would use similar concepts as those used in the FFTF secondary cooling system. Each heat exchanger would be equipped with a heater (electric or propane) to warm incoming air when needed (only at times when the VTR is shutdown) to prevent sodium freezing in the system lines (INL 2020c). System flow would be from the intermediate heat exchanger to the sodium-to-air heat exchangers to the pumps and back to the intermediate heat exchangers. Connections to the Sodium Processing System and the Cover Gas System (not shown in the figure) would be provided. System piping from within the VTR Reactor Head Access Area (but not within the reactor vessel) up to the secondary pump rooms would be double walled with the space between the walls filled with inert gas and monitored, providing an additional layer of protection between the sodium coolant and the atmosphere. HRS piping in the secondary pump rooms would have leak protections and monitoring as well. The HRS is capable of rejecting a significant amount of heat in a natural circulation mode. This passive heat rejection behavior, as well as the system providing an intact boundary, are considered safety significant functions given their role in plant defense in depth for reactor cooling in the event of a reactor trip or shutdown.

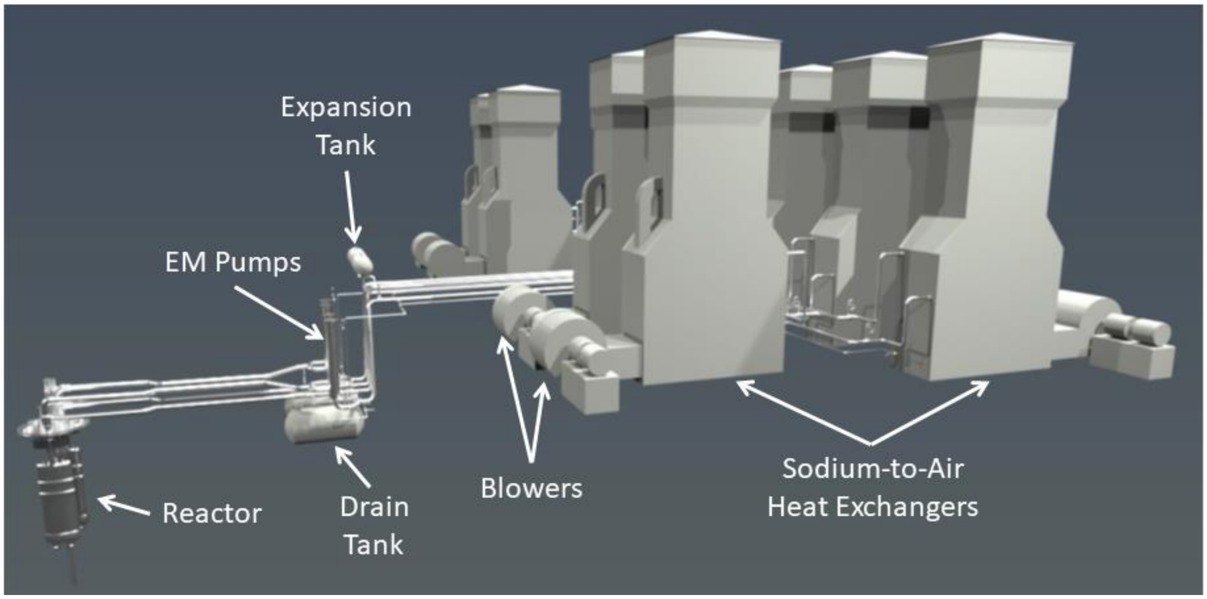


Figure B–11. Secondary Heat Removal System (One of Two Trains)

As with the PHTS, the secondary HRS would be sized to remove the required amount of heat to maintain the PHTS coolant temperature within operational limits. This includes the heat collected from the PHTS plus the thermal energy deposition from the secondary EM pumps. In addition, the PHTS and the HRS would be able to operate in conjunction in a natural circulation mode to remove reactor decay heat. Within minutes following a reactor shutdown, the heat removal capability of one of the two trains of the PHTS and HRS operating in a natural circulation mode would remove the decay heat generated by the reactor core (heat generated by the fuel, any experiments, and spent fuel stored in the reactor vessel). Therefore, sufficient heat removal capability is available to avoid significant thermal transients following a reactor shutdown. Elevation differences between the intermediate heat exchangers and the sodium-to-air heat exchangers support natural convective flow of the secondary sodium.

Table B–7 provides the coolant temperature, flow rates, and operation capacity of the system.

Table B–7. Secondary Heat Removal System Operating Parameters

Parameter	***Value***
Thermal duty (operating)	315 megawatts (thermal)
Cold leg temperature	301 °C
Hot leg temperature	462 °C
Flow per train	14,700 gallons per minute
Total flow	29,400 gallons per minute

°C = degrees Celsius.
Source: INL 2019b; GE Hitachi 2019a.

B.2.7 Reactor Vessel Auxiliary Cooling System

The RVACS would be based on the GE Hitachi PRISM RVACS design and would be a safety class, passive cooling system (no active components) that would provide decay heat removal through natural convection of air without any operator action. The RVACS would remove decay heat from the sodium pool through the reactor and guard vessel walls by radiation and convection to air outside the guard vessel. Heat would be removed to the atmosphere through the natural circulation of air due to the chimney effect. (Density differences between the cold air in the inlet and the hot air in the outlet drives the hot air up and out into the atmosphere.) The system would operate continuously, even during reactor operation. It therefore would operate in conjunction with the PHTS and HRS to remove heat during operation. In the RVACS, air is drawn in through four chimneys, circulated around the reactor guard vessel and exits through the chimney (see **Figure B–12**). All four chimneys contain both cold-air inlet chimneys and hot-air outlet chimneys. The air outlets are located at a higher elevation than the air inlets. The RVACS would be able to perform its safety function with at least one of the four stacks out of service.

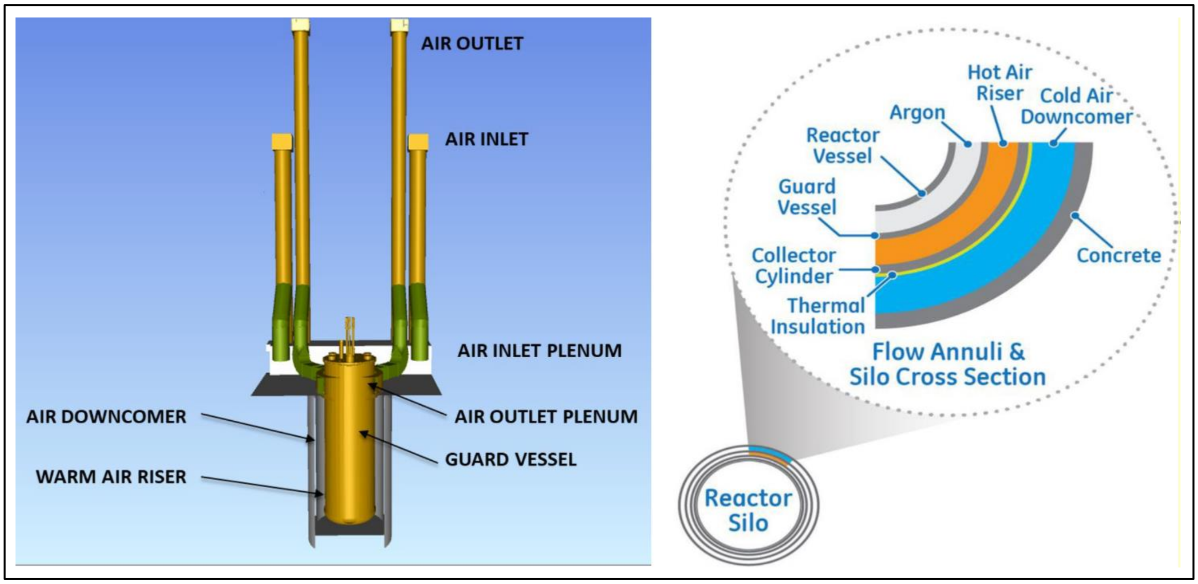


Figure B–12. Reactor Vessel Auxiliary Cooling System

The RVACS operates at a higher heat removal rate as the temperature of the primary sodium increases; alternately, as the temperature of the guard vessel outer surface decreases, so does the heat removed by the RVACS. The RVACS operates at its design capability only when it is the sole means of core heat removal, that is, only when the PHTS and secondary HRS are not functioning. The system reaches its design operation capability only after the reactor has been shut down for some period of time (on the order of a day). This means the core temperature would rise during that time before the heat removed by the RVACS would match the heat generated by the core. At equilibrium, the RVACS would remove approximately 2.8 MWth. (During power operation, the system capability would be limited to approximately 0.7 MWth).

B.2.8 Additional Systems

This section provides brief descriptions of some of the remaining VTR systems. This is not an all-inclusive set of systems (e.g., electrical systems, radiation monitoring, and control room systems are not discussed). The systems described are unique (or configured differently than in other applications) to a sodium-cooled reactor or test reactor. Additionally, the radioactive waste systems are discussed because failures associated with these systems were identified in the accident analysis as a pathway to an accidental radiological release.

Argon Gas Distribution System – The Argon Gas Distribution System would vaporize liquid argon to a suitably high pressure, filter it for removal of solid impurities, and store it under pressure as gas in a storage tank(s). An extensive distribution system of pipes, pressure regulators, and valves would deliver the argon gas to the various VTR systems and components where it would be utilized. The Argon Gas Distribution System would provide argon gas of suitable purity to:

- The reactor vessel cover gas region;
- The gap between the reactor and guard vessels;
- The cover gas regions of the secondary sodium expansion tanks and drain tanks;

- The gas space between the main and guard pipes of double-walled sodium piping;
- Driver fuel, assembly, test vehicle, and component transfer and dry storage casks; and
- Other processes requiring argon gas.

Wherever there would be a sodium system, there would be components and piping of the Argon Gas Distribution System. The system would include sodium vapor traps where needed.

Containment – The reactor and PHTS would not be enclosed inside of a containment dome structure.[9] The containment function is provided by the reactor head and the Head Access Area (see **Figure B–13**), which would be entirely below grade. Components of the Head Access Area that are part of the containment would include the area ceiling, walls, and floor; ventilation duct dampers; penetration isolation; isolation valves; and airlocks. Additionally, the outer piping of HRS double-walled piping provides containment in the event of a leak in the secondary sodium piping.

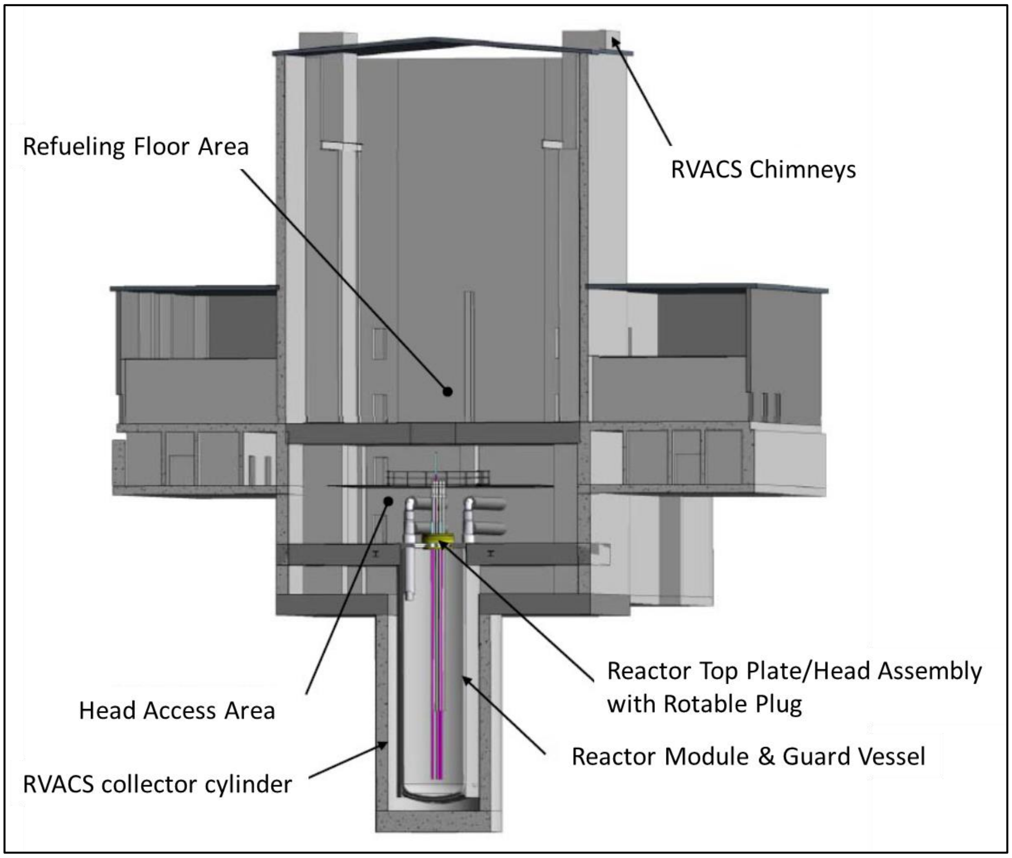


Figure B–13. View of the Versatile Test Reactor Operating Floor, Head Access Area, and Reactor

HVAC – The Reactor Facility HVAC System would provide heating, ventilation, and air conditioning for the various areas of the Reactor Facility during normal and off-normal conditions. The Reactor Facility HVAC System would also maintain humidity, pressure, and air cleanliness required for the areas served. The HVAC System would provide HVAC within the Reactor Facility by recirculating conditioned air or by once-through circulation of air. The Reactor Facility operating area and Experiment Hall and Head Access Area, as well as Reactor Facility electrical rooms would be heated and air conditioned, while other areas would

[9] The VTR operates at near atmospheric pressure. Even under post-accident conditions, reactor and containment pressures are near atmospheric. A large reinforced containment structure is not needed to prevent the release of radioactive elements to the environment under accident conditions.

be ventilated to remove heat loads with once-through circulation of air and heated with heaters, as required. Because of the potential for contamination, air from potentially contaminated spaces would be exhausted to the outside through charcoal adsorbers and high-efficiency particulate air (HEPA) filters to control the release of airborne radioactive gases and particles to the outside environment.

In-Vessel and Test Assembly Handling Systems – Movement of fuel and non-instrumented test assemblies within the reactor core would be accomplished using the IVTM. The IVTM would be used for all fuel, control, safety, reflector, and shield assemblies and non-instrumented test assembly transfer movements (except for control and safety rod movement into and out of the core) in the core, including:

- Retrieval of fresh assemblies from the transfer basket,
- Placement of fresh assemblies into the core,
- Removal of assemblies[10] from the core,
- Placement of spent driver fuel assemblies into a storage rack above the core,
- Placement of spent driver fuel assemblies into the outer row of the radial shield,
- Removal of spent driver fuel assemblies from the storage rack, and
- Placement of core assemblies in the transfer basket.

The IVTM would consist of three major parts: an upper ex-vessel drive section, a lower in-vessel section with a pantograph (a jointed framework), and a mechanical grappler. The IVTM is attached to the rotatable plug within the reactor top plate. The IVTM grappler could be positioned over any core position, over any in-vessel storage location outside of and above the core, and over the fuel transfer basket/station.

The In-Vessel Test Assembly Handling System would receive ELTA's and rabbit thimbles for transfer into and out of the reactor. This would be accomplished via a test assembly transfer cask, the building overhead bridge crane, the test assembly transfer adapter (designed to fit the test assembly ports on the rotatable plug), and the appropriate grapples and attachment mechanisms. The ELTA's/rabbit thimbles will occupy the six fixed positions provided on the rotatable plug. The In-Vessel Test Assembly Handling System would be required to:

- Raise and lock the ELTA's/rabbit thimbles into position above the core to avoid interference between test vehicles, the IVTM, and the core during refueling and experiment vehicle management;
- Unlock and lower the ELTA's/rabbit thimbles once refueling and other necessary movements are complete; and
- The ELTA's and rabbit thimbles will be designed to allow for tooling that will sever instrument cables/tubing/etc., from the ELTA's and rabbit thimbles, and the stalks can be removed, if required.

Ex-Vessel Fuel and Test Assembly Handling Systems – All fuel handling activities outside of the reactor vessel in the Reactor Facility are carried out on the Reactor and Experiment Hall operating floor, located above the reactor at grade level (see Figure B–2). The Ex-Vessel Fuel Handling System would receive fresh fuel as well as control, reflector, and shield assemblies and process spent fuel in preparation for shipment

[10] Test assemblies may be moved from the core to a storage location in the vessel to allow for decay-heat decrease before removal from the vessel. In-vessel storage would be required should the test assembly decay heat need to fall sufficiently to allow removal from the reactor vessel. All connections (power and instrumentation) would be severed before the assemblies could be moved to the storage locations. Removal of these assemblies would then be performed using the same procedure as that for any other assembly.

to a fuel treatment facility. Equipment required for Ex-Vessel Fuel Handling System operations would include:

- Assembly preheating station,
- Overhead bridge crane and assembly transfer cask,
- Fuel transfer adaptor, and
- Spent fuel washing station.

Fresh fuel would be received at the receiving and shipping area and transferred to a fresh fuel storage pit using the overhead crane. (The top of the pit is located at the floor level of the Reactor and Experiment Hall operating floor.) Prior to insertion in the core, each assembly would be transferred from the pit and placed inside a vertical preheating station filled with inert argon gas. The top of the preheating station would be at floor level of the Reactor and Experiment Hall operating floor and located near the fuel pits.

The building overhead bridge crane serving the operating floor and an overlying fuel transfer cask filled with argon would be used to transfer the fuel assembly from the preheating station to the reactor vessel. The fuel transfer cask may be capable of holding from one to three assemblies (the fuel transfer cask design has not been finalized). The bottom of the fuel transfer cask would incorporate a gate valve. A fuel transfer adaptor would be required to connect the fuel transfer cask with the fuel transport port in the reactor top plate (see Figure B–10). The fuel transfer cask would be relocated from the preheating station to the operating floor above the reactor upper head, using the building crane. Each assembly is transferred from the fuel transfer cask to the reactor using internal drives with the aid of a fuel transfer adapter. The adapter would be necessary because the fuel transfer cask is located at the refueling floor at the 0-foot elevation, and the upper head is located at the 29-foot elevation; these are connected by the transfer adapter, which would be filled with argon gas. The use of the adaptor allows for simplified movement through the Head Access Area, while protecting the reactor head and associated penetrations from potential impacts from facility cask movements. A floor valve, located at the top of the adapter, when open, would provide a conduit for lowering the assembly through the upper head and fuel transfer port into the reactor vessel transfer basket below the sodium surface.

Spent driver fuel assemblies would be removed from the reactor vessel transfer basket using the same equipment and would be transferred to a washing station located on the Reactor and Experiment Hall operating floor. Fuel pits, below the operating floor (top of pits at floor level), would be available to temporarily hold spent driver fuel assemblies after washing. The residual sodium would be removed by reacting it under tightly controlled environmental conditions and reaction rates in the washing station. The washing station top is located at floor level. A combination of nitrogen and demineralized water moisture would be used to remove sodium from the driver fuel assembly. The reaction of sodium with moisture creates hydrogen gas, as well as sodium hydroxide. The sodium hydroxide would be washed off the assembly surfaces with demineralized water. Water containing sodium hydroxide and radionuclides would be collected by the Liquid Radioactive Waste System. The assembly would be dried with heated inert gas. After washing and drying, the spent driver fuel assemblies would be loaded into transfer casks for interim storage at the fuel storage pad and eventual transfer to a fuel treatment facility. Gas containing hydrogen and radionuclides would be collected by the Gaseous Radioactive Waste System.

The Ex-Vessel Test Assembly Handling System would be similar to the Ex-Vessel Fuel Assembly Handling System. The requirements for the system would vary depending upon the content and configuration of the test assembly. However, some test assemblies would require preheating, so preheating capability would be available in some of the assembly preparation stations. The overhead crane, test assembly transfer casks, test assembly transfer adaptors (designed to fit the test assembly ports on the rotatable plug), and a test assembly washing station would all be required. Due to the length of some test assemblies, longer than fuel assemblies (e.g., ELTA), the preparation and cleaning stations and the transfer casks for this system would be taller than those for the Ex-Vessel Fuel Handling System.

As with ex-vessel fuel movements, all test assembly handling would take place on the Reactor and Experiment Hall operating floor.

Radioactive Waste Systems (Gaseous, Liquid, and Solid) – This system has not been fully designed. The following provides a conceptual design for the Gaseous Radioactive Waste System (see **Figure B–14**). The system receives radioactive argon cover gas from the reactor, radioactive argon cover gas from sodium components, and radioactive nitrogen from washing of residual sodium off of sodium components, as well as off-gas from processes. The radioactive gas would be filtered, and radionuclides such as xenon would be adsorbed and held in charcoal filters to decay. After sufficient treatment and holding time, gaseous effluents would be passed through multiple stages of HEPA filters before being released to the environment via an exhaust stack.

The system would receive air and cover gases from all VTR building systems, including radioactive reactor cover gas, sodium component cover gas, and process off-gas. The system would be sized to support the sodium removal and decontamination of a driver fuel assembly with failed fuel or a failed experiment vehicle in addition to maintenance activities.

Radioactive gases would be initially collected in a Holdup/Sampling Tank where unfiltered gas samples could be collected. Downstream of the Holdup/Sampling Tank, the Transfer Tank, a high-pressure tank, is used to maintain a constant system pressure. Located downstream of the Transfer Tank, the treatment system would consist of two 100-percent-capacity trains containing moisture separators, upstream and downstream HEPA filters, and charcoal-adsorption delay beds. A Secondary Hold-up/Sampling Tank would be located between the filtration components and the HVAC stack and would be the point where filtered gas samples could be collected. Compressors (two 100-percent capacity between the Holdup/Sampling Tank and the Transfer Tank and two 100-percent capacity downstream of the second set of HEPA filters) would provide the motive force for gases through the system.

The Liquid Radioactive Waste System would provide for collection and processing of radioactive liquid wastes from sodium removal, decontamination, equipment and area washing, and showers/washes. Through a series of pipes and drains, the radioactive liquid wastes would be collected in collection tanks, pumped through cartridge filters (two 100-percent-capacity trains), as required by treatment facility acceptance criteria, and held up in storage tanks for export via truck to be processed outside of the VTR. The Liquid Radioactive Waste System would incorporate a demineralized water supply system. Demineralized water would be provided to the moist gas generator for removal of sodium via interaction with moist gas inside the sodium washing station and other facility users. After use, the contaminated water would be collected as part of the liquid radioactive waste. The cartridge filters would be processed as solid radioactive waste.

The Solid Radioactive Waste System would receive solid radioactive waste from the other plant systems, perform any size reduction required, package the waste, and temporarily store the waste before final export from the VTR facility. The storage area would provide one outage (25 days or less) of storage space. The system would be monitored locally to ensure operating conditions are within specified parameters and that the system is configured appropriately.

***Sodium Fire Protection System* –** The Sodium Fire Protection System would include instrumentation/ detectors to detect sodium leaks and sodium fires, portable fire extinguishers for fighting sodium fires of limited size by personnel, and design features to mitigate against the effects of postulated bounding and conservative sodium fire scenarios.

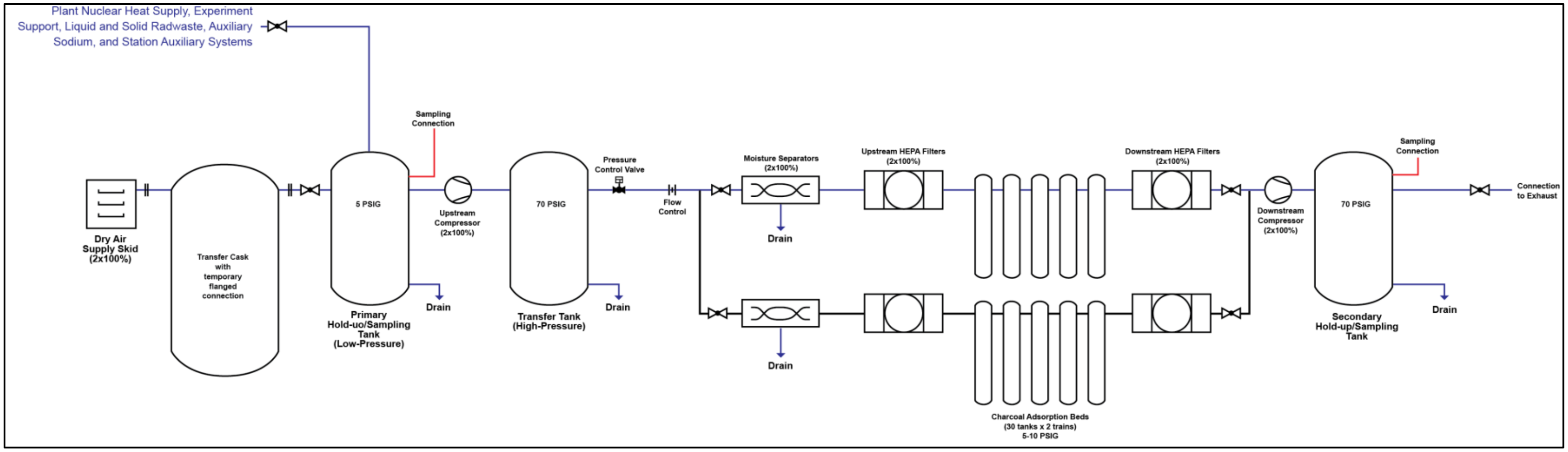


Figure B–14. Gaseous Waste Management System

Due to the low system operating pressures, any sodium leaks are expected to start as small weeping leaks in a "leak before break" failure mode. Sodium leak detectors would be provided to detect sodium leaks while they are still small, such that the affected pipe or component could be removed from service and repaired before the hole grew to a significantly larger size. For these small leaks or leaks of sodium limited in size, portable fire extinguishers containing dry powder would be provided inside areas containing sodium piping and components. If the sodium is accessible (e.g., has leaked from thermal insulation), can be observed to be burning, extinguishment is judged to be the correct action, and fighting the fire can be done safely, personnel can use the fire extinguishers to extinguish such limited fires.

In addition to the potential fire-related damage, sodium fires can result in the generation of harmful aerosols (sodium peroxide and sodium oxide) and sodium hydroxide (from chemical reaction with water) and sodium carbide (from chemical reaction with carbon dioxide), both of which are corrosive. Extinguishing a sodium fire terminates and limits the generation of these hazards.

The installation of steel catch pans or steel basins on the floor would be a mitigation design feature that would prevent released sodium from directly interacting with the concrete floor. Upon being heated, concrete could release water that would chemically interact with metallic sodium, forming hydrogen. Typically, a steel catch pan would be sized to hold more than the maximum volume of sodium that can potentially leak into a room.

As noted above, all sodium leaks are expected to start as small weeping leaks and to be detected in time such that the amount of sodium leaked remains small. However, the Sodium Fire Protection System would be designed to accommodate postulated bounding and conservative sodium-release scenarios, in which the total inventory of sodium that can potentially leak is assumed to be released.

Specific design features for preventing and mitigating sodium leaks and fires would include double-walled piping on the secondary sodium inlet and outlet main pipes from inside of the reactor Head Access Area room to the secondary pump rooms. Sodium released from a postulated leak in the main pipe would be collected in the leak-monitored and inert-gas space between the two pipes and drained into an inerted sodium collection tank, which is located inside of a vault beneath the loop sodium drain tank room. Sodium leaking outside the piping system would flow onto a catch pan on the floor with pan drains leading to the sodium collection tank. The sodium collection tank would incorporate a perforated plate with a significantly reduced area for air flow near the top, to reduce the transport of oxygen to the sodium pool surface and thereby reduce the sodium burning rate. The sodium collection tank would incorporate a vent for heated gas, would be trace heated to prevent the condensation of water moisture from air, and would enable collected sodium to be heated and melted. The piping delivering sodium to the sodium collection tank would also be trace heated to prevent sodium from freezing inside of the piping.

Sodium Purification Systems – An in-vessel Primary Sodium Purification System for the VTR is in the conceptual design phase. The Primary Sodium Purification System would remove impurities (mainly oxygen) above an established level from the PHTS sodium to maintain a desired level of purity. It also would remove radionuclides, primarily cesium, that may be released from failed fuel. The system would be a module that is installed inside the reactor vessel and would consist of two integrated purification units with a cold trap cartridge and a cesium trap cartridge. The integrated purification unit largely consists of a sodium pump, regenerative heat exchanger, non-regenerative heat exchanger, removable cartridges (to be replaced as necessary to ensure filtration capability), sodium piping, and nitrogen piping associated with the non-regenerative heat exchanger. Except for the portion which accepts insertion of a cold trap or cesium trap cartridge, the components within an integrated purification unit are largely contained within an argon-inerted and sealed vessel. To remove the necessary heat from the sodium for purification, each integrated purification unit would be associated with a closed nitrogen loop with a blower, which cools the heat exchanger and a nitrogen-to-air heat exchanger and air blower to cool the nitrogen loop. Additional concepts for outside reactor vessel cleanup, either temporarily or permanently installed, may be explored as the design progresses.

The Secondary Sodium Purification System would remove impurities above an established level (mainly oxygen) from the secondary HRS sodium to maintain a desired level of purity. A separate purification system would be provided for each of the two secondary sodium loops. The system would also support initial fill and sodium-inventory-control operations for both the PHTS and HRS. The purification system for each secondary HRS sodium loop would be equipped with the following components:

- an EM sodium pump separate from the main-loop EM sodium pumps,
- an economizer (i.e., a regenerative heat exchanger) that partially cools sodium upstream of the cold trap via heat exchange to cooler sodium exiting the cold trap,
- a cold trap in which excess oxygen is crystallized to form sodium oxide that deposits upon a structure (e.g., a stainless-steel mesh packing) inside of the cold trap,
- a cold trap air-cooling circuit incorporating an air blower,
- a plugging temperature indicator with an air-cooling circuit,
- interconnecting piping and valves,
- instrumentation, and
- valve control actuators.

The system would receive unprocessed sodium from the secondary HRS sodium loop upstream of the main-loop EM sodium pumps and from the loop drain tank. The sodium would flow through the economizer and cold trap. The system also would incorporate piping to direct a portion of the sodium flow through the plugging temperature indicator/plugging meter. Following removal or measurement of impurities, the sodium would be returned to the secondary HRS loop at the loop expansion tank. Grab samples can be taken for analysis of the radionuclides and chemical impurities present in the sodium.

B.2.9 Operations

The nominal test-cycle length for the VTR would be 100 effective full-power days, followed by a nominal 20-day refueling outage. Driver fuel assemblies would remain in the core for a number of cycles. Those further out from the core centerline would be subjected to a lower neutron flux and undergo a slower rate of burnup. Consequently, they could be left in the core for a greater number of cycles. The goal is to achieve approximately the same mean discharge burnup in all driver fuel assemblies. A VTR driver fuel assembly may be left in the core for three, four, five, or six cycles.

The VTR test cycle would require 14 to 15 fresh driver fuel assemblies for each 100-day cycle (INL 2020c). Fresh driver fuel assemblies would be delivered by truck into the truck bay at grade level. Fresh driver fuel assemblies could be stored in fuel cask pits beneath the Reactor Operating Room floor or loaded directly into the reactor vessel. The operating area above the reactor would be a long Experiment Hall interconnected to a truck bay. The operating floor inside of the Reactor Facility would be at grade level, as shown in Figure B–2. Prior to insertion into the reactor vessel, each fresh driver fuel assembly would be properly preheated to melt the sodium to form the sodium bond with the fuel before being transferred into the reactor sodium pool. Preheating prevents thermal shock to the cold assembly when it is lowered into the sodium pool and ensures that the bond sodium in the fuel pins heats from the free surface down. Following preheating and cleaning, the assembly would be raised into a heated fuel transfer cask and moved to the reactor using the overhead bridge crane. The preheated and cleaned fresh assembly would be lowered through the fuel transport port into the transfer basket, from which it is removed and placed in the core by the IVTM.

Spent driver fuel assemblies would be transferred from the core to the spent fuel storage locations within the reactor vessel (either within the outer ring of shield assemblies or above and outside the core at the level of the intermediate heat exchanger) using the IVTM. A spent fuel assembly would be stored in-vessel for a year or more, while its decay heat power level falls below a specified value. When sufficiently cooled, a spent driver fuel assembly would be raised from the transfer basket below the sodium level, through the fuel transport port in the reactor top plate, and placed inside a fuel transfer cask with an inert atmosphere and cooled by natural circulation. Movement of the spent driver fuel while on the Reactor and Experiment Hall operating floor has been discussed in Section B.2.8.

The overhead bridge crane would be used to move the fuel transfer cask to the sodium wash station. Residual sodium would be removed from the assembly inside of the wash station vessel by first exposing the assembly to inert nitrogen gas containing demineralized water moisture and then with demineralized water. Waste water containing sodium hydroxide and radionuclides would be collected by the Liquid Radioactive Waste System, while nitrogen containing hydrogen and radionuclides would be collected by the Gaseous Radioactive Waste System. The assembly would be dried with heated nitrogen gas and then raised up inside of an inerted dry storage/transfer cask which may hold up to six assemblies (cask design is not final). Clean and dried spent driver fuel assemblies would be transferred to a fuel storage pad for interim storage. At the storage pad, spent driver fuel assemblies would be stored in each spent fuel cask until decayed sufficiently to allow for fuel treatment, for a period of at least 3 years. Driver fuel assemblies would be stored for less than 5 years. At that time, the spent driver fuel assemblies would be transferred to a spent fuel treatment facility in preparation for ultimate storage. Spent fuel treatment and storage is discussed in Section B.4.

During refueling outages, it may be necessary to raise ELTA's and RTA's out of the core to an elevation sufficiently high above the core and lock them in the raised position to avoid interference with refueling operations. This is described in Section B.2.8, above.

ELTA, RTA, and NTA insertion and removal from the core follows a procedure very similar to that used for fresh and spent driver fuel assemblies. However, differences include:

- ELTA/RTA/NTA preparation would be required before preheating and cleaning;
- ELTA's and RTA's (up to 65 feet tall with the instrumentation stalk) require a tall test vehicle transfer cask; and
- ELTA's and RTA's would be inserted directly into the core through the test assembly penetrations in the rotatable plug, not through the transport port into a transfer basket.

ELTA's and RTA's could be:

- Removed directly from the core and transferred to a tall sodium wash station, or
- Disconnected from the assembly stalks and then moved using the IVTM to a transfer basket; and
- May be examined in Experiment Hall facilities.

The Reactor Facility layout facilitates ex-vessel test vehicle handling. The operating floor area at the reactor would be at grade level and open to a long Experiment Hall along the length of the building. The Experiment Hall would include an experiment support/preparation area. Test vehicles, particularly the ELTA's and RTA's, would be prepared in the horizontal position attached to a strongback. When ready, the test vehicle would be raised to a vertical position and placed inside of a deep pit. From the pit, the test assembly would be transferred to the reactor core in a tall transfer cask in a manner similar to that used for fresh driver fuel assemblies.

At the end of their irradiation, test assemblies would be removed from the reactor vessel and transferred to a washing station. The stalks from ELTA's and RTA's are particularly long. A separate sodium washing station, or purpose-built wash coffin connecting the fuel wash station incorporating a great height, would need to be included to remove residual sodium from stalks or complete test vehicles in which the ELTA or RTA is connected to its stalk. Movement of stalks or complete test vehicles from the rotatable plug to the washing station would be carried out using a tall test vehicle transfer cask. Alternatively, the stalks could be removed and sectioned/cut, and washed in the fuel wash station.

At the end of their irradiation, instrumented test assemblies may have a significant decay heat similar to fuel and may require in-vessel storage while their decay heat falls. While NTA's can be handled similar to spent fuel assemblies, the stalk of ELTA's must be disconnected or severed. Once the stalk is disconnected, the ELTA would be handled in the same manner as described above for a spent fuel assembly, when being transferred to a shielded cell.

The Experiment Hall would incorporate a shielded cell located in a pit for prompt robotic post-test examination of test assemblies. The sequencing of removing residual sodium may be specific to the particular experiment and the intent of the experimenters. The ELTA stalks may be removed from the lower test vehicle portion inside of the shielded cell to make it suitable for shipment to a DOE facility for post-irradiation examination.

B.2.10 Versatile Test Reactor at the Idaho National Laboratory Site

At the INL Site, the VTR would be built adjacent to and east of the Fuel Manufacturing Facility (FMF) and Zero Power Physics Reactor (ZPPR) protected area at the Materials and Fuels Complex (MFC). The protected area PIDAS would be extended to encompass most of the VTR structures. Construction of the VTR has been estimated to take approximately 51 months, once design activities are complete. Based on the layout of the VTR (see Section B.2.2), the VTR complex at INL would occupy about 25 acres. During construction, an additional 75 acres would be required for temporary parking and equipment laydown, assembly, and staging. About 100 acres would be impacted by VTR construction (see **Figure B–15**). There is a pygmy rabbit burrow located on the southern edge of the construction disturbance area. Chapter 3, Section 3.1.5.3 identifies this area and Chapter 4, Section 4.5.1, discusses limitations for activities in the vicinity of the pygmy rabbit burrow.

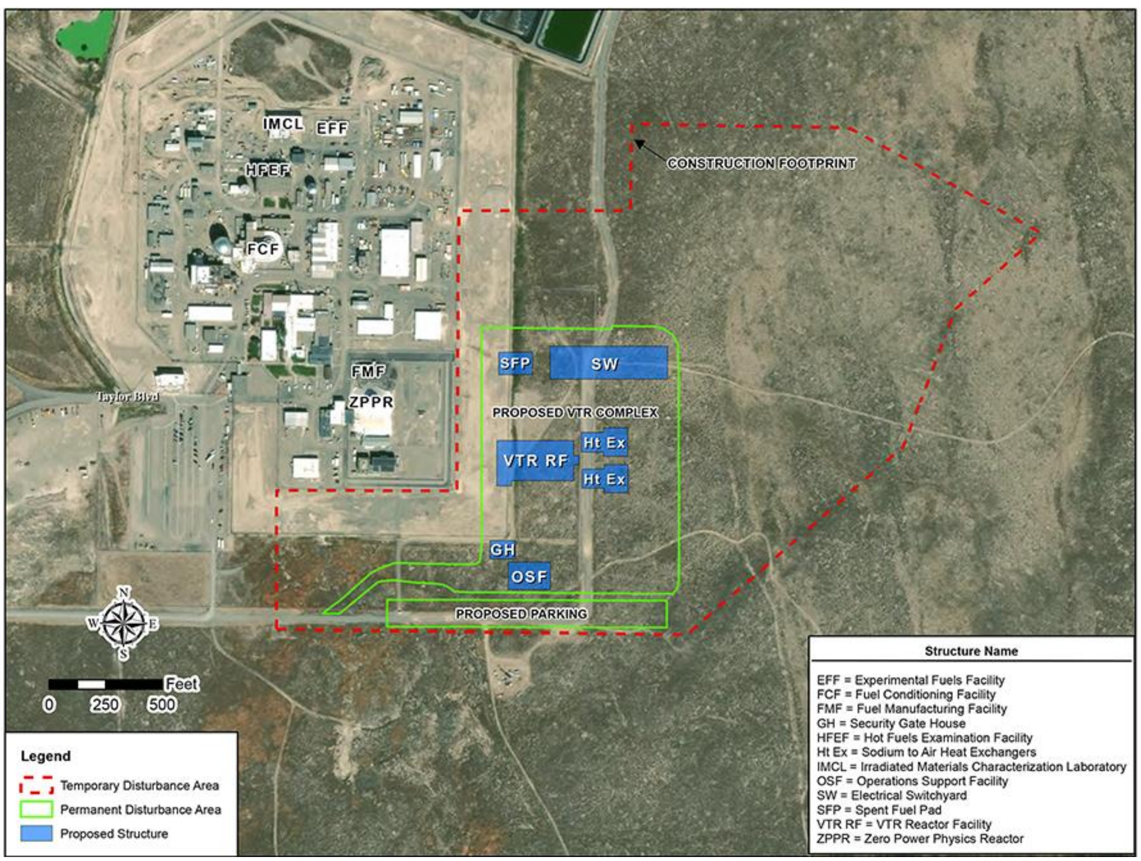


Figure B–15. Proposed Versatile Test Reactor Location at Idaho National Laboratory

VTR utility demands (electricity, water, etc.) would be supplied by existing MFC utility systems. With one exception, no modifications to the MFC utility systems would be required to support the addition of the VTR. The addition of the VTR to the MFC would require an upgrade to the electrical distribution system at the INL Site. A dynamic volt-ampere reactive device would be installed at the Advanced Test Reactor electrical substation to ensure electrical (voltage) stability for the area.

B.2.10.1 Environmental Resources – Construction

Resource Requirements

Table B–8 provides a summary of the key resources committed to the construction of the VTR facilities. The construction effort would ramp up until peaking in the third year of construction.

Table B–8. Idaho National Laboratory Resource Requirements During Versatile Test Reactor Construction

Resource	*Units*	*Annual Average Value*	*Annual Peak Value*	*Total* [a]
Staff	FTE	640	1300	2,700
Electricity	kWh	1,000,000	2,000,000	4,300,000
Gasoline	gallons	87,000	145,000	370,000
Diesel Fuel				
Road Diesel	gallons	84,000	144,000	360,000
Non-road Diesel	gallons	447,000	750,000	1,900,000
Total Diesel	gallons	531,000	894,000	2,300,000
Water				
Potable	gallons	8,000,000	16,000,000	34,000,000
Dust control, etc.	gallons	22,000,000	40,000,000	94,000,000
Total	gallons	30,000,000	56,000,000	128,000,000
Asphalt	cubic yards	---	---	1,400
Structural Concrete	cubic yards	---	---	40,000
Rebar	tons	---	---	4,350
Excavation	bank cubic yards [b]	---	---	135,000
Backfill Material	cubic yards	---	---	200,000 [c]
Landscaping	cubic yards	---	---	2,000
Structural Steel	tons	---	---	4,150
Large Bore Piping	linear feet	---	---	31,500
Cable and Wire	linear feet	---	---	1,200,000
Cable Tray	linear feet	---	---	18,000
Conduit Above Grade	linear feet	---	---	220,000
Conduit Inside Duct Banks	linear feet	---	---	53,000
Rock/Gravel	cubic yards	---	---	45,000
Temporary Concrete	cubic yards	---	---	14,000
Lumber	tons	---	---	250
Temporary Steel	tons	---	---	50
Gas [d]	bottles/cubic meters	---	---	20,000/130,000

FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] Construction duration of 51 months is assumed.

[b] A bank yard is the volume of earth or rock in its natural state, as compared to the expanded volume after excavation.

[c] Excavated material would be temporarily stored within the construction footprint and would be used as backfill. Material from a borrow site would be used for the additional 65,000 cubic yards needed.

[d] Gas bottles (cylinders) can range from 2 to 10 cubic meters in size. A typical size of 6.5 cubic meters has been used to estimate the volume of gas in the cylinders.

Source: INL 2020c.

Nonradiological Releases

Nonradiological releases are primarily associated with the operation of trucks and construction equipment (i.e., the burning of diesel fuel). However, fugitive dust contributes the majority of particulate matter emissions. Emission sources and air pollutant emissions are presented in **Table B–9**.

Table B–9. Calendar Year Nonradiological Construction Emissions – Idaho National Laboratory Versatile Test Reactor

Calendar Year/Source Type	*Air Pollutant Emissions (tons per year)*						
	VOC	*CO*	*NO_x*	*SO_2*	*PM_{10}*	*$PM_{2.5}$*	*CO_2e (metric tons)*
Year 2022							
Onsite On-road Sources	0.05	1.00	0.48	0.002	0.06	0.02	261
Onsite Nonroad Sources	0.35	2.47	4.66	0.01	0.27	0.27	1,614
Fugitive Dust	---	---	---	---	56.78	5.68	---
Offsite On-road Sources	0.08	5.12	1.00	0.006	0.20	0.05	761
Total Annual Emissions	**0.48**	**8.59**	**6.13**	**0.02**	**57.31**	**6.01**	**2,637**
Year 2023							
Onsite On-road Sources	0.08	1.46	0.78	0.004	0.09	0.04	445
Onsite Nonroad Sources	0.73	4.61	8.59	0.02	0.47	0.45	2,755
Fugitive Dust	---	---	---	---	102.21	10.22	---
Offsite On-road Sources	0.36	24.37	4.28	0.03	0.95	0.22	3,666
Total Annual Emissions	**1.16**	**30.44**	**13.64**	**0.05**	**103.72**	**10.93**	**6,866**
Year 2024							
Onsite On-road Sources	0.06	1.27	0.61	0.003	0.08	0.03	393
Onsite Nonroad Sources	0.68	4.16	8.50	0.02	0.43	0.41	2,773
Fugitive Dust	---	---	---	---	68.14	6.81	---
Offsite On-road Sources	0.32	24.32	3.91	0.03	0.98	0.22	3,763
Total Annual Emissions	**1.06**	**29.75**	**13.03**	**0.05**	**69.62**	**7.47**	**6,929**
Year 2025							
Onsite On-road Sources	0.02	0.73	0.22	0.002	0.04	0.01	182
Onsite Nonroad Sources	0.21	1.50	2.50	0.01	0.13	0.13	1,051
Fugitive Dust	---	---	---	---	34.07	3.41	---
Offsite On-road Sources	0.03	1.09	0.50	0.00	0.10	0.02	336
Total Annual Emissions	**0.26**	**3.32**	**3.21**	**0.01**	**34.33**	**3.57**	**1,569**

CO = carbon monoxide; CO_2e = carbon dioxide equivalent; NA = not applicable; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound; --- = no air pollutant emission from this source type.
Source: Derived from INL 2020c.

Waste Generation

Table B–10 provides estimates of the wastes generated during VTR construction; this includes construction of all of the facilities (Reactor Facility, switchyard, exterior HRS components, the Operational Support Facility, and associated structures). There would not be any radiological waste generated during construction of the VTR.

Table B–10. Wastes Generated During Versatile Test Reactor Construction

Waste Type	Material	Units	Value
Hazardous Waste			Assumed to be 2 percent of nonhazardous waste volumes
Nonhazardous Waste	Concrete	cubic yards	9,900
	Rebar	tons	180
	Structural steel	tons	330
	Large bore pipe	linear feet	2,500
	Small bore pipe	linear feet	2,800
	Cable and wire	linear feet	96,000
	Cable tray	linear feet	1,400
	Conduit	linear feet	26,000
	Tubing	linear feet	2,800
	Instruments	each	65
	Valves	each	30
	In-line components	each	65
	Lumber	tons	120
	Steel	tons	50
	Gas bottles	bottles	19,200

Source: INL 2020c.

B.2.10.2 Environmental Resources – Operations

The nominal test cycle length for the VTR would be 100 effective full-power days. At the end of each cycle there would be a 20-day refueling cycle during which 14 to 15 driver fuel assemblies and test assemblies at the end of their planned test exposure times would be removed from the core (INL 2020c).

Resource Requirements

Key annual resource commitments for the operation of the VTR are provided in **Table B–11**. Annual staffing requirements include both the normal operational and maintenance staff for the VTR, as well as augmented staffing during refueling. Diesel fuel would be required for testing of the site diesel generators, and electric or propane heaters would be used as the heat source for the SAHX air pre-heaters. Since the VTR would be a sodium-cooled reactor, both the PHTS and HRS would use sodium coolant. The commitment of water would be required only for staff needs and firewater (system testing, etc.). No water would be used for cooling the reactor. Only chemicals used in quantities of over 1,000 pounds are shown in the table. Other chemicals would be used in smaller quantities (INL 2020d).

Table B–11. Annual Resource Requirements During Versatile Test Reactor Operation

Resource	Units	Value Annual (Peak)
Staff	FTE	200
Electricity [a]	MWh	140,000 (170,000)
Diesel Fuel [b]	gallons	9,200
Propane [c]	Standard cubic feet	18,500 (1,500,000)
Water		
Potable	gallons	1,200,000
Fire Water	gallons	1,700,000
Demineralized Water	gallons	250,000
Total	gallons	3,100,000
Chemicals		
Sulfuric Acid	pounds	640,000
Gasoline	pounds	79,000
Oil	pounds	59,000

Resource	Units	Value
		Annual (Peak)
Fuel Maintenance	pounds	20,000
Paint	pounds	10,000
Alcohol	pounds	13,000
Vehicle Maintenance	pounds	8,000
Adhesive	pounds	7,000
Cleaner	pounds	7,500
Building Maintenance	pounds	3,000
Lubricant	pounds	9,400
Sealant	pounds	2,500
Acetone	pounds	2,200
Grounds Keeping	pounds	1,900
Metal Cleaner	pounds	2,000
Coolant	pounds	1,400
Sodium Hypochlorite	pounds	1,200
Nitric Acid	pounds	6,400
Ammonium Hydrozide	pounds	7,000
Epoxy	pounds	3,400
Antifreeze	pounds	1,700
Caulk	pounds	1,300
Gases		
Compressed Neon	liters	23,000
Suva Refrigerant	pounds	5,200
Liquid Nitrogen	standard cubic feet	3,400
P-10 Gas (argon with 10% methane)	standard cubic feet	3,100
Methane	standard cubic feet	2,900
Freon (R-410a)	pounds	1,800
Hydrogen/Air Mix	liters	1,800
Compressed Helium	standard cubic feet	1,500
Compressed Oxygen	standard cubic feet	1,200

FTE = full-time equivalent (person); MWh= megawatt-hours.

[a] Annual electricity usage was provided in MVa (mega-volt-amperes). A load factor of .9 was used to convert to MWs (megawatts).

[b] Diesel generators would operate 1 percent of the time, 88 hours per year. Fuel consumption is based on the fuel consumption rates (Leidos 2020).

[c] Propane heaters are an alternative design for preheating air in the sodium-to-air heat exchangers. Use of this alternative design would be a site-specific decision. These heaters would be used for short periods when the reactor is shutdown following a test cycle. The peak usage is associated with an extended maintenance outage, projected to be needed once every 15 years.

Source: GE Hitachi 2019b; INL 2020c.

Nonradiological Releases

The main source of nonradiological releases associated with the operation of the VTR would be the releases from operation of the site diesel generators, personal vehicles, and vehicles used to transport materials (wastes, spent fuel, test assemblies, etc.). The generators supply power to the site in the event of a loss of the normal offsite power supply. To ensure that the generators are functional, they would be tested, started and run for a period of time, several times a year. The annual emissions associated with these sources are provided in **Table B–12**.

Table B–12. Versatile Test Reactor Operational Nonradiological Emissions

Emission Source	Air Pollutant Emissions (tons per year)							
	VOC	CO	NO_x	SO_2	PM_{10}	$PM_{2.5}$	CO_2	CO_2e (metric tons)
Back-up Generators – VTR	0.03	0.50	0.10	0.00	0.00	0.00	102	93
Pre-Heaters – Normal Annual	0.00	0.00	0.00	0.00	0.00	0.00	3	3
Haul Trucks	0.03	0.15	0.55	0.00	0.08	0.02	305	277
Worker Commuter Vehicles	0.02	2.85	0.18	0.00	0.08	0.02	382	347
Total – Normal Annual Operations	**0.09**	**3.50**	**0.84**	**0.01**	**0.17**	**0.04**	**793**	**720**
Pre-Heaters –Large Component Replacement [a]	0.02	0.16	0.27	0.00	0.01	0.01	263	239
Total Annual Emissions [b]	**0.11**	**3.66**	**1.11**	**0.01**	**0.18**	**0.06**	**1,052**	**956**

CO = carbon monoxide; CO_2e = carbon dioxide equivalent; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.

[a] Large Component Replacement would occur every 15 years.

[b] Equal to sum of Back-up Generators, Haul Trucks, Worker Commuter Vehicles, and Pre-Heaters Large Component Replacement.

Source: Derived from INL 2020d.

Radiological Releases

Radiological releases were estimated assuming that the VTR operates for three test cycles per year of 100 days each, with one failed fuel pin in the core at all times. The estimated annual release activity per isotope is presented in **Table B–13**.

Table B–13. Versatile Test Reactor Operational Annual Radiological Releases

Isotope	Annual Release (curies)	Isotope	Annual Release (curies)
Argon-41 [a]	27.1	Krypton-88	8.9×10^{-06}
Cesium-135	9.0×10^{-16}	Xenon-131m	1.6×10^{-02}
Cesium-137	1.2×10^{-12}	Xenon-133	1.0×10^{-03}
Cesium-138	2.0×10^{-06}	Xenon-133m	5.4×10^{-07}
Hydrogen-3 (Tritium)	1.2	Xenon-135	4.2×10^{-05}
Krypton-83m	1.8×10^{-06}	Xenon-135m	1.5×10^{-06}
Krypton-85	0.70	Xenon-137	7.4×10^{-07}
Krypton-85m	3.5×10^{-06}	Xenon-138	4.4×10^{-06}
Krypton-87	4.8×10^{-06}		

[a] Most of the release of argon (27 curies) is through the RVACS stacks. The rest (0.01 curies) is through the facility HVAC stacks.

Source: INL 2020c.

Note that currently the only anticipated normal operation releases of radioactivity to the environment, with the exception of most of the argon, would be from the Gaseous Radioactive Waste System. The release from the Gaseous Radioactive Waste System would be inserted into the radioactive waste area HVAC system exhaust. The combined flow rate would be about 2,400 cubic meters per minute, at approximately 105 °F. The HEPA-filtered release would be through a 24-inch diameter stack, at a height of about 99 feet. The HVAC systems, Liquid Radioactive Waste System, and Solid Radioactive Waste System are not anticipated to have appreciable releases to the environment. The unfiltered releases of argon from activated air would be from the RVACS stacks would be from four 7-foot diameter RVACS stacks, outer diameter, at an elevation of approximately 98 feet, with a total flow rate of 1,000 cubic meters per minute, at a temperature less than 500 °F (INL 2020c).

Waste Generation

Annual waste generation rates, based on three test cycles per year, are presented in **Table B–14**.

Table B–14. Versatile Test Reactor Operational Annual Waste Generation

Waste Type	Category	Annual Average Volume (cubic meters)		Average Weight Maximum	
		Net	Gross	Net	Gross
Hazardous waste	NA	3.2	4.4	5,400	6,500
Industrial	NA	22	26	27,000	30,000
Universal	NA	0.88	0.99	420	490
TSCA	NA	2.3	2.4	1,300	1,900
Recyclable	NA	4.5	6.0	9,700	11,000
Low-level waste	Contact handled	160	180	58,000	98,000
Mixed low-level waste	Contact handled	4.7	5.9	7,000	8,800
	Remote handled	0.7	1.7	280	4,700
Waste Type	**Unit**	**Quantity**			
Driver fuel assemblies	assemblies	45/66 [a]			
Liquid low-level waste	gallons	250,000			
Sanitary waste	gallons	1,200,000			

NA = not applicable; TSCA = Toxic Substance Control Act material.

[a] Up to 45 assemblies could be removed during a single year consisting of three operational cycles. Sixty-six assemblies would be removed from the VTR when the final core is removed.

Source: INL 2017b, 2020d; GE Hitachi 2019b.

B.2.11 Versatile Test Reactor at Oak Ridge National Laboratory

At Oak Ridge National Laboratory (ORNL), the VTR would be built approximately a mile east of the High Flux Isotope Reactor complex. Construction of the VTR has been estimated to take approximately 51 months, once design activities are complete. Based on the layout of the VTR (see Section B.2.2), the VTR complex at ORNL would occupy about 25 acres. However, in addition to the construction of the VTR, additional test assembly examination and spent fuel treatment and storage facilities would be constructed at ORNL (see Sections B.3.4 and B.4.4). These facilities would be collocated with the VTR, and in total, would result in a land commitment to the VTR and facilities of less than 50 acres. The test assembly examination and spent fuel treatment facility, spent fuel pad, and most of the VTR structures would be enclosed in a PIDAS. During construction, an additional 100 acres would be required for temporary parking and equipment laydown, assembly, and staging. In total, up to 150 acres would be impacted by the VTR, test assembly examination facility, and fuel storage pad construction (see **Figure B–16**).

VTR utility demands (electricity, water, etc.) would be supplied by existing ORNL utility systems. Once connected, no modifications to the ORNL utility systems would be required to support the addition of the VTR.

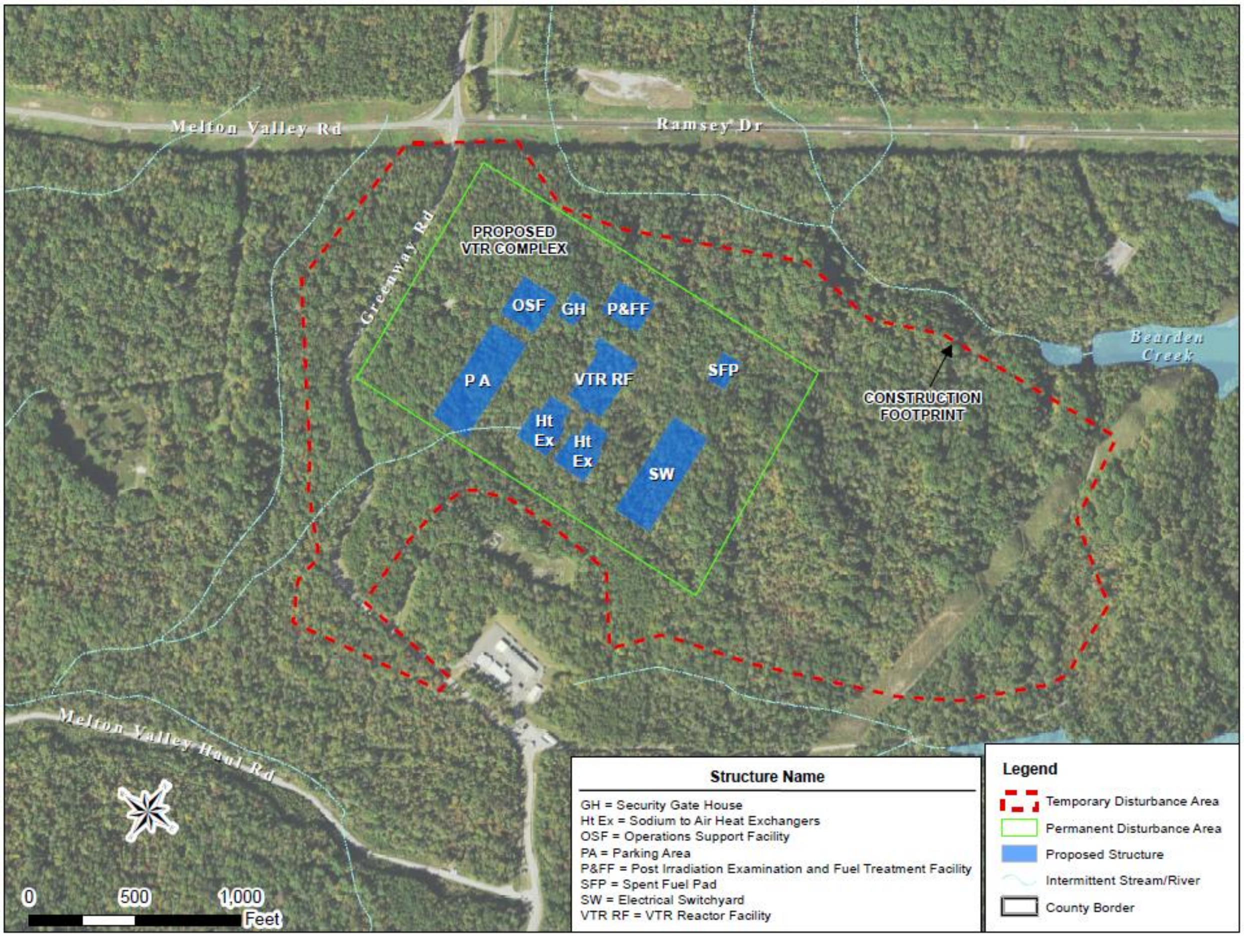


Figure B–16. Proposed Versatile Test Reactor Location at Oak Ridge National Laboratory

B.2.11.1 Environmental Resources – Construction

Resource Requirements

The environmental resources required or affected by construction of the VTR at ORNL would be similar to those described for the INL Site in Section B.2.10.1, but would include resources required for site preparation of an undisturbed, wooded area. Unlike at the INL Site, trees would need to be removed and the site more extensively graded. Resource requirements for site preparation at ORNL are presented in **Table B–15**. Once the site is prepared, resources required for the construction of the VTR facilities (VTR Reactor Facility, switchyard, sodium-to-air heat exchangers, Operational Support Facility, etc.) would be the same as those presented for VTR construction at INL (see Table B–8), with two exceptions. Construction at ORNL would involve the construction of a shorter road from existing roads to the facility parking lot, this results in a reduction in the use of asphalt (about 400 cubic yards less).[11] The construction activities at ORNL would include construction of the test assembly examination and spent fuel treatment facility. The resources affected by construction of this facility are discussed in Sections B.3.4 and B.4.4, respectively.

[11] Differences between INL and ORNL access road and parking lot construction resource utilizations for other resources are small and do not change the values presented in Table B–8.

Table B–15. Resource Requirements During Versatile Test Reactor Site Preparation at Oak Ridge National Laboratory

Site Preparation				
Resource	***Units***	***Annual Average Value***	***Annual Peak Value***	***Total*** [a]
Staff	FTE	16	NA	16
Diesel Fuel				
Road Diesel	gallons	25,000	NA	25,000
Non-road Diesel	gallons	244,000	NA	244,000
Total Diesel	gallons	269,000	NA	269,000
Gasoline	gallons	300	NA	300
Water				
Potable Water	gallons	250,000	NA	250,000
Dust Control	gallons	140,000	NA	140,000
Total Water	gallons	390,000	NA	390,000
Excavation [b]	cubic yards	690,000	NA	690,000
Fill Material	cubic yards	29,000	NA	720,000

FTE = full-time equivalent (person); NA = not applicable.

[a] Site preparation duration of about 10 months; includes 5 months for tree removal and 5 months for site grading.

[b] Excavated material would be temporarily stored within the construction footprint and would be used as backfill. Material from a borrow site would be used for the additional 29,000 cubic yards needed.

Source: Leidos 2020.

Nonradiological Releases

Nonradiological releases are associated with the operation of trucks and construction equipment (i.e., the burning of diesel fuel). Types and duration of operation for the equipment used during construction are discussed in the main body of this EIS. For construction of the VTR at ORNL, the nonradiological emissions would include those associated with site preparation as well as facility construction, and are presented in **Table B–16**.

Table B–16. Oak Ridge National Laboratory Site Preparation and Facility Construction Nonradiological Emissions

Year/Activity-Source Type	***Emissions (tons)***							***Combined HAPs*** [a]	***CO_2e (mt)***
	VOC	***CO***	***NOx***	***SO_2***	***PM_{10}***	***$PM_{2.5}$***	***CO_2***		
Year 2022									
Onsite Emissions from On-road Sources	0.01	0.14	0.15	0.00	0.02	0.01	79	0.00	72
Onsite Emissions from Nonroad Sources	0.33	1.80	0.99	0.00	0.08	0.08	300	0.05	272
Fugitive Dust	---	---	---	---	6.95	0.69	---	---	---
Offsite Emissions from On-road Sources	0.03	0.32	0.54	0.00	0.07	0.02	260	0.01	236
Slash Burning	28.88	136.80	3.06	1.91	26.64	22.64	3,065	1.09	2,787
Total 2022 Emissions	**29.26**	**139.06**	**4.75**	**1.91**	**33.76**	**23.45**	**3,704**	**1.15**	**3,367**
Year 2023									
Onsite Emissions from On-road Sources	0.08	2.99	0.62	0.00	0.10	0.03	607	0.02	552
Onsite Emissions from Nonroad Sources	0.68	4.27	8.45	0.02	0.42	0.41	2,828	0.11	2,571
Fugitive Dust	---	---	---	---	43.21	4.32	---	---	---
Offsite Emissions from On-road Sources	0.11	6.93	1.20	0.01	0.27	0.06	1,161	0.02	1,055
Total 2023 Emissions	**0.87**	**14.19**	**10.27**	**0.03**	**44.02**	**4.82**	**4,596**	**0.15**	**4,178**

Year/Activity-Source Type	Emissions (tons)							Combined HAPs [a]	CO_2e (mt)
	VOC	CO	NOx	SO_2	PM_{10}	$PM_{2.5}$	CO_2		
Year 2024									
Onsite Emissions from On-road Sources	0.18	8.06	1.46	0.01	0.28	0.08	1,621	0.04	1,474
Onsite Emissions from Nonroad Sources	1.06	6.21	12.75	0.03	0.62	0.60	4,031	0.18	3,665
Fugitive Dust	---	---	---	---	13.35	1.33	---	---	---
Offsite Emissions from On-road Sources	0.29	19.68	3.31	0.03	0.83	0.18	3,478	0.07	3,162
Total 2024 Emissions	**1.53**	**33.96**	**17.52**	**0.07**	**15.08**	**2.19**	**9,131**	**0.28**	**8,301**
Year 2025									
Onsite Emissions from On-road Sources	0.13	5.55	1.14	0.01	0.22	0.06	1,273	0.03	1,157
Onsite Emissions from Nonroad Sources	1.00	5.64	12.35	0.03	0.58	0.56	4,303	0.17	3,912
Fugitive Dust	---	---	---	---	7.32	1.08	0	---	---
Offsite Emissions from On-road Sources	0.18	13.45	2.15	0.02	0.60	0.12	2,483	0.04	2,257
Total 2025 Emissions	**1.31**	**24.64**	**15.64**	**0.06**	**8.72**	**1.83**	**8,058**	**0.24**	**7,326**

CO = carbon monoxide; CO_2 = carbon dioxide; CO_2e = carbon dioxide equivalent; HAPs = hazardous air pollutants; mt = metric tons; NOx = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound; --- = no pollutant emissions from this source type.

[a] Combined HAPs = 15/3 percent of combustive VOC/PM emissions for on-road and nonroad sources and 1/3 percent for slash burning (California Air Resources Board 2018).

Source: Derived from Leidos 2020.

Waste Generation

Estimates of the wastes generated during VTR construction at ORNL would be the same at ORNL as at INL (see Table B–10), this includes waste from the construction of the reactor facilities (Reactor Facility, switchyard, exterior HRS components, the Operational Support Facility, and associated structures). There would not be any radiological waste generated during construction of the VTR. Marketable material from the trees removed during site preparation would be shipped to a local lumberyard, the remainder mulched or burned onsite. Excavation material would be used onsite for site backfill. Therefore, the site preparation activities would not result in the generation of any waste requiring disposal.

B.2.11.2 Environmental Resources – Operations

Resource Requirements

The environmental resources required for operation of the VTR at ORNL would be the same as those described for INL in Section B.2.10.2.

Nonradiological Releases

The main source of nonradiological releases associated with the operation of the VTR would be the releases from operation of the site diesel generators, operations staff personal vehicles, and vehicles used to transport materials (wastes, spent fuel, test assemblies, etc.). The generators supply power to the site in the event of a loss of the normal offsite power supply. To ensure that the generators are functional, they would be tested, started, and run for a period of time, several times a year. The annual emissions associated with these generators are presented in **Table B–17**. Emissions presented in this table include those for all activities at the VTR site, including VTR reactor operations, post-irradiation examination, and spent fuel treatment and storage.

Table B–17. Versatile Test Reactor Operational Nonradiological Emissions

Emission Source	Air Pollutant Emissions (tons per year)							CO_2e (metric tons)
	VOC	CO	NO_x	SO_2	PM_{10}	$PM_{2.5}$	CO_2	
Back-up Generators – VTR	0.05	0.66	0.13	0.001	0.01	0.01	133	121
Pre-Heaters – Normal Annual	0.00	0.00	0.00	0.00	0.00	0.00	3	3
Haul Trucks	0.03	0.13	0.45	0.00	0.07	0.02	271	246
Worker Commuter Vehicles	0.04	4.81	0.26	0.00	0.13	0.02	624	568
Total –Normal Annual Operations	**0.11**	**5.60**	**0.84**	**0.01**	**0.20**	**0.05**	**1,031**	**938**
Pre-Heaters –Large Component Replacement [a]	0.02	0.16	0.27	0.00	0.01	0.01	263	239
Total Annual Emissions [b]	**0.13**	**5.75**	**1.11**	**0.01**	**0.22**	**0.06**	**1,291**	**1,173**

CO = carbon monoxide; CO_2 = carbon dioxide; CO_2e = carbon dioxide equivalent; HAPs = hazardous air pollutants; NO_x = nitrogen oxides; PM_{10} = particulate matter less than 10 microns in diameter; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound; VTR = Versatile Test Reactor.

[a] Large Component Replacement would occur every 15 years.

[b] Equal to sum of Back-up Generators, Haul Trucks, Worker Commuter Vehicles, and Pre-Heaters Large Component Replacement.

Source: Derived from Leidos 2020.

Radiological Releases

The radiological releases from operation of the VTR at ORNL would be the same as those described for INL in Section B.2.10.2 and presented in Table B–13.

Waste Generation

The waste generated from operation of the VTR at ORNL would be the same as those described for INL in Section B.2.10.2 and presented in Table B–14.

B.3 Test Assembly Examination

B.3.1 Introduction

Test assemblies from the VTR would be temporarily stored in the VTR Reactor Facility, within the reactor vessel, if necessary, to allow the assembly to cool sufficiently for handling and transport. Some prompt post-irradiation examination of a test assembly may be performed in a shielded cell located in a pit at the VTR Reactor Facility. Most post-irradiation examination would occur at separate facilities collocated with the VTR.

B.3.2 Post-Irradiation Examination of Test Assemblies

Concurrent with the irradiation capabilities provided by the VTR, the mission need requires the capabilities to examine the test specimens irradiated in the reactor to determine the effects of a high flux of high-energy or fast neutrons. The test specimens could include assemblies of fuel or materials often encapsulated in cartridges such that the material being tested is fully contained. The highly radioactive test specimen capsule would be removed from the reactor after a period of irradiation, ranging from days to years, depending on the nature of the test requirements, and transferred to a fully shielded facility where the test item could be analyzed and evaluated remotely. The examination facilities are "hot-cell" facilities (see **Figure B–17**). These hot cells include concrete walls several feet thick; multi-layered, leaded-glass windows several feet thick; and remote manipulators that allow operators to perform a range of tasks remotely without incurring a substantial radiation dose from the test specimens within the hot cell. In some cases, an inert atmosphere is required to prevent test specimen degradation. DOE intends that the hot-cell facilities where the test items are examined and analyzed after removal from the reactor, would be in close proximity to the VTR to minimize onsite or offsite transportation of the potentially high-radioactive specimens.

Figure B–17. Exterior and Interior Views of Hot Cell Facilities

Needed testing capabilities would include the ability to assess macro and microscopic changes to irradiated materials. Irradiated materials (test specimens) could include reactor fuels, coolants, and any other material that could be exposed to a fast flux in a demonstration or operating fast reactor (e.g., any liquid metal cooled, molten salt fueled and cooled, gas cooled). The post-irradiation examination facility must have the ability to disassemble the test assemblies and the test specimens (disassembly of a test capsule if used and the test specimen itself) and should be able to perform non-destruction examination of irradiated samples including dimensional measurements and neutron radiography (NRAD), and destructive examination including mechanical testing or microscopic examination and characterization of metals and/or ceramics.

B.3.3 Test Assembly Examination at the Idaho National Laboratory Site

B.3.3.1 Facilities

Test assembly examination at INL could be performed at the Hot Fuel Examination Facility (HFEF), the Irradiated Materials Characterization Laboratory (IMCL), the Analytical Laboratory, and the Electron Microscopy Laboratory. Test assemblies would first be transferred to the HFEF for initial disassembly and examination. Entire test specimens or portions of specimens could be transferred to the other facilities to make use of their specialized examination capabilities. The existing facilities would not require modification; although, the HFEF would need new in-cell handling equipment for experiment movements (INL 2020c). All facilities currently do test assembly examination and are able to accept casks with radioactive material. The HFEF can currently accept the test assemblies and dismantle the assemblies for shipment to other facilities (INL 2020c). The facility is linked to analytical laboratories and other facilities by pneumatic sample transfer lines (INL 2017a).

The HFEF, the largest hot-cell facility at INL, is a versatile hot cell facility that consists primarily of two adjacent shielded cells, the main cell and the decontamination cell, surrounded by offices, laboratories, and personnel-related areas in a three-story (above-ground) building. A service level is located below ground. The facility includes an air-atmosphere decontamination cell, an argon-atmosphere main cell (the main cell), decontamination areas, repair areas for hot-cell equipment, auxiliary laboratories, offices, and a high bay area (INL 2020c).

The main cell is a 70 by 30 foot stainless steel-lined gas-tight hot cell. It is fitted with two 5-ton cranes and two electromechanical manipulators. There are 15 workstations, each with a 4-foot-thick window of oil-filled, cerium-stabilized high-density leaded glass and a pair of remote manipulators for use in its purified argon atmosphere. The decontamination hot cell is an air cell that includes five workstations and a water wash spray chamber for decontaminating materials and equipment. Assemblies would be dismantled using the precision mill, a low-speed mill (INL 2017a).

Non-destructive and destructive radioactive material examination and processing is performed in the decontamination cell and main cell. The radioactive materials involved in these activities include actinides and fission products. Radioactive material examination tasks include, but are not limited to, investigation of material characteristics (microstructure) and measurement of properties (fuel length, bowing, cladding surface distortion, and radionuclide distribution). Investigations of these phenomena are performed on samples ranging in mass from milligrams to hundreds of grams. The samples may be cut, ground, and/or polished to facilitate examination (INL 2020c).

These activities utilize current capabilities housed in the HFEF, including:

- gamma scanning,
- visual examination and eddy current testing,
- gas sampling using the Gas Assay Sample and Recharge,
- accident simulation testing in the Fuel and Accident Condition furnace,
- metallic and ceramic sample preparation, and
- bench measurements.

The HFEF also houses the NRAD reactor (a 300-kilowatt TRIGA [Training, Research, Isotopes, General Atomics] reactor), located in the HFEF basement. NRAD is a neutron source for radiographs of experiment components (INL 2017a).

Radioactive material is stored in the HFEF in various storage arrangements in the main cell and consists of (1) FFTF fuel; (2) EBR-II fuel in element magazines; and (3) uranium, plutonium, and other radioactive fuels or materials in containers of various shapes and sizes (INL 2020c).

The IMCL is a 12,000-square foot research facility and is the newest of the INL MFC facilities. The IMCL focuses on microstructural and thermal characterization of irradiated nuclear fuels and materials. The IMCL's design provides customizable radiological shielding and confinement systems. The shielded instruments allow characterization of highly radioactive fuels and materials at the micro-scale and nanoscale. The IMCL was designed to facilitate evolving capabilities (i.e., its flexible modular design would simplify the adaptation of its capabilities to support VTR nuclear fuel and materials examinations). The IMCL has free space for user-defined capability, such as the VTR program. Current and future planned capabilities include:

- Preparation of minute samples for further testing,
- Precision quantitative composition analysis,
- Microstructural characterization, and
- Thermal property measurement (INL 2019c).

In addition to the HFEF and IMCL, some post-irradiation examination could occur at the Analytical Laboratory and the Electron Microscopy Laboratory. The radiochemistry laboratory has six hot cells and eight gloveboxes and general chemistry laboratories. It has the capability to examine irradiated samples including fuels. Equipment within the laboratory can be used to test fundamental physical properties of samples and includes mass spectrometers and gamma and alpha counters (INL 2020a). The Electron Microscopy Laboratory performs materials characterization using electron and optical microscopy tools.

B.3.3.2 Environmental Resources – Construction

Test assembly inspection is currently performed in existing INL facilities. Significant modification of existing facilities is not anticipated. Modifications would consist of removal of some existing legacy equipment and replacement with new equipment that meets the VTR needs. This is a routine activity that is currently performed in these facilities. Any changes to resource requirements would be minimal (e.g., minimal water usage associated with manufacturing of tooling for equipment replacement). No additional plant staff would be required during construction and any changes to resource requirements would be minimal (INL 2020c).

B.3.3.3 Environmental Resources – Operations

The nominal test cycle length for the VTR would be 100 effective full-power days. At the end of each cycle, test assemblies at the end of their test exposure times would be removed from the core. The test specimens within the assemblies would be allowed to cool within the reactor vessel for a period of time. When removed from the reactor vessel and after being cleaned (sodium washed), these test assemblies would be transferred to the post-irradiation examination facility.

Resource Requirements

Most VTR-associated activities would be encompassed by the scope of current activities. No additional staff would be required, assuming that the VTR test specimen preparation and examination activities would supplant current activities at the HFEF. Resource requirements for VTR-related activities are presented in **Table B–18**. Only chemicals used in quantities of over 1,000 pounds are shown in the table. Other chemicals and gases would be used in smaller quantities (INL 2020d).

Table B–18. Idaho National Laboratory Annual Test Assembly Facility Operational Resource Requirements

Resource	*Units*	*Value Annual*
Staff	FTE	80 [a]
Electricity	MWh	minimal
Water		
Potable – staff	gallons	1,000,000
Component wipedown	gallons	1,000
Total	gallons	1,000,000
Chemicals		
Nitric Acid	pounds	17,000
Alcohol	pounds	9,300
Lubricant	pounds	1,400
Acetone	pounds	1,200
Hydrochloric acid	pounds	1,000
Gases		
Argon liquid	standard cubic feet	61,000
Argon/carbon dioxide/hydrogen/methane/methanol	liters	7,800

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] These are all existing staff members (Nelson 2020). VTR activities would replace existing activities.

Source: INL 2020c.

Nonradiological Releases

The nonradiological releases from the HFEF are not expected to change with the addition of VTR test assembly operations. No new sources of emissions are anticipated (INL 2020c).

Radiological Releases

Radiological releases were estimated to increase by 40 percent over current post-irradiation examination operations due to VTR-related activities. The estimated annual release activity per isotope is presented in **Table B–19**. The isotopes in bold are those that contributed at least 0.1 percent of the total offsite dose from MFC operations in 2018, based on the INL Annual Site Environmental Report (INL 2019d). Other isotopes listed are limited to those with releases greater than 10^{-10} curies.

Table B–19. Idaho National Laboratory Test Assembly Examination Facility Operational Annual Radiological Releases

Isotope	*Release (curies)*	*Isotope*	*Release (curies)*
Antimony-125	3.2×10^{-5}	Krypton-85	4.4×10^{-3}
Americium-241	$\mathbf{8.4 \times 10^{-12}}$	Neptunium-237	3.2×10^{-9}
Carbon-14	$\mathbf{3.1 \times 10^{-4}}$	Phosphorus-32	2.6×10^{-5}
Cadmium-109	5.2×10^{-4}	Phosphorus-33	4.9×10^{-9}
Cadmium-115m	1.0×10^{-7}	**Plutonium-238**	$\mathbf{1.2 \times 10^{-10}}$
Chlorine-36	1.0×10^{-5}	**Plutonium-239**	$\mathbf{9.5 \times 10^{-8}}$
Cobalt-60	$\mathbf{7.9 \times 10^{-13}}$	**Plutonium-240**	$\mathbf{3.0 \times 10^{-12}}$
Cesium-134	8.0×10^{-7}	Plutonium-242	1.8×10^{-9}
Cesium-137	$\mathbf{2.5 \times 10^{-2}}$	Sodium-22	3.2×10^{-6}
Hydrogen-3 (Tritium)	$\mathbf{3.7 \times 10^{-2}}$	Sodium-24	1.7×10^{-8}
Iodine-129	$\mathbf{1.8 \times 10^{-5}}$	Sulfur-35	1.2×10^{-4}
Iodine-131	8.9×10^{-3}	**Strontium-90**	$\mathbf{3.8 \times 10^{-7}}$

Note: The isotopes in bold are those that contributed at least 0.1 percent of the total offsite dose from MFC operations in 2018, based on the INL Annual Site Environmental Report (INL 2019d).
Source: INL 2020d.

Releases of radioactivity to the environment would be through the existing release points for each of the facilities that could be used for post-irradiation examination. All test specimens would be processed through the HFEF first; individual samples could be transferred to other facilities for detailed examination. The combined flow rate would be about 35,200 cubic feet per minute at 72 °F. The release would be through a rectangular, 84 by 30-inch stack, at a height of about 95 feet.

Waste Generation

Waste from post-irradiation examination activities would involve discarding of material from driver fuel assemblies and experiments as well as low-level waste items associated with cask operations and operator protective equipment (INL 2020c). Annual waste generation rates, based on the handling of up to 60 test assemblies per year, are provided in **Table B–20**.

Table B–20. Idaho National Laboratory Test Assembly Facility Annual Waste Generation

Waste Type	*Category*	*Volume (cubic meters)*		*Weight (pounds)*	
		Net	*Gross*	*Net*	*Gross*
Hazardous	NA	1.6	4.7	1,400	2,300
Industrial	NA	1.9	1.9	1,300	1,600
Recyclable	NA	1.2	1.2	1,900	2,000
TSCA	NA	0.053	0.054	70	87
Universal	NA	0.12	0.13	83	95
Low-level waste	Contact handled	93	100	35,000	50,000
	Remote handled	2.5	2.6	1,900	2,800
Mixed low-level waste	Contact handled	6.3	8.9	7,800	9,800
Transuranic waste	Contact handled	0.67	0.75	310	540
Mixed transuranic waste	Contact handled	0.14	0.14	62	100
	Remote handled	0.073	0.11	90	470

NA = not applicable.
Source: INL 2020d.

B.3.4 Test Assembly Examination at Oak Ridge National Laboratory

B.3.4.1 Facilities

Test assembly post-irradiation examination at ORNL would make use of some existing facilities, but none of these facilities would include hot cells that operate using an inert environment; all would use an air atmosphere. Initial test assembly examination activities would need to be performed within a hot cell with an inert atmosphere. Once properly prepared, additional examination of the test specimens can be performed at existing ORNL facilities.

A new hot cell facility with inert atmosphere hot cells adjacent to the VTR would be needed. A conceptual design[12] for this facility has been developed to meet the process requirements identified in Section B.3.2, using equipment similar to that identified under the INL Alternative for the VTR (Section B.3.3). The facility would be located adjacent to the VTR, within a common protected area, and would support both test specimen post-irradiation examination and spent fuel treatment activities. In size and capability, this new post-irradiation examination facility would be similar to the INL HFEF (see Section B.3.3.1).

New Facility

The new hot cell facility would provide an inerted hot cell for post-irradiation examination (plus one for spent fuel treatment, See Section B.4.4). Each hot cell would be connected to a decontamination cell with an air atmosphere. The hot cell facility would have four levels and would be approximately rectangular with a reinforced concrete structure. The bottom portion of the hot cell facility would have a footprint of about 172 by 154 feet.

The hot cell facility would include two major structural systems: a concrete structure from the basement level up to the floor of the fourth level or high bay area, and a steel structure enclosing the fourth level high bay area.

The reinforced concrete bottom portion of the hot cell facility would consist of three floors: the service floor, an operating floor, and a second floor extending from an elevation of about -16 feet (16 feet below surface level) to the top of the second floor at 29 feet. The concrete structure would contain the test assembly hot cell, the spent fuel treatment hot cell, and the two associated decontamination cells. The top of the concrete structure forms the floor of the high bay area.

A steel-braced structure, 122 by 154 feet, would rise about 53 feet above the concrete portion of the structure. This high bay area would be constructed of metal siding and a metal roof deck, at an elevation of about 86 feet above ground level, supported by steel roof beams and tapered, built-up, steel roof girders. The steel structure would form a high bay for a 40-ton overhead crane, used to transfer equipment and material, including transfer of material between the truck lock, high bay, and cask tunnel.

A hot repair area would be an enclosed single-story area near the center of the high bay area. This area would be used for the maintenance of in-cell material-handling equipment. The area would not be shielded, as equipment would be decontaminated prior to being moved to this area. The hot repair area would be constructed of concrete-block masonry perimeter and interior walls, with a roof of steel decking covered by a thin layer of concrete.

The new hot cell facility would include a truck lock to accommodate receipt of the various materials into the facility through roll-up doors at each end. A 25-ton bridge crane in the top of the truck lock would be provided to move loads through a floor hatch into the cask tunnel for each hot cell. The ceiling of the truck lock would consist of metal covers that could be removed for access to the high bay area. A 29.5-foot-deep cylindrical cask handling pit is included. The truck lock would also be accessible to the 40-ton

[12] The conceptual designs have been developed for National Environmental Policy Act purposes only. This conceptual design is not as detailed as, nor is it to be considered, the conceptual design that is a part of the DOE facility design process.

HBA crane, which would be used to move loads between the truck lock, the high bay area, and the cask tunnel.

Transfer tunnels would be incorporated into the hot cell facility design, a cask tunnel and shielded transfer tunnels. The cask tunnel would be used to transfer material (equipment, tools, experiments, etc.) from top-opening casks into the cell complex. The cask tunnel would extend from the truck lock to the decontamination cells. The shielded transfer tunnels, located under the cell floors, would be used for the movement of large equipment and irradiated components between the decontamination cells and the inerted cells.

A central portion of the hot cell facility, measuring approximately 100 by 105 feet, would house the test assembly examination and spent fuel treatment hot cells, cask tunnel, and other facilities. These areas would have concrete walls and concrete-floor slabs for radiation-shielding purposes. The area surrounding this central cell area would house offices, labs, corridors, and other rooms. The floors in the office areas would be thin, reinforced concrete slabs, supported by reinforced concrete girder-joist systems, which, in turn, would be supported on reinforced concrete columns. The perimeter wall up to a grade elevation, would be constructed of reinforced concrete. Above this elevation, the walls and interior partitions would be concrete masonry blocks.

The test assembly examination portion of the hot cell facility would have its own set of inerted hot and decontamination cells. The test assembly examination hot cell would be a concrete-shielded, steel-lined enclosure with interior dimensions of 30 feet wide by 70 feet long by 25 feet high. It would be filled with argon gas that provides an inert, non-oxidizing atmosphere. The associated decontamination cell would be a concrete-shielded, steel-lined enclosure with interior dimensions of 30 feet wide by 20 feet long by 25 feet high; it would be filled with air. The interior surfaces of the cell would be lined with steel. A raised steel floor would extend over part of the cell. Sections of the raised floor could be removed for access to the subfloor area. Test samples and equipment would be moved using two 5-ton cranes and electromechanical manipulators. The space beneath the removable floor would be used for storage; it would also house gas ducts and filters, and serve as additional space (depth) for vertical handling of long items.

There would be penetrations in the cell walls, roof, and floor for windows, utility service, feedthroughs, in-cell handling equipment, gas ducting, transfer hatches, etc. Penetrations into each cell would be steel-lined, welded to the cell liner, and surrounded by high-density shielding closures or inserts. Closures or inserts for the penetration liners would have double seals, with the space between them pressurized with an argon purge.

The test assembly examination hot cell would have 15 work stations, each about 10 feet wide, equipped with a shielding observation window (layers of leaded glass with thin layers of mineral oil between them, plus a protective non-leaded glass plate on the cell side). Stations would be equipped with lights, utility distribution systems (electric and pneumatic), examination equipment, work tables, and up to two master/slave manipulators. The cell would be designed so that equipment could be added or removed from the work station without releasing radioactive contaminants, diluting the inert cell atmosphere, or extensively interrupting work at adjacent stations. The interior of the hot cell would be lighted, and high-intensity lighting would be provided in the cell at each active work station. In addition, emergency lighting would be provided.

Fuel material and test assembly storage would be available at various locations in and below each hot cell. There would be two 33-inch inner-diameter steel pits, extending below the level of the cell steel floor and the facility basement. These pits would be directly below the cell-roof loop-transfer penetrations for direct access. The pits can be covered when not in use.

The decontamination cell would be a shielded hot cell with an air atmosphere, maintained at a negative pressure relative to the surrounding corridors to minimize the spread of contamination. The decontamination cell would have six work stations and six leaded-glass observation windows. The decontamination cell would be separated from the inerted cell by an ordinary concrete shielding wall. The decontamination cell would be the same width and height as the inerted cell, and its outer walls would be similarly constructed. The cell floor would be lined with stainless steel, and the lower walls would be lined with carbon steel coated with epoxy paint. Electrical and pneumatic services in each decontamination cell would be generally similar to those in the inerted cell.

Support systems within the hot cell facility would be shared by the post-irradiation examination and spent fuel treatment processes.

The hot cell facility would have two distinct HVAC systems for contamination and emissions control: a cell exhaust system and a building/laboratory exhaust system. Both the cell and the building/laboratory ventilation exhausts would be HEPA-filtered.

Utility distribution systems supporting the hot cell facility include normal electrical power supplied by the commercial grid; optional standby electrical power supplied by two diesel generators; instrument and vital compressed air; fire, potable, and service water systems; and communications. Compressed gas for process applications would be supplied by standard compressed gas cylinders. Compressed argon for cell inerting would be supplied by a liquid argon tank system located outside the hot cell facility.

The control room for hot cell facility operations would be located on the operating floor. Local instrument alarm panels would be installed on, or in the vicinity of, the applicable equipment (e.g., hot cell work-station equipment, hot cell atmosphere-cooling and purification equipment, ventilation systems).

Existing Facilities

In addition to this new hot cell facility, existing facilities at ORNL would be used for supplemental and/or advanced post-irradiation examination for materials that do not require an inert environment. Hot cells within the Irradiated Fuels Examination Laboratory, Building 3525, and the Irradiated Materials Examination and Testing facility, Building 3025E, would be used to supplement the capabilities of the new post-irradiation examination facility. In addition, the Low Activation Materials Design and Analysis Laboratory (LAMDA) would be used for testing of low dose samples, samples that do not require hot cells for article examination. No modifications to the existing facilities would be required in support of the VTR post-irradiation examination of test specimens.

The Irradiated Fuels Examination Laboratory in Building 3525 is a Category 2[13] nuclear facility and contains six hot cells (including a scanning electron microscope cell, irradiated microsphere gamma analyzer cell, and a core conduction cool-down test facility cell) that are currently used for examination of a wide variety of fuels. The facility has been used for safety testing of High Temperature Gas Reactor fuel. Examination and testing capabilities include destructive and non-destructive testing of irradiated samples by techniques including metrology, optical and electron microscopy, gamma spectrometry, and other physical and mechanical property evaluation techniques (ORNL 2015).

The Irradiated Materials Examination and Testing facility in Building 3025E is a Category 3 nuclear facility that contains six hot cells (four of which are connected by transfer drawers) that are used for mechanical testing and examination of highly irradiated structural alloys and ceramics. The facility also includes a

[13] DOE defines hazard categories by the potential impacts identified by hazard analysis and has identified radiological limits (quantities of material present in a facility) corresponding to the hazard categories: Hazard Category 3 – Hazard Analysis shows the potential for only significant localized consequences; Hazard Category 2 – Hazard Analysis shows the potential for significant onsite consequences beyond localized consequences (DOE 2018a).

Specimen Prep Lab equipped with laboratory hoods and glove boxes. It is a two-story block and brick building with a two-story high bay (ORNL 2014).

LAMDA is a laboratory for the examination of materials with low radiological content (samples limited to less than 100 millirad per hour at 30 centimeters) that do not require remote manipulation. LAMDA capabilities focus on mechanical, physical, and microstructural characterization of samples. The LAMDA facility augments the capabilities in the ORNL hot cell facilities by adding a more precise and delicate sample-handling capability allowing for the study of material phenomenon not possible in a hot cell facility (ORNL 2017).

B.3.4.2 Environmental Resources – Construction

In addition to the resource requirements for the post-irradiation examination capability, these resources include the resources required for construction of the spent fuel treatment capability. Both capabilities are located within the same new facility; the Post-Irradiation Examination and Spent Fuel Treatment Facility. Estimates of environmental resources were developed for the facility, not each individual capability.

Resource Requirements

Table B–21 provides a summary of the key resources committed to the construction of the post-irradiation examination and spent fuel treatment capability. The construction effort would ramp up until peaking in the third year of construction. The resources required for site preparation have been included in the resource requirements for VTR construction at ORNL (see Section B.2.11).

Table B–21. Resource Requirements during Oak Ridge National Laboratory Post-Irradiation Examination and Spent Fuel Treatment Facility Construction

Resource	*Units*	*Annual Average Value*	*Annual Peak Value*	*Total* [a]
Staff	FTE	200	390	960
Electricity	kWh	300,000	600,000	1,300,000
Gasoline	gallons	26,000	44,000	110,000
Diesel Fuel				
Road Diesel	gallons	25,000	43,000	110,000
Non-road Diesel	gallons	130,000	230,000	570,000
Total Diesel	gallons	160,000	270,000	690,000
Water				
Potable	gallons	2,400,000	3,600,000	12,000,000
Dust Control, etc.	gallons	6,600,000	12,000,000	27,000,000
Total	gallons	9,000,000	16,000,000	39,000,000
Asphalt	cubic yards	420	NA	420
Structural Concrete	cubic yards	--	--	12,000
Rebar	tons	--	--	1,300
Excavation	bank cubic yards [b]	--	--	41,000
Backfill Material	cubic yards	--	--	60,000 [c]
Landscaping	cubic yards	--	--	600
Structural Steel	tons	--	--	1,200
Large Bore Piping	linear feet	--	--	9,500
Cable and Wire	linear feet	--	--	360,000
Cable Tray	linear feet	--	--	5,400
Conduit Above Grade	linear feet	--	--	66,000
Conduit Inside Duct Banks	linear feet	--	--	16,000
Rock/Gravel	cubic yards	--	--	14,000
Temporary Concrete	cubic yards	--	--	4,200
Lumber	tons	--	--	75
Temporary Steel	tons	--	--	15
Gas [d]	bottles/cubic meters	--	--	6,000/39,000

Resource	Units	Annual Average Value	Annual Peak Value	Total [a]

FTE = full-time equivalent (person); kWh = kilowatt-hour; NA = not applicable.

[a] Construction duration of 51 months is assumed.

[b] A bank yard is the volume of earth or rock in its natural state, as compared to the expanded volume after excavation.

[c] Excavated material would be temporarily stored within the construction footprint and would be used as backfill. Material from a borrow site would be used for the additional 19,000 cubic yards needed.

[d] Gas bottles (cylinders) can range from 2 to 10 cubic meters in size. A typical size of 6.5 cubic meters has been used to estimate the volume of gas in the cylinders.

Source: INL 2020c; Leidos 2020.

Nonradiological Releases

Nonradiological releases are associated with the operation of trucks and construction equipment (i.e., the burning of diesel fuel). Types and duration of operation for the equipment used during construction are discussed in the main body of this EIS. Emissions associated with equipment have been included in the estimates for construction of the VTR at ORNL in Table B–16.

Waste Generation

Table B–22 provides estimates of the wastes generated during facility construction. There would not be any radiological waste generated during construction of the Post-Irradiation and Spent Fuel Treatment Facility.

Table B–22. Oak Ridge National Laboratory Post-Irradiation and Spent Fuel Treatment Facility Construction Wastes

Waste Type	Material	Units	Value
Hazardous Waste			Assumed to be 2 percent of nonhazardous waste volumes
Nonhazardous Waste	Concrete	cubic yards	3,000
	Rebar	pounds	110,000
	Structural steel	tons	99
	Large bore pipe	feet	750
	Small bore pipe	feet	840
	Cable and wire	feet	29,000
	Cable tray	feet	420
	Conduit	feet	7,800
	Tubing	feet	840
	Instruments	each	20
	Valves	each	9
	In-line components	each	20
	Lumber	tons	36
	Steel	tons	15
	Gas	bottles	5,800

Source: INL 2020c; Leidos 2020.

B.3.4.3 Environmental Resources – Operations

In addition to the resource requirements for the post-irradiation examination capability, these resources include the resources required for operation of the spent fuel treatment capability. Both capabilities are located within the same new facility, the Post-Irradiation Examination and Spent Fuel Treatment Facility. Estimates of environmental resources were developed for the facility, not each individual capability.

The nominal test cycle length for the VTR would be 100 effective full-power days. At the end of each cycle, test assemblies at the end of their test exposure time would be removed from the core. The test specimens within the assemblies would be allowed to cool within the reactor vessel for a period of time. When removed from the reactor vessel and after being cleaned (sodium removal), these test assemblies would be transferred to the post-irradiation examination facility.

Resource Requirements

Key annual resource commitments for the operation of the Post-Irradiation Examination and Spent Fuel Treatment Facility are provided in **Table B–23**. Diesel fuel would be required for testing of the site diesel generators.

Table B–23. Oak Ridge National Laboratory Post-Irradiation Examination and Fuel Treatment Facility Operational Resource Requirements

Resource	*Units*	*Value*
		Annual (peak)
Staff	FTE	100
Electricity	MWh	57,000 (60,000)
Diesel Fuel [a]	gallons	2,700
Water		
Potable	gallons	1,200,000
Component Wipedown	gallons	1,000
Total	gallons	1,200,000
Chemicals		
Acetone	pounds	15,000
Alcohol	pounds	30,000
Decon	pounds	14,000
Lubricant	pounds	1,400
Hydrochloric acid	pounds	1,000
Nitric Acid	pounds	17,000
Oil	pounds	2,300
Paint/Paint Thinner	pounds	1,800
Sodium Hydroxide Solutions	pounds	7,800
Gases		
Argon Liquid	standard cubic feet	61,000
Argon/Carbon Dioxide/Hydrogen/Methane/Methanol	liters	7,800
R-22 Refrigerant in Nitrogen/Air	liters	2,700

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] Diesel generators would operate 1 percent of the time, 88 hours per year.

Source: Leidos 2020.

Nonradiological Releases

Non-radiological releases result primarily from the testing of the building diesel generators and from the operation of personal vehicles by facility staff. The emissions associated with equipment have been included in the estimates for operation of the VTR at ORNL in Table B–17.

Radiological Releases

Radiological releases were estimated based on current releases from the HFEF and estimates of the gaseous inert fission products (INL 2020c) identified for examination of VTR test specimens. These estimates are presented in **Table B–24**. All releases from the facility would pass through HEPA filters (and from the main cell additional carbon filters) before being released through the facility stack.

Table B–24. Oak Ridge National Laboratory Post-Irradiation and Spent Fuel Treatment Facility Operational Annual Radiological Releases

Isotope	*A Post-Irradiation Examination Release (curies)*	*Spent Fuel Treatment Release (curies)*
Antimony-125	3.2×10^{-5}	1.57×10^{-7}
Americium-241	8.4×10^{-12}	
Carbon-14	3.1×10^{-4}	
Cadmium-109	5.2×10^{-4}	
Cadmium-113m		4.15×10^{-10}
Cadmium-115m	1.0×10^{-7}	
Cerium-144		1.41×10^{-6}
Chlorine-36	1.0×10^{-5}	
Cobalt-60	7.9×10^{-13}	2.08×10^{-9}
Cesium-134	8.0×10^{-7}	2.62×10^{-7}
Cesium-137	2.5×10^{-2}	1.96×10^{-6}
Europim-154		1.73×10^{-10}
Europium-155		2.07×10^{-9}
Iron-55		5.50×10^{-8}
Hydrogen-3 (Tritium)	3.7×10^{-2}	510
Iodine-129	1.8×10^{-5}	
Iodine-131	8.9×10^{-3}	
Krypton-85	4.4×10^{-3}	8.250
Neptunium-237	3.2×10^{-9}	
Niclel-63		2.76×10^{-10}
Promethium-147		1.25×10^{-7}
Phosphorus-32	2.6×10^{-5}	
Phosphorus-33	4.9×10^{-9}	
Plutonium-238	1.2×10^{-10}	1.24×10^{-10}
Plutonium-239	9.5×10^{-8}	2.83×10^{-9}
Plutonium-240	3.0×10^{-12}	1.87×10^{-10}
Plutonium-241		1.17×10^{-9}
Plutonium-242	1.8×10^{-9}	
Ruthinium-106		5.66×10^{-6}
Samarium-151		8.97×10^{-10}
Sodium-22	3.2×10^{-6}	
Sodium-24	1.7×10^{-8}	
Sulfur-35	1.2×10^{-4}	
Strontium-90	3.8×10^{-7}	3.47×10^{-8}

Source: INL 2020c.

Waste Generation

Annual waste generation rates for the Post-Irradiation Examination and Spent Fuel Treatment Facility are based on three VTR test cycles per year. These estimates are provided in **Table B–25**. This table includes waste generated from post-irradiation examination of test specimens, as well as spent driver fuel treatment. In addition to the wastes listed in this table, the heavy metal from 45 spent driver fuel assemblies (66 for the final core offload at the end of the VTRs operational lifetime) would be packaged as spent fuel.

Table B–25. Oak Ridge National Laboratory Post-Irradiation and Spent Fuel Treatment Facility Annual Operational Waste Generation

Waste Type	Category	Volume (cubic meters)		Weight (pounds)	
		Net	Gross	Net	Gross
Hazardous	NA	1.6	4.7	1,400	2,300
Industrial	NA	3.7	3.9	4,600	4,900
Recyclable	NA	1.2	1.2	1,900	2,000
TSCA	NA	0.053	0.054	70	87
Universal	NA	0.12	0.13	83	95
Low-level radioactive waste	Contact handled	220	240	110,000	160,000
	Remote handled	160	170	170,000	230,000
Mixed low level radioactive waste	Contact handled	16	21	21,000	25,000
	Remote handled	16	16	14,000	20,000
Transuranic waste	Contact handled	0.67	0.74	310	530
Mixed transuranic waste	Contact handled	0.14	0.15	62	100
	Remote handled	0.073	0.11	90	470

NA = not applicable.
Source: INL 2020d.

B.4 Spent Fuel Treatment and Storage

B.4.1 Introduction

Spent fuel would be stored within the VTR reactor vessel for about 1 year, until the decay heat produced drops sufficiently to allow for transport within a fuel transport cask and treatment of the spent fuel. Spent fuel treatment includes the removal of sodium from the spent fuel and the consolidation and packaging of the fuel. The fuel would be packaged in casks suitable for transport and storage at an onsite temporary storage facility and transport to and storage at a permanent repository.

Unless otherwise noted, information in the following subsections is from the *VTR Fuel Facility Plan* (INL 2019a).

B.4.2 Spent Fuel Treatment

The fuel would contain metallic sodium between the cladding and the metallic fuel pins to improve heat transfer from the fuel to the reactor coolant through the stainless-steel cladding. When fuel is irradiated in the reactor for some period of time, the metallic fuel swells as fission products are generated. Pores form throughout the fuel as it swells under irradiation and pressure from the gaseous fission products. The fission product gases escape to a plenum in the fuel element just above the metallic fuel. As the gases escape, liquid sodium flows into these tiny pores, much like a sponge. As more pores form and grow, others are closed off from the fuel surface, including those containing sodium. Between 20 and 40 percent of the available sodium (up to 0.8 grams) may enter the fuel and become inseparable from the uranium, except by dissolving or melting the fuel.

Maintaining a small inventory of untreated spent VTR fuel, perhaps 4 years or less of discharged fuel, would require that the fuel treatment facility treat fuel at the same rate as discharged by the VTR. These material throughput rates could be as high as 2.0 metric tons of fuel alloy per year with up to 1.8 metric tons of heavy metal per year.

The proposed treatment option for the sodium-bonded fuel elements would consist of five activities:

- Assembly disassembly,
- Fuel pin chopping,

- Consolidation and vacuum distillation of chopped fuel and plenums,
- Sodium stabilization, and
- Packaging.

Prior to transfer to the fuel storage pad, driver fuel assemblies would be washed at the VTR in a sodium wash station. At the wash station, the assembly would be washed inside of the wash station vessel by exposing the assembly to inert nitrogen gas containing demineralized water moisture. The demineralized water reacts with residual sodium to form sodium hydroxide. A second wash with demineralized water is used to remove the sodium hydroxide.

Up to six spent driver fuel assemblies would be transferred in a transfer cask to a spent fuel pad for temporary storage. The spent driver fuel assemblies would be inserted into a storage module within the interim dry storage system, where they would be stored for at least 3 years. (Three years would be the minimum storage time prior to spent fuel treatment and has been selected for planning purposes; the storage time could vary.) The interim dry storage system would consist of commercially available storage casks (INL 2020c).

Following the 3 additional years of cooling time, the spent driver fuel assemblies would be removed from the storage cask and transferred to a spent fuel treatment facility. All fuel treatment activities would take place in hot cells. VTR spent driver fuel assemblies would first be disassembled in the reverse of the assembly process described in Section B.5. Following disassembly, the fuel pins would be transferred to an element chopper.

Fuel pin chopping would consist of cutting the 165-centimeter fuel pins into much shorter pieces. Pieces free of spent fuel would be separated from pieces containing spent fuel. Gases released during the chopping process would be processed through a waste gas treatment system.

The container of chopped fuel would be placed into a vacuum distillation furnace. The entire driver fuel assembly (including reflectors and other smaller components) would be melted. Melting the full driver assembly would serve three functions: (1) reduce the concentration of the fissile material in the resulting consolidated product; (2) assist with fuel melting and consolidation; and (3) produce a more durable or corrosion-resistant, stabilized fuel product. The chopped segments of sodium-bonded fuel would be heated, evaporating the sodium, including the sodium that had migrated into pores in the fuel. The sodium-free fuel product (fuel, cladding, and possibly diluent) would continue to be heated to melt the product to form a eutectic[14] mixture, which would be removed from the furnace, solidified into ingots, and transferred to a packaging station. Individual ingots would weigh about 60 kilograms and would contain less than 10 percent by weight (no more than 6 kilograms) plutonium.

The sodium-free spent fuel ingots would be packaged in metal small canisters. The ingot canisters would have a robust metal shell and would fix the ingots into a location for criticality and transportation accident considerations. The ingot canisters would be filled with inert gas (argon or helium) and close-seal welded. A number of these canisters would be loaded into a DOE dual-purpose canister, providing an added measure of containment and protection for the spent fuel. The treated spent fuel would be loaded into a transfer cask, transferred back to the spent fuel pad, and transferred to the storage casks. Each storage cask would be capable of storing 120 ingots of treated spent fuel. This would be equal to 2 years of spent fuel generated by the VTR. The treated fuel would be stored onsite until an offsite storage capability (either a temporary storage site or a permanent repository) would be available (INL 2020c).

[14] A eutectic mixture is a homogenous mixture of two or more substances that solidifies at one temperature, lower than the temperature at which the individual substances solidify.

In the bottom section of the consolidation and distillation systems, sodium would be collected in a disposable steel container and transferred for stabilization. Depending on processing conditions, some volatile and semi-volatile fission products could be collected with the condensed sodium.

Fuel-pin plenum pieces (i.e., without fuel) would also be processed in a distillation system to remove any sodium but may or may not be consolidated into stainless-steel ingots. Sodium collected from the plenum sections would also be collected and transferred for stabilization.

Sodium stabilization would be achieved in a bakeout furnace. The sodium along with a stabilization chemical would be heated to about 800 degrees Celsius (°C) in a sealed steel shell. The stabilization chemical (possibly iron chloride) would react with the sodium to create a stable compound (e.g., combined with iron chloride, the reaction would produce iron and sodium chloride [salt]).

The sealed steel shells of stabilized salt and iron would be transferred to a packaging station where they would be placed in road-ready containers for shipment to a temporary waste storage location. Iron from sodium stabilization, sodium salt, and the processed plenums (sodium-free steel clads either as ingots or as scrap metal) would be treated as remote-handled low-level radioactive waste.

B.4.3 Spent Fuel Treatment and Storage at the Idaho National Laboratory Site

B.4.3.1 Idaho National Laboratory Facilities

All fuel treatment activities would be performed in the Fuel Conditioning Facility (FCF). The FCF is used to support nuclear energy research and development for multiple customers, including DOE, and is used to support the treatment of sodium-bonded spent fuel. (The FCF also supports developmental efforts in pyroprocessing; high-temperature chemical and electrochemical methods for the separation, purification, and recovery of fissile elements.) The FCF has two heavily shielded hot cells, one rectangular with an air atmosphere and one round with an inert (argon) atmosphere. Both are equipped with remotely operated manipulators to allow safe handling of irradiated fuels and materials. The inerted cell facilitates the preparation and treatment of spent fuel elements. Additionally, the facility has equipment to decontaminate and prepare elements for treatment, transfer components to other facilities (e.g., HFEF) and test, using mockup facilities, remotely operated systems designs (INL 2016).

To accommodate the material throughput identified in Section B.4.1, the FCF would require additional in-cell equipment treatment capacity, the replacement of a cell window to accommodate the transfer of spent driver fuel assemblies into the hot cell, and a transition to a 24-hour, 7-days-per-week operations schedule.

Fuel pin chopping would use existing FCF element choppers (see **Figure B–18**). In the existing element choppers, the linear slide feed mechanism is capable of handling up to five fuel elements of EBR-II fuel. Fuel pins are fed into the electromechanical press one at a time (INL 2020b). The press cuts them into elements that are between 0.25 and approximately 1.0 inches long (INL 2020b). For the VTR fuel pins, chopped fuel elements would be collected in separate baskets for fuel-containing elements and plenum elements. The FCF element choppers were designed to chop EBR-II fuel and have previously been modified to chop FFTF fuel and may need to be modified to accommodate VTR fuel pin length and diameter.

Spent fuel consolidation and distillation would use vacuum distillation furnaces. INL currently uses similar furnaces (see **Figure B–19**), in the HFEF. To handle the expected amount of spent fuel, multiple distillation units would need to be installed at the FCF. All fuel treatment actions would be performed in the argon atmosphere hot cell.

The sodium contaminated bakeout furnace would also be located within the FCF.

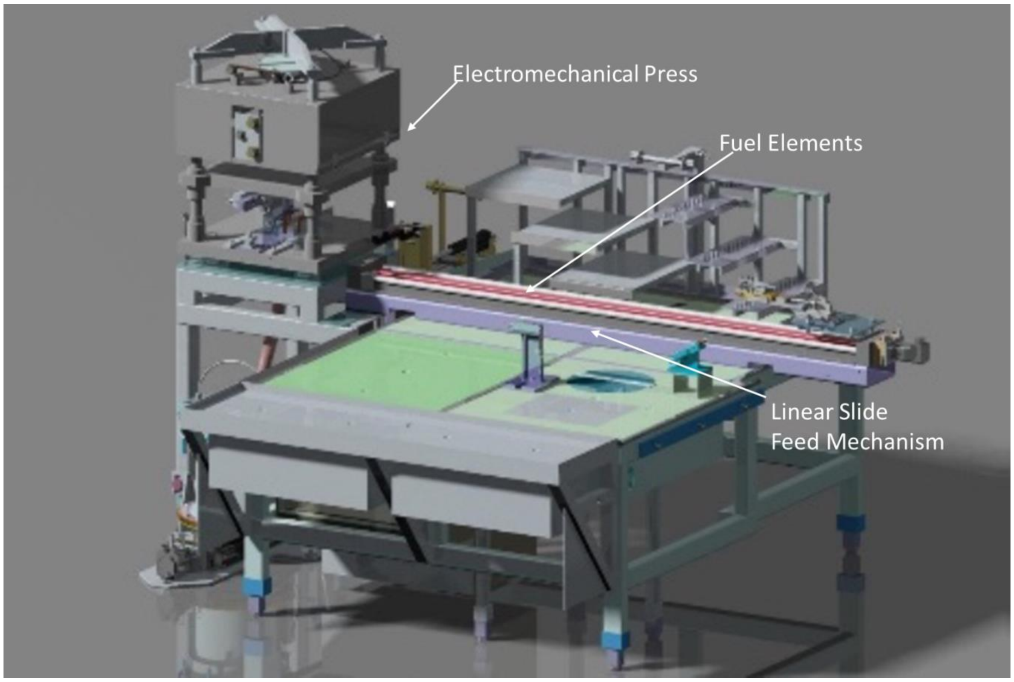


Figure B–18. Production Fuel Element Chopper in the Fuel Conditioning Facility

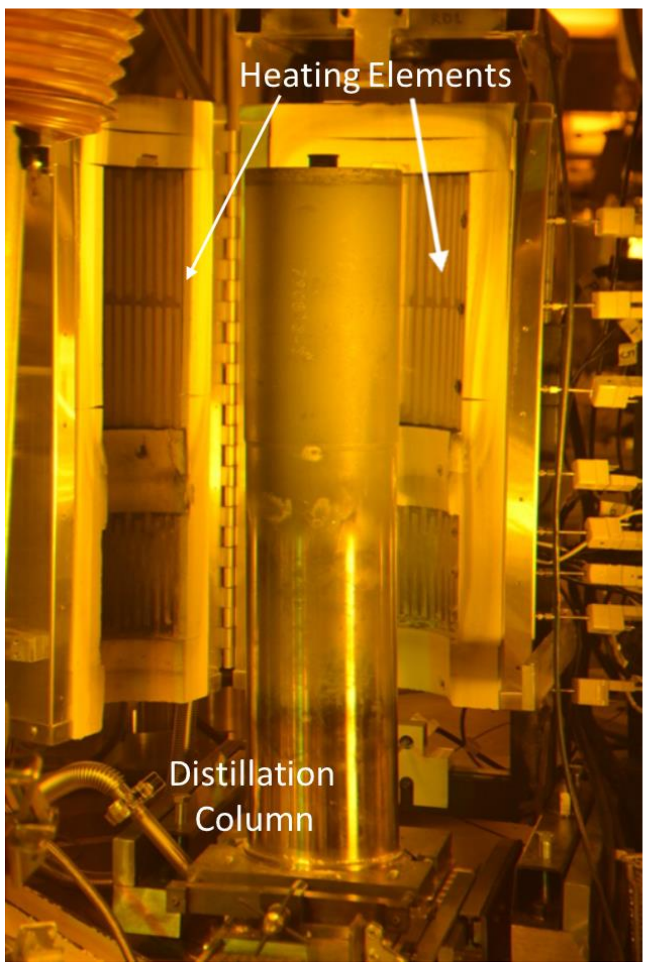


Figure B–19. Hot Fuel Examination Facility Distillation System

All products from the sodium treatment of the spent fuel would be packaged and temporarily stored (pending transfer to a permanent repository) at a facility at the MFC.

A new pad for the temporary storage of VTR spent driver fuel assemblies and treated spent fuel would be constructed on the VTR site at INL. The spent fuel pad could be required to handle all of the spent fuel from VTR operation (60 years) after treatment at the FCF.[15] Prior to the end of VTR operations, 3 years of spent fuel directly from the VTR would be stored on the pad. If sized to handle spent fuel from 60 years of VTR operations, the facility would consist of a concrete pad about 11,000 square feet (90 by 120 feet) and 4.5 feet thick. The spent fuel would be stored in qualified commercial storage casks (INL 2020c).

B.4.3.2 Environmental Resources – Construction

Resource Requirements

Resources required for the modifications to the FCF to accommodate VTR spent fuel treatment are limited to the workers needed to make the modifications and the use of potable water by these workers. INL estimates it would take a 10-person team working for 2 years to make the modifications. The workers would require 250,000 gallons of potable water during construction. Other material and utility use would be minimal.

Resource requirements for the construction of the spent fuel pad would be included in the construction of the VTR and its associated facilities. They would be a small fraction of that needed for the construction of the VTR (INL 2020c) and would not appreciably increase the resource requirements for construction of the VTR and its associated facilities.

Nonradiological Releases

Nonradiological emissions during the construction of the spent fuel treatment facility are expected to be minimal. Emissions from the construction of the spent fuel pad would not materially increase the emissions associated with construction of the VTR facilities.

Waste Generation

Replacement of an FCF hot cell window may be required to accommodate VTR fuel transfer into the hot cell. Should this modification be necessary, removal of the existing hot cell window would be expected to generate low-level waste: about 5.4 cubic meters (12,000 pounds) gross, 5.2 cubic meters (10,000 pounds) net. Construction of the spent fuel pad would result in minimal waste generation. Small amounts of excess concrete and rebar would be generated, which would be a small fraction of the waste generated from the construction of the VTR.

B.4.3.3 Environmental Resources – Operations

The nominal test cycle duration for the VTR would be 100 effective full-power days. At the end of each cycle, up to 15 spent driver fuel assemblies could be removed from the core (INL 2020c). The spent driver fuel assemblies would be allowed to cool within the reactor vessel for a period of time, nominally a year. When removed from the reactor vessel and after being cleaned (sodium removal), these spent driver fuel assemblies would be transferred to the spent fuel pad. After an additional cooling period, at least 3 years, these assemblies would be transferred to the Spent Fuel Treatment Facility within the FCF for treatment and consolidation. The resulting spent fuel waste form would be returned to and stored at the spent fuel pad until transferred to an offsite storage facility.

[15] The spent fuel pad could be smaller. The VTR program intends to ship spent fuel offsite as soon as an offsite storage option, either an interim storage facility or a permanent repository, is available.

Resource Requirements

Key annual resource commitments for the operation of the Spent Fuel Treatment Facility are provided in **Table B–26**. Only chemicals used in quantities of over 1,000 pounds are shown in the table. Other chemicals and gases would be used in smaller quantities (INL 2020d).

Table B–26. Annual Resource Requirements for Versatile Test Reactor Spent Fuel Treatment at the Fuel Conditioning Facility

Resource	*Units*	*Usage*
Staff	FTE	18 [a]
Electricity	kWh	8,300,000
Potable Water	gallons	230,000
Chemicals		
Alcohol	pounds	21,000
Acetone	pounds	14,000
Decon	pounds	14,000
Sodium hydroxide solutions	pounds	7,800
Oil	pounds	2,300
Paint/Paint thinner	pounds	1,800
Gases		
R-22 refrigerant in nitrogen/air	liters	2,700

FTE = full-time equivalent (person); kWh = kilowatt-hour.
[a] New staff; in addition, 66 current workers would be shared with existing programs.
Source: INL 2020c.

Nonradiological releases

The FCF is an existing operational facility at the MFC. The addition of VTR spent fuel treatment activities is not expected to increase the amount of nonradiological emissions from this facility.

Radiological Releases

Radiological releases were estimated based on current releases from the FCF. These estimates are presented in **Table B–27**. All releases from the facility would pass through HEPA filters (and from the main cell additional carbon filters) before being released through the facility stack. The combined flow rate would be about 34,900 cubic feet per minute at ambient temperatures. The release would be through a 60-inch diameter stack at an elevation of about 200 feet.

Table B–27. Idaho National Laboratory Spent Fuel Treatment Facility Operational Annual Radiological Releases

Isotope	*Curies*	*Isotope*	*Curies*
Antimony-125	1.57×10^{-7}	Krypton-85	8,250
Cadmium-113m	4.15×10^{-10}	Nickel-63	2.76×10^{-10}
Cerium-144	1.41×10^{-6}	Promethium-147	1.25×10^{-7}
Cesium-134	2.62×10^{-7}	Plutonium-238	1.24×10^{-10}
Cesium-137	1.96×10^{-6}	Plutonium-239	2.83×10^{-9}
Cobalt-60	2.08×10^{-9}	Plutonium-240	1.87×10^{-10}
Europium-154	1.73×10^{-10}	Plutonium-241	1.17×10^{-9}
Europium-155	2.07×10^{-9}	Ruthenium-106	5.66×10^{-6}
Iron-55	5.50×10^{-8}	Samarium-151	8.97×10^{-10}
Hydrogen-3 (Tritium)	510	Strontium-90	3.47×10^{-8}

Note: Only isotopes with a release of 1×10^{-10} curies or greater are listed.
Source: INL 2020d.

Waste generation

Annual waste generation rates for spent fuel treatment are based on the treatment of 45 driver fuel assemblies per year, a total of approximately 1.8 metric tons of heavy metal. These estimates are provided in **Table B–28**.

Table B–28. Idaho National Laboratory Spent Fuel Treatment Facility Annual Operational Waste

Waste Type	*Category*	*Volume (cubic meters)*		*Weight (pounds)*	
		Net	*Gross*	*Net*	*Gross*
Industrial	NA	1.8	2.0	4,600	4,900
Low-level waste	Contact handled	130	140	74,000	110,000
	Remote handled	160	170	170,000	230,000
Mixed low-level waste	Contact handled	10	12	13,000	15,000
	Remote handled	16	16	14,000	20,000

NA = not applicable.
Source: INL 2020d.

In addition to the waste identified here, the treated and conditioned fuel from 45 spent driver fuel assemblies, (previously identified as waste from the VTR) would be generated by spent fuel treatment. This treated fuel would be stored at the site until an offsite storage option (either an interim storage facility or a permanent repository when either becomes available for VTR fuel), at which time it would be shipped off site.

B.4.4 Spent Fuel Treatment and Storage at Oak Ridge National Laboratory

B.4.4.1 Oak Ridge National Laboratory Facilities

The storage and treatment of spent fuel at ORNL would require the construction of new facilities; no existing facilities at the site are capable of handling these activities. Spent fuel treatment of the VTR driver fuel assemblies requires the use of hot cells with an inert atmosphere. ORNL has no such hot cells. A conceptual design[16] for this facility has been developed to meet the process requirements identified in Section B.4.2, using equipment similar to that identified under the INL VTR Alternative in Section B.4.3. The spent fuel treatment activities would occur within the same facility envisioned for post-irradiation examination of test specimens (see Section B.3.4). Both the fuel treatment and temporary storage facilities would be located within the same protected area as the VTR.

The spent fuel treatment portion of the hot cell facility would have its own set of inerted hot and decontamination cells. The spent fuel treatment hot cell would be a concrete-shielded, steel-lined enclosure with interior dimensions of 30 feet wide by 70 feet long by 25 feet high. It would be filled with argon gas that provides an inert, non-oxidizing atmosphere. The associated decontamination cell would be a concrete-shielded, steel-lined enclosure with interior dimensions of 30 feet wide by 20 feet long by 25 feet high. It would be filled with air. The interior surfaces would be lined with steel. A raised steel floor would extend over part of the cell. Sections of the raised floor could be removed for access to the subfloor area. Test samples and equipment would be moved using two 5-ton cranes and electromechanical manipulators. The space beneath the removable floor would be used for storage; it would also house gas ducts and filters, and serve as additional space (depth) for vertical handling of long items.

There would be penetrations in the cell walls, roof, and floor for windows, utility service, feedthroughs, in-cell handling equipment, gas ducting, transfer hatches, etc. Penetrations into each cell would be steel-

[16] The conceptual designs have been developed for NEPA purposes only. This conceptual design is not as detailed as, nor is it to be considered, the conceptual design that is a part of the DOE facility design process.

lined, welded to the cell liner, and surrounded by high-density shielding closures or inserts. Closures or inserts for the penetration liners would have double seals, with the space between them pressurized with an argon purge.

The fuel treatment hot cell would have 15 work stations, each about 10 feet wide, equipped with a shielding observation window (layers of leaded glass with thin layers of mineral oil between them, plus a protective non-leaded glass plate on the cell side). Stations would be equipped with lights, utility distribution systems (electric and pneumatic), examination equipment, work tables, and up to two master/slave manipulators. The cell would be designed so that equipment could be added or removed from the work station without releasing radioactive contaminants, diluting the inert cell atmosphere, or extensively interrupting work at adjacent stations. The interior of the hot cell would be lighted, and high-intensity lighting would be provided in the cell at each active work station. Emergency lighting would also be provided.

The fuel treatment decontamination cell would be a shielded hot cell with an air atmosphere, maintained at a negative pressure relative to the surrounding corridors to minimize the spread of contamination. The decontamination cell would have six work stations and six leaded-glass observation windows. The decontamination cell would be separated from the inerted cell by an ordinary concrete shielding wall. The decontamination cell would be the same width and height as the inerted cell, and its outer walls similarly constructed. The cell floor would be lined with stainless steel, and the lower walls would be lined with carbon steel coated with epoxy paint. Electrical and pneumatic services in each decontamination cell would be generally similar to those in the inerted cell.

The spent fuel temporary storage facility would be similar to that proposed for use under the INL alternative, a concrete pad (see Section B.4.3.1).

B.4.4.2 Environmental Resources – Construction

Resource Requirements

Spent fuel treatment would be collocated in the same building as the post-irradiation examination capability at ORNL, the new Post-Irradiation Examination and Fuel Treatment Facility. Environmental resources associated with the construction of the Spent Fuel Treatment Facility have been included in the resources identified for the facilities used for post-irradiation examination of test specimens at ORNL (see Section B.3.4.2).

In addition to the spent fuel treatment capability, a spent fuel pad would be constructed at the VTR site at ORNL. The environmental resource requirements associated with this construction activity are presented in **Table B–29**.

Nonradiological Releases

Nonradiological releases are associated with the operation of trucks and construction equipment (i.e., the burning of diesel fuel). Types and duration of operation for the equipment used during construction are discussed in the main body of this EIS. Emissions associated with equipment have been included in the estimates for construction of the VTR at ORNL in Table B–16.

Waste Generation

Small amounts of waste would be generated during construction of the spent fuel pad. Waste would consist of 2 cubic yards of concrete and 10 cubic yards of municipal waste. It has been assumed that about 2 percent of this waste would be hazardous waste (Leidos 2020).

Table B–29. Oak Ridge National Laboratory Spent Fuel Treatment and Storage Facilities Construction Resource Requirements

Resource	Units	Total
Staff	FTE	8
Electricity	kWh	1,800
Gasoline	gallons	580
Diesel Fuel		
Road Diesel	gallons	35,000
Non-road Diesel	gallons	5,200
Total Diesel	gallons	40,000
Water		
Potable	gallons	100,000
Dust Control, etc.	gallons	NA
Total	gallons	100,000
Structural Concrete	cubic yards	2,700
Rebar	tons	72
Excavation	bank cubic yards [a]	4,700
Asphalt	tons	1,900
Backfill (rock/gravel)	cubic yards	4,600
Cable	linear feet	6,500
Conduit	linear feet	6,500
Fencing	linear feet	10,000
Isolation Area Rip Rap	cubic yards	12,200

FTE = full-time equivalent (person); kWh = kilowatt-hour; NA = Not Applicable.

[a] A bank yard is the volume of earth or rock in its natural state, as compared to the expanded volume after excavation.

Source: Leidos 2020.

B.4.4.3 Environmental Resources – Operations

The nominal test cycle duration for the VTR would be 100 effective full-power days. At the end of each cycle, up to 15 spent fuel assemblies could be removed from the core (INL 2020c). The spent fuel assemblies would be allowed to cool within the reactor vessel for a period of time, nominally a year. When removed from the reactor vessel and after being cleaned (sodium removal), these spent fuel assemblies would be transferred to the spent fuel pad. After an additional cooling period, at least 3 years, these assemblies would be transferred to the new Post-Irradiation Examination and Fuel Treatment Facility for treatment and consolidation. The resulting spent fuel would be returned to and stored at the spent fuel pad until transferred to an offsite location (either an interim storage facility or a permanent repository when either becomes available for VTR fuel), at which time it would be shipped offsite.

Spent fuel treatment would be collocated in the same building as the post-irradiation examination capability at ORNL, the new Post-Irradiation Examination and Fuel Treatment Facility. Environmental resources associated with the operation of the Spent Fuel Treatment Facility have been included in the resources identified for the facilities used for post-irradiation examination of test specimens at ORNL (see Section B.3.4.3).

B.5 Reactor Fuel Production

B.5.1 Introduction

The design of the VTR driver fuel assemblies was discussed in Section B.2.3. The driver fuel assembly and fuel pin designs are based on the most recent fuel designs for the EBR-II and metal fuel demonstrated in the FFTF. The VTR core would contain 66 driver fuel assemblies. These hexagonal assemblies would be approximately 3.85 meters in length and 11.7 centimeters wide (flat surface to flat surface). Each driver fuel assembly would contain a bundle of 217 fuel pins, upper and lower shield blocks, a grid to which the lower end plugs of the fuels are fixed and a surrounding hexagonal duct with upper and lower adaptors. Each of the fuel pins would be 1.65 meters long with a diameter of 0.625 centimeters. Within the fuel pin, there would be fuel slugs with a total length of 80 centimeters. The fuel pins would also have an 80-centimeter plenum (for a plenum-to-fuel volume ratio of approximately 1) filled with argon (and possibly a mixture of tag[17] gas isotopes) near atmospheric pressure. Upper and lower end plugs, made of the same material as the cladding, would be seal-welded to the cladding tube and the completed fuel pin would be helically wrapped with a spacer wire on a 15.2-centimeter (6-inch) pitch.

> ***Ingot*** – an oblong metallic block consisting of one of the fuel elements; plutonium, uranium, and zirconium
>
> ***Fuel slug*** – a cylindrical rod of alloyed fuel to be inserted into the fuel pin
>
> ***Fuel pin*** – a single rod of fuel. The pin consists of a cladding tube, with top and bottom end plugs, containing fuel slugs, sodium-bonded to the cladding, and an inert gas plenum above the fuel.
>
> ***Fuel assembly*** (sometimes referred to as a subassembly) – a hexagonal array of 217 fuel pins, top and bottom reflectors (shields) surrounded by an assembly duct with assorted mechanical components.

The metallic fuel (consisting of an alloy of uranium, plutonium, and zirconium) to be used in the VTR is unique and would be fabricated at a DOE facility separate from the VTR. Materials available for use in the production of the metallic fuel (feedstock) exist in several forms. Plutonium feedstock may be in the form of metals or oxides; uranium feedstock (of varying enrichments) may be in the form of metals, oxides, or nitrates. The fuel form for the fuel pin is a cast metallic cylindrical slug. The steps needed to convert these various feedstocks into VTR fuel would be:

- Conversion of feedstock from non-metallic forms to metals, if needed;
- Removal of impurities from feedstock, if needed;
- Fuel alloying and homogenization;
- Fuel slug casting and demolding;
- Assembly of the fuel slugs into fuel pins; and
- Assembly of the fuel pins into driver fuel assemblies.

The first two steps identified above would occur within a single facility, a feedstock preparation facility. The remaining steps would occur in a separate facility, the fuel fabrication facility. (If a single site were to be selected for both facilities, a single facility could be used to house both.) DOE has identified options for the siting of each of these activities, the INL Site and Savannah River Site (SRS). Separate sites could be selected for the two facilities; both could be located at the same site or either alone could be located at INL or SRS.

If sited at either INL or SRS, neither the feedstock preparation facility nor the fabrication facility would require the construction of a new facility, rather the equipment required would be installed within existing facilities (INL 2020c; SRNS 2020).

[17] Tag gas is a gas added to gas plenum used to help identify the location of any cladding leaks.

B.5.2 Versatile Test Reactor Fuel Production

The fuel needs for operation of the VTR were identified in Section B.2.3. Each year the VTR would need to replace up to 45 driver fuel assemblies. These assemblies would contain about 1,800 kilograms of fresh fuel; 400 kilograms of plutonium and 1,400 kilograms of uranium. Fuel production would require more than this amount of feed material to account for material left in the furnace during casting and rejected fuel rods (rods that do not meet fuel quality standards) that end up as fuel production waste. The efficiencies of the various fuel production operations vary, but as much as 27 percent of the fuel feedstock could end up as waste stream.[18] With this amount of feedstock becoming waste as much as 550 kilograms of plutonium and about 1,900 kilograms of uranium could be required to fabricate the 45 driver fuel assemblies per year. Over the 60-year lifetime of the VTR, this would result in the need for about 34 metric tons of plutonium and 120 metric tons of uranium feed material (SRNL 2020).

Not all of the plutonium available for the VTR exists in a form suitable for direct use in the driver fuel fabrication process. Preparation of the source material may be required to convert the plutonium into a metal and to remove impurities (polish) from the plutonium. Americium-241 is one of the primary elements targeted for removal, due to its impact on worker exposure.

Uranium is expected to be received in a form (metallic, acceptable impurity content) for use directly in the fuel fabrication process.

Feedstock Preparation

Feedstock preparation would address the first two steps in fuel production: conversion of feedstock from non-metallic forms to metals and fuel purification, removal of impurities. (Preparation is not anticipated to be required for uranium fuel feeds since metallic uranium fuel of the appropriate enrichment is commercially available.) There are several process options available for feedstock preparation. The selection of a preferred process methodology would depend upon, among other factors, the form and purity of the plutonium made available for the VTR program. Depending upon the form and quality of the plutonium feed, not all of the process steps described below may be necessary. It is even possible that plutonium with acceptably low impurity levels and in a metallic form could be available for the VTR. In that case feedstock preparation would not be necessary. In addition to the feedstock preparation processes described below, other preparation processes are available. Even within the processes described, potential variations could be utilized. A final determination of the processes that would be used for the VTR program has not been made.

Three potential feedstock preparation processes are under consideration for VTR feedstock preparation: an aqueous capability, a pyrochemical capability, and a combination of the two.[19] In the aqueous process, the plutonium feed (containing impurities) is dissolved in a nitric acid solution and put through a series of extraction and precipitation steps until a polished plutonium oxide is produced. The proposed process then converts the oxide to a metal in a direct oxide reduction process. (A potential variation of this process would be to precipitate the oxide with plutonium trifluoride and convert the cake to a mixture of plutonium dioxide and plutonium tetrafluoride that could be then reduced directly to plutonium metal, if adequate worker shielding could be provided.) In one form of the pyrochemical process (molten salt extraction [MSE]), the metallic plutonium feed is combined with a salt, the mixture is raised to the melting point, and an electrical current is passed through the solution. Impurities (such as americium) react with the salt and the purified plutonium is collected at the bottom of the reaction crucible. If the pyrochemical

[18] The highest percentage of feedstock material entering the waste stream would be associated with an option where no feedstock preparation would be necessary and no provisions were made to recapture some of the material that could otherwise end up in the waste stream. Other fuel production options could result in less waste and a smaller quantity of plutonium and uranium feedstock.

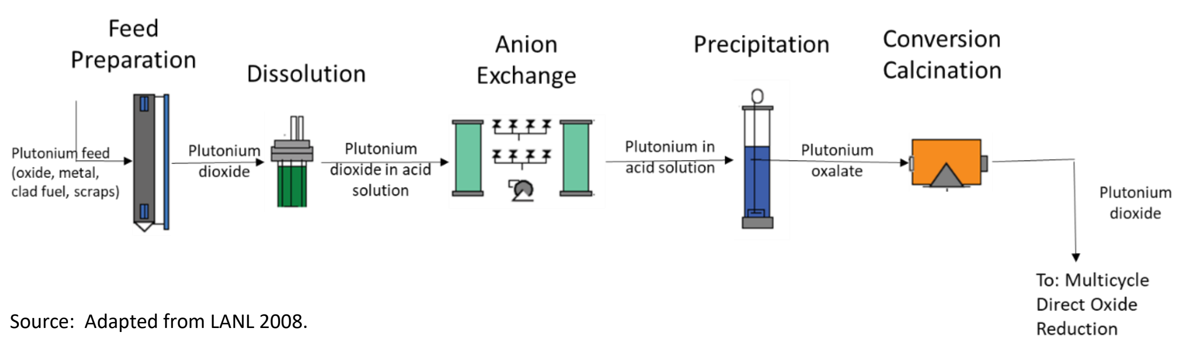
[19] Oher processing options are available, including; a trifluoride precipitation process and direct dissolution of plutonium/uranium alloys.

process were selected, a direct oxidation reduction process would also be required to convert plutonium dioxide feeds to plutonium metal. Either process (aqueous or pyrochemical) could be used to process unusable fuel from the driver fuel fabrication process. If a combination of the two processes were to be selected, a smaller aqueous line to prepare this reject fuel could be incorporated into the pyrochemical process.

Regardless of the feedstock preparation process, each step in the feedstock preparation process would take place within enclosures intended to protect workers and to help limit releases. At this stage in the design process, DOE envisions feedstock preparation being performed in gloveboxes.[20] The design for the feedstock preparation process is in an early stage of development, and hot cells may be a preferred alternative to gloveboxes to mitigate workforce exposure for some operations (SRNS 2020).

Aqueous Plutonium Processing

The aqueous process is the most mature of the three feedstock preparation processes being considered. It is also the process capable of handling the widest variety of feeds and the easiest to automate. Feed material for the aqueous process would consist of "new" feed material and scraps from the driver fuel fabrication process. Although not the only form of aqueous processing, the major steps, **Figure B–20**, in the aqueous process identified for use with VTR fuel production (SRNL 2020; INL 2020e) include the following:



Source: Adapted from LANL 2008.

Figure B–20. Major Steps in Aqueous Processing

Feed preparation – Plutonium could be received in many forms: clad fuel or unclad material and in either an oxide or metallic form. The aqueous process works best with oxide feeds; dissolving metal feeds produces an unstable residue. Any feed material received in a metallic form would be converted to plutonium dioxide. Clad material would be processed to remove the cladding. The resulting materials would be ground to facilitate dissolution.

Dissolution – The plutonium dioxide would be dissolved in a strong nitric acid solution with other solvents (e.g. fluoride) and water. The resulting solution is filtered to remove any solid material (scrap).

Anion Exchange – The resulting solution is passed through an anion exchange column where a resin bed selectively absorbs the plutonium. The resin bed is an organic polymer that has positively charged sites imbedded in the solid polymer. Negatively charged mobile ions (in this case nitrates) balance the charge of the polymer. The resin preferentially captures the negatively charged plutonium in solution with the nitric acid, displacing the nitrates, while allowing impurities (americium, uranium, fluoride, etc.) to pass through the resin bed. The plutonium would be washed from the resin using a weak (nitric) acid solution.

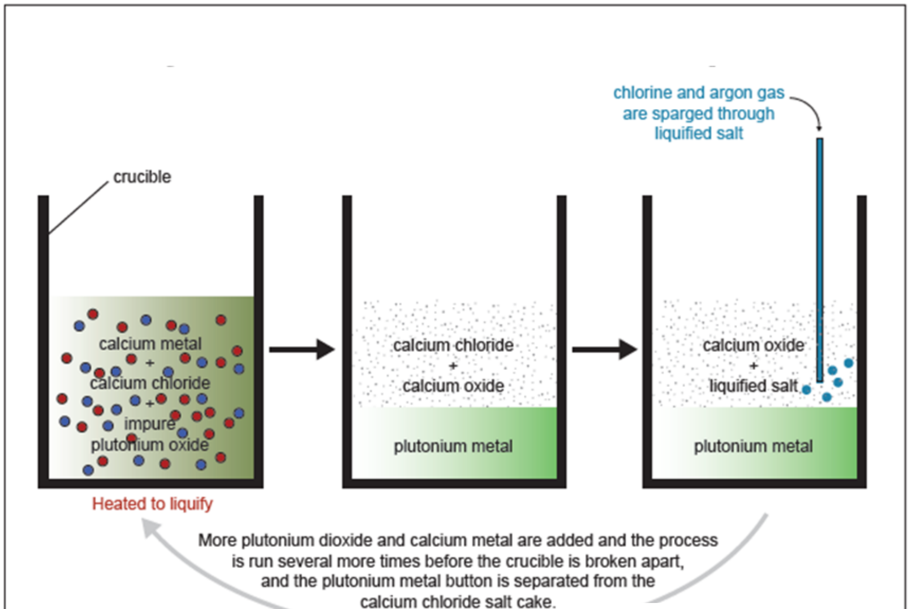
[20] Gloveboxes are sealed enclosures with gloves that allow an operator to manipulate materials and perform other tasks, while keeping the enclosed material contained. In some cases, remote manipulators may be installed in place of gloves. The gloves, glass, and siding material of the glovebox can be designed to provide worker radiation protection.

Precipitation – The product of the anion exchange is a weak acid solution that contains the purified plutonium. This solution is combined with another acid that reacts with the plutonium to produce an insoluble compound of plutonium, which is collected on a filter.

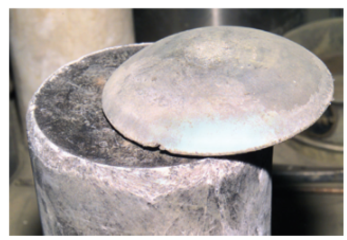
Conversion (*Calcination)* – The insoluble plutonium compound is put into a calciner, a vessel in which the plutonium is heated and dried. Oxygen is added to the calciner, reacting with the plutonium compound, creating plutonium oxide.

The plutonium oxide is the final product in the aqueous plutonium purification process. This product would be converted to metallic plutonium and cast into ingots for use in the fuel fabrication process.

Multicycle Direct Oxide Reduction (MDOR) – Direct oxide conversion (**Figure B–21**) converts oxide to metal feeds. The plutonium oxide is combined with a salt (calcium chloride) and calcium metal in a crucible within a furnace and heated to melt the mixture. The plutonium oxide and calcium react, producing plutonium metal and a mixture of calcium oxide and liquefied salt. As the mixture cools the plutonium metal (called plutonium buttons) collects at the bottom of the crucible. In a once-through process, the calcium/salt mixture retains a significant amount of the plutonium. However, the salt and calcium can be regenerated and reused in multiple oxide conversion cycles, thus reducing the amount of plutonium lost in the process.



Source: Adapted from LANL 2008.



Plutonium Button

Figure B–21. Multicycle Direct Oxide Reduction

Casting – This final step in the feedstock preparation process produces the ingots for fuel fabrication. The output of the MDOR is vacuum cast into ingots in a furnace. The furnaces use a reusable crucible for melting, a coated graphite crucible to collect the casting, and are operated at 800 °C, under vacuum. This final step removes salt and slight impurities from the buttons.

Waste Handling – Radioactive waste is generated in most of the steps of aqueous and MDOR processing. Waste material from feed preparation and plutonium dissolution would have to be dried, oxidized, and downblended or immobilized (combined with an inert material). Liquid waste from anion exchange and precipitation would be processed to recover acids and the remaining waste would be solidified via evaporation. Each of these operations would require specialized equipment operated in gloveboxes. Crucibles from the MDOR and casting (collection of the plutonium products involves breaking the crucibles) would be wastes.

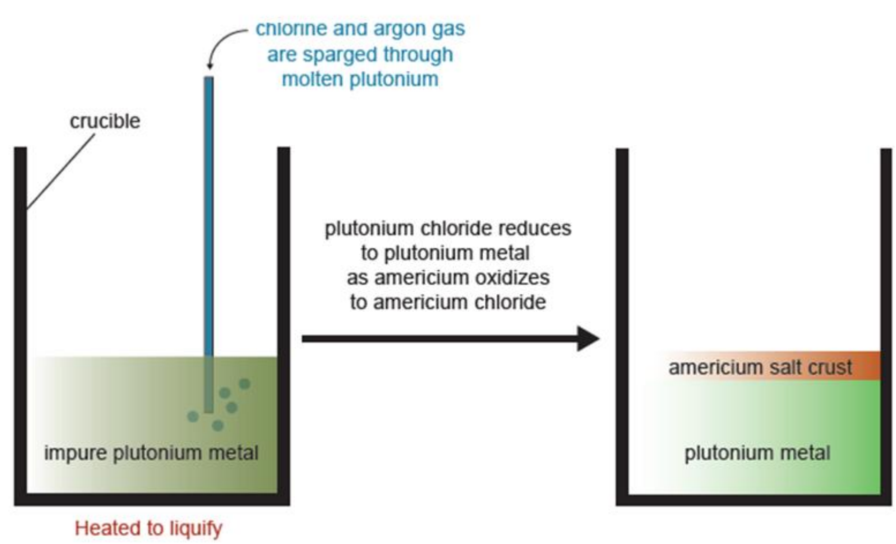
Pyrochemical and Electrorefining Plutonium Processing

The pyrochemical process would process the plutonium in metallic form rather than oxides needed for the aqueous process. The pyrochemical process has the advantage that fewer steps are involved in the purification process, and the entire operation would require less space than the aqueous process. However, the process identified for handling the majority of the plutonium does not handle feed material with higher impurity content as well as the aqueous process. Two separate processes would be utilized for VTR fuel. An MSE process would be used for "new" feed; an electrorefining process would be used for driver fuel fabrication product and "new" feed with higher impurity content.

The major steps in the preparation of the "new" plutonium feed by pyrochemical processing (SRNL 2020; INL 2020d) would include the following:

Feed Preparation – The same MDOR process described above would be used to convert any oxide feed to metal. Metallic feeds would not require any feed preparation.

Molten Salt Extraction (MSE) – In MSE (also called Metal Chlorination) (**Figure B–22**), plutonium metal is processed in batches with a salt. The mixture is heated to the melting point in a crucible, and chlorine gas is mixed with the molten mixture. This produces compounds of americium and plutonium, resulting in almost all of the americium and some of the plutonium being retained in the salt. The salt separates from the metallic plutonium, forming a salt crust that can be removed from the plutonium metal, and when mixing and heating is stopped the plutonium forms a button.



Source: Adapted from LANL 2008.

Figure B–22. Molten Salt Extraction

Vacuum Casting: Vacuum casting removes excess chloride and light metallic impurities, as described under the aqueous process. The resultant button is expected to be of sufficient purity to meet the VTR specification, without any further processing, provided the feeds were pure enough.

Waste Processing – MSE produces salt wastes (salts containing impurities such as americium and some plutonium) that would be processed with a salt scrub and salt oxidation and disposal – the scrub alloy process uses an aluminum-magnesium alloy to scrub the molten salt; impurities form a new alloy with the aluminum. The process removes most of the plutonium, essentially all of the americium, and produces a scrub alloy ingot. Crucibles from the casting process would also be processed using the salt scrub. Waste salts would be oxidized and disposed as drummed waste.

Recyclable fuel fabrication products would be processed using an electrorefining process. In addition, "new" plutonium feed that contains a higher impurity content may need to be processed using electrorefining, due to the lesser ability of MSE to remove impurities. The major steps in the electrorefining process (SRNL 2020; INL 2020e) would include the following:

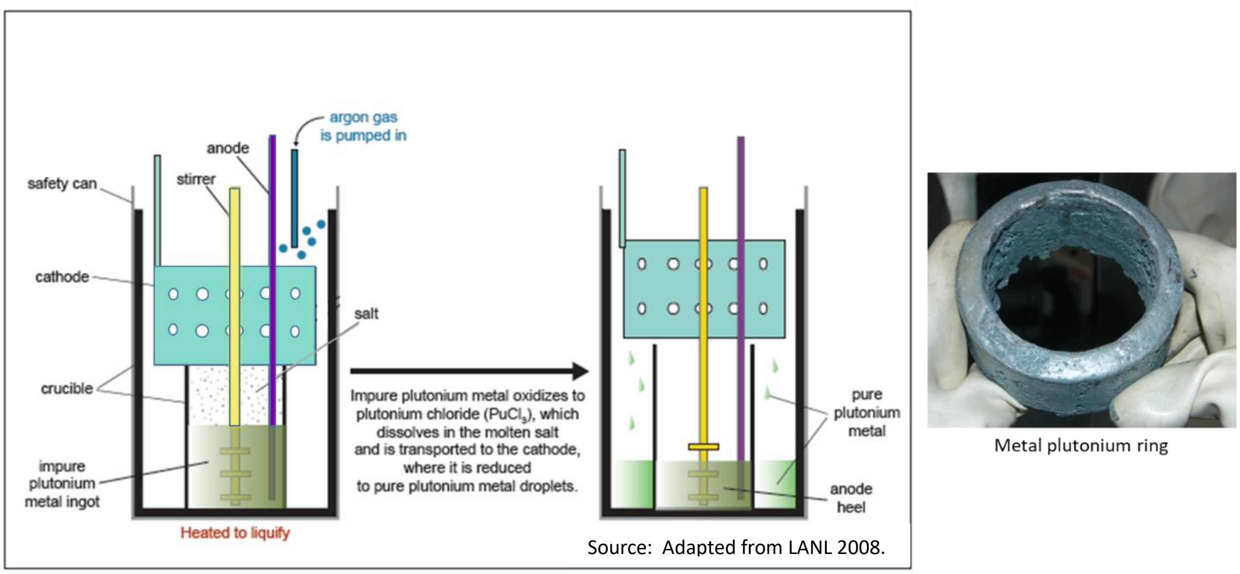
Feed Preparation – The material being dissolved would be chopped to increase its surface area. After chopping, the material would be placed in an anode basket and sent to the electrorefiner.

Electrolytic Reduction/Chlorination – "New" oxide feeds could be reduced to metal using either electrolytic reduction or chlorination. Electrorefining operates more efficiently when there are small quantities of metal chlorides in the salt mixture. A chlorination furnace could be included to produce these compounds as needed. Electrolytic reduction (essentially a single-step version of MDOR) could be used to prepare "new" oxide feeds for electrorefining. Electrolytic reduction could be used for oxide conversion, since the electrorefining process does not require the purity of feed material that the MSE process does.

Electrorefining – In the electrorefining process (**Figure B–23**), the chopped fuel is placed in a basket (or multiple baskets) in a molten salt. The basket acts as the anode (the negatively charged electrode) for the electrorefining process. A direct current is then passed between the anodes and cathodes (the positively charged electrodes), which dissolves and oxidizes the plutonium and uranium into the molten chloride salt. Multiple cathodes, at different electric potentials, allow deposition of uranium and plutonium metal onto different cathodes. In a typical arrangement, the anode is the inner section of a disc shape and the cathode is the outer ring of this disc shape.

Casting – Vacuum casting, similar to that used in the pyrochemical processing would be required to form the ingots used in fuel fabrication.

Waste Processing – Waste processing for the electrorefining process would be similar to that for the MSE process.



Source: Adapted from LANL 2008.

Figure B–23. Plutonium Electrorefining

Pyrochemical and Aqueous Plutonium Processing

The third feedstock preparation process utilizes the pyrochemical process (MDOR, MSE, and vacuum casting) in combination with a small aqueous line. The pyrochemical process would be used for "new" feeds, and the aqueous process would be used to further process products of the pyrochemical processing, as well as the unusable driver fuel. Additional processing would be required if impurity levels in the "new" feed plutonium are too high. A small electrorefining process line might be included in this option for these feeds. The processing steps would be the same as previously described for each process; although, the aqueous process would be on a smaller scale than needed if used to process all plutonium feeds.

Driver Fuel Fabrication

The driver fuel fabrication process takes the metallic uranium, plutonium, and zirconium metals and fabricates the finished driver fuel assemblies. Steps in the process include fuel alloying and homogenization, fuel slug casting and decasting, fuel pin assembly, and fuel assembly fabrication. Through pin assembly, these activities would occur in gloveboxes. (The design for the fuel fabrication process is in an early stage of development, and hotcells may be a preferred alternative to gloveboxes to mitigate workforce exposure for some operations (SRNS 2020)). Unless otherwise noted, information in this section is from the *VTR Fuel Facility Plan* (INL 2019a).

Fuel Alloying and Homogenization – An induction casting furnace would be used in the initial steps in the fuel fabrication process, alloying the elemental metallic components and producing the fuel slugs. (A possible design for the induction casting furnace is shown in **Figure B–24**.) This furnace would be contained within a glovebox with an inert gas atmosphere (see **Figure B–25**).

With the glovebox inerted, fuel constituents would be mixed together in their elemental metallic forms (i.e., as pre-weighed buttons, ingots, or chunks of uranium, plutonium, and zirconium) and melted together in a melt crucible to produce a chemically homogeneous uranium-plutonium-zirconium (U-Pu-Zr) alloy. This alloying and homogenization would take place in the casting furnace itself, without need for a separate fuel alloying process. The alloying step entails melting the alloy constituents and holding the melt at an alloying and homogenization temperature for some period of time. Inductive stirring in a U-10Zr melt has been shown to produce a homogenous mixture; however, for large batches of U-Pu-Zr, inductive stirring may not be sufficient to generate a homogenous mixture, and a tantalum stirrer may be required.

Fuel Slug Casting – The melted alloy of uranium, plutonium, and zirconium would be cast into cylindrical slugs by drawing the melt upward into quartz molds. The induction furnace glovebox would be evacuated to put all the molds under vacuum, and the furnace temperature would be adjusted from the homogenization temperature to the casting temperature. A mold palette (see **Figure B–26**), capable of producing about 135 fuel slugs,[21] would be preheated and then lowered into the melt crucible so every mold is dipped completely into the molten metal to a depth sufficient to keep the tips immersed in the melt throughout the casting process. The system would be rapidly pressurized to create a large differential pressure between the melt surface and the interior of the molds. The molds would then fill with molten metal. After sufficient time to allow the fuel alloy to begin to solidify within the molds, the mold palette would be raised to remove the molds from the melt.

[21] VTR operation would require the production of up to 19,530 usable fuel slugs per year when there are two fuel slugs per fuel pin. Initial plans call for two casting furnaces combined producing about four and a half batches per week (with 12 weeks of maintenance per year)resulting in the need (assuming non-recyclable and recyclable losses due to failed castings) for each batch to yield about 135 fuel slugs.

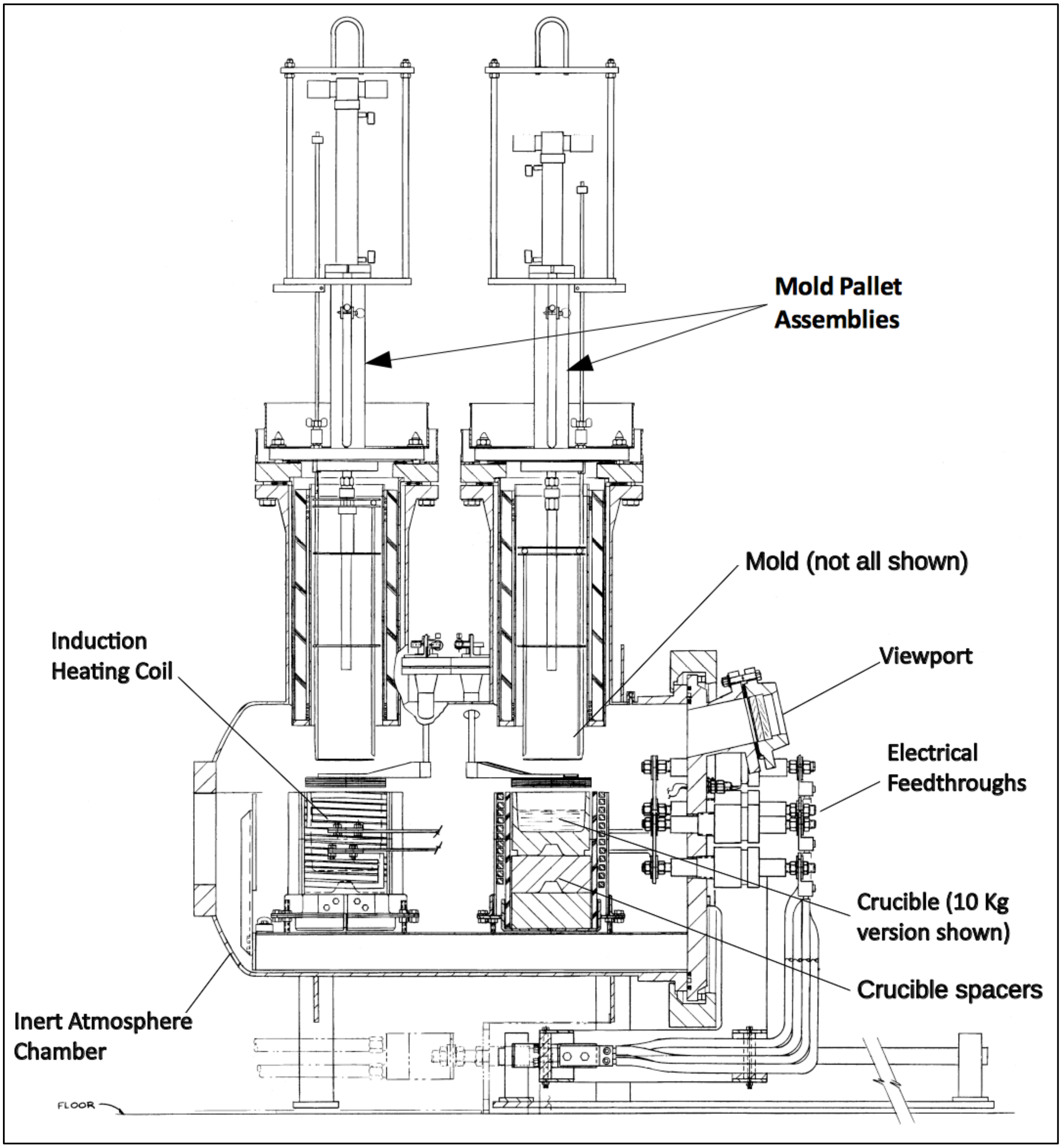


Figure B–24. Fuel Injection Casting Furnace

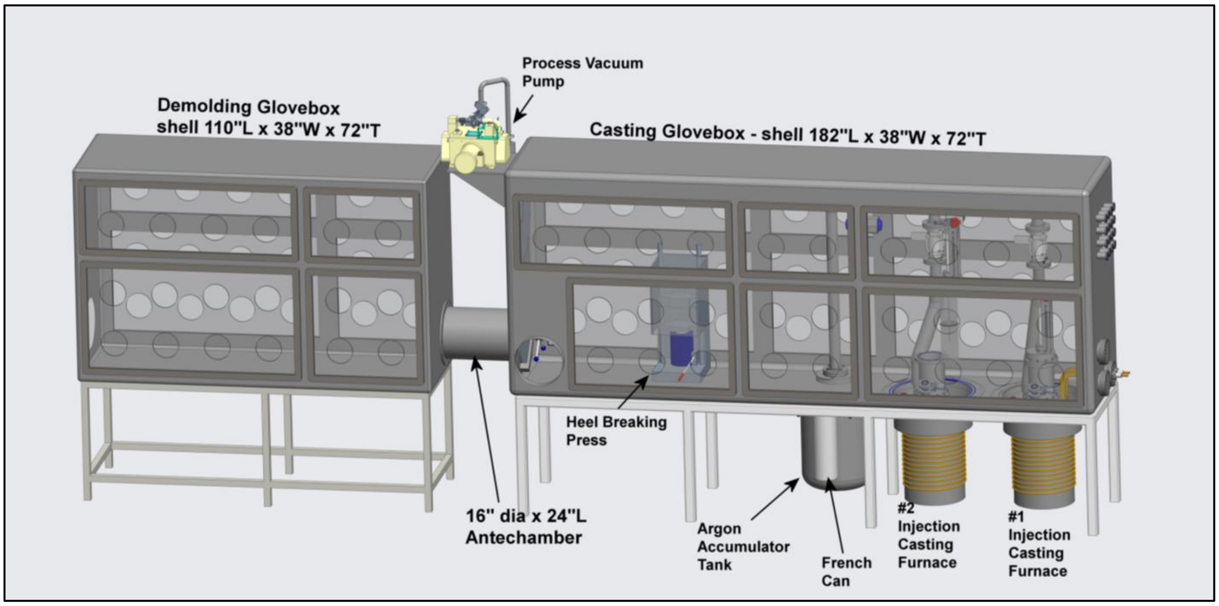


Figure B–25. Preconceptual Illustration of Slug Casting and Demolding Glovebox Line

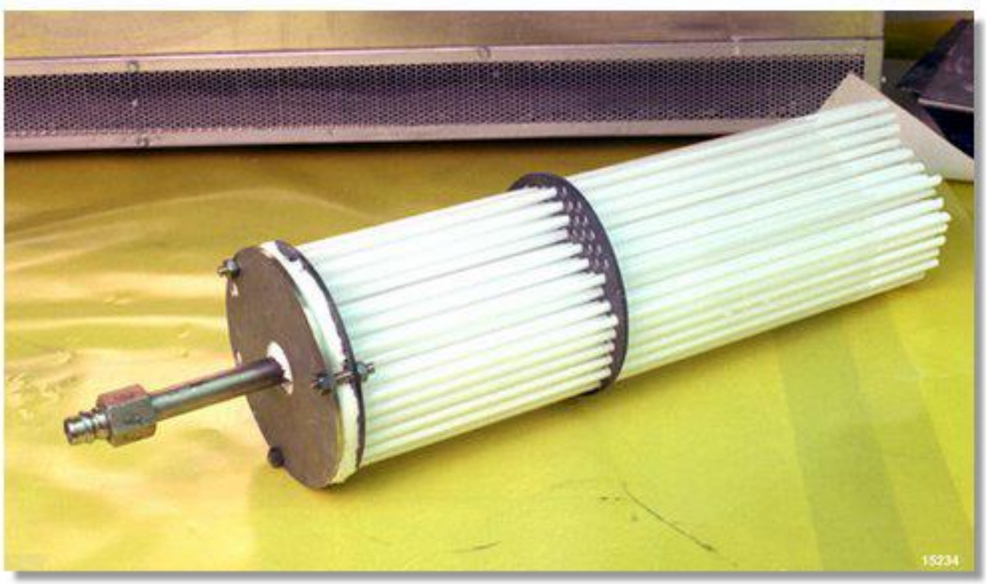


Figure B–26. Representative Casting Furnace Palette Ready for Loading into the Casting Furnace

Fuel Slug Demolding – In a separate glovebox with an inert atmosphere (see Figure B–25), the fuel slugs would be allowed to solidify and cool before removal. It may be possible to remove some fuel slugs through the palette hole, but it is expected that removing most slugs would require breaking the mold. Regardless of how the fuel slugs would be removed from the molds, molds are not reused. Once free of the mold, the fuel slug would be inspected for imperfections and surface defects. This function could be automated using machine vision to determine recoverable slug length, characterize any surface defects, and to determine straightness. Following inspection, the slug would be sheared to length, with final dimensions (length and diameter) measured by machine or manual inspection. Sheared material may be used for chemical analysis sampling, to determine that the fuel slugs meet specifications.

Fuel Pin Assembly – The prepared fuel slugs would be transferred to a third glovebox (see **Figure B–27**) with an inert atmosphere (argon with small amounts of helium) for fuel pin assembly. Fuel pin assembly would consist of loading sodium (for bonding) and fuel slugs into a cladding jacket (a fuel pin cladding tube with the lower end plug welded into place).

After empty cladding jackets are introduced into the glovebox, extruded sodium metal would be inserted to slide to the bottom of the jacket. The amount of sodium inserted, when melted, would be sufficient to fill the fuel-cladding gap and provide a 2-centimeter cover above the top of the fuel. Fuel slugs, totaling 80 centimeters in length, would then to be inserted into the cladding jacket, to rest on top of the sodium. Finally, the top end plug would be pressed into the cladding jacket and the pin seal welded. The argon/helium glovebox atmosphere is the gas composition enclosed into the 80-centimeter fuel rod plenum. Helium would be included in the plenum gas to enable leak checking of the pin for a hermetic seal. After seal closure, fuel pins would be decontaminated and cleaned, which ensures that fuel pins can be handled outside the glovebox without plutonium contamination concerns. The final step in fuel pin assembly would be to wind the HT-9 steel, wire wrap spacer around the pin. The wrap would be welded to one end plug wrapped around the fuel pin and welded to the other end plug.

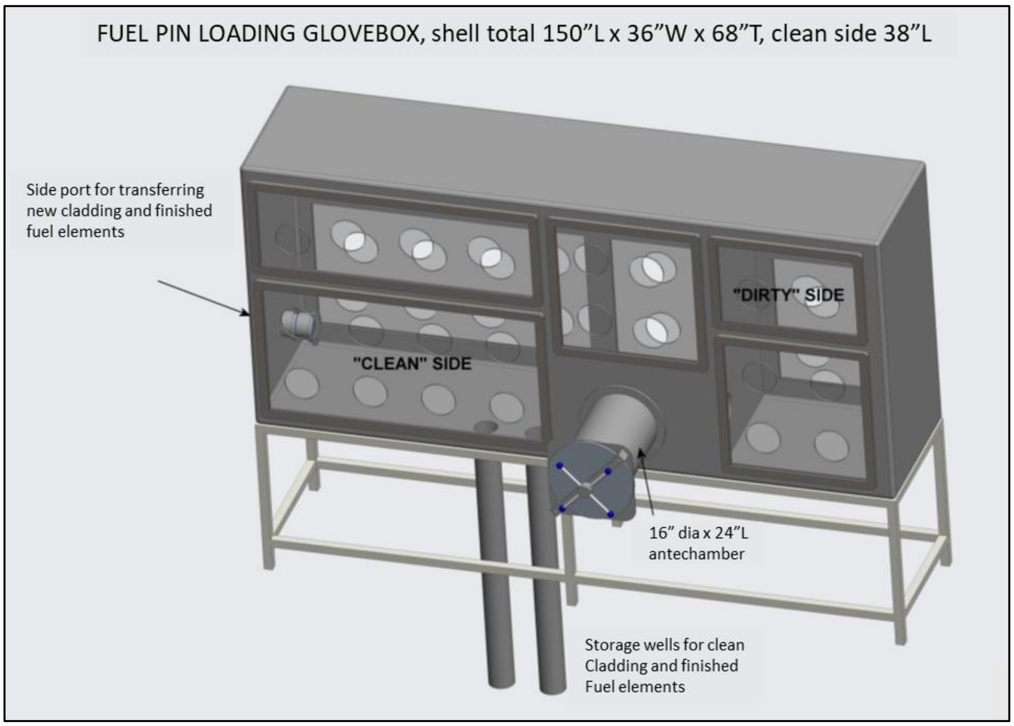


Figure B–27. Preconceptual Illustration of a Fuel Pin Loading Glovebox

Assembly Fabrication – The driver fuel assembly, described in Section B.2.3, would be fabricated using a Vertical Assembly Device, a fixture and loading station. The inlet assembly and lower shield block would be loaded into the device. A T-bar grid, providing proper spacing for the fuel pins, would be installed, and the fuel pins would be inserted into the grid such that the wire wraps properly intermesh. The upper shield block would be installed atop the fuel pins. Finally, the duct assembly (the duct, upper shield, and upper handling socket) would be inserted over the fuel pins and the duct would be secured to the assembly. The completed driver fuel assembly would be heated to melt the sodium filling the space between the fuel and cladding, providing a layer of sodium above the fuel slug. The assembly would be cooled and inspected, measured, and straightened, if needed. These operations would be carried out behind shielding; gloveboxes would not be required.

B.5.3 Idaho National Laboratory Site Reactor Fuel Production Options

Either or both feedstock preparation and driver fuel fabrication could be located at the INL site. Each option is described independently in the following sections. The equipment required for either process could not be used for the other. However, there could be some benefit, in reduced resource use, in locating both options at the same site. In particular, construction resource use for both options may be less than the sum of resource use for the two options.

As described in the following paragraphs, DOE has identified existing MFC facilities that would be capable of supporting all fuel production activities. All of these facilities are currently in use and some (e.g., the ZPPR cell) have been identified as possible locations for future programmatic missions other than VTR reactor fuel production. Based on DOE programmatic and scheduling priorities, use of these facilities by other programs may result in their being unavailable to the VTR Program. Should this happen, modifications to enlarge an existing facility or the use of other MFC or VTR facilities would be evaluated to assess their capability to support the VTR Program. Any changes to the facilities being considered to host VTR reactor fuel production would be subject to review under the National Environmental Policy Act (NEPA).

B.5.3.1 Idaho National Laboratory Site Feedstock Preparation

B.5.3.1.1 Idaho National Laboratory Site Feedstock Preparation Overview

At the INL Site, this capability would be located in the FCF (a hazard category 2 facility[22]), but not in the FCF hot cells. Equipment would be installed in the Operating Floor/High Bay, the Mockup Area, and Workshop. Additionally, some space in the outer annulus of the FCF operating floor could possibly be repurposed for feedstock preparation. Equipment and operations currently located within this portion of the FCF would be relocated within the MFC. The identified area would be suitable for pretreatment operations like molten salt removal of the americium from plutonium (polishing) and direct oxide reduction and electrorefining to convert fuel compounds (e.g., fuel oxides) into their metallic form. The facility has space available that can be used to install the equipment required for these operations (INL 2020e).

At the current level of development for this process, designs for the glovebox have not been developed. Conceptually, they would be similar to gloveboxes currently used for plutonium processing. However, differences in size (based on processing rates) or the use of automation or other mechanisms to control worker dose would be expected.

Preparing the plutonium using the aqueous process (with direct reduction of the aqueous process plutonium dioxide product to plutonium metal) requires the largest area, and this process has been used to estimate the preparation area required. If the aqueous process is selected, the equipment required for feedstock preparation would consist of the following glovebox lines (INL 2020e):

- One line for feed preparation and product staging,
- Two lines for dissolving and adjustment,
- One line for anion reaction,
- Two lines for oxide conversion,
- One line for waste immobilization,
- Two lines for acid recycle and evaporators (2 lines approximately 60-foot each), and
- One line for accessory tanks.

[22] DOE defines hazard categories by the potential impacts identified by hazard analysis and has identified radiological limits (quantities of material present in a facility) corresponding to the Hazard Categories. Hazard Category 3: Hazard Analysis shows the potential for only significant localized consequences, Hazard Category 2: Hazard Analysis shows the potential for significant onsite consequences beyond localized consequences, DOE-STD-1027-2018.

Breakdowns for the arrangement of the gloveboxes for the pyrochemical process and for the combined pyrochemical/aqueous process have not been developed.

B.5.3.1.2 Environmental Resources – Construction

Construction activities associated with the feedstock preparation facility are limited to modifications to the FCF needed to convert space from its current purpose to feedstock preparation.

Resource Requirements

Resource commitments for the modification of the FCF to house the Feedstock Preparation Facility at INL are provided in **Table B–30**. In addition to the materials identified in this table, materials used in the construction of the gloveboxes include stainless steel for structural supports, glass for glovebox windows, piping for inlet, exhaust and other gas lines, electrical cable, and conduit for power and instrument lines. Primary gases used in the gloveboxes include argon as an atmosphere and hydrogen as a mechanism to remove oxygen from the glovebox atmosphere.

Table B–30. Idaho National Laboratory Feedstock Preparation Facility Construction Resource Requirements

		Value	
Resource	***Units***	***Annual Average (peak)***	***Total*** [a]
For Modifications to Existing Facilities			
Staff	FTE	6 [a] (18 [b])	18
Electricity	kWh	Minimal [c]	Minimal
Diesel Fuel			
Forklift Fuel [d]	gallons	--	32
Mobile Crane Diesel [e]	gallons	--	120
Total Diesel	gallons	--	150
Water			
Potable	gallons	75,000	230,000
Construction Area Cleaning	gallons	1,700 (2,500)	5,000
Total	gallons	77,000	230,000
Propane, Butane		Minimal	Minimal
Gas (acetylene, oxygen)		Minimal	Minimal

FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] Construction duration of 3 years is assumed.

[b] Value represents peak number of workers at one time, not FTE.

[c] Electrical use is limited to hand held or cordless hand tools and occasional welding.

[d] Values assume 40 hours of operation and fuel consumption of 0.8 gallons per hour of operation.

[e] Values assume 30 hours of operation and fuel consumption of 4 gallons per hour of operation.

Source: INL 2020c.

Nonradiological Emissions

Nonradiological emissions during construction would be limited to emissions from personal vehicles and the cranes and forklifts used to move equipment. Emissions are presented in **Table B–31**.

Table B–31. Idaho National Laboratory Feedstock Preparation Facility Annual Nonradiological Releases During Construction

Table Calendar Year/Source Type	*Air Pollutant Emissions (tons per year)*						
	VOC	*CO*	*NO_x*	*SO_2*	*PM_{10}*	*$PM_{2.5}$*	*CO_2e (metric tons)*
Year 2024							
Onsite On-road Sources	0.000	0.02	0.002	0.000	0.000	0.000	3
Onsite Nonroad Sources	0.000	0.001	0.002	0.000	0.000	0.000	2
Offsite On-road Sources	0.001	0.13	0.01	0.000	0.004	0.001	16
Total Annual Emissions	**0.002**	**0.15**	**0.02**	**0.000**	**0.005**	**0.001**	**20**
Year 2025							
Onsite On-road Sources	0.000	0.02	0.001	0.000	0.000	0.000	3
Offsite On-road Sources	0.001	0.12	0.01	0.000	0.004	0.001	16
Total Annual Emissions	**0.001**	**0.13**	**0.01**	**0.000**	**0.004**	**0.001**	**18**

CO = carbon monoxide; CO_2e = carbon dioxide equivalent; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.
Source: Derived INL 2020d.

Waste Generation

Space within the FCF would be reallocated to support feedstock preparation. Equipment currently in this space would be relocated for use in other facilities. The removed equipment would not be waste. Waste generated during placement of the new equipment in the FCF would be minimal.

B.5.3.1.3 Environmental Resources – Operation

Resource Requirements

Key annual resource commitments for the operation of the feedstock preparation facility are provided in **Table B–32**. Resource requirements listed do not include the fuel feed material (uranium, plutonium, and zirconium).

Table B–32. Idaho National Laboratory Annual Feedstock Preparation Facility Resource Requirements

Resource	*Units*	*Value*	
		Annual	*Peak*
Staff	FTE	300	--
Electricity	MWh	6,700	--
Natural Gas	cubic feet	0	--
Heating Oil	gallons	0	--
Diesel [a]	gallons	1,500	--
Diesel (Operations)[a]	gallons	2,000	--
Water			
Potable Water [b]	gallons (thousands)	1,400	--
Process and Waste Treatment [c]	gallons (thousands)	50	--
Total	gallons (thousands)	1,500	--
Sanitary Waste Water Treatment	gallons (thousands)	1,400	--
Nitric Acid	cubic meters	88	130
Caustic	kilograms	43	64
Potassium Fluoride	kilograms	600	900
Aluminum Nitrate Nonahydrate	kilograms	300	450

Resource	Units	Value	
		Annual	Peak
Hydroxylamine Nitrate	kilograms	125	190
Polymer Resin	kilograms	40	60
Oxalic Acid	kilograms	1,400	2,100
Ascorbic Acid	kilograms	100	150
Argon	cubic meters	900,000	--
Helium	cubic meters	45,000	--
Nitrogen	cubic meters	50,000	--
Oxygen	cubic meters	5,000	--
Propane	bottles/gallons	100/470	150/700

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] Diesel fuel for one additional security vehicle and an additional diesel generator (Operations).

[b] Water use provided as gallons per minute, converted to annual assuming 8-hour work days, 5 days a week, and 50 weeks per year.

[c] Water requirements are for the aqueous processing of feedstock material. Other processes would require less.

Source: SRNS 2020.

Nonradiological Emissions

Nonradiological emissions for feedstock preparation would be associated with the transport of material to the FCF and worker vehicles. Emission data is presented in **Table B–33**.

Table B–33. Annual Nonradiological Operations Emissions from Feedstock Preparation Facilities at Idaho National Laboratory

Facility	Air Pollutant Emissions (tons per year)						
	VOC	CO	NO_x	SO_2	PM_{10}	$PM_{2.5}$	CO_2e (metric tons)
Haul Trucks	0.000	0.000	0.001	0.000	0.000	0.000	1
Worker Commuter Vehicles	0.003	0.39	0.03	0.000	0.01	0.002	48
Total Annual Emissions	**0.003**	**0.39**	**0.03**	**0.000**	**0.01**	**0.002**	**49**

CO = carbon monoxide; CO_2e = carbon dioxide equivalent; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.

Source: Derived from INL 2020c.

Radiological Releases

The VTR would require approximately 400 kilograms of plutonium each year, based on the need to replace 45 driver fuel assemblies per year. Depending upon the source of plutonium used as feed material for this process, the plutonium could contain varying quantities of impurities (especially americium-241). A representative estimate of the impurity content for the class of fuel containing the highest impurities was used to develop these estimate. Radiological releases were estimated assuming the feedstock preparation facility would process up to 580 kilograms of plutonium each year. This includes the processing of plutonium from driver fuel fabrication material (in a recycle of material unfit for use as VTR fuel) and plutonium that would be retained within wastes generated during feedstock preparation and fuel fabrication. The estimated annual release activity per isotope is presented in **Table B–34**.

Table B–34. Idaho National Laboratory Feedstock Preparation Facility Operational Annual Radiological Releases

Isotope	Release (curies)	Isotope	Release (curies)
Plutonium-238	9.5×10^{-6}	Uranium-232	5.8×10^{-12}
Plutonium-239	9.6×10^{-6}	Uranium-234	1.7×10^{-9}
Plutonium-240	1.4×10^{-5}	Uranium-235	1.5×10^{-11}
Plutonium-241	2.0×10^{-4}	Uranium-236	2.2×10^{-10}
Plutonium-242	2.2×10^{-8}	Uranium-238	4.39×10^{-11}
Americium-241	6.6×10^{-4}		

Note: Releases are based on processing 580 kilograms of plutonium and 460 kilograms of uranium each year.
Source: Adapted from SRNS 2020.

The HEPA-filtered releases of radioactivity to the environment would be through the existing FCF stack. The combined flow rate would be about 34,900 cubic feet per minute at ambient temperatures. The release would be through a 60-inch diameter stack at an elevation of about 200 feet.

Waste Generation

Annual waste generation rates, based on the steady-state production of about 45 driver fuel assemblies per year are provided in **Table B–35**. Estimated waste quantities for production (feedstock preparation and fuel fabrication) have been developed without considering any potential reduction in wastes that would result from the performance of both processes. In particular primary transuranic waste would not be doubled if both feedstock preparation and fuel fabrication were to be required. Estimated waste also may vary with the quality of the plutonium feedstock. The quantities listed here are expected to be representative of the waste generated during feedstock preparation.

Table B–35. Idaho National Laboratory Annual Feedstock Preparation Facility Operational Wastes

Waste Type	*Volume (cubic meters)*
Low-level radioactive waste	170
Mixed low-level radioactive waste [a]	2
Secondary transuranic	32
Mixed transuranic [a]	10
Primary transuranic	170
Hazardous – solid	1
Hazardous – liquid	1
Nonhazardous – solid	17
Nonhazardous – liquid	200
Universal	0.42

[a] For low-level and secondary transuranic radioactive wastes, the mixed waste volumes are included in the total waste.
Source: SRNS 2020.

B.5.3.2 Idaho National Laboratory Site Fuel Fabrication

The INL Fuel Fabrication Option includes the use of the FMF and the ZPPR to house the equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and driver fuel assembly fabrication. VTR driver fuel fabrication is projected to require sample analysis for hundreds and potentially thousands of samples in the first few years of operation. INL proposes to use existing space fitted with new equipment in the FCF (Building MFC-765) as an analytical chemistry laboratory to support VTR fuel fabrication.

B.5.3.2.1 Fuel Fabrication Overview

Under this fuel fabrication option, the ingots of each fuel component (uranium, plutonium, and zirconium) would be delivered to the INL Fuel Fabrication Facility. At the INL Site, the Fuel Fabrication Facility would consist of existing INL facilities that would house the equipment needed to fabricate driver fuel assemblies from these ingots.

The driver fuel fabrication process at the INL Site would be located in the FMF and the ZPPR of the MFC (see Figure B–15). Both facilities are located within the MFC Protected Area, within its PIDAS. The FMF, adjacent to the ZPPR, consists of multiple workrooms and a material storage vault. The FMF has the ability to develop transuranic metallic and ceramic fuels, store these fuels, and produce and remove impurities from transuranic and enriched-uranium feedstock. The reactor and auxiliary systems portion of the ZPPR have been removed, and the facility is now used, among other tasks, for the storage, inspection, and repackaging of transuranic elements and enriched uranium. The ZPPR facility includes a workroom, cell area, and a material storage vault. As proposed, the three gloveboxes needed for fuel pin fabrication (casting furnace, demolding, and pin loading) and two additional gloveboxes for slug inspection and scrap recovery would be located in the south workroom of the FMF, where the existing Neptunium Repackaging-Transuranic Breakout Glovebox train is currently located. An existing uranium glovebox in this room would be removed. Two production lines are proposed (see **Figure B–28**). An existing glovebox train would be converted for use as one scrap recovery glovebox. The remaining casting gloveboxes, demolding gloveboxes, the train 2 scrap recovery glovebox, the slug quality assurance glovebox, and the pin loading glovebox shown in the figure would all be new equipment. Space in the MFC Special Nuclear Materials Storage Vaults would be prepared for material storage of:

- Plutonium feedstock,
- Fuel slugs,
- Fuel pins,
- Driver fuel assemblies, and
- Scrap and waste storage.

Space for lag storage of casting scrap, and assembled fuel pins pending transfer to ZPPR, would be made available in the FMF vault.

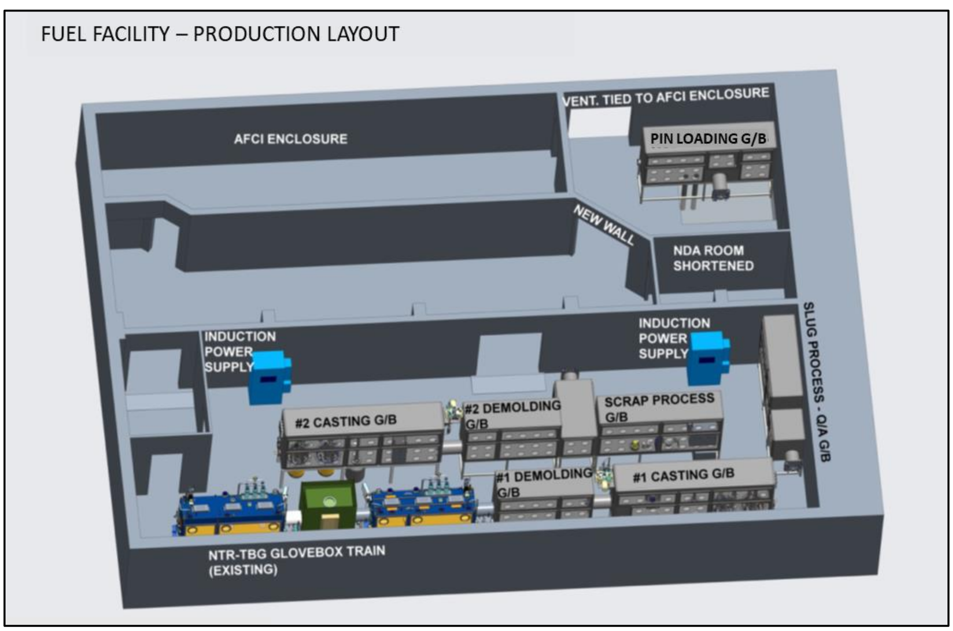


Figure B–28. Fuel Manufacturing Facility Fuel Pin Fabrication Equipment Arrangement

Upon completion of the fuel pin fabrication, fuel pins would be transferred to a storage vault or directly to the ZPPR reactor cell using a horizontal transport cask. Assembly of the fuel assembly, including bonding of the sodium to the fuel, would occur in the ZPPR reactor cell. New equipment would be installed to perform the following functions for assembly fabrication:

- Sodium bonding would be performed in a settling furnace,
- Fuel pins would be wrapped in an element (fuel pin) wire wrap station,
- Pin inspection would be performed using a profilometer and eddy current testing,
- Assembly fabrication would be performed in a vertical assembly device, and
- Assembly inspection would be performed in a vertical profilometer.

Additionally, temporary fuel pin storage racks, also located in the ZPPR reactor cell, would be required. Driver fuel assemblies could be stored in the ZPPR vault; this would require preparation of storage space, including installation of storage racks. The initial design objective for assembly storage would be sufficient capacity for 100 fresh assemblies, to ensure adequate supply for VTR operation, including the initial core load of 66 assemblies and most of the first year's reload fuel.

Driver fuel fabrication is projected to require sample analysis for hundreds and potentially thousands of samples in the first few years of operation. This workload, estimated as the analysis of 216 samples per week, and the required additional workspace would potentially overburden existing capabilities at the INL Analytical Laboratory (Buildings MFC-752). Additionally the plutonium content of samples would increase the radionuclide inventory of the Analytical Laboratory beyond the Hazard Category 3 limits currently in place. A revised safety analysis would be required to raise the facility to Hazard Category 2, before VTR fuel sampling could be done in the facility. This change would be potentially disruptive to current activities.

To minimize disruption to current activities, INL proposes to use existing space fitted with new equipment in the FCF (Building MFC-765) as an analytical chemistry laboratory to support fuel fabrication. Because the FCF is a Hazard Category-2 nuclear facility, the additional radionuclide inventory can be accommodated within the current hazard classification. **Table B–36** presents a list of equipment that would be needed to outfit the room.

Table B–36. List of Analytical Instrumentation Needed to Support Versatile Test Reactor Fuel Production

Equipment and instrumentation	*Purpose*
Class A TRU Glovebox	Manipulation of fuel samples (dissolution, dilution, disposition)
High Purity Germanium Detector System	Gamma spectrometry
Inductively Coupled Plasma-Optical Emission Spectrometer	Measurement of iron, cobalt, copper, nickel, beryllium, and other elements per fuel specifications
Ion Conductivity Probe	Measurement of chlorine and other accessible elements
Carbon, nitrogen, oxygen, and hydrogen analyzers	Light element analysis per fuel specifications
Multi-Collector – Inductively Coupled Plasma – Mass Spectrometer	High-precision measurements of uranium and plutonium isotopics (also possibly americium)
Nonradiological Fume Hood	Manipulation of nonradiological chemical reagents
Quadrupole Inductively Coupled Plasma – Mass Spectrometer [a]	Quantification of impurities per fuel specifications
Radiological Fume Hood	Preparation of dilutions and other manipulations

TRU = transuranic.
[a] Two instruments are recommended for high sample throughput and out-of-service contingency.

Initially, process qualification, development of a statistical understanding of the U-20Pu-10Zr as-cast fuel slug characteristics, and understanding phenomena such as elemental segregation during casting would require a large number of samples. The number of analytical tests would decrease as the fuel fabrication process matured.

B.5.3.2.2 Environmental Resources – Construction

Metallic feed stock would be delivered to the FMF and no new facilities would be constructed at the INL Site. The only construction activities would be the build-out of the equipment locations in the FMF, ZPPR and FCF. Construction is assumed to require 2 years.

Resource Requirements

Table B–37 presents a summary of the key resources committed to the construction of a fuel fabrication facility. In addition to the materials identified in this table, materials used in the construction of the gloveboxes include stainless steel for structural supports, glass for glovebox windows, piping for inlet, exhaust and other gas lines, electrical cable, and conduit for power and instrument lines. Primary gases used in the gloveboxes include argon as an atmosphere and hydrogen as a mechanism to remove oxygen from the glovebox atmosphere.

Nonradiological Releases

Construction of the fuel fabrication facility and feedstock preparation facility would generate similar nonradiological emissions. The annual emissions associated with fuel fabrication facility construction would be the same as those presented in Table B–31.

Waste Generation

Wastes associated with fuel fabrication construction activities would be comprised of three main types: obsolete or replaced equipment, radiologically contaminated construction wastes, and cleaning supplies and clean wastes. These are anticipated to be minimal and consistent with current facility operations and existing NEPA documentation.

Table B–37. Idaho National Laboratory Fuel Fabrication Facility Construction Resource Requirements

Resource	*Units*	*Value*	
		Annual Average (peak)	*Total* [a]
For Modifications to Existing Facilities			
Staff	FTE	6 [a] (18 [b])	18
Electricity	kWh	Minimal [c]	Minimal
Diesel Fuel			
Forklift Fuel [d]	gallons	--	32
Mobile Crane Diesel [e]	gallons	--	120
Total Diesel	gallons	--	150
Water			
Potable	gallons	75,000	230,000
Construction Area Cleaning	gallons	1,700 (2,500)	5,000
Total	gallons	77,000	230,000
Propane, Butane		Minimal	Minimal
Gas (acetylene, oxygen)		Minimal	Minimal

FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] Construction duration of 3 years is assumed.

[b] Value represents peak number of workers at one time, not FTE.

[c] Electrical use is limited to hand held or cordless hand tools and occasional welding.

[d] Values assume 40 hours of operation and fuel consumption of 0.8 gallons per hour of operation.

[e] Values assume 30 hours of operation and fuel consumption of 4 gallons per hour of operation.

Source: INL 2020c.

B.5.3.2.3 Environmental Resources – Operations

The fuel fabrication facility would produce up to 19,530 usable fuel slugs per year when each fuel pin contains two fuel slugs, sufficient to supply up to 45 fresh driver fuel assemblies per year. A portion of the fuel slugs produced would not be expected to meet VTR fuel requirements. Most of the unusable fuel slugs could be processed in the feedstock preparation facility and would be recast into fuel slugs. However, some of the material would be expected to be captured in one of the fuel fabrication waste streams.

Resource Requirements

Key annual resource commitments for the operation of the fuel fabrication facility are provided in **Table B–38**. Only chemicals used in quantities of over 1,000 pounds are shown in the table. Other chemicals and gases would be used in smaller quantities (INL 2020d).

Table B–38. Idaho National Laboratory Fuel Fabrication Facility Annual Operational Resource Requirements

Resource	*Units*	*Value*
Staff	FTE	70
Electricity	MWh	8,300-13,300 [a]
Water		
Potable	gallons	880,000
Cleaning	gallons	1,000
Chemicals		
Alcohol	pounds	1,900
Nitric Acid	pounds	1,400
Gas		
Argon, compressed	standard cubic feet	30,000
Quartz	kilograms	3,000
Ytrria	kilograms	9
Zirconia Mold Wash	kilograms	90
Graphite	kilograms	500

FTE = full-time equivalent (person); kWh = kilowatt-hour.
[a] High and low values.
Source: INL 2020c; SRNS 2020.

Nonradiological Releases

Operation of the fuel fabrication facility and feedstock preparation facility would generate similar nonradiological emissions. The annual emissions associated with fuel fabrication facility operation would be the same as those presented in Table B–33.

Radiological Releases

Radiological releases were estimated assuming the fuel fabrication facility would process about 2,500 kilograms of uranium and plutonium. This quantity includes the material needed for the fuel product and some material that would be waste from fuel fabrication. The estimated annual release activity per isotope is presented in **Table B–39**. These releases assume the use of plutonium metal that either has been prepared as described in Section B.5.2.1, lowering any impurity content of the fuel to meet the VTR fuel quality criteria, or is from feedstock material that meets the VTR fuel quality criteria.

Table B–39. Idaho National Laboratory Fuel Fabrication Facility Operational Annual Radiological Releases

Isotope	*Release (curies)*	*Isotope*	*Release (curies)*
Americium-241	3.3×10^{-4}	Uranium-232	7.3×10^{-12}
Plutonium-238	2.3×10^{-6}	Uranium-234	2.2×10^{-9}
Plutonium-239	3.7×10^{-6}	Uranium-235	1.9×10^{-11}
Plutonium-240	2.4×10^{-6}	Uranium-236	2.8×10^{-10}
Plutonium-241	5.7×10^{-5}	Uranium-238	5.4×10^{-11}
Plutonium-242	1.7×10^{-9}		

Note: Releases are based on processing 550 kilograms of plutonium and 1,900 kilograms of uranium each year.
Source: Adapted from SRNS 2020.

The HEPA-filtered releases of radioactivity to the environment would be through the existing FMF stack. The combined flow rate would be about 6,400 cubic feet per minute at 64 °F. The release would be through a 36-inch diameter stack at an elevation of about 46 feet.

Waste Generation

Annual waste generation rates, based on the production of about 45 driver fuel assemblies per year are provided in **Table B–40**. The rates shown in the table are for the fabrication of fuel directly from feedstocks; feedstocks for which no feedstock preparation would be required. These feedstocks would contain impurities at levels below the acceptable limits for the VTR fuel. Should feedstock preparation be required, the transuranic wastes generated from fuel fabrication would be much less than the values shown in Table B-40. Other wastes would be generated in quantities similar to those shown.

Table B–40. Idaho National Laboratory Fuel Fabrication Facility Annual Operational Wastes

Waste	*Volume (cubic meters)*
Low-level radioactive	170
Mixed low-level radioactive [a]	2
Secondary transuranic	32
Secondary mixed transuranic [a]	10
Primary transuranic	170
Hazardous – solid	1
Hazardous – liquid	1
Nonhazardous – solid	17
Nonhazardous – liquid	200
Universal	0.42

[a] For low-level and secondary transuranic radioactive wastes, the mixed waste volumes are included in the total waste volume.
Source: SRNS 2020.

B.5.4 Savannah River Site Reactor Fuel Production Options

Either or both feedstock preparation and driver fuel fabrication could be located at SRS. Each option is described independently in the following sections. The equipment required for either process could not be used for the other. However, there could be some benefit, in reduced resource use, in locating both options at the same site. In particular, construction resource use for both options may be less than the sum of resource use for the two options.

Reactor fuel production capabilities could be installed in either the K Area Complex or the similar L Area Complex. The reactor buildings in K Area and L Area are of the same design, and like the K-Reactor Building, the nuclear fuel and equipment needed for reactor operations have been removed from the

L-Reactor Building. This EIS specifically evaluates the potential environmental impacts of using the K Area Complex in support of the VTR project, but the impacts would be similar if the L Area Complex were used. The reactor buildings are only 2.5 miles apart and each is within a PIDAS. At either location, activities would largely occur indoors with small, previously disturbed locations outside being used for construction laydown areas or for the construction of HVAC and entry control structures. At L Area, the option exists to use either the minus-20- and minus-40-foot levels or the ground floor level for reactor fuel production. A comparative analysis shows that the offsite impacts from radiological releases would be within 3 percent of each other, with those from L Area being slightly lower.

The description that follows assumes installation of reactor fuel production capabilities at K Area. A notional equipment configuration was developed to assess the capability to house the fuel production equipment within the identified structures. But, the equipment layout that would be used has not been determined and would be finalized during the detailed design of the fuel production facility.

B.5.4.1 Feedstock Preparation

B.5.4.1.1 Savannah River Site Feedstock Preparation Overview

At SRS, this capability would be located adjacent to the location for the driver fuel fabrication capability, in the K-Reactor Building (105-K) or the 108-K buildings in the K Area Complex, mostly at the minus-20-foot level (20 feet below grade).[23] About 10,000 square feet of space would be required for feedstock preparation in either location**,** The identified area would be suitable for pretreatment operations like molten salt removal of the americium from plutonium (polishing), electrorefining, and direct oxide reduction to convert fuel compounds (e.g., fuel oxides) into their metallic form.

As discussed for feedstock preparation at the INL Site, a design of the equipment for the feedstock preparation process has not been developed. A conceptual layout for the aqueous process, using the same glovebox lines as described for feedstock preparation at INL, would require the largest amount of space of the three processes being considered. (This is one possible layout other layouts are being considered.) This process fits within the available space at the K-Reactor Building, even if the fuel fabrication process is collocated within this structure.[24] To accommodate the feedstock preparation equipment, facility modifications would be required, including the addition of a new 8,000 square foot structure to house an upgraded HVAC system. This structure could be contained within one of the 108-K buildings, placed on top of one of the buildings or located adjacent to the structures on less than 3 acres of previously disturbed land within the K-105 Reactor Building security area, depending on whether one or both of feedstock preparation and fuel fabrication were to be located at SRS.

Most of the aqueous process equipment would be located at the minus-20-foot level; the plutonium dioxide to plutonium metal conversion equipment (the pyrochemical cell) would be located at the minus-40-foot level.

Breakdowns for the arrangement of the gloveboxes for the pyrochemical process and for the combined pyrochemical/aqueous process have not been developed.

B.5.4.1.2 Environmental Resources – Construction

Resource Requirements

Key annual resource commitments for the modifications in the K-Reactor Building to enable its use as the feedstock preparation facility are provided in **Table B–41**. In addition to the materials identified in this

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[23] The location of the 108-K Buildings relative to the 105-5 Reactor Building is shown in figures provided in the discussion of fuel fabrication at SRS, Section B.5.4.2.

[24] The layouts for feedstock preparation and driver fuel fabrication depicted in this appendix were developed independently, neither considers the location of the other activity. The layouts would differ if both activities were to be located at SRS. However, there is sufficient space that both activities could be located within the K-Reactor Building structures.

table, materials used in the construction of the gloveboxes include stainless steel for structural supports, glass for glovebox windows, piping for inlet, exhaust and other gas lines, electrical cable, and conduit for power and instrument lines. Primary gases used in the gloveboxes include argon as an atmosphere and hydrogen as a mechanism to remove oxygen from the glovebox atmosphere.

Table B–41. Savannah River Site Feedstock Preparation Facility Construction Resource Requirements

		Value	
Resource	***Units***	***Annual***	***Total***
Staff	FTE	120	360
Electricity	MWh	minimal	minimal
Diesel	gallon	1,500	4,500
Gasoline	gallon	2,500	7,500
Water Supply			
Potable	gallons (thousands)	1,000	3,000
Construction	gallons (thousands)	2,000	6,000
Total	gallons (thousands)	3,000	9,000
Waste Water Treatment	gallons (thousands)	1,000	3,000
Cement	tons	--	800
Steel (tons)	tons	--	600
Conduit	linear feet	--	74,000
Cable Tray	linear feet	--	2,400
Power/Control Cable	linear feet	--	83,000
Piping	linear feet	--	14,000
Facilities	square feet	--	40,000
Ductwork	pounds	--	51,000
Formwork	square feet	--	36,000
Sand, Cone, Aggregate	cubic yards	--	880
Gravel, Crushed Stone, etc.	cubic yards	--	660
Soil – Fill Material	cubic yards	--	3,700
Gases			
Acetylene	cubic meters	--	53
Oxygen	cubic meters	--	240
CO_2/Argon	cubic meters	--	80
Nitrogen	cubic meters	--	160
Argon	cubic meters	--	1,300
Helium	cubic meters	--	33
Other			
Epoxy Floor Covering	square feet	--	48,000
Macropoxy (concrete wall covering)	square feet	--	17,000
Enamel Paint	square feet	--	50,000
Intumescent Coating (steel deck coating)	square feet	--	8,300

CO_2 = carbon dioxide; FTE = full-time equivalent (person); MWh = megawatt-hour.
Source: SRNS 2020.

Nonradiological Releases

Nonradiological releases are associated with the operation of the forklifts, construction vehicles, concrete mixers, cranes and other smaller equipment (i.e., the burning of diesel fuel and worker personal vehicle use). The total construction related emissions associated with these items are provided in **Table B–42**.

Table B–42. Savannah River Site Feedstock Preparation Facility Construction Nonradiological Emissions

Facility	Emissions (tons)							Combined HAPs [a]	CO_2e (metric tons)
	VOC	CO	NOx	SO2	PM_{10}	$PM_{2.5}$	CO_2		
Onsite Emissions from On-road Sources	0.02	1.62	0.20	0.002	0.05	0.01	221	0.005	201
Onsite Emissions from Nonroad Sources	0.04	0.85	0.24	0.001	0.02	0.01	63	0.01	57
Offsite Emissions from On-road Sources	0.05	3.24	0.44	0.003	0.10	0.02	458	0.01	416
Total 2025 Emissions	**0.11**	**5.71**	**0.88**	**0.01**	**0.17**	**0.05**	**742**	**0.02**	**674**

CO = carbon monoxide; CO_2= carbon dioxide; CO_2e = carbon dioxide equivalent; HAPs = hazardous air pollutants; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.

[a] Combined HAPs = 15/3 percent of combustive VOC/PM emissions for on-road and nonroad sources and 1/3 percent for slash burning (California Air Resources Board 2018).

Source: Adapted from SRNS 2020.

Waste Generation

Areas within the K-Reactor Building structures would be modified to make room for the feedstock preparation equipment. This would involve the removal of existing equipment and some structural modifications. Estimates for waste generation from this modification effort are shown in **Table B–43**.

Table B–43. Savannah River Site Feedstock Fabrication Facility Construction Wastes

Waste Type	Units	Value
Toxic Substance Control Act Waste	cubic meters	28
Universal Waste	cubic meters	7.5
Nonhazardous Waste		
From Construction Activities	gallons/cubic meters	90,000/340
Equipment Removed	metric tons/cubic meters	100/5,000
Low-level Radioactive Waste	cubic meters	380

Source: SRNS 2020.

B.5.4.1.3 Environmental Resources – Operations

Resource Requirements

Key annual resource commitments for the operation of the feedstock preparation facility are provided in **Table B–44**. Resource requirements listed do not include the fuel feed material (uranium, plutonium, and zirconium.)

Table B–44. Savannah River Site Annual Feedstock Preparation Facility Resource Requirements

Resource	Units	Value	
		Annual	Peak
Staff	FTE	300	--
Electricity	MWh	6,700	--
Natural Gas	cubic feet	0	--
Heating Oil	gallon	0	--

Resource	Units	Value	
		Annual	Peak
Diesel (Centerra) [a]	gallon	1,500	--
Diesel (Operations)[a]	gallon	2,000	--
Water			
Potable [b]	gallons (thousands)	1,400	--
Process and Waste Treatment [c]	gallons (thousands)	50	--
Total	Gallons (thousands)	1,500	--
Sanitary Waste Water Treatment	gallons (thousands)	1,400	--
Nitric Acid	cubic meters	88	130
Caustic	kilograms	43	64
Potassium Flouride	kilograms	600	900
Aluminum Nitrate Nonahydrate	kilograms	300	450
Hydroxylamine Nitrate	kilograms	125	190
Polymer Resin	kilograms	40	60
Oxalic Acid	kilograms	1,400	2,100
Ascorbic Acid	kilograms	100	150
Argon	cubic meters	900,000	--
Helium	cubic meters	45,000	--
Nitrogen	cubic meters	50,000	--
Oxygen	cubic meters	5,000	--
Propane	bottles/gallons	100/470	150/700

FTE = full-time equivalent (person); MWh = megawatt-hour.

[a] Diesel fuel for one additional security vehicle (Centerra) and an additional diesel generator (Operations).

[b] Water use provided as gallons per minute, converted to annual assuming 8 hour work days, 5 days a week, and 50 weeks per year.

[c] Water requirements are for the aqueous processing of feedstock material. Other processes would require less.

Source: SRNS 2020.

Nonradiological Releases

Nonradiological emissions for feedstock preparation would be associated with the transport of material to the K-Reactor Building and worker vehicles. Emission data is presented in **Table B–45**.

Table B–45. Savannah River Site Feedstock Preparation Facility Annual Operational Nonradiological Emissions

Facility	Emissions (tons)							Combined HAPs [a]	CO_2e (mt)
	VOC	CO	NOx	SO2	PM_{10}	$PM_{2.5}$	CO_2		
Onsite Emissions from On-road Sources	0.02	0.23	0.03	0.0003	0.001	0.001	46	0.002	42
Onsite Emissions from Nonroad Sources	0.002	0.01	0.03	0.0001	0.004	0.001	16	0.0004	15
Offsite Emissions from On-road Sources	0.07	7.58	0.39	0.007	0.19	0.04	1,000	0.02	909
Total 2025 Emissions	**0.08**	**7.82**	**0.45**	**0.01**	**0.20**	**0.04**	**1,062**	**0.02**	**965**

CO = carbon monoxide; CO_2 = carbon dioxide; CO_2e = carbon dioxide equivalent; HAPs = hazardous air pollutants; MT = metric tons; NO_x = nitrogen oxides; $PM_{2.5}$ = particulate matter less than 2.5 microns in diameter; PM_{10} = particulate matter less than 10 microns in diameter; SO_2 = sulfur dioxide; VOC = volatile organic compound.

[a] Combined HAPs = 15/3 percent of combustive VOC/PM emissions for on-road and nonroad sources and 1/3 percent for slash burning (California Air Resources Board 2018).

Source: Adapted from SRNS 2020.

Radiological Releases

Radiological releases for feedstock preparation at SRS would be the same as described for that activity at the INL Site. See Table B–34 in Section B.5.3.1.3.

HEPA-filtered releases of radioactivity to the environment would be through a stack installed for the driver fuel fabrication facility. The combined flow rate would be about 18,000 cubic feet per minute at an elevation of about 124 feet (SRNS 2020).

Waste Generation

Annual waste generation rates, based on the steady state production of about 45 driver fuel assemblies per year are provided in **Table B–46**. Estimated waste quantities for production (feedstock preparation and fuel fabrication) have been developed without considering any potential reduction in wastes that would result from the performance of both processes. In particular primary transuranic waste would not be doubled if both feedstock preparation and fuel fabrication were to be required. Estimated waste also may vary with the quality of the plutonium feedstock. The quantities listed here are expected to be representative of the waste generated during feedstock preparation.

Table B–46. Savannah River Site Annual Feedstock Preparation Facility Operational Wastes

Waste	*Volume (cubic meters)*
Low-level radioactive	170
Mixed low-level radioactive [a]	2
Secondary transuranic	32
Secondary mixed transuranic [a]	10
Primary transuranic	170
Hazardous – solid	1
Hazardous – liquid	1
Nonhazardous – solid	17
Nonhazardous – liquid	200
Universal	0.42

[a] For low-level and secondary transuranic radioactive wastes, the mixed waste volumes are included in the total waste volume.

Source: SRNS 2020.

B.5.4.2 Savannah River Site Fuel Fabrication

Under the SRS fuel fabrication option, driver fuel fabrication would be performed in the K-Reactor Building (105-K) in the K Area Complex. All equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and driver fuel assembly fabrication would be located on two below-ground levels within the building.

Under the SRS Fuel Processing and Conversion Option, this capability would be located adjacent to the location for the fuel fabrication capability, in the K-Reactor Building (105-K) in the K Area Complex. All of the equipment for fuel processing and conversion would be newly constructed.

B.5.4.2.1 Savannah River Site Fuel Fabrication Facilities Overview

At SRS, the fuel fabrication facility would be located on the minus-20- and minus-40-foot levels (20 and 40 feet below grade) of the K-Reactor Building, Building 105-K. The facility is located within a Protected Area and includes a Material Access Area with the physical security infrastructure that satisfies requirements for handling and storage of Category I special nuclear material. This area is currently used to store drums of heavy water and pumps (SRNS 2020).

Approximately 17,000 square feet and 22,600 square feet of space would be made available at the minus-40- and minus-20-levels, respectively. Material and equipment to be removed are expected to be radiologically clean. A portion of the space at the minus-20-foot level has a high bay area that would allow for the vertical assembly of driver fuel assemblies. The identified area would be suitable for the fuel fabrication glovebox processes being designed at the INL Site. The facility could support feed material purification, ingot manufacturing, and/or the fabrication of fuel from ingots. New equipment would be provided for fuel slug casting, slug trimming and inspection, fuel rod loading and inspection, fuel bundle assembly and packaging, and waste handling. Other infrastructure to be supplied would include material storage areas (including an area to store fully assembled driver fuel assemblies), special nuclear material measurement equipment, analytical support, and other infrastructure services such as glovebox and room ventilation and electrical distribution (SRNS 2020).

The facility design would be based on the conceptual design developed for the fuel fabrication facility at the INL Site. While a specific layout has not been established, the following is a notional layout to convey the type and size of equipment and the representative space needed for operations. Structural modifications to the facility would be required to accommodate fuel fabrication. At SRS, fuel ingots would be received at ground level and transferred via an existing, but to be upgraded, elevator to a small lag vault located in one of the motor rooms at the minus-40-foot level. Two process lines for alloy mixing, slug casting, and pin assembly would be located at the -40-foot level within the existing Cross-over Area and the Process and Pump Rooms (see **Figure B–29**). Additionally, equipment for fuel pin non-destructive analysis, waste processing,[25] and analytical support would be located at this level. Assembled fuel pins would be transferred to a high bay area at the minus-20-foot level for preparation and assembly into complete driver fuel assemblies (see **Figure B–30**). (Alternately final assembly could be done in the K-108 Building or at the -40-foot level (provided some heat exchangers were removed from this area). Since SRS is not a proposed site for the VTR, completed assemblies would be loaded into a shielded transfer cask at the minus-20-foot Assembly Area Basement. The shielded transfer cask would be raised up out of the Assembly Area Basement and then loaded into a shipping container for shipment (SRNS 2020).

Although the VTR modifications have not been designed, based on similar K Area upgrade projects, the space needed for support facilities for the needed HVAC, fire suppression, etc. are expected to be substantial. At least one and possibly two, of the adjacent 108-K buildings could be needed for these support operations. The addition of a new 8,000-square foot structure to house an upgraded HVAC system would be required. This structure could be contained within one of the 108-K buildings, placed on top of one of the buildings or placed on a previously disturbed area (less than 3 acres) within the K-Reactor Building security area, depending on whether one or both of feedstock preparation and fuel fabrication were to be located at SRS. (This is the same HVAC capability described under SRS feedstock preparation.) Additional modifications could include construction of a new facility stack (the preconceptual design includes a 124-foot stack) for the VTR fuel production activities and construction of a new entry control structure.

Should SRS be selected as the site for fuel fabrication, a demonstration facility would still be built at INL. The demonstration facility would be located in the existing INL FMF at the same location as the proposed production facility. It would consist of a single line of furnace, demolding, and pin-assembly gloveboxes. Scrap processing, waste handling, and fuel slug quality assurance gloveboxes would also be installed.

[25] Scrap unsuitable for reuse would be transferred to the oxidation/blenddown line where the alloy would be oxidized and blended down to meet the Waste Isolation Pilot Plant facility disposal and safeguards and security criteria.

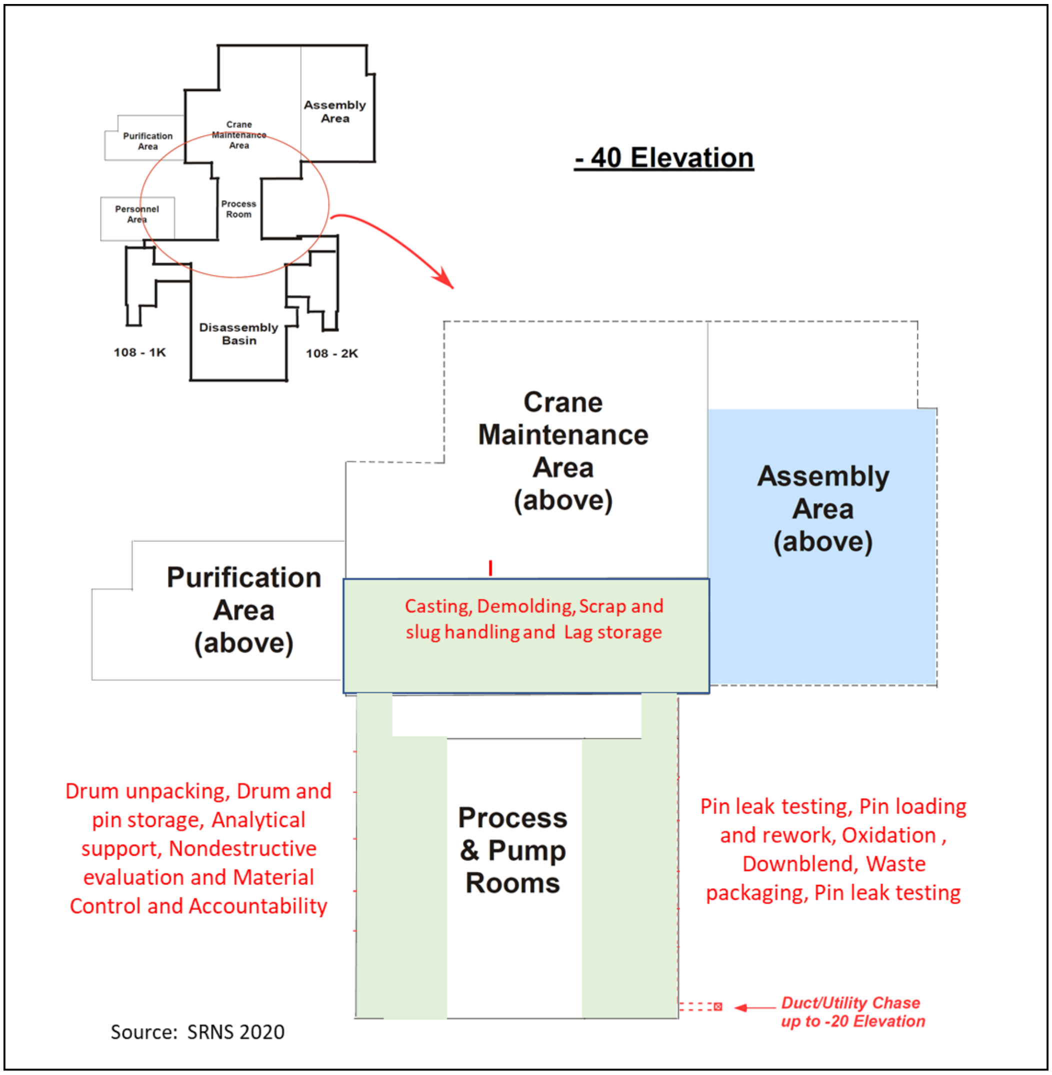

Figure B–29. Savannah River Site Proposed Fuel Fabrication Facility Minus-40-Foot Level of K-Reactor Building

B.5.4.2.2 Environmental Resources – Construction

Metallic feed stock would be delivered to the K-Reactor Building (K-105), and no new facilities would be constructed at SRS. The only construction activities would be the build-out of the equipment locations within K-Reactor Building and the removal of existing equipment. Construction is assumed to require 3 years. A few (three) small, previously disturbed areas, totaling less than an acre) within the K-105 security fencing have been identified as potential construction laydown areas.

Resource Requirements

Table B–47 provides a summary of the key resources committed to the modification of the K-Reactor Building to enable its use as a fuel fabrication facility. In addition to the materials identified in this table, materials used in the construction of the gloveboxes include stainless steel for structural supports, glass for glovebox windows, piping for inlet, exhaust and other gas lines, electrical cable, and conduit for power and instrument lines. Primary gases used in the gloveboxes include argon as an atmosphere and hydrogen as a mechanism to remove oxygen from the glovebox atmosphere.

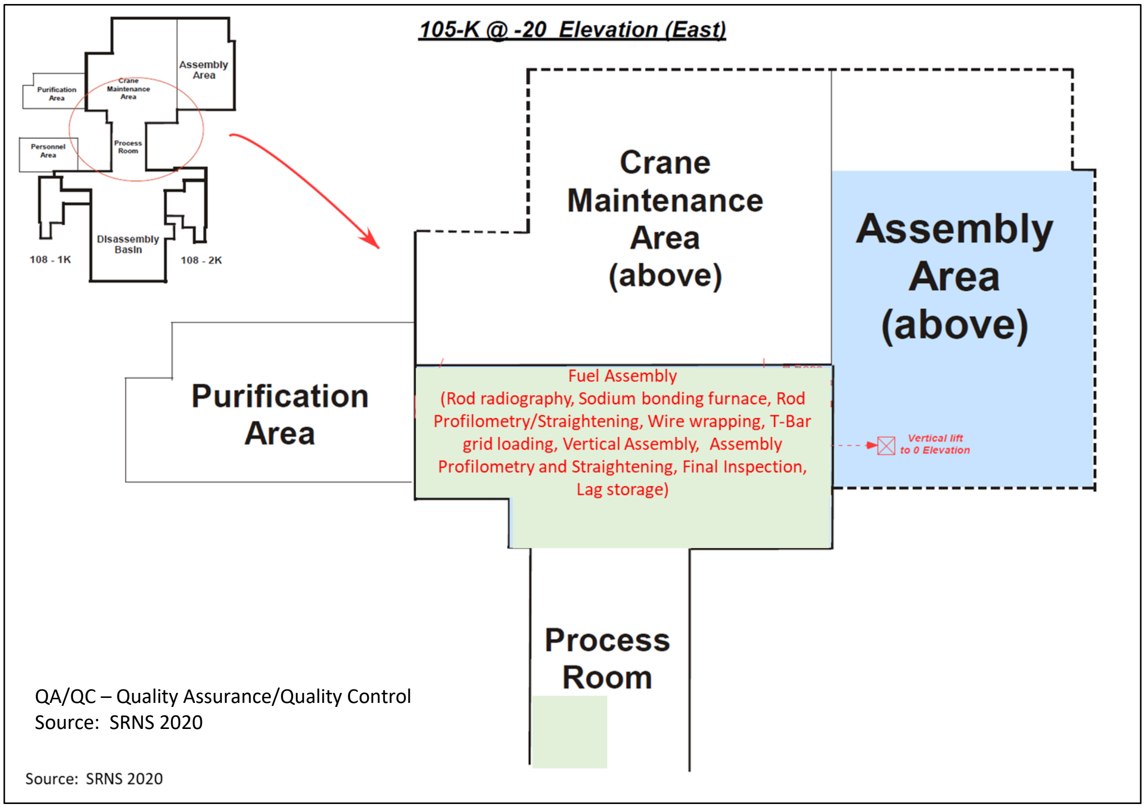

Figure B–30. Savannah River Site Proposed Fuel Fabrication Facility Minus-20-Foot Level of K-Reactor Building

Table B–47. Savannah River Site Fuel Fabrication Facility Construction Resource Requirements

Resource	*Units*	*Value*	
		Annual Average	*Total* [a]
For Modifications to Existing Facilities			
Staff	FTE	120	360
Electricity	kWh	Minimal	Minimal
Diesel Fuel	gallons	1,500	4,500
Gasoline	gallons	2,500	7,500
Water [b]			
Potable	gallons (thousands)	1,000	3,000
Construction	gallons (thousands)	2,000	6,000
Total	gallons (thousands)	3,000	9,000
Construction Materials			
Cement	tons	--	800
Steel (tons)	tons	--	600
Conduit	linear feet	--	74,000
Cable Tray	linear feet	--	2,500
Power/Control Cable	linear feet	--	83,000
Piping	linear feet	--	14,000
Facilities	square feet	--	40,000
Ductwork	pounds	--	51,000
Formwork	square feet	--	36,000
Sand, Cone, Aggregate	cubic yards	--	880
Gravel, Crushed Stone, etc.	cubic yards	--	660
Soil – Fill Material	cubic yards	--	3,600

Resource	Units	Value	
		Annual Average	Total [a]
Gases			
Acetylene	cubic meters	--	53
Oxygen	cubic meters	--	240
CO_2/Argon	cubic meters	--	80
Nitrogen	cubic meters	--	160
Argon	cubic meters	--	1,300
Helium	cubic meters	--	33
Other			
Epoxy Floor Covering	square feet	--	48,000
Macropoxy (concrete wall covering)	square feet	--	117,000
Enamel Paint	square feet	--	50,000
Intumescent Coating (steel deck coating)	square feet	--	8,300

CO_2 = carbon dioxide; FTE = full-time equivalent (person); kWh = kilowatt-hour.

[a] A 3-year construction period.

[b] Water use provided as gallons per minute, converted to annual assuming 10-hour work days, 5 days a week, and 50 weeks per year and is based on the peak construction workforce.

Source: SRNS 2020.

Nonradiological Releases

Nonradiological releases are associated with the operation of the forklifts, construction vehicles, concrete mixers, cranes and other smaller equipment (i.e., the burning of diesel fuel and worker personal vehicle use). The annual emissions associated with these items would be about the same as those associated with feedstock preparation (see Table B–42).

Waste Generation

Table B–48 provides waste generation information for construction of the fuel fabrication facility. Wastes associated with construction activities would be comprised of three main types: obsolete or replaced equipment, radiologically contaminated construction wastes, and cleaning supplies and clean wastes.

Table B–48. Savannah River Site Fuel Fabrication Facility Construction Wastes

Waste Type	Units	Value
Toxic Substance Control Act Waste	cubic meters	28
Universal Waste	cubic meters	7.5
Nonhazardous Waste		
From Construction Activities	gallons/cubic meters	90,000/340
Equipment Removed	metric tons/cubic meters	100/5,000
Low-level Radioactive Waste	cubic meters	770

Source: SRNS 2020.

The majority of the dismantlement and removal (D&R) items to be removed from the minus-40-foot motor rooms and crossover and the minus-20-foot pipe corridors and crossover are expected to be nonradioactive. There are a few contamination areas that have the potential to generate low-level radioactive waste (LLW). Radiological control operations personnel will be involved in determining which items can be free released, which items fall under the metals moratorium, and which items may have to be treated as LLW due to unknown history. In addition, all items will require evaluations for asbestos, polychlorinated biphenyls and Resource Conservation and Recovery Act constituents prior to determining a final disposition path (SRNS 2020).

It is anticipated that asbestos will be encountered during D&R activities. An inspection will be conducted by a licensed inspector prior to initiation of D&R activities and as needed during D&R when suspect materials are encountered to properly identify asbestos-containing materials and presumed asbestos-containing materials (SRNS 2020).

Although detailed estimates of the decontamination and decommissioning waste are not available, the mass of the removed material could be as high as 100 metric tons and 5,000 cubic meters in packaged form[26] (SRNS 2020). This material would be disposed at either onsite LLW sites or onsite construction and demolition landfill disposal sites.

B.5.4.2.3 Environmental Resources – Operations

The fuel fabrication facility would produce up to 19,530 usable fuel slugs per year when each fuel pin contains two fuel slugs, sufficient to supply up to 45 fresh driver fuel assemblies per year. A portion of the fuel slugs produced would not be expected to meet VTR fuel requirements. Most of the unusable fuel slugs could be processed in the feedstock preparation facility and would be recast into fuel slugs. However, some of the material would be expected to be captured in one of the fuel fabrication waste streams.

Should SRS be selected as the site for fuel fabrication, a demonstration fuel fabrication line would be built at INL. Environmental resources associated with the operation of this demonstration line for the full duration of its operation would be bound by the resources associated with one year of operation of the INL fuel fabrication facility. These operational environmental resources are discussed in Section B.5.3.2.3.

Resource Requirements

Key annual resource commitments for the operation of the fuel fabrication facility are provided in **Table B–49**. Resource requirements listed do not include the fuel fabrication material (uranium, plutonium, zirconium, sodium, and HT-9 stainless steel)

Table B–49. Savannah River Site Annual Fuel Fabrication Facility Resource Requirements

Resource	*Units*	*Value*
Staff	FTE	300
Electricity	MWh	8,300-13,300 [a]
Diesel		
Centerra [b]	gallon	3,000
Operations [b]	gallon	4,000
Total	gallon	7,000
Water Supply [c]	gallons (thousands)	1,400
Wastewater Treatment	gallons (thousands)	1,400
Argon	cubic meters	600,000
helium	cubic meters	30,000
Nitrogen	cubic meters	30,000
Oxygen	cubic meters	30,000
Propane	bottles/gallons	100/470
Quartz	kilograms	3,000
Ytrria	kilograms	9
Zirconia Mold Wash	kilograms	90
Graphite	kilograms	500

FTE = full-time equivalent (person); MWh = megawatt-hour.
[a] High and low of estimated values.
[b] Diesel fuel for one additional security vehicle (Centerra) and an additional diesel generator (Operations).
[c] Water use provided as gallons per minute, converted to annual assuming 8-hour work days, 5 days a week, and 50 weeks per year.
Source: INL 2020c; SRNS 2020.

[26] If the heat exchangers are removed from the minus-40-foot level, an additional 18 truckloads of debris would be generated.

Nonradiological Releases

Nonradiological emissions for fuel fabrication would be associated with the transport of material to the K-Reactor Building and worker vehicles. Emission data would be similar to that for INL feedstock preparation, see Table B–45.

Radiological Releases

HEPA-filtered radiological releases would be the same as for fuel fabrication at INL. See Section B.5.3.2.3, Table B–39.

Releases of radioactivity to the environment would be through a stack installed for the VTR fuel fabrication facility or an existing stack. The combined flow rate would be about 18,000 cubic feet per minute at an elevation of about 124 feet (SRNS 2020).

Waste generation

Annual waste generation rates, based on the production of about 45 driver fuel assemblies per year are provided in **Table B–50**. The rates shown in the table are for the fabrication of fuel directly from feedstocks; feedstocks for which no feedstock preparation would be required. These feedstocks would contain impurities at levels below the acceptable limits for the VTR fuel. Should feedstock preparation be required, the transuranic wastes generated from fuel fabrication would be much less than the values shown in Table B–50. Other wastes would be generated in quantities similar to those shown.

Table B–50. Savannah River Site Annual Fuel Fabrication Facility Operational Wastes

Waste	*Volume (cubic meters)*
Low-level radioactive	170
Mixed low-level radioactive [a]	2
Secondary transuranic	32
Secondary mixed transuranic [a]	10
Primary transuranic	170
Hazardous – solid	1
Hazardous – liquid	1
Nonhazardous – solid	17
Nonhazardous – liquid	200
Universal	0.42

[a] For low-level and secondary transuranic radioactive wastes, the mixed waste volumes are included in the total waste volume.

Source: SRNS 2020.

B.6 References

Bumgardner, J., 2019, VTR Program Manager, Idaho National Laboratory, Versatile Test Reactor Overview presentation, Idaho Falls, Idaho, May 13.

California Air Resources Board, 2018, Speciation Profiles Used in ARB Modeling (available at https://ww3.arb.ca.gov/ei/speciate/speciate.htm).

Crawford, D., 2019, VTR Fuel Cycle Manager/MFC Chief Scientist, Idaho National Laboratory, VTR Fuel Cycle Update, Idaho Falls, Idaho, June 6. **OUO**

DOE (U.S. Department of Energy), 2018a, DOE Standard, Hazard Categorization of DOE Nuclear Facilities, DOE-STD-1027-2018, Washington, DC, November.

DOE (U.S. Department of Energy), 2018b, *Mission Need Statement for the Versatile Test Reactor (VTR) Project, A Major Acquisition Project*, Office of Nuclear Technology Research and Development, Office of Nuclear Energy, December.

GE Hitachi, 2019a, VTR Quarterly Integration Meeting: Conceptual Design Update Presentation, Wilmington, North Carolina, June 5.

GE Hitachi, 2019b, Nuclear - Liquid Radwaste – GEH Input, Design Input Request - Task ECK10-DIR-N100, Wilmington, North Carolina.

INL (Idaho National Laboratory), 2013, *Post Irradiation Examination Capabilities Guide* (available at https://nsuf.inl.gov/documents/INL%20PIE%20Capabilities%20Guide_web.pdf).

INL (Idaho National Laboratory), 2016, Fuel Conditioning Facility Factsheet, Idaho Falls, Idaho.

INL (Idaho National Laboratory), 2017a, Hot Fuel Examination Facility Factsheet, Post-irradiation examination of fuels and materials, Idaho Falls, Idaho.

INL (Idaho National Laboratory), 2017b, Utility Study for the Versatile Test Reactor (VTR) at MFC, EM-10300-1, Rev. 5, TEV No: 3542, Idaho Falls, Idaho.

INL (Idaho National Laboratory), 2019a, VTR Fuel Facility Plan, INL/LTD-19-54001, Idaho Falls, Idaho, May. **OUO**

INL (Idaho National Laboratory), 2019b, Conceptual Design Report for the Versatile Test Reactor (VTR), INL/LTD-19-56681, Rev. 0, Idaho Falls, Idaho, November. **OUO, Export Controlled Information**

INL (Idaho National Laboratory), 2019c, Irradiated Materials Characterization Laboratory Factsheet, Idaho Falls, Idaho.

INL (Idaho National Laboratory), 2019d, *Idaho National Laboratory Site Environmental Report Calendar Year 2018*, DOE/ID-12082(18), Environmental Surveillance, Education, and Research Program, Idaho Operations Office, Idaho Falls, Idaho, September.

INL (Idaho National Laboratory), 2020a, Analytical Laboratory Factsheet, Idaho Falls, Idaho.

INL (Idaho National Laboratory), 2020b, Materials and Fuel Complex, Production Element Chopper, Fuel Conditioning Facility (accessed January 9, 2020 at https://mfc.inl.gov/SitePages/Instruments/Fuel%20Conditioning%20Facility/Production%20element%20chopper.aspx).

INL (Idaho National Laboratory), 2020c, VTR Hazards and Impacts Information in Support of National Environmental Policy Act Data Needs, INL/EXT-19-55588, July.

INL (Idaho National Laboratory), 2020d, TEV-3976 Versatile Test Reactor (VTR Waste and Material Data for Environmental Impact Statement (EIS), Idaho Falls, Idaho.

INL (Idaho National Laboratory), 2020e, VTR Fuel Polishing, TEV No.: 4077, Rev. 0, Idaho Falls, Idaho, August 4.

LANL (Los Alamos National Laboratory), 2008, "Plutonium Processing at Los Alamos," *Actinide Research Quarterly*, 3rd Quarter 2008, LALP-08-004, Los Alamos, New Mexico.

Leidos, 2020, *Information for Construction and Operation of the Versatile Test Reactor Complex at the Oak Ridge National Laboratory*, VTR-RPT-01, Rev. 0, Reston, Virginia.

Nelson, L., 2020, Idaho National Laboratory, Personal communication (teleconference), with K. Owens, D. Outlaw, D. Gallagher, Leidos, Re: Staffing Numbers, July 6.

ORNL (Oak Ridge National Laboratory), 2014, Irradiated Material Examination and Testing Facility Factsheet, Oak Ridge, Tennessee, July.

ORNL (Oak Ridge National Laboratory), 2015, Irradiated Fuels Examination Laboratory Factsheet, Oak Ridge, Tennessee, April.

ORNL (Oak Ridge National Laboratory, 2017, Low Activation Materials Development and Analysis Laboratory (LAMDA) Factsheet, Oak Ridge, Tennessee, April.

SRNL (Savannah River National Laboratory), 2020, Conceptual Assessment of VTR Add-on Processing Capability, SRNL-TR-2020-00171, Rev. 2, Aiken, South Carolina, July 22.

SRNS (Savannah River Nuclear Solutions, LLC), 2020, Savannah River Site Data Call Response for the Versatile Test Reactor Fuel Fabrication Facility, SRNS-RP-2020-00286, Rev. 2, Aiken, South Carolina, July 22.

TerraPower, 2019, VTR Pool Reactor and Metal Fuel Overview Presentation.

Appendix C
Evaluation of Human Health Effects from Normal Operations

APPENDIX C
EVALUATION OF HUMAN HEALTH EFFECTS FROM NORMAL OPERATIONS

C.1 Introduction

This appendix presents detailed information on the potential impacts on humans associated with incident-free (normal) releases of radioactivity from the U.S. Department of Energy (DOE) facilities proposed in this *Versatile Test Reactor Environmental Impact Statement* (VTR EIS). This appendix also presents information on the calculation of worker doses that would be received as a result of performing facility modifications and operation of the Versatile Test Reactor (VTR) and associated facilities. Chapter 2 of this VTR EIS presents descriptions of the alternatives and the fuel preparation and fabrication options that contribute to the doses evaluated in this appendix. Appendix B provides descriptions of the VTR facilities: the VTR building, fuel preparation and fabrication facilities, post-irradiation examination facilities, and spent fuel treatment and temporary storage facilities. The analysis in this appendix supports the human health risk assessments described in Chapter 4, Section 4.10. Site-specific input data used in the evaluation of these human health impacts are provided or referenced, as appropriate. Resulting impacts can be compared to criteria invoked in DOE Order 458.1 for protection of the public (10 millirem per year from airborne pathways and 100 millirem per year total from all pathways) and Title 10 of the *Code of Federal Regulations* (CFR), Part 835, for protection of workers (5,000 millirem per year) at the three sites considered as alternative locations for VTR-related activities: the Idaho National Laboratory (INL) Site, Oak Ridge National Laboratory (ORNL), and Savannah River Site (SRS). Worker doses would be monitored and controlled below the regulatory limit to ensure that individual doses are less than 2,000 millirem per year and as low as reasonably achievable (ALARA).

The rest of this section provides information to aid the reader in understanding the impacts from the radiological dose assessments. The text box on the following page presents basic information about the sources, types, and nature of radiation and units of measurement. Subsequent subsections address the sources of radiation protection guidelines, radiation exposure limits applicable to DOE operations, and the assessment of health effects from exposure to radiation.

C.1.1 Radiation Protection Guides

Various organizations have issued radiation protection guides. The two organizations most directly responsible for the development of radiological requirements and exposure criteria associated with the operation of DOE facilities are DOE and the U.S. Environmental Protection Agency (EPA).

DOE. Radiological protection of the public and site workers from the operation of DOE facilities is primarily the responsibility of DOE. DOE establishes and enforces requirements for radiological protection at DOE sites in regulations and orders. Requirements for worker protection are included in “Occupational Radiation Protection Program” (10 CFR Part 835). Radiological protection of the public and environment is addressed in “Radiation Protection of the Public and the Environment” (DOE Order 458.1).

EPA. The EPA has published a series of documents under the title *Radiation Protection Guidance to Federal Agencies*. This guidance is used as a benchmark by a number of Federal agencies, including DOE, for the purpose of ensuring that regulation of public and occupational workforce exposures is protective, reflects the best available scientific information, and is carried out in a consistent manner. In addition, the EPA has established a regulatory limit of 10 millirem per year for exposure of the public to emissions from DOE facilities (40 CFR Part 61, Subpart H).

Radiation Basics

What is radiation? Radiation is energy emitted from unstable (radioactive) atoms in the form of atomic particles or electromagnetic waves. This type of radiation is also known as ionizing radiation because it can produce charged particles (ions) in matter.

What is radioactivity? Radioactivity is produced by the process of radioactive atoms trying to become stable (a process termed "decay"), the splitting of atoms (fission), and the combination of atoms (fusion). Radiation is emitted in the process. In the United States, radioactivity is commonly measured in units called curies, where 1 curie is equal to 3.7×10^{10} disintegrations (decay transformations) per second. Internationally, radioactivity is generally measured in units called becquerels, where 1 becquerel is equal to 1 disintegration per second (1 curie = 3.7×10^{10} becquerels).

What is radioactive material? Radioactive material is any material containing unstable atoms that emit radiation.

What are the four basic types of ionizing radiation?

> ***Alpha particles*** — Alpha particles consist of two protons and two neutrons. They can travel only a few centimeters in air and can be stopped easily by a sheet of paper or by the skin's surface.
>
> ***Beta particles*** — Beta particles are smaller and lighter than alpha particles and have the mass of a single electron. A high-energy beta particle can travel a few meters in the air. Beta particles can pass through a sheet of paper, but may be stopped by a thin sheet of aluminum foil or glass.
>
> ***Gamma rays*** — Gamma rays (and x-rays), unlike alpha or beta particles, are waves of pure energy. Gamma radiation is very penetrating and can travel several hundred feet in the air. Gamma radiation requires a thick wall of material such as concrete, lead, or steel to stop it.
>
> ***Neutrons*** — A neutron is an atomic particle that has about one-quarter the weight of an alpha particle. Like gamma radiation, it can easily travel several hundred feet in the air. Neutron radiation is most effectively stopped by materials with high hydrogen content, such as water or plastic.

What are the sources of radiation?

> ***Natural sources of radiation*** — Sources include cosmic radiation from the sun and outer space, natural radioactive elements in the Earth's crust, natural radioactive elements in the human body, and radon gas from the radioactive decay of uranium that is naturally present in the soil.
>
> ***Man-made sources of radiation*** — Sources include medical radiation (x-rays, medical isotopes), consumer products (TVs, luminous dial watches, smoke detectors), nuclear technology (nuclear power plants, industrial x-ray machines), and worldwide fallout from past nuclear weapons tests or accidents.

What is radiation dose? Radiation dose is the amount of energy in the form of ionizing radiation absorbed per unit mass of any material. For people, radiation dose is the amount of energy absorbed in human tissue. In the United States, radiation dose is commonly measured in units called rem; a smaller fraction of the rem is the millirem (1/1,000 of 1 rem). Internationally, radiation dose is generally measured in units called sieverts, where 1 rem = 0.01 sieverts.

Person-rem (or person-sievert) is a unit of collective radiation dose applied to populations or groups of individuals; it is the sum of the doses received by all the individuals of a specified population.

What is the average annual radiation dose from natural and man-made sources? Globally, humans are exposed constantly to radiation from the solar system and the Earth's rocks and soil. This natural radiation contributes to the natural background radiation that always surrounds us. Man-made sources of radiation also exist, including medical and dental x-rays, household smoke detectors, and materials released from nuclear and coal-fired power plants. The average individual in the United States annually receives about 625 millirem of radiation dose from all background sources, of which about half is received from natural sources such as cosmic and terrestrial radiation and radon-220 and -222 in homes. Most of the remaining radiation dose from man-made sources is received from diagnostic x-rays and nuclear medicine (NCRP 2009).

What are the effects of radiation on humans? Radiation can cause a variety of adverse health effects in humans. Health impacts of radiation exposure, whether from external or internal sources, generally are identified as somatic (i.e., affecting the exposed individual) or genetic (i.e., affecting descendants of the exposed individual). Radiation is more likely to produce somatic than genetic effects. The somatic risks of most importance are induced cancers. Except for leukemia, which can have an induction period (time between exposure to the carcinogen and cancer diagnosis) of 2 to 7 years, most cancers have an induction period of more than 20 years.

For uniform irradiation of the body, cancer incidence varies among organs and tissues; the thyroid and skin demonstrate a greater sensitivity than other organs. Such cancers, however, also produce relatively low mortality rates because they are relatively amenable to medical treatment. Because fatal cancer is the most serious effect of environmental and occupational radiation exposures, estimates of cancer fatalities, rather than cancer incidence, are presented as a measure of impact in this document. These estimates are referred to as "latent cancer fatalities" (LCFs), because the cancer may take many years to develop.

Several organizations, in addition to DOE and EPA, continually evaluate the impacts of radiation and provide radiation protection guidance. The responsibilities of the main radiation safety organizations, particularly those that affect policies in the United States, are summarized below.

International Commission on Radiological Protection (ICRP). The ICRP is responsible for providing guidance in matters of radiation safety.

National Council on Radiation Protection and Measurements. In the United States, this council is the national organization that formulates and disseminates guidance and recommendations on radiation protection and measurements that represent the consensus of leading scientific thinking.

National Research Council/National Academy of Sciences. The National Research Council integrates the broad science and technology community with the Academy's mission to further knowledge and advise the Federal Government. The National Research Council's Biological Effects of Ionizing Radiation (BEIR) Committee prepares reports to advise the Federal Government on the health consequences of radiation exposure.

U.S. Nuclear Regulatory Commission (NRC). NRC regulates nuclear power plants and the use of source materials, special nuclear materials, and byproduct materials by commercial and certain governmental entities.

C.1.2 Radiation Exposure Limits

Radiation exposure limits for members of the public and radiation workers are derived from ICRP recommendations. The EPA considers NCRP and ICRP recommendations in setting specific annual exposure limits (usually lower than those specified by the ICRP) in its radiation protection guidance to Federal agencies. The various exposure limits set by DOE and EPA for radiation workers and members of the public are given in **Table C–1**.

Table C–1. Radiation Exposure Limits for Members of the Public and Radiation Workers

Regulation/DOE Order/Standard (organization)	*Public Exposure Limits at the Site Boundary*	*Worker Exposure Limits*
10 CFR Part 835 (DOE)	–	5,000 millirem per year [a]
DOE-STD-1098-2017 (DOE)	–	2,000 millirem per year [b]
DOE Order 458.1 (DOE) [c]	100 millirem per year (all pathways) 10 millirem per year (all air pathways) 4 millirem per year (drinking-water pathway)	–
40 CFR Part 61, Subpart H (EPA)	10 millirem per year (all air pathways)	–
40 CFR Part 141 (EPA)	4 millirem per year (drinking-water pathway)	–

CFR = Code of Federal Regulations; EPA = U.S. Environmental Protection Agency.

[a] Although this measurement is a limit (or level) that is enforced by DOE, worker doses must be managed in accordance with as low as reasonably achievable principles. Refer to footnote b.

[b] This is an administrative control level; exceeding this level generally requires approval of senior management. DOE established this level to assist in achieving its goal of maintaining radiation doses as low as reasonably achievable. DOE recommends that facilities adopt a more limiting Administrative Control Level (DOE 2017). Facility operators must make reasonable attempts to maintain individual worker doses below these levels.

[c] Consistent with 10 CFR Part 20.

C.1.3 Human Health Effects Due to Exposure to Radiation

This section discusses the basic concepts used in the evaluation of radiation effects. Radiation can cause a variety of damaging health effects in humans, both somatic and genetic. Somatic effects (those that affect the exposed individual) are more probable. The most significant effect is induced cancer fatalities. These are called LCFs because the onset of cancer may take many years to develop after the radiation dose is received. In this VTR EIS, LCFs are used as the measure of estimated risk due to radiation exposure.

Cancer is a group of diseases characterized by the uncontrolled growth and spread of abnormal cells. Cancer is caused by both external factors (e.g., tobacco, excessive body weight, infectious organisms, alcohol consumption, and radiation) and internal factors (inherited mutations, hormones, immune conditions, and mutations that occur from metabolism). For the U.S. population of about 310 million, the American Cancer Society estimates that, in 2020, about 1.8 million new cancer cases would be diagnosed and about 606,520 cancer deaths would occur. Just under 20 percent of U.S. cancer deaths are estimated to be caused by tobacco use and slightly less are related to excess weight or obesity, physical inactivity, and poor nutrition. The average U.S. resident has about 4 chances in 10 of developing an invasive cancer over his or her lifetime (40 percent probability for males, 39 percent for females) (American Cancer Society 2020). About 21 percent of all deaths in the United States are due to cancer (CDC 2020).

In 2002, the Interagency Steering Committee on Radiation Standards (ISCORS) recommended that Federal agencies use conversion factors of 0.0006 fatal cancers per rem for mortality and 0.0008 cancers per rem for morbidity (incidences of cancer) when making qualitative or semi-quantitative estimates of risk from radiation exposure to members of the general public. No separate values were recommended for workers. The DOE Office of Environmental and Policy Guidance subsequently recommended that DOE personnel and contractors use the risk factors recommended by ISCORS, stating that, for most purposes, the value for the general population (0.0006 fatal cancers per rem) could be used for both workers and members of the public in National Environmental Policy Act analyses (DOE 2003a).

Publications by both the BEIR Committee and the ICRP support the continued use of the ISCORS-recommended risk values. *Health Risks from Exposure to Low Levels of Ionizing Radiation: BEIR VII Phase 2* (National Research Council 2006) reported fatal cancer risk factors of 0.00048 per rem for males and 0.00066 per rem for females in a population with an age distribution similar to that of the entire U.S. population (average value of 0.00057 per rem for a population with equal numbers of males and females). ICRP Publication 103 (Valentin 2007) recommends nominal cancer risk coefficients of 0.00041 and 0.00055 per rem for adults and the general population, respectively.

Accordingly, a risk factor of 0.0006 LCFs per rem was used in this VTR EIS to estimate risk impacts due to radiation doses from normal operations and accidents. For high, acute individual doses (greater than or equal to 20 rem), the health risk factor is multiplied by 2 (NCRP 1993). The presentation of risks from radiation exposure associated with VTR EIS activities are the increased risks of developing a cancer; that is, they are in addition to the risk of cancer from all other causes.

Using the risk factors discussed above, a calculated dose can be used to estimate the risk of an LCF. For example, if each member of a population of 100,000 people were exposed to a one-time dose of 100 millirem (0.1 rem), the collective dose would be 10,000 person-rem (100,000 persons times 0.1 rem). Using the risk factor of 0.0006 LCFs per person-rem, this collective dose is expected to cause 6 additional LCFs in this population (10,000 person-rem times 0.0006 LCFs per person-rem).

Calculations of the number of LCFs sometimes do not yield whole numbers and may yield a number less than one. For example, if each individual of a population of 100,000 people were to receive an annual dose of 1 millirem (0.001 rem), the collective dose would be 100 person-rem, and the corresponding risk of an LCF would be 0.06 (100,000 persons times 0.001 rem times 0.0006 LCFs per person-rem). A fractional result should be interpreted as a statistical estimate. That is, 0.06 is the average number of LCFs expected if many groups of 100,000 people were to experience the same radiation exposure situation. For most groups, no LCFs would occur; in a few groups, one LCF would occur; in a very small number of groups, two or more LCFs would occur. The average number of LCFs over all of the groups would be 0.06. In this VTR EIS, LCFs calculated for a population are presented as both the rounded whole number, representing the most likely outcome for that population, and the calculated statistical estimate of risk, which is presented in parentheses.

The numerical estimates of LCFs presented in this VTR EIS were obtained using a linear extrapolation from the nominal risk estimated for lifetime total cancer mortality resulting from a dose of 0.1 grays (10 rad). This results in the use of a "linear no-threshold" model. Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of LCFs. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation. Studies of human populations exposed to low doses are inadequate to demonstrate the actual level of risk. However, the latest recommendations of the National Research Council support use of a "linear no-threshold" risk model in which the risk of cancer proceeds in a linear fashion at lower doses without a threshold i.e., any non-zero dose results in an increased risk of cancer (National Research Council 2006).

C.2 Assessment Approach

The dose assessments performed for this VTR EIS were based on site-specific environmental data, facility-specific data, and assumptions related to various exposure parameters. The GENII Version 2 (GENII Environmental Dosimetry System, Version 2) computer code (Version 2.10) was used to calculate the projected doses from normal operations at the INL site, ORNL, and SRS. The GENII computer code complies with quality assurance plans based on the American National Standards Institute Standard NQA-1. This code is one of the toolbox models that meets DOE Order 414.1C, and is overseen by DOE's Office of Quality Assurance Policy and Assistance. All steps of code development were documented and tested, and hand calculations verified the code's implementation of major transport and exposure pathways for a subset of the radionuclide library. The code was reviewed by the EPA Science Advisory Board and a separate, EPA-sponsored, independent peer review panel. The quality assurance of GENII Version 2 has been reviewed by DOE (DOE 2003b) and continues to be rigorously reviewed with each updated version released by Pacific Northwest National Laboratory, the developer of the code.

C.2.1 Meteorological Data

The meteorological data used in the INL, ORNL, and SRS dose assessments are joint frequency distribution (JFD) files created from site-specific meteorological data. A JFD file is a table listing the percentage of time the wind blows from a certain direction, within a certain range of speeds, and within a certain stability class. JFD data for the INL Site were based on measurements taken from the National Oceanographic and Aeronautics Administration/INL Mesonet tower at the Materials and Fuels Complex (MFC) over a 5-year period (2015 through 2019) at a height of 15 meters. JFD data for ORNL were based on measurements taken at ORNL Meteorological Tower A over a 5 year period (2015 through 2019) at a height of 15 meters. JFD data for SRS were based on measurements taken at the H Tower over a 5-year period (2007 through 2011) at a height of 10 meters. Meteorological station parameters and wind-speed midpoints were used in the normal operational assessments. **Tables C–2** through **C–4** present the JFD data used in the INL, ORNL, and SRS analyses, respectively.

Table C–2. Idaho National Laboratory Site Joint Frequency Distribution Data [a]

Average Wind Speed (m/s)	Stability Class	Direction From Which the Wind Blows															
		N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
1.23	A	0.06	0.05	0.08	0.03	0.03	0.02	0.00	0.03	0.02	0.04	0.06	0.07	0.09	0.11	0.09	0.07
	B	0.63	0.60	0.49	0.24	0.13	0.11	0.06	0.11	0.13	0.20	0.26	0.38	0.39	0.36	0.44	0.64
	C	0.11	0.11	0.09	0.06	0.03	0.03	0.02	0.03	0.05	0.09	0.07	0.07	0.09	0.06	0.07	0.07
	D	0.59	0.52	0.59	0.47	0.28	0.19	0.20	0.24	0.30	0.37	0.30	0.25	0.24	0.22	0.27	0.34
	E	0.36	0.40	0.39	0.24	0.13	0.09	0.08	0.11	0.13	0.20	0.19	0.17	0.11	0.14	0.17	0.30
	F	0.62	0.86	1.31	1.11	0.75	0.50	0.53	0.74	0.74	0.83	0.64	0.48	0.36	0.33	0.32	0.41
2.92	A	0.00	0.01	0.01	0.01	0.00	0.00	0.00	0.01	0.01	0.02	0.02	0.02	0.01	0.00	0.00	0.00
	B	0.12	0.21	0.36	0.12	0.04	0.02	0.03	0.09	0.20	0.38	0.39	0.27	0.12	0.04	0.05	0.05
	C	0.64	1.40	1.39	0.34	0.08	0.07	0.07	0.21	0.30	0.56	0.54	0.52	0.23	0.17	0.21	0.33
	D	0.97	1.76	2.67	2.02	0.65	0.29	0.43	1.09	1.66	1.71	1.46	0.78	0.42	0.32	0.34	0.51
	E	0.10	0.24	0.51	0.44	0.16	0.06	0.08	0.17	0.30	0.29	0.19	0.10	0.07	0.04	0.05	0.06
	F	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00
4.94	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.01	0.04	0.11	0.03	0.01	0.01	0.00	0.02	0.11	0.23	0.20	0.04	0.01	0.00	0.01	0.01
	C	0.10	0.26	0.45	0.10	0.03	0.01	0.01	0.09	0.28	0.60	0.52	0.17	0.05	0.05	0.07	0.07
	D	0.62	1.07	1.36	1.09	0.41	0.24	0.44	2.06	2.03	2.87	2.25	0.73	0.21	0.19	0.29	0.51
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
7.38	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	C	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.03	0.07	0.09	0.01	0.00	0.00	0.00	0.00
	D	0.40	1.00	0.96	0.34	0.08	0.08	0.08	0.67	1.91	3.80	4.19	0.92	0.10	0.13	0.15	0.31
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
10.34	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	C	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02	0.03	0.09	0.01	0.00	0.00	0.00	0.00
	D	0.23	0.38	0.21	0.02	0.01	0.00	0.00	0.05	0.55	1.52	2.69	0.56	0.04	0.03	0.01	0.06
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
13.11	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	C	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.08	0.01	0.00	0.00	0.00	0.00
	D	0.06	0.09	0.04	0.00	0.00	0.00	0.00	0.00	0.09	0.34	1.01	0.18	0.00	0.00	0.01	0.01
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
16.42	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	C	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02	0.00	0.00	0.00	0.00	0.00
	D	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.04	0.24	0.03	0.00	0.00	0.00	0.00
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00

E = east; ENE = east-northeast; ESE = east-southeast; INL = Idaho National Laboratory; m/s = meters per second; MFC = Materials and Fuels Complex; N = north; NE = northeast; NNE = north-northeast; NNW = north-northwest; NW = northwest; S = south; SE = southeast; SSE = south-southeast; SSW = south-southwest; SW = southwest; W = west; WNW = west-northwest; WSW = west-southwest.

[a] MFC: 15 meter tower height. Based on 2015 to 2019 meteorological data.

Note: To convert meters per second to miles per hour, multiply by 2.2369; meters to feet, by 3.2808.

Table C–3. Oak Ridge National Laboratory Joint Frequency Distribution Data [a]

Average Wind-speed (m/s)	Stability Class	Direction From Which the Wind Blows															
		S	***SSW***	***SW***	***WSW***	***W***	***WNW***	***NW***	***NNW***	***N***	***NNE***	***NE***	***ENE***	***E***	***ESE***	***SE***	***SSE***
0.55	A	0.03	0.02	0.05	0.08	0.10	0.04	0.03	0.06	0.06	0.07	0.06	0.06	0.02	0.01	0.02	0.01
	B	0.11	0.14	0.23	0.23	0.21	0.20	0.16	0.19	0.24	0.32	0.31	0.29	0.18	0.13	0.10	0.09
	C	0.02	0.04	0.07	0.13	0.07	0.02	0.01	0.01	0.01	0.03	0.07	0.11	0.07	0.04	0.01	0.00
	D	0.31	0.45	0.90	1.30	0.52	0.21	0.18	0.17	0.23	0.41	0.61	1.12	0.75	0.41	0.27	0.27
	E	0.16	0.26	0.50	0.73	0.34	0.17	0.15	0.15	0.24	0.25	0.42	0.72	0.58	0.27	0.15	0.18
	F	0.93	1.51	2.38	2.42	1.34	0.86	0.61	0.71	0.86	0.92	1.17	2.19	1.99	0.79	0.65	0.69
	G	0.32	0.44	0.76	0.94	0.71	0.51	0.52	0.50	0.48	0.44	0.58	0.70	0.64	0.26	0.20	0.18
1.44	A	0.12	0.18	0.27	0.30	0.25	0.22	0.18	0.21	0.29	0.57	0.58	0.36	0.21	0.12	0.06	0.09
	B	0.12	0.21	0.55	0.69	0.43	0.18	0.16	0.16	0.33	0.60	0.89	0.60	0.37	0.21	0.18	0.13
	C	0.03	0.04	0.13	0.37	0.12	0.04	0.02	0.03	0.02	0.17	0.38	0.37	0.20	0.07	0.03	0.03
	D	0.11	0.16	0.79	1.30	0.23	0.06	0.04	0.05	0.09	0.32	0.69	1.11	0.72	0.21	0.12	0.07
	E	0.18	0.23	0.63	0.83	0.31	0.21	0.15	0.16	0.25	0.50	0.76	0.84	0.72	0.49	0.28	0.18
	F	0.08	0.18	0.77	1.33	0.16	0.05	0.03	0.05	0.13	0.24	0.45	0.97	0.54	0.13	0.06	0.04
	G	0.00	0.01	0.10	0.15	0.02	0.00	0.00	0.00	0.00	0.01	0.02	0.07	0.09	0.01	0.01	0.00
2.42	A	0.02	0.02	0.08	0.08	0.06	0.03	0.00	0.00	0.03	0.14	0.18	0.06	0.03	0.04	0.02	0.02
	B	0.04	0.08	0.25	0.26	0.10	0.03	0.02	0.02	0.03	0.26	0.43	0.19	0.10	0.11	0.09	0.03
	C	0.05	0.09	0.30	0.53	0.15	0.04	0.02	0.03	0.08	0.48	0.89	0.43	0.36	0.28	0.08	0.05
	D	0.12	0.15	0.64	0.76	0.14	0.03	0.03	0.05	0.12	0.57	1.24	0.82	0.87	0.64	0.24	0.13
	E	0.03	0.08	0.21	0.25	0.02	0.01	0.00	0.01	0.04	0.07	0.10	0.07	0.06	0.13	0.14	0.07
	F	0.01	0.01	0.07	0.11	0.00	0.00	0.00	0.00	0.01	0.03	0.04	0.01	0.00	0.02	0.01	0.01
	G	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
3.58	A	0.00	0.00	0.01	0.02	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02	0.00
	B	0.01	0.02	0.14	0.14	0.03	0.00	0.00	0.00	0.01	0.19	0.26	0.05	0.03	0.07	0.07	0.02
	C	0.02	0.05	0.17	0.16	0.03	0.00	0.00	0.00	0.04	0.30	0.42	0.14	0.15	0.21	0.07	0.02
	D	0.02	0.05	0.32	0.20	0.02	0.02	0.00	0.02	0.07	0.50	0.89	0.34	0.50	0.52	0.24	0.06
	E	0.01	0.01	0.03	0.02	0.00	0.00	0.00	0.00	0.00	0.02	0.00	0.00	0.00	0.01	0.08	0.02
	F	0.00	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.00
	G	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
5.06	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.02	0.03	0.00	0.00	0.01	0.01	0.00
	C	0.00	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.01	0.09	0.09	0.01	0.03	0.03	0.04	0.01
	D	0.01	0.02	0.05	0.01	0.00	0.00	0.00	0.00	0.04	0.19	0.20	0.09	0.11	0.18	0.12	0.01
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	G	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
6.49	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00
	C	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	D	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.07	0.03	0.01	0.02	0.03	0.02	0.00
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	G	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00

m/s = meters per second.

[a] ORNL: Tower A, 15 meter tower height. Based on 2015 to 2019 meteorological data.

Note: To convert meters per second to miles per hour, multiply by 2.2369; meters to feet, by 3.2808.

Table C–4. Savannah River Site Joint Frequency Distribution Data [a]

Average Wind-speed (m/s)	Stability Class	Direction From Which the Wind Blows															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
1.14	A	0.05	0.08	0.10	0.12	0.12	0.12	0.08	0.11	0.10	0.10	0.09	0.08	0.08	0.08	0.09	0.07
	B	0.00	0.01	0.01	0.03	0.01	0.04	0.02	0.02	0.01	0.02	0.02	0.02	0.02	0.02	0.02	0.02
	C	0.03	0.02	0.04	0.06	0.05	0.04	0.05	0.04	0.04	0.04	0.09	0.06	0.04	0.05	0.04	0.04
	D	0.16	0.17	0.16	0.12	0.18	0.13	0.12	0.15	0.18	0.22	0.28	0.26	0.20	0.17	0.19	0.16
	E	0.26	0.30	0.22	0.24	0.25	0.22	0.21	0.17	0.15	0.18	0.33	0.30	0.26	0.27	0.24	0.30
	F	0.77	0.90	0.75	0.68	0.50	0.51	0.52	0.52	0.49	0.55	0.69	0.33	0.26	0.36	0.44	0.68
2.07	A	0.31	0.32	0.40	0.55	0.55	0.38	0.28	0.21	0.21	0.25	0.27	0.35	0.37	0.29	0.27	0.26
	B	0.06	0.08	0.12	0.16	0.13	0.14	0.06	0.05	0.05	0.07	0.11	0.16	0.12	0.10	0.08	0.07
	C	0.14	0.17	0.20	0.30	0.18	0.21	0.10	0.08	0.07	0.10	0.25	0.24	0.20	0.15	0.12	0.08
	D	0.60	0.43	0.63	0.68	0.58	0.57	0.41	0.29	0.37	0.58	1.18	0.96	0.75	0.59	0.46	0.39
	E	1.47	1.46	1.31	1.27	1.11	0.78	0.49	0.34	0.34	0.44	0.80	0.88	0.86	0.82	0.93	1.05
	F	0.26	0.37	0.20	0.21	0.13	0.13	0.05	0.06	0.15	0.32	0.08	0.02	0.02	0.07	0.10	0.13
3.19	A	0.08	0.07	0.09	0.18	0.23	0.11	0.05	0.06	0.04	0.05	0.07	0.11	0.12	0.09	0.04	0.04
	B	0.16	0.21	0.30	0.61	0.57	0.33	0.10	0.10	0.10	0.12	0.25	0.40	0.38	0.26	0.16	0.05
	C	0.42	0.49	0.59	0.99	0.70	0.51	0.20	0.13	0.21	0.34	0.81	0.73	0.49	0.25	0.22	0.16
	D	2.09	1.33	1.54	1.52	1.62	1.32	0.60	0.34	0.52	0.87	1.85	1.56	1.20	0.90	0.97	1.36
	E	0.59	0.42	0.40	0.29	0.16	0.10	0.05	0.03	0.07	0.14	0.04	0.10	0.12	0.09	0.11	0.23
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
4.85	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	C	0.16	0.22	0.31	0.63	0.66	0.57	0.18	0.05	0.09	0.14	0.40	0.34	0.10	0.05	0.06	0.07
	D	0.70	0.60	0.55	0.61	0.84	1.18	0.27	0.09	0.21	0.27	0.27	0.17	0.10	0.07	0.13	0.66
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
7.59	A	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	B	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	C	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	D	0.02	0.01	0.00	0.04	0.07	0.05	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	E	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	F	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00

m/s = meters per second.

[a] SRS: Tower H, 10 meter tower height. Based on 2007 to 2011 meteorological data.

Note: To convert meters per second to miles per hour, multiply by 2.2369; meters to feet, by 3.2808.

C.2.2 Population Data

The INL Site, ORNL, and SRS population distributions were based on data from the 2010 census and the 2017 five-year American Community Survey update for areas within 50 miles of the locations for the proposed facilities. The 2010 populations derived from the census were projected to the year 2050, which was selected as the representative year for full-scale operations, by calculating a linear trend developed using data from the 2000 and 2010 decennial censuses and the 2017 American Community Survey (Census 2020a, 2020b, 2020c). The populations were spatially distributed on a circular grid with 16 directions and 10 radial distances out to 50 miles. The grids were centered at the proposed location for the VTR at the INL Site (just east of the Zero Power Physics Reactor [ZPPR] in the MFC), at the proposed location for the VTR at ORNL (less than a mile north-east of the High Flux Isotope Reactor [HFIR]), and at the K-Reactor Building in the K Area Complex (K Area) at SRS; the locations from which radionuclides were assumed to be released during incident-free operations at INL,[1] ORNL, and SRS, respectively. During the population distribution allocation process, those individuals who were geographically situated within a sector that was entirely on the INL Site, ORNL, or SRS property were moved (for the analysis) to an adjoining sector to ensure that no individuals were assessed as if they were living on DOE property. **Tables C–5**, **C–6**, and **C–7** present the population data used for the dose assessments.

Potential maximally exposed individual (MEI) locations at each site boundary for all 16 compass directions were evaluated to determine the boundary location with the highest total dose for all facilities associated with the alternatives evaluated in this VTR EIS. (This location differs from locations for the MEI from current operations.) This analysis was performed using population estimates for the year 2050. With an increasing population, an individual could live closer to the border than the current MEI (e.g., INL identifies the MEI as living 1.4 miles south of the INL border and northeast of the East Butte [a farm and cattle operation]). Therefore, a location at the site boundary was used. It was determined that an INL Site boundary location 3.1 miles south of the proposed VTR location at the MFC, yielded the highest annual MEI dose. For ORNL, the boundary location is 1.6 miles to the southeast of the proposed VTR location, on the east bank of Melton Valley Lake. For SRS (K Area), the boundary location is 6.6 miles south-southwest of the K Area. These are the distances and compass directions to this MEI location used in the GENII Version 2 modeling.

Table C–5. Estimated Population Within 50 Miles of INL-MFC in the Year 2050

Direction	*Distance (miles)*									
	0–1	*1–2*	*2–3*	*3–4*	*4–5*	*5–10*	*10–20*	*20–30*	*30–40*	*40–50*
E	0	0	0	0	0	73	223	1,974	20,268	55,812
ENE	0	0	0	0	0	0	286	409	512	738
ESE	0	0	0	0	0	94	398	10,151	126,556	7,896
N	0	0	0	0	0	0	0	104	42	63
NE	0	0	0	0	0	0	287	378	311	51
NNE	0	0	0	0	0	0	281	319	123	51
NNW	0	0	0	0	0	0	35	36	51	72
NW	0	0	0	0	0	0	33	143	328	367
S	0	0	0	8	4	38	153	7,320	13,636	45,759
SE	0	0	0	0	12	112	736	9,092	34,303	1,275
SSE	0	0	0	8	4	58	433	3,622	5,604	698
SSW	0	0	0	0	0	51	147	181	1,937	7,133
SW	0	0	0	0	0	0	178	175	234	573
W	0	0	0	0	0	0	34	35	52	139
WNW	0	0	0	0	0	0	0	158	283	425
WSW	0	0	0	0	0	0	114	56	105	220
Total Population										**363,570**

Note: Centered on 43.592755 degrees latitude N, 112.651649 degrees longitude W.
Source: Census 2020a, 2020b, 2020c.

[1] Additional sources of VTR-related releases at the INL Site include the Hot Fuel Examination Facility, Fuel Manufacturing Facility, Fuel Conditioning Facility, and Zero Power Physics Reactor. All of these facilities are located within the MFC, relatively close to the proposed location of the VTR. Separate population distributions centered on these facilities were not generated.

Table C–6. Estimated Population Surrounding Oak Ridge National Laboratory in the Year 2050

	Distance (miles)									
Direction	**0–1**	**1–2**	**2–3**	**3–4**	**4–5**	**5–10**	**10–20**	**20–30**	**30–40**	**40–50**
E	0	19	215	369	523	18,661	138,371	113,771	74,799	94,315
ENE	0	109	62	270	397	10,735	90,465	105,804	25,278	29,539
ESE	0	22	253	480	810	26,451	53,546	52,846	19,227	49,227
N	0	0	0	92	985	8,820	1,302	1,894	3,254	6,286
NE	0	46	0	7	0	5,957	22,494	15,732	12,145	11,378
NNE	0	0	0	0	402	16,223	9,824	20,221	17,109	6,882
NNW	0	0	0	328	1,150	5,732	1,997	1,528	9,190	9,257
NW	0	0	0	22	340	2,245	9,316	3,057	3,125	9,136
S	0	8	45	231	354	15,643	17,236	16,269	11,975	3,622
SE	0	76	255	478	1,458	18,195	30,638	51,497	962	1,741
SSE	0	27	180	300	701	10,067	11,359	11,586	2,873	4,354
SSW	0	0	25	188	246	3,381	16,525	19,607	32,906	17,150
SW	0	0	0	112	151	1,238	4,177	5,272	10,244	15,883
W	0	0	0	0	0	1,503	12,370	7,538	18,003	36,793
WNW	0	0	0	0	0	1,441	6,713	3,150	10,299	12,036
WSW	0	0	0	20	73	1,994	12,510	7,482	10,446	11,916
Total Population										**1,617,562**

Note: Centered on 35.925707 degrees latitude N, 84.290790 degrees longitude W.
Source: Census 2020a, 2020b, 2020c.

Table C–7. Estimated Population Surrounding Savannah River Site K Area in the Year 2050

	Distance (miles)									
Direction	**0–1**	**1–2**	**2–3**	**3–4**	**4–5**	**5–10**	**10–20**	**20–30**	**30–40**	**40–50**
E	0	0	0	0	0	0	4,912	2,575	4,444	3,774
ENE	0	0	0	0	0	6	2,299	4,411	5,750	22,437
ESE	0	0	0	0	0	51	1,026	1,942	2,194	3,221
N	0	0	0	0	0	0	15,044	45,274	8,158	16,239
NE	0	0	0	0	0	0	1,975	4,101	4,021	15,810
NNE	0	0	0	0	0	0	2,845	3,787	7,060	28,309
NNW	0	0	0	0	0	0	10,997	51,820	21,221	8,431
NW	0	0	0	0	0	94	6,368	124,148	177,640	11,342
S	0	0	0	0	0	62	523	1,607	5,153	7,461
SE	0	0	0	0	0	66	352	1,931	4,974	6,788
SSE	0	0	0	0	0	95	252	436	1,793	2,110
SSW	0	0	0	0	0	80	1,325	2,037	3,951	6,506
SW	0	0	0	0	0	168	1,026	1,905	1,974	2,430
W	0	0	0	0	0	186	2,642	4,954	4,223	4,376
WNW	0	0	0	0	0	152	3,953	75,035	76,496	20,717
WSW	0	0	0	0	0	242	2,558	7,519	1,756	5,364
Total Population										**888,904**

Note: Centered on 33.211800 degrees latitude N, 81.663915 degrees longitude W.
Source: Census 2020a, 2020b, 2020c.

Population distributions for use in the Environmental Justice analysis were developed in the same manner as described above for the total populations. These population distributions are presented in Attachment C1 to this appendix.

C.2.3 Agricultural Data

Ingestion exposures from atmospheric transport include ingestion of farm products and inadvertent ingestion of soil. Farm products include leafy vegetables, other vegetables, cereal grains, fruit, cow's milk, beef, poultry, and eggs. The concentration in plants at the time of harvest was evaluated as the sum of contributions from deposition onto plant surfaces, as well as uptake through the roots. Pathways by which animal products may become contaminated include animal ingestion of contaminated plants, water, and soil. Site-specific agricultural data were not developed. This analysis used the generic agricultural production data and the human consumption rates provided in the GENII code for both the population and MEI calculations.

C.2.4 Source Term Data

Table C–8 presents the stack parameters for INL, ORNL, and SRS facilities. Stack heights, sizes, velocities, temperatures, and release locations were provided in the responses to the facility data requests supporting this VTR EIS (INL 2020a; SRNS 2020). These parameters affect the distribution of radioactive emissions from the stacks. **Table C–9** identifies the VTR-related activities associated with each facility identified in Table C–8.

Table C–8. Stack Parameters

Stack Parameter	*VTR (Radwaste HVAC)*	*VTR (RVACS exhaust)*	*HFEF*	*FMF*	*FCF*	*ZPPR*	*ORNL Hot Cell Facility* [a]	*K-Reactor Building* [b]
Height (feet)	99	98	95	46	200	75	95	124
Area (square feet)	3.1	38 [c]	17.4	7.1	19.6	3.0	20	7.1
Flow Velocity (feet/second)	450 [d]	3.9 [d]	40.7	15.1 [d]	29.6	27.9	44	42
Average Temperature (°F)	105	<500	72.3	64	72 [e]	68	72.3	72

FCF = Fuel Conditioning Facility; FMF = Fuel Manufacturing Facility; HFEF = Hot Fuel Examination Facility; HVAC = heating, ventilation, and air conditioning; RVACS = Reactor Vessel Auxiliary Cooling System; VTR = Versatile Test Reactor, ZPPR = Zero Power Physics Reactor.

[a] Parameters for this facility were estimated based on the HFEF stack parameters and adjusted for the larger size of the facility

[b] Final height, area, and location would be refined when a design is finalized. Some parameters are estimated based on existing K-Reactor Building stack.

[c] RVACS has four stacks, exhaust area is for each stack.

[d] Calculated based on area and flow rates provided.

[e] Discharge is at ambient temperature.

Table C–9. Locations used for VTR-Related Activity Stack Emissions

Activity	*Facility Location*		
	INL	*ORNL*	*SRS*
VTR Operation [a]	VTR	VTR	–
Post-Irradiation Examination	HFEF	Hot Cell Facility	–
Spent Fuel Treatment	FCF	Hot Cell Facility	–
Feedstock Preparation	FCF	–	K Area
Fuel Fabrication	FMF/ZPPR	–	K Area

FCF = Fuel Conditioning Facility; FMF = Fuel Manufacturing Facility; HFEF = Hot Fuel Examination Facility; VTR = Versatile Test Reactor; ZPPR = Zero Power Physics Reactor.

[a] Most emissions are from the facility HVAC stack. However, activated argon is in the air emitted from the RVACS stack.

As discussed in Appendix B, at INL the final fuel assembly fabrication step would be performed in ZPPR. At ZPPR fuel assemblies would be fabricated using fuel rods produced at FMF. Few if any emissions would result from this portion of the fuel assembly fabrication process. At INL, irradiated test articles would be transferred to the Hot Fuel Examination Facility (HFEF) for decontamination, initial examination, and preparation for transfer to additional examination facilities. While other MFC facilities would be used for further examination, the emissions from HFEF are used to represent the total post-irradiation examination emissions. Similarly, at ORNL where additional existing facilities would be used for post-irradiation examination, the post-irradiation examination releases are modeled as coming from the new hot cell facility.

Tables C–10 through **C–14,** respectively, present the estimated incident-free radiological releases (source terms), based on the following activities: VTR operation, test article post-irradiation examination, VTR spent fuel treatment, VTR feedstock preparation, and reactor fuel fabrication. The source terms were provided in responses to facility data requests supporting this VTR EIS (INL 2020a; SRNS 2020). The source terms are based on emissions from INL and SRS facilities or proposed projects and scaled to adjust for the types and quantities of expected emissions from VTR facilities.

All releases, except for the argon released from the VTR RVACS, would be filtered through high efficiency particulate air (HEPA) filters. The real-world performance of multiple stages of HEPA filters has been well demonstrated and experimental testing confirms the performance of HEPA filters for uranium and plutonium particles. The independent Defense Nuclear Facilities Safety Board thoroughly evaluated the use of HEPA filters by DOE) and has issued multiple reports on the performance of HEPA filters within the DOE complex. HEPA filters used in support of the VTR activities would conform to the latest version of DOE Standard "Specifications for HEPA Filters Used by DOE Contractors," DOE-STD-3020-2015. Performance testing required by this standard for all HEPA filters credited for safety would ensure that the filters meet or exceed the performance requirements assumed in safety evaluations.

For the post-irradiation examination operational releases (Table C–11), the isotopes in bold are those that contributed at least 0.1 percent of the total offsite dose from MFC operations in 2018 based on the INL Annual Site Environmental Report (INL 2019). Other isotopes listed are those with releases greater than 10^{-10} curies. Spent fuel treatment releases, Table C–12, are limited to those with releases greater than 10^{-10} curies per year.

Source terms were determined to be independent of the location for the VTR and its associated facilities, feedstock preparation, and fuel fabrication, e.g., the VTR source term would be the same whether the VTR were located at the INL Site or ORNL.

Table C–10. Annual Radiological Releases from Versatile Test Reactor Operation

Isotope	*Curies*	*Isotope*	*Curies*
Argon-41	27.1 [a]	Krypton-88	8.9×10^{-6}
Cesium-135	9.0×10^{-16}	Xenon-131m	1.6×10^{-2}
Cesium-137	1.2×10^{-12}	Xenon-133	1.0×10^{-3}
Cesium-138	2.0×10^{-6}	Xenon-133m	5.4×10^{-7}
Hydrogen-3 (Tritium)	1.2	Xenon-135	4.2×10^{-5}
Krypton-83m	1.8×10^{-6}	Xenon-135m	1.5×10^{-6}
Krypton-85	0.70	Xenon-137	7.4×10^{-7}
Krypton-85m	3.5×10^{-6}	Xenon-138	4.4×10^{-6}
Krypton-87	4.8×10^{-6}		

[a] Argon is released through both the VTR plant stack (0.14 curies) and the RVACS stacks (27 curies due to air activation).

Source: INL 2020a, 2020b.

Table C–11. Annual Radiological Releases from Post-Irradiation Examination Operations

Isotope	*Release (curies)*	*Isotope*	*Release (curies)*
Antimony-125	3.2×10^{-5}	Krypton-85	4.4×10^{-3}
Americium-241	**8.4×10^{-12}**	Neptunium-237	3.2×10^{-9}
Carbon-14	**3.1×10^{-4}**	Phosphorus-32	2.6×10^{-5}
Cadmium-109	5.2×10^{-4}	Phosphorus-33	4.9×10^{-9}
Cadmium-115m	1.0×10^{-7}	**Plutonium-238**	**1.2×10^{-10}**
Chlorine-36	1.0×10^{-5}	**Plutonium-239**	**9.5×10^{-8}**
Cobalt-60	**7.9×10^{-13}**	**Plutonium-240**	**3.0×10^{-12}**
Cesium-134	8.0×10^{-7}	Plutonium-242	1.8×10^{-9}
Cesium-137	**2.5×10^{-2}**	Sodium-22	3.2×10^{-6}
Hydrogen-3 (Tritium)	**3.7×10^{-2}**	Sodium-24	1.7×10^{-8}
Iodine-129	**1.8×10^{-5}**	Sulfur-35	1.2×10^{-4}
Iodine-131	8.9×10^{-3}	**Strontium-90**	**3.8×10^{-7}**

Source: INL 2020a, 2020b.

Table C–12. Annual Radiological Releases from Spent Fuel Treatment

Isotope	*Curies*	*Isotope*	*Curies*
Cadmium-113m	4.2×10^{-10}	Nickel-63	2.8×10^{-10}
Cerium-144	1.4×10^{-6}	Promethium-147	1.3×10^{-7}
Cesium-134	2.6×10^{-7}	Plutonium-238	1.2×10^{-10}
Cesium-137	2.0×10^{-6}	Plutonium-239	2.8×10^{-9}
Cobalt-60	2.1×10^{-9}	Plutonium-240	1.9×10^{-10}
Europium-154	1.7×10^{-10}	Plutonium-241	1.2×10^{-9}
Europium-155	2.1×10^{-9}	Ruthenium-106	5.7×10^{-6}
Iron-55	5.5×10^{-8}	Antimony-125	1.6×10^{-7}
Hydrogen-3 (Tritium)	5.1×10^{2}	Samarium-151	9.0×10^{-10}
Krypton-85	8.3×10^{3}	Strontium-90	3.5×10^{-8}

Source: INL 2020a, 2020b.

Table C–13. Annual Radiological Releases from Feedstock Preparation

Isotope	*Curies*	*Isotope*	*Curies*
Americium-241	6.6×10^{-4}	Uranium-232	5.8×10^{-12}
Plutonium-238	9.5×10^{-6}	Uranium-234	1.7×10^{-9}
Plutonium-239	9.6×10^{-6}	Uranium-235	1.5×10^{-11}
Plutonium-240	1.4×10^{-5}	Uranium-236	2.2×10^{-10}
Plutonium-241	2.0×10^{-4}	Uranium-238	4.3×10^{-11}
Plutonium-242	2.2×10^{-8}		

Source: Adapted from SRNS 2020.

Table C–14. Annual Radiological Releases from Fuel Fabrication

Isotope	*Curies*	*Isotope*	*Curies*
Americium-241	3.3×10^{-4}	Uranium-232	7.3×10^{-12}
Plutonium-238	2.3×10^{-6}	Uranium-234	2.2×10^{-9}
Plutonium-239	3.7×10^{-6}	Uranium-235	1.9×10^{-11}
Plutonium-240	2.4×10^{-6}	Uranium-236	2.8×10^{-10}
Plutonium-241	5.7×10^{-5}	Uranium-238	5.4×10^{-11}
Plutonium-242	1.7×10^{-9}		

Source: Adapted from SRNS 2020.

Because activities associated with spent fuel storage only involve movement and storage of materials within certified containers, no significant airborne radiological emissions would result from these activities.

C.2.5 Other Calculation Assumptions

To estimate the radiological impacts of incident-free operation of the VTR facilities, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977):

- All receptors were assumed to be exposed to radioactive material deposited on the ground from facility emissions. Exposure pathways include direct exposure from air immersion and ground exposure, inhalation, and translocation through the food chain.
- The annual exposure time to the plume (for inhalation and immersion) and soil contamination was assumed to be 0.7 years for the MEI.
- The annual exposure time to the plume (for inhalation and immersion) and soil contamination was assumed to be 0.5 years for the population.
- The annual exposure time to the plume (for inhalation and immersion) was assumed to be 1 year for the MEI, average individual and general population.
- Noninvolved worker exposure was limited to the plume and resuspension pathways; ingestion exposure pathways were not considered. The annual exposure time to the plume (for inhalation and immersion) was assumed to be 2,500 hours.
- All receptors were assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of adult humans.
- The GENII model uses a finite plume (i.e., Gaussian) model for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of animal products.
- The calculated internal doses were assumed to be the 50-year committed effective dose equivalent from 1 year of emissions.
- At the INL Site all releases relating to post-irradiation examination were modeled as being emitted from the HFEF. Most post-irradiation examination releases are anticipated to be from this facility.
- Two release points exist for the VTR, the HVAC exhaust and the RVACS stack. Only one release point was modeled, the HVAC exhaust. The release from the RVACS stack (argon-41) was combined with the other radionuclides released from the HVAC exhaust.

In addition to the calculation assumptions listed above, a risk estimator of 0.0006 latent cancer fatalities per rem or person-rem (600 cancer deaths per 1 million rem or person-rem) received by workers or members of the public is used in the impact assessments (DOE 2003a).

C.3 Results for Idaho National Laboratory

The following subsections present the potential incident-free radiological impacts that could occur from VTR operation, feedstock preparation, and reactor fuel fabrication at the INL Site. Radiological impacts from VTR operation include impacts from operation of the VTR, test article post-irradiation examination, and spent fuel treatment and storage. Human health risks from construction and normal operations are evaluated for several individuals, including a noninvolved worker, a hypothetical MEI at the site boundary, and an average member of the public. Human health risk from construction and normal operations are also evaluated for the offsite population within 50 miles of the MFC.

For the purposes of this VTR EIS, an involved worker[2] (worker) is a facility worker who is directly or indirectly involved with operations at a facility and might receive an occupational radiation exposure due to direct radiation (neutron, x-ray, beta, or gamma) or through radionuclides released as a part of normal VTR-related operations. Noninvolved workers are assumed to be outside of the facility would not be subject to direct radiation exposure due to building shielding and appreciable distances between operational facilities, but could be exposed to operational releases.

Materials released from VTR-related operations activities include both particulates and fission product gases. All material would be released through facility stacks. Particulates would be filtered through HEPA filters and gases would be absorbed by charcoal bed absorbers in the VTR exhaust system. Normal releases would be very small, in the millicurie to less than millicurie-per-year quantities, in most cases. But argon, tritium, and krypton would be released in curie quantities.

Materials released due to feedstock preparation and fuel fabrication would be particulates (primarily plutonium and uranium isotopes and americium-241) that would be released through tall stacks. Particulates would be filtered through HEPA filters before being released. These filter systems are designed to protect the onsite workforce and the public from normal and accidental releases. Normal releases from all facilities would be very small—in the microcurie to less than millicurie-per-year.

C.3.1 Construction

There would be no radiological risk to members of the public from potential construction of the VTR or modification of the INL MFC facilities. VTR construction would occur in an undeveloped area adjacent to the ZPPR at MFC where worker exposures would be to background radiation only. Modifications to equipment within facility hot cells (fuel and sampling equipment replacement within HFEF and the Fuel Conditioning Facility [FCF]) are anticipated. Modification work within the HFEF would not result in worker exposures beyond those currently being experienced. However, modification work in the FCF to support spent fuel treatment and in the Fuel Manufacturing Facility (FMF) and ZPPR to support reactor fuel fabrication occurs in an area where workers would be expected to receive an operational dose. Equipment would be designed, assembled, and tested in radiologically clean areas. Installation of the equipment within the hot cells would be performed remotely. To enable feedstock preparation and fuel fabrication, new gloveboxes and supporting equipment would be installed in the FMF and FCF, and new fuel pin handling and fuel assembly fabrication and handling equipment would be installed in ZPPR. Radiological and nonradiological worker impacts associated with these construction efforts are provided in Chapter 4.

C.3.2 Operations

Under the INL VTR Alternative and the INL reactor fuel production options, the following program activities could occur at the INL Site and could result in doses to the public and a noninvolved worker:

VTR Operation. Operation would include:

- *VTR Reactor Operations*. Multiple fuel cycles would be run each year. Reactor operation would be the principle source of potential normal releases. Fuel and test article handling, washing, and movement would also occur, but these activities would be performed within fuel and test article casks.

[2] Involved worker impacts are calculated for the VTR project. However, the analysis of worker dose (average individual and collective worker population dose) is performed simply based on the number of workers involved in VTR-related activities, and the expected exposure received during these activities. Involved worker doses are estimated based on existing environmental conditions at the sites or based upon analysis of worker activities resulting in exposure. Radiological impacts to involved workers are provided in Chapter 4 of this EIS.

- *Post-Irradiation Examination*. Test articles would be transferred to the HFEF for decontamination and initial post-irradiation examination. Test articles, in whole or in part, could be sent to additional INL facilities for further examination
- *Spent Fuel Treatment*. Fuel removed from the reactor core would be stored, for up to a year within the VTR reactor vessel and for at least 3 years on a storage pad. Fuel would then be transferred to the FCF for treatment (sodium bond removal and fuel downblending with a diluent) and repackaging. Treated spent fuel would be returned to the storage pad pending transfer to an offsite repository.

Feedstock Preparation. Non-metallic plutonium feed would be converted into metal and plutonium with unacceptable levels of impurities would be polished. All activities would be performed in gloveboxes designed and installed specifically for feedstock preparation within the FCF. The polished metallic plutonium product would be used as feed for fuel fabrication.

Fuel Fabrication. Fuel fabrication would include plutonium, uranium, and zirconium melting and alloying, casting of fuel pins, and fabrication of fuel assemblies. Melting, alloying and pin casting would all be performed in gloveboxes. All operations would take place in FMF and ZPPR.

Additional details about these operations are provided in Appendix B.

Tables C–15 through **C–17** present the projected incident-free radiological impacts on a noninvolved worker and the public from VTR-related operations at MFC.

Table C–15. Radiological Impacts on a Noninvolved Worker and the Public from the INL VTR Alternative

	Noninvolved Worker	***Maximally Exposed Individual***	***Population***	***Average Individual***
Annual dose	0.0021 millirem	0.0068 millirem	0.044 person-rem	1.2×10^{-4} millirem [a]
Regulatory dose limit [b]	--	10 millirem	--	10 millirem
Annual LCF risk [c]	1×10^{-9}	4×10^{-9}	0 (3×10^{-5})	Less than 1×10^{-10}
Percent of natural background radiation [d]	0.0006	0.002	3×10^{-5}	3×10^{-5}

FCF = Fuel Conditioning Facility; HFEF = Hot Fuel Examination Facility; LCF = latent cancer fatality; VTR = Versatile Test Reactor.

[a] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the INL facilities in 2050 (approximately 379,265 for MFC).

[b] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

[c] LCF risk for individuals; projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[d] The annual natural background radiation dose assumed for the INL Site is 383 millirem for the average individual (INL 2019); the population within 50 miles of MFC in 2050 would receive a dose of about 145,000 person-rem.

Table C–16. Radiological Impacts on a Noninvolved Worker and the Public from the INL Feedstock Preparation Option

	Noninvolved Worker	*Maximally Exposed Individual*	*Population*	*Average Individual*
Annual dose	0.0017 millirem	0.0012 millirem	0.012 person-rem	3.2×10^{-5} millirem [a]
Regulatory dose limit [b]	--	10 millirem	--	10 millirem
Annual LCF risk [c]	1×10^{-9}	7×10^{-10}	0 (7×10^{-6})	Less than 1×10^{-10}
Percent of natural background radiation [d]	0.0004	0.0003	8×10^{-6}	8×10^{-6}

LCF = latent cancer fatality.

[a] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the INL facilities in 2050 (approximately 379,265 for MFC).

[b] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

[c] LCF risk for individuals; projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[d] The annual natural background radiation dose assumed for the INL Site is 383 millirem for the average individual (INL 2019); the population within 50 miles of MFC in 2050 would receive a dose of about 145,000 person-rem.

Table C–17. Radiological Impacts on a Noninvolved Worker and the Public from the INL Fuel Fabrication Option

	Noninvolved Worker	*Maximally Exposed Individual*	*Population*	*Average Individual*
Annual dose	0.067 millirem	0.0016 millirem	0.0053 person-rem	1.5×10^{-5} millirem [a]
Regulatory dose limit [b]	--	10 millirem	--	10 millirem
Annual LCF risk [c]	1×10^{-7}	1×10^{-9}	0 (3×10^{-6})	Less than 1×10^{-10}
Percent of natural background radiation [d]	0.02	0.0004	4×10^{-6}	4×10^{-6}

FCF = Fuel Conditioning Facility; LCF = latent cancer fatality; ZPPR = Zero Power Physics Reactor.

[a] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the INL facilities in 2050 (approximately 379,265 for MFC).

[b] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

[c] LCF risk for individuals, projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[d] The annual natural background radiation dose assumed for the INL Site is 383 millirem for the average individual (INL 2019); the population within 50 miles of MFC in 2050 would receive a dose of about 145,000 person-rem.

As indicated by the results for the MEI, the annual potential doses from normal releases (on the order of 0.0035 to 0.0057 millirem) are small fractions (less than or about 0.002 percent) of the natural background radiation dose of 383 millirem per year (see Chapter 3, Section 3.1.10). A conservative estimate of the dose to a noninvolved INL worker is also calculated. Assuming no shielding, a location within MFC near about 200 meters from the VTR INL fuel fabrication facility release point would result in the highest dose to the noninvolved worker, an incremental annual dose of about 0.067 millirem. (Doses to the noninvolved worker located 300 meters from VTR operations and 400 meters from feedstock preparation, would have lower annual doses from the operation of these facilities.) This dose is small relative to the dose from natural background radiation and much smaller than the dose an involved worker would receive.

Worker impacts for these operations are provided in Chapter 4.

C.4 Results for Oak Ridge National Laboratory

The following subsections present the potential incident-free radiological impacts that could occur from VTR operations at ORNL under the ORNL VTR Alternative. Human health risks from construction and normal operations are evaluated for several individual and population groups; including noninvolved workers, a hypothetical MEI at the site boundary, and an average member of the public. Human health risk from construction and normal operations are also evaluated for the offsite population within 50 miles of the proposed VTR location. As stated in Section C.4 the impacts to involved workers are discussed in Chapter 4.

Materials released from VTR-related operations activities include both particulates and fission product gases. All material would be released through facility stacks. Particulates would be filtered HEPA filters and gases would be absorbed by charcoal bed absorbers in the VTR exhaust system. Most material would be released in less than millicurie-per-year quantities, although argon, tritium, and krypton would be release in curie quantities.

C.4.1 Construction

There would be no radiological risk to members of the public or workers from construction of the VTR and the Hot Cell Facility at ORNL. Construction would occur in an undeveloped area where worker exposures would be to background radiation only. No radiological emissions that could impact the public would result from construction.

Nonradiological worker impacts for these operations are provided in Chapter 4.

C.4.2 Operations

Under the ORNL VTR Alternative the following program activities could occur at ORNL and could result in doses to the public:

- *VTR Reactor Operations*. Multiple fuel cycles would be run each year. Reactor operation would be the principle source of potential normal releases. Fuel and test article handling, washing, and movement would also occur, but these activities would be performed within fuel and test article casks.
- *Post-Irradiation Examination*. Test articles would be transferred to the Hot Cell Facility for decontamination and initial post-irradiation examination. Test articles, in whole or in part, could be sent to additional ORNL facilities for further examination
- *Spent Fuel Treatment*. Fuel removed from the reactor core would be stored for up to a year within the VTR reactor vessel and for at least 3 years on a storage pad. Fuel would then be transferred to the Hot Cell Facility for treatment (sodium bond removal and fuel downblending with a diluent) and repackaging. Treated spent fuel would be returned to the storage pad pending transfer to an offsite repository.

Additional details about these operations are provided in Appendix B.

Tables C–18 presents the projected incident-free radiological impacts on a noninvolved worker and the public from operations at the VTR and Hot Cell Facility.

As indicated by the results for the MEI, the annual potential doses from normal releases (on the order of 0.031 millirem) would be a small fraction (approximately 0.01 percent) of the natural background radiation dose of 300 millirem per year (see Chapter 3, Section 3.3.10). A conservative estimate of the dose to a noninvolved ORNL worker is also calculated. The proposed VTR site would not be within any of the currently developed areas of ORNL. The nearest continuously occupied area to the proposed VTR site would be the HFIR complex (about 4,700 feet), the 7000 area (about 3,700 feet), the ORNL main campus (about 5,400 feet) and the Energy System Test Complex (about 4,800 feet). Assuming no shielding, a location at the ORNL Energy System Test Complex would result in the highest dose to the noninvolved worker, an incremental annual dose of about 0.0048 millirem. This dose is small compared to the dose from natural background radiation and much smaller than the dose an involved worker would receive.

Worker impacts for these operations are provided in Chapter 4.

Table C–18. Radiological Impacts on a Noninvolved Worker and the Public from the ORNL VTR Alternative

	Noninvolved Worker	*Maximally Exposed Individual*	*Population*	*Average Individual*
Annual dose	0.0048 millirem	0.031 millirem	0.58 person-rem	3.6×10^{-4} millirem [a]
Regulatory dose limit [b]	--	10 millirem	--	10 millirem
Annual LCF risk [c]	3×10^{-9}	2×10^{-8}	0 (3×10^{-4})	2×10^{-10}
Percent of natural background radiation [d]	0.002	0.01	0.0001	0.0001

LCF = latent cancer fatality.

[a] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the ORNL facilities in 2050 (approximately 1,553,177 for the proposed VTR site).

[b] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

[c] LCF risk for individuals; projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[d] The annual natural background radiation dose assumed for ORNL is 300 millirem for the average individual (ORO 2019); the population within 50 miles of the proposed VTR site in 2050 would receive a dose of about 466,000 person-rem.

C.5 Results for Savannah River Site

The following subsections present the potential incident-free radiological impacts that could occur from feedstock preparation and reactor fuel fabrication at SRS. Human health risks from construction and normal operations are evaluated for several individuals, including a noninvolved worker, a hypothetical MEI at the site boundary, and an average member of the public. Human health risk from construction and normal operations are also evaluated for the offsite population within 50-miles of the SRS K Area. As stated in Section C.4 the impacts to involved workers are discussed in Chapter 4.

All of the materials released due to feedstock preparation and fuel fabrication would be particulates (primarily plutonium and uranium isotopes and americium-241) that would be released through a new facility stack. Particulates would be filtered through HEPA filters before being released. These filter systems are designed to protect the onsite workforce and the public from normal and accidental releases. Normal releases would be very small—in the microcurie to less than millicurie-per-year range.

C.5.1 Construction

There would be no radiological risk to members of the public from potential construction or modification of facilities at the K Area. Construction worker exposures to radiation derived from other activities at the site, past or present, would be kept ALARA. Construction workers would be monitored (badged), as appropriate. Limited demolition, removal, and decontamination actions within the K-Reactor buildings would be required to support installation of new equipment. Construction activities would include 3 years of decontamination and equipment removal from K Area. To enable feedstock preparation and fuel fabrication, new gloveboxes and supporting equipment would be installed in the K Area. Radiological and nonradiological worker impacts associated with this construction effort are provided in Chapter 4.

C.5.2 Operations

Under the fuel production options the following possible program activities could occur at SRS and would result in doses to the public:

- *Feedstock Preparation.* Non-metallic plutonium feed would be converted into metal and plutonium with unacceptable levels of impurities would be polished. All activities would be performed in gloveboxes designed and installed specifically for feedstock preparation located in the K Area. The polished metallic plutonium product would be used as feed for fuel fabrication.
- *Fuel Fabrication*. Fuel fabrication would include plutonium, uranium, and zirconium melting and alloying, casting of fuel pins, and fabrication of fuel assemblies. Melting, alloying, and pin casting would all be performed in gloveboxes. All operations would take place in the K Area.

Additional details about these operations are provided in Appendix B.

Tables C–19 and **C–20** present the projected incident-free radiological impacts on a noninvolved worker and the public from operations at the K Area.

Table C–19. Radiological Impacts on a Noninvolved Worker and the Public from the SRS Feedstock Preparation Option

	Noninvolved Worker	*Maximally Exposed Individual*	*Population*	*Average Individual*
Annual dose	0.0061 millirem	0.0015 millirem	0.042 person-rem	4.7×10^{-5} millirem [a]
Regulatory dose limit [b]	–	10 millirem	–	10 millirem
Annual LCF risk [c]	4×10^{-9}	9×10^{-10}	0 (2×10^{-5})	Less than 1×10^{-10}
Percent of natural background radiation [d]	0.002	0.0005	2×10^{-5}	2×10^{-5}

LCF = latent cancer fatality.

[a] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the SRS facilities in 2050 (approximately 885,150 for K Area).

[b] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

[c] LCF risk for individuals; projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[d] The annual natural background radiation dose assumed for SRS is 311 millirem for the average individual (SRNS 2019); the population within 50 miles of K Area in 2050 would receive a dose of about 277,000 person-rem.

Table C–20. Radiological Impacts on a Noninvolved Worker and the Public from SRS Fuel Fabrication Option

	Noninvolved Worker	*Maximally Exposed Individual*	*Population*	*Average Individual*
Annual dose	0.0030 millirem	0.00071 millirem	0.020 person-rem	2.3×10^{-5} millirem [a]
Regulatory dose limit [b]	--	10 millirem	--	10 millirem
Annual LCF risk [c]	2×10^{-9}	4×10^{-10}	0 (1×10^{-5})	Less than 1×10^{-10}
Percent of natural background radiation [d]	0.001	0.0002	7×10^{-6}	7×10^{-6}

LCF = latent cancer fatality.

[a] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the SRS facilities in 2050 (approximately 885,150 for K Area).

[b] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

[c] LCF risk for individuals; projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[d] The annual natural background radiation dose assumed for SRS is 311 millirem for the average individual (SRNS 2019); the population within 50 miles of K Area in 2050 would receive a dose of about 277,000 person-rem.

As indicated by the results for the MEI, the annual potential doses from normal releases (between 0.00071 and 0.0015 millirem) would be small fractions (less than 0.001 percent) of the natural background radiation dose of 311 millirem per year (see Chapter 3, Section 3.3.10). A conservative estimate of the dose to a noninvolved SRS worker is also calculated. Assuming no shielding, a location 500 meters from the K-Reactor Building would result in the highest dose to the noninvolved worker, an incremental annual dose of between 0.003 and 0.006 millirem. This dose is small compared to the dose from natural background radiation and much smaller than the dose an involved worker would receive.

Worker impacts for these operations are provided in Chapter 4.

C.6 Environmental Justice Results

Tables C–21 through **C–23** present the results of the population and average individual impact assessments for minority and low-income populations. These impacts are calculated in the same manner as the impacts for the total populations. The population distributions for the Total Minority, African American, Native American, White Hispanic, Other Minority, and Low-Income groups (provided in Attachment C1 to this appendix) are used in place of the total population distributions. All other exposure parameters for the population from the general population analysis are used.

A separate MEI calculation was not performed for these groups. The MEI is assumed to be an individual located at the site boundary where the highest individual dose occurs. No specific attributes of a minority or low-income individual were identified that would indicate that the exposure parameters should be modified to address unique characteristics of a minority or low-income MEI.

Table C–21. Radiological Impacts on Minority and Low-Income Populations from VTR-Related Operations at INL – MFC

	Total Minority	*African American*	*Native American*	*White Hispanic*	*Other Minority*	*Low Income*
INL VTR Alternative						
Population						
Annual dose (person-rem)	0.0084	0.00027	0.00073	0.0028	0.0060	0.0062
Annual LCF [a]	0 (5×10^{-6})	0 (2×10^{-7})	0 (4×10^{-7})	0 (2×10^{-6})	0 (4×10^{-6})	0 (4×10^{-6})
Annual dose from natural background radiation [b] (person-rem)	29,000	830	1,700	7,800	18,000	21,000
Percent of natural background radiation	3×10^{-5}	3×10^{-5}	4×10^{-5}	4×10^{-5}	3×10^{-5}	3×10^{-5}
Average Individual [c]						
Annual dose (millirem) [d]	0.00011	0.00013	0.00017	0.00014	0.00013	0.00011
Annual LCF Risk [c]	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}
INL Feedstock Preparation Option						
Population						
Annual dose (person-rem)	0.0026	7.3×10^{-5}	0.00019	0.00072	0.0016	0.0016
Annual LCF [a]	0 (2×10^{-6})	0 (4×10^{-8})	0 (1×10^{-7})	0 (5×10^{-7})	0 (1×10^{-6})	0 (1×10^{-6})
Annual dose from natural background radiation [b] (person-rem)	29,000	830	1,700	7,800	18,000	21,000
Percent of natural background radiation	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}
Average Individual [c]						
Annual dose (millirem) [d]	3.5×10^{-5}	3.4×10^{-5}	4.4×10^{-5}	3.7×10^{-5}	3.4×10^{-5}	3.0×10^{-5}
Annual LCF Risk [c]	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}

	Total Minority	*African American*	*Native American*	*White Hispanic*	*Other Minority*	*Low Income*
INL Fuel Fabrication Option						
Population						
Annual dose (person-rem)	0.0012	3.3×10^{-5}	8.4×10^{-5}	0.00034	0.00073	0.00072
Annual LCF [a]	0 (7×10^{-7})	0 (2×10^{-8})	0 (5×10^{-8})	0 (2×10^{-7})	0 (4×10^{-7})	0 (4×10^{-7})
Annual dose from natural background radiation [b] (person-rem)	29,000	830	1,700	7,800	18,000	21,000
Percent of natural background radiation	4×10^{-6}	4×10^{-6}	5×10^{-6}	4×10^{-6}	4×10^{-6}	3×10^{-6}
Average Individual [c]						
Annual dose (millirem) [d]	1.6×10^{-5}	1.5×10^{-5}	1.9×10^{-5}	1.7×10^{-5}	1.5×10^{-5}	1.3×10^{-5}
Annual LCF Risk [c]	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}

LCF = latent cancer fatality.

[a] LCF risk for individuals, projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[b] The annual natural background radiation dose assumed for the INL Site is 383 millirem for the average individual (INL 2019).

[c] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the proposed INL VTR facilities at MFC in 2050 (populations are provided in Attachment C1 to this appendix).

[d] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

Table C–22. Radiological Impacts on Minority and Low-Income Populations from the ORNL VTR Alternative

	Total Minority	*African American*	*Native American*	*White Hispanic*	*Other Minority*	*Low Income*
Population						
Annual dose (person-rem)	0.093	0.023	0.0019	0.017	0.051	0.087
Annual LCF [a]	0 (6×10^{-5})	0 (1×10^{-5})	0 (1×10^{-6})	0 (1×10^{-5})	0 (3×10^{-5})	0 (5×10^{-5})
Annual dose from natural background radiation [b] (person-rem)	90,000	22,000	2,100	12,000	54,000	78,000
Percent of natural background radiation	1×10^{-4}	1×10^{-4}	9×10^{-5}	1×10^{-4}	1×10^{-4}	1×10^{-4}
Average Individual [c]						
Annual dose (millirem) [d]	0.00031	0.00033	0.00027	0.00042	0.00028	0.00034
Annual LCF Risk [a]	2×10^{-10}	2×10^{-10}	2×10^{-10}	3×10^{-10}	2×10^{-10}	2×10^{-10}

LCF = latent cancer fatality.

[a] LCF risk for individuals, projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[b] The annual natural background radiation dose assumed for ORNL is 300 millirem for the average individual (ORO 2019).

[c] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the proposed ORNL VTR facilities in 2050 (populations are provided in Attachment C1 to this appendix).

[d] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

Table C–23. Radiological Impacts on Minority and Low-Income Populations from the SRS Fuel Production Options

	Total Minority	*African American*	*Native American*	*White Hispanic*	*Other Minority*	*Low Income*
SRS Feedstock Preparation Option						
Population						
Annual dose (person-rem)	0.022	0.017	0.00011	0.0011	0.0034	0.0079
Annual LCF [a]	0 (1×10^{-5})	0 (1×10^{-5})	0 (7×10^{-8})	0 (7×10^{-7})	0 (2×10^{-6})	0 (5×10^{-6})
Annual dose from natural background radiation [b]	140,000	100,000	700	8,100	25,000	49,000
Percent of natural background radiation	2×10^{-5}	2×10^{-5}	2×10^{-5}	1×10^{-5}	1×10^{-5}	2×10^{-5}
Average Individual [c]						
Annual dose (millirem) [d]	5.0×10^{-5}	5.2×10^{-5}	5.0×10^{-5}	4.3×10^{-5}	4.2×10^{-5}	5.1×10^{-5}
Annual LCF Risk [a]	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}
SRS Fuel Fabrication Option						
Population						
Annual dose (person-rem)	0.011	0.0083	5.6×10^{-5}	0.00054	0.0017	0.0039
Annual LCF [a]	0 (6×10^{-6})	0 (5×10^{-6})	0 (3×10^{-8})	0 (3×10^{-7})	0 (1×10^{-6})	0 (2×10^{-6})
Annual dose from natural background radiation [b]	140,000	100,000	700	8,100	25,000	49,000
Percent of natural background radiation	8×10^{-6}	8×10^{-6}	8×10^{-6}	7×10^{-6}	7×10^{-6}	8×10^{-6}
Average Individual [c]						
Annual dose (millirem) [b]	2.4×10^{-5}	2.5×10^{-5}	2.5×10^{-5}	2.1×10^{-5}	2.0×10^{-5}	2.5×10^{-5}
Annual LCF Risk [a]	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}	Less than 1×10^{-10}

LCF = latent cancer fatality.

[a] LCF risk for individuals, projected number of fatalities for the population. Numbers of LCFs in the population are whole numbers; the statistically calculated values are provided in parentheses.

[b] The annual natural background radiation dose assumed for SRS is 311 millirem for the average individual (SRNS 2019).

[c] Obtained by dividing the population dose by the number of people projected to live within 50 miles of the proposed SRS fuel preparation facilities in 2050 (populations are provided in Attachment C1 to this appendix).

[d] 40 CFR Part 61, Subpart H, establishes an annual limit of 10 millirem via the air pathway to any member of the public from DOE operations.

C.7 References

American Cancer Society, 2020, *Cancer Facts & Figures 2020*, Atlanta, Georgia.

CDC (Centers for Disease Control and Prevention), 2020, Mortality in the United States 2018, NCHS Data Brief No. 355, National Center for Health Statistics, January.

Census (U.S. Census Bureau), 2020a, 2019 Tiger/Line Shapefiles: Block Groups (available at https://www.census.gov/cgi-bin/geo/shapefiles/index.php?year=2019&layergroup=Block+Groups), March.

Census (U.S. Census Bureau), 2020b, American FactFinder, B03002 Hispanic or Latin Origin by Race (available at https://factfinder.census.gov/faces/nav/jsf/pages/searchresults.xhtml?refresh=t#none), March.

Census (U.S. Census Bureau), 2020c, American FactFinder, C17002 Ratio of Income to Poverty Level in the past 12 Months (available at https://factfinder.census.gov/faces/nav/jsf/pages/searchresults.xhtml?refresh=t#none), March.

DOE (U.S. Department of Energy), 2003a, *Estimating Radiation Risk from Total Effective Dose Equivalent (TEDE), ISCORS Technical Report No. 1,* DOE/EH-412/0015/0802, Rev. 1, Office of Environmental Policy and Guidance, Washington, DC, January.

DOE (U.S. Department of Energy), 2003b, *GENII Computer Code Application Guidance for Documented Safety Analysis, Interim Report*, DOE-EH-4.2.1.4-Interim-GENII, Rev. 1, Office of Environment, Safety and Health, Washington, DC, September.

DOE (U.S. Department of Energy), 2017, *Radiological Control*, DOE-STD-1098-2017, Change Notice 1, Washington, DC, May.

INL (Idaho National Laboratory), 2019, *Idaho National Laboratory Site Environmental Report Calendar Year 2018*, DOE/ID-12082(18), Environmental Surveillance, Education, and Research Program, Idaho Operations Office, Idaho Falls, Idaho, September.

INL (Idaho National Laboratory), 2020a, VTR Hazards and Impacts Information in Support of National Environmental Policy Act Data Needs, INL/EXT-19-55588, July.

INL (Idaho National Laboratory), 2020b, TEV-3976 Versatile Test Reactor (VTR Waste and Material Data for Environmental Impact Statement (EIS), Idaho Falls, Idaho. DRAFT

National Research Council, 2006, *Health Risks from Exposure to Low Levels of Ionizing Radiation: BEIR VII Phase 2*, Committee on the Biological Effects of Ionizing Radiation, National Academy Press, Washington, DC.

NCRP (National Council on Radiation Protection and Measurements), 1993, *Risk Estimates for Radiation Protection,* NCRP Report No. 115, Bethesda, Maryland, December 31.

NCRP (National Council on Radiation Protection and Measurements), 2009, *Ionizing Radiation Exposure of the Population of the United States*, NCRP Report No. 160, Bethesda, Maryland, March 3.

NRC (U.S. Nuclear Regulatory Commission), 1977, *Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR 50*, Appendix I, Regulatory Guide 1.109, Rev. 1, Office of Standards Development, Washington, DC, October.

ORO (U.S. Department of Energy - Oak Ridge Operations Office), 2019, *Oak Ridge Reservation Annual Site Environmental Report 2018*, DOE/ORO-2512, Oak Ridge, Tennessee, September.

SRNS (Savannah River Nuclear Solutions), 2019, *Savannah River Site Environmental Report 2018*, SRNS-RP-2019-00022, Aiken, South Carolina.

SRNS (Savannah River Nuclear Solutions), 2020, Savannah River Site Data Call Response for the Versatile Test Reactor Fuel Fabrication Facility, SRNS-RP-2020-00286, Rev. 2, Aiken, South Carolina, July 22.

Valentin, J., 2007, *The 2007 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 103, Elsevier Ltd.

Attachment C1:
Environmental Justice Population Distributions

Minority and Low-Income Populations

This attachment to Appendix C presents the projected 2050 population distributions for Total Minority, African American, Native American, Other Minority, White Hispanic, and low-income populations. The subject populations are presented in **Tables C–24** through **C–29** for the INL Site; **Tables C–30** through **C–35** for ORNL; and **Tables C–36** through **C–41** for SRS.

Table C–24. Estimated Total Minority Population Within 50 Miles of the Proposed VTR Complex at the INL–MFC in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	0	23	87	470	2,016	8,285
ENE	0	0	0	0	0	0	110	127	115	41
ESE	0	0	0	0	0	30	149	1,992	29,643	754
N	0	0	0	0	0	0	0	34	26	22
NE	0	0	0	0	0	0	111	93	80	34
NNE	0	0	0	0	0	0	109	87	43	34
NNW	0	0	0	0	0	0	8	8	13	16
NW	0	0	0	0	0	0	8	10	13	21
S	0	0	0	1	0	5	20	2,637	4,479	8,578
SE	0	0	0	0	1	33	229	1,913	4,997	57
SSE	0	0	0	1	0	12	121	648	842	458
SSW	0	0	0	0	0	7	21	41	596	3,855
SW	0	0	0	0	0	0	29	41	67	411
W	0	0	0	0	0	0	8	8	13	18
WNW	0	0	0	0	0	0	0	17	11	10
WSW	0	0	0	0	0	0	22	13	15	93
Total Minority Population										**74,940**

Table C–25. Estimated African American Population Within 50 Miles of the Proposed VTR Complex at INL–MFC in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	0	0	1	0	38	575
ENE	0	0	0	0	0	0	1	1	0	9
ESE	0	0	0	0	0	0	0	74	798	0
N	0	0	0	0	0	0	0	13	2	6
NE	0	0	0	0	0	0	1	0	0	1
NNE	0	0	0	0	0	0	1	1	0	1
NNW	0	0	0	0	0	0	7	8	12	15
NW	0	0	0	0	0	0	7	6	5	1
S	0	0	0	0	0	0	0	169	100	212
SE	0	0	0	0	0	0	0	10	11	0
SSE	0	0	0	0	0	0	0	1	1	1
SSW	0	0	0	0	0	0	0	0	0	1
SW	0	0	0	0	0	0	0	0	0	0
W	0	0	0	0	0	0	7	8	12	7
WNW	0	0	0	0	0	0	0	13	5	1
WSW	0	0	0	0	0	0	3	7	3	0
Total African American Population										**2,156**

Table C–26. Estimated Native American Population Within 50 Miles the Proposed VTR Complex at INL–MFC in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	0	0	1	4	27	17
ENE	0	0	0	0	0	0	2	1	0	3
ESE	0	0	0	0	0	0	0	7	200	53
N	0	0	0	0	0	0	0	2	10	6
NE	0	0	0	0	0	0	2	0	5	14
NNE	0	0	0	0	0	0	2	1	9	14
NNW	0	0	0	0	0	0	0	0	0	0
NW	0	0	0	0	0	0	0	0	0	0
S	0	0	0	0	0	0	0	28	876	1,573
SE	0	0	0	0	0	0	0	58	17	0
SSE	0	0	0	0	0	0	0	11	218	364
SSW	0	0	0	0	0	0	0	0	227	704
SW	0	0	0	0	0	0	0	0	0	0
W	0	0	0	0	0	0	0	0	0	0
WNW	0	0	0	0	0	0	0	0	0	0
WSW	0	0	0	0	0	0	0	0	0	0
Native American Total Population										**4,456**

Table C–27. Estimated Other Minority Population Within 50 Miles of the Proposed VTR Complex at INL–MFC in the Year 2050 [a]

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	0	14	40	230	1,649	5,065
ENE	0	0	0	0	0	0	45	50	39	19
ESE	0	0	0	0	0	28	147	954	22,419	240
N	0	0	0	0	0	0	0	7	6	4
NE	0	0	0	0	0	0	46	32	25	7
NNE	0	0	0	0	0	0	45	31	12	7
NNW	0	0	0	0	0	0	0	0	0	0
NW	0	0	0	0	0	0	0	0	0	11
S	0	0	0	0	0	1	5	1,671	1,899	3,834
SE	0	0	0	0	0	29	204	1,292	3,464	31
SSE	0	0	0	0	0	8	101	528	262	84
SSW	0	0	0	0	0	2	7	40	230	2,400
SW	0	0	0	0	0	0	17	41	65	407
W	0	0	0	0	0	0	0	0	0	3
WNW	0	0	0	0	0	0	0	0	1	5
WSW	0	0	0	0	0	0	13	5	3	78
Total Other Minority Population										**47,902**

[a] Includes people who identified as Asian, Native Hawaiian and other Pacific Islanders, Some other race, and two or more races.

Table C–28. Estimated White Hispanic Population Within 50 Miles of the Proposed VTR Complex at INL–MFC in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	0	9	45	236	302	2,628
ENE	0	0	0	0	0	0	62	75	76	10
ESE	0	0	0	0	0	2	2	957	6,226	461
N	0	0	0	0	0	0	0	12	8	6
NE	0	0	0	0	0	0	62	61	50	12
NNE	0	0	0	0	0	0	61	54	22	12
NNW	0	0	0	0	0	0	1	0	1	1
NW	0	0	0	0	0	0	1	4	8	9
S	0	0	0	1	0	4	15	769	1,604	2,959
SE	0	0	0	0	1	4	25	553	1,505	26
SSE	0	0	0	1	0	4	20	108	361	9
SSW	0	0	0	0	0	5	14	1	139	750
SW	0	0	0	0	0	0	12	0	2	4
W	0	0	0	0	0	0	1	0	1	8
WNW	0	0	0	0	0	0	0	4	5	4
WSW	0	0	0	0	0	0	6	1	9	15
Total White Hispanic Population										**20,426**

Table C–29. Estimated Low-Income Population Surrounding the Proposed VTR Complex at INL–MFC in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	0	11	22	394	1,749	17,511
ENE	0	0	0	0	0	0	23	35	28	33
ESE	0	0	0	0	0	25	124	1,039	17,198	382
N	0	0	0	0	0	0	0	10	9	9
NE	0	0	0	0	0	0	23	19	18	12
NNE	0	0	0	0	0	0	23	18	12	12
NNW	0	0	0	0	0	0	3	3	5	8
NW	0	0	0	0	0	0	3	32	71	76
S	0	0	0	0	0	1	3	1,049	2,677	6,481
SE	0	0	0	0	0	11	52	827	2,504	121
SSE	0	0	0	0	0	2	27	237	363	95
SSW	0	0	0	0	0	1	4	11	284	988
SW	0	0	0	0	0	0	6	11	17	11
W	0	0	0	0	0	0	3	3	5	21
WNW	0	0	0	0	0	0	0	30	49	49
WSW	0	0	0	0	0	0	5	5	18	27
Total Low-Income Population										**54,938**

Table C–30. Estimated Total Minority Population Within 50 Miles of the Proposed VTR Complex at ORNL in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	4	7	26	2,318	37,033	39,778	26,086	37,756
ENE	0	1	1	4	33	1,292	15,132	15,969	1,287	3,704
ESE	0	0	4	56	153	4,043	9,009	6,806	2,388	27,373
N	0	0	0	29	315	2,052	10	65	88	411
NE	0	1	0	0	0	1,029	1,702	565	614	467
NNE	0	0	0	0	160	4,330	522	996	560	532
NNW	0	0	0	68	274	568	50	51	390	160
NW	0	0	0	4	60	133	2,464	73	66	397
S	0	0	0	1	3	4,027	1,105	1,410	1,472	167
SE	0	1	4	60	266	3,011	2,396	4,187	53	157
SSE	0	0	1	2	17	1,120	874	669	236	1,240
SSW	0	0	1	4	5	292	3,775	2,113	4,731	1,454
SW	0	0	0	1	1	48	488	578	859	1,343
W	0	0	0	0	0	114	989	644	1,162	3,503
WNW	0	0	0	0	0	225	292	124	227	583
WSW	0	0	0	0	0	85	997	478	441	2,013
Total Minority Population										**299,518**

Table C–31. Estimated African American Population Within 50 Miles of the Proposed VTR Complex at ORNL in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	2	859	13,180	24,232	945	351
ENE	0	0	0	0	7	284	6,122	7,057	256	467
ESE	0	0	0	5	12	692	1,801	1,412	61	256
N	0	0	0	10	116	679	0	6	3	213
NE	0	0	0	0	0	85	339	119	141	0
NNE	0	0	0	0	68	801	151	105	54	188
NNW	0	0	0	16	44	100	30	0	232	33
NW	0	0	0	1	11	18	2,224	10	6	10
S	0	0	0	1	2	297	158	290	7	3
SE	0	0	0	6	13	312	378	945	0	0
SSE	0	0	1	2	1	55	16	0	0	9
SSW	0	0	0	0	0	40	508	610	796	362
SW	0	0	0	0	0	23	91	83	92	261
W	0	0	0	0	0	0	301	106	247	576
WNW	0	0	0	0	0	12	136	24	51	0
WSW	0	0	0	0	0	7	279	31	112	507
Total African American Population										**71,565**

Table C–32. Estimated Native American Population Within 50 Miles of the Proposed VTR Complex at ORNL in the Year 2050

	Distance (miles)									
Direction	***0-1***	***1-2***	***2-3***	***3-4***	***4-5***	***5-10***	***10-20***	***20-30***	***30-40***	***40-50***
E	0	0	0	0	0	37	471	268	62	261
ENE	0	0	0	0	0	16	216	352	95	44
ESE	0	0	0	0	0	13	227	318	73	199
N	0	0	0	0	0	10	3	4	9	17
NE	0	0	0	0	0	11	18	12	3	13
NNE	0	0	0	0	1	34	17	54	30	51
NNW	0	0	0	0	0	0	3	30	57	33
NW	0	0	0	0	0	0	7	8	1	3
S	0	0	0	0	0	0	0	120	535	28
SE	0	0	0	0	0	25	86	101	13	141
SSE	0	0	0	0	0	47	54	43	137	1,080
SSW	0	0	0	0	0	0	17	71	116	10
SW	0	0	0	0	0	6	83	40	2	30
W	0	0	0	0	0	43	117	0	76	492
WNW	0	0	0	0	0	0	3	11	14	124
WSW	0	0	0	0	0	41	29	12	22	200
Native American Total Population										**7,050**

Table C–33. Estimated Other Minority Population Within 50 Miles of the Proposed VTR Complex at ORNL in the Year 2050

	Distance (miles)									
Direction	***0-1***	***1-2***	***2-3***	***3-4***	***4-5***	***5-10***	***10-20***	***20-30***	***30-40***	***40-50***
E	0	0	1	2	16	1,130	18,536	12,380	23,833	34,492
ENE	0	0	0	1	20	608	6,664	6,714	665	1,945
ESE	0	0	1	40	123	2,254	5,513	2,817	1,793	25,976
N	0	0	0	19	198	1,131	6	43	58	131
NE	0	0	0	0	0	827	1,211	386	355	279
NNE	0	0	0	0	91	2,502	275	516	350	285
NNW	0	0	0	41	174	366	6	1	50	87
NW	0	0	0	2	35	74	159	14	48	341
S	0	0	0	0	1	2,083	244	376	704	64
SE	0	0	1	43	222	2,265	1,333	2,404	40	14
SSE	0	0	0	0	10	676	511	289	97	98
SSW	0	0	1	4	5	111	1,377	1,007	2,468	615
SW	0	0	0	1	1	11	223	345	631	570
W	0	0	0	0	0	62	518	394	459	1,190
WNW	0	0	0	0	0	154	141	36	106	350
WSW	0	0	0	0	0	37	553	398	155	703
Total Other Minority Population										**179,686**

[a] Includes people who identified as Asian, Native Hawaiian and other Pacific Islanders, Some other race, and two or more races.

Table C–34. Estimated White Hispanic Population Within 50 Miles of the Proposed VTR Complex at ORNL in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	3	5	8	292	4,846	2,898	1,246	2,652
ENE	0	1	1	3	6	384	2,130	1,846	271	1,248
ESE	0	0	3	11	18	1,084	1,468	2,259	461	942
N	0	0	0	0	1	232	1	12	18	50
NE	0	1	0	0	0	106	134	48	115	175
NNE	0	0	0	0	0	993	79	321	126	8
NNW	0	0	0	11	56	102	11	20	51	7
NW	0	0	0	1	14	41	74	41	11	43
S	0	0	0	0	0	1,647	703	624	226	72
SE	0	1	3	11	31	409	599	737	0	2
SSE	0	0	0	0	6	342	293	337	2	53
SSW	0	0	0	0	0	141	1,873	425	1,351	467
SW	0	0	0	0	0	8	91	110	134	482
W	0	0	0	0	0	9	53	144	380	1,245
WNW	0	0	0	0	0	59	12	53	56	109
WSW	0	0	0	0	0	0	136	37	152	603
Total White Hispanic Population										**41,217**

Table C–35. Estimated Low-Income Population Within 50 Miles of the Proposed VTR Complex at ORNL in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	1	16	27	43	1,406	25,460	30,989	12,183	15,794
ENE	0	8	4	20	17	604	12,602	13,646	3,330	4,927
ESE	0	2	18	49	78	1,481	4,075	7,889	2,992	5,464
N	0	0	0	22	231	1,936	548	313	784	2,084
NE	0	3	0	1	0	288	3,076	2,126	2,630	2,099
NNE	0	0	0	0	119	2,844	1,800	2,807	4,067	2,066
NNW	0	0	0	30	82	798	533	367	2,359	2,574
NW	0	0	0	1	16	282	1,196	653	657	1,905
S	0	1	4	20	35	3,266	1,870	3,513	2,676	262
SE	0	6	19	51	61	593	2,787	6,168	109	247
SSE	0	2	15	24	45	781	1,053	1,196	260	793
SSW	0	0	5	36	46	471	2,859	4,255	6,329	2,833
SW	0	0	0	20	27	155	503	863	1,554	2,907
W	0	0	0	0	0	210	2,808	1,588	2,149	7,166
WNW	0	0	0	0	0	214	1,403	711	825	2,192
WSW	0	0	0	3	13	105	1,357	1,001	1,668	2,532
Total Low-Income Population										**259,087**

Table C–36. Estimated Total Minority Population Within 50 Miles of K Area at SRS in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	0	0	2,870	700	3,121	1,846
ENE	0	0	0	0	0	3	1,171	3,360	3,900	15,350
ESE	0	0	0	0	0	33	701	1,702	951	1,199
N	0	0	0	0	0	0	4,292	20,399	3,174	8,835
NE	0	0	0	0	0	0	824	2,849	1,474	6,053
NNE	0	0	0	0	0	0	1,030	937	2,718	6,816
NNW	0	0	0	0	0	0	2,814	14,531	10,803	4,386
NW	0	0	0	0	0	39	2,217	83,785	60,303	3,506
S	0	0	0	0	0	27	219	681	2,316	2,049
SE	0	0	0	0	0	54	329	1,878	3,323	4,102
SSE	0	0	0	0	0	59	153	291	1,041	1,054
SSW	0	0	0	0	0	36	635	494	1,882	2,133
SW	0	0	0	0	0	94	566	912	830	989
W	0	0	0	0	0	110	1,165	2,210	1,731	1,863
WNW	0	0	0	0	0	72	2,113	66,490	41,438	8,343
WSW	0	0	0	0	0	143	1,704	4,662	984	3,726
Total Minority Population										**441,593**

Table C–37. Estimated African American Population Within 50 Miles of K Area at SRS in the Year 2050

Direction	Distance (miles)									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
E	0	0	0	0	0	0	2,618	642	2,790	1,484
ENE	0	0	0	0	0	3	1,087	3,078	3,646	12,330
ESE	0	0	0	0	0	30	656	1,664	794	1,030
N	0	0	0	0	0	0	2,291	15,980	2,396	5,898
NE	0	0	0	0	0	0	635	2,676	1,222	4,265
NNE	0	0	0	0	0	0	653	787	1,996	3,007
NNW	0	0	0	0	0	0	1,660	9,974	4,442	2,664
NW	0	0	0	0	0	31	1,559	71,050	34,343	2,362
S	0	0	0	0	0	27	214	588	1,898	1,505
SE	0	0	0	0	0	45	269	1,849	2,704	3,540
SSE	0	0	0	0	0	54	145	283	744	863
SSW	0	0	0	0	0	36	626	449	1,297	1,531
SW	0	0	0	0	0	93	563	853	579	894
W	0	0	0	0	0	108	949	1,805	1,389	1,534
WNW	0	0	0	0	0	60	1,808	58,300	27,145	5,341
WSW	0	0	0	0	0	140	1,668	4,097	782	3,353
Total African American Population										**331,871**

Table C–38. Estimated Native American Population Within 50 Miles of K Area at SRS in the Year 2050

	Distance (miles)									
Direction	***0-1***	***1-2***	***2-3***	***3-4***	***4-5***	***5-10***	***10-20***	***20-30***	***30-40***	***40-50***
E	0	0	0	0	0	0	52	2	88	37
ENE	0	0	0	0	0	0	12	5	14	30
ESE	0	0	0	0	0	2	23	8	0	9
N	0	0	0	0	0	0	18	10	3	37
NE	0	0	0	0	0	0	4	26	17	48
NNE	0	0	0	0	0	0	22	6	36	78
NNW	0	0	0	0	0	0	1	41	39	33
NW	0	0	0	0	0	0	7	60	174	22
S	0	0	0	0	0	0	0	1	23	71
SE	0	0	0	0	0	7	47	12	6	16
SSE	0	0	0	0	0	3	5	0	8	13
SSW	0	0	0	0	0	0	0	22	217	63
SW	0	0	0	0	0	0	0	0	2	1
W	0	0	0	0	0	0	3	5	3	1
WNW	0	0	0	0	0	0	9	63	645	41
WSW	0	0	0	0	0	0	0	0	2	1
Native American Total Population										**2,254**

Table C–39. Estimated Other Minority Population Within 50 Miles of K Area at SRS in the Year 2050

	Distance (miles)									
Direction	***0-1***	***1-2***	***2-3***	***3-4***	***4-5***	***5-10***	***10-20***	***20-30***	***30-40***	***40-50***
E	0	0	0	0	0	0	168	43	221	265
ENE	0	0	0	0	0	0	48	266	236	2,777
ESE	0	0	0	0	0	1	18	15	147	159
N	0	0	0	0	0	0	1,407	3,792	576	2,050
NE	0	0	0	0	0	0	169	117	199	1,393
NNE	0	0	0	0	0	0	93	56	508	2,571
NNW	0	0	0	0	0	0	680	2,678	5,897	1,547
NW	0	0	0	0	0	7	463	8,658	19,958	594
S	0	0	0	0	0	0	4	55	333	348
SE	0	0	0	0	0	2	10	14	517	503
SSE	0	0	0	0	0	2	1	4	178	153
SSW	0	0	0	0	0	0	9	4	87	381
SW	0	0	0	0	0	0	1	39	226	74
W	0	0	0	0	0	0	187	323	281	273
WNW	0	0	0	0	0	11	266	7,375	8,613	2,575
WSW	0	0	0	0	0	0	24	490	73	248
Total Other Minority Population										**81,461**

[a] Includes people who identified as Asian, Native Hawaiian and other Pacific Islanders, Some other race, and two or more races.

Table C–40. Estimated White Hispanic Population Within 50 Miles of K Area at SRS in the Year 2050

	Distance (miles)									
Direction	**0-1**	**1-2**	**2-3**	**3-4**	**4-5**	**5-10**	**10-20**	**20-30**	**30-40**	**40-50**
E	0	0	0	0	0	0	32	13	22	60
ENE	0	0	0	0	0	0	24	11	4	213
ESE	0	0	0	0	0	0	4	15	10	1
N	0	0	0	0	0	0	576	617	199	850
NE	0	0	0	0	0	0	16	30	36	347
NNE	0	0	0	0	0	0	262	88	178	1,160
NNW	0	0	0	0	0	0	473	1,838	425	142
NW	0	0	0	0	0	1	188	4,017	5,828	528
S	0	0	0	0	0	0	1	37	62	125
SE	0	0	0	0	0	0	3	3	96	43
SSE	0	0	0	0	0	0	2	4	111	25
SSW	0	0	0	0	0	0	0	19	281	158
SW	0	0	0	0	0	1	2	20	23	20
W	0	0	0	0	0	2	26	77	58	55
WNW	0	0	0	0	0	1	30	752	5,035	386
WSW	0	0	0	0	0	3	12	75	127	124
Total White Hispanic Population										**26,007**

Table C–41. Estimated Low-Income Population Within 50 Miles of K Area at SRS in the Year 2050

	Distance (miles)									
Direction	**0-1**	**1-2**	**2-3**	**3-4**	**4-5**	**5-10**	**10-20**	**20-30**	**30-40**	**40-50**
E	0	0	0	0	0	0	1,340	434	910	650
ENE	0	0	0	0	0	1	475	1,362	1,370	4,743
ESE	0	0	0	0	0	9	225	625	381	487
N	0	0	0	0	0	0	1,482	8,405	1,198	3,898
NE	0	0	0	0	0	0	389	1,541	740	3,323
NNE	0	0	0	0	0	0	624	849	1,544	5,590
NNW	0	0	0	0	0	0	874	7,898	2,326	1,198
NW	0	0	0	0	0	14	976	33,352	14,140	1,245
S	0	0	0	0	0	17	147	326	1,227	1,405
SE	0	0	0	0	0	16	94	548	899	1,068
SSE	0	0	0	0	0	19	58	65	239	485
SSW	0	0	0	0	0	22	420	436	1,341	1,232
SW	0	0	0	0	0	68	317	503	467	633
W	0	0	0	0	0	80	625	986	1,056	863
WNW	0	0	0	0	0	24	883	17,258	7,513	4,856
WSW	0	0	0	0	0	105	1,065	2,153	237	1,522
Total Low-Income Population										**155,896**

Appendix D
Human Health Impacts from Facility Accidents

APPENDIX D
HUMAN HEALTH IMPACTS FROM FACILITY ACCIDENTS

D.1 Impact Assessment Methods for Facility Accidents

D.1.1 Introduction

This appendix presents the methodology and assumptions used for estimating potential impacts and risks associated with both radiological and hazardous material releases, due to postulated accidents, at the facilities being considered for the Versatile Test Reactor (VTR), VTR fuel production facilities, VTR post-irradiation examination facilities, VTR spent fuel treatment facilities, and VTR spent fuel storage facilities. Accidents involving transuranic (TRU) waste generated during operation of the VTR and associated facilities are also considered. Considerations for accident selection are presented in Section D.2. Selection of postulated accidents for the various alternatives and options and determination of the associated radiological source terms are presented in Section D.3. Radiological impacts from the accidents are presented in Section D.4. Assessment of hazardous material releases is discussed in Section D.5. Information regarding the impacts of normal operations, along with background information on the health impacts from exposure to ionizing radiation and exposure to hazardous materials, is provided in Appendix C.

D.1.1.1 Consequences and Risks

Metrics commonly used in environmental impact statements (EISs) to present the potential impacts of accidents are consequences and risks. The consequences are the potential impacts that would result if the accident were to occur. Accident consequences may be presented as impacts on individuals or a specified population (e.g., residents within 50 miles of an accident) and in terms of dose (e.g., rem or person-rem) or health effects (e.g., latent cancer fatalities [LCFs]). Risk is defined as the product of the consequences and estimated frequency of a given accident. The accident frequency is the number of times the accident is expected to occur over a given time (e.g., per year).

Accident consequences are determined using the MELCOR Accident Consequence Code System, Generation 2] (MACCS2) computer program/code (NRC 1990, 1998). A number of specific types of risk can be directly calculated from the output of the MACCS2 computer program. The risk to a noninvolved worker or to a maximally exposed member of the public (MEI) can be calculated. The MACCS2 computer code yields a dose to the noninvolved worker and the MEI. Using the risk factor of 0.0006 LCFs per rem (DOE 2003), the consequence in terms of the likelihood of an LCF can be calculated. The risk to this hypothetical individual is calculated by multiplying the consequence in terms of an LCF by the estimated accident frequency. For example, if an accident has an estimated frequency of 0.001 per year and the dose from the accident is 1 rem, the risk is $0.001 \times (1 \times 0.0006) = 6 \times 10^{-7}$ LCFs per year.

Calculation of the population risk is also possible. Population risk, which is the product of the total consequences experienced by the population and accident frequency, is a measure of the expected number of LCFs experienced by the population as a whole over the course of a year.[1] For example, if an accident has a frequency of 0.001 per year and the consequence of the accident is 5 LCFs in the population, then the population risk is $0.001 \times 5 = 0.005$ LCFs per year.

[1] Population distribution data for each facility considered in this VTR EIS is presented in Appendix C.

D.1.1.2 Uncertainties and Conservatism

The analyses of accidents are based on calculations relevant to hypothetical sequences of events and models of their effects. The models provide estimates of the frequencies, source terms, pathways for dispersion, exposures, and effects on human health and the environment that are as realistic as possible within the scope of the analysis. In many cases, minimal experience with the postulated accidents leads to uncertainty in the calculation of their consequences and frequencies. This fact has prompted the use of models or input values that yield conservative estimates of consequence and frequency. All alternatives have been evaluated using uniform methods and data, allowing for a fair comparison of all alternatives (DOE 2004b).

One method for evaluating alternatives involves comparing individual and population risks that can be calculated from the information in this VTR EIS. The equations for such calculations involve accident frequency and accident consequences, parameters that are subject to uncertainty. The uncertainty in estimates of the frequency of events can vary over several orders of magnitude. Similarly, consequence calculations depend on inputs that can vary by several orders of magnitude. The generally accepted practice is to report accident frequencies qualitatively, in terms of broad frequency bins, as opposed to numerically and to report consequences as numeric values from the consequence calculations. Therefore, in this VTR EIS, the consequence metrics have been preserved as the primary accident analysis results. Likewise, accident frequencies have been identified qualitatively, to provide a perspective on risk that does not imply an unjustified level of accuracy.

D.1.2 Safety Strategy

The VTR is being designed with the concept of ensuring safety throughout the proposed operating regions as well as being resilient to potential accidents or upsets whether they are caused by internal hazards (such as human errors, equipment failures, or fires) or external hazards (such as seismic events, vehicle impacts, or wind loading). Consistent with U.S. Department of Energy (DOE) guidance for safety in design, the VTR design focus is on providing capabilities that reduce or eliminate hazards, a bias towards preventive, as opposed to mitigative, design features. The design also demonstrates a preference for passive systems over active systems. This general approach creates a design which is reliable, resilient to upset, and has low potential consequences of accidents.

Safe operation of the VTR depends on reliable systems designed to ensure the key reactor safety functions are achieved. These key safety functions can be summarized as (1) reactivity control, (2) fission and decay heat removal, (3) protection of engineered fission product boundaries, and (4) shielding. The first three of these safety functions are generally relevant for safe VTR operations, both within the reactor and outside the reactor. Given the anticipated distances between the VTR candidate sites and the public boundaries, the shielding function for VTR facilities and activities is primarily only of concern for involved workers. However, shielding is also a potential concern for noninvolved workers and the public during irradiated material transports.

For VTR fuel production facilities, spent fuel storage and treatment facilities, and post-irradiation examination facilities, including the hot cell and glovebox facilities like those evaluated in this EIS, the general safety strategy requires the following:

- Confinement of fuel and fission product materials at all times that prevents the materials from reaching the environment.
- Minimization of energy sources that are large enough to disperse the plutonium and fission products and threaten confinement.

This basic strategy means that operational accidents, including spills, impacts, fires, and operator errors, never have sufficient energy available to challenge the confinement. The final layer of confinement is the system of barriers and multiple stages of high-efficiency particulate air (HEPA) filters that limit the amount of material that could be released to the environment.

The operational events that present the greatest threats to confinement are large-scale internal fires that, if they were to occur, could present heat and smoke loads that threaten the building's HEPA filter systems. For modern plutonium facilities,[2] the safety strategy is (1) to prevent large internal fires by limiting energy sources, such as flammable gases and other combustible materials, to the point that a wide-scale, propagating fire is not physically possible and (2) to defeat smaller internal fires with fire-suppression systems.

Modern hot cell and glovebox facilities for plutonium and fission product operations are designed and operated such that the estimated frequency of any large fire within the facility would fall into the extremely unlikely category and would require multiple violations of safety procedures to introduce sufficient flammable materials into the facility to support such a fire. Any postulated large-scale fire in a modern plutonium facility that would be expected to result in severe consequences if it occurred would be categorized as a "beyond-design-basis" event and would fall into the "beyond extremely unlikely" category.

Earthquakes present the greatest design challenges for these facilities due to the requirement to prevent substantial releases of radioactive materials to the environment during and after an earthquake. For safety analysis purposes, occurrence of a substantial release of radioactive material within the facility is often assumed in response to a postulated earthquake that exceeds the design loading levels of the facility equipment, enclosures, and building structure and confinement. This assumption allows designers and safety analysts to determine that design features are adequate to ensure confinement of the radioactive material at risk (MAR) for any seismic event up to and through the design-basis earthquake.

D.1.3 Selection of Accidents and Determination of Radiological Source Terms

Potential accident scenarios have been identified for the Idaho National Laboratory (INL) Materials and Fuels Complex (MFC), the Oak Ridge National Laboratory (ORNL) and the Savannah River Site (SRS) facilities. Alternatives for siting the VTR include locations at the INL MFC and ORNL. Options for siting the VTR fuel production facility include locations at the INL MFC and SRS. Alternatives for siting the post-irradiation examination facility and the spent fuel treatment and storage pad include locations at the INL MFC and ORNL.

The analysis of accidents is based on calculations relevant to hypothetical sequences of events and models of their effects. The analysis in this appendix includes accident scenarios that address duration of release, elevation of release, MAR, source terms (quantities of radioactive and hazardous materials released to the environment), and consequences. The models provide estimates of the frequencies, source terms, pathways for dispersion, exposures, and effects on human health and the environment that are as realistic as possible within the scope of the analysis. In many cases, minimal experience with the postulated accidents leads to uncertainty in the calculation of their consequences and frequencies. This fact has prompted the use of models or input values that yield conservative estimates of consequence and frequency. However, given the preliminary nature of the designs under consideration, quantitatively assessing the frequency of occurrence of the events addressed is not possible. Consequently, the frequency of occurrence is qualitatively assessed, and the event frequency is assigned to a bin.

In this analysis, four frequency bins/categories are defined. The frequency bins are selected based on DOE guidance for safety analyses and National Environmental Policy Act (NEPA) documents for facilities with similar operations. Accident frequencies are grouped into the bins of "anticipated," "unlikely,"

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[2] Because the VTR fuel would be 20 percent plutonium, the facilities for feedstock preparation, fuel fabrication, and spent fuel treatment would be plutonium facilities.

"extremely unlikely," and "beyond extremely unlikely," with estimated frequencies of greater than 1×10^{-2}, 1×10^{-2} to 1×10^{-4}, 1×10^{-4} to 1×10^{-6}, and less than 1×10^{-6} per year, respectively. In this EIS, the frequency estimate includes both the initiating event and conditional events/conditions leading to the release. For example, an aircraft crash includes not only the frequency of an aircraft impacting a facility, but also the probability of the containment being breached and system damage resulting in core damage. The accident analysis considers accident scenarios that represent the spectrum of reasonably foreseeable accidents, including low-frequency/high-consequence accidents and higher frequency/(usually) lower consequence accidents. Typically, accidents with a frequency of less than 10^{-7} per year are not considered reasonably foreseeable and do not need to be examined. However, because of the effectiveness of advanced reactor safety systems, low-frequency/high-consequence accidents for the VTR typically have a frequency of less than 10^{-7} per year. Consequently, accidents with a frequency of less than 10^{-7} per year are considered in this EIS in order to provide insight into accidents with greater impacts (DOE 2008). All alternatives have been evaluated using uniform methods and data, allowing for a fair comparison of all alternatives (DOE 2004b).

D.1.3.1 Accident Analysis Background

All of the analyzed accidents involve either a release of respirable radioactive material (that is plutonium, uranium, fission products, and activation products as particles and gases) or exposure to direct gamma and neutron radiation. To a lesser extent, accidents may involve the release of fission products from a nuclear criticality. With the exception of uranium and sodium, the quantities of hazardous materials to be handled are small relative to those of many industrial facilities. Consequently, no major chemical accidents are identified, but uranium and sodium are evaluated in terms of hazardous material releases.

For each accident category, a conservative preliminary assessment of consequence is made and, where consequences are significant, one or more bounding accident scenarios are postulated. The building confinement and fire-suppression systems would be adequate to reduce the risks of most spills and minor fires. The systems would be designed to prevent, to the extent practicable, larger fires and explosions. Great efforts have always been made to prevent inadvertent nuclear criticalities, which have the potential to kill workers in the immediate vicinity. In all cases, implementation of a Criticality Safety Program and standard practices are expected to keep the frequency of accidental nuclear criticalities less than or equal to extremely unlikely.

D.1.3.2 Considerations for Accident Scenarios and Frequencies

A range of design-basis and beyond-design-basis accident scenarios has been identified for the VTR and support facilities. For each technology, the process-related accidents possible during operation of the facility have been evaluated to ensure that either their consequences are small or their frequency of occurrence is extremely low. Design features and operating practices would limit the extent of any accidents and mitigate the consequences for the workers, public, and environment. The general categories of process-related accidents considered include the following:

- Drops or spills of materials within and outside the gloveboxes
- Fires involving process equipment or materials, as well as room or building fires
- Explosions initiated by the process equipment or materials or by conditions or events external to the process
- Nuclear criticalities.

The proposed VTR facilities are being designed to meet or exceed the requirements of *Facility Safety,* DOE Order 420.1B or 420.1C (as applicable) (DOE 2005, 2012a). Design-basis and beyond-design-basis natural-phenomenon-initiated accidents are considered for VTR facilities. In accordance with DOE orders and standards, the design of DOE nuclear facilities is based upon the potential consequences of failure due to natural phenomena initiated events. Specifically for the VTR, the critical plant systems for ensuring reactor safety are designed to withstand the highest classifications of natural phenomenon demands. Seismic events are anticipated to impose the greatest demand on the VTR facilities. Because of potential consequences associated with facilities containing plutonium, the facilities are of robust construction.

Because buildings that contain plutonium are of robust construction, few events outside the buildings would have sufficient energy to threaten the building's confinement. The principal concern for events initiated outside the building would be a collision between a large commercial or military aircraft and the facility.

Several facilities evaluated in this VTR EIS have had documented safety analyses (DSAs) prepared. A central focus of the DSA process is to demonstrate that sufficient safety controls are in place, as opposed to quantifying an absolute value of risk. In general, DSAs do not attempt to establish best estimates of the probabilities or consequences of potential accidents. Consistent with their purpose, source terms and other assumptions used for bounding DSA frequency and consequence estimates are conservative. In other words, the DSA process accounts for the inherent uncertainties associated with quantifying risk by requiring that conservative assumptions be made to ensure that the final safety control set is comprehensive and adequate. In contrast, the goal of the accident analysis in this VTR EIS is to present consistent estimates of accident risks between facilities so that fair comparisons can be made among alternatives. If the accident risks between facilities or alternatives are based on differing levels of conservatism, balanced comparisons are not possible.

D.1.3.3 Identification of Material at Risk

For each accident scenario, the radioactive MAR is identified. MAR has a wide range of chemical and isotopic forms. The isotopic composition of MAR will vary, depending on the feed source. Low-enriched uranium (LEU) is also present. For the accidents considered in this VTR EIS, the contribution to dose from LEU releases is a second-order effect when released in conjunction with plutonium. The susceptibility of material to be released under accident conditions generally depends on the form of that material, the degree and robustness of confinement, and the energetics of the potential accident scenario (DOE 1999). For example, plutonium stored in strong, tight storage containers is not generally vulnerable to simple drops or spills, but may be vulnerable in a total structural collapse, as postulated in an earthquake scenario.

<u>Plutonium-239 dose equivalents</u>: The VTR project is considering a wide range of plutonium for use in the production of VTR fuel. Potential fuel sources include excess plutonium metal and oxide stored in K Area of SRS. Plutonium metal might also be available from the excess pit plutonium, excess unirradiated reactor fuel, plutonium from the processing of foreign reactor fuel, or other sources. From an accident perspective, the key differences are the radiological dose impacts from release of a given amount of one material versus another.

Many safety analyses have adopted the strategy of using a convenient surrogate, plutonium-239 dose equivalents, in place of the actual quantities and isotopic composition of the materials. With this approach, the masses or activities of certain quantities of material, such as weapons-grade plutonium, can be expressed in terms of the amount of plutonium-239 that would result in the same radiological dose upon inhalation. Quantities of other materials, such as uranium, can also be expressed in terms of plutonium-239 dose equivalents.

The relative inhalation hazard of source material for VTR fuel production is highly dependent on the amount of each of the plutonium isotopes and on the amount of americium-241 in the mixture. Adding uranium to the fuel mixture is a second-order effect when assessing the inhalation hazard and, therefore, is not considered. Plutonium-241 in the fuel mixture is less of an inhalation hazard than americium-241, but over time, plutonium-241 decays into americium-241 thereby creating an increased inhalation hazard.

To account for the relative inhalation hazard of fuel mixtures, **Table D–1** summarizes the isotopic composition and plutonium-239 dose equivalency of several possible mixtures of plutonium source materials for production of VTR fuel. Dose calculations for accidents evaluated in this VTR EIS use SRS K Area Material Storage Area/K Area Interim Surveillance (KIS)-type plutonium (87.8 percent plutonium-239 and 6.25 percent americium-241) as the reference fuel material. As shown in Table D–1 for the SRS KIS fuel mix, inhalation of 1 gram of the KIS fuel mixture would give the same dose as inhalation of 9.83 grams of plutonium-239.

Table D–1. Potential VTR Plutonium Feed Materials and Plutonium-239 Dose Equivalency

Isotope	***Mid-Range Reactor-Grade Mix*** [a]	***Bounding Reactor-Grade Mix*** [a]	***Mid-Range Weapons-Grade Mix*** [a]	***Bounding Weapons-Grade Mix*** [a]	***SPD SEIS SRS KIS Mix*** [b]	***Mid-Range Non-Weapons-Grade*** [a]	***Bounding Non-Weapons-Grade*** [a]
	Weight Fraction						
Pu-238	$2.50×10^{-3}$	$2.10×10^{-2}$	$3.00×10^{-4}$	$5.00×10^{-4}$	$4.00×10^{-4}$	$1.90×10^{-3}$	$9.00×10^{-3}$
Pu-239	$6.94×10^{-1}$	$6.22×10^{-1}$	$9.38×10^{-1}$	$9.29×10^{-1}$	$8.78×10^{-1}$	$8.36×10^{-1}$	$8.30×10^{-1}$
Pu-240	$2.70×10^{-1}$	$2.51×10^{-1}$	$6.01×10^{-2}$	$6.00×10^{-2}$	$1.15×10^{-1}$	$1.48×10^{-1}$	$1.47×10^{-1}$
Pu-241	$8.60×10^{-3}$	$4.00×10^{-2}$	$1.40×10^{-3}$	$1.00×10^{-2}$	$3.70×10^{-3}$	$7.90×10^{-3}$	$7.80×10^{-3}$
Pu-242	$2.54×10^{-2}$	$6.60×10^{-2}$	$4.00×10^{-4}$	$1.00×10^{-3}$	$2.60×10^{-3}$	$6.10×10^{-3}$	$6.10×10^{-3}$
Am-241	$4.68×10^{-2}$	$8.40×10^{-2}$	$4.50×10^{-3}$	$7.00×10^{-3}$	$6.25×10^{-2}$	$2.30×10^{-2}$	$6.50×10^{-2}$
Sum:	1.05	1.08	1.00	1.01	1.06	1.02	1.07
Pu-239 Dose Equivalent (g Pu-239/g mix) [c]	**8.83**	**18.6**	**1.90**	**2.45**	**9.83**	**5.23**	**12.5**
Ratio: Pu-239 Dose Equivalent Mixture/SRS KIS Mix	**0.90**	**1.89**	**0.19**	**0.25**	**1.00**	**0.53**	**1.27**

Am = americium; g = grams; INL = Idaho National Laboratory; KIS = K Area Interim Surveillance; Pu = plutonium; SPD SEIS = Surplus Plutonium Disposition Supplemental EIS (DOE 2015); SRS = Savannah River Site; VTR = Versatile Test Reactor.

[a] SRNL-TR-2020-00171 (SRNL 2020), sum is greater than 1 to bound, from a dose perspective, the mixtures of plutonium.

[b] DOE/NNSA 2012, NEPA Source Document – sum is greater than 1 to bound the K Area mixture of plutonium.

[c] Plutonium-239 dose equivalency based on Federal Guidance Report 13 (EPA 2002).

Hazardous chemicals: The occupational risks of using hazardous chemicals at the VTR and associated facilities are generally limited to material handling and are managed under the required industrial hygiene program. With the exception of uranium and sodium, the quantities of hazardous materials to be used for the VTR project are small relative to those of many industrial facilities. Consequently, no major chemical accidents are identified, but uranium and sodium are evaluated in terms of hazardous material releases.

D.1.3.4 Identification of Material Potentially Released to the Environment

Material must be aerosolized in order to be released to the environment. The amount and particle size of material aerosolized in an accident generally depends on the form of that material and the energetics of the potential accident scenario. Once the material is aerosolized, it must bypass the confinement and filtration systems in order to result in a release to the environment. The types and amounts of radioactive or hazardous material released to the environment following an accident are referred to as the source term.

The standard DOE formula is used to estimate the source term for each accident at each of the proposed VTR facilities:

$$\text{Source Term} = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF}$$

where:

MAR = material at risk (curies or grams)
DR = damage ratio
ARF = airborne release fraction
RF = respirable fraction[3]
LPF = leak path factor

MAR is the amount of radionuclides (measured in curies or grams of each radionuclide) available for release when acted upon by a given physical stress or accident. MAR is specific to a given process and potential accident sequence in the facility of interest. It is not necessarily the total quantity of material present; rather, it is that amount of material in the scenario of interest postulated to be available for release.

The damage ratio (DR) is the fraction of MAR exposed to the effects of the energy, force, or stress generated by the postulated event. For the accident scenarios discussed in this analysis, the value of the DR varies depending on the details of the accident scenario but can range up to 1.0.

The airborne release fraction (ARF) is the fraction of material that becomes airborne due to the accident. The respirable fraction (RF) is the fraction of the material with a particulate aerodynamic diameter less than or equal to 10 microns that could be retained in the respiratory system following inhalation. The value of each of these factors depends on the details of the specific accident scenario postulated. ARFs and RFs are estimated according to reference material in *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE 1994).

The leak path factor (LPF) accounts for the action of removal mechanisms (e.g., confinement systems, filtration, and deposition) to reduce the amount of airborne radioactivity ultimately released to occupied spaces in the facility or the environment.

Accident scenarios are identified that result in an airborne release of material. No accident scenarios are identified that would result in a substantial release of plutonium or other radionuclides via liquid pathways. Effects of fission products released during a nuclear criticality accident have been extensively studied. The principal concern is ingestion of iodine-131 via milk that becomes contaminated due to milk cows ingesting contaminated feed. This pathway can be controlled and, in terms of the effects of an accidental criticality, doses from this pathway would be small.

Since it is still early in the VTR design phase, detailed emergency response planning has not occurred. For purposes of the EIS, the radiological impacts are conservatively estimated with the assumption that no emergency planning mitigation measures are taken into account. Thus, the impact estimates do not model sheltering in place, evacuation, and similar measures typically implemented around power reactors. Similarly, for purposes of the EIS, no interdiction or mitigation is assumed to prevent or mitigate long-term exposures to contaminated areas, and foods. But in the event of a large release, such measures would likely occur. Total postulated radiological impact reported includes both the near-term and long-term impacts without mitigation. This conservative approach may result in overestimating health effects within an exposed population.

[3] Respirable fractions are not applied in the assessment of doses based on non-inhalation pathways, such as criticality.

D.1.4 Evaluation of Accident Consequences

D.1.4.1 Potential Receptors

For each potential accident, consequences (in doses) and frequencies are provided for three receptors: (1) a noninvolved worker, (2) the MEI, and (3) the offsite population. The doses are calculated for mean meteorological conditions. Meteorology specific to INL/MFC, ORNL, and SRS is used in the evaluation. Site-specific meteorological data is in the form of hourly readings for wind speed, wind direction, stability class, and rain rate. Dispersion of the materials is highly dependent on the release location, specific weather conditions at the time, and the effects of buildings and other obstructions to enhance or reduce the concentrations.

In practice, there is no way to predict realistically what the actual exposures to the receptors might be. Hence, the DOE safety process uses a prescribed method to calculate impacts and establish safety controls that limit potential radiological impacts on the receptors. Consequences for receptors as a result of plume passage are determined without regard for emergency response measures and, thus, are more conservative than would be expected if evacuation, sheltering, or other measures to reduce or prevent impacts were explicitly modeled. For purposes of this VTR EIS, the hypothetical receptors are assumed to be unaware of the accident and to remain in the plume for the entire passage with no emergency actions taken for protection. Assuming exposure to the entire release maximizes the predicted impacts. If the release of materials from an accident were through a stack, the release would likely be HEPA filtered, thereby reducing the radiological impacts on the receptors. If the release were to occur through a tall stack, the highest exposures could occur several times the stack height downwind. Most of the accidents evaluated for the VTR and support facility operations at INL/MFC and ORNL and fuel production operations at SRS assume failed containment and ground-level releases when evaluating the impacts on the hypothetical receptors. Deviations from the failed containment and ground-level release assumptions are identified in the specific accident evaluations. Given all of these factors, the reported impacts on the receptors are not necessarily realistic, but they are conservative, and application of common assumptions allows a fair comparison among the alternatives.

The first receptor, a noninvolved worker, is a hypothetical individual working on site, but not involved in the proposed activity. The noninvolved worker is assumed to be 330 feet downwind from the accidental releases at the INL/MFC, ORNL, and SRS sites. The evaluation of the potential consequences to a noninvolved worker give some measure of the potential consequences to an onsite worker who might not hear the emergency alarms and take proper protective responses as training would dictate. Realistically, evaluating the potential impacts on a noninvolved worker is highly uncertain. At most DOE sites, including the sites for the proposed VTR and supporting facilities, all workers within the operations area would likely be warned of a potentially significant radiological incident and take proper precautions and response, like taking shelter or evacuating.

The second receptor, an MEI, is a hypothetical individual assumed to be at a location along the site boundary or at a point within the site boundary where a member of the public has unrestricted access. There, this person would be exposed to the entire release and receive the largest dose. Exposures received by this individual are intended to represent the highest potential dose to a member of the public. For the VTR at INL/MFC, the assumed MEI location is on U.S. Highway 20 that runs through the site. The distance from the proposed VTR site to the road is about 3.1 miles. For the VTR at ORNL, the specific location of the VTR and its support facilities is uncertain but the proposed MEI location is about 0.5 miles east of the proposed VTR site on a publicly accessible location on Melton Hill Reservoir. For fuel production operations at the SRS K Area Complex, the MEI is 5.5 miles from the facility.

The third receptor, the offsite population, comprises all members of the public within 50 miles of the accident location. Population projections to 2050 are used to evaluate the population dose for each accident scenario and site. The general public living within a 50-mile radius of the facility and residing directly downwind of the accident receive the maximum exposure via inhalation, ingestion, air immersion, and ground-surface pathways.

For workers directly involved in the processes under consideration, consequences are addressed qualitatively rather than quantitatively. The uncertainties involved in quantifying accident consequences for the involved worker become overwhelming for most radiological accidents due to the high sensitivity of dose values to assumptions about the details of the release and the location and behavior of the affected worker.

D.1.4.2 Modeling of Dispersion of Releases to the Environment

The MACCS2 computer program (WinMACCS, Version 3.11.2) is used to calculate radiation doses and health risks to the noninvolved worker, the maximally exposed offsite individual, and the population within 50 miles of the release point (SNL 2007). SecPop (Sector Population), a U.S. Nuclear Regulatory Commission (NRC) computer program, provides estimates of population, land use, and economic values related to a specific site. It creates a site file that is needed by MACCS2 to perform a site-specific offsite consequence analysis of the health, economic, and environmental impacts of a hypothetical, atmospheric release of radioactive material from a nuclear facility (NRC 2019b). The population for the year 2050 within 50 miles of the approximate midpoint of proposed VTR operations at either the INL/MFC site or ORNL site is estimated using the latest census and past population growth information. A detailed description of the MACCS2 model is available in two NRC documents: *MELCOR Accident Consequence Code System* (NRC 1990) and *Code Manual for MACCS2* (NRC 1998). Originally developed to model the radiological consequences of nuclear reactor accidents, this computer code has been used for the analysis of accidents in many EISs and other safety documentation. A detailed description of the SecPop model is available in NRC's *Sector Population, Land Fraction, and Economic Estimation Program* (NRC 2019b). These computer programs are considered applicable to the analysis of accidents associated with the VTR.

MACCS2 models the consequences of an accident that releases a plume of radioactive materials into the atmosphere. Specifically, it models the degree of dispersion versus distance as a function of historical wind direction, speed, and atmospheric conditions. Were such an accidental release to occur, the radioactive gases and aerosols in the plume would be transported by the prevailing wind, dispersed in the atmosphere, and would expose the population to radiation. MACCS2 generates the downwind doses at specified distances, as well as the population doses out to 50 miles.

To calculate population doses, the region around the facility is divided by a polar-coordinate grid, centered on the facility itself. The user specifies the number of radial divisions and their endpoint distances. The angular divisions used to define the spatial grid correspond to the 16 directions of the compass. Dose distributions are calculated in a probabilistic manner. Releases during each of the 8,760 hours of a year are simulated, resulting in a distribution of dose reflecting variations in weather conditions at the time of the postulated accidental release. The code outputs the conditional probability of exceeding an individual or population dose as a function of distance. The mean consequences are analyzed in this VTR EIS.

The standard MACCS2 dose library is used. This library is based on *Cancer Risk Coefficients for Environmental Exposure to Radionuclides:* Federal Guidance Report 13 (EPA 1999) inhalation dose conversion factors. For exposure to radioactive heavy metal, such as in VTR fuel, the dominant pathway for exposure is inhalation of micron-sized, respirable particles. Absorption through the skin is not a significant pathway for plutonium dose. Overall, the values reported in this EIS are both conservative and internally consistent. The uncertainties in the estimated source terms far outweigh the differences in the modeling and dose conversion factor models used in this EIS.

A key factor in predicting concentration of radioactive particles downwind of a release is the "dry deposition velocity," which is a measure of how fast particles settle to the ground due to gravity. The phenomenon has been well studied and a key factor in the effect of gravity is the size of the particles released. If fine particles (micron or pollen-sized) are released, the settling or deposition velocity is low, and the particles can remain airborne for an extended time. For larger particles, in the range of tens of microns, many of the particles would reach the ground within short distances from the release point. For most types of accidental releases that might occur with the VTR and support facilities, the size of particles that might be released is unknown and speculative. For purposes of this release, a dry deposition of 0.1 centimeters per second is adopted. Using a low value like this likely over predicts the 50-mile population dose, but it is realistic for the MEI dose. This dry deposition velocity is consistent with DOE guidance and recommendations for unmitigated and unfiltered particulate releases for safety analyses using MACCS2 (DOE 2004a). An even more conservative deposition velocity used in some previous DOE EISs of 0 centimeters per second (no deposition) results in 50-mile population doses about 10 percent higher than the 0.1 centimeters per second value used in this EIS.

MACCS2 has the capability of modeling the potential near-term impacts from the initial plume passage and the long-term impacts of the radionuclides remaining after the plume passage. In MACCS2 terms, the near-term phase is called the "early" phase. Because power reactor accidents have the potential for very high radiological impacts during this initial plume phase, and major releases can be delayed for hours or days after the initial event, MACCS2 allows for complicated modeling of the mitigating effects of evacuation of persons within the emergency planning zones around power reactors. For this EIS, the MACCS2 modeling assumptions used for evaluation of initial plume passage or early doses are the default assumptions provided in the MACCS2 sample problem "NRC Point Estimates Linear No-Threshold (LNT)" which evaluates potential power reactor accidents. The MACCS2 LNT input file implements the standard linear, no-threshold, hypothesis model. In the linear, no-threshold model, the relationship between dose and health effect is linear, even for infinitesimal doses. Specification of wet and dry deposition parameters in the input file also serves to achieve conservative calculation results. These assumptions should be conservative and over-predict the initial consequences. Emergency actions, such as evacuation, which could lower the potential public consequences are not modeled.

MACCS2 also allows for the detailed modeling of potential long-term (over decades) radiological impacts from the radiological materials that remain in the surrounding area after initial plume passage. In MACCS2, this long term is called the "chronic" phase and includes the combined effects of direct exposure (inhalation) to residual particulates, ingestion of contaminated foods, radiation exposure from residual material on the ground (ground shine), inhalation of disturbed, residual ground-level particulates (resuspension), and ingestion of contaminated water, as applicable. This modeling approach has evolved primarily to evaluate the potential long-term impacts of nuclear power plant accidents, such as Chernobyl and Fukushima. Due to the nature of the problem of modeling long-term consequences, there are great uncertainties and many site-specific differences that affect many of the calculations. Many of the modeling assumptions require detailed, site-specific understanding of local food growth and consumption patterns to estimate food and water dose; local soil conditions to predict resuspension; and an array of variables to predict chronic direct and ground-shine exposures. For these reasons, the estimates of long-term doses or chronic doses should be viewed with caution and the recognition that they are highly uncertain. For this EIS, the MACCS2 modeling assumptions used for evaluation of chronic doses are the default assumptions provided in the MACCS2 sample problem "NRC Point Estimates LNT," which evaluates potential power reactor accidents. These assumptions should be conservative and over-predict the long-term consequences.

As implemented in this EIS for accidents at VTR facilities, the MACCS2 model evaluates doses due to inhalation of aerosols containing respirable radionuclides and to direct exposure from radionuclides in the passing plume. This model represents the major portion of the dose that a noninvolved worker or member of the public would receive from a VTR or support facility accident. The long-term effects from exposure to radionuclides deposited on the ground and surface waters, from resuspension and inhalation of radionuclides, and from ingestion of contaminated crops also are modeled. These long-term pathways have been studied and found not to contribute as significantly to dose as inhalation, and they would be controllable through interdiction. For purposes of this EIS, both the near-term (early) and long-term (chronic) impacts are reported.

Long-term effects of fission products released during a nuclear criticality, reactor accidents, and accidents involving used or spent fuel after removal from a reactor have been extensively studied. For reactor accidents, ingestion of contaminated food can be the major pathway for long-term exposure. For accidents involving reactor fuel, ingestion of isotopes of cesium, strontium, and iodine can result in exposures much larger than the initial exposures during plume passage. As such, mitigation measures are often planned for power reactors to reduce the potential public impacts. The modeling results in this EIS report the long-term doses to the population within 50 miles.

Radiological consequences may vary somewhat because of variations in the duration of release. For longer releases, there is a greater chance of plume meander caused by variations in wind direction over the duration of release. MACCS2 models plume meander by increasing the lateral dispersion coefficient of the plume for longer release durations, thus lowering the centerline dose reported by MACCS2. For perspective, centerline doses from a homogenous 1-hour release would be 30 percent lower than those from a 10-minute release because of plume meander. Centerline doses from a 2-hour release would be 46 percent lower. The other effect of longer release durations is involvement of a greater variety of meteorological conditions in a given release, which reduces the variance of the resulting dose distributions. This would tend to lower high-percentile doses, raise low-percentile doses, and have no effect on the mean dose.

A duration of 10 minutes is assumed for all VTR facility accident releases. This is consistent with the accident phenomenology expected for all scenarios, with the possible exception of fire. Depending on the circumstances, the time between fire ignition and extinction may be considerably longer, particularly for the larger beyond-design-basis fires. However, even in a fire of long duration, it is possible to release substantial fractions of the total radiological source term in short periods as the fire consumes areas of high MAR concentrations. The assumption of a 10-minute release duration for fire is intended to represent this circumstance.

D.1.4.3 Long-Term Consequences of Releases

The probability coefficients for determining the likelihood of fatal cancer, given a dose, are taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991) and DOE guidance (DOE 2004b). For low doses or low dose rates, probability coefficients of 6.0×10^{-4} fatal cancers per rem and person-rem are applied for workers and the general public, respectively (DOE 2003). For cases where the individual dose would be equal to or greater than 20 rem, the LCF risk is doubled (NCRP 1993). For severe accidents where members of the public could receive more than 20 rem, the total and early cancers from the MACCS2 calculations are reported. Additional information about radiation and its effects on humans is provided in Appendix C.

D.2 Considerations for Accident Selection

D.2.1 Background

High-energy events would be expected to damage some of the confinement barriers provided in the facility design and could remove their ability to provide mitigation of potential releases. Medium-energy events would be expected to reduce the effectiveness of the barriers but would not be expected to defeat them. However, low-energy events would have almost no impact on the ability of the confinement barriers to perform their function. A review of the accident scenarios indicated that only severe accident conditions (e.g., accidents involving confinement failure) could result in a significant release of radioactive material to the environment or an increase in radiation levels. These severe accident conditions are associated with beyond-design-basis events, combinations of events for which the facility is not specifically designed. While these events would be expected to have consequences larger than those associated with design-basis events, their frequency would be expected to be much lower than the design-basis event frequency. Natural phenomena (e.g., earthquake) events and fire accidents creating a direct path for releases to the environment represent the situations with the highest consequences to the public. Some types of events, such as procedure violations and spills of small quantities of material containing radioactive particles and most other types of accidents caused by human error, occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material to result in a significant release to the environment, although the impact on operational personnel may be significant. The airborne particles from a process-related accident would normally pass through at least one bank and possibly two to four banks of HEPA filters before entering the environment. Spent nuclear fuel handling operations are performed inside confinement barriers, such as hot cells or canyon walls. The hot cells are equipped with significant safety features, such as flow and pressure control and in some cases, an inert gas atmosphere. These features are modeled in the analysis when their operability is not compromised by the sequence of events associated with the accident progression. While severe accidents (also referred to as beyond-design-basis events) are expected to have the most significant impact, that is, the highest consequences on the population, these accidents may not have as significant a risk impact on all receptors as higher-frequency, lower-consequence accidents. For this reason, higher-frequency accident scenarios are included in the accident analysis.

D.2.2 Accident Scenario Consistency

In preparing the accident analysis for this VTR EIS, the primary objective is to ensure consistency in the accident analyses so that the results of the analyses for the proposed VTR and supporting facility alternatives and options impacts can be fairly compared. The accidents selected for analysis provide a consistent basis for comparing the alternatives and options in this EIS. The following sections discuss various accident categories.

Aircraft crash. Frequencies of an aircraft crash into each facility evaluated in this VTR EIS under each alternative are developed in accordance with the *Accident Analysis for Aircraft Crash into Hazardous Facilities* (DOE 2006). External events, such as the crash of a large aircraft into a VTR facility structure with an ensuing fuel-fed fire, are conceivable. At most locations away from major airports, however, the likelihood of a large aircraft crash is less than 1 in 10 million per year (1×10^{-7} per year). In most cases, the building is considered to provide sufficient structural strength and shielding such that a release of radioactive material would not be likely.

Inadvertent Nuclear Criticality. The source term for an inadvertent nuclear criticality is based on a fission yield of 1.0×10^{19} fissions, which is used for all facilities analyzed in this VTR EIS. The source term is based on that given in *DOE Handbook: Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE-HDBK-3010-94) (DOE 1994). The estimated frequency of an inadvertent nuclear criticality is often assigned to the extremely unlikely bin to be conservative.

Design-basis earthquake. All of the existing and proposed facilities that are considered in this VTR EIS would be expected to have seismic evaluations demonstrating that they meet the seismic evaluation requirements for a design-basis earthquake.

Beyond-design-basis earthquake. All of the proposed operations would be in either existing or new facilities that are designed to meet or exceed the requirements of DOE Order 420.1B or 420.1C (DOE 2005, 2012a), as applicable. Facilities would also meet applicable requirements of DOE-STD-1020, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* (DOE 2002, 2012b, 2016) for reducing the risks associated with natural phenomenon hazards. The frequency of beyond-design-basis earthquakes for all facilities may be reported in this VTR EIS as extremely unlikely to beyond extremely unlikely. The proposed facilities would be characterized as Natural Phenomena Hazard (NPH)

Design Category 3 (NDC-3) or Performance Category 3 (PC-3) for fuel production and VTR experiment hall facilities and NDC-5 for the VTR reactor and supporting safety class systems.[4]

The numerical seismic design requirements detailed in DOE-STD-1020 are structured such that there is assurance that specific performance goals would be met. For PC-3, NDC-3, and NDC-5 designated facilities, the performance goal is to ensure occupant safety and hazard confinement for earthquakes with an annual probability of occurrence exceeding approximately 4×10^{-4}, 1×10^{-4}, or 1×10^{-5}, respectively. There is sufficient conservatism in the design of the buildings and the structures, systems, and components that are important to safety that this goal should be met, given that they are designed to withstand earthquakes with an estimated mean annual probability of 4×10^{-4}, 1×10^{-4}, or 1×10^{-5}.

By contrast, nonnuclear structures at these sites and the surrounding community would be constructed to the regional standards of the *Uniform Building Code* or *International Building Code* at the time of construction. Specifically, peak acceleration values are 50 to 82 percent of the design requirements for nuclear facilities in the same area and correspond approximately to DOE PC-1 or PC-2 facilities with 500-year return intervals. During major earthquakes, structures built to these *Uniform Building Code* or *International Building Code* requirements are expected to suffer significantly more damage than reinforced structures designed for nuclear operations.

For the VTR, a site-specific probabilistic seismic hazard analysis and probabilistic risk assessment will be performed to characterize the site for potential seismically-induced ground motions with ground motion magnitude and frequency estimates for return periods up to 100,000 years. The details of potential earthquakes with return periods in this range is uncertain and uncertainty bounds and estimates are accounted for in the definition of the seismic hazard, which governs the system design and analyses. Even though details of seismic events with return periods greater than 100,000 years are uncertain, some evaluations of facility margins to these events can be assessed based upon the ratio of NPH-induced system stresses to their actual design capacity. For purposes of this VTR EIS, it is assumed that, at all the candidate sites, earthquakes with return periods in the 10,000- to 10-million-year range might result in sufficient ground motion to cause major damage to the facilities classified as SDC-3 or PC-3. The level of damage necessary for failure of the SDC-5 structures would be so large that the overall infrastructure damage and consequences of such an event to the nearby area infrastructure would likely be catastrophic negating any VTR consequences. Such catastrophic damage is evidenced by the results of the 2011 Tohoku earthquake in Japan, which destroyed many homes and much infrastructure. However, the nearby reactors, which were designed to be equivalent of SDC-5, performed as expected and designed until impacted by the tsunami. Neither of the potential VTR candidate sites are subject to tsunami hazards. VTR reactor safety systems are designed to the most rigorous construction standards and will be evaluated to ensure that the annual probability of seismically-induced failure is outside the frequency of concern when considering the frequency of the events and the component fragilities of the key safety systems.

Filtration efficiency. This VTR EIS analysis assumes the exhaust from most facilities, including the existing K-Reactor and MFC facilities, as well as the proposed ORNL Post-Irradiation Examination/Spent Fuel

[4] Each structure, system, and component in a DOE facility is assigned to one of five NPH Design Category categories, or to one of four performance categories, depending on its safety importance. Structures, systems, and components designated as SDC-3 or PC-3 or higher are those for which failure to perform their safety function could pose a potential hazard to public health, safety, and the environment from release of radioactive or hazardous materials. Design considerations for this category are to limit facility damage as a result of design-basis natural phenomena events (for example, an earthquake) so that hazardous materials can be controlled and confined, occupants are protected, and the functioning of the facility is not interrupted (DOE 2002, 2012c, 2016). The return periods or other criteria under which the design and analysis needs to demonstrate confinement of radioactive or hazardous material are progressively higher as the design category goes from a 1 to a 5 and are set based upon estimates of the public and worker consequences as a result of failure caused by the specific NPH.

facility, would be directed through two stages of testable HEPA filters to a stack. A building LPF of 1.0×10^{-5} is used for particulate releases with HEPA filters unless otherwise noted (DOE 1999). Under design-basis accident conditions, the HEPA filters are assumed to remain functional but with degraded efficiency and a building LPF of 5×10^{-3} is assumed. Under more severe conditions, such as major fires, the HEPA filters are assumed for purposes of this EIS to be severely degraded. Under these severely degraded conditions, a building LPF of 1×10^{-1} is assumed. Under beyond-design-basis conditions where structural failure of a building is assumed, a building LPF of 1 is conservatively assumed.

For the hypothetical beyond-design-basis earthquake and fire accident scenarios, a consistent LPF is assumed across the facilities evaluated. In this VTR EIS, the hypothetical beyond-design-basis earthquake accidents are not based on detailed analysis, and are postulated simply to show a bounding level of impacts should the safety design and operational controls fail. For NEPA purposes, the goal is to show the impacts of realistic, physically possible events even if it is believed their probability is extremely low.

For comparison purposes, it is postulated that:

- The hypothetical beyond-design-basis accident is an earthquake that exceeds the design-basis earthquake (for example, SDC-3 for the experiment hall and SDC-5 for the VTR) by a sufficient margin that gloveboxes fail, fire suppression systems fail, power fails, and some building confinement is lost. It is further assumed that a room-wide fire or multiple local fires might occur. The overall probability of the event, considering the conditional probabilities of fires following a beyond-design-basis earthquake, is expected to be in the 1×10^{-6} to 1×10^{-7} per year range or even lower.
- For new facilities and significantly upgraded facilities, a building LPF of 0.1 could be assumed and expected to be conservative. This factor should adequately represent an LPF for cracks in the building or transport through rubble. Nevertheless, this VTR EIS assumes an LPF of 1 for these facilities even though the LPF could be several times lower than this.
- For older, existing facilities that have not been or are not planned to be upgraded, it is not generally known how they might fail in a beyond-design-basis earthquake but an LPF of 1 is considered unrealistic because even a rubble pile in a total building collapse offers some impediment to particulates being released to the environment. Nevertheless, this VTR EIS assumes an LPF of 1 for these facilities even though the LPF could be several times lower than this.
- For all facilities, an LPF of 1 is assumed for gaseous releases.

The real-world performance of multiple stages of HEPA filters has been well demonstrated and experimental testing confirms the performance of HEPA filters for uranium and plutonium particles. The independent Defense Nuclear Facilities Safety Board (DNFSB) thoroughly evaluated the use of HEPA filters by DOE and issued multiple reports on the performance of HEPA filters within the DOE complex. HEPA filters used in support of the VTR activities would conform to the latest version of DOE Standard "Specifications for HEPA Filters Used by DOE Contractors," DOE-STD-3020-2015. Performance testing required by this standard for all HEPA filters installed for safety would ensure that the filters meet or exceed the performance requirements assumed in safety evaluations.

D.2.3 Accident Scenario Differences among Alternatives and Options

The accident scenarios defined for the different alternatives and options have the same initiating events and scenario descriptions. The differences among the alternative are due to different distances to the noninvolved worker and MEI, different population distributions, different meteorological conditions, and different release heights for elevated releases. Other parameters for radionuclide inventory, release factors, release duration, wet and dry deposition, and food ingestion factors are the same in a set of accidents for the alternatives and a set of accidents for the options.

D.2.3.1 INL VTR Alternative

At INL, the VTR would be built adjacent to and east of the currently protected area at MFC. Fuel manufacturing would occur in the Fuel Manufacturing Facility (FMF) and Zero Power Physics Reactor (ZPPR) protected area at the MFC. Post-irradiation examination and spent fuel treatment would occur in the Hot Fuel Examination Facility (HFEF) and the Fuel Conditioning Facility (FCF). Spent fuel would be stored in casks on the spent fuel storage pad (Marschman et al. 2020).

D.2.3.2 ORNL VTR Alternative

At ORNL, the VTR and a new Hot Cell Building would be built on land approximately a mile east of the High Flux Irradiation Reactor (HFIR) complex. Fuel manufacturing would not be performed at ORNL. Post-irradiation examination and spent fuel treatment would occur in the Hot Cell Building. Spent fuel would be stored in casks on the spent fuel storage pad.

D.2.3.3 VTR Fuel Production Option at Savannah River Site

Under the SRS fuel production option, fuel production would be performed in the K-Reactor Building (105-K) in the K Area Complex. All equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and fuel assembly fabrication would be located on two below-ground levels within the building. An induction-casting furnace would be used in the initial steps in the fuel fabrication process, alloying the elemental metallic components and producing the fuel slugs. This furnace would be contained within a glovebox with an inert gas atmosphere. Under the SRS feedstock preparation option, this capability would be located adjacent to the location for the fuel fabrication capability, in the K-Reactor Building (105-K) in the K Area Complex.

D.2.3.4 VTR Fuel Production Option at Idaho National Laboratory

The INL Fuel Production Option includes the use of the FMF and the ZPPR to house the equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and fuel assembly fabrication. All of the equipment for feedstock preparation would be newly constructed. VTR fuel production is projected to require sample analysis for hundreds and potentially thousands of samples in the first few years of operation. INL proposes to use existing space fitted with new equipment in the FCF as an analytical chemistry laboratory to support VTR fuel production. Like the SRS option, an induction-casting furnace at INL would be used in the initial steps in the fuel fabrication process, alloying the elemental metallic components and producing the fuel slugs. This furnace would be contained within a glovebox with an inert gas atmosphere.

D.2.3.5 No Action Alternative

Under the No Action Alternative for the VTR and the No Action Option for fuel production, no new facilities would be built or modified at INL, ORNL, or SRS. Existing test reactors and support facilities would continue operating under either the no action or action alternatives and options, so there would be no incremental change to the associated impacts. Impacts of continued operation are included in those representing the existing affected environment.

D.3 Facility Accident Scenarios

D.3.1 Reactor Fuel Production Accidents at Idaho National Laboratory and Savannah River Site

As discussed in Section D.1.3.3, the VTR project is considering a wide range of plutonium for use in the production of VTR fuel. Radiological impact calculations for VTR feedstock preparation and fuel fabrication activities at SRS or INL assume that the fuel material is from the KIS mixture of plutonium oxide. This is a hypothetical mixture of plutonium developed for safety basis analysis at the K-Reactor Area and is expected to adequately represent the plutonium materials stored in K Area. From an accident perspective, the key differences among accident scenarios are the radiological dose impacts from release of a given amount of one plutonium mixture versus another. The differences in radiological impacts of one mixture of plutonium versus another are partially due to the build-up of americium-241 from the decay of plutonium-241. To show the effect of one plutonium mixture versus another, the radiological dose impacts of each potential plutonium type have been evaluated and compared to the radiological dose impacts of the KIS mixture of plutonium oxide as shown in Table D–1. Table D–1 provides a summary of potential sources of plutonium for VTR fuel production, the isotopic composition of the fuel, the plutonium-239 dose equivalency of the fuel, and the ratio of plutonium-239 dose equivalency of the fuel to the SRS KIS mixture of plutonium oxide.

The reference metallic fuel (consisting of an alloy of uranium, plutonium, and zirconium) to be used in the VTR is unique and would be fabricated at either the SRS or the INL MFC. Materials available for use in the production of the metallic fuel (feedstock) exist in several forms. Plutonium feedstock may be in the form of metals or oxides while uranium feedstock may be in metal form.

Depending on the feed material, additional processing of the plutonium may be necessary. Additional processing could include:

- Conversion of plutonium oxide to metal
- Removal of gallium from plutonium metal to improve desirability for reactor fuel
- Removal of americium-241, a significant source of radiological extremity dose to glovebox workers.

The fuel form for the fuel pin is a cast metallic cylindrical slug. Depending on the form of the feedstock, the steps needed to fabricate the fuel slug would be:

- Feed material polishing, removal of impurities;
- Conversion of feedstock from non-metallic forms to metals;
- Fuel alloying and homogenization;
- Fuel slug casting and demolding;
- Assembly of the fuel slugs into fuel pins; and
- Assembly of the fuel pins into fuel assemblies.

The fuel production facility would need to provide space for material storage. Materials to be stored include:

- Plutonium feedstock;
- Uranium feedstock;
- Zirconium feedstock;
- Fuel slugs;
- Fuel pins;
- Fuel cladding;
- Sodium;
- Fuel assemblies; and
- Scrap and waste.

Under the SRS fuel production option, fuel fabrication would be performed in the K-Reactor Building in the K Area Complex. All equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and fuel assembly fabrication would be located on two below-ground levels within the building. The fuel production facility would be located on the minus-20 and minus-40-foot levels (20 and 40 feet below grade) of the K-Reactor Building. Approximately 11,500 square feet and

12,000 square feet of space would be made available at the minus-40 and minus-20 levels, respectively. The facility could support feed material purification, ingot manufacturing, and/or the fabrication of fuel from ingots. New equipment would be provided for fuel slug casting, slug trimming and inspection, fuel rod loading and inspection, fuel bundle assembly and packaging, and waste handling. Other infrastructure to be supplied would include material storage areas, special nuclear material measurement equipment, analytical support, and other infrastructure services, such as glovebox and room ventilation and electrical distribution. Because SRS is not a proposed site for the VTR, completed assemblies (or fuel pins if assemblies are fabricated at the reactor site) would be loaded into a shielded transfer cask and then loaded into a shipping container for shipment.

The INL MFC fuel production option includes the use of the existing FMF and ZPPR to house the equipment necessary to support fuel alloying and homogenization, fuel slug casting, fuel pin assembly, and fuel assembly fabrication. Under this option, the ingots of each fuel component (uranium, plutonium, and zirconium) would be delivered to INL. The FMF has capabilities to produce and purify transuranic and enriched-uranium feedstock, to develop transuranic metallic and ceramic fuels and to store these fuels. Space for lag storage of casting scrap, and assembled fuel pins pending transfer to ZPPR would be made available in the FMF vault. Additional facilities at MFC would be required to support fuel preparation (conversion of plutonium oxide to plutonium metal, and plutonium processing ("polishing") to remove undesirable impurities).

Activities at either SRS or MFC could result in accidents involving the fuel production. Fuel production activities would be conducted in gloveboxes. Because of the design of the gloveboxes, fuel production accidents would not be expected to result in radioactive or hazardous material releases to the room. Fuel production accidents would result primarily in an elevated release of materials. The material would be released to the glovebox and then exhausted through HEPA filters and out of a stack. Accident scenarios potentially relevant to the proposed fuel production activities are discussed in the SRS and INL Data Reports (INL 2020a; SRNS 2020).

In addition to specific fuel production activities, other potential activities at the SRS and INL fuel production facilities include conversion of plutonium oxide to plutonium metal, and processing ("polishing") the plutonium oxide or metal to remove undesirable impurities, specifically gallium and americium-241. The specific polishing process that would be at SRS and INL has not been finalized at this time but an aqueous process similar to that used for the mixed oxide (MOX) plant and a pyrochemical process similar to the electrorefining process at INL are both potential processes.

The MOX plant at SRS had a front-end polishing process to remove impurities from the plutonium oxide feed material (DOE 2015; NRC 2005; ORNL 1998). The "aqueous polishing" process used in the MOX plant is described and evaluated in detail in the *Environmental Impact Statement on the Construction and Operation of a Proposed Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina* (MFFF EIS) (NRC 2005). The MOX facility was designed (at least initially) to process up to 3.5 metric tons of plutonium annually so its throughput was likely a factor of ~7 times the VTR needs. The MOX aqueous polishing process thus had about 7 times the annual throughput needed for the VTR fuel production facility but the technology and accident scenarios could be similar.

Principal accidents identified in the MFFF EIS (NRC 2005) and the DOE *Surplus Plutonium Disposition Supplemental Environmental Impact Statement* (SPD Supplemental EIS) (DOE 2015) include spills or fires involving plutonium solutions. Two relevant accident scenarios are identified in the DOE SPD Supplemental EIS (DOE 2015) for the aqueous polishing portion of the MFFF. The scenarios involved a leak of liquid organic solvent containing the maximum plutonium concentration (a MAR of 40 grams of plutonium) and a liquid spill of concentrated aqueous plutonium solution (a MAR of 5,000 grams of plutonium). In addition, ORNL 1998 identified additional bounding accidents, including a thermal excursion in an ion exchange column. The postulated thermal excursion in an ion exchange column resulted in the highest potential release, as evaluated in this EIS.

The 2015 SPD Supplemental EIS (DOE 2015) evaluated a range of potential accidents in K Area associated with handling 3013 cans of plutonium oxide. The highest consequence operational or natural phenomena-initiated accidents are all in the extremely unlikely or lower frequency range and involved over-pressurization of a DOE standard 3013 can of plutonium oxide. These containers are extremely robust and consist of two welded, nested stainless-steel containers. Because of their robust construction, they can pressurize to the point of rupture if subjected to a fire of sufficient magnitude and length. This could cause a high-pressure release of plutonium oxide. For the oxide-to-metal conversion processes, all of the operations would be criticality-safety limited, so MAR also would be limited. No accidents involving an oxide-to-metal conversion process (spills, fires, explosions, etc.) would result in a higher amount of material becoming airborne than the over-pressurization of the 3013 can of oxide.

To evaluate the possible effects of fuel production accidents, a criticality and several fires were evaluated. In addition, potential accidents associated with converting plutonium oxide to metal and "polishing" operations are discussed. Polishing operations, whether aqueous or pyrochemical, would result in a high americium-241 content waste stream. This waste stream could ultimately be converted to a solid form for disposal as high-activity TRU waste. The evaluation discusses the release factors used to determine the source term for the accident. An LPF of 1 is used to model the effects when confinement fails. Under severe accident conditions, such as fires, the HEPA filters are assumed, for purposes of this EIS, to be degraded. Under these degraded conditions, a building LPF of 0.1 is assumed to model the accident effects when confinement is intact. The evaluation also considers hazardous material releases.

D.3.1.1 Criticality while Melting Plutonium Metal and Adding Uranium and Zirconium (Fuel Received as Plutonium Metal)

An inadvertent nuclear criticality is assumed to occur in the fuel fabrication glovebox line while melting plutonium metal and adding LEU and zirconium. The criticality results from a seismic event that also causes failure of confinement. The source term for this criticality is based on a fission yield of 1.0×10^{19} fissions. This source term, which is used for all facilities, is based on that given in DOE-HDBK-3010-94 (DOE 1994). Criticality safety controls should prevent this accident from occurring and the materials involved should remain at less than a critical mass. Sufficient acceleration to cause this much damage would require an earthquake with accelerations higher than the design-basis requirements for the structure and failures of the building and equipment would be expected. Thus, this accident is identified as extremely unlikely. The scenario represents a metal criticality. The metal is postulated to soften during the process, resulting in a 100 percent release of gaseous fission products generated in the criticality. However, the scenario does not assume release of aerosolized, respirable metal fragments. Engineered and administrative controls would be available to ensure that the double-contingency principles are in place for all portions of the process. For purposes of this VTR EIS, the DR is 1, and the bounding ARF and RF value of 1 is assumed for noble gases. No radionuclide deposition during transport through the building or the accident debris is assumed, so an LPF of 1 is modeled. The plume rise option is not activated for this scenario. Because the building structure remains intact, a stack release is modeled. **Table D–2** summarizes the MAR, release fractions, and source terms for VTR fuel fabrication facilities.

D.3.1.2 Fire Impingement on Fuel Material (Intact Confinement)

In this scenario, a fire would affect the solid fuel material in a glovebox in the fuel fabrication line. The plume rise option is not activated for this scenario. Because the building structure remains intact, a stack release is modeled. The frequency of the fire is extremely unlikely. MAR is assumed to be 5,000 grams of KIS plutonium mixture. All material in the fuel fabrication line would be at risk, and the DR is 1. The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively, are based on the airborne release of plutonium particulates formed by oxidation at elevated temperatures without self-sustained oxidation (DOE 1994). Reduction in the source term due to the fuel fabrication line confinement is modeled (LPF=0.1). Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.3 Fire impingement on Fuel Material with Seismically-Induced Confinement Failure

The fire selected for analysis would affect the solid fuel material in a glovebox in the fuel fabrication line. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. This much damage would require an earthquake with accelerations substantially higher than the design-basis requirements for the structure and major failures of the building and equipment would be expected. The frequency of the earthquake is extremely unlikely to beyond extremely unlikely. MAR is assumed to be 5,000 grams of KIS plutonium mixture. All material in the fuel fabrication line would be at risk, and the DR is 1. The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively, are based on the airborne release of plutonium particulates formed by oxidation at elevated temperatures without self-sustained oxidation (DOE 1994). A building LPF of 1 is assumed, although a more realistic value is likely to be at least a factor of several lower. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure

A seismically-induced spill of molten plutonium-uranium mixture while heating or casting is assumed to occur. The spilled material is then postulated to rapidly oxidize or burn. This much damage would require an earthquake with accelerations substantially higher than the design-basis requirements for the structure and major failures of the building and equipment would be expected. The frequency of the earthquake is extremely unlikely to beyond extremely unlikely. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. For this accident, a MAR of 4,500 grams of KIS plutonium mixture, a DR of 1, an ARF of 5×10^{-4}, and an RF of 0.5 are estimated, which would result in a release of 1.1 grams to the building. A building LPF of 1.0 is assumed, although a more realistic value is likely to be at least a factor of several lower. Using this LPF would result in a release of 1.1 grams of plutonium mixture to the environment. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.5 Plutonium Oxide-to-Metal Conversion – Explosion of 3013 Container of Plutonium Oxide

The bounding mitigated explosion event identified in the 2015 SPD Supplemental EIS (DOE 2015) for K Area plutonium oxide handling activities is a postulated deflagration or detonation in the glovebox that occurs just as a 3013 container is being punctured for sampling purposes. This accident scenario is applicable to both K Area and MFC if oxide would be used as a feedstock material. The plume rise option is not activated for this scenario. Because the building structure remains intact, a stack release is modeled. The EIS indicates that the internal pressure should be within the 3013-container design rupture limit of 700 pounds per square inch (gauge) unless subjected to an external fire. For this pressure, the expected ARF × RF is 0.022. This ARF × RF yields approximately 99 grams of plutonium mixture from a drum containing 4,500 grams (88.4 percent plutonium from 5,090 grams plutonium oxide, 11.6 percent oxygen) of KIS plutonium mixture. Given a scenario where that mixture is released to the building exhaust system, the building HEPA filters would reduce the amount released to the stack. A building LPF of 0.1 is assumed for one stage of HEPA filters. Therefore, the mitigated release to the environment through the stack would be approximately 9.9 grams of plutonium mixture or 97.3 grams of plutonium-239 equivalent. A release of this magnitude would fall in the extremely unlikely category. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.6 Beyond-Design-Basis Fire Involving a Transuranic Waste Drum Fire

A beyond-design-basis fire has been postulated in the K Area Complex and for MFC that would involve an unmitigated TRU waste drum fire on the loading dock that burns with sufficient intensity and duration that all of the material in the drum is consumed. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The expected ARF x RF is 0.0005, which corresponds to approximately 0.2 grams (0.007 ounces) of plutonium from a drum containing 398 grams (14 ounces) of plutonium mixture (88.4 percent plutonium from 450 grams of plutonium oxide, 11.6 percent oxygen). Because this fire is postulated to occur outside the building, an LPF of 1 is assumed. This accident is conservatively estimated to have a frequency of extremely unlikely to beyond extremely unlikely. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.7 Aqueous/Electrorefining Fuel Preparation

Fires, spills, and explosions for the aqueous plutonium polishing are evaluated. The scenario is based on an analysis for MFFF (ORNL 1998). The following are the dominant accident scenarios identified (ORNL 1998).

Fire. It is assumed that the liquid organic solvent leaks as a spray into the glovebox, builds to a flammable concentration, and is contacted by an ignition source. A MAR of 1 liter of solvent containing 40 grams of plutonium is postulated (ORNL 1998). The combined ARF and RF value for this scenario is 1.0×10^{-2} for quiescent burning to self-extinguishment (DOE 1994; ORNL 1998). A release of 0.4 grams of plutonium to the glovebox is postulated. Under these accident conditions, a building LPF of 0.1 is assumed for one stage of HEPA filters. The frequency of this accident is in the unlikely range.

Spill. Leakage of liquids from process equipment must be considered as an anticipated event. However, with multiple confinement barriers, a release from the process room would be extremely unlikely. A bounding scenario involves a liquid spill of concentrated aqueous plutonium solution (100 grams per liter plutonium) (ORNL 1998), with 50 liters containing 5,000 grams of plutonium accumulating before the leak is stopped. The ARF and RF values used for this scenario are 2.0×10^{-4} and 0.5, respectively (DOE 1994; ORNL 1998). A release of 0.5 grams of plutonium to the glovebox is postulated. A building LPF of 5.0×10^{-3} is assumed for one stage of HEPA filters (ORNL 1998; DOE 2015).

Uncontrolled Reaction/Explosion. The highest consequence operational accident identified by ORNL for the aqueous polishing portion of MFFF is a thermal excursion in a nitrate anion exchange column (ORNL 1998). This scenario examines the potential effects of a thermal excursion within an ion exchange column while the aqueous process is operating. The thermal excursion is postulated to result from off-normal operations, degraded resin, or a glovebox fire. It is also assumed that the column venting/pressure relief fails to vent the overpressure causing the column to rupture violently. The overpressure releases plutonium nitrate solution as an aerosol within the affected glovebox that in turn is processed through the ventilation system. If the overpressure also breeches the glovebox, a fraction of the aerosol will be released within the room as well.

The total mass of KIS plutonium mixture that could be contained in an ion exchange column is 1,000 grams on the resin and 246 grams in nitrate solution. These quantities are based on the maximum intended plutonium loading of the resin and the maximum intended plutonium concentration in solution after pH-adjustment, respectively. DOE 1994 lists ARF/RF values of 9×10^{-3} for burning resin and 6×10^{-3} for liquid behaving as a flashing spray upon depressurization. ORNL 1998 assumes 10 percent of the resin is assumed to burn upon release, or a DR of 0.1. Thus, the release from the resin fire is 1,000 grams × 0.1 × 9×10^{-3} or 0.9 grams of plutonium. The release from the liquid ejected from the column because of overpressure from the flashing solution is 246 grams × 1 × 6×10^{-3} = 1.5 grams of plutonium. An electrorefining process is assumedly operating while the aqueous process is operating. In the electrorefining process, the chopped fuel is placed in anode basket(s) and a current passing between the anode(s) and cathode(s) anodically dissolves the actinides into the molten chloride salt at 1,202 degree Fahrenheit. Multiple cathodes at different electric potentials allow deposition of TRU isotopes on different cathodes. The ARF and RF values used for the release from the electrorefiner are $1×10^{-3}$ and 0.4 for heavy metal salts (DOE 1994:3-25). The total mass of KIS plutonium mixture that could be contained in the electrorefiner is 3,500 grams. A DR of 1 is assumed. The release from the electrorefiner into the glovebox is 1.4 grams of plutonium (3,500 grams × 1 × $1×10^{-3}$ × 0.4). Summing these three airborne source terms from the resin, the solution, and the electrorefiner gives a total of 3.8 grams of plutonium mixture aerosol in the glove box. A building LPF of $5.0×10^{-3}$ is assumed for one stage of HEPA filters. The plume rise option is not activated for this scenario. Because the building structure remains intact, a stack release is modeled. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.1.8 Aircraft Crash into VTR Fuel Production Facility

A crash of a large, heavy commercial or military aircraft directly into the fuel production facility might damage the structure sufficiently to breach confinement and disperse material into the environment. A subsequent fuel-fed fire could provide energy to further damage structures and equipment, aerosolize material, and drive materials into the environment. Source terms are highly speculative but could be similar to those from the beyond-design-basis earthquake if the structure is not sufficiently robust.

The probabilities of aircraft crashes into SRS facilities, including the K-Reactor Building, have been previously evaluated (Radder 2006) and reported in the K Area DSA (SRNS 2016) and the SPD Supplemental EIS (DOE 2015). The K Area DSA dismisses events initiated by a large or military aircraft because those were shown to be beyond extremely unlikely. The K Area DSA includes evaluation of small aircraft/helicopter (security) crashes since those probabilities are in the extremely unlikely range.

At MFC, the feedstock preparation activities would occur in the FCF followed by fuel fabrication within the FMF and the ZPPR. Due to the remote location of MFC, the probability is sufficiently low that impacts from aircraft crashes are not addressed in the facility safety documents. For purposes of this VTR EIS, MAR within these facilities is assumed vulnerable to a direct aircraft impact. An aircraft impact into the fuel production building with a release of the magnitude projected is considered to be beyond extremely unlikely.

The degree of damage incurred and any subsequent release of radioactive materials depends on the size and speed of the aircraft involved. MAR identified in the above scenarios is assumed to be involved and vulnerable to release due to failure of the inert gloveboxes and confinement and subsequent fires. MAR for the feedstock and finished product is assumed stored in an area that is not affected by the crash. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. MAR includes 5,000 grams of plutonium metal that oxidizes after the loss of confinement and inert atmosphere (Section D.3.1.4). MAR contains 4,500 grams of molten plutonium metal released from a casting operations fire (Section D.3.1.5) and 4,500 grams of plutonium from the 3013-container overpressure (Section D.3.1.5). It also includes 398 grams of plutonium from a TRU waste drum fire (Section D.3.1.6), plus 1,000 grams from resin, 246 grams from liquid, and 3,500 grams from an electrorefiner explosion in aqueous/electrorefining fuel preparation (Section D.3.1.7). Deposition during transport through the building or the accident debris is not assumed, so an LPF of 1 is modeled. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario.

Table D–2. Accident Scenarios and Source Terms for Idaho National Laboratory and K Area VTR Fuel Production Operations

Accident Scenario	*MAR Grams of Pu or Other*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF*	*RF*	*LPF*	*Source Term Dose Equivalent Scale Factor*	*Source Term Pu-239 Dose Equivalent (FP, grams Pu-239)*
D.3.1.1 Criticality while alloying the three components of the metal fuel, uranium, plutonium, and zirconium	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44	Extremely Unlikely	1	1.0	1	1	NA	See Table D-44
D.3.1.2 Fire Impingement on Fuel Material (Intact confinement)	5,000	KIS-grade Pu, ~ 5 kg Pu per assembly	Extremely Unlikely	1	3.0×10^{-5}	0.04	0.1	9.83	5.90×10^{-3}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	5,000	KIS-grade Pu, ~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	1	3.0×10^{-5}	0.04	1	9.83	5.90×10^{-2}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	4,500	5,090 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	1	5.0×10^{-4}	0.5	1	9.83	1.11×10^{1}
D.3.1.5 Plutonium Oxide-to-Metal Conversion– Explosion of 3013 Container of PuO_2	4,500	5,090 g KIS grade PuO_2 in one 3013 container	Extremely Unlikely	1	2.2×10^{-2}	1	0.1	9.83	97.3
D.3.1.6 Beyond-Design-Basis Fire Involving TRU Waste Drum	398	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	1	5.0×10^{-4}	1	1	9.83	1.96

Accident Scenario	MAR Grams of Pu or Other	Assumed Material Type	Frequency (per year)	DR	ARF	RF	LPF	Source Term Dose Equivalent Scale Factor	Source Term Pu-239 Dose Equivalent (FP, grams Pu-239)
D.3.1.7 Aqueous/ Electrorefining Fuel Preparation	1,000 resin 246 nitrate solution 3,500 electrorefiner	KIS-grade Pu	Extremely Unlikely	0.1 resin 1.0 liquid 1 electro-refiner	9×10^{-3} resin 6×10^{-3} liquid 1×10^{-3} electro-refiner	1 1 0.4	0.005	9.83	1.86×10^{-1}
D.3.1.8 Aircraft Crash into VTR Fuel Production Facility	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	varies	varies	varies	1	9.83	1.02×10^{3}
D.3.1.9 Beyond-Design-Basis Earthquake and Fire Involving All of VTR Fuel Production MAR	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	varies	varies	varies	1	9.83	1.02×10^{3}

ARF = airborne release fraction; DR = damage ratio; FP = fission products; g = grams; INL= Idaho National Laboratory; kg = kilograms; KIS = K Area Interim Surveillance; LPF = leak path factor; MAR = material at risk; NA = not applicable; Pu = plutonium; PuO_2 = plutonium oxide; RF = respirable fraction; TRU = transuranic; VTR = Versatile Test Reactor.

D.3.1.9 Beyond-Design-Basis Earthquake and Fire Involving All of Versatile Test Reactor Fuel Production Material at Risk

A beyond-design-basis earthquake is assumed to occur and involve all of the VTR fuel production activities. The MAR identified in the above scenarios is vulnerable to release because of confinement failure of the inert gloveboxes and subsequent fires. The feedstock and finished product are assumed to have a secondary effect on the source term because of the form of the material and are not included in the event. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. MAR includes the same material identified for the aircraft crash scenario (Section D.1.3.8). Deposition during transport through the building or the accident debris is not assumed, so an LPF of 1 is modeled. An earthquake that results in this much damage would require accelerations substantially higher than the design-basis requirements for the structure and major failures of the building and equipment would be expected. This event would be characterized as beyond extremely unlikely. Table D–2 summarizes MAR, release fractions, and source term for this accident scenario

D.3.2 Transuranic Waste Handling Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

TRU waste is waste that is contaminated with alpha emitting TRU radionuclides (radionuclides with atomic numbers greater than that of uranium) that have half-lives greater than 20 years and in which the TRU radionuclide concentrations are greater than 100 nanocuries per gram of waste. TRU waste consists of tools, rags, protective clothing, and other materials contaminated with radioactive elements, such as plutonium. TRU waste would be generated mostly from fuel production and spent fuel treatment. VTR operations and post-irradiation examination of test assemblies would generate less TRU waste than fuel production and spent fuel treatment.

D.3.2.1 Fire Outside Confinement (Waste from Fuel Production or Spent Fuel Treatment)

In this scenario, a fire is postulated to occur in a 1.2 × 1.2 × 2.4-meter (4 × 4 × 8-foot) solid TRU waste box because of spontaneous combustion, pyrophoric material, vehicle accident, electrical failure, or poor housekeeping. The accident is assumed to occur outdoors during handling. The release occurs at ground level over 10 minutes. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The assumed accident frequency is extremely unlikely. MAR is assumed to be one box of TRU waste. MAR is based on one spent fuel pin, with the nuclide distribution of a spent fuel assembly. The material is decayed for a period of four years. A fuel pin is a small item that is representative of activities in either the fuel production or spent fuel treatment facilities. The fuel pin is assumed to represent a bounding quantity of material and is considered to represent the material that would be a contaminant on the items in a TRU waste box. Even though spent fuel has some burnup and contains fission products and activation products, the material still contains a bounding quantity of material for waste from fuel production. The material in a TRU waste box would potentially consist of items, such as contaminated tools, personal protective equipment, and materials (for example rags and blotter paper) used for decontamination activities. The DR is assumed to be 0.001, based on the design of the TRU waste box. Failure of the waste box is assumed to occur because of a seal breach, crack, or puncture and not a massive rupture. The bounding ARF and RF value of 1 is assumed for noble gases. For the other radionuclides, the bounding ARF and RF values of 5×10^{-4} and 1, respectively are based on thermal stresses to packaged surface contaminated waste from DOE-HDBK-3010-94 (DOE 1994). Since the fire is outside building confinement, the LPF is 1. **Table D–3** summarizes MAR, release fractions, and source term for this accident scenario.

Table D–3. Accident Scenarios and Source Terms for the Transuranic Waste Handling Accidents at Savannah River Site, Idaho National Laboratory and Oak Ridge National Laboratory

Accident Scenario	*MAR Grams of Pu or Other*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF*	*RF*	*LPF*	*Source Term Dose Equivalent Scale Factor*	*Source Term No. of Assemblies or grams Pu-239*
D.3.2.1 Fire Outside Confinement (Waste from fuel production or spent fuel treatment)	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	0.001	Noble Gas: 1 Other: 5×10^{-4}	1	1	1.00	4.61×10^{-6}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	3,346 grams	Pu-239 equivalent	Unlikely	0.5	6.0×10^{-3}	0.01	1	1.00	1.00×10^{-1}

Am = americium; ARF = airborne release fraction; DR = damage ratio; INL= Idaho National Laboratory; LPF = leak path factor; MAR = material at risk; ORNL = Oak Ridge National Laboratory; Pu = plutonium; RF = respirable fraction; SRS = Savannah River Site.

D.3.2.2 Fire Outside Involving a Waste Drum with 23 Grams of Americium-241

TRU waste containing americium-241 is generated by fuel production operations. Production waste is placed in containers and temporarily stored (or staged) pending shipment to an offsite disposal facility. A fire is postulated to occur in a 55-gallon drum because of poor housekeeping. The accident is assumed to occur outside confinement during handling. The release occurs at ground level over 10 minutes. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The assumed accident frequency is unlikely. Based on the Waste Isolation Pilot Plant (WIPP) Acceptance Criteria, remote-handled waste is limited to 55-gallon drums with a maximum allowed load of 80 curies or about 23 grams of americium-241. To comply with the WIPP criteria, MAR is assumed to be 1 drum of TRU waste containing 23 grams of americium-241 or 3,346 grams of plutonium-239 equivalent. The DR is assumed to be 0.5 because only a portion of the waste in the container is assumed to be involved in the fire. The bounding ARF and RF values of 6×10^{-3} and 0.01, respectively, are based on thermal stresses to composite solids from DOE-HDBK-3010-94 (DOE 1994). Since the fire is outside building confinement, the LPF is 1. Involved workers could inhale some radioactive material before evacuating the area. The effects from the fire outside confinement bound the effects from a corresponding fire inside confinement or a drop of the drum either inside confinement or outside confinement. Table D–3 summarizes MAR, release fractions, and source term for this accident scenario

D.3.3 Versatile Test Reactor Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

The selection of accidents for the VTR at either INL or ORNL considers various sources of background information. Information in Sections D.3.3.1 through D.3.3.4 is used to select accidents that provide representative radiation and hazardous material effects to receptors associated with siting the VTR at either INL or ORNL. The analysis for these accidents is presented in Section D.3.3.5.

D.3.3.1 Review of Sodium-Cooled Reactor Accidents and Operations

Sodium-cooled reactors have been operated for a number of years. Analysis of sodium-cooled reactor accidents provides insight into possible accident initiators as discussed below (Cahalan 2008).

Experimental Breeder Reactor-1 (EBR-I) – During a test to investigate the prompt positive component of the power coefficient, an unanticipated power excursion resulted in fuel melting. Subsequent investigations identified fuel rod bowing as the source of the positive reactivity feedback.

Fermi-1 – Investigations revealed fuel melting in two adjacent assemblies. Another adjacent assembly was bent, with no internal damage. A crumpled zirconium plate was found in the inlet plenum. Flowing coolant apparently caused the loose zirconium plate to partially or completely cover the inlet nozzle of various assemblies during the multiple start-ups.

Phenix – Four rapid, large, negative reactivity excursions triggered automatic scrams due to power reduction. Given the amplitude and speed of the events, only core movements could cause the observed behavior. The final explanation attributed the reactivity excursions to outward (radial) expansion of the assembly lattice followed by a return to the original position.

Super Phenix – Leaking sodium was contained by the storage tanks' guard vessel. Investigation showed cracks at support plates and tank walls' weld beads. Cracking was associated with the steel material in conjunction with microcracking in zones of high hardness, residual stresses close to the elastic limit of the material, and the hydrogen embrittlement.

MONJU – A sodium leak was detected at a thermocouple well. The thermocouple well tip failed due to flow-induced cycle fatigue.

Fast Flux Test Facility (FFTF) – Analysis of observations during the FFTF acceptance/startup testing program (Wootan et al. 2017) reveals items that may be pertinent to the VTR.

1. Periodic flow and pressure oscillations occurred in the secondary main heat transport system loops. The oscillations were caused by periodic vortex formation and release from piping tees near the inlets of the dump heat exchangers.
2. Fuel cladding breaches can release radioactive cesium (cesium-134 and cesium-137) to the coolant and become a significant source of radiological dose in the plant. Cesium can be effectively removed from the sodium using cesium traps. Because of the relatively high vapor pressure of cesium at reactor operating temperatures, cesium is likely to transport to and throughout the reactor cover's gas system.
3. Gas entrainment and accumulation in the primary coolant could result in a positive reactivity insertion in the reactor if a significant volume of gas were to pass through the core. A similar reactivity insertion could occur in the VTR.
4. FFTF experienced a significant increase in the primary system's sodium pressure drop during its early operation. Pressure normally decreases between inlet and outlet (pressure drop) but the pressure decrease was greater than expected. Observers believed the increased pressure drop was caused by the deposition of silicon-based crystals in the fuel assembly's inlet orifices.
5. Under low flow conditions, sodium can stratify, resulting in large vertical thermal gradients and the potential for undesirable thermal stresses.
6. An inert gas, typically argon, is used as a cover gas over the sodium in the reactor vessel and other components in liquid metal reactors. Due to the high sodium temperature, the cover gas may become saturated with sodium vapor and contain significant quantities of sodium aerosol.

EBR-II – A wide variety of mild undercooling and overpower tests were safely conducted at EBR-II. Identification and investigation of major safety and availability issues surrounding sodium-cooled fast reactor operation with breached fuel elements was conducted through a vigorous run-beyond-cladding-breach testing program. Safety-related experimental programs and modifications in EBR-II resulted in the development and defense of a safety philosophy and documentation to support operation of sodium-cooled fast reactors and development of the first set of technical specifications for a fast reactor power plant in the United States. Additionally operating history and demonstration tests, such as those performed at EBR-II, demonstrated the ability for metal-fueled sodium reactors to be designed in such a manner that core reactivity decreases as temperature increases. This feature would reduce reactor fission power to levels matching heat rejection rates.

A series of tests, originally intended to qualify EBR-II for continued operation, evolved into an experimental program supporting the safety and design of advanced liquid metal reactors. Testing ranged from demonstration of removal of decay heat by natural circulation to whole-plant simulation of unprotected (without scram) loss-of flow (ULOF) and loss-of-heat-sink (ULOHS) accidents from full power and flow. The ULOF and ULOHS tests demonstrated the ability of pool-type, metal-fueled liquid metal reactors to provide self-protection in beyond-design-basis accidents.

D.3.3.2 Current Versatile Test Reactor Safety Basis

The VTR is being designed with the concept of ensuring safety throughout proposed operating conditions, as well as being resilient under potential accident or upset conditions. Consistent with DOE guidance for safety in design, the focus of design is to reduce or eliminate hazards, with a bias towards preventive, as opposed to mitigative, design features and a preference for passive over active safety systems. This general approach creates a design, which is reliable, resilient to upset, and has low potential consequences of accidents.

Safe operation of the VTR is ensured by reliable systems design to ensure preservation of the key reactor safety functions. These key safety functions can be summarized as (1) reactivity control, (2) fission- and decay-heat removal, (3) protection of engineered fission product boundaries, and (4) shielding.

Various systems, which function separately and independently, provide reactivity control in the VTR. The first element of reactivity control is the design of a fuel and core system so that the core experiences a negative reactivity feedback when core temperatures increase. This provides stability in the reactor by ensuring that the power in the reactor cannot "run away" and that deliberate actions are necessary to increase the level of power in the reactor. In addition to this fundamental design expectation, the VTR is designed to be operated without automatic control rod removal. This means that evolutions to increase reactor power are initiated by the operator and are not controlled by an automatic system. This design eliminates automatic rod withdrawal as a potential upset condition. In the event that a plant upset or accident condition results in the need for shutting the reactor down, the reactor can be shut down either through the normal control system slow cutback function or through the scram function. The slow cutback function would drive control rods into the reactor's core until the condition clears or the plant is shutdown. The scram function would shut down the reactor through a VTR Plant Protection System actuation. This system causes an immediate delatching of the control rod drivelines from the control rod drive motors and allows for a spring-assisted gravitational insertion of the control rods into the core. The weight of the control rod and driveline are sufficient to insert the control rod into the core. However, the use of the supporting spring decreases the response time and minimizes the potential for any binding or sticking of control rods. The Plant Protection System, from sensing through logic, and into actuation is an extremely reliable system with independence and diversity in each layer. For additional safety, the VTR core is designed so that the engineered reactivity feedbacks passively reduce reactor power and match residual heat rejection capability simply as a function of the core response to temperatures elevated above the operating temperature, but below cladding failure temperatures (BEA 2020).

Adequate fission product and decay-heat cooling are provided in the design of the VTR through a number of design decisions that ensure robust and resilient mechanisms for heat removal from the reactor and supporting systems. The first design decision is the choice of metal fuel with a sodium bond. The excellent heat transfer capabilities of the metal fuel with sodium bond result in steady state fuel and coolant operating temperature differences, which are much lower than other fuels that have gas gaps and lower heat conduction properties. Use of metal fuel with sodium bond allows greater temperature margins between the fuel temperature and the potential failure temperatures of critical components. In the event that the active mode of the Heat Rejection System fails, the system is designed for each of the loops to have the capacity to reject sufficient heat to prevent fuel damage while functioning in an entirely passive mode. Thus, redundant systems exist that ensure appropriate heat removal capacity. In the event that common issues or very low probability events compromise the ability for both loops of the heat rejection system to reject heat, the ultimate heat removal system, the Reactor Vessel Auxiliary Cooling System (RVACS), would reject the heat. This system allows the heat from the reactor vessel to radiate outward to the collector cylinder. Heat from the collector cylinder and reactor guard vessel would be removed by convection as cold air from outside runs down the cold leg of the RVACS inlet and is heated. The heated air is then exhausted out the RVACS's hot leg stack. One benefit of both the passive mode of the heat removal system and of the RVACS system is that both mechanisms function in a passive state relying only on temperature differences to reject heat. Thus, they are not susceptible to increased temperatures because higher temperatures would result in elevated heat rejection through the dominant heat removal mechanisms of convection and radiation behaviors, respectively (BEA 2020).

Integrity of the barriers designed to provide retention of fission and activation products would be ensured by keeping temperatures and pressures below design limits. A number of barriers would provide mitigation to ensure control of radioactive materials. First, the metal fuel matrix and sodium compatibility would minimize transportation of fission products into the fuel gas gap. Then, the fuel cladding would provide a leak-tight boundary that would contain radioactive materials within the fuel. In the event that some event causes failure of the fuel cladding, fission products would be held up in the primary sodium coolant, the reactor vessel and cover gas cleanup systems, the facility confinement boundary and then, if necessary, the experiment hall confinement boundary and filtration systems (BEA 2020).

A thorough evaluation of the potential abnormal conditions and associated accidents are evaluated in the facility's safety basis documents. The safety analysis of the VTR conceptual design (INL 2019) evaluates accidents involving loss of offsite power, loss of flow, transient overpower, and loss of heat sink with and without the Reactor Protection System (RPS) functioning as designed and demonstrates that fuel failure temperatures would not be exceeded.

D.3.3.3 Key Conclusions from Versatile Test Reactor Safety Analyses

Deterministic safety analyses for the current state of the VTR conceptual design (INL 2019) are being evaluated to support design decisions and analyses that will be required for the future preliminary safety analysis report. The full spectrum of event initiators and subsequent accident sequences would be defined through a probabilistic risk assessment (PRA) of the plant design following the Licensing Modernization Project guidelines. At the current stage of design, transient simulation results for the VTR indicate that large safety margins exist for many event initiators. However, several "enabling" assumptions have been made, in terms of both design features and design limits, in order to perform the analyses; these assumptions may need to be revised as the design matures. Additional events may be evaluated in the future as the VTR design evolves.

Analyses indicate that the offsite source terms for VTR accident scenarios tend to be small because of the melting temperature of the fuel, retention of fission products within the sodium pool (Bucknor et al. 2017), and the small driving force from temperature and pressure in the VTR for release from the confinement to the environment. Transient scenarios for the VTR are presented below.

D.3.3.3.1 Protected Transient Scenarios

The protected transient overpower (PTOP) scenario evaluated in *Safety Analysis for the Versatile Test Reactor Conceptual Design* (ECAR-4733) (INL 2019) is initiated by an unintentional withdrawal of the most reactive control rod. The RPS responds to off-normal conditions by initiating a full system shutdown. The PTOP scenario is a bounding event for perturbations of the reactor core through reactivity changes. The protected-loss-of-heat-sink (PLOHS) accident evaluated in ECAR-4733 is initiated by a simultaneous trip of the secondary electromagnetic (EM) pumps, which significantly reduces heat rejection through the internal heat exchangers (IHXs).

Complete loss of sodium-to-air heat exchanger (SAHX) heat rejection demonstrates how the system responds with the RVACS providing the only source of heat rejection before the RPS detects elevated system conditions. Once elevated temperatures are detected, the RPS responds. The PLOHS is a bounding event for perturbations of the reactor core through core inlet temperature changes.

The protected-loss-of-flow (PLOF)/protected-station-blackout (PSBO) transient evaluated in ECAR-4733 is initiated by an assumed loss of electrical power to all plant systems. Forced circulation in the primary and secondary loops is lost, and heat rejection through the SAHXs is assumed to decrease to zero. The RPS responds to off-normal conditions by initiating a full system shutdown. When the primary pumps trip, core flow decreases to 60 percent and coasts down with a 12-second initial flow halving time thereafter. In the secondary loops, pump head is assumed to decrease from 100 to 0 percent in 1 second. The PSBO is a bounding event for perturbations of the reactor core through mass flow rate changes.

A loss of offsite power (LOOP) considers interruptions of normal power to the electrical buses, which would result in failure of the primary and intermediate EM pumps and reactor scram. Plant-centered LOOP data includes failures of primary safety-class alternating current buses in the plant, which would result in the need to start diesel generators. The LOOP event sequence assumes that all EM pumps are tripped with a LOOP event.

As evaluated in ECAR-4733 (INL 2019), immediately following a seismically-induced LOOP and shutdown of four EM pumps, a scram signal is generated to initiate control rod insertion, and control rods successfully insert. This transient scenario assumes that a reactor shutdown is initiated after detection of a seismic P wave. An assumed five seconds after the shutdown sequence is initiated, the slower, stronger S wave reaches the plant and is assumed to disable the primary pump coast-down mechanisms. Temperatures in the core increase, but because power is significantly reduced and the power-to-flow ratio is less than before the transient began, in-core temperatures do not increase back to the pre-transient values. Even with the coast-down mechanisms being disabled only five seconds after pump trip, the VTR would be able maintain low temperatures and successfully transition to natural circulation.

The PTOP transient described above is analyzed in ECAR-4733 with the addition of an assumed "stuck rod" failure to simulate a malfunction of equipment. That is, in addition to the withdrawal of the most reactive control rod, one of the remaining control or safety rods fails to insert when the RPS initiates a scram. With one of the five control rods that did not withdraw sticking during the RPS-induced scram, the negative reactivity introduced by the scram is reduced. The maximum temperatures predicted during the transients are the same whether or not one rod sticks.

A single coast-down failure during a PTOP or PLOHS transient would not impact peak temperatures because the coast-down failure occurs as part of the transient mitigation by the RPS, that is the reactor is decreasing in power due to insertion of control rods and subsequently, the coast down is being employed. During a PSBO transient, a coast-down failure would occur as part of the transient initiator, so the impact would be more significant. The pump without a functional coast down provides an open flow path back to the cold pool, so some flow would be diverted from the core and instead flow back through the failed pump. Flow in the core would drop quickly and the RPS would initiate a reactor shutdown. All EM pumps and SAHXs would have already tripped, so the only remaining action for the RPS to take would be to scram the control rods. This timing would be the same as for the standard PSBO. The system then would transition to natural circulation and decay heat would slowly decrease.

The review of experiments indicates potential sequences that can lead to an experiment malfunction and a positive reactivity insertion. These include:

1. Insertion from an instrumented assembly
2. Insertion from a non-instrumented assembly
3. Insertion from a test loop
4. Insertion from a rabbit insertion tube
5. Instrumentation or gas line failures that can lead to material discharge into the core

Bounding estimates of the potential design-basis accident reactivity insertion for the above sequences are likely comparable to the transient overpower scenarios analyzed.

D.3.3.3.2 Unprotected Transient Scenarios

A preliminary set of criteria has been developed for evaluating the VTR's response to the unprotected transient scenarios analyzed below. The three unprotected transient scenarios are assessed against criteria roughly based on the PRISM Preliminary Safety Information Document criteria for extremely unlikely events.

The primary considerations for extremely unlikely events are that coolable fuel pin geometry must be maintained and radioactive materials contained. Both can be ensured if the multiple layers of physical protection (i.e., the cladding, reactor vessel, guard vessel, and confinement) remain intact and damage to the reactor does not occur. In these events, a limited amount of fuel melting may be tolerable if the cladding remains intact, and the coolant boundary is not at risk of failure.

The unprotected-transient-overpower (UTOP) scenario evaluated in ECAR-4733 is initiated by an unintentional withdrawal of the most reactive control rod. The UTOP is a bounding event for perturbations of the reactor core through reactivity changes without any mitigating actions by the RPS or reactor operators.

The unprotected-loss-of-heat-sink (ULOHS) transient evaluated in ECAR-4733 is assumed to be initiated by a simultaneous trip of all secondary EM pumps, significantly reducing heat rejection through the IHXs. Additionally, heat rejection through the SAHXs is assumed to be lost. Assuming zero SAHX heat rejection is conservative because, even if the dampers for all ten SAHXs were completely closed, there would still be parasitic heat losses. As with the other unprotected transients, the RPS is assumed to fail to take any action in response to the off-normal conditions. Reactor operator action is also neglected. Therefore, control rods do not scram, and the primary pumps do not trip during the ULOHS. The primary pumps remaining on is the main difference between the ULOHS transient and the unprotected-station-blackout (USBO) transient analyzed below.

The ULOF/USBO transient evaluated in ECAR-4733 is initiated by an assumed loss of electrical power to all plant systems. Forced circulation in the primary and secondary loops is lost, and heat rejection through the SAHXs is assumed to decrease to zero. This transient is initiated by the same failures as the PSBO transient, except that the RPS is assumed to fail to take any action in response to the off-normal conditions. Therefore, control rods would not be scrammed during the USBO. The USBO is a bounding event for perturbations of the reactor core through mass flow rate changes without any mitigating actions by the RPS or reactor operators. The station blackout (SBO) is considered more severe than a loss of flow (LOF) because it also includes a loss-of-heat-sink (LOHS) component due to the simultaneous tripping of the secondary pumps and the loss of heat rejection through the SAHXs. The RVACS is responsible for long-term heat rejection.

D.3.3.4 Versatile Test Reactor Defense-in-Depth

The VTR safety design approach implements the defense-in-depth strategy by adopting the traditional five layers of defense-in-depth to the VTR as follows:

Layer 1: Prevention of abnormal operation and failures

Layer 2: Control of abnormal operation and detection of failures

Layer 3: Control of accidents within the design-basis

Layer 4: Control of severe facility conditions

Layer 5: Mitigation of radiological consequences.

D.3.3.4.1 Prevention of Abnormal Operation and Failures

At the first level, the VTR is designed to operate with a high level of reliability, so that accident initiators are prevented from occurring. The conceptual VTR plant takes advantage of well-established fast reactor metal fuel, sodium coolant, and structural materials that are stable and compatible. VTR design considers the proposed operational ranges for systems and components and ensures that material selection provides for reliable operations during normal operations. Thermal and operational duty cycles are considered in this analysis to ensure that fatigue failures and aging degradations are both understood and minimized. Plant control systems (PCS) are designed as high reliability industrial systems to keep high plant availability. Additionally, adequate instrumentation to ensure reliable plant control and early recognition of abnormal conditions is provided. PCSs are designed with the intent of ensuring that stable plant states are maintained during plant control changes and control variables are measured and evaluated to ensure that changes resulting in abnormal operations are minimal.

D.3.3.4.2 Control of Abnormal Operation and Detection of Failures

Plant instrument and control systems provide the initial protection for ensuring that abnormal operations and deviations are minimized and that conditions are appropriately corrected when failures occur. The PCS contains provisions to detect and provide alarms associated with equipment failures. Actions initiated by the PCS could, as appropriate, decrease reactor power by inserting control rods or throttle primary and secondary EM pumps to ensure heat balance.

In addition, the RPS would initiate a reactor scram, shut off the primary EM pumps, and initiate confinement isolation if identified setpoints were exceeded. Reliability of ensuring the ability of core flow and the transition from forced to natural circulation would be provided by the EM pump coast-down machines, which facilitate applying appropriate control in events where core flow is impacted.

In addition to these active systems, the VTR design benefits from strong favorable reactivity feedbacks that together with the low-pressure sodium coolant and reference metallic fuel, provide passive shutdown and passive safety behavior under various reactor upset conditions.

D.3.3.4.3 Control of Accidents within the Design Basis

This layer of defense-in-depth for the VTR is achieved by conservative design and engineered safety systems for reactor shutdown, decay heat removal, and confinement of radioactive materials. Success in meeting the objectives in this layer is shown by virtue of the fact that all safety basis events are well within the acceptable range, and all design-basis accidents analyzed are successfully mitigated by the safety-class structures, systems, and components performing their intended safety functions. Successive, multiple physical barriers are in place for protection against release of radioactivity and hazardous materials. Multiple, diverse, and independent means are provided to accomplish safety functions. Reactor shutdown systems, actuated by the RPS, and shutdown heat-removal systems provide high reliability protection functions. The selection of liquid sodium coolant and metallic fuel with a pool-type primary system arrangement provides a highly reliable reactor system with large operational safety margins. The coolant thermophysical properties provide superior heat removal and transport characteristics at low operating pressure with a large temperature margin to boiling. The metallic fuel operates at a relatively low temperature, below the coolant boiling point, due to its high thermal conductivity. The pool-type primary system confines all significantly radioactive materials within a single vessel and allows for shutdown heat removal by natural circulation within the vessel and through the RVACS.

D.3.3.4.4 Control of Severe Facility Conditions

This layer's objective is to control severe plant conditions and mitigate beyond extremely unlikely event consequences. The “unprotected” TOP (UTOP), LOHS (ULOHS), and SBO (USBO) transients are initiated by the same sequences of events defined for the protected transients, except the RPS is assumed to fail to take any action. The inherent and passive features of the system are responsible for bringing the system to a stable state at safe temperatures. The UTOP, ULOHS, and USBO scenarios are bounding events for perturbations of the reactor core through reactivity changes, core inlet temperature changes, and core mass flow rates changes, respectively, without any mitigating actions by the RPS or reactor operators. The passive performance mechanisms for ensuring reactivity control and cooling provide performance with generally stronger feedbacks as temperatures increase. These design features help control the level of severity of facility upsets. Additionally, the various levels of confinement barriers (cladding, coolant, reactor vessel, confinement, and experiment hall structure) provide thresholds that serve to control the release of radioactive material if facility conditions are severe enough to result in fuel failures and releases.

D.3.3.4.5 Mitigation of Radiological Consequences

The fifth layer of defense applies to severe accidents where significant releases of radiological or hazardous material could occur. High accident doses to workers and the public could result from severe accidents. However, DOE requirements for emergency planning and the generally large distances to site boundaries at DOE sites, as well as additional safety management programs, provide the ability to mitigate consequences from these extremely low probability events.

D.3.3.4.6 VTR Facility Seismic Hazard Classification

The VTR would satisfy the applicable requirements and criteria contained in DOE-STD-1020-2016. Based on being a hazard category 1 facility, the VTR and support safety-class systems are categorized as SDC-5, and the experiment hall and other facility handling systems are categorized as SDC-3 or less, per the criteria in DOE-STD-1020-2016 (DOE 2016). Such facilities would have to be designed or evaluated for an SDC-3 design-basis earthquake with a mean annual exceedance probability of 1×10^{-4}, corresponding to a return period of 10,000 years or an SDC-5 design-basis earthquake with a mean annual exceedance probability of 1×10^{-5}, corresponding to a return period of 100,000 years.

D.3.3.5 Analysis of Versatile Test Reactor Reactor-Building Accidents

The analysis of reactor building accidents considers accidents that have a low frequency of occurrence, but large consequences and a spectrum of other accidents that have higher frequencies of occurrence and smaller consequences. As indicated in Sections D.3.3.2 and D.3.3.3, most VTR-specific accidents are either prevented or mitigated such that there would be no release of radioactive materials to the environment. An early PRA for the VTR identified a few accident sequences that, at very low probabilities, had the potential for fuel damage and confinement damage. These sequences were predominately initiated by severe seismic events substantially above the seismic design-basis. These beyond extremely unlikely, high-consequence accident sequences presented the highest public radiological risks (probability × consequences) because most of the traditional accidents (for example, loss of coolant accidents) were either prevented or mitigated and did not result in radiological releases.

In addition to the VTR-specific design-basis accidents identified with no radiological releases, accidents resulting in releases from the VTR and the adjoining experiment hall require consideration in NEPA documents. In the VTR building, one possible pathway to the environment would be from releases from the cover gas. These releases would, by design be minuscule. Accidents in the experiment hall involving fuel handling are also a possibility. The ex-vessel hazards and accidents have been determined to involve a mishandling or malfunction of fresh or spent fuel during fuel movements or storage outside the reactor vessel. The VTR may also be vulnerable to beyond-design-basis events, such as severe seismic events, that might fail reactor cooling and confinement. Each of these categories of accidents is addressed in the following sections.

In the analysis presented in the following subsections for VTR-building accidents, the source term for the spent fuel that has 6 percent burnup with no decay time is found in *Probabilistic Risk Assessment, Preliminary Mechanistic Source Term Analysis* (ECAR-4737) (Grabaskas 2019) and is shown in Table D–42 (Attachment D1). The source term for the fresh fuel is calculated with the isotopic compositions shown in the *Evaluation of the VTR Ex-Vessel Inhalation Dose Consequences* (INL 2020b). The source term for the fresh fuel includes uranium at 5 percent enrichment and the isotopic distribution shown in **Table D–4**. (Note that americium is added to account for a 30-year plutonium-241 decay.) **Table D–5** details the pin and assembly mass in grams.

Table D–4. VTR Fresh Assembly

Isotope	*Weight Percent*	*Grams*
Plutonium-238	0.10%	8.9
Plutonium-239	68.70%	6112.2
Plutonium-240	26.40%	2348.8
Plutonium-241	3.40%	75.6
Plutonium-242	1.40%	124.6
Americium-241	75% of 241	226.9
Uranium-234		11.1
Uranium-235	5%	1770.0
Uranium-238		33589.9

VTR = Versatile Test Reactor.

Table D–5. VTR Pin and Assembly Mass

204	grams per fuel pin
41	grams plutonium per pin
217	pins per assembly
8,897	grams plutonium per assembly
35,371	grams uranium per assembly

VTR = Versatile Test Reactor.

D.3.3.5.1 Versatile Test Reactor Design-Basis-Accidents with Releases

The safety analyses for the VTR based on the conceptual design (INL 2019, 2020c), as well as the preliminary *Versatile Test Reactor Probabilistic Risk Assessment Summary* (VTR PRA) (GEH 2019) indicate that most of the accident sequences do not lead to a release of radionuclides from the reactor. The potential of a radioactive material release in the cover gas can be an important factor for operational safety. Leaks may result in the release of radioactive gases from the primary system, including fission gases from failed fuel and activated sodium vapor/aerosols (sodium-23 and argon-41), into the reactor room, other reactor facilities, or the environment. While the cover gas system is at low pressure, portions of it will likely be at relatively high temperature (approximately the temperature of the reactor cover gas within the vessel). Although the system is at low pressure with relatively benign contents consisting mostly of noble gases, the length of piping and number of valves associated with the system provides potential avenues for system leaks.

Releases from the cover gas cleanup system are assessed as part of two different scenarios because the system extends from the reactor head in the reactor room to a service room outside the reactor room. The first scenario examines a major leak of the cover gas cleanup system within the reactor room. A leak in this location would result in the complete release of the contents of the reactor cover gas. However, parts of the system outside the reactor room would be isolated by system check valves. The second scenario assesses a leak from the cover gas cleanup system outside the reactor room, which is where the cesium traps and noble-gas-removal components are located. Half of the cesium inventory is located within the cesium traps and all of the noble gases from the pins are within the noble gas retention components. With a system leak, release of all of the cesium and noble gases within these components is conservatively assumed. In addition, no retention within the service room is assumed, representing an immediate release to the environment with no time for decay. The parts of the cover gas cleanup system within the reactor room are assumed to isolate from the section of the system in the service room.

Cover Gas System Failure with Intact Confinement

The accident is assumed to occur because of an equipment failure in the cover gas system. The assumed accident frequency is unlikely. MAR is assumed to be the gaseous inventory from three fuel pins and activation products in the cover gas. The DR is 0.5 for cesium and 1 for all other isotopes. For generation of vapors plus release from physical confinement, the recommended ARF is 1.0 (DOE 1994). All materials in the gaseous state can be transported and inhaled; therefore, an RF of 1.0 is recommended (DOE 1994). Reduction in the source term because of the reactor confinement is modeled, and an LPF of 6×10^{-5} is used. **Table D–6** summarizes MAR, release fractions, and source term for this accident scenario (BEA 2020).

Table D–6. Source Term for Cover Gas Cleanup System Leak in the Reactor Room

	Material at Risk		*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
	Isotope	*Curies*					
Activation Gases	Ne-23[a]	0	1	1	1	6×10^{-5}	0
	Ar-41	1.12×10^{-3}	1	1	1	6×10^{-5}	6.72×10^{-8}
	Na-22	1.18×10^{-4}	1	1	1	6×10^{-5}	7.08×10^{-9}
	Na-24	1.43×10^{-3}	1	1	1	6×10^{-5}	8.58×10^{-8}
Failed Fuel Pins	Cs-134	3.04×10^{1}	0.5	1	1	6×10^{-5}	9.12×10^{-4}
	Cs-135	1.59×10^{-3}	0.5	1	1	6×10^{-5}	4.77×10^{-8}
	Cs-136	1.09×10^{2}	0.5	1	1	6×10^{-5}	3.27×10^{-3}
	Cs-137	1.06×10^{2}	0.5	1	1	6×10^{-5}	3.18×10^{-3}
	Cs-138	1.13×10^{-4}	0.5	1	1	6×10^{-5}	3.39×10^{-9}
	Kr-83m	2.67	1	1	1	6×10^{-5}	1.60×10^{-4}
	Kr-85	1.62×10^{1}	1	1	1	6×10^{-5}	9.72×10^{-4}
	Kr-85m	1.12×10^{3}	1	1	1	6×10^{-5}	6.72×10^{-2}
	Kr-87	4.44×10^{-5}	1	1	1	6×10^{-5}	2.66×10^{-9}
	Kr-88	5.96×10^{1}	1	1	1	6×10^{-5}	3.58×10^{-3}
	Xe-131m	2.64×10^{3}	1	1	1	6×10^{-5}	1.58×10^{-1}
	Xe-133	4.00×10^{3}	1	1	1	6×10^{-5}	2.40×10^{-1}
	Xe-133m	5.18×10^{3}	1	1	1	6×10^{-5}	3.11×10^{-1}
	Xe-135	5.57×10^{3}	1	1	1	6×10^{-5}	3.34×10^{-1}
	Xe-135m	4.91×10^{3}	1	1	1	6×10^{-5}	2.95×10^{-1}
	Xe-138	3.87×10^{-5}	1	1	1	6×10^{-5}	2.32×10^{-9}

Ar = argon; Cs = cesium; Kr = krypton; Na = sodium; Ne = neon; Xe = xenon.
[a] Ignored due to short half-life (~37 seconds).
Source: BEA 2020.

Release from Cover Gas System Failure with Failed Confinement

The accident is assumed to occur when an earthquake causes a failure of the cover gas piping. The frequency of the earthquake is extremely unlikely. For purposes of this scenario, MAR is assumed as the gaseous inventory from 10 fuel pins even though a fuel failure rate of this magnitude would be a significant operational concern and would likely prompt reactor outages to understand the issues. The DR is 0.5 for cesium and 1 for all other isotopes. For generation of vapors plus release from physical confinement, the recommended ARF is 1.0 (DOE 1994). All materials in the gaseous state can be transported and inhaled; therefore, an RF of 1.0 is recommended (DOE 1994). There is no reduction in the source term because of reactor confinement, and an LPF of 1 is used. **Table D–7** summarizes MAR, release fractions, and source term for this accident scenario (BEA 2020).

Table D–7. Source Term for Cover Gas Cleanup System Leak Outside the Reactor Room

	Material at Risk		*Damage Ratio*	*Airborne Release Fraction*	*Respirable Fraction*	*Leak Path Factor*	*Source Term (curies)*
	Isotope	*Curies*					
Noble Gas Retention Components	Kr-85	2.90×10^{-1}	1	1	1	1	2.90×10^{-1}
	Kr-87	1.10×10^{-5}	1	1	1	1	1.10×10^{-5}
	Kr-88	3.60×10^{-5}	1	1	1	1	3.60×10^{-5}
	Xe-133	8.60×10^{-3}	1	1	1	1	8.60×10^{-3}
	Xe-135	7.30×10^{-4}	1	1	1	1	7.30×10^{-4}
Cesium Traps	Cs-134	2.00×10^{-1}	0.5	1	1	1	1.00×10^{-1}
	Cs-136	1.88×10^{-3}	0.5	1	1	1	9.40×10^{-4}
	Cs-137	12.2	0.5	1	1	1	6.10
	Cs-138	1.14×10^{-4}	0.5	1	1	1	5.70×10^{-5}

Cs = cesium; Kr = krypton; Xe = xenon.
Source: BEA 2020.

D.3.3.5.2 Versatile Test Reactor Experiment Hall Accidents

A range of operations involving irradiated fuel assemblies and irradiated test assemblies would occur in the experiment hall. These operations could expose the fuel to a range of hazards, including fires and drops due to mechanical failures or human error. These could be initiated by a range of events, including operational accidents and a seismic event. The experiment hall would be designed and evaluated for an SDC-3 design-basis earthquake with a mean annual exceedance probability of 4×10^{-4}, corresponding to a return period of 2,500 years. Thus, seismic events in the extremely unlikely or lower frequency category could possibly be expected to initiate accidents. Four accidents are postulated, including a beyond-design-basis seismic event so severe that collapse of the experiment hall would be possible.

D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire

A fire involving a test assembly in the experiment hall is postulated. The frequency of the seismically-induced fire is extremely unlikely. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. MAR is based on one spent fuel pin, with the nuclide distribution of a spent fuel assembly decayed for 220 days. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For plutonium, americium and other actinides in MAR, the bounding ARF and RF values of 3×10^{-5} and 0.04, respectively (DOE 1994). For uranium, activation products, and fission products in MAR, the bounding ARF and RF values of 1×10^{-3} and 1, respectively (DOE 1994). There is no reduction in the source term because of reactor confinement, so an LPF of 1 is used. **Table D–10** summarizes MAR, release fractions, and source term for this accident scenario.

D.3.3.5.2.2 Fire Involving Versatile Test Reactor Spent Fuel Assemblies

A fire involving spent fuel in the experiment hall is postulated. DOE considers releases from fires involving fresh fuel or releases from fires involving a test assembly to be bounded by fires involving spent fuel. A room fire is assumed to occur and impinge upon a safety-class spent fuel cask designed to withstand the thermal stress from a fire. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The frequency of the imitating fire is unlikely, but a fire of the magnitude postulated coupled with failure of a safety-class cask and heating the spent fuel to eutectic temperatures (800 degrees Celsius (1,470 degrees Fahrenheit) is considered extremely unlikely to beyond extremely unlikely. MAR is assumed as the amount of fuel in 3 spent fuel assemblies that have cooled for 220 days. The DR is 1. The fire event's release fractions are taken from a fuel assembly at eutectic[5] pin failure, but the fuel does not create a self-sustained oxidizing reaction. The bounding ARF and RF values by isotope group are from **Table D–8**. There is no reduction in the source term from the experiment hall confinement, and an LPF of 1 is modeled. Table D–10 summarizes MAR, release fractions, and source term for this accident scenario.

Table D–8. Release Conditions for the Eutectic Spent Fuel Fire in the VTR Experiment Hall

Isotope Group	***Isotope Group ARF × RF***	***Material, Release Conditions, and Reference***
Noble Gases Group: (iodine, bromine)	0.67	The eutectic release fractions are taken from 800 degrees Celsius (1,472 degrees Fahrenheit) eutectic pin failure event. The fractions are the migration fractions interpolated to 6 percent burnup. The resulting fractions may be conservative for a fire involving metal fuel that does not create a self-sustained oxidizing reaction outside of the reactor vessel because it assumes the estimated releases are entirely airborne and respirable.
Alkali Metals Group: (cesium, rubidium)	0.1	
Tellurium Group: (tellurium, antimony, selenium)	0.42	
Barium, strontium Group: (barium, strontium)	6.0×10^{-3}	
Noble Metals Group: (ruthenium, rhodium, palladium, molybdenum, technetium, cobalt)	0.24 6.0×10^{-4}	
Cerium Group: (cerium, plutonium, neptunium)	6.0×10^{-4}	
Lanthanides Group: (lanthanum, zirconium, neodymium, niobium, promethium, samarium, yttrium, curium, americium)	6.0×10^{-5}	
Europium Group: (europium)	0.42	
Remaining elements	1.0×10^{-3}	

ARF = airborne release fraction; RF = respirable fraction; VTR = Versatile Test Reactor.
Source: ECAR 4777 Rev. 0, page 3 (INL 2020b).

D.3.3.5.2.3 Versatile Test Reactor Fuel Assembly Drop in Experiment Hall

A drop of spent fuel in the experiment hall is postulated. Drops of fresh fuel are considered to have no release fractions. A drop of spent fuel bounds releases from a drop of a test assembly. The release fractions for the spent fuel drop are listed in **Table D–9**.

[5] The term eutectic refers to a mixture of materials that has a melting point that is lower than the melting points of the separate materials. The term eutectic is important in the fire because the VTR reactor fuel may have a lower melting temperature and larger release fractions than fuel from reactors other than the VTR.

Table D–9. Release Factors for the Spent Fuel Drop in the VTR Experiment Hall

Isotope Group	*Isotope Group ARF × RF*	*Material, Release Conditions, and Reference*
Drop of spent fuel Cold gap Release		A cold gap release occurs when spent fuel element are mechanically damaged, but the temperature is low enough that the cladding does not suffer any thermal damage.
Noble Gases Group:(xenon, krypton)	0.4	
Halogens Group: (iodine, bromine)	0.003	
Alkali Metals Group: (cesium, rubidium)	0.003	
Tellurium Group: (tellurium, antimony, selenium)	1.0×10^{-4}	
Barium, Strontium Group: (barium, strontium)	6.0×10^{-7}	
Noble Metals Group: (ruthenium, rhodium, palladium, molybdenum, technetium, cobalt)	6.0×10^{-7}	
Cerium Group: (cerium, plutonium, neptunium)	6.0×10^{-7}	
Lanthanides Group: (lanthanum, zirconium, neodymium, niobium, promethium, samarium, yttrium, curium, americium)	6.0×10^{-7}	

ARF = airborne release fraction; RF = respirable fraction; VTR = Versatile Test Reactor.
Source: ECAR 4777 Rev. 0, page 4 (INL 2020b).

Given the need for irradiated fuel to be in a cask and the robustness of the casks, the frequency of the drop is assumed to be beyond extremely unlikely. MAR is assumed to be the amount of fuel in 1 spent fuel assembly that has cooled for 220 days. The DR is assumed to be 1. The bounding ARF and RF values are from Table D–9. For isotopes not listed in Table D–9, an ARF of 4×10^{-5} and an RF of 1 for aerodynamic entrainment and resuspension of surface contaminated waste is used (DOE 1994). Reduction in the source term due to experiment hall confinement is modeled (an LPF of 0.005 is used). Table D–10 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.3.5.2.4 Versatile Test Reactor Seismic Event Resulting in Collapse of the Experiment Hall

One accident scenario considers the consequences of a severe seismic event that would collapse the VTR experiment hall. Releases based upon the collapse of the experiment hall would necessarily be limited to the fresh and spent fuels. These fuels would be on their way into or out of the reactor. Given the robust structural protections provided by the Reactor Head Access area floor and other SDC-3 components, the VTR design would meet the satisfaction of the three key safety functions in a passive mode under NPH conditions. This scenario assumes 12 fuel assemblies stored in casks and 6 fuel assemblies in the process of being unloaded or washed.

The release occurs at ground level over 10 minutes. The assumed seismic accident frequency is beyond extremely unlikely. MAR is the equivalent of 18 spent fuel assemblies with the inventory decayed for 220 days. The DR is 0.1, because the building collapse would not be expected to cause extensive damage to spent fuel in casks or in the cleaning station. The ARF and RF for noble gases are 1. For the remaining radionuclides in MAR, an ARF of 1×10^{-3} with an RF of 1.0 is assessed to bound the suspension of (set the limits of) surface contamination from non-brittle solid material for this phenomenon (DOE 1994). The LPF is 1. Confinement in the experiment hall is not modeled. Depending on the location of the drop, nearby workers could inhale the airborne radioactive materials before evacuating the area. Table D–10 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.4 Spent Fuel Handling and Treatment Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

The selection of accidents for spent fuel handling and treatment at either INL or ORNL considers various sources of background information. Because spent fuel treatment involving assemblies or fuel pins occurs in a hot cell filled with inert gas, no releases occur unless the hot cell confinement is breached. Release of fission product gases is anticipated as part of the facility design due to the release that occurs when the spent fuel is melted. The analysis for these accidents is presented in Sections D.3.4.1 through D.3.4.2.

Table D–10. Accident Scenarios and Source Terms for Versatile Test Reactor Operations at Idaho National Laboratory/Materials and Fuels Complex and Oak Ridge National Laboratory

Accident Scenario	*MAR*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term Number of Assemblies*
VTR Operational Accidents								
D.3.3.5.1.1 Cover Gas System Failure with Intact Confinement	3 fuel pins	Gases See Table D–6.	Unlikely	0.5 for Cs, 1 for other gases	1.0	1	6×10^{-5}	Not Applicable
D.3.3.5.1.2 Cover Gas System Failure with Failed Confinement	10 fuel pins	See Table D–7.	Extremely Unlikely	0.5 for Cs, 1 for other gases	1.0	1	1	Not Applicable
Spent Fuel Accidents in the VTR Experiment Hall								
D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire in Experiment Hall	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	220-day cooled spent fuel assembly See Table D-42.	Extremely Unlikely	1	Noble gases: 1; Pu, Am, and other Actinides: 3×10^{-5} U, Activation Products, Fission Products: 1×10^{-3}	0.04 for Pu, Am, and other Actinides 1 for remaining	1	4.61×10^{-3}
D.3.3.5.2.2 Fire at Eutectic Temperature Involving VTR Fuel Assemblies	3 assemblies	220-day cooled spent fuel assembly See Table D-42.	Extremely Unlikely to Beyond Extremely Unlikely	1	Noble Gases: 0.67 Halogens: 0.1 Cs/Rb: 0.42 Te/Sb: 6×10^{-3} Ba/Sr: 0.24 Noble Metals: 6×10^{-4} Ce/Pu: 6×10^{-4} Lanth: 6×10^{-5} Eu: 0.42 Rest 1×10^{-3}	1	1	3
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	1 assembly	220-day cooled spent fuel assembly See Table D-42.	Beyond Extremely Unlikely	1	Noble Gas: 0.4 Halogens: 0.003 Cs/Rb: 0.003 Te/Sb: 1×10^{-4} Ba/Sr: 6×10^{-7} Noble Metals: 6×10^{-7} Ce/Pu: 6×10^{-7} Lanth: 6×10^{-7} Rest: 4×10^{-5}	1	0.005	1

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	DR	ARF [a]	RF	LPF	Source Term Number of Assemblies
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	12 assemblies in casks and 6 assemblies being unloaded or washed. Total MAR 18	220-day cooled spent fuel assembly See Table D-42.	Beyond Extremely Unlikely	0.1	Noble Gas: 1 Halogens: 1×10^{-3} Cs/Rb: 1×10^{-3} Te/Sb: 1×10^{-3} Ba/Sr: 1×10^{-3} Noble Metals: 1×10^{-3} Ce/Pu: 1×10^{-3} Lanth: 1×10^{-3} Eu: 1×10^{-3}	1	1	1.8

Am = americium; ARF = airborne release fraction; Ba = barium; Ce = cerium; Cs = cesium; DR = damage ratio; Eu = europium; HRS = heat removal system; KIS = K Area Interim Surveillance; Lanth = lanthanum; LPF = leak path factor; MAR = material at risk; Pu = plutonium; Rb = rubidium; RF = respirable fraction; RVACS = Reactor Vessel Auxiliary Cooling System; Sb = antimony; Sr = strontium; U = uranium; VTR = Versatile Test Reactor.

[a] Isotope groups are defined in Tables D-8 and D–9.

D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)

A scenario is postulated to examine the consequences of an inadvertent nuclear criticality while spent nuclear fuel is being treated. The criticality results from a seismic event that also causes failure of confinement. The source term for this criticality is based on a fission yield of 1.0×10^{19} fissions. This source term, which is used for all facilities, is based on that given in DOE-HDBK-3010-94 (DOE 1994). Criticality safety controls should prevent this accident from occurring and the materials involved should remain at less than a critical mass. Thus, this accident is identified as extremely unlikely. The scenario represents a metal criticality. A 100 percent release of fission products generated in the criticality is assumed. However, the scenario did not postulate aerosolized, respirable metal fragments to be released. Engineered and administrative controls should be available to ensure that the double-contingency principles are in place for all portions of the process. For purposes of this VTR EIS, the DR is 1, and the bounding ARF and RF value of 1 is assumed for noble gases. Deposition during transport through the building or debris is not assumed, so an LPF of 1 is modeled. **Table D–11** summarizes MAR, release fractions, and source term for this accident scenario.

Table D–11. Accident Scenarios and Terms for Spent Fuel Handling and Treatment Activities at Oak Ridge National Laboratory and Idaho National Laboratory/Materials and Fuels Complex

Accident Scenario	*MAR grams of Pu or other*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term Number of Ingots*
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44	Extremely Unlikely	1	1.0	1	1	See Table D-44
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	1 molten metal spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1	Noble Gas: 1 other: 2×10^{-5}	1	1	1
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	1 spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1	Noble Gas: 1 Pu, Am, and other Actinides: 3×10^{-5} U, Activation Products, Fission Products: 1×10^{-3}	0.04 for Pu, Am, and other Actinides 1 for remaining	1	1

Am = americium; ARF = airborne release fraction; DR = damage ratio; LPF = leak path factor; MAR = material at risk; Pu = plutonium; RF = respirable fraction U = uranium.

[a] Isotope groups are defined in Table D–8 and Table D–9.

D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure

A molten fuel spill into the spent fuel treatment hot cell is assumed to occur when an earthquake causes failure of the hot cell confinement. The frequency of the earthquake and event is extremely unlikely. The hot cell enclosure and the offgas exhaust ventilation system would be expected to fail and allow the release of radioactive material released in the spill. MAR is the amount of fuel in one fuel assembly. The fuel is postulated to have a decay time of 4 years. The DR is assumed to be 1. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For the remaining radionuclides in MAR, the bounding ARF and RF values of 2.0×10^{-5} and 1, respectively are based on a free-fall spill of aqueous solutions (<3 m) with a density >1.2 g/cm^3 from DOE-HDBK-3010-94 (DOE 1994). There is no reduction in the source term because of hot cell confinement; therefore, an LPF of 1 is used. Table D–11 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure

The fire selected for analysis would affect the sodium and fuel material in one spent fuel assembly. The event is assumed to occur when an earthquake causes confinement failure in conjunction with the drop of an assembly that breaches the assembly cladding. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. The frequency of the sodium fire is extremely unlikely. MAR is assumed to be the amount of fuel in one assembly. The fuel is postulated to have a decay time of 4 years, and the DR is 1. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For the plutonium, americium, and other actinides in MAR, the bounding ARF and RF values are 3×10^{-5} and 0.04, respectively (DOE 1994). For uranium, activation products, and fission products in MAR, the bounding ARF and RF values are 1×10^{-3} and 1, respectively (DOE 1994). There is no reduction in the source term based on confinement, and an LPF of 1 is used. Table D–11 summarizes MAR, release fractions, and source term for this accident scenario.

D.3.5 Post-Irradiation Examination Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

The selection of accidents for post-irradiation examination at either INL or ORNL considers various sources of background information. Because post-irradiation examination involving assemblies or fuel pins occurs in a hot cell filled with inert gas, no releases occur unless the hot cell confinement is breached. Release of fission product gases is anticipated as part of the facility design due to the release that occurs when the experiment is prepared for further examination.

D.3.5.1 Fire Involving Test Assembly (Seismically-Induced Confinement Failure)

The fire selected for analysis impacts the test assembly during post-irradiation examination. The plume rise option is not activated for this scenario. Using a ground-level release for the fire provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS. Because post-irradiation examination is performed in a hot cell with an inert argon atmosphere, a fire would not occur unless the hot cell confinement is breached. For this scenario, an earthquake is postulated to breach the hot cell confinement. The frequency of the earthquake is extremely unlikely. MAR is assumed to be the amount of fuel in one-half of a spent fuel assembly. Selecting one-half of a spent fuel assembly as MAR accounts for multiple VTR experiments being examined concurrently in the hot cell. The fuel is postulated to have a decay time of 4 years, and the DR is 1. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For the remaining radionuclides in MAR, the bounding ARF and RF values of 3×10^{-5} and 0.04, respectively are for airborne release of particulates during complete oxidation of metal mass (DOE 1994). The release factors are specifically for an airborne release of plutonium particulates formed by oxidation at elevated temperatures without self-sustained oxidation (DOE 1994). There is no reduction in the source term from the hot cell confinement, and an LPF of 1 is used. **Table D–12** summarizes MAR, release fractions, and source term for this accident scenario.

Table D–12. Accident Scenarios and Source Terms for Post-Irradiation Examination Activities at Idaho National Laboratory/Materials and Fuels Complex and Oak Ridge National Laboratory

Accident Scenario	*MAR Grams of Pu or Other*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term Number of Assemblies*
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	0.5 assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1	Noble Gas: 1 Other: 3×10^{-5}	0.04	1	0.5

ARF = airborne release fraction; DR = damage ratio; LPF = leak path factor; MAR = material at risk; Pu = plutonium; RF = respirable fraction.

D.3.6 Spent Fuel Storage Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

The selection of accidents for spent fuel storage at either INL or ORNL considers various sources of background information. The analysis for these accidents is presented in Sections D.3.6.1 through D.3.6.3.

D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask

After spent fuel has been washed, it is loaded into spent fuel storage casks and stored on a spent fuel storage pad pending transfer to the fuel treatment facility where it will be treated and conditioned for eventual disposal. Additionally, until a disposal site has been identified, the spent fuel storage casks containing conditioned spent fuel ingots are stored on a spent fuel storage pad. The accident is assumed to occur when an earthquake causes failure of the spent fuel storage casks. The release occurs because of damage to the contents of several spent fuel storage casks. The release occurs at ground level over 10 minutes. The assumed accident frequency is beyond extremely unlikely. MAR is the equivalent of six spent fuel assemblies with an inventory decay time of 4 years, and the DR is 0.5, because damage to material in the cask is not expected to be extensive. For purposes of this VTR EIS, the ARF and RF value of 1 is assumed for noble gases. For the remaining radionuclides in MAR, an ARF of 4×10^{-5} and an RF of 1 for aerodynamic entrainment and resuspension of surface contaminated waste is used (DOE 1994). The LPF is assumed to be 1. The source term is not reduced because of the confinement provided by a damaged spent fuel storage cask. Depending on the location of the drop, nearby workers could inhale the airborne radioactive materials before evacuating the area. **Table D–13** summarizes MAR, release fractions, and source term for this accident scenario.

Table D–13. Accident Scenarios and Source Terms for Spent Fuel Storage Activities at Idaho National Laboratory/Materials and Fuels Complex and Oak Ridge National Laboratory

Accident Scenario	*MAR*	*Assumed Material Type*	*Frequency (per year)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term Number of Assemblies*
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	6 spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43.	Beyond Extremely Unlikely	0.5	Noble Gas: 1 Halogens: 4×10^{-5} Cs/Rb: 4×10^{-5} Te/Sb: 4×10^{-5} Ba/Sr: 4×10^{-5} Noble Metals: 4×10^{-5} Ce/Pu: 4×10^{-5} Lanth: 4×10^{-5} Eu: 4×10^{-5}	1	1	3
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	1.0×10^{19} fissions, Three spent fuel assemblies	Nobles+ Iodine See Table D-44.	Extremely Unlikely	1	1.0	1	1	See Table D-44
D.3.6.3 Drop of Fuel-Loaded Cask	6 spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43.	Extremely Unlikely	0.5	Noble Gas: 1 Halogens: 4×10^{-5} Cs/Rb: 4×10^{-5} Te/Sb: 4×10^{-5} Ba/Sr: 4×10^{-5} Noble Metals: 4×10^{-5} Ce/Pu: 4×10^{-5} Lanth: 4×10^{-5} Eu: 4×10^{-5}	1	0.5	3

ARF = airborne release fraction; Ba = barium; Ce = cerium; Cs = cesium; DR = damage ratio; Eu = europium; Lanth = lanthanum; LPF = leak path factor; MAR = material at risk; Pu = plutonium; Rb = rubidium; RF = respirable fraction; Sb = antimony; Sr = strontium; Te = tellurium.

[a] Isotope groups are defined in Table D–8 and Table D–9.

D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask

An inadvertent nuclear criticality is postulated to occur in untreated fuel from a spent fuel storage cask. The criticality results from a seismic event that also causes failure of the spent fuel storage cask. The source term for this criticality is based on a fission yield of 1.0×10^{19} fissions. This source term, which is used for all facilities, is based on that given in DOE-HDBK-3010-94 (DOE 1994). The scenario represents a metal criticality. The metal is postulated to soften, resulting in a 100 percent release of fission products stored in the fuel and fission products generated in the criticality. The fission products stored in the fuel are from three spent fuel assemblies with a decay time of four years. However, no aerosolized, respirable metal fragments are predicted to be released. This accident is conservatively estimated to have a frequency of extremely unlikely. For purposes of this VTR EIS, the DR is 1, and the bounding ARF and RF value of 1 is assumed for noble gases. Deposition debris from spent fuel storage cask does not reduce the source term, so an LPF of 1 is assumed. Table D–13 summarizes MAR, release fractions, and source term for this accident scenario

D.3.6.3 Drop of Fuel-Loaded Cask

After spent fuel has been washed, it is loaded into spent fuel storage casks pending transfer to the fuel treatment site. The accident is assumed to occur when a spent fuel storage cask is dropped during handling. The cask drop is assumed to be caused by equipment failure or human error. The release occurs from damaging the contents of one spent fuel storage cask. The release occurs at ground level over 10 minutes. The assumed accident frequency is extremely unlikely. MAR is the equivalent of six spent fuel assemblies. The DR is assumed to be 0.5 because not all fuel in the cask is expected to be involved. For purposes of this VTR EIS, the bounding ARF and RF value of 1 is assumed for noble gases. For the remaining radionuclides in MAR, an ARF of 4×10^{-5} and an RF of 1 for aerodynamic entrainment and resuspension of surface contaminated waste is used (DOE 1994). The dropped cask is assumed to continue providing some confinement after the drop. Consequently, the LPF is assumed 0.5. Depending on the location of the drop, nearby workers could inhale the airborne radioactive materials before evacuating the area. Table D–13 summarizes MAR, release fractions, and source term for this accident scenario.

D.4 Radiological Impacts of Facility Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

The following sections summarize the consequence of accidents described in Sections D.3.1 through D.3.6. The radiological and hazardous material consequences are presented for accidents at the SRS, INL, and ORNL. Because few sources of energy and only limited materials would be present, the likelihood of a major accident is extremely remote. Most incidents would not involve much energy. Any criticality, fire, drop, or spill would be confined, with little or no radiological impact. For bounding accidents, radiological impacts on workers in the immediate vicinity of the incident and on those exposed to released material could be relatively high. The radiological impacts from beyond-design-basis earthquakes on involved and noninvolved workers could be high as well.

For most of the proposed activities, accidents that could potentially affect noninvolved workers, the maximally exposed offsite individual, and the public are extremely unlikely or beyond extremely unlikely, i.e., an annual probability of one in 10,000 or less. Accidents with releases of this magnitude are not expected to occur during the lifetime of the project. Impacts on the noninvolved worker are modeled assuming that worker is 330 feet downwind of the release point. Dispersion modeling of impacts so close to a release point are highly dependent on assumptions and does not include protection of buildings and other structures, turbulence, personnel taking emergency actions, etc. As a result, these models are, therefore, highly uncertain. They do offer an opportunity to compare the relative impacts of each accident but are much less useful for comparing impacts at different sites because any real differences due to meteorology or location would be small compared to the uncertainties in modeling close-in effects.

The potential near-term impacts from the initial plume passage are reported as the "Near-Term-Dose" in the following tables while the long-term impacts of exposure to the radionuclides after the plume passage are added to the "Near-Term-Dose" and reported as the "Near+Long-Term Dose." The long term (or chronic) dose includes the combined effects of ingesting contaminated foods, direct radiation exposure from residual material on the ground (ground shine), inhalation of disturbed, residual ground-level particulates (resuspension), and ingestion of contaminated water.

D.4.1 Radiological Impacts from Accidents at Idaho National Laboratory and Oak Ridge National Laboratory Versatile Test Reactor-Related Facilities

Table D–14 through **Table D–19** summarize the impacts related to various accident scenarios for reactor fuel production, TRU waste handling, VTR operation, spent fuel handling, post-irradiation examination, and spent fuel storage at MFC.

D.4.1.1 Accident Impacts from VTR Fuel Production Capability at Idaho National Laboratory

Two phases of reactor fuel production are evaluated. The first phase is a feedstock preparation capability. This phase involves the receipt of plutonium that contains high levels of impurities, e.g., americium-241 (which is present as the result of the decay of plutonium-241 in "older" plutonium) or plutonium in other than metal form (e.g., plutonium oxide). The feedstock preparation capability would address the issues of plutonium polishing and conversion to metal. The second phase of reactor fuel production would involve the alloying of plutonium with uranium and zirconium, and fabricating fuel pins and fuel assemblies.

Table D–14 presents the potential impacts from accidents associated with implementation of VTR fuel production capability at INL. For the VTR fuel fabrication activities at either the INL Site or SRS, the bounding operational accident is a fire that results in heating and over pressurization of a 3013 can of plutonium oxide. Releases from other accidents such as a fire and spill involving molten uranium and plutonium while being cast into fuel would be filtered before release to the environment. During all of the VTR fuel fabrication operations, the radiological materials are either in metal form in a container designed to contain the radionuclides in virtually all accidents or in an inert glovebox. As such, only controlled, filtered releases would be expected. Workers would be protected from routine accidents within the gloveboxes and the ventilation system that directs glovebox releases through HEPA filters and to an outside stack.

For accidents associated with the INL VTR metal preparation portion of the fuel fabrication option, the bounding accident from plutonium "polishing" operations would be an uncontrolled reaction during portions of the aqueous operations. This could release radioactive materials to the glovebox, but any releases to from the glovebox or room to the environment would be filtered and have low impacts. If plutonium oxide were used as a feed material, the bounding operational event is a high-pressure rupture of a welded, DOE standard 3013, plutonium-oxide storage container. This could occur if the container were exposed to a fire that burns sufficiently long to raise the internal pressure of the container to the point of rupture. This scenario, while theoretically possible, would be extremely unlikely and normal fire prevention and mitigation practices should reduce the chance of it occurring.

In a severe seismic event, the building and glovebox structures could be damaged enough that an unfiltered release could occur. Even so, most of the material not specifically being processed at the time of the earthquake would be in metal form and would not result in a substantial release even with loss of glovebox and building integrity.

A beyond-design-basis earthquake was also postulated that could threaten unfiltered releases of all of the MAR in the Fuel Preparation and Fabrication area, including the molten plutonium in casting, liquid plutonium in the polishing operation, and plutonium oxide in the conversion operations. In a severe seismic event, the building and glovebox structures could be damaged enough that an unfiltered release could occur. Even so, most of the material not specifically being processed at the time of the earthquake would be in metal form and would not result in a substantial release even with loss of glovebox and building integrity.

D.4.1.2 Accident Impacts from VTR Transuranic Waste Activities at Idaho National Laboratory

Table D–15 presents the potential impacts from accidents associated with implementation of VTR fuel production capability at INL. The bounding accident is a fire outside of a building's confinement system involving americium-241 waste. The waste container would be designed to prevent such fires.

D.4.1.3 Accident Impacts from Reactor Operations at Idaho National Laboratory

Table D–16 presents the potential impacts from accidents associated with operation of the VTR at INL. Both reactor-specific and VTR building accidents are considered. The potential impacts of a hypothetical, beyond-design-basis reactor accident with loss of cooling are presented in Section D.4.9.

D.4.1.3.1 Reactor Accidents

As discussed in Section D.3.3.5.1, the results of the safety analyses for the VTR based on the conceptual design (INL 2019, 2020c), as well as the preliminary VTR PRA (GEH 2019) indicate that most of the accident sequences do not lead to a release of radionuclides from the reactor. The only accidents with potential releases at probabilities of 10^{-6} per year or greater are leaks from the cover gas system which lead to minor accidental releases and are not evaluated in detail.

D.4.1.3.2 Reactor Building Fuel Accidents

As discussed in Section D.3.3.5.2, the results of the safety analyses indicate that accidents associated with operations involving irradiated fuel assemblies and irradiated test assemblies in the experiment hall area of the reactor building could occur and bound, in terms of consequences, all reactor-area accidents. As indicated in Table D–16, all of the accidents evaluated involve spent fuel assemblies or test assemblies. Most handling accidents involving these assemblies would not result in substantial releases. The bounding accidents evaluated are assumed to be initiated by a major event such as a major fire or severe, beyond-design-basis seismic event. As indicated in the Table D–16, the largest impacts are associated with a major fire that heats spent fuel assemblies inside a safety-class cask (designed to withstand fires) to the point of failure of the cask. This combination of events is expected to fall in the extremely unlikely to beyond extremely unlikely frequency category since it requires multiple failures of safety systems. The potential radiological impact, are relatively high, with a dose of 0.24 Rem to the MEI and a near-term population dose of 36 person-rem. The fire, assumed to be 800 degrees Celsius (1,470 degrees Fahrenheit), is sufficient high to release relatively high fractions of isotopes of elements such as cesium and strontium that could result in high doses if ingested. Without mitigation measures, the long term dose due primarily to ingestion is 4,300 person-rem or 3 LCFs among the 50-mile population. It is expected that as the design of the VTR fuel handling systems within the experiment hall progresses, designs with ensure that this accident is prevented or mitigated. The other bounding accident is a seismic event that results in collapse of the experiment hall. Because all of the spent fuel would be stored in safety class casks, only spent fuel being handled might be vulnerable. The potential radiological impact, are moderately high, with a dose of 0.071 rem to the MEI and a near-term population dose of 13 person-rem. Without mitigation measures, the long term dose due to resuspension and ingestion is 27 person-rem and no LCFs among the 50-mile population.

D.4.1.4 Accident Impacts from Treatment at Idaho National Laboratory

Table D–17 presents the potential impacts from accidents associated with spent fuel handling and treatment at MFC. Because operations occur in a hot cell designed to contain high quantities of radionuclides, uncontrolled releases to the environment do not occur unless there is a major breach of confinement. Accidental handling and spills within the hot would not result in uncontrolled releases. Each of the accident scenarios selected for further evaluation requires breach of hot cell confinement, likely be a major seismic event. Such an event could lead to ingress of oxygen into the hot cell and sodium fires. The highest impacts are associated with a seismically-initiated hot-cell confinement failure and sodium fire involving a spent fuel assembly and cladding. Such an event is considered extremely unlikely.

D.4.1.5 Accident Impacts from Post-Irradiation Examination at Idaho National Laboratory

Table D–18 presents the potential impacts from accidents associated with post-irradiation examination at MFC. Because operations occur in a hot cell designed to contain high quantities of radionuclides, uncontrolled releases to the environment do not occur unless there is a major breach of confinement. The accident scenarios selected for further evaluation requires breach of hot cell confinement, likely be a major seismic event. Such an event could lead to ingress of oxygen into the hot cell and a fire with a test assembly under examination. Such an event is considered extremely unlikely.

D.4.1.6 Accident Impacts from Spent Fuel Storage at Idaho National Laboratory

Table D–19 presents the potential impacts from accidents associated with spent fuel handling and storage at MFC. VTR spent fuel would be stored in robust canisters designed to withstand a wide-range of accidents. As such, most types of handling accidents, including impacts and handling accidents, would not be expected to result in releases. The accidents identified result in small releases.

D.4.2 Radiological Impacts from Accidents at Versatile Test Reactor-Related Facilities at Oak Ridge National Laboratory

Table D–20 through **Table D–24.** summarize the impacts related to various accident scenarios for TRU waste handling, VTR operation, spent fuel handling, post-irradiation examination, and spent fuel storage at ORNL. Because few sources of energy and only limited materials would be present, the likelihood of a major accident is extremely remote. Most incidents would not involve much energy, and any criticality, fire, drop, or spill would be confined, with little or no radiological impact. For the bounding accidents, radiological impacts on workers in the immediate vicinity of the incident and on those exposed to released material could be relatively high. The radiological impacts from beyond-design-basis earthquakes on noninvolved workers could be high as well.

The accident scenarios for the VTR alternative and supporting operations at ORNL are identical to those presented for the comparable facilities at INL. No new or substantially different accident scenarios were identified with the placement of the facilities at ORNL. While there are differences in the sites that could affect the probabilities of certain scenarios, particularly natural phenomena-initiate events such as wind, flooding, seismic, and volcanism, the dominant accident scenarios remain the same.

There are, however, differences in potential impacts due to population distributions around the VTR location, distances to the nearest potential offsite individual, and differences in meteorological conditions. Impacts on the maximally exposed member of the public, assumed to be a boater on an arm of Melton Lake about 0.5 miles from the VTR complex are higher than those for the INL site, where the MEI is assumed be on the nearby highway about 3.1 miles away. The offsite population density also is quite different between the two VTR sites. Most of the differences in population impacts between the two sites are directly due to a lower 50-mile population at INL and few people residing within 10 miles of the proposed VTR site.

D.4.2.1 Accident Impacts from VTR Transuranic Waste Activities at Oak Ridge National Laboratory

Table D–20 presents the potential impacts from accidents associated with implementation of VTR fuel production capability at ORNL.

D.4.2.2 Accident Impacts from Reactor Operations at Oak Ridge National Laboratory

Table D–21 presents the potential impacts from accidents associated with operation of the VTR at ORNL. The accident scenarios for the VTR alternative and supporting operations at ORNL are identical to those presented for the comparable facilities at INL. No new or substantially different accident scenarios were identified with the placement of the facilities at ORNL. While there are differences in the sites that could affect the probabilities of certain scenarios, particularly natural phenomena-initiate events such as wind, flooding, seismic, and volcanism, the dominant accident scenarios remain the same. The VTR and support facilities would be designed to the same standards at either site. For natural-phenomena events, the standards are based to return intervals or frequency per year. Thus a VTR at ORNL may be designed to a higher seismic loading in order to meet the DOE standards. The likelihood of beyond-design-basis events should be the same for both sites.

D.4.2.3 Accident Impacts from VTR Support Activities at Oak Ridge National Laboratory

Table D–22, D–23, and D–24 presents the potential impacts from accidents associated VTR support activities, including spent fuel handling and treatment, post-irradiation examination, and spent fuel storage at ORNL. The accident scenarios are identical to those identified for INL.

D.4.3 Radiological Impacts from Accidents at Versatile Test Reactor-Related Fuel Production Activities at K-Reactor Complex at Savannah River Site

Table D–25 and **D–26** summarize the impacts related to various accident scenarios for fuel production at SRS. The accident scenarios for the VTR fuel fabrication operations at SRS are identical to those presented for the comparable facilities at INL. No new or substantially different accident scenarios were identified with the placement of the facilities at SRS. While there are differences in the sites that could affect the probabilities of certain scenarios, particularly natural phenomena-initiate events such as wind, flooding, seismic, and volcanism, the dominant accident scenarios remain the same.

There are, however, differences in potential impacts due to population distributions around the VTR location, distances to the nearest potential offsite individual, and differences in meteorological conditions.

Because few sources of energy and only limited materials would be present, the likelihood of a major accident is extremely remote. Most incidents would not involve much energy, and any criticality, fire, drop, or spill would be confined, with little or no radiological impact. For the bounding accidents, radiological impacts on workers in the immediate vicinity of the incident and on those exposed to released material could be relatively high. The radiological impacts from beyond-design-basis earthquakes on involved and noninvolved workers could be high as well. Impacts on the noninvolved worker are included for the receptor at 330 feet and 3,300 feet. The impacts for the receptor at 330 feet are presented for consistency with the other calculations in this VTR EIS, while the impacts for the receptor at 3,300 feet are presented for historical consistency. Previous SRS EIS documents reported calculation results for a noninvolved worker at 3,300 feet.

Table D–14. Accident Impacts for the VTR Fuel Production Capability at Idaho National Laboratory

Accident Scenario	MAR Grams of Pu or Other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.1.1 Criticality while alloying the three components of the metal fuel (uranium, plutonium, and zirconium)	1.0×10^{19} fissions	Nobles+Iodine See Table D-44	Extremely Unlikely	4.8×10^{-2}	3×10^{-5}	3.6×10^{-3}	2×10^{-6}	1.3×10^{-1}	8×10^{-5}	7.7×10^{-1}	5×10^{-4}
D.3.1.2 Fire Impingement on Fuel Material (intact confinement)	5,000	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely	1.6×10^{-5}	1×10^{-8}	6.4×10^{-6}	4×10^{-9}	1.1×10^{-3}	7×10^{-7}	1.6×10^{-3}	9×10^{-7}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	5,000	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	4.8×10^{-2}	3×10^{-5}	6.5×10^{-5}	4×10^{-8}	1.0×10^{-2}	6×10^{-6}	1.5×10^{-2}	9×10^{-6}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	4,500	5,090 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	9.0	5×10^{-3}	1.2×10^{-2}	7×10^{-6}	2.0	1×10^{-3}	2.9	2×10^{-3}
D.3.1.5 Plutonium Oxide-to-Metal Conversion–Explosion of 3013 Container of PuO_2	4,500	5,090 g KIS grade PuO_2 in one 3013 container	Extremely Unlikely	2.7×10^{-1}	2×10^{-4}	1.1×10^{-1}	6×10^{-5}	18	1×10^{-2}	26	2×10^{-2}
D.3.1.6 Beyond-Design-Basis Fire Involving a TRU Waste Drum	398	450 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	1.6	1×10^{-3}	2.2×10^{-3}	1×10^{-6}	0.35	2×10^{-4}	0.51	3×10^{-4}
D.3.1.7 Aqueous/ Electrorefining Fuel Preparation	1,000 resin 246 nitrate solution 3,500 electro-refiner	KIS-grade Pu	Extremely Unlikely	5.2×10^{-4}	3×10^{-7}	2.0×10^{-4}	1×10^{-7}	3.4×10^{-2}	2×10^{-5}	5.0×10^{-2}	3×10^{-5}

Accident Scenario	MAR Grams of Pu or Other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.1.8 Aircraft Crash into VTR Fuel Production Facility	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	8.3×10^{2}	1	1.1	7×10^{-4}	1.8×10^{2}	1×10^{-1}	2.7×10^{2}	2×10^{-1}
D.3.1.9 Beyond-Design-Basis Earthquake and Fire Involving All of VTR Fuel Production MAR	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	8.3×10^{2}	1	1.1	7×10^{-4}	1.8×10^{2}	1×10^{-1}	2.7×10^{2}	2×10^{-1}

g = gram; kg = kilogram; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; Pu = plutonium; PuO_2 = plutonium oxide; rem = roentgen equivalent man; TRU = transuranic; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–2. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Near+Long-Term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–15. Accident Impacts from VTR Transuranic Waste Activities at Idaho National Laboratory

Accident Scenario	MAR Grams of Pu or Other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.2.1 Fire Outside Confinement (waste from fuel production or spent fuel treatment)	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	6.0×10^{-5}	4×10^{-8}	7.4×10^{-8}	4×10^{-11}	1.3×10^{-5}	8×10^{-9}	2.6×10^{-5}	2×10^{-8}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	3,346	Pu-239 Equivalent	Unlikely	8.0×10^{-2}	5×10^{-5}	1.1×10^{-4}	7×10^{-8}	1.8×10^{-2}	1×10^{-5}	1.8×10^{-2}	1×10^{-5}

Am = americium; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man; Pu = plutonium; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–3. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less.

Table D–16. Accident Impacts for VTR Accidents at Idaho National Laboratory/Materials and Fuels Complex

Accident Scenario	*MAR*	*Assumed Material Type*	*Frequency (per year)*	*Impacts on Noninvolved Worker (330 feet)*		*Impacts on an MEI at 3.1 Miles*		*Near-Term Impacts on Population within 50 Miles*		*Near+Long-Term Impacts on Population within 50 Miles*	
				Dose (rem) [a]	*Probability of an LCF* [b]	*Dose (rem)* [a]	*Probability of an LCF* [b]	*Early Dose (rem)* [a]	*LCFs* [c]	*Early + Chronic Dose (rem)* [a]	*LCFs* [c]
D.3.3.5.1.1 Cover Gas System Failure with Intact Confinement	3 fuel pins	gases see Table D–7	Unlikely	The source term for this scenario is very small in relation to the source terms for other VTR accidents and no calculation is performed since the results from other calculations are enveloping.							
D.3.3.5.1.2 Cover Gas System Failure with Failed Confinement	10 fuel pins	see Table D–8	Extremely Unlikely	The source term for this scenario is very small in relation to the source terms for other VTR accidents and no calculation is performed since the results from other calculations are enveloping.							
D.3.3.5.2.1 Test Assembly Failure Following Seismically-Induced Fire	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	220-day cooled spent fuel assembly See Table D-42	Extremely Unlikely	5.2×10^{-2}	3×10^{-5}	6.5×10^{-5}	4×10^{-8}	1.2×10^{-2}	7×10^{-6}	3.6×10^{-2}	2×10^{-5}
D.3.3.5.2.2 Release from Fire at Eutectic Temperature Involving VTR Fuel Assemblies	3 assemblies	220-day cooled spent fuel assembly See Table D-42	Extremely Unlikely to Beyond Extremely Unlikely	1.6×10^{2}	2×10^{-1}	2.4×10^{-1}	1×10^{-4}	3.6×10^{1}	2×10^{-2}	4.3×10^{3}	3
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	1 assembly	220-day cooled spent fuel assembly See Table D-42	Beyond Extremely Unlikely	7.3×10^{-4}	4×10^{-7}	1.3×10^{-6}	8×10^{-10}	1.7×10^{-4}	1×10^{-7}	4.6×10^{-2}	3×10^{-5}
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	12 assemblies in casks and 6 assemblies being unloaded or washed. Total MAR 18	220-day cooled spent fuel assembly See Table D-42	Beyond Extremely Unlikely	5.8×10^{1}	7×10^{-2}	7.1×10^{-2}	4×10^{-5}	1.3×10^{1}	8×10^{-3}	2.7×10^{1}	2×10^{-2}

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]

HRS = heat removal system; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man; RVACS = Reactor Vessel Auxiliary Cooling System; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–11. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less.

Table D–17. Accident Impacts for the Spent Fuel Handling and Treatment at Idaho National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44	Extremely Unlikely	1.0	6×10^{-4}	3.9×10^{-3}	2×10^{-6}	1.3×10^{-1}	8×10^{-5}	7.6×10^{-1}	5×10^{-4}
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	1 molten metal spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	6.6×10^{-1}	4×10^{-4}	8.0×10^{-4}	5×10^{-7}	1.4×10^{-1}	9×10^{-5}	2.9×10^{-1}	2×10^{-4}
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	1 spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	9.0	5×10^{-3}	1.1×10^{-2}	7×10^{-6}	2.0	1×10^{-3}	7.8	5×10^{-3}

INL= Idaho National Laboratory; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–12. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–18. Accident Impacts for Post-Irradiation Examination at Idaho National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	One-half of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1.6×10^{-2}	9×10^{-6}	1.9×10^{-5}	1×10^{-8}	3.5×10^{-3}	2×10^{-6}	6.9×10^{-3}	4×10^{-6}

INL= Idaho National Laboratory; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–13. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–19. Accident Impacts for Spent Fuel Storage at Idaho National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 3.1 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	Six spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43	Beyond Extremely Unlikely	3.1	2×10^{-3}	3.9×10^{-3}	2×10^{-6}	6.9×10^{-1}	4×10^{-4}	1.4	8×10^{-4}
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44 4-year cooled spent fuel assembly	Extremely Unlikely	1.0	6×10^{-4}	3.9×10^{-3}	2×10^{-6}	1.3×10^{-1}	8×10^{-5}	7.6×10^{-1}	5×10^{-4}
D.3.6.3 Drop of Fuel-Loaded Cask	Six spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1.6	9×10^{-4}	1.9×10^{-3}	1×10^{-6}	3.5×10^{-1}	2×10^{-4}	6.9×10^{-1}	4×10^{-4}

LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–14. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–20. Accident Impacts from VTR Transuranic Waste Activities at Oak Ridge National Laboratory

Accident Scenario	MAR grams of Pu or other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.2.1 Fire Outside Confinement (Waste from fuel production or spent fuel treatment)	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1.5×10^{-4}	9×10^{-8}	5.4×10^{-6}	3×10^{-9}	5.2×10^{-4}	3×10^{-7}	7.2×10^{-4}	4×10^{-7}
D.3.2.5 Fire Outside Involving a Waste Drum with 23 grams of Am-241	3,346	Pu-239 Equivalent	Unlikely	2.1×10^{-1}	1×10^{-4}	7.2×10^{-3}	4×10^{-6}	0.69	4×10^{-4}	0.69	4×10^{-4}

INL= Idaho National Laboratory; Kg = kilogram; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; Pu = plutonium; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man; Pu = plutonium; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–3. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–21. Accident Impacts for VTR Accidents at Oak Ridge National Laboratory

Accident Scenario	*MAR*	*Assumed Material Type*	*Frequency (per year)*	*Impacts on Noninvolved Worker (330 feet)*		*Impacts on an MEI at 0.5 Miles*		*Near-Term Impacts on Population within 50 Miles*		*Near+Long-Term Impacts on Population within 50 Miles*	
				Dose (rem) [a]	*Probability of an LCF* [b]	*Dose (rem)* [a]	*Probability of an LCF* [b]	*Early Dose (rem)* [a]	*LCFs* [c]	*Early + Chronic Dose (rem)* [a]	*LCFs* [c]
D.3.3.5.1.1 Cover Gas System Failure with Intact Confinement	Three fuel pins	gases see Table D–6	Unlikely	The source term for this scenario is very small in relation to the source terms for other VTR accidents and no calculation is performed since the results from other calculations are enveloping.							
D.3.3.5.1.2 Cover Gas System Failure with Failed Confinement	Ten fuel pins	see Table D–7	Extremely Unlikely	The source term for this scenario is very small in relation to the source terms for other VTR accidents and no calculation is performed since the results from other calculations are enveloping.							
D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire	1 pin or 1/217 (or 4.6×10^{-3}) of an assembly	220-day cooled spent fuel assembly See Table D-42	Extremely Unlikely	1.3×10^{-1}	8×10^{-5}	4.6×10^{-3}	3×10^{-6}	4.5×10^{-1}	3×10^{-4}	7.1×10^{-1}	4×10^{-4}
D.3.3.5.2.2 Fire at Eutectic Temperature Involving VTR Fuel Assemblies	Three assemblies	220-day cooled spent fuel assembly See Table D-42	Extremely Unlikely to Beyond Extremely Unlikely	4.0×10^{2}	5×10^{-1}	1.4	9×10^{-3}	1,400	9×10^{-1}	15,000	9
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	One assembly	220-day cooled spent fuel assembly See Table D-42	Beyond Extremely Unlikely	1.9×10^{-3}	1×10^{-6}	6.7×10^{-5}	4×10^{-8}	6.7×10^{-3}	4×10^{-6}	1.3×10^{-1}	8×10^{-5}
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	12 assemblies in casks and 6 assemblies being unloaded or washed. Total MAR 18	220-day cooled spent fuel assembly See Table D-42	Beyond Extremely Unlikely	1.5×10^{2}	2×10^{-1}	5.1	3×10^{-3}	380	2×10^{-1}	700	4×10^{-1}

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]

HRS = heat removal system; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man; RVACS = Reactor Vessel Auxiliary Cooling System; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–11. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–22. Accident Impacts for the Spent Fuel Handling and Treatment at Oak Ridge National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	1.0×10^{19} fissions	Nobles+Iodine See Table D-44	Extremely Unlikely	2.5	2×10^{-3}	1.5×10^{-1}	9×10^{-5}	4.8	3×10^{-3}	6.2	4×10^{-3}
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	one molten metal spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	1.7	1×10^{-3}	5.8×10^{-2}	4×10^{-5}	5.6	3×10^{-3}	7.9	5×10^{-3}
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	one spent fuel assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	23	3×10^{-2}	8.0×10^{-1}	5×10^{-4}	77	5×10^{-2}	110	7×10^{-2}

LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–12. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–23. Accident Impacts for Post-Irradiation Examination at Oak Ridge National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	one-half of an assembly	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	4.0×10^{-4}	2×10^{-5}	1.4×10^{-3}	8×10^{-7}	1.3×10^{-1}	8×10^{-5}	1.9×10^{-1}	1×10^{-4}

LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–13. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–24. Accident Impacts for Spent Fuel Storage at Oak Ridge National Laboratory

Accident Scenario	MAR	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI at 0.5 Miles		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	Six spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43	Beyond Extremely Unlikely	8.0	5×10^{-3}	2.8×10^{-1}	2×10^{-4}	27	2×10^{-2}	138	2×10^{-2}
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	1.0×10^{19} fissions Three spent fuel assemblies	Nobles+ Iodine See Table D-44 4-year cooled spent fuel assembly	Extremely Unlikely	2.5	2×10^{-3}	1.5×10^{-1}	9×10^{-5}	4.8	3×10^{-3}	6.2	4×10^{-3}
D.3.6.3 Drop of Fuel-Loaded Cask	Six spent fuel assemblies	4-year cooled spent fuel assembly See Table D-43	Extremely Unlikely	4.0	2×10^{-3}	1.4×10^{-1}	8×10^{-5}	13	8×10^{-3}	19	1×10^{-2}

LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; rem = roentgen equivalent man.

[a] Calculated using the source terms in Table D–14. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–25. Accident Impacts for the VTR Fuel Production Capability at Savannah River Site

Accident Scenario	MAR grams of Pu or other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on Noninvolved Worker (3,300 feet)		Impacts on an MEI at Site Boundary		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.1.1 Criticality while alloying the three components of the metal fuel, uranium, plutonium, and zirconium)	1.0×10^{19} fissions	Nobles+ Iodine See Table D-44	Extremely Unlikely	4.0×10^{-2}	2×10^{-5}	2.4×10^{-2}	1×10^{-5}	1.5×10^{-3}	9×10^{-7}	8.9×10^{-1}	5×10^{-4}	1.7	1×10^{-3}
D.3.1.2 Fire Impingement on Fuel Material (Intact confinement)	5,000	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely	4.4×10^{-6}	3×10^{-9}	4.7×10^{-5}	3×10^{-8}	2.5×10^{-6}	1×10^{-9}	7.0×10^{-3}	4×10^{-6}	9.7×10^{-3}	6×10^{-6}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	5,000	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	4.7×10^{-2}	3×10^{-5}	1.3×10^{-3}	8×10^{-7}	2.5×10^{-5}	1×10^{-8}	6.8×10^{-2}	4×10^{-5}	9.4×10^{-2}	6×10^{-5}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	4,500	5,090 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	8.8	5×10^{-3}	2.4×10^{-1}	1×10^{-4}	4.6×10^{-3}	3×10^{-6}	13	8×10^{-3}	18	1×10^{-2}
D.3.1.5 Plutonium Oxide-to-Metal Conversion	4,500	5,090 g KIS grade PuO_2 in one 3013 container	Extremely Unlikely	7.2×10^{-2}	4×10^{-5}	7.8×10^{-1}	5×10^{-4}	4.1×10^{-2}	2×10^{-5}	120	7×10^{-2}	160	9×10^{-2}

Accident Scenario	MAR grams of Pu or other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on Noninvolved Worker (3,300 feet)		Impacts on an MEI at Site Boundary		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.1.6 Beyond-Design-Basis Fire Involving a TRU Waste Drum	398	450 g KIS-grade PuO_2	Extremely Unlikely to Beyond Extremely Unlikely	1.5	9×10^{-4}	4.2×10^{-2}	3×10^{-5}	8.2×10^{-4}	5×10^{-7}	2.2	1×10^{-3}	3.1	2×10^{-3}
D.3.1.7 Aqueous/ Electrorefining Fuel Preparation	1,000 resin 246 nitrate solution 3,500 electrorefiner	KIS-grade Pu	Extremely Unlikely	1.4×10^{-4}	8×10^{-8}	1.5×10^{-3}	9×10^{-7}	7.8×10^{-5}	5×10^{-8}	2.2×10^{-1}	1×10^{-4}	3.0×10^{-1}	2×10^{-4}
D.3.1.8 Aircraft Crash into VTR Fuel Production Facility	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu,	Beyond Extremely Unlikely	8.1×10^{2}	1	22	3×10^{-2}	4.3×10^{-1}	3×10^{-4}	1.2×10^{3}	7×10^{-1}	1,600	1
D.3.1.8 Beyond-Design-Basis Earthquake and Fire Involving All of VTR Fuel Production MAR	5,000 metal 4,500 molten Pu fire 4,500 container overpressure 396 TRU waste drum fire 1,000 resin + 246 liquid + 3,500 electrorefiner	KIS-grade Pu	Beyond Extremely Unlikely	8.1×10^{2}	1	22	3×10^{-2}	4.3×10^{-1}	3×10^{-4}	1.2×10^{3}	7×10^{-1}	1,600	1

g = grams; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; rem = roentgen equivalent man; SRS = Savannah River Site; Pu = plutonium; PuO_2 = plutonium oxide; TRU = transuranic; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–2. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

Table D–26. Accident Impacts from VTR Transuranic Waste Activities at Savannah River Site

Accident Scenario	MAR grams of Pu or other	Assumed Material Type	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on Noninvolved Worker (3,300 feet)		Impacts on an MEI at Site Boundary		Near-Term Impacts on Population within 50 Miles		Near+Long-Term Impacts on Population within 50 Miles	
				Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (rem) [a]	LCFs [c]	Early + Chronic Dose (rem) [a]	LCFs [c]
D.3.2.1 Beyond-Design-Basis TRU Waste Drum Fire Outside Confinement (Waste from fuel production)	396 (450g PuO_2)	KIS-grade Pu,~ 5 kg Pu per assembly	Extremely Unlikely to Beyond Extremely Unlikely	7.7×10^{-1}	5×10^{-4}	2.1×10^{-2}	1×10^{-5}	4.1×10^{-4}	2×10^{-7}	1.1	7×10^{-4}	1.5	9×10^{-4}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	3,346	Pu-239 Equivalent	Unlikely	8.0×10^{-2}	5×10^{-5}	2.2×10^{-3}	1×10^{-6}	4.2×10^{-5}	3×10^{-8}	1.2×10^{-1}	7×10^{-5}	1.6×10^{-1}	1×10^{-4}

Am = americium; g = gram; kg = kilogram; KIS = K Area Interim Surveillance; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; ; Pu = plutonium; PuO_2 = plutonium oxide; rem = roentgen equivalent man; SRS = Savannah River Site; TRU = transuranic; VTR = Versatile Test Reactor.

[a] Calculated using the source terms in Table D–3. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

D.4.4 Comparison of the Radiological Impacts from Accidents at Versatile Test Reactor-Related Facilities at Idaho National Laboratory and Oak Ridge National Laboratory

In order to facilitate comparisons of accident impacts for comparable activities at INL and ORNL, the key impacts from Sections D.4.1 and D.4.2 have been extracted to generate tables to enable direct comparisons. Impacts are expressed in terms of the potential LCFs if an accident occurred. Because the noninvolved worker at both sites is 330 feet away from the release point, the differences in modeling impacts should be within the overall uncertainty in meteorology and dispersion. Hence, any differences are not real discriminators and are not reported in these comparison tables. For the MEI, the reported LCF can be interpreted as the probability that the affected individual would ultimately die of an LCF. Unless the exposure is quite high (~1,000 rem), the expected LCF would be 0.

Table D–27 summarizes the impacts related to various accident scenarios for VTR operation, spent fuel handling, post-irradiation examination, TRU waste handling, and spent fuel storage at INL and ORNL. For the VTR at INL or ORNL, the reactor would be designed and operations would be conducted so that for credible accidents, no fuel would melt and accidents would result in negligible releases. A fire involving VTR spent fuel assemblies in the VTR experiment hall is postulated as the bounding operational accident at the VTR. This type of accident is modeled as extremely unlikely to beyond extremely unlikely but may not be credible.

For the VTR support activities, including the post-irradiation examination and spent fuel handling, conditioning, and storage, the bounding operational accidents are a criticality, which results in unfiltered releases, and a filtered release from fire and spills involving molten spent fuel. In a severe seismic event, building structures, process enclosures, and process equipment could be damaged enough that unfiltered releases could occur. This event has lower impacts than a fire involving VTR spent fuel assemblies in the VTR experiment hall. The impacts from the hypothetical beyond-design-basis earthquake at the VTR dominate releases that might occur from all VTR support activities. Results differ between the INL VTR Alternative and the ORNL VTR Alternative because each has unique meteorology, receptor distance, and population distribution.

D.4.5 Comparison of Radiological Impacts from Accidents at Versatile Test Reactor-Related Fuel Production Activities at Idaho National Laboratory and K-Reactor Complex at Savannah River Site

Table D–28 summarizes the impacts related to various accident scenarios for reactor fuel production at INL and SRS. As with Table D–27, this table presents side-by-side results of similar accidents at the two sites. For the VTR fuel production activities at either INL or SRS, the bounding operational accident is a high-pressure explosion of a 3013 container of plutonium oxide Although this event is considered extremely unlikely, it does have the potential for a large release. Releases from other accidents involving liquid, oxide, or molten forms of plutonium would be filtered before release to the environment. In a severe, beyond-design-basis earthquake, severe structural and process equipment damage was postulated. This damage could result in spillage and unfiltered release of liquid, oxide, and molten forms of plutonium. In a severe seismic event, the building and glovebox structures could be damaged enough that an unfiltered release could occur. Even so, most of the material not specifically being processed at the time of the earthquake would be in metal form and would not result in a substantial release even with loss of glovebox and building integrity. Results differ between the INL options and the SRS options because of differences in stack height, meteorology, receptor distance, and population distribution.

Table D–27. Summary of Potential Annual Radiological Impacts for VTR Operational Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

Accident Scenario	Frequency (per year)	MEI (probability of an LCF)		Population (Near-Term) (LCFs) [a,b]		Population (Near+Long-Term) (LCFs) [a,b]	
		INL	ORNL	INL	ORNL	INL	ORNL
VTR Accident Impacts							
D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire in Experiment Hall	Extremely Unlikely	4×10^{-8}	3×10^{-6}	7×10^{-6}	3×10^{-4}	2×10^{-5}	4×10^{-4}
D.3.3.5.2.2 Fire at Eutectic Temperature Involving VTR Fuel Assemblies	Extremely Unlikely to Beyond Extremely Unlikely	1×10^{-4}	9×10^{-3}	2×10^{-2}	9×10^{-1}	3	9
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	Beyond Extremely Unlikely	8×10^{-10}	4×10^{-8}	1×10^{-7}	4×10^{-6}	3×10^{-5}	8×10^{-5}
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	Beyond Extremely Unlikely	4×10^{-5}	3×10^{-3}	8×10^{-3}	3×10^{-1}	2×10^{-2}	4×10^{-1}
Spent Fuel Handling and Treatment							
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	Extremely Unlikely	2×10^{-6}	9×10^{-5}	8×10^{-5}	3×10^{-3}	5×10^{-4}	4×10^{-3}
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	Extremely Unlikely	5×10^{-7}	4×10^{-5}	9×10^{-5}	3×10^{-3}	2×10^{-4}	5×10^{-3}
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	Extremely Unlikely	7×10^{-6}	5×10^{-4}	1×10^{-3}	5×10^{-2}	5×10^{-3}	7×10^{-2}
Transuranic Waste Accident Impacts							
D.3.2.1 Fire Outside Confinement (Waste from fuel INL or spent fuel treatment)"	Extremely Unlikely	4×10^{-11}	3×10^{-9}	8×10^{-9}	3×10^{-7}	2×10^{-8}	4×10^{-7}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	Unlikely	7×10^{-8}	4×10^{-6}	1×10^{-5}	4×10^{-4}	2×10^{-5}	6×10^{-4}
Post-Irradiation Examination Accident Impacts							
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	Extremely Unlikely	1×10^{-8}	8×10^{-7}	2×10^{-6}	8×10^{-5}	4×10^{-6}	1×10^{-4}
Spent Fuel Storage Accident Impacts							
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	Beyond Extremely Unlikely	2×10^{-6}	2×10^{-4}	4×10^{-4}	2×10^{-2}	8×10^{-4}	2×10^{-2}
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	Extremely Unlikely	2×10^{-6}	9×10^{-5}	8×10^{-5}	3×10^{-3}	5×10^{-4}	4×10^{-3}
D.3.6.3 Drop of Fuel-Loaded Cask	Extremely Unlikely	1×10^{-6}	8×10^{-5}	2×10^{-4}	8×10^{-3}	4×10^{-4}	1×10^{-2}

Am = americium; INL= Idaho National Laboratory; LCF = latent cancer fatality; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; VTR = Versatile Test Reactor.

[a] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less. For the MEI, the reported LCF can be interpreted in the probability that the affected individual would ultimately die of a LCF. Unless the exposure is quite high (~1,000 rem), the expected LCF would be 0.

[b] Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage without mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passes. These doses include ingestion of contaminated foods,

Accident Scenario	Frequency (per year)	MEI (probability of an LCF)		Population (Near-Term) (LCFs) [a,b]		Population (Near+Long-Term) (LCFs) [a,b]	
		INL	ORNL	INL	ORNL	INL	ORNL

water, etc., direct exposure to deposited material, and resuspension and inhalation of deposited materials. For purposes of the EIS, no interdiction or mitigation is assumed, but such measures would likely occur. The long-term impacts reported include both the near-term and long-term impacts without mitigation.

Table D–28. Summary of Potential Radiological Impacts for Fuel Fabrication Accidents at Idaho National Laboratory and Savannah River Site

Accident Scenario	Frequency (per year)	MEI (probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	SRS	INL	SRS	INL	SRS
Fuel Fabrication – Accident Impacts							
D.3.1.1 Criticality while alloying the three components of the metal fuel, uranium, plutonium, and zirconium	Extremely Unlikely	2×10^{-6}	9×10^{-7}	8×10^{-5}	5×10^{-4}	5×10^{-4}	1×10^{-3}
D.3.1.2 Fire Impingement on Fuel Material (Intact confinement)	Extremely Unlikely	4×10^{-9}	1×10^{-9}	7×10^{-7}	4×10^{-6}	9×10^{-7}	6×10^{-6}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	Extremely Unlikely to Beyond Extremely Unlikely	4×10^{-8}	1×10^{-8}	6×10^{-6}	4×10^{-5}	9×10^{-6}	6×10^{-5}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	Extremely Unlikely to Beyond Extremely Unlikely	7×10^{-6}	3×10^{-6}	1×10^{-3}	8×10^{-3}	2×10^{-3}	1×10^{-2}
D.3.1.5 Plutonium Oxide-to-Metal Conversion - Explosion of 3013 Container of PuO_2	Extremely Unlikely	6×10^{-5}	2×10^{-5}	1×10^{-2}	7×10^{-2}	2×10^{-2}	1×10^{-1}
D.3.1.6 Beyond-Design-Basis Fire Involving TRU Waste Drum	Extremely Unlikely to Beyond Extremely Unlikely	1×10^{-6}	5×10^{-7}	2×10^{-4}	1×10^{-3}	3×10^{-4}	2×10^{-3}
Fuel Fabrication – Feedstock Preparation – Accident Impacts							
D.3.1.7 Aqueous/Electrorefining Fuel Preparation	Extremely Unlikely	1×10^{-7}	5×10^{-8}	2×10^{-5}	1×10^{-4}	3×10^{-5}	2×10^{-4}
Fuel Fabrication + Feedstock Preparation: Combined Beyond-Design-Basis Earthquake Accident Impacts							
D.3.1.8 Aircraft Crash into VTR Fuel Fabrication Facility	Beyond Extremely Unlikely	7×10^{-4}	3×10^{-4}	1×10^{-1}	7×10^{-1}	2×10^{-1}	1
D.3.1.9 Beyond-Design-Basis Earthquake Involving All VTR Fuel Fabrication and Preparation MAR	Beyond Extremely Unlikely	7×10^{-4}	3×10^{-4}	1×10^{-1}	7×10^{-1}	2×10^{-1}	1

INL= Idaho National Laboratory; LCF = latent cancer fatality; MAR = material at risk; MEI = maximally exposed individual; PuO_2 = plutonium oxide; SRS = Savannah River Site; TRU = transuranic; VTR = Versatile Test Reactor.

[a] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

[b] Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage without mitigation measures, such as sheltering in place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passes. These doses include ingestion of contaminated foods, water, etc., direct exposure to deposited material, and resuspension and inhalation of deposited materials. For

Accident Scenario	Frequency (per year)	MEI (probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	SRS	INL	SRS	INL	SRS

purposes of the EIS, no interdiction or mitigation is assumed, but such measures would likely occur. The long-term impacts reported include both the near-term and long-term impacts without mitigation.

D.4.6 Comparison of the Annual Radiological Risks from Accidents at Versatile Test Reactor-Related Facilities at Idaho National Laboratory and Oak Ridge National Laboratory

Table D–29 summarizes the annual radiological accident risks for specific accident scenarios for VTR operation, spent fuel handling, post-irradiation examination, TRU waste handling, and spent fuel storage at INL and ORNL. Risks are expressed in terms of the product of the estimated annual probability of the accident times the projected impacts, in terms of LCFs, if these accidents were to occur. It is recognized that there are large uncertainties involved in both the probabilities of specific accidents as well as the potential consequences. Nevertheless, expressing the potential impacts in terms of an annual risk to the MEI or general population does provide a useful tool for comparing disparate events, such as severe reactor accidents and simple glovebox accidents. It also allows comparison of events that differ significantly in probability and consequence. For example, the risk to the MEI from both unlikely events with low consequences can be compared to the risks from extremely unlikely events with higher consequences.

For purposes of calculating risk in this EIS, events categorized as "unlikely" are assigned a probability of 10^{-2}. Events categorized as "extremely unlikely" are assigned a probability of 10^{-4}. Events categorized as "extremely unlikely to beyond extremely unlikely" are assigned a probability of 10^{-5}. Events categorized as "beyond extremely unlikely" are assigned a probability of 10^{-6}.

Use of LCF risks allows comparison of individual accidents. For the VTR at INL or ORNL, operational reactor-specific accidents are all managed so that no fuel would melt and releases would be negligible. A fire involving VTR spent fuel assemblies in the VTR experiment hall (D.3.3.5.2.2 Fire Involving VTR Fuel Assemblies) is postulated as the highest risk, operational accident at the VTR. This accident is certainly in the extremely unlikely to beyond extremely unlikely category and may not be credible once the VTR design is completed. Other accidents that contribute substantially to the overall risk are the sodium fire involving spent fuel (D.3.4.3) and the waste drum fire with americium-241 (D.3.2.2). Table D–29 indicates that the overall, near-term LCF risk to the population within 50 miles is 5×10^{-7} at INL and 2×10^{-5} at ORNL.

Some of the accidents release radionuclides that would result in longer-term exposures due to ingestion if no interdiction or mitigation occurred. At both sites, the highest-risk accident for long-term exposure is the fire involving spent fuel assemblies in the experiment hall, which presents the dominant long-term risk to the public. That fire is projected to release substantial quantities of cesium and strontium isotopes that are the dominant contributors to long-term dose.

Also presented at the end of Table D–29 is the annual LCF risk to the MEI and the population within 50 miles. For the MEI, the annual probability of a LCF from VTR reactor and support operations at INL and ORNL is estimated to be 3×10^{-9} and 2×10^{-7}, respectively. The risk is higher at ORNL because the MEI is calculated for a point on Melton Hill Lake within the ORR where the public has access, though by boat. For the population within 50 miles, the annual probability of a LCF from VTR reactor and support operations at INL and ORNL due to near and long-term exposures is estimated to be 3×10^{-5} and 1×10^{-4}, respectively. Here, the primary difference is due to the higher population density near ORNL. The long-term risks to the public from accidents at either site do not take into account mitigation actions DOE would take if an accident occurred.

Table D–29. Summary of Annual Risks for VTR Operational Accidents at Idaho National Laboratory and Oak Ridge National Laboratory

Accident Scenario	*Frequency (per year)*	*MEI (Probability of an LCF)*		*Population (Near-Term) (LCFs)* [a, b]		*Population (Near+Long-Term) (LCFs)* [a, b]	
		INL	*ORNL*	*INL*	*ORNL*	*INL*	*ORNL*
VTR Accident LCF Risks [c]							
D.3.3.5.2.1 Test Assembly Failure following Seismically-Induced Fire in Experiment Hall	Extremely Unlikely	4×10^{-12}	3×10^{-10}	7×10^{-10}	3×10^{-8}	2×10^{-9}	4×10^{-8}
D.3.3.5.2.2 Fire Involving VTR Fuel Assemblies	Extremely Unlikely to Beyond Extremely Unlikely	1×10^{-9}	9×10^{-8}	2×10^{-7}	9×10^{-6}	3×10^{-5}	9×10^{-5}
D.3.3.5.2.3 VTR Fuel Assembly Drop in Experiment Hall	Beyond Extremely Unlikely	8×10^{-16}	4×10^{-14}	1×10^{-13}	4×10^{-12}	3×10^{-11}	8×10^{-11}
D.3.3.5.2.4 VTR Seismic Event Resulting in Collapse of the Experiment Hall	Beyond Extremely Unlikely	4×10^{-11}	3×10^{-9}	8×10^{-9}	3×10^{-7}	2×10^{-8}	4×10^{-7}
Spent Fuel Handling and Treatment Accident LCF Risks [c]							
D.3.4.1 Criticality Involving Melted Spent Fuel (Failed Confinement)	Extremely Unlikely	2×10^{-10}	9×10^{-9}	8×10^{-9}	3×10^{-7}	5×10^{-8}	4×10^{-7}
D.3.4.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	Extremely Unlikely	5×10^{-11}	4×10^{-9}	9×10^{-9}	3×10^{-7}	2×10^{-8}	5×10^{-7}
D.3.4.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	Extremely Unlikely	7×10^{-10}	5×10^{-8}	1×10^{-7}	5×10^{-6}	5×10^{-7}	7×10^{-6}
Transuranic Waste Accident LCF Risks [c]							
D.3.2.1 Fire Outside Confinement (Waste from fuel fabrication or spent fuel treatment)"	Extremely Unlikely	4×10^{-15}	3×10^{-13}	8×10^{-13}	3×10^{-11}	2×10^{-12}	4×10^{-11}
D.3.2.2 Fire Outside Involving a Waste Drum with 23 grams of Am-241	Unlikely	7×10^{-10}	4×10^{-8}	1×10^{-7}	4×10^{-6}	2×10^{-7}	6×10^{-6}
Post-Irradiation Examination Accident LCF Risks [c]							
D.3.5.1 Fire Involving Test Assembly (Seismically-induced confinement failure)	Extremely Unlikely	1×10^{-12}	8×10^{-11}	2×10^{-10}	8×10^{-9}	4×10^{-10}	1×10^{-8}
Spent Fuel Storage Accident LCF Risks [c]							
D.3.6.1 Seismic Event Causes Failure of Spent Fuel Storage Cask	Beyond Extremely Unlikely	2×10^{-12}	2×10^{-10}	4×10^{-10}	2×10^{-8}	8×10^{-10}	2×10^{-8}
D.3.6.2 Seismic Event Causes Criticality in Fuel from Spent Fuel Storage Cask	Extremely Unlikely	2×10^{-10}	9×10^{-9}	8×10^{-9}	3×10^{-7}	5×10^{-8}	4×10^{-7}
D.3.6.3 Drop of Fuel-Loaded Cask	Extremely Unlikely	1×10^{-10}	8×10^{-9}	2×10^{-8}	8×10^{-7}	4×10^{-8}	1×10^{-6}
Total Annual LCF Risk from Reactor and Support Facility Operations without Hypothetical Beyond-Design-Basis Reactor Accident [d]		$\mathbf{3\times10^{-9}}$	$\mathbf{2\times10^{-7}}$	$\mathbf{5\times10^{-7}}$	$\mathbf{2\times10^{-5}}$	$\mathbf{3\times10^{-5}}$	$\mathbf{1\times10^{-4}}$

Accident Scenario	Frequency (per year)	MEI (Probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	ORNL	INL	ORNL	INL	ORNL

Am = americium; INL= Idaho National Laboratory; LCF = latent cancer fatality; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; VTR = Versatile Test Reactor.

[a] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[b] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[c] Risks are the product of the probability and impacts. For purposes of this EIS, events categorized as "unlikely" are assigned a probability of 10^{-2}. Events categorized as "extremely unlikely" are assigned a probability of 10^{-4} Events categorized as "extremely unlikely to beyond extremely unlikely" are assigned a probability of 10^{-5}. Events categorized as "beyond extremely unlikely" are assigned a probability of 10^{-6}.

[d] The total annual risk from reactor and support facility operations is the sum of all accidents evaluated. For the MEI, it represents a reasonable estimate of the annual risk that the MEI might receive sufficient dose to result in a LCF. For the population within 50 miles, it represents a reasonable estimate of the annual number of LCFs that might occur within that population due to accidents.

D.4.7 Comparison of the Annual Radiological Risks from Accidents at Versatile Test Reactor-Related Fuel Production Activities at Idaho National Laboratory and K-Reactor Complex at Savannah River Site

Table D–30 summarizes the annual radiological accident risks for specific accident scenarios for reactor fuel production at INL or SRS. For the VTR fuel production activities at either INL or SRS, the highest risk bounding operational accident is a high-pressure explosion of 3013 container of plutonium oxide. Although this event is considered extremely unlikely, it does have the potential for a large radiological release.

Releases from other accidents involving liquid, oxide, or molten forms of plutonium would be filtered before release to the environment. In a beyond-design-basis earthquake, severe damage to the structures and process equipment is postulated. This could result in spillage and unfiltered release of liquid, oxide, and molten forms of plutonium. In a severe seismic event, the building and glovebox structures could be damaged enough that an unfiltered release could occur. Even so, most of the material not specifically being processed at the time of the earthquake would be in metal form and would not result in a substantial release even with loss of glovebox and building integrity. Results differ between the INL option and the SRS option because of stack height, meteorology, receptor distance, and population distribution.

Table D–30. Summary of Annual Risks for Reactor Fuel Production Accidents at Idaho National Laboratory and Savannah River Site

Accident Scenario	Frequency (per year)	MEI (Probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	SRS	INL	SRS	INL	SRS
Fuel Fabrication – Accident LCF Risks [c]							
D.3.1.1 Criticality while alloying the three components of the metal fuel, uranium, plutonium, and zirconium)	Extremely Unlikely	2×10^{-10}	9×10^{-11}	8×10^{-9}	5×10^{-8}	5×10^{-8}	1×10^{-7}

Accident Scenario	Frequency (per year)	MEI (Probability of an LCF)		Population (Near-Term) (LCFs) [a, b]		Population (Near+Long-Term) (LCFs) [a, b]	
		INL	SRS	INL	SRS	INL	SRS
D.3.1.2 Fire Impingement on Fuel Material (Intact confinement)	Extremely Unlikely	4×10^{-13}	1×10^{-13}	7×10^{-11}	4×10^{-10}	9×10^{-11}	6×10^{-10}
D.3.1.3 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	Extremely Unlikely to Beyond Extremely Unlikely	4×10^{-13}	1×10^{-13}	6×10^{-11}	4×10^{-10}	9×10^{-11}	6×10^{-10}
D.3.1.4 Spill and Oxidation of Molten Plutonium-Uranium Mixture while Heating or Casting with Seismically-Induced Confinement Failure	Extremely Unlikely to Beyond Extremely Unlikely	7×10^{-11}	3×10^{-11}	1×10^{-8}	8×10^{-8}	2×10^{-8}	1×10^{-7}
D.3.1.5 Plutonium Oxide-to-Metal Conversion - Explosion of 3013 Container of PuO_2	Extremely Unlikely	6×10^{-9}	2×10^{-9}	1×10^{-6}	7×10^{-6}	2×10^{-6}	1×10^{-5}
D.3.1.6 Beyond-Design-Basis Fire Involving TRU Waste Drum	Extremely Unlikely to Beyond Extremely Unlikely	1×10^{-11}	5×10^{-12}	2×10^{-9}	1×10^{-8}	3×10^{-9}	2×10^{-8}
Feedstock Preparation– Accident LCF Risks [c]							
D.3.1.7 Aqueous/Electro-refining Fuel Preparation	Extremely Unlikely	1×10^{-11}	5×10^{-12}	2×10^{-9}	1×10^{-8}	3×10^{-9}	2×10^{-8}
Fuel Fabrication + Feedstock Preparation: Combined Beyond-Design-Basis Earthquake Accident Risks [c]							
D.3.1.8 Aircraft Crash into VTR Fuel Fabrication Facility	Extremely Unlikely to Beyond Extremely Unlikely	7×10^{-10}	3×10^{-10}	1×10^{-7}	7×10^{-7}	2×10^{-6}	1×10^{-6}
D.3.1.9 Beyond-Design-Basis Earthquake Involving All VTR Fuel Fabrication and Preparation MAR	Beyond Extremely Unlikely	7×10^{-10}	3×10^{-10}	1×10^{-7}	7×10^{-7}	2×10^{-7}	1×10^{-6}
Total Annual LCF Risk from Fuel Fabrication Operations [d]		**8×10^{-9}**	**3×10^{-9}**	**1×10^{-6}**	**8×10^{-6}**	**2×10^{-6}**	**1×10^{-5}**

INL= Idaho National Laboratory; LCF = latent cancer fatality; MEI = maximally exposed individual; PuO_2 = plutonium oxide; SRS = Savannah River Site; VTR = Versatile Test Reactor.

[a] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[b] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[c] Risks are the product of the probability and impacts. For purposes of this EIS, events categorized as “unlikely” are assigned a probability of 10^{-2} Events categorized as “extremely unlikely” are assigned a probability of 10^{-4}. Events categorized as “extremely unlikely to beyond extremely unlikely” are assigned a probability of 10^{-5}. Events categorized as “beyond extremely unlikely” are assigned a probability of 10^{-6}.

[d] The total annual risk from reactor and support facility operations is the sum of all accidents evaluated. For the MEI, it represents a reasonable estimate of the annual risk that the MEI might receive sufficient dose to result in a LCF. For the population within 50 miles, it represents a reasonable estimate of the annual number of LCFs that might occur within that population due to accidents.

D.4.8 Comparison of the Annual Radiological Risks from Accidents at Versatile Test Reactor-Related Activities at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site to Other Radiological Risks

Table D–31 summarizes the total annual risk from reactor and support facility operations at INL, ORNL, and SRS, taking into account the probability of occurrence of each accident. For each option, the values are simply the sum of the LCF risk of all accidents evaluated. For the MEI, the summary represents a reasonable estimate of the annual risk that the MEI might receive sufficient dose to result in a LCF. For the population within 50 miles, the summary represents a reasonable estimate of the annual number of LCFs that might occur within that population due to accidents.

Table D–31. Summary of the Total Annual Risks for VTR-Related Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

	MEI (annual LCF risk) [a,b]	*Population (Near+Long-Term) (annual person-rem risk)* [a,b,c,d]	*Population (Near+Long-Term) (annual LCF risk)* [a,b,c,d]
Total Risk: Reactor and Support Operations at INL	3×10^{-9}	4×10^{-2}	3×10^{-5}
Total Risk: Reactor and Support Operations at ORNL	2×10^{-7}	2×10^{-1}	1×10^{-4}
Total Risk: Reactor Fuel Production Operations at INL	8×10^{-9}	3×10^{-3}	2×10^{-6}
Total Risk: Reactor Fuel Production Operations at SRS	3×10^{-9}	2×10^{-2}	1×10^{-5}
VTR, Support, and Reactor Fuel Production Operations at INL	1×10^{-9}	5×10^{-2}	3×10^{-5}

INL= Idaho National Laboratory; LCF = latest cancer fatality; MEI = maximally exposed individual; ORNL = Oak Ridge National Laboratory; SRS = Savannah River Site; VTR = Versatile Test Reactor.

[a] Annual LCF risks presented in this table are the product of the probability of the accident occurring and its impacts. For purposes of this EIS, the probability of an event categorized as "unlikely" is 10^{-2}; and "extremely unlikely" is 10^{-4}. "Extremely unlikely to beyond extremely unlikely" is 10^{-5}; and "beyond extremely unlikely" is 10^{-6}.

[b] The total annual risk from reactor and support operations and the total annual risk for operations from reactor fuel production is the sum of all accidents evaluated. For the MEI, it represents a reasonable estimate of the annual risk that the MEI might receive sufficient dose from an accident to result in a LCF. For the population within 50 miles, it represents a reasonable estimate of the annual number of LCFs that might occur within that population due to accidents.

[c] Numbers of LCFs in the population are typically whole numbers. The statistically calculated values are provided when the reported result is 1 or less.

[d] Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage without mitigation measures, such as sheltering in place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passes. These doses include ingestion of contaminated foods, water, etc., direct exposure to deposited materials, and resuspension and inhalation of deposited materials. For purposes of the EIS, no interdiction or mitigation is assumed, but such measures would likely occur. The long-term risk reported includes both the near-term and long-term impacts without mitigation.

It is important to note that no near-term mitigation, such as emergency notifications to shelter in place or evacuate, or long-term restrictions on access to contaminated areas and interdiction of food supplies are assumed in evaluating the consequences.

Potential MEI Risks from VTR Activities at INL, ORNL, and SRS

The potential annual risks to the hypothetical MEI are low at all sites. For VTR and support operations at INL and ORNL, the MEI annual risks are 3×10^{-9} and 2×10^{-7} LCFs per year, respectively. The principal difference is the MEI for ORNL is assumed to be a boater on Melton Hill Lake within the overall ORO property boundary. The nearest residence is much farther away.

For VTR fuel production operations at INL and SRS, the potential annual MEI risks are 8×10^{-9} and 3×10^{-9} LCFs per year, respectively. In this case, the MEI for SRS is further away than at INL.

Potential Population Risks from VTR Activities at INL, ORNL, and SRS

The potential annual risks to the populations within 50 miles are low at all sites. For VTR and support operations at INL and ORNL, the annual population risks without the hypothetical beyond-design-basis reactor accident are 3×10^{-5} and 1×10^{-4} LCFs per year, respectively. The principal difference in the population risk is a higher population density near ORNL than near INL.

For VTR fuel production operations at INL and SRS, the potential annual population risks are 2×10^{-6} and 1×10^{-5} LCFs per year, respectively. The principal difference in the population risk is a higher population density near SRS than near INL.

If both VTR and reactor fuel production operations were at INL, the near-term plus long-term annual population risk would be 3×10^{-5} LCFs.

D.4.9 Versatile Test Reactor Beyond-Design-Basis Reactor Accidents

By design, the VTR is able to withstand a wide range of accidents. Most events that could affect safe operation of the VTR are mitigated by the VTR design. This section addresses potential beyond-design-basis accidents that have the potential for high consequences even though the probability is very low (1×10^{-6} to 1×10^{-8} per year). These accidents represent events in which the consequences can be in the hundreds or thousands of rem to the public while probabilities are less than one in a million per year. Consideration of these very low-probability but potentially high-consequence accidents provides valuable insight for the public and decision-makers in understanding the overall risks of operation, siting decisions, and the need for emergency preparedness.

Deterministic safety analyses for the VTR, based on the conceptual design (INL 2019, 2020c) and the preliminary VTR PRA (GEH 2019), are ongoing and are expected to continue to evolve with the VTR design. As part of the on-going safety analysis process (INL 2019), a more elaborate VTR PRA with a larger spectrum of events than currently quantified in terms of both probability and consequences is needed. These events will include the full spectrum of design-basis events and beyond-design-basis events, including accidents initiated by human failures, natural phenomena, and external events. Preliminary results from the ongoing PRA for the VTR conceptual design do not quantify the probabilities and consequences for beyond-design-basis accident sequences. Given that the time is not ripe to prepare a full VTR PRA on a yet-to-be finalized VTR design, hypothetical, beyond-design-basis accidents are postulated for the VTR in order to fulfill the requirements of NEPA.

D.4.9.1 Potential Hypothetical Beyond-Design-Basis Accident Scenarios

The purpose of considering these scenarios is to ensure that events that have potentially large offsite consequences are sufficiently understood and properly categorized. This process typically leads to design changes to prevent or mitigate any of these events. For the VTR EIS, a range of potential operational, external, and natural-phenomena hazards have been considered that might lead to severe accident conditions, including

- Human failures
- Plant design or construction errors
- Aircraft crash
- Seismic hazards
- Extreme straight-line wind, tornado, and hurricane hazards
- Flood, seiche, tsunami, and other flood-related hazards
- Extreme precipitation hazards
- Volcanic eruption hazards.

In typical power reactors, radiological risks to the public are dominated by loss-of-cooling, reactor-core-damage, and failure-of-confinement events with estimated frequencies in the one in ten thousand to one in a hundred thousand per year range. The safety analyses for the VTR conceptual design indicate that the pool-type, sodium-cooled design of the VTR and other design features provide substantial protection to the VTR from the severe accidents that would lead to core-melt and containment-failure accidents in commercial LWR power plants.

The hypothetical, beyond-design-basis reactor accident with loss of cooling is considered in this VTR EIS to provide a reasonable but bounding estimate of the potential impacts from very low-probability, high-consequence accidents. The probability of this event is estimated to be ~10^{-7} or less. As the VTR design evolves past the conceptual design phase, additional event initiators and subsequent accident sequences may be developed but this postulated, hypothetical event is expected to provide a reasonable estimate of the impacts of such an event. This event would bound a possible intentionally destructive act involving an aircraft.

D.4.9.2 Material at Risk for Severe Accidents in the Versatile Test Reactor

The MAR is based on fuel assemblies with 6 percent burnup. Current projections indicate that fuel burnup will not exceed 6 percent. The reactor vessel is designed to contain 110 fuel assemblies of which 66 fuel assemblies are in the reactor core and 44 used fuel assemblies are in out-of- core storage. The 66 in-core fuel assemblies are assumed to have no decay while the 44 out-of-core fuel assemblies are assumed to have decayed for 220 days.

By necessity, the assumptions used in modeling the impacts of this hypothetical event are quite conservative. Chemical and physical properties that would be expected to mitigate and limit a release are not considered. All of the active and passive cooling is assumed to be lost and all of fuel in the reactor is assumed to melt and be released quickly. Because the release is assumed to occur very quickly, no decay of even very short-lived isotopes is assumed. These very short-lived isotopes, with half lives of seconds to minutes or even hours contribute substantially to close-in doses. In reality, the major portion of a release, even with total loss of cooling, may not occur for hours or days after reactor shut down. In contrast, severe LWR accidents such as loss-of-cooling accidents typically assume that the major releases to the environment occur many hours or days after the initial event shutting off the reactor, allowing time for the short-lived isotopes to decay and for emergency actions, such as evacuation, to be implemented.

D.4.9.3 Release Fractions for Severe Accidents in the Versatile Test Reactor

For most VTR accidents in which fuel fails and fission products are released, nongaseous fission products would be retained in the sodium (Bucknor et al. 2017; Brunett et al. 2016; Grabaskas et al. 2016a). However, if bulk boiling of the sodium were to occur, retention of radionuclides in the sodium may be limited. The actual airborne release fractions and respirable fractions by isotope group for specific VTR events are unknown. Various ARFs and RFs have been assumed in the past. More recently, there have been efforts to better understand the potential phenomena during severe accidents and develop mechanistic source terms for severe accidents at sodium-cooled fast reactors. An ongoing program at Argonne National Laboratory is researching mechanistic source terms for fast reactors (Brunett et al. 2016; Clark et al. 2017; Grabaskas et al. 2015, 2016a, 2016b, 2017; Middleton et al. 2011) and sophisticated modeling codes have been developed.

The fraction of radionuclides in VTR fuel that might ultimately be released to the environment is speculative, but, if all active and passive cooling failed, could be substantial. Experimental data for releases of various isotopes from metal fuel in a pool of sodium has been reviewed and used to postulate mechanistic release fractions from a metal fuel, pool-type sodium fast reactor for various temperature ranges that might occur under operating and accident conditions (Grabaskas et al. 2016a). Potential release fractions and uncertainty estimates for key isotopic groups are postulated for normal operation temperatures (~500 °C), eutectic formation temperatures (~700 °C), fuel melting temperatures (~1100 °C), and very high temperatures (≥~1300 °C). **Table D–32** summarizes the postulated release fractions and uncertainty estimates.

Other releases fractions were considered, including

- PRISM: During the early licensing efforts for PRISM site suitability, the initial PRISM source term assumed complete core damage (including the spent fuel in the storage area) because of the lack of data on the behavior of metal fuel in severe liquid-metal-reactor accidents. The analysis did not include retention of radionuclides in the sodium pool. The near instantaneous release to confinement was modeled. The source term used for the site suitability assessment was 100 percent of the noble gas inventory, 0.1 percent of halogens, 0.1 percent of the volatiles, and 0.01 percent of the transuranic nuclides (plutonium) (PSID 1987).
- GNEP Draft EIS-0396 (DOE 2008), Table D.1.4-1—*Release Parameters for Reactor Beyond Design Basis Earthquakes and Aircraft Crashes* used noble gases (xenon, krypton) 1.0; halogens (iodine, bromine) 0.4; alkali metals (cesium, rubidium) 0.3; tellurium metals (tellurium, antimony, selenium) 0.05; barium, strontium 0.02; noble metals (ruthenium, rhodium, palladium, molybdenum, technetium, cobalt) 2.5×10^{-3}; lanthanides (lanthanum, zirconium, neodymium, niobium, promethium, samarium, yttrium, curium, americium) 2×10^{-4} based on Table 1 of NRC Regulatory Guide 1.183 (NRC 2000).
- Clinch River Breeder Reactor (CRBR): For purposes of determining the suitability of the proposed site for construction and operation of the CRBR in accordance with 10 CFR Part 100, a hypothetical core disruptive accident was evaluated. The radiological source term associated with this hypothetical release was specific in terms of percentages of fission products and fuel material released from the core to the reactor confinement building. The source term used for site suitability assessment was 100 percent of noble gas inventory. The inventory included 50 percent halogen (25 percent airborne), 1 percent solid fission product, and 1 percent plutonium inventory (PMC 1976). The PSAR further states that "The source term specified by NRC not only envelopes all design basis accidents considered in Chapter 15, but further envelopes a wide range of conservatively hypothesized core-related events. Evidence, both analytical and experimental, supports the applicant's position that compliance with the requirements of 10 CFR 100 could be demonstrated with a less stringent source term" (PMC 1976).

For purposes of this VTR EIS, the release fractions in Table D–32 for the fuel melting region of ~1,100 degrees Celsius are assumed for the fuel and fission products. These release fractions are higher than presented above for the 1987 PRISM analyses (PSID 1987) and are more experiment-based than those assumed for the CRBR analyses (PMC 1976). Except for the lanthanides, these release fractions for the most dose-significant isotope groups are similar to what might be expected for core-melt with containment failure accidents with LWRs. At both INL and ORNL, the lanthanides are a major contributor to the population doses. The lanthanide release fractions are about 30 percent for a sodium-cooled reactor with fuel melting accident and about 1 to 1.5 percent (Grabaskas et al. 2015) for a loss-of-coolant severe accident in an LWR. As a result, without allowance for decay, the lanthanides contribute about 74 and 63 percent of the near- and long-term population dose at ORNL, and about 77 and 22 percent of the near- and long-term population dose at INL. As a sensitivity analysis, source terms and impacts are also evaluated assuming the release fractions for VTR fuel are in the very-high-temperature region (≥~1,300 degrees Celsius). The resulting source terms would have the potential for impacts several times higher than projected for the fuel-melting region (~1,100 degrees Celsius).

Table D–32. Release Fractions for Isotope Groups for Metal-Fuel Sodium Fast Reactors

Isotope Group [a]	Release Fraction and Uncertainty Estimate							
	Normal Operation		Eutectic Formation		Fuel Melting		Very High Temperatures	
	Temperature ~500°C	Uncertainty	Temperature ~700°C	Uncertainty	Temperature ~1100°C	Uncertainty	Temperature ≥ 1300°C	Uncertainty
Noble Gases Group: (xenon, krypton)	≤ 0.85	Low	≤ 1	Medium	~ 1	Low	~ 1	Low
Halogens Group: (iodine, bromine)	≤ 0.15	Medium	≤ 0.20	Medium	≤ 0.3	Medium	≤ 1	Low
Alkali Metals Group: (cesium, rubidium)	≤ 0.55	Low	≤ 0.6	Medium	≤ 1	Medium	≤ 1	Low
Tellurium Group: (tellurium, antimony, selenium)	≤ 0.01	Medium	≤ 0.01	Medium	≤ 0.05	High	No Data	No Data
Barium, Strontium Group: (barium)	≤ 0.05	Medium	≤ 0.1	Medium	≤ 0.15	High	≤ 0.2	Medium
Barium, Strontium Group: (strontium)	≤ 0.001	Medium	≤ 0.05	Medium	≤ 0.2	High	≤ 0.2	High
Noble Metals Group: (ruthenium, rhodium, palladium, molybdenum, technetium, cobalt)	≤ 0.001	Low	≤ 0.01	Medium	≤ 0.05	Medium	≤ 0.05	Medium
Lanthanides Group [a]: (lanthanum, zirconium, neodymium, niobium, promethium, praseodymium, samarium, yttrium, curium, americium)	≤ 0.001	Medium	≤ 0.01	High	≤ 0.3	High	≤ 0.3	High
Lanthanides Group [a]: (europium)	≤ 0.55	Low	≤ 0.6	Medium	≤ 1	Medium	≤ 1	Low
Cerium Group [a]: (cerium)	≤ 0.01	Medium	≤ 0.05	High	≤ 0.1	High	≤ 0.15	High
Cerium Group [a]: (uranium, plutonium, neptunium)	≤ 0.001	Low	≤ 0.001	Medium	≤ 0.001	Medium	≤ 0.001	Medium

[a] Elements are grouped by general chemical characteristics typically used for development of reactor accident releases. For this analysis, the reference Grabaskas et al. 2016a further divided some of the isotope groups because of the specific melting characteristics of the elements at these temperatures.

Source: Adapted from Sections 5.1.1–4.1.8 of Grabaskas et al. 2016a.

D.4.9.4 Potential Releases from a Hypothetical, Beyond-Design-Basis Reactor Accident with Loss of Cooling

In order to fulfill the requirements of NEPA and for VTR EIS purposes, potential releases and impacts are evaluated for a hypothetical beyond-design-basis reactor accident of unknown cause in which all active (heat removal system [HRS]) and passive (RVACS) cooling systems are disrupted. For the sequence of events with total loss of heat removal capabilities, that is loss of both RVACS and HRS or the loss of heat sink, bulk sodium boiling and release of radionuclides from melted fuel in the reactor core is assumed. Because both the VTR reactor vessel and reactor room are not designed to withstand pressurization due to bulk sodium boiling, the confinement systems are assumed to fail (GEH 2019:117).

If a beyond-design-basis reactor accident with loss of cooling were to occur, the releases are assumed to occur initially at ground level but to rise as an elevated plume due to the residual heat of the reactor core and boiling sodium. For evaluation purposes, modeling was performed for a range of potential feasible heats and impacts associated with the highest population impacts are reported.

D.4.9.5 Potential Impacts and Risks from a Hypothetical Beyond-Design-Basis Reactor Accident at Idaho National Laboratory and Oak Ridge National Laboratory

Potential radiological impacts and risks to the MEI and public from the hypothetical beyond-design-basis accident are presented in **Tables D–33** and **D–34**. As expected, without allowing for pre-release decay and emergency actions, the results of the MACCS modeling indicate very high, likely fatal doses near the reactor site. An individual remaining at the assumed location of the MEI for the entire plume passage would receive a fatal dose. Individuals, including members of the public that remained near the reactor site, could receive very high and potentially fatal doses.

At INL, the projected public population within 10 miles of the VTR is less than 500, while at ORNL it is about 160,000. This makes a large difference in the projected unmitigated LCFs among the population, with MACCS projections of less than 10 for the INL site, but 3,200 for the ORNL site.

The combined effects of the conservative accident and modeling assumptions result is an over statement of the potential hypothetical VTR accident by a substantial amount. For example, evacuation at ORNL out to 10 miles could reduce the LCFs from ~8,400 (0-50 miles) by 3,200 (0-10 miles), leaving ~5,200 LCFs. The INL location is sufficiently remote that evacuation would not result in a substantial reduction in impacts.

The radiological impacts on the population are dominated by the early release of fission products from the reactor core and not the longer-term doses from ingestion of contaminated food. With the release fractions for fuel melting from Table D–32 assumed, the early doses are dominated by radioisotopes from the lanthanides (75 percent of the dose), halogens (8 percent of the dose), and barium/strontium (6 percent of the dose). About 30 percent of the lanthanides are assumed to be released from the molten fuel due to their relatively lower melting temperature as compared to the fuel itself.

Potential impacts on the offsite populations are proportionally higher for the reactor accident with loss of cooling than for other VTR accidents due to the increased MAR, the potentially higher airborne release fractions if active and passive heat removal were disrupted and sodium boiling were to occur, and the availability of short-lived isotopes within the reactor core. For the beyond-design-basis reactor accident with loss of cooling, the consequences could also be several orders of magnitude greater than those calculated for the highest-consequence design-basis earthquake for either the VTR or its support facilities.

Table D–34 presents the annual risks to the MEI and public from the unmitigated, hypothetical beyond-design-basis reactor accident. These risks are calculated by multiplying the projected impacts from Table D–33 by the estimated probability of the accident of 1×10^{-7} per year. Even without mitigation, these LCF risks are quite small.

Table D–33. Impacts for an Unmitigated, Hypothetical Beyond-Design-Basis Reactor Accident at Idaho National Laboratory and Oak Ridge National Laboratory

Location	Frequency (per year)	Impacts on Noninvolved Worker (330 feet)		Impacts on an MEI – No Mitigation		Near-Term Impacts on Population within 50 Miles – No Mitigation		Near + Long-Term Impacts on Population within 50 Miles – No Mitigation	
		Dose (rem) [a]	Probability of an LCF [b]	Dose (rem) [a]	Probability of an LCF [b]	Early Dose (person rem) [a]	LCFs [c]	Early + Chronic Dose (person rem) [a]	LCFs [c]
INL	Hypothetical – Estimated ~10^{-7} or less	5.2×10^{5} [d]	1 [d]	7.9×10^{2} [d]	1 [d]	1.3×10^{5} [d]	220 [d]	1.9×10^{5} [d]	260 [d]
ORNL		1.3×10^{6} [d]	1 [d]	4.8×10^{4} [d]	1 [d]	7.2×10^{6} [d]	8,000 [d]	7.9×10^{6} [d]	8,400 [d]

INL= Idaho National Laboratory; ORNL=Oak Ridge National Laboratory; LCF = latent cancer fatality; MEI = maximally exposed individual; rem = roentgen equivalent man; VTR = Versatile Test Reactor.

[a] Calculated using the source terms derived from Sections D.4.9.2 and D.4.9.3. Near-term impacts on the 50-mile population include the potential radiological exposures due to the initial plume passage within mitigation measures, such as sheltering-in-place or evacuation. Long-term impacts include doses due to radiological exposures over a longer period after the plume passages. These doses include ingestion of contaminated foods, water, etc., resuspension of materials deposited materials, on other pathways. For purposes of the EIS, no interdiction or mitigation is assumed but such measures would likely occur. The total reported includes both the near-term and long-term impacts without mitigation.

[b] For hypothetical individual doses equal to or greater than 20 rem, the probability of an LCF is doubled.

[c] Numbers of LCFs in the population are typically whole numbers; the statistically calculated values are provided when the reported result is 1 or less. These results are the MACCS-calculated LCFs.

[d] For NEPA purposes, a hypothetical, beyond-design-basis reactor accident with loss of cooling is included in this VTR EIS to provide a reasonable but bounding estimate of the potential impacts from very low-probability, high-consequence accidents. The probability of this event is estimated to be ~10^{-7} or less. As the VTR design evolves past the conceptual design phase, additional event initiators and subsequent accident sequences may be developed. Accident probabilities and consequences are quantified as a part of the future VTR safety analysis and probabilistic risk assessment processes.

Table D–34. Annual Risks for an Unmitigated, Hypothetical Beyond-Design-Basis Reactor Accident at Idaho National Laboratory and Oak Ridge National Laboratory

Accident Scenario	Frequency (per year)	MEI LCF Risk – No Mitigation (Probability of an LCF) [a]		Population LCF Risk – No Mitigation (Near-Term) (LCFs) [a]		Population LCF Risk – No Mitigation (Near+Long-Term) (LCFs) [a]	
		INL	ORNL	INL	ORNL	INL	ORNL
VTR Accident LCF Risks							
Hypothetical Beyond-Design-Basis Reactor Accident with Loss of Cooling [b]	Hypothetical - Estimated ~10^{-7} or less	1×10^{-7}	1×10^{-7}	2×10^{-5}	8×10^{-4}	3×10^{-5}	8×10^{-4}

INL= Idaho National Laboratory; ORNL= Oak Ridge National Laboratory; LCF=latent cancer fatality; MEI = maximally exposed individual; rem = roentgen equivalent man; VTR = Versatile Test Reactor.

[a] Calculated using the projected impacts in Table D–33.

[b] For NEPA purposes, a hypothetical, beyond-design-basis reactor accident with loss of cooling is included in this VTR EIS to provide a reasonable but bounding estimate of the potential impacts from very low-probability, high-consequence accidents. The probability of this event is estimated to be ~10^{-7} or less. As the VTR design evolves past the conceptual design phase, additional event initiators and subsequent accident sequences may be developed. Accident probabilities and consequences are to be quantified as a part of the future VTR safety analysis and probabilistic risk assessment processes.

D.4.9.6 Comparison to U.S. Nuclear Regulatory Commission-Licensed Power Reactor Risks

Although the VTR would be regulated by the DOE and not the NRC,[6] comparing the potential VTR accident risks to those of LWRs regulated by the NRC provides insight into the relative risks from the VTR. The NRC has recently released an EIS (NRC 2019a; NUREG-2226) that includes a summary of environmental risks from severe accidents for current nuclear power plants. Table 5-18 of that EIS summarizes the core damage frequencies and 50-mile population dose risk. The table indicates that, based on over 70 current plants at over 40 sites, the current mean reactor core damage frequency is 3.1×10^{-5} per year with an annual 50-mile population dose risk of 15 person-rem (or ~2×10^{-2} LCFs) per reactor. The range of 50-mile population dose risk for current operating power reactors is 0.55 to 69 person-rem per year, or approximately 7×10^{-4} to 8×10^{-2} LCFs per year. All of the NRC estimates were modeled with early evacuation and calculated using MACCS2.

Table D–35 compares the annual population risks from operation of the VTR to that of commercial LWRs. It is important to note that the VTR risks are based on conservative assumptions that do not consider decay of short-lived isotopes, mitigation to limit releases, or emergency actions such as evacuation or sheltering-in-place. Thus the potential VTR impacts are likely over stated. On the other hand, the NRC-evaluated risks are based on more realistic assumptions that consider preventative and mitigation features of the LWRs, including evacuation of persons within the typical 10-mile radius emergency planning zones surrounding the LWRs. Severe accident modeling for LWRs also considers for radioisotope decay for releases that occur hours or days after the reactor shuts down.

Table D–35. Summary of the Total Annual Risks for LWR and VTR Severe Accidents

Severe Accidents with VTR and LWRs	*Severe Accident Frequency (per year)* [a,b]	*50-Mile Population Risk (person-rem per year)* [a,b]	*50-Mile Population Risk (LCF risk per year)* [a,b,c]
Unmitigated, Hypothetical Beyond-Design-Basis Reactor Accident: INL	Hypothetical - Estimated ~10^{-7} or less	0.019	3×10^{-5}
Unmitigated, Hypothetical Beyond-Design-Basis Reactor Accident: ORNL	Hypothetical - Estimated ~10^{-7} or less	0.79	8×10^{-4}
Highest Risk: Upper Range of NRC LWRs – with Evacuation	2.4×10^{-4}	69	8×10^{-2}
Mean Risk: NRC LWRs – with Evacuation	3.1×10^{-5}	15	2×10^{-2}
Lowest Risk: Lower Range of NRC LWRs – with Evacuation	1.9×10^{-6}	0.55	7×10^{-4}

INL= Idaho National Laboratory; ORNL= Oak Ridge National Laboratory; LCF=latent cancer fatality; MEI = maximally exposed individual; rem = roentgen equivalent man; VTR = Versatile Test Reactor, NRC=Nuclear Regulatory Commission.

[a] VTR risk calculated using the projected impacts in Table D–33.

[b] LWR data from Table 5018 of NUREG-2226 (NRC 2019a).

[c] LCF risks for VTR calculated from projected LCFs in MACCS. LCF risks for LWRs calculated by multiplying reported person-rem per year in NRC 2019a, Table 5-18 by 2 times 0.0006 LCFs per person-rem.

[6] Under the Atomic Energy Act (AEA) of 1954 and its amendments, and the AEA Energy Reauthorization Act (ERA) of 1974, DOE has the authority to develop, construct and operate its own reactors. Under this authority, DOE plans to conduct the safety review for the VTR and authorize its construction and operation. DOE research reactor facilities, such as the VTR, are exempt from Nuclear Regulatory Commission (NRC) licensing in accordance with Section 110 of the ERA and Title 10, Code of Federal Regulations 50.11, Exceptions and Exemptions from Licensing Requirements. The VTR Project will consider guidance developed for non-LWRs, such as NRC Regulatory Guide 1.232, Guidance for Developing Principal Design Criteria for non-Light Water Reactors, and the recently developed NEI 18-04, Risk-Informed Performance-Based Technology Guidance for non-Light Water Reactors, in the development of VTR safety basis documentation.

Table D–35 indicates that the 50-mile person-rem risk for the VTR is much smaller than that of the typical LWR. A portion of this reduction is due to lower power levels in the VTR versus a LWR, but the major factor leading to the reduction is the much lower annual core damage frequency, on the order of 10^{-7} or less for the VTR versus 3.1×10^{-5} for the mean risk LWR.

D.4.9.7 Comparison of Versatile Test Reactor Hypothetical Beyond-Design-Basis Accident Risk to DOE Nuclear Safety Policy

To provide further insight into the safety of the VTR, the VTR risk estimates for the unmitigated, hypothetical, beyond-design-basis accident are compared to the DOE nuclear safety policy and Safety Goals for all DOE Operations (DOE 2011, DOE P 420.1).

DOE Nuclear Safety Policy and Safety Goals

DOE has established a nuclear safety policy (DOE P 420.1) and established safety goals for the conduct of its operations. The DOE Safety Goal is "to conduct its operations such that (a) Individual members of the public be provided a level of protection from the consequences of DOE operations such that individuals bear no significant additional risk to life and health to which members of the general population are normally exposed, and (b) DOE workers' health and safety are protected to levels consistence with or better than that achieved for workers in similar industries."

The following two quantitative safety objectives for public protection are established as "aiming points" (not requirements) in support of the Safety Goal that guides the development of DOE's nuclear safety requirements and standards:

- The risk to an average individual in the vicinity of a DOE nuclear facility for prompt fatalities that might result from accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the population are generally exposed. For evaluation purposes, individuals are assumed to be located within one mile of the site boundary (DOE P 420.1).
- The risk to the population in the area of a DOE nuclear facility for cancer fatalities that might result from operations should not exceed one-tenth of one percent (0.1%) of the sum of all cancer fatality risks resulting from all other causes. For evaluation purposes, individuals are assumed to be located within 10 miles of the site boundary (DOE P 420.1).

NRC Safety Goals

Even though the VTR is not subject to NRC regulation, consideration of the NRC safety goals provides additional assurance that the VTR can be safely operated. The NRC has set safety goals for average individual early fatality and LCF risks from reactor accidents in the Safety Goal Policy Statement (51 FR 30028). The Safety Goal Policy Statement expressed the Commission's policy regarding the acceptance level of radiological risk from nuclear power plant operation as follows (NRC 2019a):

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative health objectives are used in determining achievement of the NRC safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

The two DOE quantitative safety objectives for public are similar to the NRC quantitative health objectives. NRC interpreted the two quantitative health objectives by translating them into two numerical objectives (NRC 2019a):

- The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," is about 4.0×10^{-4} per year, including a 1.3×10^{-4} per year risk associated with transportation accidents; one-tenth of 1 percent of these figures implies that the individual risk of prompt fatality from a reactor accident should be less than 4×10^{-7} per reactor-year (NRC 2019a).
- "The sum of cancer fatality risks resulting from all other causes" for an individual is taken to be the cancer fatality rate in the United States, which is about 1 in 500 or 2×10^{-3} per year; one-tenth of 1 percent of this implies that the risk of cancer to the population in the area near a nuclear power plant because of its operation should be limited to 2×10^{-6} per reactor-year (NRC 2019a).

Since the DOE and NRC safety objectives are the same, the NRC quantitative risk numbers can be used for both DOE and NRC comparisons.

Comparison VTR Risks to DOE and NRC Safety Goals

As indicated above, the quantitative individual risk objective or goal and the population objective or goal for both DOE nuclear facility operations and NRC power plant operations are essentially identical and can be used for comparison to VTR risks. MACCS2 calculates average individual early fatality and LCF risks. The average individual early fatality risk is calculated using the population distribution within 1 mile of the site boundary. The average individual LCF risk is calculated using the population distribution within 10 miles of the site.

> <u>Individual Risk Goal Comparisons</u>**:** For the VTR at INL, the average individual (within I mile of the site boundary) risk of a prompt facility from the hypothetical beyond-design-basis reactor accident is 2×10^{-2} early fatalities × 1×10^{-7} accidents per year or 2×10^{-9} early fatalities per reactor year. At ORNL, the early fatality risk is 8×10^{-2} early fatalities × 1×10^{-7} accidents per year or 6×10^{-9} early fatalities per reactor year. Thus the early fatality risk from the hypothetical beyond-design-basis accident is 0.60 percent and 1.5 percent, respectively, of the DOE and NRC safety goals for prompt fatalities.

> <u>Population Risk Goal Comparisons</u>**:** For the VTR at INL, the average individual (within 10 miles of the site boundary) risk of an LCF from the hypothetical beyond-design-basis reactor accident is 1×10^{-2} LCFs × 1×10^{-7} accidents per year or 1×10^{-9} LCFs per reactor year. At ORNL, the LCF risk is 2×10^{-2} LCFs × 1×10^{-7} accidents per year or 2×10^{-9} LCFs per reactor year. Thus the LCF risk from the hypothetical beyond-design-basis accident is 0.06 percent and 0.10 percent, respectively, of the DOE and NRC safety goals for LCFs.

DOE Safety Goal Conclusion: The VTR sited at either INL or ORNL would meet the DOE the safety goals. The safety goals for prompt fatalities or latent cancers would be met by a wide margin, even with the many conservative assumptions used in the accident source term and impact calculations. It is expected that as the VTR design progresses and a seismic PRA is conducted, the design will evolve to demonstrate that the hypothetical accident postulated here both overstates the potential damage from beyond extremely unlikely events, such as a far beyond-design-basis earthquake and the likelihood of such an event.

D.4.9.8 Economic Costs of Severe Accidents

The most severe accident postulated, in terms of potential radiological consequences, is the unmitigated, hypothetical beyond-design-basis reactor accident with loss of cooling. It is expected that the ultimate design of the VTR would prevent or substantially mitigate releases if an event occurred that had the potential for partial or total loss of cooling of the reactor. Thus, both the estimated probability of this event, one in ten million or less per year and the extent of damage to the reactor core and fraction of the core released to the environment should be quite conservative. If the event were initiated by a severe earthquake, the expected damage at MFC, INL, and the region would be extensive with most structures (homes, businesses, and infrastructure) severely damaged. Initial emergency response to the earthquake damage would likely focus on saving of lives.

The economic impacts of the hypothetical beyond-design-basis reactor accident with loss of cooling are speculative. The MACCS2 computer program, which is used for the accident impact evaluations, has the capability to project economic costs, including population-dependent costs, farm dependent costs, decontamination costs, interdiction costs, emergency phase costs, and milk and crop disposal costs. These economic models were developed by Sandia National Laboratory, the MACCS2 model developer, and the NRC. The models have been used for U.S. nuclear power plant evaluations for decades. Evaluations using this MACCS2 model incorporated INL and ORNL-specific regional data developed with the SECPOP companion computer code to MACCS2. The models projected economic costs within 50 miles for the severe accidents to be 290 and 3,500 million dollars at INL and ORNL, respectively. The models' projected economic costs for the ORNL region are much higher primarily due to the higher population density and the more varied land use in that area.

D.5 Hazardous Material Releases

D.5.1 Source Terms for Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

The hazardous material source terms for accidents at either SRS, INL, or ORNL are determined for a representative set of accidents. Accidents are selected from fuel fabrication, VTR operations, spent fuel handling and treatment, post-irradiation examination, and spent fuel cask operations. Plume rise is not considered for any of the fire scenarios. Using a ground-level release for the fires provides conservative and consistent consequence results to allow a fair comparison of alternatives in this EIS.

D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure

The fire selected for analysis would affect the solid fuel material and sodium in the fuel fabrication line. The frequency of the earthquake is assumed to be extremely unlikely to beyond extremely unlikely. MAR is one assembly. All material in the fuel fabrication line would be at risk and the DR is 1. For sodium, the bounding ARF of 1×10^{-2} and an RF of 1 for combustible liquid is used (DOE 1994). The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively, are based on the airborne release of plutonium particulates formed by oxidation at elevated temperatures without self-sustained oxidation (DOE 1994). A building LPF of 1 is assumed. However, because the building debris would provide some confinement, a more realistic value is expected to be much lower. **Table D–36** summarizes MAR, release fractions, and source term for this accident scenario.

D.5.1.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure

A spill of melted spent fuel from the spent fuel treatment hot cell is assumed to occur when an earthquake causes failure of the hot cell's confinement. The frequency of such an earthquake is extremely unlikely. The hot cell enclosure and the offgas exhaust ventilation system would be expected to fail and allow the release of hazardous material released in the spill. MAR is one assembly and the DR is 1. The bounding ARF and RF values of 2.0×10^{-5} and 1, respectively, are based on a free-fall spill (less than a 3-meter drop) of aqueous solutions with a density more than 1.2 gram per cubic centimeter (DOE 1994). There is no reduction in the source term because of the hot cell confinement, and an LPF of 1 is used. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

D.5.1.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure

The fire selected for analysis would affect the sodium and fuel material during spent fuel handling and treatment. The frequency of the sodium fire is extremely unlikely. MAR is one assembly, and the DR is 1. For sodium, the bounding ARF of 1×10^{-2} and an RF of 1 for combustible liquid is used (DOE 1994). The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively, are based on the airborne release of particulates formed by oxidation at elevated temperatures (DOE 1994). There is no reduction in the source term because of confinement, and an LPF of 1 is used. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

D.5.1.4 Fire Involving Test Assembly during Seismically-Induced Confinement Failure

The fire selected for analysis would impact the test assembly during post-irradiation examination. The frequency of the earthquake is extremely unlikely. MAR is one pin of a fuel assembly, and the DR is 1. For sodium, the bounding ARF of 1×10^{-2} and an RF of 1 for combustible liquid are used (DOE 1994). The bounding ARF and RF values of 3×10^{-5} and 0.04, respectively are based on the airborne release of particulates formed by oxidation at elevated temperatures (DOE 1994). There is no reduction in the source term because of the fuel fabrication line's confinement, and an LPF of 1 is used. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

D.5.1.5 Drop of Fuel-Loaded Cask

After spent fuel has been washed, it is loaded into spent fuel storage casks pending transfer to the fuel treatment site. The accident is assumed to occur when a spent fuel storage cask is dropped during handling. Equipment failure or human error is assumed to cause the cask drop. The release is assumed to occur because of damage to the contents of one spent fuel storage cask. The release is assumed to occur at ground level over 10 minutes. The assumed accident frequency is extremely unlikely. MAR is the equivalent of six assemblies. The DR is 0.5 because not all fuel in the cask is expected to be involved. The ARF of 4×10^{-5} and an RF of 1 for aerodynamic entrainment and resuspension of surface contaminated waste are used (DOE 1994). The ARF and RF are 1 for sodium. The dropped cask is assumed to continue providing some confinement after the drop. Consequently, the LPF is assumed 0.5. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

Table D–36. Hazardous Material Source Terms

Accident	Material at Risk	Damage Ratio	Airborne Release Fraction	Respirable Fraction	Leak Path Factor	Sodium Source Term (grams)	Uranium Source Term (grams)
D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	1 Assembly	1	3×10^{-5}	0.04	1		0.042
		1	1×10^{-2}	1	1	9.5	
D.5.1.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	1 Assembly	1	2×10^{-5}	1	1	0.019	0.71
D.5.1.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	1 Assembly	1	3×10^{-5}	0.04	1		0.042
		1	1×10^{-2}	1	1	9.5	
D.5.1.4 Fire Involving Test Assembly (Seismically-Induced Confinement Failure)	1 Pin	1	3×10^{-5}	0.04	1		2.0×10^{-4}
		1	1×10^{-2}	1	1	0.044	
D.5.1.5 Drop of Fuel-Loaded Cask	6 Assemblies	0.5	4×10^{-5}	1	0.5	0.057	2.1
D.5.1.6 Sodium Fire due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building	4,000 kg Na	1	1×10^{-2}	1	1	40,000	-

Kg = kilogram; Na = sodium.
Notes: From Appendix B: 945.252 grams of sodium per assembly; 35,371 grams of uranium per assembly.

D.5.1.6 Sodium Fire due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building

The accident is assumed to occur when an earthquake causes a breach of the secondary heat removal system main branch piping. The flowrate in each loop is approximately 800 kilograms per second. A full breach of the piping in one loop is assumed to continue for five seconds. Thus, approximately 4,000 kilograms of sodium could be released and oxidized. The release is assumed to occur at ground level over 10 minutes. The assumed accident frequency is extremely unlikely. MAR is 4,000 kilograms of sodium. The DR is 1 because all of the sodium is assumed involved. The bounding ARF of 1×10^{-2} and an RF of 1 for combustible liquid is used (DOE 1994). Because the sodium is released outside of the reactor building, an LPF of 1 is assumed. Table D–36 summarizes MAR, release fractions, and source term for this accident scenario.

D.5.2 Hazardous Material Impacts of Facility Accidents at Idaho National Laboratory, Oak Ridge National Laboratory, and Savannah River Site

Sodium and uranium are potential materials of concern for VTR-related activities. Hazardous material releases are evaluated by calculating the concentration of sodium hydroxide (the product formed by reaction of sodium with water) and uranium. The concentrations are compared to the DOE protective action criteria (PAC) to evaluate the hazard to human health. The concentration is calculated using the χ/q value from the site-specific MACCS2 calculation in conjunction with the release rate for the hazardous material. The release rate is calculated assuming that the quantity of the hazardous material is released over ten minutes (600 seconds). The resulting concentration is:

Concentration = Release Rate x χ/q

Where

Concentration	= hazardous material concentration (mg/m3 [milligrams per cubic meter])
Release Rate	= ST x (1000 milligrams per gram) per second)
ST	= source term (grams)
t	= release duration (seconds)
χ/q	= dispersion coefficient (second per cubic meter)

The ground-level dispersion coefficients from MACCS2 for INL, ORNL, and SRS are shown in **Table D–37**. The noninvolved worker is assumed to be downwind at a point of 330 feet from the accident. The MEI is located downwind at a point of 3.1 miles, 0.5 miles, and 5.5 miles from the accident at INL, ORNL, and SRS, respectively.

Table D–37. Ground Level Dispersion Coefficients χ/q

Site	*Ground-Level Dispersion Coefficients χ/q (s/m^3)*	
	Noninvolved Worker	*Maximally Exposed Individual*
INL	1.45×10^{-3}	1.84×10^{-6}
ORNL	3.81×10^{-3}	1.46×10^{-4}
SRS	3.41×10^{-3}	1.81×10^{-6}

INL= Idaho National Laboratory; ORNL = Oak Ridge National Laboratory; s/m^3 = seconds per cubic meter; SRS = Savannah River Site.

The PACs for sodium hydroxide and uranium are based on Emergency Response Planning Guidelines (ERPGs) and Temporary Emergency Exposure Limits (TEELs), respectively. The ERPGs and TEELs estimate the concentrations at which most people will begin to experience health effects if they are exposed to a hazardous airborne chemical for one hour. (Sensitive members of the public—such as children, seniors, or the chronically ill—are not covered by these guidelines. They may experience adverse effects at concentrations below the ERPG or TEEL values.) A chemical may have up to three ERPG or TEEL values, each of which corresponds to a specific tier of health effects. TEELs are intended for use until Acute Exposure Guideline Levels (AEGLs) or ERPGs are adopted for chemicals. The three ERPG tiers and TEEL tiers are defined as follows:

- ERPG-3 is the maximum concentration in air below which nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects.
- ERPG-2 is the maximum concentration in air below which nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.
- ERPG-1 is the maximum concentration in air below which nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.
- TEEL-3 is the airborne concentration (expressed as ppm [parts per million] or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, when exposed for more than one hour, could experience life-threatening adverse health effects or death.
- TEEL-2 is the airborne concentration (expressed as ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, when exposed for more than one hour, could experience irreversible or other serious, long-lasting, adverse health effects or an impaired ability to escape.
- TEEL-1 is the airborne concentration (expressed as ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, when exposed for more than one hour, could experience notable discomfort, irritation, or certain asymptomatic, non-sensory effects. However, these effects are not disabling, but they are transient and are reversible upon cessation of exposure.

The ERPG values for sodium hydroxide and the TEEL values for uranium are shown in **Table D–38** (DOE 2018).

Table D–38. Emergency Response Planning Guideline Values for Sodium and Uranium

Sodium Hydroxide				*Uranium*		
ERPG-1 (mg/m³)	***ERPG-2 (mg/m³)***	***ERPG-3 (mg/m³)***		***TEEL-1 (mg/m³)***	***TEEL-2 (mg/m³)***	***TEEL-3 (mg/m³)***
0.5	5	50		0.6	5	30

ERPG = Emergency Response Planning Guidelines; mg/m³ = milligrams per cubic meter; TEEL = Temporary Emergency Exposure Limit.

Sodium hydroxide (1.739 grams sodium hydroxide per gram of sodium) and uranium concentrations are presented in **Table D–39, D–40,** and **D-41** for INL, ORNL, and SRS (respectively) along with ERPG values, fractions of ERPG values, TEEL values, and fractions of TEEL values. Inspection of the concentrations in the tables shows that all of the uranium concentrations for the noninvolved worker are less than the TEEL-2 value for uranium. All of the uranium concentrations for the MEI are less than the TEEL-1 value for uranium. For the noninvolved worker and the MEI, the sodium hydroxide concentrations are less than the corresponding ERPG-2 and ERPG-1 values for sodium hydroxide for all events except the "Sodium Fire Due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building" event.

Table D–39. Sodium and Uranium Concentration for Accidents at Idaho National Laboratory

Accident	*Sodium Release Rate (mg/s)*	*Uranium Release Rate (mg/s)*	*Noninvolved Worker*		*Maximally Exposed Individual*	
			NaOH Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of ERPG-2 [a]	*U Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of TEEL-2*	*NaOH Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of ERPG-1* [a]	*Uranium Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of TEEL-1*
D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	16	0.070	0.040 / 5 / 0.0081	1.0×10^{-4} / 5 / 2.0×10^{-5}	5.1×10^{-5} / 0.5 / 1.0×10^{-4}	1.3×10^{-7} / 0.6 / 2.1×10^{-7}
D.5.1.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	0.032	1.2	8.1×10^{-5} / 5 / 1.6×10^{-5}	0.0017 / 5 / 3.5×10^{-4}	1.0×10^{-7} / 0.5 / 2.0×10^{-7}	2.2×10^{-6} / 0.6 / 3.7×10^{-6}
D.5.1.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	16	0.070	0.040 / 5 / 0.0081	1.0×10^{-4} / 5 / 2.0×10^{-5}	5.1×10^{-5} / 0.5 / 1.0×10^{-4}	1.3×10^{-7} / 0.6 / 2.1×10^{-7}
D.5.1.4 Fire Involving Test Assembly During Seismically-Induced Confinement Failure	0.073	3.3×10^{-4}	1.8×10^{-4} / 5 / 3.7×10^{-5}	4.8×10^{-7} / 5 / 9.6×10^{-8}	2.3×10^{-7} / 0.5 / 4.7×10^{-7}	6.1×10^{-10} / 0.6 / 1.0×10^{-9}
D.5.1.5 Drop of Fuel-Loaded Cask	0.095	3.5	2.4×10^{-4} / 5 / 4.8×10^{-5}	0.0051 / 5 / 0.0010	3.0×10^{-7} / 0.5 / 6.1×10^{-7}	6.4×10^{-6} / 0.6 / 1.1×10^{-5}
D.5.1.6 Sodium Fire Due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building	67,000	-	170 / 5 / 34	-	0.21 / 0.5 / 0.43	-

ERPG = Emergency Response Planning Guidelines; HRS = heat removal system; mg/m^3 = milligrams per cubic meter; mg/s = milligrams per second; NaOH = sodium hydroxide; RVACS = Reactor Vessel Auxiliary Cooling System; TEEL = Temporary Emergency Exposure Limit.

[a] 1.739 grams NaOH per gram Na.

Table D–40. Sodium and Uranium Concentration for Accidents at Oak Ridge National Laboratory

Accident	Sodium Release Rate (mg/s)	Uranium Release Rate (mg/s)	Noninvolved Worker		MEI	
			NaOH Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of ERPG-2 [a]	Uranium Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of TEEL-2	NaOH Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of ERPG-1 [a]	Uranium Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of TEEL-1
D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	16	0.070	0.11 / 5 / 0.021	2.7×10^{-4} / 5 / 5.3×10^{-5}	0.0041 / 0.5 / 0.0081	1.0×10^{-5} / 0.6 / 1.7×10^{-5}
D.5.1.2 Spill of Melted Spent Fuel with Seismically-Induced Confinement Failure	0.032	1.2	2.1×10^{-4} / 5 / 4.2×10^{-5}	0.0046 / 5 / 9.1×10^{-4}	8.1×10^{-6} / 0.5 / 1.6×10^{-5}	1.8×10^{-4} / 0.6 / 2.9×10^{-4}
D.5.1.3 Sodium Fire Involving Spent Fuel with Cladding and Confinement Failure	16	0.070	0.11 / 5 / 0.021	2.7×10^{-4} / 5 / 5.3×10^{-5}	0.0041 / 0.5 / 0.0081	1.0×10^{-5} / 0.6 / 1.7×10^{-5}
D.5.1.4 Fire Involving Test Assembly During Seismically-Induced Confinement Failure	0.073	3.3×10^{-4}	4.8×10^{-4} / 5 / 9.7×10^{-5}	1.3×10^{-6} / 5 / 2.5×10^{-7}	1.9×10^{-5} / 0.5 / 3.7×10^{-5}	4.8×10^{-8} / 0.6 / 8.0×10^{-8}
D.5.1.5 Drop of Fuel-Loaded Cask	0.095	3.5	6.3×10^{-4} / 5 / 1.3×10^{-4}	0.013 / 5 / 0.0027	2.4×10^{-5} / 0.5 / 4.8×10^{-5}	5.1×10^{-4} / 0.6 / 8.5×10^{-4}
D.5.1.6 Sodium Fire Due to Failure of the Secondary Heat Removal System Main Branch Piping Outside of the Reactor Building	67,000	-	440 / 5 / 89	-	17 / 0.5 / 34	-

ERPG = Emergency Response Planning Guidelines; HRS = heat removal system; MEI = maximally exposed individual; mg/m^3 = milligrams per cubic meter; mg/s = milligrams per second; NaOH = sodium hydroxide; RVACS = Reactor Vessel Auxiliary Cooling System; TEEL = Temporary Emergency Exposure Limit.

[a] 1.739 grams NaOH per gram Na.

Table D–41. Sodium and Uranium Concentration for Accidents at Savannah River Site

Accident	Sodium Release Rate (mg/s)	Uranium Release Rate (mg/s)	Noninvolved Worker		MEI	
			NaOH Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of ERPG-2 [a]	Uranium Concentration (mg/m^3) / ERPG-2 (mg/m^3) / Fraction of TEEL-2	NaOH Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of ERPG-1 [a]	Uranium Concentration (mg/m^3) / ERPG-1 (mg/m^3) / Fraction of TEEL-1
D.5.1.1 Fire Impingement on Fuel Material with Seismically-Induced Confinement Failure	16	0.070	0.095 / 5 / 0.019	2.4×10^{-4} / 5 / 4.8×10^{-5}	5.0×10^{-5} / 0.5 / 1.0×10^{-4}	1.3×10^{-7} / 0.6 / 2.1×10^{-7}

ERPG = Emergency Response Planning Guidelines; MEI = maximally exposed individual; mg/m^3 = milligrams per cubic meter; mg/s = milligrams per second; NaOH = sodium hydroxide; SRS = Savannah River Site; TEEL = Temporary Emergency Exposure Limit.

[a] 1.739 grams NaOH per gram Na.

D.6 References

BEA (Battelle Energy Alliance), 2020, "VTR Accidents Overview-Review-Fin.docx," email from J. Andrus, INL, to D. Outlaw, Leidos, February 13.

Brunett, A. J., M. Bucknor, and D. Grabaskas, 2016, *Toward a Mechanistic Source Term in Advanced Reactors: Characterization of Radionuclide Transport and Retention in a Sodium Cooled Fast Reactor*. Conference Paper Manuscript, 2016 International Congress on Advances in Nuclear Power Plants ICAPP 2016 Nuclear Engineering Division, Argonne National Laboratory, Argonne, Illinois, April.

Bucknor, M., M. Farmer, and D. Grabaskas, 2017, *An Assessment of Fission Product Scrubbing in Sodium Pools Following a Core Damage Event in a Sodium Cooled Fast Reactor*, International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development, FR17. Nuclear Engineering Division, Argonne National Laboratory, Argonne, Illinois, June.

Cahalan, J., 2008, "Sodium Fast Reactor Safety #2," Argonne National Laboratory, DOE/HQ October 31, 2007, NRC/White Flint November 1, 2007, Rev. 1, October.

Clark, A. J., M. R. Denman, and D. Grabaskas, 2017, *Mechanistic Source Term Modeling for Sodium Fast Reactors*, SAND2017-0454C, Sandia National Laboratories, Albuquerque, New Mexico.

DOE (U.S. Department of Energy), 1994*, DOE Handbook: Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, Vol. 1: Analysis of Experimental Data, DOE-HDBK-3010, Reaffirmed 2013, December.

DOE (U.S. Department of Energy), 1999, S*urplus Plutonium Disposition Final Environmental Impact Statement,* DOE/EIS-0283, Office of Fissile Materials Disposition, Washington, DC, November.

DOE (U.S. Department of Energy), 2002, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities,* DOE-STD-1020-2002, Washington, DC, January.

DOE (U.S. Department of Energy), 2003, *Estimating Radiation Risk from Total Effective Dose Equivalent (TEDE), ISCORS Technical Report No. 1,* DOE/EH-412/0015/0802, Rev. 1, Office of Environmental Policy and Guidance, Washington, DC, January.

DOE (U.S. Department of Energy), 2004a, *MACCS2 Computer Code Application Guidance for Documented Safety Analysis*, DOE-EH-4.2.1.1-MACCS2-Code Guidance, Office of Environmental, Safety, and Health, Washington, DC, June.

DOE (U.S. Department of Energy), 2004b, *Recommendations for the Preparation of Environmental Assessments and Environmental Impact Statements,* Second Edition, Office of Environment, Safety and Health, Office of NEPA Policy and Compliance, Washington, DC, December.

DOE (U.S. Department of Energy), 2005, *Facility Safety*, DOE Order 420.1B, Office of Environment, Safety and Health, Washington, DC, December 22.

DOE (U.S. Department of Energy), 2006, *Accident Analysis for Aircraft Crash into Hazardous Facilities,* DOE-STD-3014-2006, Washington, DC, May.

DOE (U.S. Department of Energy), 2008, Global Nuclear Energy Partnership Programmatic Environmental Impact Statement, DOE/EIS-0396, Draft, October 1.

DOE (U.S. Department of Energy), 2011, *Department of Energy Nuclear Safety Policy*, DOE P 420.1, Office of Environment, Safety and Health, Washington, DC, February 8.

DOE (U.S. Department of Energy), 2012a, *Facility Safety*, DOE Order 420.1C, Office of Environment, Safety and Health, Washington, DC, December 4.

DOE (U.S. Department of Energy), 2012b, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities,* DOE-STD-1020-2012, Washington, DC, December

DOE (U.S. Department of Energy), 2015, *Final Surplus Plutonium Disposition Supplemental Environmental Impact Statement*, DOE/EIS-0283-S2, Office of Environmental Management and National Nuclear Security Administration, Office of Material Management and Minimization, Washington, DC, April.

DOE (U.S. Department of Energy), 2016, *Natural Phenomena Hazards Analysis and Design Criteria for DOE Facilities,* DOE-STD-1020-2016, Washington, DC, December.

DOE (U.S. Department of Energy), 2018, *Protective Action Criteria (PAC) with AEGLs, ERPGs, & TEELs*, Office of Environment, Health, Safety and Security, Rev. 29a, Washington, DC, June.

DOE/NNSA (U.S. Department of Energy/National Nuclear Security Administration), 2012, *NEPA Source Document for Pit Disassembly and Conversion (PDC) Project*, SRB-25.02-12-0001, Rev. 1, Office of Defense Nuclear Nonproliferation, Denver, Colorado, June 18.

EPA (U.S. Environmental Protection Agency), 1999, *Cancer Risk Coefficients for Environmental Exposure to Radionuclides*, Federal Guidance Report No. 13, EPA 402-R-99-001, Office of Radiation and Indoor Air, Washington, DC, September.

EPA (U.S. Environmental Protection Agency), 2002, *Federal Guidance Report No. 13: CD Supplement* (accessed March 4, 2019, at https://www.epa.gov/radiation/federal-guidance-report-no-13-cd-supplement), Washington, DC.

GEH (GE Hitachi Nuclear Energy), 2019, Versatile Test Reactor (VTR) Probabilistic Risk Assessment (PRA) Summary, GEH ID: 005N3526 VTR PRA Summary Report, Revision D Draft, GEH Services Engineering - PRA Group, October. **Controlled Information**

Grabaskas, D., A. Brunett, M. Bucknor, J. Sienicki, and T. Sofu, 2015, *Regulatory Technology Development Plan, Sodium Fast Reactor, Mechanistic Source Term Development,* ANL-ART-3, Nuclear Engineering Division, Argonne National Laboratory, Argonne, Illinois, February 28.

Grabaskas, D., M. Bucknor, and J. Jerden, 2016a, *Regulatory Technology Development Plan, Sodium Fast Reactor, Mechanistic Source Term – Metal Fuel Radionuclide Release*, ANL-ART-38, Nuclear Engineering Division, Argonne National Laboratory, Argonne, Illinois, February.

Grabaskas, D., M. Bucknor, J. Jerden, A. J. Brunett, M. Denman, A. Clark, and R. S. Denning, 2016b, *Regulatory Technology Development Plan, Sodium Fast Reactor, Mechanistic Source Term – Trial Calculation*, ANL-ART-49, Vol. 1, Nuclear Engineering Division, Argonne National Laboratory, Argonne, Illinois, October.

Grabaskas, D., M. Bucknor, J. Jerden, and R. Denning, 2017, A Mechanistic Source Term Calculation for a Metal Fuel Sodium Fast Reactor, Conference Paper Manuscript APT#: 132620, Nuclear Engineering Division, Argonne National Laboratory, Argonne, Illinois, June.

Grabaskas, D., 2019, Versatile Test Reactor (VTR), Probabilistic Risk Assessment (PRA), *Preliminary Mechanistic Source Term Analysis*, ANL-VTR-48, ECAR-4737, Rev. 0, Nuclear Science and Engineering Division, Argonne National Laboratory, Argonne, Illinois, September.

ICRP (International Commission on Radiological Protection), 1991, "1990 Recommendations of the International Commission on Radiological Protection," ICRP Publication 60, *Annals of the ICRP*, Pergamon Press, Elmsford, New York.

INL (Idaho National Laboratory), 2019, "Safety Analysis for the Versatile Test Reactor Conceptual Design," ECAR No.: 4733, Rev. 1, Project No. 32833, November 4.

INL (Idaho National Laboratory), 2020a, VTR Hazards and Impacts Information in Support of National Environmental Policy Act Data Needs, INL/EXT-19-55588, July.

INL (Idaho National Laboratory), 2020b, "Evaluation of the VTR Ex-Vessel Inhalation Dose Consequences," ECAR No.: 4777, Rev. 0, Project No. 32833.

INL (Idaho National Laboratory), 2020c, *Conceptual Safety Design Report for the Versatile Test Reactor,* Draft, CSDR-422, Rev. 0. **OUO, Export Controlled Information**

Marschman, S. C., C. E. Baily, K. C. Marsden, and D. C. Crawford, 2020, VTR Fuel Facility Plan, INL/LTD-19-54001, Idaho National Laboratory, Materials and Fuels Complex, Idaho Falls, Idaho, May. **OUO**

Middleton, B. D., E. J. Parma, T. J. Olivier, J. Phillips, and J. L. LaChance, 2011, *The Development of a Realistic Source Term for Sodium-Cooled Fast Reactors: Assessment of Current Status and Future Needs*, SAND2011-3404, Sandia National Laboratories, Albuquerque, New Mexico.

NCRP (National Council on Radiation Protection and Measurements), 1993, *Risk Estimates for Radiation Protection*, NCRP Report No. 115, Bethesda, Maryland, December 31.

NRC (U.S. Nuclear Regulatory Commission), 1990, *MELCOR Accident Consequence Code System (MACCS),* NUREG/CR-4691 Vol. 3, Division of Systems Research, Office of Nuclear Regulatory Research, Washington, DC, February.

NRC (U.S. Nuclear Regulatory Commission), 1998, *Code Manual for MACCS2: Volume 1, User's Guide,* NUREG/CR-6613, Office of Nuclear Regulatory Research, Washington, DC, May.

NRC (U.S. Nuclear Regulatory Commission), 2000, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, Regulatory Guide 1.183, Office of Nuclear Regulatory Research, Washington, DC, July.

NRC (U.S. Nuclear Regulatory Commission), 2005, *Environmental Impact Statement on the Construction and Operation of a Proposed Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina,* NUREG-1767, Office of Nuclear Material Safety and Safeguards, Washington, DC, January.

NRC (U.S. Nuclear Regulatory Commission), 2019a, *Final Environmental Impact Statement for an Early Site Permit at the Clinch River Nuclear Site*, NUREG-2226, Vol. 1 and 2, Office of New Reactors, Washington, DC, and U.S. Army Corps of Engineers, Nashville District, Nashville, Tennessee, April.

NRC (U.S. Nuclear Regulatory Commission), 2019b, "SecPop Version 4: Sector Population, Land Fraction, and Economic Estimation Program User Guide, Model Manual, and Verification Report," NUREG/CR-6525, Rev. 2, Office of Nuclear Regulatory Research, Washington, DC, June.

ORNL (Oak Ridge National Laboratory), 1998, *Technical Report for Generic Site Add-On Facility for Plutonium Polishing, Fissile Materials Disposition Program*, ORNL/TM-13662, June.

PMC (Project Management Corporation), 1976, Clinch River Breeder Reactor Project, *Preliminary Safety Analysis Report*, p. 15 A-1, Amendment 40.

PSID (Preliminary Safety Information Document), 1987, *PRISM Preliminary Safety Information Document*, Volume 1, Chapters 1-4, GEFR-00793, UC-87TA, December 31.

Radder, J. A, 2006, *Accident Analysis for Aircraft Crash into 105-K*, S-CLC-K-00182, Rev. 2, Washington Savannah River Company, Aiken, South Carolina, June.

SNL (Sandia National Laboratories), 2007, *WinMACCS, a MACCS2 Interface for Calculating Health and Economic Consequences from Accidental Release of Radioactive Materials into the Atmosphere User's Guide and Reference Manual for WinMACCS Version 3*, U.S. Nuclear Regulatory Commission, Washington, DC, August.

SRNL (Savannah River National Laboratory), 2020, *Conceptual Assessment of VTR Add-on Processing Capability*, SRNL-TR-2020-00171, Rev. 2, Aiken, South Carolina, July 22.

SRNS (Savannah River Nuclear Solutions, LLC), 2016, *K-Area Complex Documented Safety Analysis*, WSRC-SA-2002-00005, Rev. 12, Aiken, South Carolina, December. UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION

SRNS (Savannah River Nuclear Solutions, LLC), 2020, Savannah River Site Data Call Response for the Versatile Test Reactor Fuel Fabrication Facility, SRNS-RP-2020-00286, Rev. 1, Aiken, South Carolina, July 22.

Wootan, D. W., R. P. Omberg, and C. Grandy, 2017, Lessons Learned from Fast Flux Test Facility Experience, Published in the IAEA Proceedings of the International Conference on Fast Reactor and Related Fuel Cycles, Yekaterinburg, Russia, June 26-29.

D.7 Attachment D1: Isotopic Composition in Reactor-VTR Fuel and 220-Day Cooled VTR Fuel

Table D–42. Isotopic Composition In-Reactor and 220-Day Cooled VTR Fuel

Isotope	*Single 6% Burnup Assembly mass (grams)*	*Single 6% Burnup Assembly (curies)*	*VTR Assembly Decayed 220 Days mass (grams)*	*VTR Assembly Decayed 220 Days (curies)*
Am-241	$1.43{\times}10^{1}$	$4.91{\times}10^{1}$	$2.34{\times}10^{1}$	$8.04{\times}10^{1}$
Am-242	$1.95{\times}10^{-1}$	$1.58{\times}10^{5}$	0.00	0.00
Am-242m	0.00	0.00	0.00	0.00
Am-243	5.50	1.10	0.00	0.00
Am-244	$1.61{\times}10^{-4}$	$2.03{\times}10^{2}$	0.00	0.00
Am-245	$3.36{\times}10^{-9}$	$2.10{\times}10^{-2}$	0.00	0.00
Am-246	$7.53{\times}10^{-14}$	$1.48{\times}10^{-6}$	0.00	0.00
Ba-139	$2.04{\times}10^{-2}$	$3.34{\times}10^{5}$	0.00	0.00
Ba-140	3.44	$2.52{\times}10^{5}$	$2.20{\times}10^{-5}$	1.61
Ba-141	$4.23{\times}10^{-3}$	$3.09{\times}10^{5}$	0.00	0.00
Ba-142	$2.16{\times}10^{-3}$	$2.68{\times}10^{5}$	0.00	0.00
Br-83	$1.41{\times}10^{-3}$	$2.23{\times}10^{4}$	0.00	0.00
Br-84	$5.23{\times}10^{-4}$	$3.68{\times}10^{4}$	0.00	0.00
Ce-141	8.57	$2.44{\times}10^{5}$	$7.92{\times}10^{-2}$	$2.26{\times}10^{3}$
Ce-143	$3.87{\times}10^{-1}$	$2.57{\times}10^{5}$	0.00	0.00
Ce-144	$3.76{\times}10^{1}$	$1.20{\times}10^{5}$	$2.20{\times}10^{1}$	$7.02{\times}10^{4}$
Cm-242	$6.07{\times}10^{-1}$	$2.01{\times}10^{3}$	$3.02{\times}10^{-1}$	$1.00{\times}10^{3}$
Cm-243	$5.34{\times}10^{-3}$	$2.76{\times}10^{-1}$	$5.26{\times}10^{-3}$	$2.72{\times}10^{-1}$
Cm-244	$4.16{\times}10^{-1}$	$3.37{\times}10^{1}$	$4.08{\times}10^{-1}$	$3.30{\times}10^{1}$
Cm-245	$9.79{\times}10^{-3}$	$1.68{\times}10^{-3}$	$9.79{\times}10^{-3}$	$1.68{\times}10^{-3}$
Cm-246	$1.49{\times}10^{-4}$	$4.58{\times}10^{-5}$	$1.49{\times}10^{-4}$	$4.58{\times}10^{-5}$
Cs-134	1.70	$2.20{\times}10^{3}$	1.39	$1.80{\times}10^{3}$
Cs-135	$9.98{\times}10^{1}$	$1.15{\times}10^{-1}$	$1.00{\times}10^{2}$	$1.15{\times}10^{-1}$
Cs-136	$1.08{\times}10^{-1}$	$7.92{\times}10^{3}$	$1.00{\times}10^{-6}$	$7.33{\times}10^{-2}$
Cs-137	$8.83{\times}10^{1}$	$7.69{\times}10^{3}$	$8.71{\times}10^{1}$	$7.58{\times}10^{3}$
Eu-152m	0.00	0.00	0.00	0.00
Eu-154	$7.33{\times}10^{-1}$	$1.93{\times}10^{2}$	$6.98{\times}10^{-1}$	$1.84{\times}10^{2}$
Eu-155	2.52	$1.17{\times}10^{3}$	2.31	$1.07{\times}10^{3}$
Eu-156	$1.41{\times}10^{-1}$	$8.08{\times}10^{4}$	$6.30{\times}10^{-6}$	3.61
I-129	$1.82{\times}10^{1}$	$3.21{\times}10^{-3}$	$1.82{\times}10^{1}$	$3.21{\times}10^{-3}$
I-130	$2.21{\times}10^{-3}$	$4.31{\times}10^{3}$	0.00	0.00
I-131	1.54	$1.91{\times}10^{5}$	$8.65{\times}10^{-9}$	$1.07{\times}10^{-3}$
I-132	$2.80{\times}10^{-2}$	$2.89{\times}10^{5}$	0.00	0.00
I-133	$3.31{\times}10^{-1}$	$3.75{\times}10^{5}$	0.00	0.00
I-134	$1.51{\times}10^{-2}$	$4.03{\times}10^{5}$	0.00	0.00
I-135	$1.01{\times}10^{-1}$	$3.55{\times}10^{5}$	0.00	0.00
Kr-83m	0.00	0.00	0.00	0.00
Kr-85	1.30	$5.10{\times}10^{2}$	$1.07{\times}10^{-5}$	$4.20{\times}10^{-3}$
Kr-85m	0.00	0.00	0.00	0.00
Kr-87	$2.43{\times}10^{-3}$	$6.88{\times}10^{4}$	0.00	0.00
Kr-88	$7.70{\times}10^{-3}$	$9.65{\times}10^{4}$	0.00	0.00
La-140	$4.40{\times}10^{-1}$	$2.45{\times}10^{5}$	$3.33{\times}10^{-6}$	1.85
La-141	$5.47{\times}10^{-2}$	$3.09{\times}10^{5}$	0.00	0.00
La-142	$1.91{\times}10^{-2}$	$2.73{\times}10^{5}$	0.00	0.00
La-143	$2.76{\times}10^{-3}$	$2.55{\times}10^{5}$	0.00	0.00
Mo-93	0.00	0.00	0.00	0.00

Isotope	*Single 6% Burnup Assembly mass (grams)*	*Single 6% Burnup Assembly (curies)*	*VTR Assembly Decayed 220 Days mass (grams)*	*VTR Assembly Decayed 220 Days (curies)*
Mo-99	6.98×10^{-1}	3.35×10^{5}	0.00	0.00
Nb-93m	0.00	0.00	0.00	0.00
Nb-94	1.61×10^{-4}	3.02×10^{-5}	1.61×10^{-4}	3.02×10^{-5}
Nb-95	5.96	2.33×10^{5}	1.10	4.30×10^{4}
Nb-96	1.36×10^{-3}	1.90×10^{3}	0.00	0.00
Nb-97	1.09×10^{-2}	2.93×10^{5}	0.00	0.00
Nb-97m	0.00	0.00	0.00	0.00
Nd-147	1.28	1.04×10^{5}	1.19×10^{-6}	9.63×10^{-2}
Nd-149	6.25×10^{-3}	7.60×10^{4}	0.00	0.00
Nd-151	4.51×10^{-4}	4.51×10^{4}	0.00	0.00
Np-237	1.09×10^{1}	7.69×10^{-3}	1.13×10^{1}	7.97×10^{-3}
Np-238	1.68×10^{-2}	4.35×10^{3}	0.00	0.00
Np-239	9.66	2.24×10^{6}	4.73×10^{-6}	1.10
Np-240	1.62×10^{-4}	1.95×10^{3}	0.00	0.00
Pd-107	2.85×10^{1}	1.47×10^{-2}	2.84×10^{1}	1.46×10^{-2}
Pd-109	3.86×10^{-2}	8.27×10^{4}	0.00	0.00
Pm-146	1.47×10^{-3}	6.51×10^{-1}	1.36×10^{-3}	6.02×10^{-1}
Pm-147	2.68×10^{1}	2.49×10^{4}	2.39×10^{1}	2.22×10^{4}
Pr-143	3.03	2.04×10^{5}	4.55×10^{-5}	3.06
Pr-144	1.59×10^{-3}	1.20×10^{5}	9.25×10^{-4}	6.99×10^{4}
Pr-144m	0.00	0.00	3.68×10^{-6}	6.67×10^{2}
Pr-145	5.01×10^{-2}	1.81×10^{5}	0.00	0.00
Pu-237	6.55×10^{-5}	7.97×10^{-1}	2.32×10^{-6}	2.82×10^{-2}
Pu-238	8.69	1.49×10^{2}	9.13	1.56×10^{2}
Pu-239	5.55×10^{3}	3.45×10^{2}	5.57×10^{3}	3.46×10^{2}
Pu-240	2.45×10^{3}	5.59×10^{2}	2.45×10^{3}	5.59×10^{2}
Pu-241	3.16×10^{2}	3.26×10^{4}	3.06×10^{2}	3.15×10^{4}
Pu-242	1.36×10^{2}	5.34×10^{-1}	1.36×10^{2}	5.34×10^{-1}
Pu-243	5.58×10^{-3}	1.45×10^{4}	0.00	0.00
Pu-244	3.43×10^{-4}	6.09×10^{-9}	3.43×10^{-4}	6.09×10^{-9}
Pu-245	1.72×10^{-8}	2.10×10^{-2}	0.00	0.00
Pu-246	3.02×10^{-11}	1.48×10^{-6}	0.00	0.00
Rb-86	8.44×10^{-3}	6.91×10^{2}	2.35×10^{-6}	1.92×10^{-1}
Rb-87	1.24×10^{1}	1.08×10^{-6}	1.24×10^{1}	1.08×10^{-6}
Rb-88	8.31×10^{-4}	9.97×10^{4}	0.00	0.00
Rb-89	9.14×10^{-4}	1.08×10^{5}	0.00	0.00
Rh-105	2.96×10^{-1}	2.50×10^{5}	0.00	0.00
Rh-106	2.78×10^{-5}	9.90×10^{4}	1.84×10^{-5}	6.55×10^{4}
Rh-107	1.87×10^{-3}	1.52×10^{5}	0.00	0.00
Ru-105	3.82×10^{-2}	2.57×10^{5}	0.00	0.00
Ru-106	2.98×10^{1}	9.97×10^{4}	1.97×10^{1}	6.59×10^{4}
Sb-125	2.40	2.48×10^{3}	2.07	2.14×10^{3}
Sb-126	6.31×10^{-3}	5.27×10^{2}	2.74×10^{-8}	2.29×10^{-3}
Sb-126m	0.00	0.00	0.00	0.00
Sb-127	1.15×10^{-1}	3.07×10^{4}	0.00	0.00
Sb-129	1.38×10^{-2}	7.76×10^{4}	0.00	0.00
Sb-130	6.51×10^{-4}	2.35×10^{4}	0.00	0.00
Se-79	4.01×10^{-1}	2.79×10^{-2}	4.01×10^{-1}	2.79×10^{-2}
Sm-147	3.90	8.95×10^{-8}	0.00	0.00
Sm-151	1.09×10^{1}	2.87×10^{2}	0.00	0.00
Sm-153	5.50×10^{-2}	2.41×10^{4}	0.00	0.00

Isotope	Single 6% Burnup Assembly mass (grams)	Single 6% Burnup Assembly (curies)	VTR Assembly Decayed 220 Days mass (grams)	VTR Assembly Decayed 220 Days (curies)
Sr-89	3.56	1.03×10^{5}	1.74×10^{-1}	5.05×10^{3}
Sr-90	2.47×10^{1}	3.37×10^{3}	2.43×10^{1}	3.32×10^{3}
Sr-91	4.67×10^{-2}	1.69×10^{5}	0.00	0.00
Sr-92	1.54×10^{-2}	8.90×10^{6}	0.00	0.00
Tc-101	2.73×10^{-3}	3.58×10^{5}	0.00	0.00
Tc-104	3.30×10^{-3}	3.28×10^{5}	0.00	0.00
Tc-99	6.03×10^{1}	1.02	6.10×10^{1}	1.03
Tc-99m	0.00	0.00	0.00	0.00
Te-125m	0.00	0.00	2.65×10^{-2}	4.77×10^{2}
Te-127	1.12×10^{-2}	2.96×10^{4}	1.70×10^{-5}	4.49×10^{1}
Te-127m	0.00	0.00	4.85×10^{-3}	4.57×10^{1}
Te-129	3.47×10^{-3}	7.27×10^{4}	2.04×10^{-8}	4.27×10^{-1}
Te-129m	1.43×10^{1}	0.00	2.25×10^{-5}	6.78×10^{-1}
Te-131	3.32×10^{-3}	1.91×10^{5}	0.00	0.00
Te-131m	0.00	0.00	0.00	0.00
Te-132	9.37×10^{-1}	2.84×10^{5}	0.00	0.00
Te-133	1.99×10^{-3}	2.26×10^{5}	0.00	0.00
Te-133m	0.00	0.00	0.00	0.00
Te-134	8.85×10^{-3}	2.97×10^{5}	0.00	0.00
U-232	9.03×10^{-6}	1.93×10^{-4}	2.38×10^{-5}	5.10×10^{-4}
U-233	1.13×10^{-5}	1.09×10^{-7}	1.31×10^{-5}	1.27×10^{-7}
U-234	4.18×10^{-1}	2.61×10^{-3}	4.61×10^{-1}	2.88×10^{-3}
U-235	1.20×10^{3}	2.60×10^{-3}	1.20×10^{3}	2.60×10^{-3}
U-236	7.93×10^{1}	5.13×10^{-3}	7.94×10^{1}	5.14×10^{-3}
U-237	3.82×10^{-1}	3.12×10^{4}	9.54×10^{-6}	7.79×10^{-1}
U-238	2.98×10^{4}	1.00×10^{-2}	2.97×10^{4}	9.99×10^{-3}
Xe-131m	1.43×10^{1}	0.00	1.32×10^{-7}	1.11×10^{-2}
Xe-133	0.00	3.41×10^{5}	5.20×10^{-13}	9.73×10^{-8}
Xe-133m	0.00	0.00	0.00	0.00
Xe-135	0.00	4.01×10^{5}	0.00	0.00
Xe-135m	0.00	0.00	0.00	0.00
Xe-138	2.80×10^{-3}	2.67×10^{5}	0.00	0.00
Y-90	1.43×10^{1}	3.62×10^{3}	6.18×10^{-3}	3.36×10^{3}
Y-91	5.60	1.37×10^{5}	4.17×10^{-1}	1.02×10^{4}
Y-91m	0.00	0.00	0.00	0.00
Y-92	2.03×10-2	1.95×10^{5}	0.00	0.00
Y-93	7.13×10-2	2.38×10^{5}	0.00	0.00
Y-94	2.32×10^{-3}	2.43×10^{5}	0.00	0.00
Y-95	1.45×10^{-3}	2.68×10^{5}	0.00	0.00
Zr-93	4.13×10^{1}	1.04×10-1	4.15×10^{1}	1.04×10^{-1}
Zr-95	1.06×10^{1}	2.28×10^{5}	9.89×10-1	2.12×10^{4}
Zr-97	1.51×10-1	1.20×10^{4}	0.00	0.00
Total per Assembly	**4.02×10^{4}**	**2.35×10^{7}**	**3.99×10^{4}**	**4.30×10^{5}**

Note: These are the radiologically significant 143 isotopes of the 768 isotope values provided.
Source: ECAR-4777 (INL 2020b), CSDR December 2019 (INL 2020c), or Grabaskas 2019 (ECAR-4737) for single 6 percent burnup assembly mass; email from Jason Andrus (BEA 2020) for 220 day decayed spent fuel assembly; Federal Guidance Report 13 for isotope half-life values.

D.8 Attachment D2: Isotopic Composition of 4-Year Cooled VTR Fuel

The following table presents the isotopic composition of a 4-year cooled VTR fuel assembly.

Table D–43. Isotopic Composition of a 4-Year Cooled VTR Assembly

Isotope	VTR Assembly Decayed 4 Years (curies)	Isotope	VTR Assembly Decayed 4 Years (curies)	Isotope	VTR Assembly Decayed 4 Years (curies)
Ac-225	6.65×10^{-11}	La-138	3.58×10^{-11}	Pu-243	1.45×10^{-10}
Ac-227	1.34×10^{-8}	Nb-93m	1.61×10^{-2}	Pu-244	6.20×10^{-9}
Am-241	$2.38\times10^{+2}$	Nb-94	3.02×10^{-5}	Pu-246	2.74×10^{-20}
Am-243	$1.10\times10^{+0}$	Nb-95	6.94×10^{-2}	Ra-222	1.33×10^{-29}
Am-245	7.01×10^{-17}	Nb-95m	3.60×10^{-4}	Ra-223	1.34×10^{-8}
Am-246m	2.74×10^{-20}	Nd-144	6.42×10^{-11}	Ra-224	9.40×10^{-4}
At-217	6.65×10^{-11}	Np-237	8.13×10^{-3}	Ra-225	6.65×10^{-11}
Ba-137m	$6.62\times10^{+3}$	Np-239	$1.10\times10^{+0}$	Ra-226	1.02×10^{-10}
Bi-210	4.47×10^{-11}	Np-240	7.42×10^{-12}	Rb-86	1.77×10^{-21}
Bi-211	1.34×10^{-8}	Np-240m	6.19×10^{-9}	Rb-87	1.06×10^{-6}
Bi-212	9.40×10^{-4}	Pa-231	2.19×10^{-7}	Rh-102	1.65×10^{-2}
Bi-213	6.65×10^{-11}	Pa-233	8.13×10^{-3}	Rh-103m	1.75×10^{-6}
Bi-214	1.02×10^{-10}	Pa-234	1.60×10^{-5}	Rh-106	$6.48\times10^{+3}$
Bk-249	4.83×10^{-12}	Pa-234m	9.98×10^{-3}	Rn-218	2.03×10^{-17}
Cd-113m	1.90×10^{-5}	Pb-209	6.65×10^{-11}	Rn-219	1.34×10^{-8}
Cd-115m	5.19×10^{-13}	Pb-210	4.47×10^{-11}	Rn-220	9.40×10^{-4}
Ce-139	2.51×10^{-3}	Pb-211	1.34×10^{-8}	Rn-222	1.02×10^{-10}
Ce-141	7.43×10^{-9}	Pb-212	9.40×10^{-4}	Ru-103	1.77×10^{-6}
Ce-144	$3.42\times10^{+3}$	Pb-214	1.02×10^{-10}	Ru-106	$6.48\times10^{+3}$
Cm-241	7.90×10^{-17}	Pd-107	1.46×10^{-2}	Sb-124	7.79×10^{-6}
Cm-242	$5.12\times10^{+0}$	Pm-146	3.95×10^{-1}	Sb-125	$9.16\times10^{+2}$
Cm-243	2.45×10^{-1}	Pm-147	$9.03\times10^{+3}$	Se-79	6.16×10^{-3}
Cm-244	$2.90\times10^{+1}$	Po-210	4.38×10^{-11}	Sm-146	2.16×10^{-8}
Cm-245	1.68×10^{-3}	Po-211	3.69×10^{-11}	Sm-147	5.09×10^{-7}
Cm-246	4.55×10^{-5}	Po-212	6.02×10^{-4}	Sm-151	$2.80\times10^{+2}$
Cm-247	1.45×10^{-10}	Po-213	6.50×10^{-11}	Sn-119m	1.63×10^{-7}
Cm-248	4.51×10^{-11}	Po-214	1.02×10^{-10}	Sn-121	4.70×10^{-9}
Cs-134	$5.75\times10^{+2}$	Po-215	1.34×10^{-8}	Sn-121m	6.06×10^{-9}
Cs-135	1.15×10^{-1}	Po-216	9.40×10^{-4}	Sn-123	2.15×10^{-12}
Cs-137	$6.99\times10^{+3}$	Po-218	1.02×10^{-10}	Sr-89	2.08×10^{-4}
Eu-152	7.88×10^{-1}	Pr-144	$3.42\times10^{+3}$	Sr-90	$3.10\times10^{+3}$
Eu-154	$1.43\times10^{+2}$	Pr-144m	$3.27\times10^{+1}$	Tc-98	6.90×10^{-7}
Eu-155	$6.83\times10^{+2}$	Pu-236	2.23×10^{-2}	Tc-99	$1.04\times10^{+0}$
Fr-221	6.65×10^{-11}	Pu-237	1.85×10^{-10}	Te-125m	$2.24\times10^{+2}$
Fr-223	1.85×10^{-10}	Pu-238	$1.57\times10^{+2}$	Te-127	1.69×10^{-2}
I-129	3.22×10^{-3}	Pu-239	$3.45\times10^{+2}$	Te-127m	1.72×10^{-2}
In-115m	5.52×10^{-17}	Pu-240	$5.56\times10^{+2}$	Te-129	3.32×10^{-12}
Kr-81	1.76×10^{-8}	Pu-241	$2.69\times10^{+04}$	Te-129m	5.27×10^{-12}
Kr-85	3.35×10^{-3}	Pu-242	5.38×10^{-1}	Th-226	1.33×10^{-29}
Th-227	1.32×10^{-8}				
Th-228	9.40×10^{-4}				
U-238	9.98×10^{-3}				
U-240	6.19×10^{-9}				

Isotope	***VTR Assembly Decayed 4 Years (curies)***	***Isotope***	***VTR Assembly Decayed 4 Years (curies)***	***Isotope***	***VTR Assembly Decayed 4 Years (curies)***
Xe-127	2.19×10^{-14}				
Y-89m	2.00×10^{-8}				
Y-90	$3.10\times10^{+3}$				
Y-91	4.27×10^{-3}				
Zr-93	1.04×10^{-1}				
Zr-95	3.15×10^{-2}				

VTR = Versatile Test Reactor.
Source: ECAR No. 5093, *VTR Assembly Source Term at Four-Years Decay Time*, TEM-326, Rev. 1, 01/07/2020.

D.9 Attachment D3: Nuclides Released from Inadvertent Nuclear Criticality

Table D–44. Curies of Important Nuclides Released During Nuclear Excursion Involving Plutonium Solution

Nuclide	Half-life	Radioactivity, Ci [a]		
		0 to 0.5 hour	0.5 to 8 hour	Total
Kr-83m	1.8 h	1.5×10^{1}	9.5×10^{1}	1.1×10^{2}
Kr-85m	4.5 yr	9.4	6.1×10^{1}	7.1×10^{1}
Kr-85	1.7 yr	1.2×10^{-4}	7.2×10^{-4}	8.1×10^{-3}
Kr-87	76.3 m	6.0×10^{1}	3.7×10^{2}	4.3×10^{2}
Kr-88	2.8 h	3.2×10^{1}	2.0×10^{2}	2.3×10^{2}
Kr-89	3.2 m	1.8×10^{3}	1.1×10^{4}	1.3×10^{4}
Xe-131m	11.9 d	1.4×10^{-2}	8.6×10^{-2}	1.0×10^{-1}
Xe-133m	2.0 d	3.1×10^{-1}	1.9	2.2
Xe-133	5.2 d	3.8	2.3×10^{1}	2.7×10^{1}
Xe-135m	15.6 m	4.6×10^{2}	2.8×10^{3}	3.3×10^{3}
Xe-135	9.1 h	5.7×10^{1}	3.5×10^{2}	4.1×10^{2}
Xe-137	3.8 m	6.9×10^{3}	4.2×10^{4}	4.9×10^{4}
Xe-138	14.2 m	1.5×10^{3}	9.5×10^{3}	1.1×10^{4}
I-131	8.0 d	1.5	9.5	1.1×10^{1}
I-132	2.3 h	1.7×10^{2}	1.0×10^{3}	1.2×10^{3}
I-133	20.8 h	2.2×10^{1}	1.4×10^{2}	1.6×10^{2}
I-134	52.6 m	6.0×10^{2}	3.7×10^{3}	4.3×10^{3}
I-135	6.6 h	6.3×10^{1}	3.9×10^{2}	4.5×10^{2}
Pu-238 [b]				5.9×10^{-4}
Pu-239 [b]				2.7×10^{-5}
Pu-240 [b]				5.8×10^{-5}
Pu-241 [b]				1.8×10^{-2}
Pu-242 [b]				4.3×10^{-7}
Am-241 [b]				2.4×10^{-5}

Am = americium; Ci = curies; d = day; h = hour; I = iodine; Kr = krypton; m = minute; Pu = plutonium; Xe = xenon; y = year.

[a] Total Ci, except for Pu and Am, are based on cumulative yield for fission energy spectrum. The assumption of cumulative yield is conservative, i.e., it does not consider appropriate decay schemes. Calculations regarding individual nuclide yields and decay schemes may be considered on an individual basis. Data in this table do not include the iodine reduction factor.

[b] Total radioactivity assumes the isotopic mix to be the equilibrium mix for recycled plutonium and 1 milligram of plutonium oxide released.

Source: DOE 1994.

APPENDIX E
EVALUATION OF HUMAN HEALTH EFFECTS FROM TRANSPORTATION

APPENDIX E
EVALUATION OF HUMAN HEALTH EFFECTS FROM TRANSPORTATION

E.1 Introduction

Transportation of any commodity involves a risk to both transportation crew members and members of the public. This risk results directly from transportation-related accidents and indirectly from increased levels of pollution from vehicle emissions, regardless of the cargo. The transport of certain materials, such as hazardous or radioactive waste, can pose an additional risk due to the unique nature of the material itself. To permit a complete appraisal of the environmental impacts of the *Versatile Test Reactor Environmental Impact Statement* (VTR EIS) alternatives and options, this appendix assesses the human health risks associated with the transportation of radioactive materials and wastes, as well as nonradioactive construction materials and hazardous waste, on public highways.

This appendix provides an overview of the approach used to assess the human health risks that could result from transportation of VTR-related materials. The topics in this appendix include the scope of the assessment, packaging and determination of potential transportation routes, the analytical methods used for the risk assessment (e.g., computer models), and important assessment assumptions. In addition, to aid in understanding and interpreting the results, specific areas of uncertainty are described with an emphasis on how those uncertainties may affect comparisons of the EIS alternatives.

The risk assessment results are presented in this appendix in terms of "per-shipment" risk factors, as well as the total risks for a given alternative. Per-shipment risk factors provide an estimate of the risk from a single shipment. The total risks for a given alternative are estimated by multiplying the expected number of shipments by the appropriate per-shipment risk factors.

E.2 Scope of Assessment

The scope of the transportation human health risk assessment, including transportation activities, potential radiological and nonradiological impacts, transportation modes, and receptors, is described in this section. This evaluation focuses on using offsite public highways. Additional details of the assessment are provided in the remaining sections of this appendix.

E.2.1 Transportation-Related Activities

The transportation risk assessment is limited to estimating the human health risks related to transportation for each alternative. This includes incident-free risks related to being in the vicinity of a shipment during transport or at stops, as well as accident risks. The impacts of increased transportation levels on local traffic flow or on transportation infrastructure are addressed in Chapter 4, Section 4.13, of this VTR EIS.

E.2.2 Radiological Impacts

For each alternative, radiological risks (i.e., those risks that result from the radioactive nature of the materials) are assessed for both incident-free (normal) and transportation accident conditions. The radiological risk associated with incident-free transportation conditions would result from the potential exposure of people to external radiation in the vicinity of a shipment. The radiological risk from transportation accidents would come from the potential release and dispersal of radioactive material into the environment during an accident and the subsequent exposure of people or from an accident where there is no release of radioactive material but there is external radiation exposure to the unbreached container.

All radiological impacts are calculated in terms of radiation dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent, which is the sum of the effective dose equivalent from external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure (see Title 10 of the *Code of Federal Regulations* [CFR], Part 20 [10 CFR Part 20]). Radiation doses are presented in units of roentgen equivalent man (rem) for individuals and person-rem for populations. The impacts are further expressed as health risks in terms of latent cancer fatalities (LCFs) in exposed individuals or populations using dose-to-risk conversion factors recommended by the Interagency Steering Committee on Radiation Standards guidance (DOE 2003b). A health risk conversion factor of 0.0006 LCFs per rem or person-rem of exposure is used for both the public and workers (DOE 2003b).

E.2.3 Nonradiological Impacts

In addition to radiological risks posed by transportation activities, vehicle-related risks are also assessed for nonradiological causes. (That is to say, nonradiological causes would be related to the transport vehicles, not to the radioactive cargo.) The nonradiological transportation risks, which would be incurred for similar shipments of any commodity, are assessed for accidents involving transport of radioactive and nonradioactive waste and construction materials. The nonradiological accident risk refers to the potential occurrence of transportation accidents that result in fatalities unrelated to the radioactive characteristics (e.g., radioactive nature) of the cargo.

Nonradiological risks during incident-free transportation conditions could also be caused by potential exposure to increased vehicle exhaust emissions. As explained in Section E.6.2, these emission impacts, in terms of excess latent mortalities, were not considered.

E.2.4 Transportation Modes

All shipments of radioactive and nonradioactive waste and construction materials are assumed to take place by exclusive use truck and a Motor Carrier Evaluation Program approved commercial carrier. In addition to the use of commercial carriers for transport of radioactive waste and certain types of radioactive materials, shipment of several types of radioactive materials are assumed to occur using the National Nuclear Security Administration (NNSA) Secure Transportation Asset (STA), which consists of truck transport only. (No rail transport is analyzed because rail is not part of the STA used to transport radioactive materials, and the radioactive wastes to be generated would not be transported in large enough quantities to justify rail.) Onsite and offsite shipments involving transport of special nuclear material[1] such as plutonium oxide or metal are assumed to occur using STA. Transport of unirradiated VTR fuel is also assumed to occur using the STA.

For the purpose of transporting special nuclear material, such as plutonium oxide or metal, the STA may use a specially designed tractor-trailer. Although details of vehicle enhancements and some operational aspects are classified, key elements are as follows (DOE 1999):

- Enhanced structural characteristics and a tie-down system to protect the cargo from impact
- Heightened thermal resistance to protect the cargo in case of fire

[1] Special nuclear material – as defined in Section 11 of the Atomic Energy Act: "(1) plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the U.S. Nuclear Regulatory Commission determines to be special nuclear material, or (2) any material artificially enriched by any of the foregoing."

- Established operational and emergency plans and procedures governing the shipment of nuclear materials
- Federal agents who are armed officers and have received vigorous specialized training
- An armored tractor component that provides Federal agents protection against attack and contains advanced communications equipment
- Specially designed escort vehicles containing advanced communications equipment and additional Federal agents
- 24-hour-a-day, real-time communications to monitor the location and status of all STA shipments
- Significantly more stringent maintenance standards than those for commercial transport equipment

E.2.5 Receptors

Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck crew members involved in transportation and inspection of the packages. The general public includes all persons who could be exposed to a shipment while it is moving or stopped during transit. For incident-free operation, the affected population includes individuals living within 0.5 miles of each side of the highway. Potential risks are estimated for the affected populations and the hypothetical maximally exposed individual (MEI). For incident-free operation, the MEI would be a resident living near the highway who is exposed to all shipments transported on the road. For accident conditions, the affected population includes individuals residing within 50 miles of the accident, and the MEI would be an individual located 330 feet directly downwind from the accident. The risk to the affected population is a measure of the radiological risk posed to society as a whole by the alternative being considered. As such, the impact on the affected population is used as the primary means of comparing various alternatives.

E.3 Packaging and Transportation Regulations

This section provides a general summary of radioactive materials packaging and transportation regulations. The packaging and transportation of radioactive materials are highly regulated. The U.S. Department of Transportation (DOT) and the U.S. Nuclear Regulatory Commission (NRC) have primary responsibility for Federal regulations governing commercial radioactive materials transportation. In addition, the U.S. Department of Energy (DOE) works with DOT and NRC in developing requirements and standards for radioactive materials transportation. DOE, including NNSA, has broad authority under the Atomic Energy Act of 1954, as amended, to regulate all aspects of activities involving radioactive materials that are undertaken by DOE or on its behalf, including the transportation of radioactive materials. However, in most cases that do not involve national security, DOE does not exercise its authority to regulate DOE shipments. Instead DOE uses commercial carriers that undertake shipments of DOE materials under the same terms and conditions as those used for commercial shipments. These shipments are subject to regulation by DOT and NRC. As a matter of policy, however, even in the limited circumstances where DOE exercises its Atomic Energy Act authority for shipments, DOE requirements mandate that all DOE shipments be undertaken in accordance with the requirements and standards that apply to comparable commercial shipments, unless there is a determination that national security or another critical interest requires different action.

The regulatory standards for packaging and transporting radioactive materials are designed to achieve the following four primary objectives:

- Protect persons and property from radiation emitted from packages during transportation by placing specific limitations on the allowable radiation levels.
- Contain radioactive material in the package (achieved by packaging design requirements based on performance-oriented packaging integrity tests and environmental criteria).
- Prevent nuclear criticality (an unplanned nuclear chain reaction that could occur as a result of concentrating too much fissile material in one place).
- Provide physical protection against theft and sabotage during transit.

The detailed CFR regulations pertaining to the transportation of radioactive materials are published by DOT at 49 CFR Parts 106, 107, and 171 to 178; and NRC at 10 CFR Parts 20, 61, 71, and 73. For the U.S. Postal Service, Publication 52, "Hazardous, Restricted, or Perishable Mail," specifies the quantities of radioactive material prohibited in surface mail. Interested readers are encouraged to visit the cited resources for the most current regulations or to review DOT's *Radioactive Material Regulations Review* for a comprehensive discussion on radioactive material regulations (DOT 2008).

E.3.1 Packaging Regulations

The primary regulatory approach to promote safety from radiological exposure is the specification of standards for the packaging of radioactive materials. Packaging represents the primary barrier between the radioactive materials being transported and radiation exposure to the public, workers, and the environment. Transportation packaging for radioactive materials must be designed, constructed, and maintained to contain and shield its contents during normal transport conditions. For highly radioactive material, such as high-level radioactive waste or spent nuclear fuel, packaging must contain and shield the contents in the event of a severe accident. The type of packaging used is determined by the total radioactive hazard presented by the material within the packaging. Four basic types of packaging are used: Excepted, Industrial, Type A, and Type B. Specific requirements for these packages are detailed in 49 CFR 173, Subpart I, "Class 7 (Radioactive) Materials." All packages are designed to protect and retain their content under normal operations.

Excepted packaging is limited to transporting materials with extremely low levels of radioactivity and very low external radiation. Industrial packaging is used to transport materials that, because of low levels of radioactivity, present a limited hazard to the public and the environment. Type A packaging is designed to protect and retain its contents under normal transport conditions. Because Type A packages are used to transport materials with higher radioactive content, they must maintain sufficient shielding to limit radiation exposure to handling personnel. Type A packaging, typically a 55-gallon drum or standard waste box, is commonly used to transport radioactive materials with higher concentrations or amounts of radioactivity than materials transported in Excepted or Industrial packages. Type B packaging is used to transport materials with the highest radioactivity levels and is designed to protect and retain its contents under transportation accident conditions. (These conditions are described in more detail in later sections). Packaging requirements are an important consideration for transportation risk assessment.

Radioactive materials shipped in Type A containers, or packagings, are subject to specific radioactivity limits identified as A1 and A2 values in 49 CFR 173.435 ("Table of A1 and A2 values for radionuclides"). In addition, external radiation limits, as prescribed in 49 CFR 173.441 ("Radiation level limitations and exclusive use provisions"), must be met. If the A1 or A2 limits are exceeded, the material must be shipped in a Type B package unless it can be demonstrated that the material meets the definition of "low specific activity." If the material qualifies as low specific activity, as defined in 10 CFR Part 71 ("Packaging and

Transportation of Radioactive Material") and 49 CFR Part 173 ("Shippers-General Requirements for Shipments and Packagings"), it may be shipped in a shipping container such as Industrial or Type A Packaging (49 CFR 173.427). See also DOT's *Radioactive Material Regulations Review* (DOT 2008). Type B packages, or casks, are subject to the radiation limits in 49 CFR 173.441.

Type A packaging is designed to retain its radioactive contents in normal transport. Design and test conditions that a Type A package must withstand include the following:

- Operating temperatures ranging from -40 degrees Fahrenheit (°F) to 158 °F
- External pressures ranging from 3.5 to 20 pounds per square inch
- Normal vibration experienced during transportation
- Simulated rainfall of 2 inches per hour for 1 hour
- Free fall from 1 to 4 feet, depending on the package weight
- Water immersion tests
- Impact of a 13-pound steel cylinder with rounded ends dropped from 3.3 feet onto the most vulnerable surface
- A compressive load of 5 times the mass of the gross weight of the package for 24 hours, or the equivalent of 1.9 pounds per square inch, multiplied by the vertically projected area of the package for 24 hours

Type B packaging is designed to retain its radioactive contents in both normal and accident conditions. In addition to the normal conditions outlined above, a Type B package must withstand accident conditions simulated by the following:

- Free drop from 30 feet onto an unyielding surface in a position most likely to cause damage
- Free drop from 3.3 feet onto the end of a 6-inch-diameter vertical steel bar
- Exposure to temperatures of 1,475 °F for at least 30 minutes
- For all packages, immersion in at least 50 feet of water
- For some packages, immersion in at least 3 feet of water in an orientation most likely to result in leakage
- For some packages, immersion in at least 660 feet of water for 1 hour

Compliance with these requirements is demonstrated by using a combination of simple calculation methods, computer modeling techniques, or scale-model or full-scale testing of transportation packages or casks.

E.3.2 Transportation Regulations

DOT regulates the transportation of hazardous materials in interstate commerce by land, air, and water. DOT specifically regulates the carriers of radioactive materials and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements. DOT also regulates the labeling, classification, and marking of radioactive material packagings.

NRC regulates the packaging and transportation of radioactive material for its licensees, including commercial shippers of radioactive materials. In addition, under an agreement with DOT, NRC sets the standards for packages containing fissile materials and Type B packagings.

DOE, through its management directives, orders, and contractual agreements, ensures the protection of public health and safety by imposing on its transportation activities standards that meet those of DOT and NRC. DOT recognizes in 49 CFR 173.7(d) that packagings made by or under the direction of DOE may be used for transporting Class 7 materials (radioactive materials) when the packages are evaluated, approved, and certified by DOE against packaging standards equivalent to those specified in 10 CFR Part 71.

DOT also has requirements that help reduce transportation impacts. Some requirements affect drivers, packaging, labeling, marking, and placarding. Others specify the maximum dose rate from radioactive material shipments to help reduce incident-free transportation doses.

E.4 Emergency Response

The U.S. Department of Homeland Security (DHS) is responsible for establishing and coordinating policies for civil emergency management, planning, and interaction with Federal Executive agencies that have emergency response functions in the event of a transportation incident. In the event that a transportation incident involving nuclear material occurs, guidelines for response actions have been outlined in the *National Response Framework* (DHS 2019).

The Federal Emergency Management Agency, an organization within DHS, coordinates Federal and State participation in developing emergency response plans and is responsible for the development and maintenance of the *Nuclear/Radiological Incident Annex* (DHS 2016) to the *National Response Framework* (DHS 2019). The *Nuclear/Radiological Incident Annex* to the *National Response Framework* describes the policies, situations, concepts of operations, and responsibilities of the Federal departments and agencies governing the immediate response and short-term recovery activities for incidents involving release of radioactive materials.

DHS has the authority to activate Nuclear Incident Response Teams, which include DOE Radiological Assistance Program Teams that can be dispatched from regional DOE Offices in response to a radiological incident. These teams provide first-responder radiological assistance to protect the health and safety of the general public, responders, and the environment. They assist in the detection, identification and analysis, and response to events involving radiological/nuclear material. Deployed teams provide traditional field monitoring and assessment support, as well as a search capability.

DOE uses DOE Order 151.1D *Comprehensive Emergency Management System* (DOE 2016b) as a basis for establishing a comprehensive emergency management program. The program's order provides detailed, hazard-specific planning and preparedness measures to minimize the health impacts of accidents involving loss of control over radioactive material or toxic chemicals. DOE provides technical assistance to other Federal agencies and to State and local governments. Contractors are responsible for maintaining emergency plans and response procedures for all facilities, operations, and activities under their jurisdiction and for implementing those plans and procedures during emergencies. Contractor and State and local government plans are fully coordinated and integrated. In addition, DOE established the Transportation Emergency Preparedness Program to ensure that its operating contractors and State, Tribal, and local emergency responders are prepared to respond promptly, efficiently, and effectively to accidents involving DOE shipments of radioactive material. This program is a component of the overall emergency management system established by DOE Order 151.1D.

In the event of a release of radiological cargo from a shipment along a route, local emergency response personnel would be first to arrive at the accident scene. It is expected that response actions would be taken in the context of the *Nuclear/Radiological Incident Annex* protocols. Based on their initial assessment at the scene, trained and fully equipped first responders would involve State and Federal resources as necessary. First responders or State and Federal responders would initiate actions in accordance with the DOT *2016 Emergency Response Guidebook* (DOT 2016) to isolate the incident and perform any actions necessary to protect human health and the environment. (Responses could include evacuations or other steps to reduce or prevent impacts on the public.) Cleanup actions are the responsibility of the carrier. DOE would partner with the carrier, shipper, and applicable State and local jurisdictions to ensure that cleanup actions meet regulatory requirements.

To mitigate the possibility of an accident, DOE issued DOE Manual 460.2-1A, *Radioactive Material Transportation Practices Manual for Use with DOE O 460.2A* (DOE 2008a). As specified in this manual, carriers are expected to exercise due caution and care in dispatching shipments. According to the manual, the carrier determines the acceptability of weather and road conditions, whether a shipment should be held before departure, and when actions should be taken while en route. The manual emphasizes that shipments should not be dispatched if severe weather or bad road conditions make travel hazardous. Current weather conditions, the weather forecast, and road conditions would be considered before dispatching a shipment. Conditions at the point of origin and along the entire route would be considered.

E.5 Methodology

The transportation risk assessment is based on the alternatives described in Chapter 2 of the VTR EIS. **Figure E–1** summarizes the transportation risk assessment methodology. After the alternatives were identified and the requirements of the shipping campaign were understood, data were collected on material characteristics, transportation routes, and accident parameters.

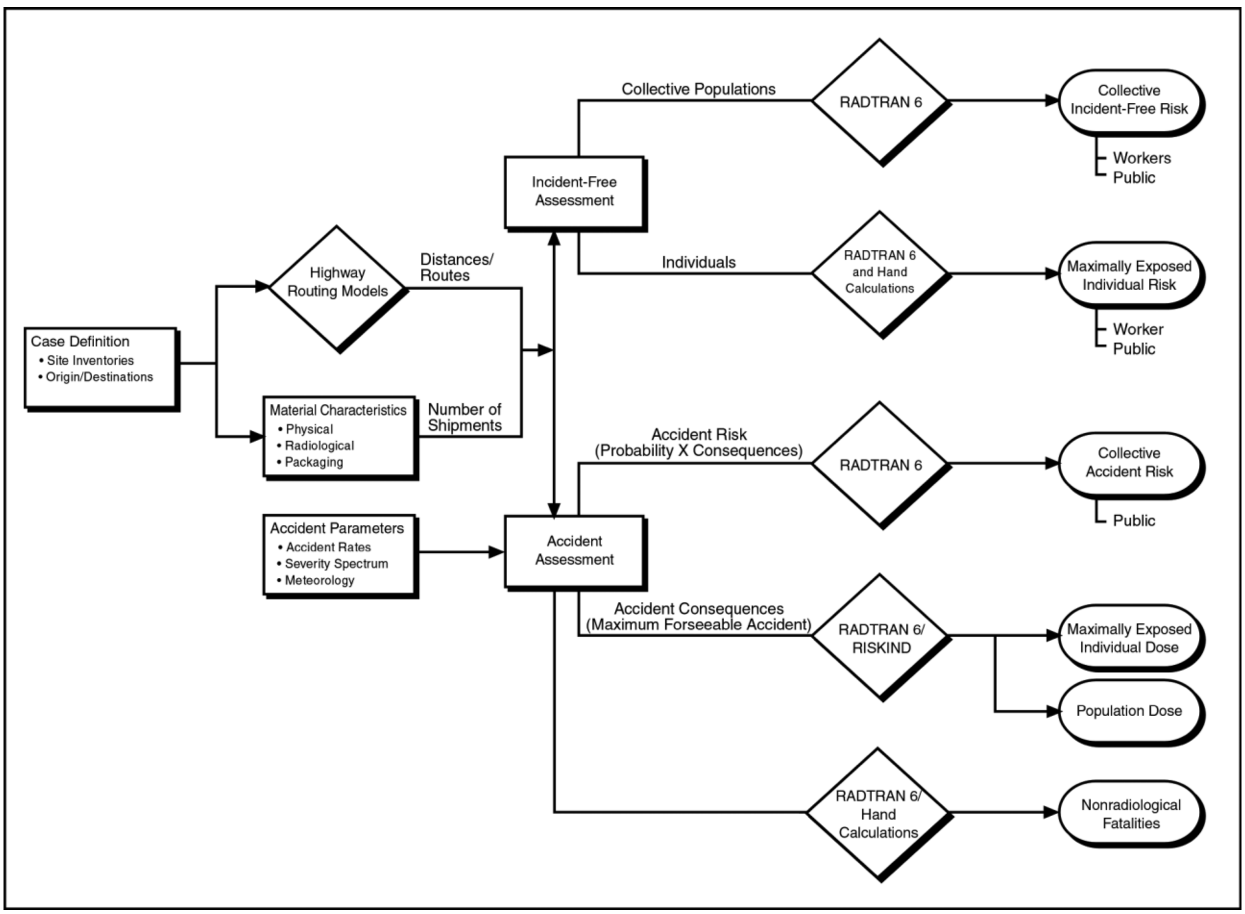

Figure E–1. Transportation Risk Assessment

Transportation impacts calculated for the VTR EIS are presented in two parts: impacts from incident-free or routine transportation and impacts from transportation accidents. Impacts of incident-free transportation and transportation accidents are further divided into nonradiological and radiological impacts. Nonradiological impacts could result from transportation accidents in terms of traffic fatalities.

Radiological impacts of incident-free transportation include impacts on members of the public and crew from radiation emanating from materials in the shipment. Radiological impacts from accident conditions consider all foreseeable scenarios that could damage transportation packages and lead to releases of radioactive materials to the environment.

The impact of transportation accidents is expressed in terms of probabilistic risk. Probabilistic risk is the probability of an accident multiplied by the consequences of that accident and summed over all reasonably conceivable accident conditions. Hypothetical transportation accident conditions ranging from low-speed "fender-bender" collisions to high-speed collisions with or without fires were analyzed. The frequencies of accidents and consequences were evaluated using a method developed by NRC and originally published in the *Final Environmental Impact Statement on the Transportation of Radioactive Materials by Air and Other Modes*, NUREG-0170 (NRC 1977) and subsequently in *Shipping Container Response to Severe Highway and Railway Accident Conditions*, NUREG/CR-4829 (NRC 1987) and *Reexamination of Spent Fuel Shipping Risk Estimates*, NUREG/CR-6672 (NRC 2000). Hereafter, these reports are cited as *Radioactive Material Transport Study,* NUREG-0170; *Modal Study*, NUREG/CR-4829; and *Reexamination Study*, NUREG/CR-6672. Radiological accident risk is expressed in terms of additional LCFs, and nonradiological accident risk is expressed in terms of additional traffic fatalities. Incident-free risk is also expressed in terms of additional LCFs.

Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck crew members involved in the actual transportation. The general public includes all persons who could be exposed to a shipment while it is moving or stopped during transit.

The first step in the ground transportation analysis was to determine the distances and populations along the routes. The Web Transportation Routing Analysis Geographic Information System (WebTRAGIS) computer program (Peterson 2018) was used to identify routes and the associated distances and populations for purposes of analysis. This information, along with the properties of the material being shipped and route-specific accident frequencies, was entered into the Radioactive Material Transportation Risk Assessment (RADTRAN) 6 computer code (Weiner et al. 2013, 2014), which calculates incident-free transport and accident risks on a per-shipment basis. The risks under each alternative were determined by summing the products of per-shipment risks for each radioactive materials shipment type by the number of shipments of that material.

The RADTRAN 6 computer code was used for incident-free and accident risk assessments to estimate the impacts on populations, as well as for incident-free assessments associated with MEIs. RADTRAN 6 was developed by Sandia National Laboratories to calculate individual and population risks associated with the transportation of radioactive materials by a variety of modes, including truck, rail, air, ship, and barge.

The RADTRAN 6 population risk calculations include both the consequences and probabilities of potential exposure events. The RADTRAN 6 code consequence analyses include the following exposure pathways: cloud shine, ground shine, direct radiation (from loss of shielding), inhalation (from dispersed materials), and resuspension (inhalation of resuspended materials) (Weiner et al. 2013, 2014). The collective population risk is a measure of the total radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing the various alternatives.

The Risks and Consequences of Radioactive Material Transport (RISKIND) computer code was used to estimate the doses to MEIs and populations for the worst-case maximum reasonably foreseeable transportation accident (Yuan et al. 1995). The RISKIND computer code was developed for the DOE Office of Civilian Radioactive Waste Management to estimate potential radiological consequences and health risks to individuals and the collective population from exposures associated with the transportation of spent nuclear fuel. This code is also applicable to the transportation of other cargo types, as the code can model complex atmospheric dispersion and estimate radiation doses to MEIs near the accident. Use of the RISKIND computer code as implemented in this VTR EIS is consistent with direction provided in *A Resource Handbook on DOE Transportation Risk Assessment* (DOE 2002b).

RISKIND calculations were conducted to supplement the collective risk results calculated with RADTRAN 6. Whereas the collective risk results provide a measure of the overall risks of each alternative, the RISKIND calculations are meant to address areas of specific concern to individuals and population subgroups. Essentially, the RISKIND analyses address "What if" questions, such as "What if I live next to a site access road?" or "What if an accident happens near my town?"

E.5.1 Transportation Routes

To assess incident-free and transportation accident impacts, route characteristics were determined for the following offsite shipments that could occur as part of routine operations:

- Plutonium materials (weapons-grade in metal or oxide form) from either Los Alamos National Laboratory (LANL) to the Savannah River Site (SRS) or the Idaho National Laboratory (INL) or from SRS to INL;[2]
- Plutonium materials (reactor-grade in oxide form) from Europe (France, United Kingdom, or both) through Joint Base Charleston-Weapons Station in South Carolina to SRS or INL;
- Transuranic (TRU) waste (both contact-handled [CH] and remote-handled [RH]) from SRS, INL, and Oak Ridge National Laboratory (ORNL), as applicable, to the Waste Isolation Pilot Plant (WIPP) in New Mexico;
- Unirradiated VTR fuel assemblies from SRS to INL or ORNL, or from INL to ORNL;
- Low-level and mixed low-level radioactive wastes from SRS, INL, and ORNL to offsite Federal or commercial disposal facilities. For purposes of analysis in this EIS, the disposal site was assumed to be the Nevada National Security Site (NNSS) near Las Vegas, Nevada; Energy*Solutions* near Clive, Utah; or Waste Control Specialists, near Andrews, Texas;
- Low-enriched uranium (LEU) (5-percent) from a commercial fuel fabrication facility (e.g., Nuclear Fuel Services, Inc. (NFS), in Erwin, Tennessee) to SRS or INL;
- Adulterant from a commercial vendor from an assumed distance of 3,000 miles or from diluent from a DOE site to INL or SRS, for dilution of plutonium wastes in critically controlled overpacks for transport to the WIPP facility;
- Construction materials shipped to SRS, INL, or ORNL (nonradiological impacts only);
- Hazardous waste from SRS, INL, and ORNL to an offsite treatment, storage, and disposal facility (nonradiological impacts only).

These sites constitute the locations where the majority of shipments would be transported.

For offsite transport, highway routes were determined using the routing program WebTRAGIS (Peterson 2018). WebTRAGIS is a geographic information system-based transportation analysis computer program used to identify the highway, rail, and waterway routes for transporting radioactive materials within the United States that were used in the analysis. Both the road and rail network are 1:100,000-scale databases, which were developed from the U.S. Geological Survey digital line graphs and the U.S. Bureau

[2] The weapons-grade plutonium would be available from LANL or SRS after pit disassembly at either site. The impacts of transporting surplus pit to either site were evaluated in the SPD SEIS (DOE 2015a).

of the Census Topological Integrated Geographic Encoding and Referencing System. The features in WebTRAGIS allow users to determine routes for shipment of radioactive materials that conform to DOT regulations as specified in 49 CFR Part 397. The population densities along each route were derived from 2010 Census Bureau data (Peterson 2018). State-level U.S. Census data for 2010 (Census 2010) was used in relation to the 2000 census data to project the population densities to 2050 levels.

Offsite Route Characteristics

Route characteristics that are important to the radiological risk assessment include the total shipment distance and population distribution along the route. The specific route selected determines both the total potentially exposed population and the expected frequency of transportation-related accidents. Route characteristics for routes analyzed in this VTR *EIS* are summarized in **Table E–1**. Rural, suburban, and urban areas are characterized according to the following breakdown (Peterson 2018):

- Rural population densities range from 0 to 54 persons per square kilometer (0 to 140 persons per square mile)
- Suburban population densities range from 55 to 1,284 persons per square kilometer (140 to 3,326 persons per square mile)
- Urban population densities include all population densities greater than 1,284 persons per square kilometer (3,326 persons per square mile)

The affected population for route characterization and incident-free dose calculation includes all persons living within 0.5 miles of each side of the transportation route.

Analyzed truck routes for offsite shipments of radioactive materials and wastes for the INL VTR Alternative and ORNL VTR Alternative are shown in **Figure E–2** and **Figure E–3**, respectively. **Figure E–4** shows additional routes that are common to both alternatives.

Table E–1. Offsite Transport Truck Route Characteristics

Origin	Destination	Nominal Distance (kilometers)	Distance Traveled in Zones (kilometers)			Population Density in Zone [a] (number per square kilometer)			Number of Affected Persons [b]
			Rural	Suburban	Urban	Rural	Suburban	Urban	
INL	NNSS	1,330	1,178	129	22	15	951	3,608	**354,070**
INL	ORNL [c]	3,320	2,624	639	57	21	626	2,342	**944,151**
SRS	INL [d]	3,753	2,809	838	107	23	712	2,806	**1,534,658**
INL	WIPP	2,285	1,935	297	54	21	769	3,551	**733,501**
NFS [e]	INL	3,545	2,747	726	71	23	633	2,344	**1,101,435**
ORNL	NNSS	3,466	2,837	564	66	18	593	2,951	**929,802**
SRS	ORNL [c]	621	327	250	45	36	858	3,454	**609,287**
SRS	NNSS	3,890	3,105	760	115	20	682	3,161	**1,502,998**
ORNL	WIPP	2,082	1,527	502	54	25	722	2,888	**887,811**
NFS [e]	SRS	488	287	192	9	35	549	2,525	**220,027**
LANL	SRS	2,722	1,980	652	90	25	655	3,119	**1,211,384**
LANL	INL	1,895	1,519	322	54	26	743	3,551	**751,812**
SRS	WIPP	2,307	1,596	681	30	25	587	2,745	**836,719**
JWS	SRS	222	145	67	10	17	948	3,034	**154,511**
INL	Energy*Solutions*	511	381	108	22	27	992	3,608	**317,354**
INL	WCS	2,365	2,007	303	55	20	772	3,521	**748,407**

Origin	Destination	Nominal Distance (kilometers)	Distance Traveled in Zones (kilometers)			Population Density in Zone [a] (number per square kilometer)			Number of Affected Persons [b]
			Rural	**Suburban**	**Urban**	**Rural**	**Suburban**	**Urban**	
ORNL	Energy*Solutions*	3,145	2,458	615	73	21	668	2,704	**1,054,278**
ORNL	WCS	1,963	1,415	496	52	26	719	2,903	**872,653**
SRS	Energy*Solutions*	3,572	2,636	814	122	22	747	2,962	**1,644,834**
SRS	WCS	2,182	1,478	675	29	27	584	2,764	**821,614**
DOE site 1	INL	3,387	2,674	647	66	21	594	2,483	**966,123**
DOE site 1	SRS	947	489	435	24	31	637	2,682	**568,421**
DOE site 2	INL	2,864	2,303	511	51	20	611	2,351	**767,812**
DOE site 2	SRS	930	528	347	55	32	838	3,226	**777,857**

INL = Idaho National Laboratory, JWS = Joint Base Charleston-Weapon Station; LANL = Los Alamos National Laboratory; NFS = Nuclear Fuel Services, Inc.; NNSS = Nevada National Security Site; OH = Ohio; SRS = Savannah River Site; WCS = Waste Control Specialists; WIPP = Waste Isolation Pilot Plant.

[a] Population densities have been projected to 2050 using State-level data from the 2010 census (Census 2010) and assuming State population growth rates from 2000 to 2010 continue to 2050.

[b] For offsite shipments, the estimated number of persons residing within 800 meters (0.5 miles) along the transportation route, projected to 2050.

[c] Shipments of VTR fuel assemblies would be from SRS or INL to ORNL, if VTR is at ORNL

[d] Shipments of plutonium materials would be made from SRS or LANL to INL, or from LANL to SRS, depending on the options for feedstock preparation and fuel production facilities (e.g., at INL or at SRS).

[e] Shipment of 5-percent enriched uranium metal is assumed to be from Nuclear Fuel Services, Inc., in Erwin, Tennessee.

Note: To convert from kilometers to miles, multiply by 0.6214; to convert from number per square kilometer to number per square mile, multiply by 2.59. Rounded to nearest kilometer.

Figure E–2. Analyzed National and Regional Truck Routes for the INL VTR Alternative

Figure E–3. Analyzed National and Regional Truck Routes for the ORNL VTR Alternative

Figure E–4. Additional Routes that are Common to Both Alternatives

E.5.2 Radioactive Material Shipments

Transportation of all material and waste types is assumed to occur in certified or certified-equivalent packaging on dedicated-use vehicles. Use of legal-weight, heavy combination trucks is assumed in this appendix for highway transportation. Type A packages are transported on common flatbed or covered trailers. Type B packages are generally shipped on trailers designed specifically for the packaging being used. For transportation by truck, the maximum payload weight is considered to be about 48,000 pounds, based on the Federal gross vehicle weight limit of 80,000 pounds (23 CFR 658.17). While there are large numbers of multi-trailer combinations (known as longer combination vehicles) with gross weights in excess of the Federal limit in operation on rural roads and turnpikes in some States (DOT 2000), for evaluation purposes, the load limit for the legal truck was based on the Federal gross vehicle weight. The width restriction is about 8.5 feet (23 CFR 658.15). Length restrictions vary by State, but are assumed for purposes of analysis to be no more than 48 feet.

Several types of containers would be used to transport radioactive materials and waste. The various wastes that would be transported under the alternatives in this VTR EIS include low-level and mixed low-level radioactive waste, CH-TRU waste, demolition and construction debris, and hazardous waste. **Table E–2** lists the types of containers assumed for the analysis along with their volumes and the number of containers in a shipment. A shipment is defined as the amount of waste transported on a single truck.

In general, the number of shipping containers per shipment was estimated on the basis of the dimensions and weight of the shipping containers, the Transport Index,[3] which is the dose rate at 3.3 feet from the container, and the transport vehicle dimensions and weight limits. The various materials and wastes were assumed to be transported on standard truck semi-trailers in a single stack.

Special nuclear material would be transported using STAs. Special nuclear material transports include plutonium in the form of metal or oxides, enriched uranium, and VTR fuel. These shipments would occur to support production of VTR fuel fabrication and its transport to the VTR site. The numbers of shipments associated with the transport of plutonium, and uranium (low- or high-enriched) were determined using up-to-date information regarding the types of transport packages to be used and forecasted VTR assembly needs. These materials would be transported in Type B packages. While it is assumed that a specific Type B package would be used for each type of nuclear material being transported, more than one particular package design could be used. Use of different Type B packages that are applicable to a particular cargo would not significantly change the impacts presented in this analysis because the designs and shipping configurations of the Type B packages are similar. For unirradiated VTR fuel, the number of shipments is based on three assemblies per transport package, one transport package per shipment (INL 2020c).

For the LEU (5-percent enriched), the quantities required for the VTR are assumed to be transported in ES 3100 packages using STAs. If LEU metal is used, then, the required materials are assumed to be shipped from a fabrication facility in Erwin, Tennessee (NFS) to SRS or INL.

[3] The Transport Index is a dimensionless number (rounded up to the next tenth) placed on label of a package, to designate the degree of control to be exercised by the carrier. Its value is equivalent to the maximum radiation level in millirem per hour at 3.3 feet from the package (10 CFR 71.4 and 49 CFR 173.403).

Table E–2. Material or Waste Type and Associated Container Characteristics [a]

Material or Waste Type	***Container***	***Container Volume (cubic meters)*** [b]	***Container Mass (kilograms)*** [c]	***Shipment Description***
MLLW	55-gallon drum	0.2	399	80 drums per truck
LLW	B-25 box	2.55	4,536	5 boxes per truck
CH-TRU waste	55-gallon drum	0.2	142 [d]	14 drums per TRUPACT-II; 3 TRUPACT-IIs per truck
CH-TRU waste	Pipe overpack container [e]	0.2	142 [d]	14 containers per TRUPACT-II; 3 TRUPACT-IIs per truck
Special nuclear material	Type B package	0.13 to 0.30	183-318	1 to 30 packages per STA
Unirradiated VTR fuel	Type B package [f]	9.3	6,350	1 transport cask per STA
TRU waste associated with the diluted processed plutonium	Criticality control container [g]	0.2	142	14 containers per TRUPACT-II
RH-TRU wastes	55-gallon drum	0.2	399	3 drums per RH-72B cask, 1 cask per truck
RH-LLW/MLLW	55-gallon drum	0.2	399	10 drums per CNS 10-160B cask, 1 cask per truck
LLW/MLLW	B-25 box	2.92	3,630	5 boxes per truck
LLW/MLLW	B-12 box	1.46	3,630	5 boxes per truck
LLW/MLLW	16-foot container	29	Not applicable	1 container per truck
Diluent	Type A package	4.04	13,800	1 cylinder per truck
Construction/demolition debris	Roll-on/Roll-off dumpster	15.30	Not applicable	1 load per truck
Hazardous waste	55-gallon drum	0.2	399	40 drums per truck

CH-TRU = contact-handled transuranic; CNS = Chem-Nuclear Systems, Inc.; LLW = low-level radioactive waste; MLLW = mixed low-level radioactive waste; RH = Remote-handled; STA = Secure Transportation Asset; TRU = transuranic; TRUPACT-II = Transuranic Package Transporter Model 2.

[a] Containers and transport packages identified in this table were used to determine the transportation impacts for purposes of analysis. Specific Type B packages, while not identified in this table, were assumed for specific material or waste types to conduct the analysis. Other containers and transportation packages may be used in addition to, or in lieu of, those shown.

[b] Container exterior volume. To convert from cubic meters to cubic feet, multiply by 35.315; from liters to gallons, by 0.26417.

[c] Filled container maximum mass. Container mass includes the mass of the container shell, its internal packaging, and the materials within. To convert from kilograms to pounds, multiply by 2.2046.

[d] For the 14 drums per TRUPACT-II and three TRUPACT-II per shipment, the average weight of the drum is limited to 142 kilograms.

[e] TRU waste consisting of plutonium would be packaged in pipe overpack containers (POCs), which would be the same size as a 55-gallon drum.

[f] Packages for transporting VTR fuel assemblies are assumed to be the Hanford Unirradiated Fuel Package.

[g] Diluted processed plutonium oxide would be packaged in the criticality control containers, which would be the same size as a 55-gallon drum.

The quantities of the uranium and plutonium source material (e.g., feedstock) needs are based on the VTR fuel design specifications as discussed in the *VTR Fuel Facility Plan* (INL 2019a), the *Conceptual Design Report for the Versatile Test Reactor* (INL 2019b), and the SRS Data Call Response (SRNS 2020). Essentially, the VTR operation requires 45 fresh fuel assemblies per year. Depending on the type of fuel (e.g., a clean weapons-grade fuel with low impurities or the more common plutonium materials with impurities, (especially the in-growth americium-241), different fabrication or pre-fabrication processing would be needed to produce plutonium feed materials that meet the VTR fuel specification needs (e.g., americium-241 content of less than 1 percent). Given the processing efficiency (SRNL 2020), between 460 to 550 kilograms of plutonium and 1,610 to 1,920 kilograms LEU per year would be needed for feedstock.

For radioactive waste to be transported to a radioactive waste disposal site, it is assumed that the wastes would meet the disposal facility's waste acceptance criteria. For purposes of analysis, it is assumed that all of the low-level radioactive waste generated at INL, ORNL, or SRS would be transported to NNSS, Energy*Solutions*, or WCS for disposal.

TRU waste would be transported to WIPP for disposal. TRU waste would consist of secondary waste resulting from VTR fuel production (plutonium preparation and fuel fabrication) activities and treatment of spent nuclear fuel. These materials could be packaged in drums, pipe overpack containers (POCs), or criticality control overpacks (CCOs). Use of CCOs for disposal of plutonium materials allows a higher concentration, thereby reducing the number of shipments and disposal volume.

Radionuclide inventories are used to determine accident risks associated with a release of the radioactive or contaminated cargo. **Table E–3** provides the container radionuclide inventory concentration assumed for low-level and mixed low-level radioactive waste. It is assumed that these two waste types would have the same radioisotopic composition with the mixed low-level radioactive waste having a hazardous component. The list of radionuclides in the table is limited to those that would be expected from the plutonium wastes during the fuel fabrication and spent fuel treatment activities. The composition of the waste is the average curie concentration per radioisotope as measured in the year 2010. This composition is assumed to be representative of the low-level and mixed low-level radioactive waste streams generated by plutonium processing and disposition activities (DOE 2015a).

Table E–3. Low-level and Mixed Low-level Radioactive Waste Radionuclide Concentrations [a] from Fuel Fabrication

Nuclide	*Curies per Cubic Meter*
Americium-241	0.000050
Plutonium-238	0.00038
Plutonium-239	0.00011
Plutonium-240	0.000049
Plutonium-241	0.00048
Technetium-99	0.0000052

[a] These isotopes are the primary isotopes to be expected in offsite shipments of low-level and mixed low-level radioactive waste. The concentrations are representative of what historically has been generated at SRS.

Source: DOE 2015a; SRNS 2012.

The various wastes that would be generated from the VTR operation, and its support facilities, including the post-irradiation examination operations, are estimated in *Versatile Test Reactor Wastes and Material Data for Environmental Impact Statement* (INL 2020b). This INL report provides the estimated volumes of different wastes from each facility operation, along with the expected radionuclide inventories for each type of waste from each facility. This compilation of waste data would lead to about 20 different waste-radionuclide combinations. For the purposes of this VTR EIS, the analysis in this appendix assigns a set of radionuclides to each waste type, regardless of its origin. This action reduces the waste-radionuclide combinations to four categories: CH- and RH-low-level; mixed low-level wastes; CH- and RH-TRU; and mixed TRU wastes. The selected lists of radionuclides are based on information in the INL report, in which the transportation accident would lead to a maximum population dose for each selected waste type.

The various wastes from the VTR and its support facility operations are assumed to be packaged for transportation to an offsite disposal facility by considering these four factors:

1. CH-low-level and mixed low-level wastes are packaged in B-12 boxes (20 percent), B-25 boxes (20 percent), and 16-foot ISO containers (60 percent), for transport to a disposal facility.

2. RH-low-level and mixed low-level wastes are packaged in 55–gallon drums and placed in a Type B shielded casks for transport to a disposal facility; CNS 10-160B (COC-71-9204 2020) was used a representative transport package.
3. RH-TRU and mixed TRU wastes are packaged in 55-gallon drums and placed in Type B shielded casks for transport to WIPP; RH-72B (COC-71-9212 2019) was used as a representative transport package.
4. CH-TRU and mixed TRU wastes are packaged in 55-gallon drums and placed in TRUPACT-II for transport to WIPP.

Given the feedstock preparation and VTR fuel production efficiency (SRNL 2020), the VTR operation would require up to 34 metric tons of plutonium feedstock materials.[4] The U.S. excess plutonium inventory of more than 50 metric tons would be sufficient to meet fueling needs for the VTR lifetime operation of 60 years. This inventory includes metallic, weapons-grade plutonium managed by the National Nuclear Security Administration, as well as non-weapons-grade material and material in different physical forms. Therefore, the sources for needed plutonium could range from domestic surplus U.S. weapons-grade and non-weapons-grade forms to optional reactor-grade material procured from Europe (France, United Kingdom, or both).

For transport of the weapons-grade plutonium from LANL, the reactor-grade plutonium from Europe, and the LEU from NFS to the VTR fuel production facilities, it was assumed that the contents of one Type B package would be released in the event of an accident.

Table E–4 shows the number of curies per transport package assumed for the new (unirradiated) VTR fuel assembly. For transport of the new fuel assemblies, it was assumed that the Hanford Unirradiated Fuel Package would be used (INL 2020c). This package was constructed for transporting Fast Flux Test Facility (FFTF) and Experimental Breeder fuel with an assembly length up to 12 feet (CH2MHILL 2009). The use of this package would require some reassembly of non-fuel part of the VTR fuel, because of the overall assembly length differences between the VTR and FFTF fuel.

Table E–4. Radioisotopic Content of Transport Packages Containing New VTR Fuel Assemblies

Radioisotope	VTR Fuel Assemblies Curies per Package [a]		Radioisotope	VTR Fuel Assemblies Curies per Package [a]
	Weapons-Grade Plutonium	Reactor-Grade Plutonium		
Americium-241	3.20	36.9	Uranium-232	0.00391
Plutonium-238	227	9,540	Uranium-234	1.61
Plutonium-239	1,530	1,030	Uranium-235	0.010
Plutonium-240	362	1,510	Uranium-236	0.150
Plutonium-241	27,400	110,000	Uranium-238	0.0289
Plutonium-242	0.104	6.89		

[a] Each package is assumed to contain three VTR fuel assemblies.

For the disposition of the plutonium wastes from fabrication without the need for feedstock preparation (e.g., use of cleaner weapons-grade plutonium feed), the waste would be oxidized and repackaged and sent to WIPP for disposal. For purposes of analysis, it was assumed there would be 150 grams of plutonium per POC and 300 grams of plutonium per CCO. A shipment would consist of three TRUPACT-II [Transuranic Package Transporter Model 2] packages, each containing 13 containers. [The selection of 13 containers per TRUPACT-II is based on the uncertainty of the total mass limit of the drums within the package. This will lead to a slightly larger number of shipments.]

[4] This is an upper estimate based on the fuel production efficiency of about 73 percent for fabrication without feedstock preparation. As the production efficiency improves, the need for the feedstock plutonium could be reduced.

If the plutonium feed requires pre-processing for the removal of impurities prior to fuel fabrication, three potential cases are considered (SRNL 2020):

1. Case 1 Aqueous processing
2. Case 2 Pyro-chemical processing with aqueous processing
3. Case 3 Pyro-chemical processing

The generated wastes in Cases 1 and 3 envelope the range of potential waste values for disposition. It was considered that Case 1 would generate cemented drums of americium-plutonium content limited to a total of 80 curies (minus the uncertainty, which was assumed to be 13 percent) per drum, whereas Case 3 would generate metal drums of americium-plutonium content limited to 80 curies (minus the uncertainty, which was assumed to be 22 percent) per drum (SRNL-2020). For the transport to WIPP, because of the limitations on container loads, it was considered that there would be 12 cemented americium-plutonium waste containers per shipment, and 28 metal americium-plutonium waste containers per shipment.

For the secondary TRU waste generated from processing of weapons-grade plutonium, it was assumed there would be 20 grams of plutonium per drum. For TRU waste generated from processing non-weapons-grade plutonium, it was assumed there would be 10 grams of plutonium per drum. A shipment of TRU waste would consist of three TRUPACT-II packages.

The feedstock (plutonium and uranium) could be in the form of metal, powder, or both. The European plutonium is in oxide powder form. There is also a domestic weapons-grade plutonium that is in oxide form. Therefore, for analysis purposes and to conservatively envelop the risk of transporting plutonium and uranium source materials to the VTR fuel production facility option locations, it was assumed that these source materials (e.g., feedstock) would be in oxide form (e.g., powder) to maximize the accident risks. In addition, the impact analysis is based on the weapons-grade (lowest risk) and European (highest risk) plutonium materials, as these provide an enveloping risk for all other potential domestic plutonium that could be transported between the affected sites.

E.6 Incident-free Transportation Risks

E.6.1 Radiological Risk

During incident-free transportation of radioactive materials, a radiological dose results from exposure to the external radiation field that surrounds the shipping containers. The population dose is a function of the number of people exposed, their proximity to the containers, their length of time of exposure, and the intensity of the radiation field surrounding the containers.

Radiological impacts were determined for crew members and the general population during incident-free transportation. For truck shipments, the crew members are the drivers of the shipment vehicle. The general population is composed of the persons residing within 0.50 miles of the truck route (off-link), persons sharing the road (on-link), and persons at stops. Exposures to workers who would load and unload the shipments are not included in this analysis, but are included in the occupational estimates for plant workers (see Chapter 4, Section 4.10 of the VTR EIS). Exposures to inspectors are evaluated and presented separately in this appendix.

Collective doses for the crew and general population were calculated by using the RADTRAN 6 computer code (Weiner et al. 2013, 2014). The radioactive material shipments were assigned an external dose rate based on their radiological characteristics. Offsite transportation of the radioactive material has a defined regulatory limit of 10 millirem per hour at about 6.6 feet from the outer lateral surfaces of the vehicle (10 CFR 71.47 and 49 CFR 173.441). If a waste container shows a high external dose rate that could exceed this limit, it is categorized as an exclusive use shipment with further transport and dose rate limitations as defined in these regulations, and the cargo would be transported in a shielded Type A or Type B shipping container. The waste container dose rate at 3.3 feet from its surface, or its Transport Index, is dependent on the distribution and quantities of radionuclides, waste density, shielding provided by the packaging, and self-shielding provided by the waste mixture.

Dose rates for packages containing CH- and RH-low-level and mixed low-level radioactive waste were assigned a dose rate of 2 and 10 millirem per hour at 3.3 feet, and the LEU was assigned a dose rate of 2 millirem per hour at 3.3 feet. The dose rate for packages containing unirradiated VTR fuel is assumed to be 1 millirem per hour at 3.3 feet from the transport vehicle. For the plutonium oxide, the dose rate is assumed to be 5 millirem per hour at 3.3 feet from the transport vehicle. A dose rate of 1 millirem per hour at 3.3 feet was assigned to packages containing diluent. The dose rates for CH-TRU and RH-TRU waste were assumed to be 4 and 7 millirem per hour at 3.3 feet, respectively (DOE 1997). In all cases, the maximum external dose rate would be less than or equal to the regulatory limit of 10 millirem per hour at 6.6 feet from each container.

To calculate the collective dose, a unit risk factor was developed to estimate the impact of transporting one shipment of radioactive material over a unit distance of travel in a given population density zone. The unit risk factors were combined with routing information, such as the shipment distances in various population density zones, to determine the risk for a single shipment (a shipment risk factor) between a given origin and destination. Unit risk factors were developed on the basis of travel on interstate highways and freeways (49 CFR Parts 171 to 178 requires use of these roadways for highway-route-controlled quantities of radioactive material) within rural, suburban, and urban population zones by using RADTRAN 6 and its default data. In addition, it was assumed for the analysis that, for 10 percent of the time, travel through suburban and urban zones would encounter rush-hour conditions, leading to lower average speed and higher traffic density.

The radiological risks from transporting the waste are estimated in terms of the number of LCFs among the crew and the exposed population. A health risk conversion factor of 0.0006 LCFs per rem or person-rem of exposure is used for both the public and workers (DOE 2003b).

E.6.2 Nonradiological Risk

Nonradiological risks, or vehicle-related health risks, resulting from incident-free transport may be associated with the generation of air pollutants by transport vehicles during shipment and are independent of the radioactive nature of the shipment. The health risk associated with these emissions under incident-free transport conditions is the excess latent mortality due to inhalation of vehicle emissions. Unit risk factors for pollutant inhalation in terms of mortality have been developed, as described in *A Resource Handbook on DOE Transportation Risk Assessment* (DOE 2002b). This analysis was not performed for this *EIS* because the results cannot be placed into context by comparison with a standard or measured data. The amounts of vehicle emissions are estimated for each alternative in Chapter 4, Section 4.4.

E.6.3 Maximally Exposed Individual Exposure Scenarios

The maximum individual doses for routine offsite transportation were estimated for transportation workers, as well as for members of the general population.

For truck shipments, three hypothetical scenarios were evaluated to determine the MEI in the general population. These scenarios are as follows (DOE 2002a):

- A person caught in traffic and located 4 feet from the surface of the shipping container for 30 minutes

- A resident living 98 feet from the highway used to transport the shipping container
- A service station worker at a distance of 52 feet from the shipping container for 50 minutes

The hypothetical MEI doses were accumulated over a single year for all transportation shipments. However, for the scenario involving an individual caught in traffic next to a shipping container, the radiological exposures were calculated for only one event because it was considered unlikely that the same individual would be caught in traffic next to all containers for all shipments. For truck shipments, the maximally exposed transportation worker would be a truck crew member who could be a DOE employee or a driver for a commercial carrier. In addition to following DOT requirements, a DOE employee would also need to comply with DOE regulations in 10 CFR Part 835 ("Occupational Radiation Protection") which limits worker radiation doses to 5 rem per year. However, DOE's goal is to maintain radiological exposures as low as reasonably achievable. DOE has, therefore, established the administrative control level of 2 rem per year per person (DOE 2017a). This limit would apply to any non-TRU waste shipment conducted by DOE personnel. Drivers of TRU waste shipments to WIPP have an administrative control level of 1 rem per year (WIPP 2006). Commercial drivers are subject to Occupational Safety and Health Administration regulations, which limits the whole body dose to 5 rem per year (29 CFR 1910.1996(b)), and the DOT requirement of 2 millirem per hour in the truck cab (49 CFR 173.411). Commercial drivers typically do not transport radioactive materials that have high dose rates external to the package. Therefore, for purposes of analysis, a maximally exposed driver would not be expected to exceed the DOE administrative control level of 2 rem per year for non-TRU waste shipments. Other workers include inspectors who would inspect the truck and its cargo along the route. One inspector was assumed to be at a distance of 3.3 feet from the cargo for a duration of 1 hour.

E.7 Transportation Accident Risks

E.7.1 Methodology

The offsite transportation accident analysis considers the impact of accidents during the transportation of materials. Under accident conditions, impacts on human health and the environment could result from the release and dispersal of radioactive material. Transportation accident impacts were assessed using an accident analysis methodology developed by NRC. This section provides an overview of the methodologies. Detailed descriptions of various methodologies are found in the *Radioactive Material Transportation Study,* NUREG-0170, *Modal Study*, NUREG/CR-4829, and *Reexamination Study,* NUREG/CR-6672 (NRC 1977, 1987, 2000). Accidents that could potentially breach the shipping container are represented by a spectrum of accident severities and radioactive release conditions. Historically, most transportation accidents involving radioactive materials have resulted in little or no release of radioactive material from the shipping container. Consequently, the analysis of accident risks takes into account a spectrum of accidents ranging from high-probability accidents of low severity to hypothetical high-severity accidents that have a correspondingly low probability of occurrence. The accident analysis calculates the probabilities and consequences from this spectrum of accidents.

To provide DOE and the public with a reasonable assessment of radioactive waste transportation accident impacts, two types of analysis were performed. First, an accident risk assessment was performed that takes into account the probabilities and consequences of a spectrum of potential accident severities using a methodology developed by NRC (NRC 1977, 1987, 2000). For the spectrum of accidents considered in the analysis, accident consequences in terms of collective "dose risk" to the population within 50 miles were determined using the RADTRAN 6 computer program (Weiner et al. 2013, 2014). The RADTRAN 6 code sums the product of consequences and probability over all accident severity categories to obtain a probability-weighted risk value referred to in this appendix as "dose risk," which is expressed in units of person-rem. Second, to represent the maximum reasonably foreseeable impacts on individuals and populations should an accident occur, maximum radiological consequences were calculated in an urban or suburban population zone for an accidental release with a likelihood of occurrence greater than 1-in-10 million per year using the RISKIND computer program (Yuan et al. 1995).

For accidents where a waste container or the cask shielding was undamaged, population and individual radiation exposure from the waste package was evaluated for the duration that would be needed to recover and resume shipment. The collective dose over all segments of transportation routes was evaluated for an affected population within a distance of 0.5 miles from the accident location. This dose is an external dose, and is inversely proportional to the square of the distance of the affected population from an accident. Any additional dose to those residing beyond 0.5 miles from the accident would be negligible. The dose to an individual (first responder) was calculated assuming that the individual would be located at 6.6 to 33 feet from the package.

E.7.2 Accident Rates

Whenever material is shipped, the possibility exists of a traffic accident that could result in vehicular damage, injury, or death. Even when drivers are trained in defensive driving techniques, there is a risk of traffic accidents. DOE and its predecessor agencies have a successful 50-year history of transporting radioactive materials.

To calculate accident risks, vehicle accident and fatality rates were taken from data provided in *State-Level Accident Rates for Surface Freight Transportation: A Reexamination*, ANL/ESD/TM-150 (Saricks and Tompkins 1999). Accident rates are generically defined as the number of accident involvements (or fatalities) in a given year per unit of travel in that same year. Therefore, the rate is a fractional value, with accident involvement count as the numerator of the fraction and vehicular activity (total travel distance in truck kilometers) as its denominator. Accident rates were generally determined for a multi-year period. For assessment purposes, the total number of expected accidents or fatalities was calculated by multiplying the total shipment distance for a specific case by the appropriate accident or fatality rate. No reduction in accident or fatality rates was assumed, even though radioactive material carrier drivers are better trained and have better maintained equipment than other truck drivers.

For truck transportation, the rates presented are specifically for heavy combination trucks involved in interstate commerce (Saricks and Tompkins 1999). Heavy combination trucks are rigs composed of a separable tractor unit containing the engine and one to three freight trailers connected to each other. Heavy combination trucks are typically used for radioactive material shipments. Truck accident rates were computed for each State based on statistics compiled by the Federal Highway Administration, Office of Motor Carriers, from 1994 to 1996. A fatality caused by an accident is the death of a member of the public who is killed instantly or dies within 30 days due to the injuries sustained in the accident.

For offsite transportation of radioactive materials and wastes, separate route-specific accident rates and accident fatality risks were used. The values selected were the total State-level accident and fatality rates provided in ANL/ESD/TM-150 (Saricks and Tompkins 1999). The State-level rates were adjusted based on the distance traveled in each State to derive a route-specific accident and fatality rate per car-kilometer.

Review of the truck accidents and fatalities reports by the Federal Carrier Safety Administration indicated that State-level accidents and fatalities were underreported (Blower and Matteson 2003). For the years 1994 through 1996, which formed the bases for the analysis in the Saricks and Tompkins report, the review identified that accidents were underreported by about 39 percent and fatalities were underreported by about 36 percent (UMTRI 2003). Therefore, State-level truck accident and fatality rates in the Saricks and Tompkins report were increased by factors of 1.64 and 1.57, respectively, to account for the underreporting.

For transport by STA, the DOE operational experience between 1975 and 1998 was used to determine an accident rate of 2.7×10^{-7} accident per kilometer (4.4×10^{-7} accident per mile) (DOE 2002a). The route-specific commercial truck accident rates were adjusted to reflect the STA accident rate. Accident fatalities for STAs were estimated using the commercial truck transport fatality per accident ratios within each zone.

E.7.3 Accident Severity Categories and Conditional Probabilities

Accident severity categories for potential radioactive waste transportation accidents are described in the *Radioactive Material Transportation Study* (NRC 1977) for radioactive waste in general, and the *Modal Study* (NRC 1987), and the *Reexamination Study* (NRC 2000) for used nuclear fuel. The methods described in the *Modal Study* and the *Reexamination Study* are applicable to transportation of radioactive materials in a Type B spent fuel cask. The accident severity categories presented in the *Radioactive Material Transportation Study* would be applicable to all other waste transported off site.

The *Radioactive Material Transportation Study* (NRC 1977) originally was used to estimate conditional probabilities associated with accidents involving transportation of radioactive materials. The *Modal Study* and the *Reexamination Study* (NRC 1987, 2000) were initiatives taken by NRC to refine more precisely the analysis presented in the *Radioactive Material Transportation Study* for used nuclear fuel shipment casks.

Whereas the *Radioactive Material Transportation Study* (NRC 1977) analysis was primarily performed using best engineering judgments and presumptions concerning cask response, the later studies rely on sophisticated structural and thermal engineering analysis and a probabilistic assessment of the conditions that could be experienced in severe transportation accidents. The latter results are based on representative used nuclear fuel casks assumed to have been designed, manufactured, operated, and maintained according to national codes and standards. Design parameters of the representative casks were chosen to meet the minimum test criteria specified in 10 CFR Part 71. The study is believed to provide realistic, yet conservative, results for radiological releases under transport accident conditions.

In the *Modal Study* and the *Reexamination Study*, potential accident damage to a cask is categorized according to the magnitude of the mechanical forces (impact) and thermal forces (fire) to which a cask may be subjected during an accident. Because all accidents can be described in these terms, severity is independent of the specific accident sequence. In other words, any sequence of events that results in an accident in which a cask is subjected to forces within a certain range of values is assigned to the accident severity region associated with that range. The accident severity scheme is designed to take into account all potential foreseeable transportation accidents, including accidents with low probabilities but high consequences, and those with high probabilities but low consequences.

As discussed earlier, the accident consequence assessment considers the potential impacts of severe transportation accidents. In terms of risk, the severity of an accident must be viewed in terms of potential radiological consequences, which are directly proportional to the fraction of the radioactive material within a cask that is released to the environment during the accident. Although accident severity regions span the entire range of mechanical and thermal accident loads, they are grouped into accident categories that can be characterized by a single set of release fractions and are, therefore, considered together in the accident consequence assessment. The accident category severity fraction is the sum of all conditional probabilities in that accident category.

For the accident risk assessment, accident "dose risk" was generically defined as the product of the consequences of an accident and the probability of occurrence of that accident, an approach consistent with the methodology used by RADTRAN 6 computer code. The RADTRAN 6 code sums the product of consequences and probabilities over all accident categories to obtain a probability-weighted risk value referred to in this appendix as "dose risk," which is expressed in units of person-rem.

E.7.4 Atmospheric Conditions

Because it is impossible to predict the specific location of an offsite transportation accident, generic atmospheric conditions were selected for the risk and consequence assessments. On the basis of observations from National Weather Service surface meteorological stations at over 177 locations in the United States, on an annual average, neutral conditions (Pasquill Stability Classes C and D) occur 58.5 percent of the time, and stable (Pasquill Stability Classes E, F, and G) and unstable (Pasquill Stability Classes A and B) conditions occur 33.5 percent and 8 percent of the time, respectively (DOE 2002a). The neutral weather conditions predominate in each season, but most frequently in the winter (nearly 60 percent of the observations).

Neutral weather conditions (Pasquill Stability Class D) compose the most frequently occurring atmospheric stability condition in the United States and are thus most likely to be present in the event of an accident involving a radioactive waste shipment. Neutral weather conditions are typified by moderate wind speeds, vertical mixing within the atmosphere, and good dispersion of atmospheric contaminants. Stable weather conditions are typified by low wind speeds, very little vertical mixing within the atmosphere, and poor dispersion of atmospheric contaminants. The atmospheric condition used in RADTRAN 6 is an average weather condition that corresponds to a stability class spread between Class D (for near distance) and Class E (for farther distance).

The accident consequences for the maximum reasonably foreseeable accident (an accident with a likelihood of occurrence greater than 1 in 10 million per year) were assessed for both stable (Class F with a wind speed of 3.3 feet per second) and neutral (Class D with a wind speed of 13 feet per second) atmospheric conditions. The population dose was evaluated under neutral atmospheric conditions and the MEI dose under stable atmospheric conditions. The MEI dose would represent an accident under weather conditions that result in a conservative dose (i.e., a stable weather condition, with minimum diffusion and dilution). The population dose would represent an average weather condition.

E.7.5 Radioactive Release Characteristics

Radiological consequences were calculated by assigning radionuclide release fractions on the basis of the type of waste, the type of shipping container, and the accident severity category. The release fraction is defined as the fraction of the radioactivity in the container that could be released to the atmosphere in a given severity of accident. Release fractions vary according to the waste type and the physical or chemical properties of the radioisotopes. Most solid radionuclides are nonvolatile and are, therefore, relatively non-dispersible.

Representative release fractions were developed for each waste and container type on the basis of DOE and NRC reports (DOE 1994, 2002b, 2003a; NRC 1977, 2000, 2005). The severity categories and corresponding release fractions provided in these documents cover a range of accidents from no impact (zero speed) to impacts with speed in excess of 120 miles per hour onto an unyielding surface. Traffic accidents that could occur at the facility would be of minor impact due to lower local speed, with no release potential.

For radioactive wastes transported in a Type B cask, the particulate release fractions were developed consistent with the models in the *Reexamination Study* (NRC 2000) and adapted in the *Final West Valley Demonstration Project Waste Management Environmental Impact Statement* (DOE 2003a). For wastes transported in Type A containers (e.g., 55-gallon drums and boxes), the fractions of radioactive material released from the shipping container were based on recommended values from the *Radioactive Material Transportation Study* and DOE Handbook on *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facility* (NRC 1977, DOE 1994). For CH-TRU and RH-TRU waste, the release fractions corresponding to the *Radioactive Material Transportation Study* severity categories as adapted in the *Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement* were used (DOE 1997).

For those accidents where the waste container or cask shielding are undamaged and no radioactive material is released, it is assumed that it would take 12 hours to recover from the accident and resume shipment for commercial shipments, and 6 hours for STA shipments. During this period, no individual would remain close to the cask. A first responder is assumed to stay 6.6 to 33 feet from the package for 1 hour (DOE 2002b).

E.7.6 Acts of Sabotage or Terrorism

In the aftermath of the tragic events of September 11, 2001, DOE is continuing to assess measures to minimize the risk or potential consequences of radiological sabotage. While it is not possible to determine terrorists' motives and targets with certainty, DOE considers the threat of terrorist attack to be real, and makes all efforts to reduce any vulnerability to this threat.

Nevertheless, DOE has evaluated the impacts of acts of sabotage and terrorism on transportation of used nuclear fuel and high-level radioactive waste shipments (DOE 1996, 2002a). The sabotage event evaluated in the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada (Yucca Mountain EIS)* was considered as the enveloping analysis for this VTR EIS. The event was assumed to involve either a truck or rail cask containing light water reactor used nuclear fuel. The consequences of such an act were calculated to result in an MEI dose (at 460 feet) of 40 to 110 rem for events involving a rail- or truck-sized cask, respectively (DOE 2002a). DOE's reassessment of the potential releases in a sabotage event in the *Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE 2008b) concluded that the consequence of a sabotage event in the *Yucca Mountain EIS* could be overstated by a factor of between 2.5 and 12. Considering a minimum factor of 2 overestimation in the calculated MEI doses, and the fact that any individual dose above 20 rem would lead to a factor of 2 increase in the dose risk conversion factor of 0.0006 LCF per rem, the *Yucca Mountain EIS* MEI dose of 40 to 110 rem would lead to an increase in risk of fatal cancer to the MEI by 2 to 7 percent. The quantity of radioactive materials transported under all alternatives considered in this VTR *EIS* would be less than that considered in the *Yucca Mountain EIS* analysis. Therefore, estimates of risk in the *Yucca Mountain EIS* envelop the risks from an act of sabotage or terrorism involving the radioactive material transported under all alternatives considered in this VTR *EIS*.

E.8 Risk Analysis Results

Per-shipment risk factors have been calculated for the collective populations of exposed persons and for the crew for all anticipated routes and shipment configurations. Radiological risks are presented in doses per shipment for each unique route, material, and container combination. Radiological risk factors per shipment for incident-free transportation and accident conditions are presented in **Table E–5**. These factors have been adjusted to reflect the projected population in 2050. For incident-free transportation, both dose and LCF risk factors are provided for the crew and exposed population. The radiological risks would result from potential exposure of people to external radiation emanating from the packaged waste. The exposed population includes the off-link public (people living along the route), on-link public (pedestrian and car occupants along the route), and public at rest and fuel stops. LCF risk factors were calculated by multiplying the accident dose risks by a health risk conversion factor of 0.0006 cancer fatalities per person-rem of exposure (DOE 2003b).

Table E–5. Risk Factors per Shipment of Radioactive Material and Waste

			Incident-Free				*Accident*	
Material or Wastes	***Origin***	***Transport Destination***	***Crew Dose (person-rem)***	***Crew Risk (LCF)***	***Population Dose (person-rem)***	***Population Risk (LCF)***	***Radiological Risk (LCF)***	***Non-radiological Risk (traffic fatalities)***
Plutonium [a, b]	LANL	SRS	0.034	$2.0{\times}10^{-5}$	0.12	$7.4{\times}10^{-5}$	$1.0{\times}10^{-7}$	0.000075
TRU waste (primary) in POCs containing WG plutonium material [c]	SRS	WIPP	0.089	$5.3{\times}10^{-5}$	0.075	$4.5{\times}10^{-5}$	$1.3{\times}10^{-8}$	0.00014
TRU waste (secondary) with 20 grams WG per drum [d]	SRS	WIPP	0.089	$5.3{\times}10^{-5}$	0.075	$4.5{\times}10^{-5}$	$2.3{\times}10^{-8}$	0.00014
TRU waste (primary) in CCOs containing WG plutonium material [e]	SRS	WIPP	0.089	$5.3{\times}10^{-5}$	0.075	$4.5{\times}10^{-5}$	$2.4{\times}10^{-8}$	0.00014
TRU (Am-241 in POCs) [c]	SRS	WIPP	0.089	$5.3{\times}10^{-5}$	0.075	$4.5{\times}10^{-5}$	$1.1{\times}10^{-7}$	0.00014
Plutonium [a, b]	LANL	INL	0.023	$1.4{\times}10^{-5}$	0.083	$5.0{\times}10^{-5}$	$7.2{\times}10^{-8}$	0.000033
Plutonium [a, b]	SRS	INL	0.047	$2.8{\times}10^{-6}$	0.17	$1.0{\times}10^{-4}$	$1.4{\times}10^{-7}$	0.000084
TRU waste (primary) in POCs containing WG plutonium material [c]	INL	WIPP	0.087	$5.2{\times}10^{-5}$	0.072	$4.3{\times}10^{-5}$	$1.2{\times}10^{-8}$	0.000095
TRU waste (secondary) with 20 grams WG per drum [d]	INL	WIPP	0.087	$5.2{\times}10^{-5}$	0.072	$4.3{\times}10^{-5}$	$2.1{\times}10^{-8}$	0.000095
TRU waste (primary) in CCOs containing WG plutonium material [e]	INL	WIPP	0.087	$5.2{\times}10^{-5}$	0.072	$4.3{\times}10^{-5}$	$2.1{\times}10^{-8}$	0.000095
TRU (Am-241 in POCs) [c]	INL	WIPP	0.087	$5.2{\times}10^{-5}$	0.072	$4.3{\times}10^{-5}$	$1.1{\times}10^{-7}$	0.000095
LLW/MLLW (B-25) [f]	SRS	NNSS	0.078	$4.7{\times}10^{-5}$	0.052	$3.1{\times}10^{-5}$	$4.0{\times}10^{-10}$	0.00018
MLLW [f, g]	SRS	NNSS	0.094	$5.6{\times}10^{-5}$	0.10	$6.2{\times}10^{-5}$	$7.8{\times}10^{-10}$	0.00018
MLLW [f, g]	INL	NNSS	0.032	$1.9{\times}10^{-5}$	0.034	$2.0{\times}10^{-5}$	$2.2{\times}10^{-10}$	0.000055
LLW/MLLW (B-25) [f]	INL	NNSS	0.026	$1.6{\times}10^{-5}$	0.017	$1.0{\times}10^{-5}$	$1.1{\times}10^{-10}$	0.000055
LLW/MLLW (B-25) [f]	INL	Energy*Solutions*	0.011	$6.2{\times}10^{-6}$	0.011	$6.4{\times}10^{-6}$	$1.2{\times}10^{-10}$	0.000059
LLW/MLLW (B-25) [f]	INL	WCS	0.047	$2.8{\times}10^{-5}$	0.043	$2.6{\times}10^{-5}$	$2.7{\times}10^{-10}$	0.00011
LLW/MLLW (B-25) [f]	SRS	Energy*Solutions*	0.072	$4.3{\times}10^{-5}$	0.073	$4.4{\times}10^{-5}$	$6.7{\times}10^{-11}$	0.00019
LLW/MLLW (B-25) [f]	SRS	WCS	0.044	$2.6{\times}10^{-5}$	0.044	$2.7{\times}10^{-5}$	$4.0{\times}10^{-11}$	0.00014
TRU waste	ORNL	WIPP	0.08	$4.8{\times}10^{-5}$	0.069	$4.1{\times}10^{-5}$	$8.9{\times}10^{-10}$	0.00014
5%-Enriched Uranium [a, b]	NFS	SRS	0.0028	$1.7{\times}10^{-6}$	0.0088	$5.3{\times}10^{-6}$	$6.6{\times}10^{-11}$	0.000013
5%-Enriched Uranium [a, b]	NFS	INL	0.02	$1.2{\times}10^{-5}$	0.06	$3.6{\times}10^{-5}$	$3.0{\times}10^{-10}$	0.000073
VTR Fuel Assemblies	SRS	INL	0.0039	$2.4{\times}10^{-6}$	0.014	$8.5{\times}10^{-6}$	$4.1{\times}10^{-9}$ ($8.6{\times}10^{-10}$) [h]	0.000072
VTR Fuel Assemblies	SRS	ORNL	0.0007	$4.0{\times}10^{-7}$	0.0027	$1.6{\times}10^{-6}$	$1.3{\times}10^{-9}$ ($2.7{\times}10^{-10}$) [h]	0.000011
VTR Fuel Assemblies	INL	ORNL	0.0035	$2.1{\times}10^{-6}$	0.012	$7.2{\times}10^{-6}$	$3.0{\times}10^{-9}$ ($6.3{\times}10^{-10}$) [h]	0.000067
Plutonium From Europe [i]	JWS	SRS	0.0025	$1.5{\times}10^{-6}$	0.0052	$3.1{\times}10^{-6}$	$6.7{\times}10^{-8}$ ($2.7{\times}10^{-8}$) [j]	0.0000061
Plutonium (European) [k]	SRS	INL	0.047	$2.8{\times}10^{-6}$	0.17	$1.0{\times}10^{-4}$	$8.7{\times}10^{-7}$ ($3.5{\times}10^{-7}$) [j]	0.000084
TRU waste (primary) in POCs containing RG plutonium material [c]	SRS	WIPP	0.089	$5.3{\times}10^{-5}$	0.075	$4.5{\times}10^{-5}$	$7.3{\times}10^{-8}$	0.00014
TRU waste (secondary)with 10 grams RG per drum [d]	SRS	WIPP	0.089	$5.3{\times}10^{-5}$	0.075	$4.5{\times}10^{-5}$	$3.5{\times}10^{-8}$	0.00014
TRU waste (primary) in CCOs containing RG plutonium material [e]	SRS	WIPP	0.089	$5.3{\times}10^{-5}$	0.075	$4.5{\times}10^{-5}$	$1.4{\times}10^{-7}$	0.00014

			Incident-Free				Accident	
Material or Wastes	Origin	Transport Destination	Crew Dose (person-rem)	Crew Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)	Radiological Risk (LCF)	Non-radiological Risk (traffic fatalities)
TRU waste (primary) in POCs containing RG plutonium material [c]	INL	WIPP	0.087	5.2×10^{-5}	0.072	4.3×10^{-5}	7.0×10^{-8}	0.000095
TRU waste (secondary)with 10 grams RG per drum [d]	INL	WIPP	0.087	5.2×10^{-5}	0.072	4.3×10^{-5}	3.3×10^{-8}	0.000095
TRU waste (primary) in CCOs containing RG plutonium material [e]	INL	WIPP	0.087	5.2×10^{-5}	0.072	4.3×10^{-5}	1.4×10^{-7}	0.000095
TRU: CASE-1 WG drummed waste [L]	INL	WIPP	0.087	5.2×10^{-5}	0.072	4.3×10^{-5}	2.2×10^{-8}	0.000095
TRU: CASE-1 WG drummed waste	SRS	WIPP	0.089	5.3×10^{-5}	0.075	4.5×10^{-5}	1.9×10^{-8}	0.00014
TRU: CASE-1 RG drummed waste	INL	WIPP	0.087	5.2×10^{-5}	0.072	4.3×10^{-5}	3.4×10^{-8}	0.000095
TRU: CASE-1 RG drummed waste	SRS	WIPP	0.089	5.3×10^{-5}	0.075	4.5×10^{-5}	3.0×10^{-8}	0.00014
TRU: CASE-3 WG drummed waste [m]	INL	WIPP	0.087	5.2×10^{-5}	0.072	4.3×10^{-5}	1.7×10^{-8}	0.000095
TRU: CASE-3 WG drummed waste	SRS	WIPP	0.089	5.3×10^{-5}	0.075	4.5×10^{-5}	1.5×10^{-8}	0.00014
TRU: CASE-3 RG drummed waste	INL	WIPP	0.087	5.2×10^{-5}	0.072	4.3×10^{-5}	2.6×10^{-8}	0.000095
TRU: CASE-3 RG drummed waste	SRS	WIPP	0.089	5.3×10^{-5}	0.075	4.5×10^{-5}	2.3×10^{-8}	0.00014
LLW (B-25)-VTR Operation [n]	INL	NNSS	0.026	1.6×10^{-5}	0.017	1.0×10^{-5}	3.3×10^{-10}	0.000055
LLW (B-12)-VTR Operation	INL	NNSS	0.023	1.4×10^{-5}	0.017	1.0×10^{-5}	2.1×10^{-10}	0.000055
LLW (16’-Iso)-VTR Operation	INL	NNSS	0.044	2.7×10^{-5}	0.019	1.2×10^{-5}	5.9×10^{-10}	0.000055
LLW (B-25)-VTR Operation	INL	Energy*Solutions*	0.011	6.2×10^{-6}	0.011	6.4×10^{-6}	3.7×10^{-10}	0.000059
LLW (B-12)-VTR Operation	INL	Energy*Solutions*	0.009	5.4×10^{-6}	0.011	6.4×10^{-6}	2.4×10^{-10}	0.000059
LLW (16’-Iso)-VTR Operation	INL	Energy*Solutions*	0.017	1.0×10^{-5}	0.009	5.3×10^{-6}	7.0×10^{-10}	0.000059
LLW (B-25)-VTR Operation	INL	WCS	0.047	2.8×10^{-5}	0.043	2.6×10^{-5}	9.0×10^{-10}	0.00011
LLW (B-12)-VTR Operation	INL	WCS	0.041	2.5×10^{-5}	0.043	2.6×10^{-5}	5.8×10^{-10}	0.00011
LLW (16’-Iso)-VTR Operation	INL	WCS	0.079	4.8×10^{-5}	0.036	2.2×10^{-5}	1.7×10^{-9}	0.00011
LLW (B-25)-VTR Operation	ORNL	NNSS	0.069	4.2×10^{-5}	0.064	3.9×10^{-5}	6.7×10^{-10}	0.00015
LLW (B-12)-VTR Operation	ORNL	NNSS	0.061	3.6×10^{-5}	0.064	3.9×10^{-5}	4.3×10^{-10}	0.00015
LLW (16’-Iso)-VTR Operation	ORNL	NNSS	0.12	7.0×10^{-5}	0.053	3.2×10^{-5}	4.0×10^{-10}	0.00015
LLW (B-25)-VTR Operation	ORNL	Energy*Solutions*	0.063	3.8×10^{-5}	0.061	3.6×10^{-5}	1.5×10^{-9}	0.00017
LLW (B-12)-VTR Operation	ORNL	Energy*Solutions*	0.055	3.3×10^{-5}	0.061	3.6×10^{-5}	9.8×10^{-10}	0.00017
LLW (16’-Iso)-VTR Operation	ORNL	Energy*Solutions*	0.11	6.4×10^{-5}	0.050	3.0×10^{-5}	2.6×10^{-9}	0.00017
LLW (B-25)-VTR Operation	ORNL	WCS	0.04	2.4×10^{-5}	0.04	2.4×10^{-5}	1.3×10^{-9}	0.00011
LLW (B-12)-VTR Operation	ORNL	WCS	0.035	2.1×10^{-5}	0.04	2.4×10^{-5}	9.2×10^{-10}	0.00011
LLW (16’-Iso)-VTR Operation	ORNL	WCS	0.067	4.0×10^{-5}	0.033	2.0×10^{-5}	2.1×10^{-9}	0.00011
RH-LLW-VTR Operation [o, n]	INL	NNSS	0.03	1.8×10^{-5}	0.037	2.2×10^{-5}	3.7×10^{-11}	0.000055
RH-LLW-VTR Operation	INL	Energy*Solutions*	0.017	1.0×10^{-5}	0.017	1.0×10^{-5}	3.8×10^{-11}	0.000059
RH-LLW-VTR Operation	INL	WCS	0.053	3.2×10^{-5}	0.068	4.1×10^{-5}	8.9×10^{-11}	0.00011
RH-LLW-VTR Operation	ORNL	NNSS	0.078	4.7×10^{-5}	0.10	6.0×10^{-5}	6.9×10^{-11}	0.00015
RH-LLW-VTR Operation	ORNL	Energy*Solutions*	0.071	4.3×10^{-5}	0.095	5.7×10^{-5}	1.8×10^{-10}	0.00017
RH-LLW-VTR Operation	ORNL	WCS	0.045	2.7×10^{-5}	0.062	3.7×10^{-5}	9.7×10^{-11}	0.00011
RH-TRU-VTR Operation [p, n]	INL	WIPP	0.092	5.5×10^{-5}	0.09	5.4×10^{-5}	2.4×10^{-9}	0.000094
RH-TRU-VTR Operation	ORNL	WIPP	0.085	5.1×10^{-5}	0.09	5.4×10^{-5}	2.3×10^{-9}	0.00014
Diluent [q]	DOE site 1	INL	0.004	2.4×10^{-6}	0.009	5.3×10^{-6}	9.8×10^{-9}	0.00019

Material or Wastes	Origin	Transport Destination	Incident-Free				Accident	
			Crew Dose (person-rem)	Crew Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)	Radiological Risk (LCF)	Non-radiological Risk (traffic fatalities)
Diluent [q]	DOE site 1	SRS	0.001	6.9×10^{-7}	0.002	1.3×10^{-6}	5.0×10^{-9}	0.000065
Diluent [q]	DOE site 2	INL	0.003	2.0×10^{-6}	0.008	4.5×10^{-6}	8.2×10^{-9}	0.00016
Diluent [q]	DOE site 2	SRS	0.001	6.7×10^{-6}	0.002	1.4×10^{-6}	7.4×10^{-9}	0.000053

CASE-1 = aqueous plutonium processing; CASE-3 = pyro-chemical plutonium processing; CCO = criticality control overpack; HEU = highly enriched uranium; JWS= Joint Base Charleston-Weapon Station; LANL = Los Alamos National Laboratory; LCF = latent cancer fatality; LLW = low-level radioactive waste; MLLW = mixed low-level radioactive waste; NFS = Nuclear Fuel Services, Inc.; NNSS = Nevada National Security Site; POC = pipe overpack container; RG = reactor-grade (French) plutonium feed; RH = remote-handled; SRS = Savannah River Site; STA = Secure Transportation Asset; TRU = transuranic; VTR = Versatile Test Reactor; WCS = Waste Control Specialists; WG = weapons-grade plutonium feed; WIPP = Waste Isolation Pilot Plant.

[a] Transported in Type B packages; for analysis purposes, assumed to be shipped in oxide powder form for maximum accident impacts.
[b] Transported by STA.
[c] Transported in 55-gallon drums in 3 TRUPACT-IIs per shipment.
[d] Transported in 55-gallon drums in 3 TRUPACT-IIs per shipment.
[e] Transported in three TRUPACT-IIs per shipment.
[f] Transported in Type A B-25 steel boxes with 5 boxes per shipment; contains fuel fabrication wastes.
[g] MLLW if transported in 55-gallon drums.
[h] The cited values are for the reactor-grade (weapons-grade) fuel.
[i] The plutonium from Europe (France or United Kingdom) will be in 9975 packages within 20-foot ISO containers.
[j] The cited values are for the French (United Kingdom) plutonium accident risks.
[k] It was assumed that the plutonium from France and United Kingdom will be transported from the U.S. port of entry to SRS and then reconfigured and transported to the INL Site.
[L] CASE-1 drummed wastes are cemented wastes in 55-gallon drums, assumed to be 12 drums per shipment, and are in 3 TRUPACT-II for maximizing incident-free population doses.
[m] CASE-3 drummed wastes are metal wastes in 55-gallon drums, assumed to be 28 drums per shipment, and are in 3 TRUPACT-II for maximizing incident free population doses.
[n] The LLW also includes MLLW. All entries with the VTR operation wastes include those generated from the operation of the reactor, its support facilities, and the post-irradiation examination activities. These wastes are transported in a combination of Type A B-25 and B-12 steel boxes with 5 boxes per shipment and in 16-foot ISO containers with 1 container per shipment.
[o] The RH-LLW also includes RH-MLLW. These wastes are transported in a shielded Type B cask. CNS10-160B used as an example.
[p] The RH-TRU also includes RH-MTRU. These wastes are transported in a shielded Type B cask. RH-72B is used.
[q] This material is used to diluent plutonium/uranium-235 waste for transport in CCOs to WIPP. The need is expected to be one shipment every 5 years when the reactor fuel production uses feedstock with no preprocessing activities.

For transportation accidents, the risk factors are given for both radiological impacts, in terms of potential LCFs in the exposed population, and nonradiological impacts, in terms of number of traffic fatalities. LCFs represent the number of additional latent fatal cancers among the exposed population. Under accident conditions, the population would be exposed to radiation from released radioactivity (if the package were damaged) and would receive a direct dose (even if the package is unbreached). For accidents that had no release, the analysis conservatively assumed that it would take about 12 hours to remove the package or commercial vehicle from the accident area (DOE 2002a); 6 hours was assumed for STA shipments. The nonradiological risk factors are for nonoccupational traffic fatalities resulting from transportation accidents.

As stated earlier (see Section E.7.3), the accident dose is called "dose risk" because the values incorporate the spectrum of accident severity probabilities and associated consequences (e.g., dose). The accident dose risks are very low because accident severity probabilities (i.e., the likelihood of accidents leading to confinement breach of a package or shipping cask and release of its contents) are small, and the content and form of the wastes (i.e., solids) are such that a breach would lead to a nondispersible and mostly noncombustible release. Although persons are residing within 50 miles of the transportation route, they are generally quite far from the route. Because RADTRAN 6 uses an assumption of "homogeneous population," it would greatly overestimate the actual doses because this assumption theoretically places people directly adjacent to the route where the highest doses would be present.

As indicated in Table E–5, all per-shipment risk factors are less than one. This means that no LCF or traffic fatalities are expected to occur during each transport. For example, risk factors to the truck crew and population for transporting one shipment of plutonium from LANL to SRS are given as 2.0×10^{-5} and 7.4×10^{-5} LCFs, respectively. This risk can also be interpreted as meaning that there is a chance of 2 in 100,000 that an additional latent fatal cancer could be experienced among the exposed workers from exposure to radiation during one shipment of this waste. Similarly, there is a chance of 1 in 13,500 that an additional latent fatal cancer could be experienced among the exposed population residing along the transport route due to 1 shipment. These chances are essentially equivalent to zero risk. It should be noted that the maximum allowable dose rate in the truck cab is less than or equal to 2 millirem per hour.

Table E–6 shows the annual risks of transporting radioactive materials and wastes under each VTR alternative, and the VTR fuel production options, if the weapons-grade plutonium feedstock were from LANL. **Table E–7** shows the annual risks of transporting radioactive materials and wastes under each VTR alternative, and the VTR fuel production options, if the weapons-grade plutonium feedstock were from SRS. The risks are calculated by multiplying the previously given per-shipment factors by the number of shipments expected to occur in a year and, for radiological doses, by the health risk conversion factors. The number of shipments for the different waste types was calculated using the estimated waste volumes generated during VTR and support facility operations (INL 2020b) and VTR fuel production facility operations (SRNS 2020) and the waste container and shipment characteristics provided in Section E.5.2 and Table E–2. The total annual shipments and associated impacts include transport of VTR fuel assemblies from the fuel production sites under each alternative, of source materials to the fuel production sites, and of generated wastes to the disposal facilities.

Comparison of the results in Tables E–6 and E–7 indicate that the option of fuel production at SRS would generally have higher radiological risk to the population during incident-free transportation than the option of fuel production at INL, due to the greater distances for shipment of the same source (plutonium and uranium) and waste materials. It should be noted that if the weapons-grade plutonium were available at SRS, the annual weapons-grade plutonium-related transports would be lower (about seven shipments), if the VTR fuel production were also at the SRS.

The No Action Alternative, which does not include the installation of VTR facility and its support facilities, would have no additional impacts on the operational facilities at any of the affected sites (i.e., INL, ORNL, and SRS).

Nonradiological accident risks (the potential for fatalities as a direct result of traffic accidents) present the greatest risks, with an estimate of up to 4 fatalities over all the 60-year operation of VTR if fuel production were to occur at SRS and the INL VTR Alternative were selected. Considering that the transportation activities analyzed in this VTR EIS would occur over about 63 years and the average number of traffic fatalities in the United States over the last 10 years (2008 through 2017 calendar years) is about 34,660 per year (DOT 2019a), the traffic fatality risk under all alternatives would be very small. See Section E.13.5 for further discussion of traffic accident fatality rates.

Table E–6. Annual Risks of Transporting Radioactive Material and Waste Under Each Alternative and Reactor Fuel Production Option (Weapons-Grade Plutonium Feedstock at LANL)[a]

Route	Transport Mode	Number of Shipments	One-way Kilometers Traveled	Incident-Free: Crew Dose (person-rem)	Incident-Free: Crew Risk [b]	Incident-Free: Population Dose (person-rem)	Incident-Free: Population Risk [b]	Accident: Radiological Risk [b]	Accident: Non-radiological Risk [b]
No Action Alternative [c]									
All shipments	None	0	0	0	0	0	0	0	0
Total	**Truck**	**0**	**0**	**0**	**0**	**0**	**0**	**0**	**0**
INL VTR Alternative									
INL VTR and Support Facility Operations									
Transuranic (CH and RH) waste to WIPP	Truck	0.23 [d]	534	0.02	0.00001	0.02	0.00001	1×10^{-6}	0.00002
Low-level (CH and RH) waste transport									
INL to Energy*Solutions*	Truck	130	66,491	2.1	0.001	1.9	0.001	2×10^{-8}	0.008
INL to NNSS	Truck	130	172,837	3.9	0.002	4.3	0.003	2×10^{-8}	0.007
INL to WCS	Truck	130	307,511	7.0	0.004	7.9	0.005	4×10^{-8}	0.01
Subtotal [e]	**Truck**	**130**	**308,046**	**7.0**	**0.004**	**8.0**	**0.005**	2×10^{-6}	**0.01**
INL VTR Operations plus Reactor Fuel Production Options									
Total 1 = INL VTR/Support Facility Operations plus INL Reactor Fuel Production									
Total 1 – Fab only [f]-WG Pu	**Truck**	**187**	**444,586**	**10.3**	**0.006**	**11.5**	**0.007**	2×10^{-6}	**0.02**
Total 1 – Prep and Fab-WG Pu (Case 1)	**Truck**	**204**	**483,178**	**12.0**	**0.007**	**12.8**	**0.008**	3×10^{-6}	**0.02**
Total 1 – Prep and Fab-WG Pu (Case 3)	**Truck**	**197**	**467,181**	**11.4**	**0.007**	**12.3**	**0.007**	3×10^{-6}	**0.02**
Total 1 – Fab only-RG Pu [g]	**Truck**	**195**	**461,142**	**10.5**	**0.006**	**12.2**	**0.007**	9×10^{-6}	**0.02**
Total 1 – Prep and Fab-RG Pu (Case 1)	**Truck**	**415**	**963,636**	**29.8**	**0.02**	**28.2**	**0.02**	2×10^{-5}	**0.04**
Total 1 – Prep and Fab-RG Pu (Case 3)	**Truck**	**325**	**757,965**	**22.0**	**0.01**	**21.7**	**0.01**	1×10^{-5}	**0.03**
Total 2 = INL VTR/Support Facility Operations plus SRS Reactor Fuel Production									
Total 2 – Fab only-WG Pu	**Truck**	**202**	**520,996**	**11.2**	**0.007**	**12.3**	**0.007**	3×10^{-6}	**0.02**
Total 2 – Prep and Fab-WG Pu (Case 1)	**Truck**	**219**	**550,768**	**12.6**	**0.008**	**13.4**	**0.008**	3×10^{-6}	**0.02**
Total 2 – Prep and Fab-WG Pu (Case 3)	**Truck**	**212**	**534,622**	**11.9**	**0.007**	**12.8**	**0.008**	3×10^{-6}	**0.02**
Total 2 – Fab only-RG Pu	**Truck**	**203**	**503,613**	**11.0**	**0.007**	**11.5**	**0.007**	3×10^{-6}	**0.02**
Total 2 – Prep and Fab-RG Pu (Case 1)	**Truck**	**423**	**1,001,621**	**30.4**	**0.02**	**27.9**	**0.02**	1×10^{-5}	**0.05**
Total 2 – Prep and Fab-RG Pu (Case 3)	**Truck**	**333**	**794,028**	**22.4**	**0.01**	**21.1**	**0.01**	6×10^{-6}	**0.04**
ORNL VTR Alternative									
ORNL VTR and Support Facility Operations									
Transuranic (CH and RH) waste to WIPP	Truck	0.23 [d]	487	0.02	0.00001	0.02	0.00001	2×10^{-6}	0.00003
Low-level (CH and RH) waste transport									

				Incident-Free				***Accident***	
				Crew		***Population***			
Route	***Transport Mode***	***Number of Shipments***	***One-way Kilometers Traveled***	***Dose (person-rem)***	***Risk [b]***	***Dose (person-rem)***	***Risk [b]***	***Radiological Risk [b]***	***Non-radiological Risk [b]***
ORNL to Energy*Solutions*	Truck	130	408,852	9.3	0.006	11.1	0.007	7×10^{-8}	0.02
ORNL to NNSS	Truck	130	450,619	10.2	0.006	11.7	0.007	2×10^{-8}	0.02
ORNL to WCS	Truck	130	255,208	5.8	0.004	7.3	0.004	5×10^{-8}	0.01
Subtotal [e]	Truck	**130**	**451,106**	**10.2**	**0.006**	**11.8**	**0.007**	**2×10^{-6}**	**0.02**
ORNL VTR Operations plus Reactor Fuel Production Options									
Total 1 = ORNL VTR/Support Facility Operations plus INL Reactor Fuel Production									
Total 1 – Fab only-WG Pu	**Truck**	**202**	**616,966**	**13.2**	**0.008**	**15.0**	**0.009**	**3×10^{-6}**	**0.03**
Total 1 – Prep and Fab-WG Pu (Case 1)	**Truck**	**219**	**676,042**	**15.3**	**0.009**	**16.8**	**0.01**	**3×10^{-6}**	**0.03**
Total 1 – Prep and Fab-WG Pu (Case 3)	**Truck**	**212**	**660,045**	**14.7**	**0.009**	**16.3**	**0.01**	**3×10^{-6}**	**0.03**
Total 1 – Fab only-RG Pu	**Truck**	**210**	**654,006**	**13.8**	**0.008**	**16.2**	**0.01**	**9×10^{-6}**	**0.03**
Total 1 – Prep and Fab-RG Pu (Case 1)	**Truck**	**430**	**1,156,500**	**33.1**	**0.02**	**32.1**	**0.02**	**2×10^{-5}**	**0.05**
Total 1 – Prep and Fab-RG Pu (Case 3)	**Truck**	**340**	**950,829**	**25.3**	**0.02**	**25.7**	**0.02**	**1×10^{-5}**	**0.04**
Total 2 = ORNL VTR/Support Facility Operations plus SRS Reactor Fuel Production									
Total 2 – Fab only-WG Pu	**Truck**	**202**	**617,072**	**14.4**	**0.009**	**15.9**	**0.01**	**3×10^{-6}**	**0.03**
Total 2 – Prep and Fab-WG Pu (Case 1)	**Truck**	**219**	**646,844**	**15.8**	**0.009**	**17.1**	**0.01**	**3×10^{-6}**	**0.03**
Total 2 – Prep and Fab-WG Pu (Case 3)	**Truck**	**212**	**630,698**	**15.1**	**0.009**	**16.5**	**0.01**	**3×10^{-6}**	**0.03**
Total 2 – Fab only-RG Pu	**Truck**	**203**	**599,689**	**14.2**	**0.009**	**15.2**	**0.009**	**3×10^{-6}**	**0.03**
Total 2 – Prep and Fab-RG Pu (Case 1)	**Truck**	**423**	**1,097,697**	**33.6**	**0.02**	**31.5**	**0.02**	**1×10^{-5}**	**0.06**
Total 2 – Prep and Fab-RG Pu (Case 3)	**Truck**	**333**	**890,104**	**25.6**	**0.02**	**24.8**	**0.01**	**6×10^{-6}**	**0.05**
INL Reactor Fuel Production Option									
STA transportation									
All STA routes (with U.S. WG Pu)	STA	13	34,530	0.29	0.0002	0.9	0.0006	5×10^{-7}	0.0007
All STA routes (with European RG Pu) [g]	STA	21	51,086	0.49	0.0003	1.7	0.001	7×10^{-6}	0.001
Low-level waste transport									
INL to NNSS	Truck	15	19,943	0.40	0.0002	0.4	0.0002	2×10^{-9}	0.0008
INL to Energy*Solutions*	Truck	15	7,672	0.15	0.00009	0.2	0.0001	2×10^{-9}	0.0009
INL to WCS	Truck	15	35,482	0.71	0.0004	0.7	0.0004	4×10^{-9}	0.002
Transuranic waste transport									
INL to WIPP (Secondary waste)	Truck	4	9,141	0.35	0.0002	0.3	0.0002	8×10^{-8}	0.0004
INL to WIPP (POCs) Fab only [f]-WG Pu	Truck	13	29,708	1.13	0.0007	0.9	0.0006	2×10^{-7}	0.001
INL to WIPP (diluted PuO_2 in CCOs) [h] – Fab only- WG Pu	Truck	12	27,679	0.87	0.0005	0.7	0.0004	2×10^{-7}	0.001
INL to WIPP (diluted PuO_2 in CCOs) [i] – Fab only-WG Pu	Truck	10	23,530	0.87	0.0005	0.7	0.0004	2×10^{-7}	0.001

Route	Transport Mode	Number of Shipments	One-way Kilometers Traveled	Incident-Free				Accident	
				Crew		Population			
				Dose (person-rem)	Risk [b]	Dose (person-rem)	Risk [b]	Radiological Risk [b]	Non-radiological Risk [b]
INL to WIPP – Prep and Fab-WG Pu (Case 1)	Truck	42	95,980	3.65	0.002	3.0	0.002	8×10^{-7}	0.004
INL to WIPP – Prep and Fab-WG Pu (Case 3)	Truck	35	79,983	3.04	0.002	2.5	0.002	5×10^{-7}	0.003
INL to WIPP – Prep and Fab-RG Pu (Case 1)	Truck	245	559,881	21.28	0.01	17.6	0.01	9×10^{-6}	0.02
INL to WIPP – Prep and Fab-RG Pu (Case 3)	Truck	155	354,211	13.47	0.008	11.1	0.007	5×10^{-6}	0.01
Total reactor fuel production transport									
Total – Fab only-WG Pu	**Truck**	**57**	**136,540**	**3.34**	**0.002**	**3.5**	**0.002**	$\mathbf{1 \times 10^{-6}}$	**0.005**
Total – Prep and Fab-WG Pu (Case 1)	**Truck**	**74**	**175,132**	**4.99**	**0.003**	**4.9**	**0.003**	$\mathbf{1 \times 10^{-6}}$	**0.007**
Total – Prep and Fab-WG Pu (Case 3)	**Truck**	**67**	**159,136**	**4.38**	**0.003**	**4.4**	**0.003**	$\mathbf{1 \times 10^{-6}}$	**0.006**
Total – Fab only-RG Pu [g]	**Truck**	**65**	**153,096**	**3.54**	**0.002**	**4.3**	**0.003**	$\mathbf{8 \times 10^{-6}}$	**0.005**
Total – Prep and Fab-RG Pu (Case 1)	**Truck**	**285**	**655,590**	**22.83**	**0.01**	**20.2**	**0.01**	$\mathbf{2 \times 10^{-5}}$	**0.03**
Total – Prep and Fab-RG Pu (Case 3)	**Truck**	**195**	**449,919**	**15.01**	**0.009**	**13.7**	**0.008**	$\mathbf{1 \times 10^{-5}}$	**0.02**
VTR Fuel Assemblies to ORNL	**Truck**	**15**	**49,804**	**0.05**	**0.00003**	**0.2**	**0.0001**	$\mathbf{5 \times 10^{-8}}$	**0.001**
SRS Reactor Fuel Production Option									
STA transportation									
All STA routes (with U.S. WG Pu)	STA	13	21,976	0.3	0.0002	0.9	0.0005	7×10^{-7}	0.0006
All STA routes (with European RG Pu)	STA	14	4,593	0.04	0.00002	0.12	0.00007	5×10^{-7}	0.0001
Low-level waste transport									
SRS to NNSS	Truck	15	58,343	1.2	0.0007	1.1	0.0007	6×10^{-9}	0.003
SRS to Energy*Solutions*	Truck	15	53,578	1.1	0.0006	1.1	0.0007	1×10^{-9}	0.003
SRS to WCS	Truck	15	32,723	0.7	0.0004	0.7	0.0004	6×10^{-10}	0.002
Transuranic waste transport									
SRS to WIPP (secondary waste)	Truck	4	9,226	0.4	0.0002	0.1	0.00005	2×10^{-8}	0.0006
SRS to WIPP (POCs) Fab only [f]-WG Pu	Truck	13	29,986	1.2	0.0007	1.0	0.0006	2×10^{-7}	0.002
SRS to WIPP (diluted PuO_2 in CCOs) [h] – Fab only WG Pu	Truck	12	27,893	0.9	0.0005	0.8	0.0005	2×10^{-7}	0.002
SRS to WIPP (diluted PuO_2 in CCOs) [i] – Fab-WG Pu	Truck	10	23,255	0.9	0.0005	0.8	0.0005	2×10^{-7}	0.001
SRS to WIPP – Prep and Fab-WG Pu (Case 1)	Truck	42	96,876	3.74	0.002	3.15	0.002	8×10^{-7}	0.006
SRS to WIPP – Prep and Fab-WG Pu (Case 3)	Truck	35	80,730	3.06	0.002	2.55	0.002	8×10^{-7}	0.004
SRS to WIPP – Prep and Fab-RG Pu (Case 1)	Truck	245	565,113	21.81	0.01	18.39	0.01	8×10^{-6}	0.03
SRS to WIPP – Prep and Fab-RG Pu (Case 3)	Truck	155	357,520	13.80	0.008	11.63	0.007	4×10^{-6}	0.02
Total reactor fuel production transport									
Total – Fab only-WG Pu	**Truck**	**57**	**156,650**	**4.2**	**0.003**	**4.2**	**0.002**	$\mathbf{2 \times 10^{-6}}$	**0.008**
Total – Prep and Fab-WG Pu (Case 1)	**Truck**	**74**	**186,422**	**5.52**	**0.003**	**5.28**	**0.003**	$\mathbf{2 \times 10^{-6}}$	**0.01**

Route	Transport Mode	Number of Shipments	One-way Kilometers Traveled	Incident-Free: Crew Dose (person-rem)	Incident-Free: Crew Risk [b]	Incident-Free: Population Dose (person-rem)	Incident-Free: Population Risk [b]	Accident: Radiological Risk [b]	Accident: Non-radiological Risk [b]
Total – Prep and Fab-WG Pu (Case 3)	**Truck**	**67**	**170,276**	**4.84**	**0.003**	**4.67**	**0.003**	**2×10^{-6}**	**0.008**
Total – Fab only-RG Pu) [g]	**Truck**	**58**	**139,267**	**3.97**	**0.002**	**3.37**	**0.002**	**1×10^{-6}**	**0.008**
Total – Prep and Fab-RG Pu (Case 1)	**Truck**	**278**	**637,275**	**23.37**	**0.01**	**19.73**	**0.01**	**8×10^{-6}**	**0.04**
Total – Prep and Fab-RG Pu (Case 3)	**Truck**	**188**	**429,682**	**15.36**	**0.009**	**12.97**	**0.008**	**5×10^{-6}**	**0.03**
VTR Fuel Assemblies to INL	**Truck**	**15**	**56,300**	**0.06**	**0.00004**	**0.21**	**0.0001**	**6×10^{-8}**	**0.001**
VTR Fuel Assemblies to ORNL	**Truck**	**15**	**9,316**	**0.010**	**0.000006**	**0.041**	**0.00002**	**2×10^{-8}**	**0.0002**

Case 1 = aqueous plutonium processing; Case 3 = pyro-chemical plutonium processing; CCO = criticality control overpack; CH = contact-handled; Fab = fuel fabrication; LANL = Los Alamos National Laboratory; LLW = low-level radioactive waste; MLLW = mixed low-level radioactive waste; NNSS = Nevada National Security Site; ORNL = Oak Ridge National Laboratory; POC = pipe overpack container; Prep and Fab = feedstock preparation and fuel fabrication; Pu = plutonium; PuO_2 = plutonium oxide; RG = reactor-grade (European) feed; RH = remote-handled; SRS = Savannah River Site; STA = Secure Transportation Asset; VTR = Versatile Test Reactor; WCS = Waste Control Specialists; WG = weapons-grade feed; WIPP = Waste Isolation Pilot Plant.

[a] For each shipment category, the cited values are annual impact values. The reactor fuel production facilities are to be operational three years before the start of the VTR. The VTR requires about 110 driver fuel assemblies (a full load plus one year of refueling needs) prior to start of operations.

[b] Risk is expressed in terms of LCFs, except for the nonradiological risk, where it refers to the number of traffic accident fatalities. Radiological risk is calculated for one-way travel while nonradiological risk is calculated for two-way travel. Accident dose-risk can be calculated by dividing the risk values by 0.0006 (DOE 2003b). Risks are rounded to one non-zero digit.

[c] Under the No Action Alternative, there would be no new activities and, therefore, no shipments.

[d] Shipments that would occur once every few years are presented as fractional annual shipments.

[e] This subtotal reflects the maximum risk from transporting the LLW/MLLW to NNSS, Energy*Solutions*, or WCS.

[f] Fabrication only is used for the clean weapons-grade plutonium feedstock materials.

[g] Includes impacts from transporting the reactor-grade (European [French or United Kingdom]) plutonium materials, which are assumed to be transported to SRS for repackaging and then transported to INL, if applicable.

[h] Includes impacts from transport of two shipments of adulterants from an assumed distance of 3,000 miles to INL or SRS for dilution of plutonium in CCOs.

[i] Includes impacts from transport of a shipment of diluent from a DOE site (one in 5 years) to INL or SRS for dilution of plutonium in CCOs.

Notes: Totals may differ from the sum of individual entries due to rounding.
All STA routes are the sum of the plutonium and low-enriched uranium transports.
Crew doses are for the truck drivers, assumed to be two drivers for each transport.
To convert kilometers to miles, multiply by 0.6214.
Bolded entries are sums.

Table E–7. Annual Risks of Transporting Radioactive Material and Waste Under Each Alternative and Reactor Fuel Production Option (Weapons-Grade Plutonium Feedstock at SRS)[a]

Route	Transport Mode	Number of Shipments	One-way Kilometers Traveled	Incident-Free: Crew Dose (person-rem)	Incident-Free: Crew Risk [b]	Incident-Free: Population Dose (person-rem)	Incident-Free: Population Risk [b]	Accident: Radiological Risk [b]	Accident: Non-radiological Risk [b]
No Action Alternative [c]									
All shipments	None	0	0	0	0	0	0	0	0
Total	**Truck**	**0**	**0**	**0**	**0**	**0**	**0**	**0**	**0**
INL VTR Alternative									
INL VTR and Support Facility Operations									
Transuranic (CH and RH) waste to WIPP	Truck	0.23 [d]	534	0.02	0.00001	0.02	0.00001	1×10^{-6}	0.00002
Low-level (CH and RH) waste transport									
INL to Energy*Solutions*	Truck	130	66,491	2.1	0.001	1.9	0.001	2×10^{-8}	0.008
INL to NNSS	Truck	130	172,837	3.9	0.002	4.3	0.003	2×10^{-8}	0.007
INL to WCS	Truck	130	307,511	7.0	0.004	7.9	0.005	4×10^{-8}	0.01
Subtotal [e]	Truck	**130**	**308,046**	**7.0**	**0.004**	**8.0**	**0.005**	$\mathbf{2 \times 10^{-6}}$	**0.01**
INL VTR Operations plus Reactor Fuel Production options									
Total 1 = INL VTR/Support Facility Operations plus INL Reactor Fuel Production									
Total 1 – Fab only [f]-WG Pu	**Truck**	**187**	**457,598**	**10.5**	**0.006**	**12,.1**	**0.007**	$\mathbf{3 \times 10^{-6}}$	**0.02**
Total 1 – Prep and Fab-WG Pu (Case 1)	**Truck**	**204**	**496,190**	**12.1**	**0.007**	**13.4**	**0.008**	$\mathbf{3 \times 10^{-6}}$	**0.02**
Total 1 – Prep and Fab-WG Pu (Case 3)	**Truck**	**197**	**480,193**	**11.5**	**0.007**	**12.9**	**0.007**	$\mathbf{3 \times 10^{-6}}$	**0.02**
Total 1 – Fab only-RG Pu [g]	**Truck**	**195**	**461,142**	**10.5**	**0.006**	**12.2**	**0.007**	$\mathbf{9 \times 10^{-6}}$	**0.02**
Total 1 – Prep and Fab-RG Pu (Case 1)	**Truck**	**415**	**963,636**	**29.8**	**0.02**	**28.2**	**0.02**	$\mathbf{2 \times 10^{-5}}$	**0.04**
Total 1 – Prep and Fab-RG Pu (Case 3)	**Truck**	**325**	**757,965**	**22.0**	**0.01**	**21.7**	**0.01**	$\mathbf{1 \times 10^{-5}}$	**0.03**
Total 2 = INL VTR/Support Facility Operations plus SRS Reactor Fuel Production									
Total 2 – Fab only-WG Pu	**Truck**	**197**	**501,945**	**11.0**	**0.007**	**11.5**	**0.007**	$\mathbf{2 \times 10^{-6}}$	**0.02**
Total 2 – Prep and Fab-WG Pu (Case 1)	**Truck**	**212**	**531,717**	**12.3**	**0.008**	**12.6**	**0.008**	$\mathbf{2 \times 10^{-6}}$	**0.02**
Total 2 – Prep and Fab-WG Pu (Case 3)	**Truck**	**205**	**515,571**	**11.6**	**0.007**	**12.0**	**0.008**	$\mathbf{2 \times 10^{-6}}$	**0.02**
Total 2 – Fab only-RG Pu	**Truck**	**203**	**503,613**	**11.0**	**0.007**	**11.5**	**0.007**	$\mathbf{3 \times 10^{-6}}$	**0.02**
Total 2 – Prep and Fab-RG Pu (Case 1)	**Truck**	**423**	**1,001,621**	**30.4**	**0.02**	**27.9**	**0.02**	$\mathbf{1 \times 10^{-5}}$	**0.05**
Total 2 – Prep and Fab-RG Pu (Case 3)	**Truck**	**333**	**794,028**	**22.4**	**0.01**	**21.1**	**0.01**	$\mathbf{6 \times 10^{-6}}$	**0.04**
ORNL VTR Alternative									
ORNL VTR and Support Facility Operations									
Transuranic (CH and RH) waste to WIPP	Truck	0.23 [d]	487	0.02	0.00001	0.02	0.00001	2×10^{-6}	0.00003
Low-level (CH and RH) waste transport									
ORNL to Energy*Solutions*	Truck	130	408,852	9.3	0.006	11.1	0.007	7×10^{-8}	0.02
ORNL to NNSS	Truck	130	450,619	10.2	0.006	11.7	0.007	2×10^{-8}	0.02

Route	Transport Mode	Number of Shipments	One-way Kilometers Traveled	Incident-Free				Accident	
				Crew		Population			
				Dose (person-rem)	Risk [b]	Dose (person-rem)	Risk [b]	Radiological Risk [b]	Non-radiological Risk [b]
ORNL to WCS	Truck	130	255,208	5.8	0.004	7.3	0.004	5×10^{-8}	0.01
Subtotal [e]	Truck	**130**	**451,106**	**10.2**	**0.006**	**11.8**	**0.007**	**2×10^{-6}**	**0.02**
ORNL VTR Operations plus Reactor Fuel Production Options									
Total 1 = ORNL VTR/Support Facility Operations plus INL Reactor Fuel Production									
Total 1 – Fab only-WG Pu	**Truck**	**202**	**650,461**	**13.8**	**0.008**	**16.1**	**0.01**	**3×10^{-6}**	**0.03**
Total 1 – Prep and Fab-WG Pu (Case 1)	**Truck**	**219**	**689,054**	**15.4**	**0.009**	**17.4**	**0.01**	**4×10^{-6}**	**0.03**
Total 1 – Prep and Fab-WG Pu (Case 3)	**Truck**	**212**	**673,057**	**14.8**	**0.009**	**16.9**	**0.01**	**3×10^{-6}**	**0.03**
Total 1 – Fab only-RG Pu	**Truck**	**210**	**654,006**	**13.8**	**0.008**	**16.2**	**0.01**	**9×10^{-6}**	**0.03**
Total 1 – Prep and Fab-RG Pu (Case 1)	**Truck**	**430**	**1,156,500**	**33.1**	**0.02**	**32.1**	**0.02**	**2×10^{-5}**	**0.05**
Total 1 – Prep and Fab-RG Pu (Case 3)	**Truck**	**340**	**950,829**	**25.3**	**0.02**	**25.7**	**0.02**	**1×10^{-5}**	**0.04**
Total 2 = ORNL VTR/Support Facility Operations plus SRS Reactor Fuel Production									
Total 2 – Fab only-WG Pu	**Truck**	**195**	**598,021**	**13.8**	**0.008**	**16.1**	**0.01**	**3×10^{-6}**	**0.03**
Total 2 – Prep and Fab-WG Pu (Case 1)	**Truck**	**212**	**627,793**	**15.5**	**0.009**	**16.2**	**0.01**	**2×10^{-6}**	**0.03**
Total 2 – Prep and Fab-WG Pu (Case 3)	**Truck**	**205**	**611,647**	**14.9**	**0.009**	**15.6**	**0.009**	**2×10^{-6}**	**0.03**
Total 2 – Fab only-RG Pu	**Truck**	**203**	**599,689**	**14.2**	**0.009**	**15.2**	**0.009**	**3×10^{-6}**	**0.03**
Total 2 – Prep and Fab-RG Pu (Case 1)	**Truck**	**423**	**1,097,697**	**33.6**	**0.02**	**31.5**	**0.02**	**1×10^{-5}**	**0.06**
Total 2 – Prep and Fab-RG Pu (Case 3)	**Truck**	**333**	**890,104**	**25.6**	**0.02**	**24.8**	**0.01**	**6×10^{-6}**	**0.05**
INL Reactor Fuel Production Option									
STA transportation									
All STA routes (with U.S. WG Pu)	STA	13	47,542	0.45	0.0003	1.5	0.0009	1×10^{-6}	0.0009
All STA routes (with European RG Pu) [g]	STA	21	51,086	0.49	0.0003	1.7	0.001	7×10^{-6}	0.001
Low-level waste transport									
INL to NNSS	Truck	15	19,943	0.40	0.0002	0.4	0.0002	2×10^{-9}	0.0008
INL to Energy*Solutions*	Truck	15	7,672	0.15	0.00009	0.2	0.0001	2×10^{-9}	0.0009
INL to WCS	Truck	15	35,482	0.71	0.0004	0.7	0.0004	4×10^{-9}	0.002
Transuranic waste transport									
INL to WIPP (Secondary waste)	Truck	4	9,141	0.35	0.0002	0.3	0.0002	8×10^{-8}	0.0004
INL to WIPP (POCs) Fab only [f]-WG Pu	Truck	13	29,708	1.13	0.0007	0.9	0.0006	2×10^{-7}	0.001
INL to WIPP (diluted PuO_2 in CCOs) [h] – Fab only- WG Pu	Truck	12	27,679	0.87	0.0005	0.7	0.0004	2×10^{-7}	0.001
INL to WIPP (diluted PuO_2 in CCOs) [i] – Fab only-WG Pu	Truck	10	23,530	0.87	0.0005	0.7	0.0004	2×10^{-7}	0.001
INL to WIPP – Prep and Fab-WG Pu (Case 1)	Truck	42	95,980	3.65	0.002	3.0	0.002	8×10^{-7}	0.004
INL to WIPP – Prep and Fab-WG Pu (Case 3)	Truck	35	79,983	3.04	0.002	2.5	0.002	5×10^{-7}	0.003
INL to WIPP – Prep and Fab-RG Pu (Case 1)	Truck	245	559,881	21.28	0.01	17.6	0.01	9×10^{-6}	0.02
INL to WIPP – Prep and Fab-RG Pu (Case 3)	Truck	155	354,211	13.47	0.008	11.1	0.007	5×10^{-6}	0.01

Route	Transport Mode	Number of Shipments	One-way Kilometers Traveled	Incident-Free: Crew Dose (person-rem)	Incident-Free: Crew Risk [b]	Incident-Free: Population Dose (person-rem)	Incident-Free: Population Risk [b]	Accident: Radiological Risk [b]	Accident: Non-radiological Risk [b]
Total reactor fuel production transport									
Total – Fab only-WG Pu	**Truck**	**57**	**149,552**	**3.50**	**0.002**	**4.1**	**0.002**	**1×10^{-6}**	**0.005**
Total – Prep and Fab-WG Pu (Case 1)	**Truck**	**74**	**188,144**	**5.15**	**0.003**	**5.5**	**0.003**	**2×10^{-6}**	**0.007**
Total – Prep and Fab-WG Pu (Case 3)	**Truck**	**67**	**172,148**	**4.54**	**0.003**	**5.0**	**0.003**	**2×10^{-6}**	**0.006**
Total – Fab only-RG Pu [g]	**Truck**	**65**	**153,096**	**3.54**	**0.002**	**4.3**	**0.003**	**8×10^{-6}**	**0.005**
Total – Prep and Fab-RG Pu (Case 1)	**Truck**	**285**	**655,590**	**22.83**	**0.01**	**20.2**	**0.01**	**2×10^{-5}**	**0.03**
Total – Prep and Fab-RG Pu (Case 3)	**Truck**	**195**	**449,919**	**15.01**	**0.009**	**13.7**	**0.008**	**1×10^{-5}**	**0.02**
VTR Fuel Assemblies to ORNL	**Truck**	**15**	**49,804**	**0.05**	**0.00003**	**0.2**	**0.0001**	**5×10^{-8}**	**0.001**
SRS Reactor Fuel Production Option									
STA transportation									
All STA routes (with U.S. WG Pu)	STA	6	2,925	0.02	0.00001	0.1	0.00003	4×10^{-10}	0.00008
All STA routes (with European RG Pu)	STA	14	4,593	0.04	0.00002	0.12	0.00007	5×10^{-7}	0.0001
Low-level waste transport									
SRS to NNSS	Truck	15	58,343	1.2	0.0007	1.1	0.0007	6×10^{-9}	0.003
SRS to Energy*Solutions*	Truck	15	53,578	1.1	0.0006	1.1	0.0007	1×10^{-9}	0.003
SRS to WCS	Truck	15	32,723	0.7	0.0004	0.7	0.0004	6×10^{-10}	0.002
Transuranic waste transport									
SRS to WIPP (secondary waste)	Truck	4	9,226	0.4	0.0002	0.1	0.00005	2×10^{-8}	0.0006
SRS to WIPP (POCs) Fab only [f]-WG Pu	Truck	13	29,986	1.2	0.0007	1.0	0.0006	2×10^{-7}	0.002
SRS to WIPP (diluted PuO_2 in CCOs) [h] – Fab only WG Pu	Truck	12	27,893	0.9	0.0005	0.8	0.0005	2×10^{-7}	0.002
SRS to WIPP (diluted PuO_2 in CCOs) [i] – Fab-WG Pu	Truck	10	23,255	0.9	0.0005	0.8	0.0005	2×10^{-7}	0.001
SRS to WIPP – Prep and Fab-WG Pu (Case 1)	Truck	42	96,876	3.74	0.002	3.15	0.002	8×10^{-7}	0.006
SRS to WIPP – Prep and Fab-WG Pu (Case 3)	Truck	35	80,730	3.06	0.002	2.55	0.002	8×10^{-7}	0.004
SRS to WIPP – Prep and Fab-RG Pu (Case 1)	Truck	245	565,113	21.81	0.01	18.39	0.01	8×10^{-6}	0.03
SRS to WIPP – Prep and Fab-RG Pu (Case 3)	Truck	155	357,520	13.80	0.008	11.63	0.007	4×10^{-6}	0.02
Total reactor fuel production transport									
Total – Fab only-WG Pu	**Truck**	**50**	**137,599**	**3.90**	**0.002**	**3.30**	**0.002**	**9×10^{-7}**	**0.008**
Total – Prep and Fab-WG Pu (Case 1)	**Truck**	**67**	**167,371**	**5.28**	**0.003**	**4.42**	**0.003**	**8×10^{-7}**	**0.009**
Total – Prep and Fab-WG Pu (Case 3)	**Truck**	**60**	**151,225**	**4.61**	**0.003**	**3.82**	**0.002**	**8×10^{-7}**	**0.007**
Total – Fab only-RG Pu) [g]	**Truck**	**58**	**139,267**	**3.97**	**0.002**	**3.37**	**0.002**	**1×10^{-6}**	**0.008**
Total – Prep and Fab-RG Pu (Case 1)	**Truck**	**278**	**637,275**	**23.37**	**0.01**	**19.73**	**0.01**	**8×10^{-6}**	**0.04**
Total – Prep and Fab-RG Pu (Case 3)	**Truck**	**188**	**429,682**	**15.36**	**0.009**	**12.97**	**0.008**	**5×10^{-6}**	**0.03**
VTR Fuel Assemblies to INL	**Truck**	**15**	**56,300**	**0.06**	**0.00004**	**0.21**	**0.0001**	**6×10^{-8}**	**0.001**
VTR Fuel Assemblies to ORNL	**Truck**	**15**	**9,316**	**0.010**	**0.000006**	**0.041**	**0.00002**	**2×10^{-8}**	**0.0002**

Route	Transport Mode	Number of Shipments	One-way Kilometers Traveled	Incident-Free				Accident	
				Crew		Population			
				Dose (person-rem)	Risk [b]	Dose (person-rem)	Risk [b]	Radiological Risk [b]	Non-radiological Risk [b]

Case 1 = aqueous plutonium processing; Case 3 = pyro-chemical plutonium processing; CCO = criticality control overpack; CH = contact-handled; Fab = fuel fabrication; INL= Idaho National Laboratory; NNSS = Nevada National Security Site; ORNL = Oak Ridge National Laboratory; POC = pipe overpack container; Pu = plutonium; PuO_2 = plutonium oxide; Prep and Fab = feedstock preparation (processing) and fuel fabrication; RG = reactor-grade (European) feed; RH = remote-handled; SRS = Savannah River Site; STA = Secure Transportation Asset; VTR = Versatile Test Reactor; WCS = Waste Control Specialists; WG = weapons-grade feed; WIPP = Waste Isolation Pilot Plant.

[a] For each shipment category, the cited values are annual impact values. The reactor fuel production facilities are to be operational three years before the start of the VTR. The VTR requires about 110 driver fuel assemblies (a full load plus 1 year of refueling needs) prior to start of operations.

[b] Risk is expressed in terms of LCFs, except for the nonradiological risk, where it refers to the number of traffic accident fatalities. Radiological risk is calculated for one-way travel while nonradiological risk is calculated for two-way travel. Accident dose-risk can be calculated by dividing the risk values by 0.0006 (DOE 2003b). Risks are rounded to one non-zero digit.

[c] Under the No Action Alternative, there would be no new activities and, therefore, no shipments.

[d] Shipments that would occur once every few years are presented as fractional annual shipments.

[e] This subtotal reflects the maximum risk from transporting the LLW/MLLW to NNSS, Energy*Solutions*, or WCS.

[f] Fabrication only is used for the clean weapons-grade plutonium feed materials.

[g] Includes impacts from transporting the reactor-grade (European [French or United Kingdom]) plutonium materials, which are assumed to be transported to SRS for repackaging and then transported to INL, if applicable.

[h] Includes impacts from transport of two shipments of adulterants from an assumed distance of 3,000 miles to INL or SRS for dilution of plutonium in CCOs.

[i] Includes impacts from transport of a shipment of diluent from a DOE site (1 in 5 years) to INL or SRS for dilution of plutonium in CCOs.

Notes: Totals may differ from the sum of individual entries due to rounding.
All STA routes are the sum of the plutonium and low-enriched uranium transports.
Crew doses are for the truck drivers. Analysis assumed two drivers for each transport.
To convert kilometers to miles, multiply by 0.6214.
Bolded entries are sums.

The risks to various exposed individuals under incident-free transportation conditions have been estimated for the hypothetical exposure scenarios identified in Section E.6.3. The maximum estimated doses to workers and the public MEIs are presented in **Table E–8**, considering all shipment types. Doses are presented on a per-event basis (rem per event, per exposure, or per shipment), because it is generally unlikely that the same person would be exposed to multiple events. For those individuals that could have multiple exposures, the cumulative dose could be calculated. The maximum dose to a crew member is based on the assumption that the same individual is responsible for driving every shipment for the duration of the campaign. Note that the potential exists for larger individual exposures under one-time events of a longer duration. For example, the maximum dose to a person stuck in traffic next to a shipment of RH-low-level radioactive waste for 1 hour is calculated to be 0.024 rem (24 millirem). This is generally considered a one-time event for that individual, although this individual may encounter another exposure of a similar or longer duration in his/her lifetime. An inspector inspecting the conveyance and its cargo would be exposed to a maximum dose rate of 0.028 rem (or 28 millirem) per hour if the inspector stood within about 3.3 feet of the cargo for the duration of the inspection.

A member of the public living along the route would likely receive multiple exposures from passing shipments during the period analyzed. The cumulative dose to this resident is calculated by assuming all the shipments pass his or her home. The cumulative dose is calculated assuming that the resident is present for every shipment and is unshielded at a distance of about 98 feet from the route. Therefore, the cumulative dose depends on the number of shipments passing a particular point and is independent of the actual route being considered. If one assumes the maximum resident dose provided in Table E–8 applies to all radioactive transport types, then the maximum dose to this resident (if all the materials were shipped via this route [a total of 430 shipments]) would be about 0.14 millirem annually, with a risk of developing an LCF of about 8.3×10^{-8}. This corresponds to the maximum annual dose that would occur for truck shipments under the ORNL VTR Alternative, which includes an estimated 430 shipments per year.

Table E–8. Estimated Dose to Maximally Exposed Individuals Under Incident-Free Transportation Conditions

Receptor	*Dose to Maximally Exposed Individual*
Workers	
Crew member (truck driver)	2 rem per year [a]
Inspector	0.028 rem per event per hour of inspection
Public	
Resident (along the truck route)	3.2×10^{-7} rem per event
Person in traffic congestion	0.012 rem per event per half an hour stop
Person at a rest stop/gas station	0.0002 rem per event per hour of stop
Gas station attendant	0.00053 rem per event

[a] In addition to complying with DOT requirements, a DOE employee would also need to comply with 10 CFR Part 835 that limits worker radiation doses to 5 rem per year. However, DOE's goal is to maintain radiological exposure as low as reasonably achievable. DOE has, therefore, established the administrative control level of 2 rem per year (DOE 2017a). Based on the number of commercial shipments and the total crew dose to two drivers in Tables E–6 and E–7, a commercial driver dose would not exceed this administrative control limit. Therefore, the administrative control limit is reflected in this table for the maximally exposed truck crew member.

The accident risk assessment and the impacts shown in Tables E–6 and E–7 take into account the entire spectrum of potential accidents, from the fender-bender to the extremely severe. To provide additional insight into the severity of accidents in terms of the potential dose to an MEI and the public, an accident consequence assessment has been performed for a maximum reasonably foreseeable hypothetical transportation accident with a likelihood of occurrence greater than 1 in 10 million per year.

The following assumptions were used to estimate the consequences of maximum reasonably foreseeable offsite transportation accidents:

- The accident is the most severe with the highest release fraction (high-impact and high-temperature fire accident [highest severity category]).
- The individual is 330 feet downwind from a ground-release accident.
- The individual is exposed to airborne contamination for 2 hours and ground contamination for 24 hours with no interdiction or cleanup. A stable weather condition (Pasquill Stability Class F) with a wind speed of 2.2 miles per hour is assumed.
- The population is assumed to have a uniform density to a radius 50 miles and to be exposed to the entire plume passage. A neutral weather condition (Pasquill Stability Class D) with a wind speed of 8.8 miles per hour is assumed. Because the consequence is proportional to the population density, the accident is first assumed to occur in an urban[5] area with the highest density (see Table E–1).
- The type and number of containers involved in the accident is listed in Table E–2. When multiple Type B or shielded Type A shipping casks are transported in a shipment, a single cask is assumed to have failed in the accident. It is unlikely that a severe accident would breach multiple casks.

Table E–9 provides the estimated dose and potential LCFs that could result for an individual and population from a maximum reasonably foreseeable truck transportation accident with the highest consequences under each alternative. (Only those accidents with a probability greater than 1×10^{-7} per year are analyzed.) The accident is assumed to involve a severe impact (collision) in conjunction with a long-duration fire. The highest consequences for the maximum reasonably foreseeable accident based on population dose are from transportation accidents occurring in a rural area involving weapons-grade plutonium oxide powder from LANL to SRS and in a suburban area involving reactor-grade (European fuel) plutonium oxide powder from SRS to INL.

E.9 Impact of Hazardous Waste and Construction and Operational Material Transport

This section evaluates the impacts of transporting hazardous wastes, as well as materials required to construct new facilities. The risks from transporting the construction and nonradiological wastes are estimated in terms of the number of traffic fatalities. For construction materials, it was assumed that materials would be transported 62 miles one way. Hazardous wastes were assumed to be transported about 1,240 miles. The truck accident and fatality rates that were assumed for construction materials were based on the State-level accident and fatality data with appropriate corrections for missing information) (Saricks and Tompkins 1999; UMTRI 2003). This assumption leads to truck accident and fatality rates of 7.69 accidents per 10 million truck-kilometers travelled and 4.08 fatalities per 100 million truck-kilometers travelled for SRS, 6.45 accidents per 10 million truck-kilometers travelled and 3.83 fatalities per 100 million truck-kilometers travelled for INL, and 2.61 accidents per 10 million truck-kilometers travelled and 2.0 fatalities per 100 million truck-kilometers travelled for ORNL, respectively. The truck accident and fatality rates assumed for transport of hazardous materials were 5.77 accidents per 10 million truck-kilometers travelled and 2.34 fatalities per 100 million truck-kilometers travelled (Saricks and Tompkins 1999; UMTRI 2003), which is reflective of the national mean.

[5] If the likelihood of an accident in an urban area is less than 1-in-10 million per year, then the accident is evaluated for a suburban area, and if that also has a likelihood of less than 1-in-10 million per year, then the accident is evaluated for a rural area.

Table E–9. Estimated Dose to the Population and to Maximally Exposed Individuals Under the Maximum Reasonably Foreseeable Accident

Transport Mode	Material or Waste in the Accident With the Highest Consequences	Applicable Alternatives	Range of Likelihood of the Accident (per year) [a]	Population Zone [a]	Population [b] Dose (person-rem)	Population [b] LCF	MEI [c] Dose (rem)	MEI [c] LCF
Truck transport to WIPP [d]	Secondary TRU waste in a TRUPACT II-WG (RG)	All	2.1×10^{-7} to 4.8×10^{-7}	Suburban	1.8 (5.7)	1×10^{-3} (3×10^{-3})	0.001 (0.005)	6×10^{-7} (3×10^{-6})
Truck transport to WIPP [e]	Processed plutonium as TRU waste in POCs- WG (RG)	All	2.4×10^{-7} to 6.7×10^{-7}	Suburban	13.8 (86.2)	8×10^{-3} (5×10^{-2})	0.0075 (0.072)	5×10^{-6} (4×10^{-5})
Truck transport to VTR facilities [f]	VTR fuel assemblies-WG (RG)	All	1.8×10^{-7} to 8.7×10^{-6}	Suburban	48.2 (245)	0.03 (0.15)	0.03 (0.17)	2×10^{-5} (1×10^{-4})
Truck transport to disposal sites [g]	LLW in B-25s	All	2.0×10^{-7} to 6.9×10^{-6}	Suburban	0.033	2×10^{-5}	0.00001	7×10^{-9}
Truck transport to WIPP [g]	Processed TRU waste in CCOs –WG (RG)	All	1.8×10^{-7} to 5.2×10^{-7}	Suburban	27.6 (172)	0.017 (0.10)	0.015 (0.14)	9×10^{-6} (9×10^{-5})
STA transport to SRS or INL [h]	Plutonium (in oxide powder) in a Type B package- WG (RG)	All	1.2×10^{-7} to 2.5×10^{-6}	Rural (Suburban)	348 (61,500)	0.21 (37)	4.3 (21.2)	3×10^{-3} (1×10^{-2})
Truck transport to SRS or INL [i]	Diluent for diluting plutonium waste	All	2.4×10^{-7} to 2.7×10^{-7}	Rural	0.076	5×10^{-5}	0.006	4×10^{-6}

CCO = criticality control overpack; INL= Idaho National Laboratory; LCF = latent cancer fatality; LLW = low-level radioactive waste; MEI = maximally exposed individual; NNSS = Nevada National Security Site; POC = pipe overpack container; RG = reactor-grade plutonium; SRS = Savannah River Site; STA = safeguards transporter; TRU = transuranic; TRUPACT-II = Transuranic Package Transporter Model 2; VTR=Versatile Test Reactor; WG = weapons-grade plutonium; WIPP = Waste Isolation Pilot Plant.

[a] The likelihood shown is the range of likelihood estimated among the alternatives given the number of shipments over a specific time period. If the likelihood of an accident is equal to or greater than 1 in 10 million per year for both suburban and urban population zones, then the consequences are provided for the urban population zone.

[b] Population extends at a uniform density to a radius of 50 miles. The weather condition was assumed to be Pasquill Stability Class D with a wind speed of 8.8 miles per hour.

[c] The MEI is assumed to be 330 feet downwind from the accident and exposed to the entire plume of the radioactive release. The weather condition is assumed to be Pasquill Stability Class F with a wind speed of 2.2 miles per hour.

[d] While these shipments would occur under all alternatives, the likelihood of an accident in a rural area (from INL) is greater than that in suburban area (from SRS). However, the accident for transport to WIPP from SRS has a larger population dose and risk, as indicated here.

[e] While these shipments would occur under all alternatives and even though the consequences of an accident are larger for shipments from INL than for shipments from SRS, the likelihood of an accident in a suburban area from SRS transport is greater than that from INL. Therefore, the transport from SRS would lead to a larger population risk, as indicated here.

[f] While these shipments would occur under all alternatives and even though the likelihood of an accident is greater for shipments from INL to ORNL than for shipments from SRS to INL or ORNL, the consequences of an accident in a suburban area for the SRS to INL route are larger than those from the other routes, leading to a larger population risk. Therefore, the transport from SRS to INL is indicated here.

[g] While these shipments would occur under all alternatives and even though the consequences of an accident is are larger for shipments from INL to WCS than for shipments from SRS to any disposal sites, the likelihood of an accident in a suburban area from SRS to Energy*Solutions* transport is greater than that from INL to any disposal sites. Therefore, the transport from SRS to Energy*Solutions* would lead to a larger population risk, as indicated here.

[h] While these shipments would occur under all alternatives, the likelihood of an accident in a rural area from transport from LANL to SRS is greater than that for transport to INL. [The likelihood of an accident in a rural area from LANL to INL transport is 1.9×10^{-6} per year, with a population dose consequence of 286 person-rem.] However, the population risk is higher when the plutonium is a reactor-grade (French) with the likelihood of an accident in a suburban area of 1.2×10^{-7} per year. Therefore, the transport to INL from SRS for the INL VTR fuel production option would lead to a larger population risk, as indicated here.

[i] Shipments of diluents to INL or SRS originates from two DOE sites. While these shipments would occur under all alternatives, the likelihood of an accident in a rural area to INL is greater would greater than one in 10 million (e.g., 1.0×10^{-7} per year). The likelihood of an accident on either route to SRS is less than one in 10 million. Therefore, transport along the route to INL with the greater likelihood of an accident would lead to a larger population risk, as indicated here.

Table E–10 summarizes the impacts in terms of total number of kilometers, accidents, and fatalities for the VTR alternatives and reactor fuel production options. The results indicate that there would be a smaller risk of traffic accidents and fatalities for the INL VTR Alternative that uses the existing facilities to support the VTR operation than for the ORNL VTR Alternative. For the ORNL VTR alternative, additional support facilities have to be constructed. The construction impacts of the needed support facilities would be about 30 percent of the VTR construction impacts (Leidos 2020).

Table E–10. Estimated Impacts of Construction Material and Hazardous Waste Transport

Materials	*Number of Shipments*	*Total Distance Traveled (kilometers; two-way)*	*Number of Accidents*	*Number of Fatalities*
INL VTR Alternative				
Construction	17,635	23,928,500	1.4×10^{1}	6×10^{-1}
Hazardous Wastes	115	230,000	1.3×10^{-1}	5×10^{-3}
Total	17,750	24,158,500	1.4×10^{1}	6×10^{-1}
ORNL VTR Alternative				
Construction	22,930	31,107,100	1.7×10^{1}	7×10^{-1}
Hazardous Wastes	150	299,000	1.7×10^{-1}	7×10^{-3}
Total	23,075	31,406,050	1.7×10^{1}	7×10^{-1}
INL Fuel Production Option[a]				
Construction	0.0			
Hazardous Wastes	0.0			
SRS Fuel Production Option				
Construction	1,227	245,400	1.9×10^{-1}	1.0×10^{-2}
Hazardous Wastes	1,227	245,400	1.9×10^{-1}	1.0×10^{-2}
Spent Fuel Storage Pad				
INL VTR Alternative	711	142,000	9.2×10^{-2}	6×10^{-3}
ORNL VTR Alternative	711	142,000	3.7×10^{-2}	3×10^{-3}

INL= Idaho National Laboratory; ORNL = Oak Ridge National Laboratory; SRS = Savannah River Site; VTR = Versatile Test Reactor.

[a] INL existing facilities do not require major construction to accommodate the equipment (e.g., glove boxes) for the fuel production activities.

Note: To convert kilometers to miles, multiply by 0.6214.

Source: INL 2020c; Leidos 2020; SRNS 2020.

E.10 Onsite Transports

Onsite shipment of radioactive materials and wastes would occur at the SRS, INL, and ORNL. These shipments would not have any substantial effect on members of the public because roads between the site processing areas are closed to the public or have comparatively short distances to which the public has access. The onsite waste shipments from construction and operations evaluated in this EIS would be a small fraction of the overall site waste shipments.

E.11 Conclusions

Based on the results presented in the previous sections, the following conclusions have been reached (see Tables E–6 and E–7):

- For all alternatives, the transportation of radioactive material and waste likely would result in no additional fatalities as a result of radiation, either from incident-free operation or postulated transportation accidents.
- The highest annual risk to the public due to incident-free transportation would be under the ORNL VTR Alternative with the INL Reactor Fuel Production Options, where up to 430 truck shipments of radioactive materials, wastes, and VTR fuel assemblies would be transported annually.
- The nonradiological accident risks (the potential for fatalities as a direct result of traffic accidents) are greater than the radiological accident risks.
- Under both VTR alternatives, up to four traffic fatalities would be expected over the duration of the activities (which is assumed to be 63 years, 60 years of VTR operation and 3 additional years of fuel production prior to VTR operation). For comparison, in the United States in 2017

there were over 37,133 traffic fatalities due to all vehicular crashes (DOT 2019a). The incremental increase in risk to the general population from shipments associated with the VTR program would, therefore, be very small and would not substantially contribute to cumulative impacts.

E.12 Long-term Impacts of Transportation

The *Yucca Mountain EIS* (DOE 2002a, 2008b) analyzed the cumulative impacts of the transportation of radioactive material, consisting of impacts of historical shipments of radioactive waste and used nuclear fuel, reasonably foreseeable actions that include transportation of radioactive material, and general radioactive material transportation that is not related to a particular action. The collective dose to the general population and workers was the measure used to quantify cumulative transportation impacts. This measure of impact was chosen because it may be directly related to the LCFs, using a cancer risk coefficient. The cumulative impacts information data in the Yucca EIS was updated in the 2015 *Surplus Plutonium Disposition (SPD) Supplemental EIS* (DOE 2015a), and is further updated to include the current information on various activities. The timeframe of the SPD Supplemental EIS transportation impacts analysis began in 1943 and extended to 2073. The time frame for this VTR EIS analysis is for 63 years beyond the 2028 start of VTR operation, which extends the cumulative impact period beyond 2090.

Table E–11 provides a summary of the total worker and general population collective doses from various transportation activities. The table shows that the impacts of this program are small compared with the overall transportation impacts. The total collective worker dose from all types of shipments (the alternatives in this VTR EIS; historical, reasonably foreseeable actions; and general transportation) was estimated to be about 430,000 person-rem (about 258 LCFs). The total general population collective dose was estimated to be about 441,000 person-rem (about 265 LCFs). The majority of the collective dose for workers and the general population is due to the general transportation of radioactive material. Examples of these activities are shipments of radiopharmaceuticals to nuclear medicine laboratories and shipments of commercial low-level radioactive waste to commercial disposal facilities. The total number of LCFs (among the workers and the general population) estimated to result from radioactive material transportation over the period between 1943 and 2091 is about 525, or an average of about 4 LCFs per year. Over this same period (about 148 years), approximately 88 million people would have died from cancer, based on National Center for Health Statistics data. The annual number of cancer deaths in the United States in 2017 was about 599,000 (CDC 2019) with about a 3 percent fluctuation in the number of cancer fatalities from 1 year to the next, over the last previous 10 years (2008 through 2017), and a mean of 584,000 cancer fatalities per year. The transportation-related LCFs would be 0.0006 percent of the total annual number of LCFs. Therefore, this number is indistinguishable from the natural fluctuation in the total annual death rate from cancer.

E.13 Uncertainty and Conservatism in Estimated Impacts

The sequence of analyses performed to generate the estimates of radiological risk for transportation includes: (1) determination of the inventory and characteristics, (2) estimation of shipment requirements, (3) determination of route characteristics, (4) calculation of radiation doses to exposed individuals (including estimating environmental transport and uptake of radionuclides), and (5) estimation of health effects. Uncertainties are associated with each of these steps. Uncertainties exist in the way that the physical systems being analyzed are represented by the computational models. There are also uncertainties in the data required to exercise the models (due to measurement errors, sampling errors, natural variability, or unknowns caused simply by the future nature of the actions being analyzed) and the calculations themselves (e.g., approximate algorithms used within the computer codes).

Table E–11. Cumulative Transportation-Related Radiological Collective Doses and Latent Cancer Fatalities

Category	*Collective Worker Dose (person-rem)*	*Collective General Population Dose (person-rem)*
Historical [a]	49	25
Past, Present, and Reasonably Foreseeable Future Actions (DOE) [a, b]	29,600	36,700
Additional Reasonably Foreseeable Future Actions (DOE)		
Permanent Disposal or Interim Storage of Spent Nuclear Fuel [c]	5,600–5,900	1,100–1,200
Final Greater-Than-Class C EIS [d]	180	68
Final SEIS for the Disposition of Du Oxides Conversion Product [l]	145–276	217–723
SRS Pit Production EIS [m]	581–901	334–455
SPD SEIS Proposed Action [n]	230–650	150–580
WIPP Supplemental Analysis [e]	492	383
Production of Tritium in a Commercial Light Water Reactor [f]	25–60	2.7–12
Liquid Highly Enriched Uranium Shipments from Canada [g]	17	10
Santa Susana Field Laboratory Remediation [h]	3.0	0.89
Acceptance and Disposition of Spent Nuclear Fuel from the Federal Republic of Germany [i]	0.12–10.9	0.54–4.7
Sister Rod Shipments [j]	0.27	0.75
Total Past, Present, and Reasonably Foreseeable Future Actions (DOE)	36,900–38,100	38,900–40,100
Past, Present, and Reasonably Foreseeable Future Actions (non-DOE) [a]	5,380	61,300
General Radioactive Materials Transportation [a]	384,000	338,000
Transportation Impacts in this VTR EIS [k]		
INL VTR Alternative	*624–1,915*	*699–1,777*
ORNL VTR Alternative	*832–2,117*	*945–2,022*
Total [o]	**427,000–430,000**	**439,000–441,000**
Total Latent Cancer Fatalities [p]	256–258	263–265

[a] DOE 2015a:Table 4-48, p. 4-136 and 4-137. Historical shipments are shipments that occurred in the past.
[b] DOE 2015a:Table 4-48, p. 4-136 and 4-137. Excluding the doses from shipping in the draft Greater-Than-Class C Waste EIS and the DUF6 Conversion at Paducah and Portsmouth EISs.
[c] DOE 2008b:Table 8-14, p. 8-44. For the purposes of the transportation cumulative impacts analysis, DOE considered the Yucca Mountain, Nevada, repository site as a surrogate destination for an interim storage facility or a permanent repository.
[d] DOE 2016a:Table 4.3.9-1, p. 4-68 and 4-69; DOE 2018a:3-20.
[e] DOE 2009:Table 2, p. 5.
[f] DOE 2016b:Table F-12, p. F-17. Calculated from LCFs.
[g] DOE 2013:A-11. Calculated from LCFs.
[h] DOE 2018b:Table H-9, p. H-31.
[i] DOE 2017b:Table 4-28, p. 4-68.
[j] DOE 2015b:Table 3-1, p. 24. Calculated from LCFs.
[k] From Section E.8 (Table E-6) of Appendix E, and adjusted for the 63 years of cumulative operations in this VTR EIS.
[l] DOE 2020b:Table 4-51, p 4-93. The highest disposal option impacts for rail and truck shipments.
[m] DOE 2020a:Table 5-7, for 50–80 pits per year; 50 years of operation.
[n] DOE 2015a:Table E-20; this addition is a conservative assumption as the range of alternatives in this SEIS are not implemented. The impacts of transporting surplus pits from Pantex to SRS or LANL for disassembly and related activities are a fraction of values presented here.
[o] Total values are rounded to three significant figures. (Note: the lower end of the range totals includes the lowest value from the VTR alternatives; the upper end of the range includes the highest value.)
[p] Total LCFs are calculated assuming 0.0006 LCFs per person-rem of exposure (DOE 2003b).

In principle, one can estimate the uncertainty associated with each input or computational source and predict the resultant uncertainty in each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final, or absolute, result. However, conducting such a full-scale quantitative uncertainty analysis is often impractical and sometimes impossible, especially for actions to be initiated at an unspecified time in the future. Instead, the risk analysis is designed to ensure, through uniform and judicious selection of scenarios, models, and input parameters, that relative comparisons of risk among the various alternatives are meaningful. In the transportation risk assessment, this design is accomplished by uniformly applying common input parameters and assumptions to each alternative. Therefore, although considerable uncertainty is inherent in the absolute magnitude of the transportation risk for each alternative, much less uncertainty is associated with the relative differences among the alternatives in a given measure of risk.

In the following sections, areas of uncertainty are discussed for the assessment steps enumerated above. Special emphasis is placed on identifying whether the uncertainties affect relative or absolute measures of risk. The reality and conservatism of the assumptions are addressed. Where practical, the parameters that most significantly affect the risk assessment results are identified.

E.13.1 Uncertainties in Material Inventory and Characterization

The inventories and the physical and radiological characteristics are important input parameters to the transportation risk assessment. The potential number of shipments for all alternatives is primarily based on the projected dimensions of package contents, the strength of the radiation field, and assumptions concerning shipment capacities. The physical and radiological characteristics are important in determining the material released during accidents and the subsequent doses to exposed individuals through multiple environmental exposure pathways.

Uncertainties in the inventory and characterization are reflected in the transportation risk results. If the inventory is overestimated (or underestimated), the resulting transportation risk estimates are also overestimated (or underestimated) by roughly the same factor. However, the same inventory estimates are used to analyze the transportation impacts of each of the alternatives. Therefore, for comparative purposes, the observed differences in transportation risks among the alternatives, as given in Tables E–6 and E–7 are believed to represent unbiased, reasonably accurate estimates from current information in terms of relative risk comparisons.

E.13.2 Uncertainties in Containers, Shipment Capacities, and Number of Shipments

The transportation required for each alternative is based in part on assumptions concerning the packaging characteristics and shipment capacities for commercial trucks. Representative shipment capacities have been defined for assessment purposes based on probable future shipment capacities. In reality, the actual shipment capacities may differ from the predicted capacities such that the projected number of shipments and, consequently, the total transportation risk, would change. However, although the predicted transportation risks would increase or decrease accordingly, the relative differences in risks among alternatives would remain about the same.

One factor that can influence shipment capacities for TRU waste using TRUPACT II packages, and therefore, the number of shipments, is the use of dunnage. Dunnage is secured space not occupied by waste or waste containers. Dunnage may be used to keep the entire payload from shifting position during transit or when the payload has reached one or more shipping limits for parameters such as weight, gas generation, radioactivity, or fissile mass (Casey 2007). Use of dunnage was factored into determining the number of shipments of surplus plutonium and TRU waste to WIPP. The impact of dunnage on the determination of number of shipments is highly variable among DOE sites and even among individual waste streams. However, to give an idea as to its impact, historically dunnage has comprised less than 10 percent of the TRU waste volume transported from DOE sites to WIPP. If the number of shipments of incidental TRU waste associated with this VTR EIS was increased by this amount, it would have a negligible impact on the results for each alternative. As in the case of variations in shipment capacities addressed in the previous paragraph, incorporation of factors related to dunnage into shipment calculations would not change the relative differences in risks among alternatives.

E.13.3 Uncertainties in Route Determination

Analyzed routes have been determined between all origin and destination sites considered in this VTR EIS. The routes have been determined to be consistent with current guidelines, regulations, and practices, but may not be the actual routes that would be used in the future. In reality, the actual routes could differ from the ones that are analyzed with regard to distances and total population along the routes. Moreover, because materials could be transported over an extended time starting at some time in the future, the highway infrastructure and the demographics along routes could change. These effects have not been accounted for in the transportation assessment; however, it is not anticipated that these changes would significantly affect relative comparisons of risk among the alternatives considered in this VTR EIS.

E.13.4 Uncertainties in the Calculation of Radiation Doses

The models used to calculate radiation doses from transportation activities introduce a further uncertainty in the risk assessment process. Estimating the accuracy or absolute uncertainty of the risk assessment results is generally difficult. The accuracy of the calculated results is closely related to the limitations of the computational models and to the uncertainties in each of the input parameters that the model requires. The single greatest limitation facing users of RADTRAN, or any computer code of this type, is the scarcity of data for certain input parameters. Populations (off-link and on-link) along the transportation routes, shipment surface dose rates, and individuals residing near the routes are the most uncertain data in dose calculations. In preparing these data, one makes assumptions that the off-link population is uniformly distributed; the on-link population is proportional to the traffic density, with an assumed occupancy of two persons per car; the shipment surface dose rate is the maximum allowed dose rate; and a potential exists for an individual to be residing at the edge of the highway. It is clear that not all assumptions are accurate. For example, the off-link population is mostly heterogeneous, and the on-link traffic density varies widely within a geographic zone (i.e., urban, suburban, or rural). Finally, added to this complexity are the assumptions regarding the expected distance between the public and the shipment at a traffic stop, rest stop, or traffic jam and the afforded shielding.

Uncertainties associated with the computational models are reduced by using state-of-the-art computer codes that have undergone extensive review. Because many uncertainties are recognized but difficult to quantify, assumptions are made at each step of the risk assessment process intended to produce conservative results (i.e., overestimate the calculated dose and radiological risk). Because parameters and assumptions are applied consistently to all alternatives, this model bias is not expected to affect the meaningfulness of relative comparisons of risk. However, the results may not represent risks in an absolute sense.

E.13.5 Uncertainties in Traffic Fatality Rates

Vehicle accident and fatality rates were taken from data provided in *State-Level Accident Rates for Surface Freight Transportation: A Reexamination*, ANL/ESD/TM-150 (Saricks and Tompkins 1999). Truck and rail accident rates were computed for each State based on statistics compiled by the Federal Highway Administration, Office of Motor Carriers and Federal Railroad Administration, from 1994 to 1996. The rates are provided per unit car-kilometers for each State, as well as national average and mean values. In this analysis, route-specific (origin-destination) rates were used.

Finally, it should be emphasized that the analysis was based on accident data for the years 1994 through 1996. While this data may be the best available data, future accident and fatality rates may change as a result of vehicle and highway improvements. The recent U.S. DOT national accident and fatality statistics for large trucks and buses indicates lower accident and fatality rates for recent years compared to those of 1994 through 1996 and earlier statistical data (DOT 2009, 2019b).

E.14 References

Blower, D. and A. Matteson, 2003, Evaluation of the Motor Carrier Management Information System Crash File, Phase One, UMTRI-2003-06, University of Michigan Transportation Research Institute, Ann Arbor, Michigan, Accession No. ML112650033.

Casey, S. C., 2007, U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico, *Nearly 50,000 m^3 of Waste and More Than 5000 Shipments to WIPP: What Does it All Mean?*, WM '07 Conference, February 25 – March 1, 2007, Tucson, Arizona (accessed on November 14, 2012, http://www. wmsym.org/archives/2007/pdfs/7384.pdf).

CDC (Centers for Disease Control and Prevention), 2019, *Deaths: Final Death for 2017*, *National Vital Statistics Reports*, Vol. 68, No. 9, U.S. Department of Health and Human Services, National Center for Health Statistic, National Vital Statistics System (available at www.cdc.gov/nchs/products/index.htm), June 24.

Census (U.S. Census Bureau), 2010, *Resident Population Data, Population Change 1910-2010* (accessed on December 21, 2010 at http://2010.census.gov/2010census/data/apportionment-pop-text.php).

CH2MHILL 2009, Safety Analysis Report for Packaging, Hanford Unirradiated Fuel Package, CHPRC-00164, Revision 1, November.

DHS (U.S. Department of Homeland Security), 2016, *Nuclear/Radiological Incident Annex*, Washington, DC, October.

DHS (U.S. Department of Homeland Security), 2019, *National Response Framework, Fourth Edition*, Federal Emergency Management Agency, Washington, DC, October 28.

DOE (U.S. Department of Energy), 1994, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facility*, DOE-HDBK-3010-94, (Change Notice No. 1, March 2000), Washington, DC, December.

DOE (U.S. Department of Energy), 1996, *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218F, Office of Environmental Management, Washington, DC, February.

DOE (U.S. Department of Energy), 1997, *Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement*, DOE/EIS-0026-S-2, Carlsbad Area Office, Carlsbad, New Mexico, September.

DOE (U.S. Department of Energy), 1999, *Surplus Plutonium Disposition Final Environmental Impact Statement*, DOE/EIS-0283, Office of Fissile Materials Disposition, Washington, DC.

DOE (U.S. Department of Energy), 2002a, *Final Environmental Impact Statement for a Geological Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada*, DOE/EIS-0250, Office of Civilian Radioactive Waste Management, Yucca Mountain Site Characterization Office, North Las Vegas, Nevada, February.

DOE (U.S. Department of Energy), 2002b, *A Resource Handbook on DOE Transportation Risk Assessment*, DOE/EM/NTP/HB-01, Office of Environmental Management, National Transportation Program, Albuquerque, New Mexico, July.

DOE (U.S. Department of Energy), 2003a, *Final West Valley Demonstration Project Waste Management Environmental Impact Statement*, DOE/EIS-0337F, West Valley Area Office, West Valley, New York, December.

DOE (U.S. Department of Energy), 2003b, Estimating Radiation Risk from Total Effective Dose Equivalent (TEDE), ISCORS Technical Report No. 1, DOE/EH-412/0015/0802, Rev. 1, Office of Environmental Policy and Guidance, January.

DOE (U.S. Department of Energy), 2008a, *Radioactive Material Transportation Practices Manual for Use with DOE O 460.2A*, DOE M 460.2-1A, Office of Environmental Management, Washington, DC, June 4.

DOE (U.S. Department of Energy), 2008b, *Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada*, DOE/EIS-0250F-S1, Office of Civilian Radioactive Waste Management, Las Vegas, Nevada, June.

DOE (U.S. Department of Energy), 2009, *Supplement Analysis for Waste Isolation Pilot Plant Site-Wide Operations,* Table 2, DOE/EIS- 0026-SA-7, May.

DOE (U.S. Department of Energy), 2013, *Supplement Analysis, Savanah River Site Spent Nuclear Fuel Management*, Page A-11; DOE/EIS-0279-SA-1 (DOE/EIS-0218-SA-6), March.

DOE (U.S. Department of Energy), 2015a, *Final Surplus Plutonium Disposition Supplemental Environmental Impact Statement*, DOE/EIS-0283-S2, Washington, DC, April.

DOE (U.S. Department of Energy), 2015b, *Supplement Analysis-Proposed Shipment of Commercial Spent Nuclear Fuel to DOE National Laboratories for the Research and Development Purposes*, Table 3-1, DOE/EIS-0203-SA-7 (DOE/EIS-250-S-1-S-2), December.

DOE (U.S. Department of Energy), 2016a, *Final Environmental Impact Statement for the Disposal of Greater-than-Class C (GTCC) Low-Level Radioactive Waste,* Table 4.3.9-1, p. 4-68; DOE/EIS-0375, January.

DOE (U.S. Department of Energy), 2016b, DOE Order 151.1D, *Comprehensive Emergency Management System*, Office of Emergency Operations, Washington, DC, August 11.

DOE (U.S. Department of Energy), 2017a, *DOE Standard Radiological Control*, DOE-STD-1098-2008, Change Notice 1, Washington, DC, January.

DOE (U.S. Department of Energy), 2017b, *Final Environmental Assessment for the Acceptance and Disposition of Spent Nuclear Fuel Containing the U.S.-Origin Highly Enriched Uranium from Federal Republic of Germany,* Table 4-28, p. 4-68; DOE/EA-1977, December.

DOE (U.S. Department of Energy), 2018a, *Environmental Assessment for the Disposal of Greater-than-Class C (GTCC) Low-Level Radioactive Waste and GTCC-Like Wastes at the Waste Control Specialist, Andrews County, Texas,* Section 3.9.2, p. 3-20; DOE/EA-2082, October.

DOE (U.S. Department of Energy), 2018b, *Final Environmental Impact Statement for Remediation of Area IV and the Northern Buffer Zone of the Santa Susana Field Laboratory,* Table H-9, p. H-31, DOE/EIS-0402, November.

DOE (U.S. Department of Energy), 2020a, *Final Environmental Impact Statement for Plutonium Pit Production at the Savannah River Site in South Carolina*, DOE/EIS-0541, National Nuclear Security Administration, Savannah River Site, September.

DOE (U.S. Department of Energy), 2020b, *Final Supplemental Environmental Impact Statement for Disposition of Depleted Uranium Oxide Conversion Product Generated from DOE's Inventory of Depleted Uranium Hexafluoride*, Table 4-51, p. 4-93, DOE/EIS-0359-S1 and DOE/EIS-0360-S1, April.

DOT (U.S. Department of Transportation), 2000, *Comprehensive Truck Size and Weight Study* – Volume I, Summary Report, FHWA-PL-00-029, Federal Highway Administration, August 19.

DOT (U.S. Department of Transportation), 2008, *Radioactive Material Regulations Review*, Pipeline and Hazardous Materials Safety Administration, December.

DOT (U.S. Department of Transportation), 2009, *Large Truck and Bus Crash Facts 2007*, FMCSA-RRA-09-029, Analysis Division Federal Motor Carrier Safety Administration, January.

DOT (U.S. Department of Transportation), 2016, *2016 Emergency Response Guidebook*, Pipeline and Hazardous Materials Safety Administration, Washington, DC.

DOT (U.S. Department of Transportation), 2019a, *Traffic Safety Facts 2017*, DOT HS 812 806, National Highway Traffic Safety Administration, National Center for Statistics and Analysis, Washington, DC, September.

DOT (U.S. Department of Transportation), 2019b, Traffic Safety Facts, 2017 Data, Large Trucks, DOT HS 812 663, National Highway Traffic Safety Administration, National Center for Statistics and Analysis, Washington, DC, January.

INL (Idaho National Laboratory), 2019a, VTR Fuel Facility Plan, INL/LTD-19-54001, Idaho Falls, Idaho, May 15. **OUO**

INL (Idaho National Laboratory), 2019b, Conceptual Design Report for the Versatile Test Reactor (VTR), INL/LTD-19-56681, Rev. 0, Idaho Falls, Idaho, November. **OUO, Export Controlled Information**

INL (Idaho National Laboratory), 2020a, *Recommended Representative Isotopic Compositions for Potential VTR Pu Supplies*, INL/MIS-20-57910, Rev. 1, April.

INL (Idaho National Laboratory), 2020b, Technical Evaluation Versatile Test Reactor (VTR) Waste and Material Data for Environmental Impact Statement (EIS), TEM-10300-1, Rev. 9, TEV-3976, Rev. 1, November 12.

INL (Idaho National Laboratory), 2020c, VTR Hazards and Impacts Information in Support of National Environmental Policy Act Data Needs, INL/EXT-19-55588, October.

Leidos, 2020, *Information for Construction of the Versatile Test Reactor at the Oak Ridge National Laboratory*, VTR-RPT-01, Rev. 0, November.

NRC (U.S. Nuclear Regulatory Commission), 1977, *Final Environmental Impact Statement on the Transportation of Radioactive Materials by Air and Other Modes*, NUREG-0170, Office of Standards Development, Washington, DC, December.

NRC (U.S. Nuclear Regulatory Commission), 1987, *Shipping Container Response to Severe Highway and Railway Accident Conditions*, NUREG/CR-4829, Office of Nuclear Regulatory Research, Washington, DC, February.

NRC (U.S. Nuclear Regulatory Commission), 2000, *Reexamination of Spent Fuel Shipping Risk Estimates*, NUREG/CR-6672, Office of Nuclear Material Safety and Safeguards, Washington, DC, March.

NRC (U.S. Nuclear Regulatory Commission), 2005, *Environmental Impact Statement on the Construction and Operation of a Proposed Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina*, NUREG-1767, Office of Nuclear Material Safety and Safeguards, Washington, DC, January.

Peterson, S., 2018, *WebTRAGIS: Transportation Routing Analysis Geographic Information System User's Manual*, ORNL/TM-2018/856, Oak Ridge National Laboratory, Oak Ridge, Tennessee, May.

Saricks, C., and M. M. Tompkins, 1999, *State-Level Accident Rates for Surface Freight Transportation: A Reexamination*, ANL/ESD/TM-150, Argonne National Laboratory, Energy Systems Division, Center for Transportation Research, Argonne, Illinois, April.

SRNL (Savannah River Nuclear Laboratory), 2020, Conceptual Assessment of VTR Add-on Processing Capability, SRNL-TR-2020-00171, Rev 2., Aiken, South Carolina, July 22.

SRNS (Savannah River Nuclear Solutions), 2012, *Surplus Plutonium Disposition Supplemental Environmental Impact Statement* Data Call Response, Aiken, South Carolina.

SRNS (Savannah River Nuclear Solutions), 2020, Savannah River Site Data Call for the Versatile Test Reactor Fuel Fabrication Facility, SRNS-RP-2020-00286, Rev. 2, Aiken, South Carolina, July 22.

UMTRI (University of Michigan Transportation Research Institute), 2003, *Evaluation of the Motor Carrier Management Information System Crash File, Phase One*, UMTRI-2003-6, Center for National Truck Statistics, Ann Arbor, Michigan, March.

Weiner, R. F., D. Hinojosa, T. J. Heames, C. Ottinger Farnum, and E. A. Kalinina, 2013, RADTRAN 6/RadCat 6 User Guide, Accession No. ML14286A092, SAND2013–8095, Sandia National Laboratories, Albuquerque, New Mexico.

Weiner, R. F., K. S. Neuhauser, T. J. Heames, B. M. O'Donnell, and M. L. Dennis, 2014, RADTRAN 6 Technical Manual, SAND2013-0780, Accession No. ML14286A085, Sandia National Laboratories, Albuquerque, New Mexico.

WIPP (Waste Isolation Pilot Plant), 2006, *Waste Isolation Pilot Plant Radiation Safety Manual*, WP 12-5 Rev. 10, Carlsbad, New Mexico, May 22.

Yuan, Y. C., S. Y. Chen, B. M. Biwer, and D. J. LePoire, 1995, *RISKIND - A Computer Program for Calculating Radiological Consequences and Health Risks from Transportation of Spent Nuclear Fuel*, ANL/EAD-1, Argonne National Laboratory, Argonne, Illinois, November.

APPENDIX F
TRANSPORT AND MANAGEMENT OF PLUTONIUM FROM FOREIGN COUNTRIES

APPENDIX F
TRANSPORT AND MANAGEMENT OF PLUTONIUM FROM FOREIGN COUNTRIES

F.1 Introduction

This appendix presents an assessment of the human health risks that could result from transporting plutonium materials from Europe (i.e., United Kingdom [UK] and France) to support reactor fuel production for the Versatile Test Reactor (VTR). The reactor fuel production would occur either at the Savannah River Site (SRS) or at the Idaho National Laboratory (INL). Given the projected processing efficiency of the VTR fuel fabrication (SRNL 2020), the VTR operation would require up to 34 metric tons of plutonium source materials over a 60-year span.[1] The United States has an excess plutonium inventory of more than 50 metric tons (GAO 2019) that is managed by the U.S. Department of Energy (DOE) and the National Nuclear Security Administration (NNSA). The excess DOE and NNSA plutonium would be sufficient to meet fueling needs for the VTR lifetime operation of 60 years. However, if that material cannot be made available or to supplement the domestic supply options, DOE has identified other potential sources from Europe.

This appendix considers that the plutonium would be transported by ship from the aforementioned countries in Europe to a U.S. seaport of entry. From the port of entry, the plutonium would be transported to SRS. Depending on DOE's decision on the location of feedstock preparation and fuel fabrication activities, the plutonium would remain at SRS for processing or be transported from SRS to the INL Site.

NNSA has prepared multiple environmental analysis documents (environmental impact statements [EISs], environmental assessments, and supplement analyses) for transporting various radioactive materials from foreign countries to the United States. Examples of these evaluations include the *Environmental Assessment for the Proposed Interim Storage at the Y-12 Plant, Oak Ridge, Tennessee, of Highly Enriched Uranium Acquired from Kazakhstan by the United States* (DOE/EA-1006) (DOE 1994), the *Environmental Assessment for the Transportation of Highly Enriched Uranium from the Russian Federation to the Y-12 National Security Complex and Finding of No Significant Impact* (DOE/EA-1471) (DOE 2004), the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (FRR SNF EIS) (DOE/EIS-0218) (DOE 1996a), and the *Environmental Assessment for the Gap Material Plutonium – Transport, Receipt, and Processing* (Gap Material Plutonium EA) (DOE/EA-2024) (DOE 2015b). The analyses and actions described in this appendix are consistent with the example documents; the current analysis draws on the discussions and analyses in the FRR SNF EIS and Gap Material Plutonium EA.[2]

F.2 Scope

This appendix evaluates the potential environmental impacts from transporting plutonium across the global commons to the United States. Transferring packages of plutonium onto transporters at the port

[1] This is an upper estimate based on a driver fuel production efficiency of about 73 percent for fabrication without feedstock preparation. As the production efficiency improves, the need for the feedstock plutonium would be reduced.

[2] In 2004, DOE established the Global Threat Reduction Initiative, now called Material Management and Minimization, a vital part of the U.S. national security strategy of preventing the acquisition of nuclear materials by any organizations for the use in weapons of mass destruction. As part of the initiative, NNSA analyzed and implemented several activities to manage threats by removing and disposing excess weapon-usable and radiological materials. As the program evolved, NNSA recognized that there were certain materials that were not addressed by existing initiatives. These materials were called gap materials and included some plutonium.

of entry is also evaluated here. The impacts of transporting plutonium packages within the United States are not discussed in this appendix. However, they are explicitly evaluated in Appendix E.

Sources of Plutonium Materials

Two potential sources of plutonium materials suitable for VTR reactor fuel production have been identified (INL 2020). These include inventories of plutonium separated from spent nuclear fuel (SNF) from the magnesium alloy (Magnox) reactors and the advanced gas-cooled reactors in the UK, and plutonium separated from SNF from pressurized water reactors in France. In the UK, there are about 140 metric tons of separated plutonium (of which about 110 metric tons is UK owned) stored at the Sellafield nuclear site (NDA 2019). In France, there are about 81 metric tons of plutonium in oxide form (of which 65.4 metric tons are French owned). The plutonium is mainly stored at the La Hague used fuel reprocessing facility.

Actions in the Global Commons

The scope of the analysis essentially begins when the conveyance for transporting the plutonium material to the United States enters the global commons. In this analysis, the global commons is the ocean outside the territorial waters of a country.

Transport by Ship

This appendix analyzes transportation of plutonium material by ship across the global commons to a U.S. seaport (Joint Base Charleston-Weapons Station in South Carolina). Marine transport of plutonium would be conducted using chartered, exclusive-use ships,[3] in compliance with international and national transportation standards. To make efficient use of resources, the chartered ships may transport plutonium from either the UK or France or from both countries.

Ground Transport to the DOE Sites (Savannah River or Idaho National Laboratory Sites)

This EIS analyzes the ground transport of plutonium materials in specially design transporters from the Joint Base Charleston-Weapons Station to SRS or the INL Site. The analysis includes the potential impacts of transferring plutonium from the ship to the transporters.

Receipt at the DOE Sites (Savannah River or Idaho National Laboratory Sites)

Activities at DOE sites to receive the plutonium would include unloading the packages of plutonium, repackaging as needed to meet storage requirements, and moving the packages to a storage location.

Storage and Processing

Storage would be temporary, pending processing of the plutonium to prepare it as feedstock for the fabrication of VTR driver fuel. The processing would include removal of impurities (polishing) in the plutonium, especially removal of the ingrowth of americium-241 to a level suitable for the VTR fuel; i.e., an americium concentration of less than 1 weight percent (INL 2019).

F.3 Description of Activities

DOE is considering the use of existing supplies of reactor-grade plutonium that is currently available in the UK and France, as an option to the domestic supply, for use in VTR fuel production. This appendix conservatively evaluates the transport of up 34 metric tons of plutonium from either UK or France, or both countries, to support the lifetime operation of the VTR. The action is to transport sufficient plutonium materials in each shipment to support bi-annual operation of the VTR facility, which would lead

[3] Exclusive-use ships operate as chartered vessels and are not used for the transport of any other cargo other than the plutonium they are hired to transport.

to an estimate of 1.2 metric tons per transport. Plutonium transport would occur over a 60-year period, with a total of 29 shipments.

F.3.1 Shipments to the United States

Shipments of the needed plutonium materials to the United States would occur after (1) implementation of a contract or agreement between authorized representatives of the United States and the countries or nuclear facilities possessing the plutonium, (2) receipt of all data necessary to ensure safe handling and temporary storage, and (3) satisfactory resolution of any identified issues. At the foreign sites, the plutonium would be stabilized to meet the requirements of DOE-STD-3013 (DOE 2012) and placed into containers that are compatible with the requirements of the DOE SRS or INL storage facility. The containerized plutonium would be placed within packaging appropriate for the type and quantity of material, shipped to the United States,[4] and then to SRS, in compliance with requirements for safe transport of radioactive materials of the host country, the United States, and international organizations. These standards include the International Atomic Energy Agency (IAEA) Safety Standard Series Number SSR-6, *Regulations for the Safe Transport of Radioactive Material* (IAEA 2018), and 10 CFR Part 71, *Nuclear Regulatory Commission Regulations for Packaging and Transportation of Radioactive Materials*.

The mode of transport would be chartered and exclusive-use ships, which would deliver the plutonium to the seaport at Joint Base Charleston-Weapons Station, South Carolina (**Figure F–1**). The Joint Base Charleston-Weapons Station was selected for analysis in this *Versatile Test Reactor Environmental Impact Statement* (VTR EIS) because it serves as a seaport of entry for NNSA's Foreign Research Reactor Spent Nuclear Fuel (FRR SNF) Acceptance Program after an extensive analysis in the FRR SNF EIS (DOE 1996a). Its receipt of radioactive material has been analyzed in subsequent National Environmental Policy Act (NEPA) documents (e.g., DOE 2003b, 2006a, 2009, 2010, 2015b). The Joint Base Charleston-Weapons Station has an ongoing working relationship with DOE/NNSA, and the FRR SNF Acceptance Program is actively receiving shipments through this seaport. Since the program was established in 1996, over 60 SNF shipments have been received in the United States. Most of these shipments were received at the Joint Base Charleston-Weapons Station (NNSA 2013). The SNF casks have been offloaded from ships to trucks or rail cars and transported to DOE facilities (DOE 2009). In recent years, containers with gap material plutonium have also been received at the Joint Base Charleston-Weapons Station.

F.3.2 Packaging and Shipments

Transportation of plutonium would be conducted in accordance with national and international requirements for safety and safeguards or, if determined to be in the interest of national security, in accordance with approved exceptions to those requirements. The packaging used for plutonium transport would need to be acceptable to both the host country and the United States, meaning that packaging for which a certificate of compliance has been issued in one country would have to be accepted by a competent authority of the other country. In general, individual countries' regulations conform to the IAEA *Regulations for the Safe Transport of Radioactive Material* (IAEA 2018), thereby facilitating acceptance of certified packaging by another country.

[4] Typically, the country shipping the plutonium would be responsible for arranging transport packaging and loading the plutonium into transport vehicles; complying with safety and security requirements; coordinating with local and national officials; obtaining export approvals; and making any needed transit arrangements with countries through whose territorial waters transport ships may pass.

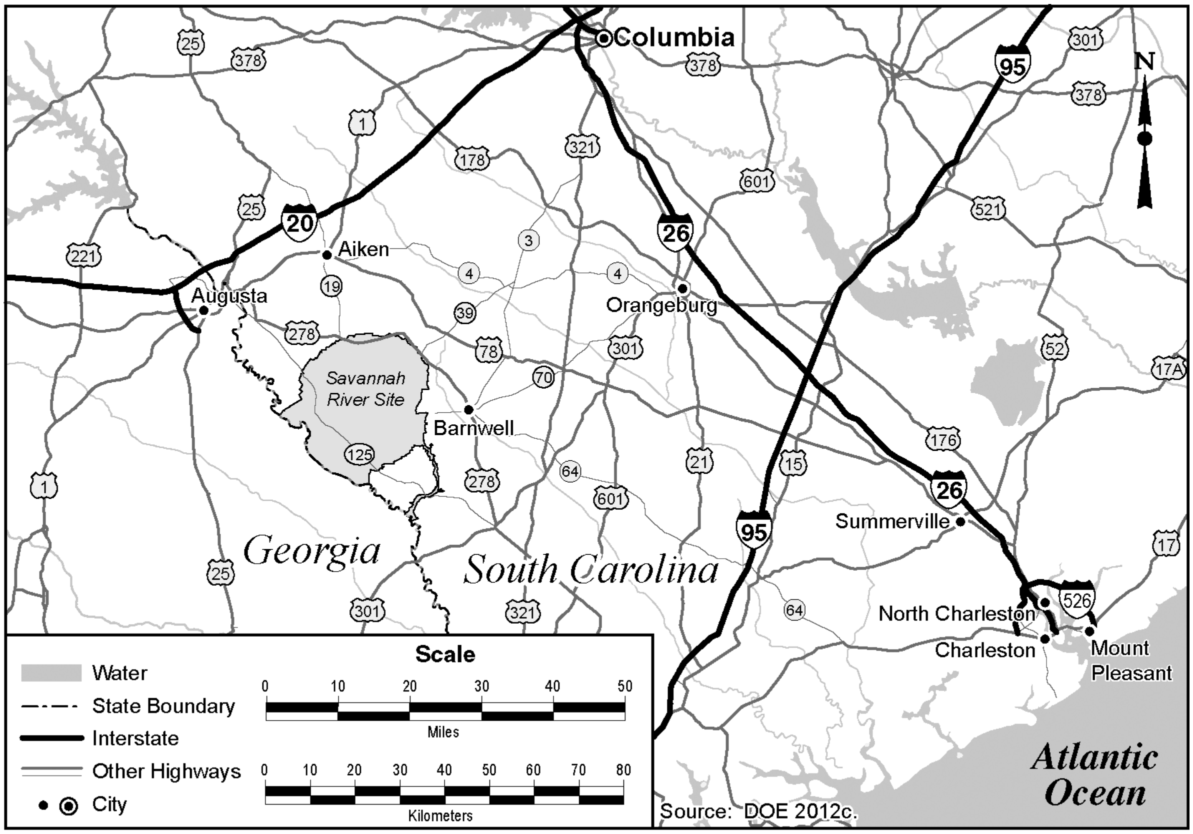

Figure F–1. Locations of the Joint Base Charleston-Weapons Station and Savannah River Site Overland Transport to Savannah River Site

All plutonium would be shipped using Type B packaging. Type B packaging must be designed and tested to withstand both normal transport and accident conditions.[5] Currently, there is only one representative Type B packaging[6] than can be used both internationally and within the United States. This packaging, the Model 9975, has been used in the United States for several years (DOT 2018:CoC USA/9975/B(M)F-96).

Model 9975 packaging (**Figure F–2**) includes an outside shell consisting of a stainless-steel, 35-gallon drum with a flange at the top for fasteners. Model 9975 packaging can hold a single container, composed of nested inner and outer stainless steel containers, of plutonium that has been stabilized pursuant to the requirements of DOE-STD-3013 (DOE 2012). One configuration housing welded containers meets DOE's standard for long-term plutonium storage (DOE 2012). A second configuration housing non-welded containers is used for interim storage. Containers of plutonium are secured in the package within primary containment vessels and secondary containment vessels that are surrounded by lead shielding and insulating material.

[5] Normal transport conditions, which may result in a package being subjected to heat, cold, vibration, changes in pressure, or other possible occurrences (e.g., being dropped, compressed under a weight, sprayed with water, or struck by objects), must not result in loss of function (e.g., containment, shielding, continuance of sub-criticality). With respect to accident conditions, there must be no substantial loss of function of the package after being subject to a series of tests that are conducted sequentially. These tests simulate being dropped from 30 feet onto an unyielding surface; being crushed or punctured; being exposed to a high heat (a temperature of at least 1,475 degrees Fahrenheit, as from a fire) for 30 minutes; and being immersed in water.

[6] In international and U.S. regulatory nomenclature, the term "package" means the packaging together with its radioactive contents as presented for transport. The term "packaging" means the assembly of components necessary to ensure compliance with packaging requirements. It may consist of one of more receptacles; absorbent materials; spacing structures; thermal insulation; radiation shielding; service equipment for filling, emptying, venting, and pressure relief; and devices for cooling or absorbing mechanical shocks.

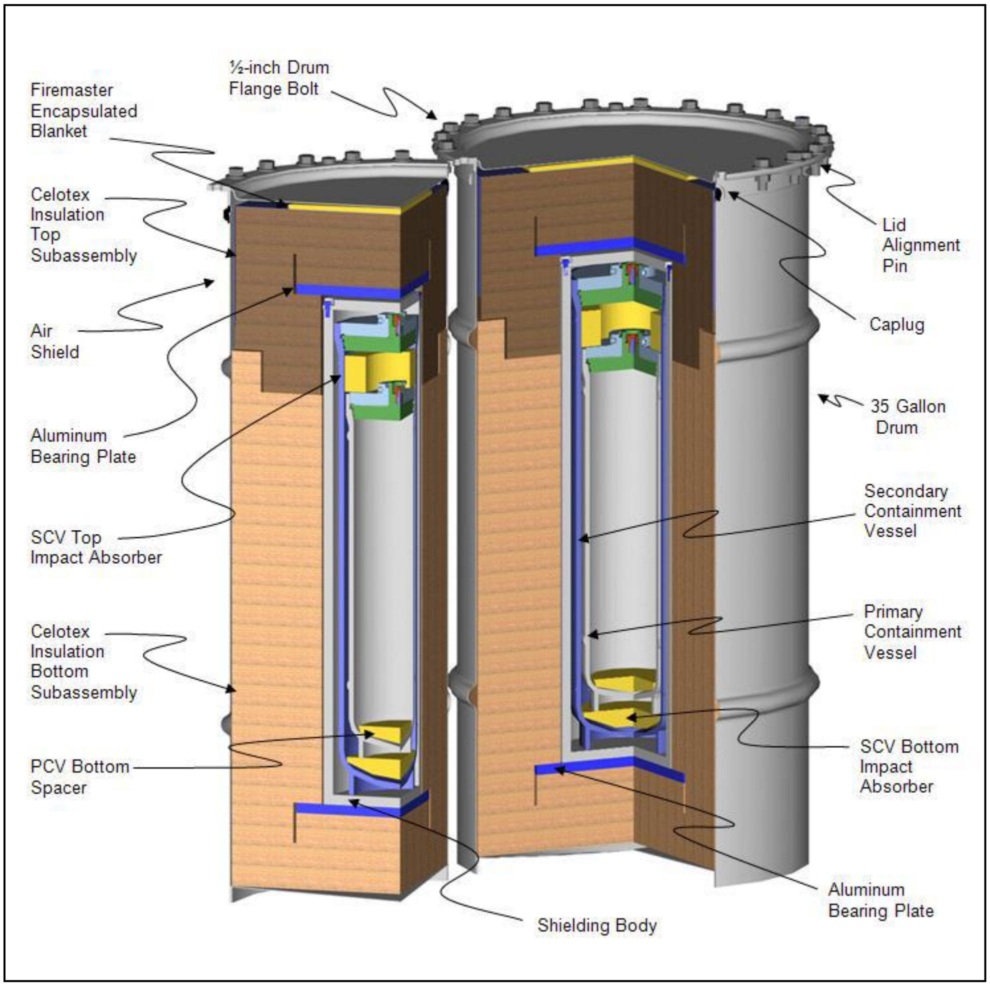

Figure F–2. Model 9975 Shipping Package

F.3.3 Ship Transport

At least 180 days before the tentative shipping date for transporting plutonium to the United States, DOE would establish a contract or agreement between DOE, representing the U.S. Government, and authorized representatives of the country or nuclear facility possessing the plutonium. Before shipment, teams of DOE or authorized contractor personnel would conduct foreign site visits that would include representative material examinations and facility and infrastructure assessments.

At the foreign sites, plutonium stabilized to meet the requirements of DOE-STD-3013 (DOE 2012), or similar characteristics meeting the requirements of the selected packaging, would be placed into containers compatible with the requirements of the U.S. storage facility. The containerized plutonium would be placed within packaging appropriate for the type and quantity of material. The packaged plutonium would be transported within the foreign countries to seaports of embarkation in compliance with local standards for safety and security. At the nuclear facility or seaport, the packages of plutonium would be securely mounted on pallets that would be secured within one or more International Organization for Standardization (ISO) shipping containers (ISO containers). Securing the packages on pallets facilitates transfer of the packages into and securing the packages within the ISO containers, removal from the ISO containers at the Joint Base Charleston-Weapons Station, and loading into specially design transporters for shipment to SRS. The ISO containers would be hoisted onto the transport ship at the foreign port and stowed securely within the ship's hold (see **Figure F–3**). DOE or contractor personnel may be present to facilitate arrangements and inspect packaging and loading operations.

Figure F–3. ISO Containers Secured within the Hold of a Ship

The number of packages placed within an ISO container may vary. Considering criticality safety requirements, the physical dimensions of the packages and their groupings on pallets, the typical dimensions of ISO containers and overland transport vehicles, and worker radiation protection, each ISO container would contain up to 25 Model 9975 packages. Each expected shipment would consist of 15 ISO containers.

The chartered ship would be certified to meet the requirements of the *International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes on Board Ships* (INF Code) (SOLAS 1999). The requirements differ depending on the ship's INF Code classification. An INF Class 1 vessel may carry INF cargo with an aggregate activity of less than 108,000 curies. An INF Class 2 vessel may carry irradiated nuclear fuel or high-level radioactive waste (HLW) with an aggregate radioactivity of less than 54 million curies or plutonium with an aggregate radioactivity less than 5.4 million curies. An INF Class 3 vessel may carry irradiated nuclear fuel, HLW, or plutonium with no restrictions on aggregate radioactivity. Design and operational requirements for the three INF ship classes are addressed in a graded manner. They include those for vessel stability after damage, fire protection, temperature control of cargo spaces, structural strength of deck areas and support arrangements, cargo securing arrangements, electrical supplies, radiological protection equipment, ship management, crew training, and emergency plans (WNTI 2007).

Prior to each shipment, a threat assessment would be conducted in accordance with a security plan developed for the specific shipment. If determined necessary, armed security personnel could be onboard the transport vessel or an escort ship.

Although members of the general public would not be exposed to radiation during loading activities or during transport across the global commons to the United States, some members of the ship crew could be exposed to external radiation. Radiation doses potentially experienced by the crew would depend on the travel time to the Joint Base Charleston-Weapons Station, the loading and placement of ISO containers within the ship's hold, any material or cargo present that could provide shielding after stowage, and crew activities during loading and transit.

For shipments from the European countries (UK or France), a transport time of 15 days was assumed, based on the distances from European ports evaluated in the FRR SNF EIS (DOE 1996a) and assuming an average cruising speed of 12 knots consistent with experience with shipping FRR SNF (DOE 1998).

The number of crew members and their activities during loading operations reflect those addressed in the FRR SNF EIS (DOE 1996a) and Gap Material Plutonium EA (DOE 2015b). Ship crew members performing loading operations would be assisted by radiation protection personnel to reduce the potential for excessive radiation exposures.

While at sea, some of the crew members would enter the hold and be in the vicinity of the ISO containers to inspect the cargo and ensure it remains securely stowed (e.g., check the tightness of the cargo tie-downs). This activity would occur daily and represents the largest potential for radiation exposure to crew members. The radiation dose received by these crew members would depend on the levels of radiation emitted by the ISO containers, the number and placement of the ISO containers within the ship's hold (for shipments containing more than one ISO container), the inspection times, and the distance maintained from the ISO container during inspections. The external radiation rates for the shipment packages were assumed to be the regulatory limit of 10 millirem per hour at 2 meters. In reality, the dose rate is expected to be well below the regulatory limit. Because the vessel used for plutonium shipment would be exclusive-use, crew members performing the inspections would understand radiation safety principles, and unauthorized crew members would be excluded from the immediate area of the radioactive cargo.

Before entering the Joint Base Charleston-Weapons Station, a vessel carrying plutonium materials would communicate with appropriate personnel at the seaport to coordinate port entry and docking activities. Measures would be taken to ensure safety and security during the passage through the port entrance channel and travel within port reaches or turning basins. A pilot may board the vessel to assist the passage to the designated wharf. Escort vessels or tugs may also assist the passage.

F.3.4 Ship to Truck Transfer at the Joint Base Charleston-Weapons Station

At Joint Base Charleston-Weapons Station, specially designed transporters would be staged to await the arrival of the ship. In accordance with the security plan, if necessary, additional security would be provided at the dock during transfer of the cargo from the ship to the transporters. Upon arrival of the ship, authorized workers, assisted by ship crew members, would enter the hold; remove the tie-downs securing the ISO containers for the ocean voyage; attach rigging; remove the ISO containers from the hold using a crane; and place and secure the ISO containers on the transporters at the dock area. During incident-free transfer of ISO containers to the transporters, authorized personnel performing or assisting in the transfer would be exposed to external radiation from the containers. Members of the public and other workers at Joint Base Charleston-Weapons Station would be restricted from the vicinity of the unloading and transfer operations. Therefore, the public and other workers would not be exposed to radiation during incident-free unloading and package transfer activities.

F.3.5 Overland Transport to SRS

The plutonium-containing ISO containers received at Joint Base Charleston-Weapons Station would immediately be transported to SRS. The transport would be in a caravan-like configuration of 15 transporters with security adequate to prevent unauthorized removal of cargo. Because of the short travel distance between Joint Base Charleston-Weapons Station and SRS (less than 150 miles), no refueling or rest stops are expected.

F.3.6 Plutonium Receipt, Processing, Storage, and Disposition

At SRS, the plutonium would be received, temporarily stored at one of the K Area Complex locations (for example, the K Area Material Storage Area, illustrated in **Figure F–4**), and prepared as needed for the VTR fuel production, if feedstock preparation occurred at SRS. If feedstock preparation were to occur at the INL Site, the plutonium packages within the ISO containers would be temporarily stored at SRS and be reconfigured for transport to the INL Site, using DOE's Secure Transportation Asset (STA) specially designed transporters.

Figure F–4. Storage of Surplus Plutonium at the K Area Complex

F.4 Affected Environment

The environments that may be affected by the activities in Section F.3 include (1) the global commons that would be traversed by ships carrying the plutonium materials (Atlantic Ocean), (2) the seaport (Joint Base Charleston–Weapons Station) at which ships would dock, (3) the overland transportation routes, and (4) the location in the United States where the plutonium would be used. Descriptions of the affected environment are incorporated by reference from the Gap Material Plutonium EA (DOE 2015b) and are not repeated here. The Gap Material Plutonium EA provides detailed descriptions for the global commons, Atlantic Ocean, and the U.S. seaport of entry, Joint Base Charleston-Weapons Station. Appendix E of this EIS provides detailed descriptions of the overland transport routes and Chapter 3 describes the affected environments at SRS and the INL Site.

F.5 Analysis and Discussions

This section analyzes the environmental consequences of transporting plutonium from foreign countries to the United States, including impacts under incident-free and accident conditions from ship transport to a United States seaport (the Joint Base Charleston-Weapons Station). The impacts of the subsequent ground transport to SRS are provided in Appendix E of this EIS.

Consistent with Executive Order 12114, *Environmental Effects Abroad of Major Federal Actions*, this appendix does not address impacts from activities involving plutonium materials within the host countries. Countries shipping the plutonium materials would be responsible for complying with applicable laws and regulations associated with activities occurring within their borders.

The plutonium composition mixture could represent a range of characteristics with respect to the relative quantities of plutonium isotopes and americium with ingrowth isotopes of 25- to 40-year old materials. Primarily because of an increase in americium-241 over time (resulting from radioactive decay of plutonium-241), the older plutonium would present the largest health risk. The radionuclide distribution and specific activities of the European plutonium used for the analysis are presented in **Table F–1** (INL 2020).

Table F–1. Assumed Composition of Plutonium Material from United Kingdom and France

	United Kingdom	*France*
Radionuclide	*Mass Fraction (percentage) (grams per gram of plutonium)*	*Mass Fraction (percentage) (grams per gram of plutonium)*
Plutonium-238	0.22	2.10
Plutonium-239	69.42	62.00
Plutonium-240	26.96	25.00
Plutonium-241	0.86	4.00
Plutonium-242	2.54	6.60
Total Plutonium	100	99.7
Americium-241 [a]	4.68	8.40

[a] The americium fraction is per plutonium + americium-241.
Note: The mass fractions in the reference document are in the 2040 time frame for the UK fuel and at 25 years after plutonium separation for the French fuel.
Source: INL 2020:INL/MIS-20-57910 Rev. 1.

The 34 metric tons of plutonium evaluated in this EIS were assumed to be transported from either UK or France, or both countries, in 1.2 metric tons shipments, over 60 years of VTR operation. The plutonium would be shipped in Model 9975 packages. Twenty-five Model 9975 packages would be placed within an ISO container, a 20-foot standard shipping container. The quantity of plutonium actually placed within a package would depend on operational factors such as the total quantity of shipped material, the isotopic distribution of the plutonium, its chemical form (e.g., metal vs. oxide), and the presence of impurities. Depending on these operational factors, the quantity of plutonium shipped within a given package could range from levels less than the authorized capacity to levels approaching the maximum capacity.

Consistent with previous analysis in the *Surplus Plutonium Disposition Supplemental Environmental Impact Statement* (DOE 2015a), it was assumed that the average plutonium content within 25 Model 9975 packages would be about 80 kilograms or about 73 percent of the authorized capacity. Given this assumption, a 1.2 metric tons plutonium per shipment would require 15 ISO containers. Assuming all packages are filled to about 73 percent of authorized capacity, there would 375 Model 9975 packages per shipment.

F.5.1 Impacts on the Global Commons

F.5.1.1 Human Health Impacts from Ship Transport under Normal Operations

This section addresses incident-free human health impacts from shipping plutonium across the global commons. The general public would not receive a radiation dose from incident-free transport of plutonium by ocean vessel. However, radiological impacts would be experienced by the crews of the ships carrying the material from exposure to radiation during loading and off-loading the ISO containers and during daily inspections of cargo. The radiological impacts from cargo inspections would depend on the duration of the voyages. As discussed in Section F.3.3, based on the distances provided in the FRR SNF EIS (DOE 1996a), a 15-day voyage for a shipment from Europe is assumed.

As explained in Section F.3.3, operational procedures for loading and unloading ISO containers containing plutonium, and for cargo inspections during transport, would be the same as those in the FRR SNF EIS for ocean shipment of FRR SNF (DOE 1996a). Consistent with the FRR SNF EIS (DOE 1996a), the assumed crew duties are summarized in **Table F–2**. As shown, a chief mate, mate on watch, bosun, and two seamen are assumed to be exposed to radiation while loading the ISO containers onto the ship, and while unloading the ISO containers at the destination seaport (the Joint Base Charleston-Weapons Station). Consistent with the FRR SNF EIS (DOE 1996a), when loading or unloading ISO containers for maximum expected shipments, the crew members are assumed to be exposed to other ISO containers in the ship's hold. Doses received by each crew member are assumed to be the same as those evaluated in the FRR SNF EIS. This assumption is based on the same loading and unloading operations would be performed as those evaluated in the FRR SNF EIS, and the radiation levels at the exterior of the ISO containers are assumed to be at the regulatory limit, the same as evaluated in the FRR SNF EIS (DOE 1996a).

Table F–2. Assumed Crew Duties for Ocean Transport of Plutonium Materials

Crew Member	*Ship Loading Operations*	*Daily Cargo Inspections*	*Ship Unloading Operations*
Chief Mate	X	X	X
Mate on Watch	X		X
Bosun	X	X	X
Seaman(2)	X		X
Engineer		X	

The chief mate, bosun, and engineer are all assumed to participate in daily inspections of the cargo. Each of these crew members is assumed to perform one cargo inspection per day during each assumed 8-hour shift (three inspections total per day). For maximum expected shipments, it was assumed that crew members performing inspections on one ISO container would be exposed to radiation from other stowed ISO containers. The configuration of the ISO containers within a hold is a function of the hold characteristics (length, width, and height) with respect to those of the stowed containers. In the absence of specific information and consistent with the regulations on safe transport of radioactive materials (IAEA 2018), the assumptions on the inspection times and the distances at which the actions are carried out are based on information in the FRR SNF EIS (DOE 1996a). Each inspection was assumed to require 20 minutes per hold. Because there would be 15 ISO containers per shipment in a purpose-built ship (PNTL 2020) with five holds, the dose values provided in the FRR SNF EIS (DOE 1996a) were adjusted, accordingly.

The estimated doses per shipment to individuals and all involved crew members are shown in **Table F–3**.

Table F–3. Per-Shipment Crew Doses and Risks for Transporting Plutonium via Chartered Vessel

Impact	*Chief Mate*	*Mate on Watch*	*Bosun*	*Seaman* [a]	*Engineer*	*Total*
Plutonium Shipment – 15 ISO Containers per 15-Day Voyage						
Maximum dose (millirem) [b]	400	80	400	140	260	1,430
LCF risk [c]	2×10^{-4}	5×10^{-5}	2×10^{-4}	9×10^{-5}	2×10^{-4}	9×10^{-4}

ISO = International Organization for Standardization; LCF = latent cancer fatality.

[a] For each voyage, two seamen would receive radiation doses from the plutonium cargo; the doses presented are per seaman.

[b] Maximum doses were determined assuming that the radiation levels at the surfaces of all ISO containers correspond to the regulatory limit (10 millirem per hour at 2 meters from the ISO container surface).

[c] Risks were determined assuming a factor of 0.0006 LCFs per rem and are presented using one significant figure (DOE 2003a).

The results in Table F–3 show doses greater than 100 millirem per voyage. These results are a function of the assumptions in the FRR SNF EIS (DOE 1996a) regarding daily inspection and dose rates. These assumptions are conservative for the current exclusive-use ships and their radiation protection and inspection practices. The radiation doses associated with at-sea inspections could be reduced by minimizing the amount of time required for inspections and by maintaining an appropriate distance from the ISO containers, consistent with inspection requirements.

Notwithstanding these caveats, it is conceivable, as indicated in Table F–3, that some members of the crew who are not radiation workers could receive a radiation dose exceeding 100 millirem in a year. DOE would extend the program described in the mitigation action plan for FRR SNF (DOE 1996b)[7] or implement a similar program as that for the gap material plutonium shipments (DOE 2015b).

F.5.1.2 Human Health Impacts from Potential Shipping Accidents

There is a small probability of an accident on the open seas involving a vessel containing plutonium materials. There is an even smaller probability that the accident would be severe enough to result in release of radioactive material (e.g., a collision with another ship that crushes packages of plutonium, followed by a fire sufficient to release radioactive material as respirable particles to the atmosphere). The probability of a severe port accident that would result in the release of plutonium is 5×10^{-9} per ship arrival in port (DOE 1996a). The probability of this accident occurring in coastal waters or the open ocean is even lower (IAEA 2001). The probability is smaller than the probability that DOE considers for analysis of maximum reasonably foreseeable accidents (1×10^{-7} or 1 chance in 10 million) (DOE 2002). Therefore, the consequences of this accident are not evaluated in this VTR EIS. This severe port accident was analyzed in previous NEPA documents addressing shipment of radioactive materials (e.g., DOE 1996a, 2006a, 2009, 2010).

F.5.1.3 Other Impacts from Ship Transport

There would be no release of radioactive material under incident-free transport, meaning there would be no radiological impacts on the global commons, including impacts on marine biota and fisheries. If an incident were to occur (for example, a collision with another ship or foundering), environmental impacts could result. Packages of plutonium could rupture and be released into the ocean.[8] The response to, and potential impacts of, such an accident would be different, depending on the location and condition of the packages following the accident (DOE 1994, 2004). Packages that did not sink below about 660 feet could be located and recovered. Undamaged packages that sink deeper than about 660 feet could be breached by the pressure of the overlying water or by corrosion, which would release their contents. As discussed in the Gap Material Plutonium EA (DOE 2015b), a number of previous NEPA evaluations have considered the potential radiological impacts of a release of radioactive material from an accident at sea. The analyses concluded that some marine organisms directly exposed to radioactive material could receive large doses of radiation and that some loss of marine life would occur. They further concluded that because of the large volumes of water involved, mixing mechanisms, existing background radiation levels, and radiation-resistance of aquatic biota, the radiological impact on marine life would be localized and minor.

Additional discussion of potential impacts from ship transports are provided in the Gap Material Plutonium EA (DOE 2015b). As noted in that EA, there is a possibility that a ship transporting plutonium

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[7] Under the mitigation program applied to shipments of FRR SNF (DOE 1996b), DOE requires that its shipping contractor obtain radiation surveys of FRR SNF casks before shipment, and use these data to ensure that the estimated dose to any crew member does not exceed 100 millirem per year. DOE also maintains a database of the actual radiation surveys for each cask and shipment, and includes clauses in its shipping contracts to minimize the likelihood that any member of a ship's crew would be exposed to more than 100 millirem during a single year.

[8] For the 5-year period between 2010 and 2014, 22 large ship collisions were reported worldwide; approximately 5 per year (Allianz 2015). The frequency of serious ship collisions is estimated at about 3.9×10^{-8} per nautical mile (IAEA 2001).

for DOE or NNSA could strike and kill or injure a federally protected Atlantic coast species (e.g., North Atlantic right whale, loggerhead sea turtle, or manatee). However, the impact on these species is expected to be minimal due to the small number of shipments (less than one per year) and adherence to speed restrictions in coastal waters and port entrance channels.

F.5.2 Impacts at the Seaport of Entry –Joint Base Charleston-Weapons Station

F.5.2.1 Human Health Impacts under Normal Port Operations

Radiation doses at the seaport would be received by the ship's crew, as well as by port workers involved in removing the ISO containers from the vessels and placing the ISO containers on the dock or on the specially designed transporters. There would be no radiation doses received by members of the public from incident-free activities at the Joint Base Charleston-Weapons Station. Activities at the seaport would occur at a secure military base. Unauthorized personnel would be excluded from locations where the ISO containers would be removed from the vessel (see Section F.3.4).

The types of involved workers participating in transfer of the ISO containers from a ship to the dock at Joint Base Charleston-Weapons Station are assumed to be the same as those evaluated in the FRR SNF EIS (DOE 1996a) for receipt of FRR SNF. It is assumed that the ISO containers unloaded from a ship would be transferred to a trailer at the dock. Involved workers include those responsible for inspection of the delivered cargo, transferring the cargo to the dock (cargo handlers), and moving the ISO containers to a staging area (staging personnel). The same radiation doses for transfer of a single ISO container were assumed for these workers as those evaluated in the FRR SNF EIS, because the same basic port activities would occur (inspection, unloading, and staging). The radiation levels of the ISO containers were assumed to be at the regulatory limit, the same as those in the FRR SNF EIS (DOE 1996a). Given these assumptions, doses and risks from shipping 15 ISO containers of plutonium are presented in **Table F–4**.[9] No worker is expected to receive a dose exceeding 100 millirem, even if all shipments were to occur in a single year. The total dose among all workers is projected to be 0.20 person-rem, with no latent cancer fatalities (LCFs) associated with these doses (calculated values: 1×10^{-4}).

Table F–4. Incident-Free Impacts for Unloading 15 ISO Containers of Plutonium Materials from Chartered Ships [a, b]

	Maximally Exposed Worker		*Worker Population*	
***Risk Group* [c]**	***Dose (millirem)***	***Risk (LCF)* [d]**	***Dose (person-rem)***	***Risk (LCF)* [d]**
Inspectors (6)	20	1×10^{-5}	0.08	5×10^{-5}
Port Cargo Handlers (4)	7	4×10^{-6}	0.02	1×10^{-5}
Port Staging Personnel (5)	6	4×10^{-6}	0.07	4×10^{-5}
Maximum [e]	20	1×10^{-5}	NA	NA
Total	NA	NA	0.20	1×10^{-4}

LCF = latent cancer fatality; NA = not applicable; rem = roentgen equivalent man.

[a] ISO container surface dose rates were assumed to be at the regulatory limit (10 millirem at 2 meters from the container surface).

[b] These results are based on the conservative assumption that each voyage carries more than one ISO container, resulting in larger doses to port personnel because of the combination of radiation fields surrounding each of the ISO containers.

[c] Numbers in parentheses are the assumed numbers of exposed personnel in each risk group.

[d] LCF risks are based on 0.0006 LCFs per rem or person-rem and are presented using one significant figure (DOE 2003a).

[e] The highest dose and risk among the risk groups.

Note: Totals may not equal the sums of table entries due to rounding.

Source: DOE 1996a for per-container radiation dose values.

[9] Doses received by cargo handlers and staging personnel were based on the assumption that ISO container unloading activities would require 65 minutes per ISO container. Experience with the FRR SNF Acceptance Program suggests that the actual unloading time would be closer to 20 minutes per ISO container (DOE 2009). The less time required to unload the ISO containers, the smaller the dose received by cargo handlers and other involved personnel.

F.5.2.2 Human Health Impacts from Potential Accidents Involving Port Operations

There is a small probability of a port accident involving a vessel containing plutonium, and an even smaller probability that the accident would be severe enough to result in release of radioactive material (e.g., a collision with another ship that crushes packages of plutonium, followed by a fire sufficient to release radioactive material as respirable particles to the atmosphere). The probability of a severe port accident that would result in the release of plutonium is 5×10^{-9} per ship arrival in port (DOE 1996a). This is smaller than the probability that DOE considers for analysis of maximum reasonably foreseeable accidents (1×10^{-7}, or 1 chance in 10 million) (DOE 2002). The consequences of this accident were not evaluated in this VTR EIS.

Other accidents could also occur during port operations, ISO container unloading, container staging, and container loading on transporters. It is conceivable that, for example, an ISO container could be dropped onto the dock while being unloaded from a ship. Any potential human health risk to a worker from such hypothetical incidents would only be associated with the physical forces of contact and not from release of radioactive material. All plutonium would be shipped within Type B packages designed and constructed to meet hypothetical accident conditions of transport without release of the package contents. Package tests include being dropped from 30 feet onto an unyielding surface; being crushed or punctured; being exposed to a high heat as from a fire; or being immersed in water. These tests and the design and construction of the packages would exceed the forces on a package that could be imposed by a dropped-package scenario at the dock.

F.5.2.3 Other Impacts from Port Operations

Shipments of plutonium materials would not affect the volume of ship traffic into or out of the port area of Charleston, meaning the shipments would have little effect on resource areas such as water quality, marine life, or socioeconomics. No more than 29 ocean voyages are expected for the maximum plutonium need of 34 metric tons over a period of 60 years. Even if all voyages occurred in a single year, 29 ocean voyages would represent about 1 percent of the 1,944 large commercial vessel and cruise ship calls at the port of Charleston in 2011 (DOT 2013a, 2013b).[10]

Shipments of plutonium would use existing infrastructure, with no need for construction or modification of Joint Base Charleston-Weapons Station facilities and no land disturbance that could potentially affect land use, biological resources, cultural resources, or geologic media. Under incident-free transport conditions, there would be no release of radioactive material to air or water. Nonradioactive waste would not be generated beyond that associated with normal operation of ships and port facilities. No pollutants, including greenhouse gases, would be discharged to the air beyond those normally released during ship and port operations. No water would be withdrawn from or discharged to surface water or groundwater beyond that authorized for normal operation of ships and port facilities. Shipments of plutonium would not affect socioeconomic conditions at the seaport. Work would be accomplished using existing DOE, seaport, and contractor personnel.

Members of the public would be placed at no radiological risk during incident-free operations because a security perimeter would be established around the ship unloading and package transfer operations, and members of the public and unauthorized seaport personnel would be excluded from the perimeter. Because all members of the public would be thus protected from radiological risk, no disproportionately high and adverse radiological risks would occur among low-income and minority populations in the vicinity of the seaport.

[10] To reach the Joint Base Charleston-Weapons Station, all ships must travel up the Cooper River past the port of Charleston. The number of annual military vessel calls at the Joint Base Charleston-Weapons Station is classified.

F.5.3 Impacts from Receipt of Plutonium Materials at the Savannah River Site

It was assumed that plutonium transported to SRS would be received and temporarily stored at the K Area Complex, where the plutonium would be unloaded from ISO containers and material control and accountability measurements taken. The packages would be transferred on metal pallets to the designated storage location if feedstock preparation for reactor fuel production is to occur at SRS. If feedstock preparation is to occur at the INL Site, the plutonium packages would be unloaded from the ISO containers and reconfigured for transport to the INL Site in STA transporters (e.g., Safeguards Transporter).

All activities involving plutonium receipt would be conducted in accordance with established radiation safety procedures and standards. Administrative and technical controls would be implemented to ensure that radiation dose rates to workers would be monitored, maintained to levels within DOE standards and guidelines, and reduced to as low as reasonably achievable levels.

F.5.3.1 Impacts on Workers

Impacts on workers could result from receiving plutonium and placing it into storage. Worker doses from receipt of plutonium would be comparable to those of daily activities for the facility operations at SRS K Area Complex, as described in Chapters 3 and 4 of this VTR EIS.

F.5.3.2 Impacts on the Noninvolved Workers and the Public

All plutonium received at SRS would be contained within Type B packages, and there would be no releases to the environment during normal receiving activities. In addition, noninvolved workers and the public would not be in direct proximity to the storage packages. The K Area Complex is more than 5.5 miles from the SRS boundary. Therefore, there would be no radiological impacts on noninvolved workers and the public from incident-free plutonium receipt.

F.6 Intentional Destructive Acts

The plutonium to be used for fabricating VTR fuel represents a potential target for diversion or terrorist actions. The following discussion relates to such intentional destructive acts associated with the transport of plutonium to the United States and its use in the VTR and rendering it unusable for weapons production.

F.6.1 Intentional Destructive Acts on the Global Commons

Maritime areas where acts of terrorism or piracy are more likely would be avoided to the extent practical. Ships passing thorough these areas would be provided with additional security as necessary. About 80 percent of all acts of piracy, for example, take place in the territorial waters of sovereign nations. In 2007, the locations with the most incidents of piracy included waters near Indonesia, Nigeria, and Somalia (Petretto 2008). Transport of material from European countries would not travel near these countries. If an intentional destructive act were to occur at sea, potential impacts would primarily be to onboard personnel. Potential impacts could range from fatalities associated with an explosion or drowning to lesser impacts from radiation exposure to untrained or uninformed personnel in the immediate vicinity of the transportation packages containing plutonium. Potential radiological impacts on people in the proximity of this accident would be similar to the analysis of intentional destructive acts during overland transport, as discussed below in Section F.6.2.

F.6.2 Intentional Destructive Acts in the United States

In accordance with DOE NEPA guidance (DOE 2006b), an analysis was performed in a classified appendix to the *Environmental Assessment for the U.S. Receipt and Storage of Gap Material – Plutonium and Finding of No Significant Impact* (DOE 2010) to consider the potential impacts of intentional destructive acts for activities related to plutonium transport. A range of scenarios involving the release of plutonium was evaluated in that EA. Each scenario involves an action by intruders during the transportation of packages within the United States. The analysis of intentional destructive acts performed for the *Environmental Assessment for the U.S. Receipt and Storage of Gap Material – Plutonium and Finding of No Significant Impact* is applicable to the action in this VTR EIS and is, therefore, incorporated by reference.

F.6.3 Mitigation of Intentional Destructive Acts

The likelihood of an intentional destructive act during transport of gap material plutonium is minimized by the security measures that would be taken to reduce knowledge of and access to the shipments. In the aftermath of the September 11, 2001 attacks, DOE, the U.S. Department of Defense (DOD), and the U.S. Department of Homeland Security implemented measures to minimize the risk and consequences of potential terrorist attacks on DOE and DOD facilities and U.S. ports. Safeguards applied to protecting facilities that contain nuclear material involve a dynamic process of enhancement needed to meet evolving threats. DOE and DOD continually re-evaluate security scenarios involving intentional destructive acts to assess potential vulnerabilities and identify improvements to security procedures and response measures. Security at these facilities is a critical priority for both DOE and DOD, which continue to identify and implement measures to deter attacks and defend against them. DOE and DOD maintain a system of regulations, orders, programs, guidance, and training that forms the basis for maintaining, updating, and testing site security to preclude and mitigate any postulated terrorist actions (Brooks 2004; DHS 2006; Pub. L. 107-296, 33 CFR Part 165, and 33 CFR Part 334).

F.7 Cumulative Impacts

Council on Environmental Quality regulations (40 CFR Parts 1500-1508) define cumulative impacts as effects on the environment that result from implementing the Proposed Action or any of its alternatives when added to other past, present, and reasonably foreseeable future actions, regardless of what agency or person undertakes the other actions (40 CFR 1508.7). The cumulative impacts of an action can be viewed as the total impacts on a resource, ecosystem, or human community of that action and all other activities affecting that resource irrespective of the source. This analysis of cumulative impacts emphasizes public health and safety impacts associated with the transport of plutonium for use as VTR driver fuel.

Transport to U.S. Seaports. Each year, there are several million worldwide shipments of radioactive materials using trucks, trains, ocean vessels, aircraft, and other conveyances, including large numbers of shipments across the global commons. Shipments of plutonium to the United States for use as VTR driver fuel would represent only a fraction of these worldwide shipments.

Collective radiation doses and risks to crews and populations for incident-free transport of 34 metric tons of plutonium from foreign countries to U.S. seaports are summarized in **Table F–5**. This table also lists the doses and risks to ship crews and dock workers from shipment of: (1) 100 kilograms of gap material plutonium by ocean vessel, as evaluated in the 2010 *Environmental Assessment for the U.S. Receipt and Storage of Gap Material – Plutonium* (DOE 2010); (2) 5 metric tons of highly enriched uranium (HEU) by ocean vessel, as evaluated in the 2006 *Supplement Analysis for the Air and Ocean Transport of Enriched Uranium between Foreign Nations and the United States* (DOE 2006a); (3) shipment of 900 kilograms of gap material plutonium in the Gap Material Plutonium EA (DOE 2015b); and (4) shipment of FRR SNF by ocean vessel under the FRR SNF Acceptance Program. Some personnel could be exposed to radiation from shipments of plutonium materials, as well as from shipment of FRR SNF or HEU in unirradiated nuclear fuel. Doses thus received as part of plutonium shipments would be mitigated, as discussed in Section 4.9 of the Gap Material Plutonium EA (DOE 2015b).

Table F–5. Cumulative Radiation Doses and Risks for Incident-Free Marine Transport of Radioactive Shipments to U.S. Seaports

Risk Receptor (scenario)	***Radiation Dose (person-rem)***	***Risk (LCF)*** [a]
Ship crew, 900 kilograms of gap material plutonium (Proposed Action) [b, c]	2.8 to 4.1	2×10^{-3}
Dock handlers, 900 kilograms of gap material plutonium (Proposed Action) [b, c]	0.20 to 0.26	1×10^{-4} to 2×10^{-4}
Ship crew, 100 kilograms of gap material plutonium [b, c, d]	1.4	8×10^{-4}
Dock handlers, 100 kilograms of gap material plutonium [b, c, d]	0.67	4×10^{-4}
Ship crew, 5,000 kilograms of unirradiated HEU [e]	0.030	2×10^{-5}
Dock handlers, 5,000 kilograms of unirradiated HEU [e]	0.13	8×10^{-5}
Ship crew, FRR SNF [f]	75.4	5×10^{-2}
Dock handlers, FRR SNF [f]	8.2	5×10^{-3}
Ship crew, 34 metric tons of plutonium from Europe **[b, g, h]**	**40.6**	$\mathbf{2 \times 10^{-2}}$
Dock handlers, 34 metric tons of plutonium from Europe **[b, g, h]**	**4.9**	$\mathbf{3 \times 10^{-3}}$
Totals	**134 to 135**	$\mathbf{8 \times 10^{-2}}$

FRR = foreign research reactor, HEU = highly enriched uranium, LCF = latent cancer fatality, rem = roentgen equivalent man; SNF = spent nuclear fuel.

[a] Risks were determined using a dose-to-risk factor of 0.0006 LCFs per person-rem and are presented using one significant figure (DOE 2003a).

[b] Conservatively assumes a surface radiation dose at International Organization for Standardization container or package array surfaces of 10 millirem per hour at 2 meters.

[c] The 2015 *Environmental Assessment for Gap Material Plutonium—Transport, Receipt, and Processing and Finding of No Significant Impact* (DOE/EA-2024) (DOE 2015b) addressed shipment of 900 kilograms of gap material plutonium to the United States under a ship transport alternative. It considered 12 shipments of gap material plutonium by chartered vessel.

[d] The 2010 *Environmental Assessment for the U.S. Receipt and Storage of Gap Material – Plutonium and Finding of No Significant Impact* (DOE/EA-1771) (DOE 2010) addressed shipment of 100 kilograms of gap material plutonium to the United States under a ship transport alternative and an aircraft transport alternative. Only the ship transport alternative is included here because the aircraft transport alternative has not been implemented.

[e] The option of shipping the same 5,000 kilograms of unirradiated HEU by military cargo or commercial aircraft was also assessed. Air shipment of all unirradiated HEU was projected to result in a collective dose to air crew members of up to 1.1 person-rem and a collective dose to ground cargo workers of up to 0.51 person-rem. The corresponding risks were 7×10^{-4} LCF and 3×10^{-4} LCF, respectively (DOE 2006a).

[f] Assumes a radiation dose of 10 millirem per hour at 2 meters for SNF, including shipment of gap material SNF (DOE 2009), and updating the dose-to-LCF factor from that assumed in the FRR SNF EIS (DOE 1996a) to 0.0006 LCFs per person-rem (DOE 2003a).

[g] The impacts values are based on the per-shipment values of 15 ISO containers per transport in Tables F–3 and F–4.

[h] Transport of 34 metric tons requires 28 shipments of 15 ISO containers and one shipment of five ISO containers. Therefore, the values reflect 28.33 times the per shipment impacts in Tables F–3 and F–4.

Note: Totals may not add due to rounding. To convert kilograms to pounds, multiply by 2.205, and metric tons to tons, multiply by 1.1023.

F.8 References

Allianz (Allianz Global Corporate & Specialty), 2015, *Safety and Shipping Review 2015*, Munich, Germany, March.

Brooks, L. F., 2004, "Testimony of Linton F. Brooks, Under Secretary for Nuclear Security and Administrator, National Nuclear Security Administration, to the Congressional Subcommittee on National Security, Emerging Threats, and International Relations, Hearing on Nuclear Security: Can DOE Meet Physical Security Requirements," Washington, DC, April 27.

DHS (U.S. Department of Homeland Security), 2006, United States Coast Guard, Press Release, "Fact Sheet: The Coast Guard since September 11, 2001," September 1.

DOE (U.S. Department of Energy), 1994, *Environmental Assessment for the Proposed Interim Storage at the Y-12 Plant, Oak Ridge, Tennessee, of Highly Enriched Uranium Acquired from Kazakhstan by the United States*, [redacted], DOE/EA-1006, October.

DOE (U.S. Department of Energy), 1996a, *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218F, Assistant Secretary for Environmental Management, Washington, DC, February.

DOE (U.S. Department of Energy), 1996b, *Mitigation Action Plan for the Implementation of a Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel*, Washington, DC.

DOE (U.S. Department of Energy), 1998, *Supplement Analysis of Acceptance of Foreign Research Reactor Spent Nuclear Fuel Under Scenarios Not Specifically Mentioned in the EIS*, DOE/EIS-0218-SA-2, Washington, DC, August 19.

DOE (U.S. Department of Energy), 2002*, Recommendations for Analyzing Accidents under the National Environmental Policy Act*, Office of NEPA Policy and Compliance, Washington, DC, July.

DOE (U.S. Department of Energy), 2003a, *Estimating Radiation Risk from Total Effective Dose Equivalent (TEDE), ISCORS Technical Report No. 1*, DOE/EH-412/0015/0802, Rev. 1, Office of Environmental Policy and Guidance, January.

DOE (U.S. Department of Energy), 2003b, *Supplement Analysis – Fabrication of Mixed Oxide Fuel Lead Assemblies in Europe*, DOE/EIS-0229-SA3, National Nuclear Security Administration, Office of Fissile Materials Disposition, Washington, DC, November.

DOE (U.S. Department of Energy), 2004, *Environmental Assessment for the Transportation of Highly Enriched Uranium from the Russian Federation to the Y-12 National Security Complex and Finding of No Significant Impact*, DOE/EA-1471, National Nuclear Security Administration, Washington, DC, January.

DOE (U.S. Department of Energy), 2006a, *Supplement Analysis for the Air and Ocean Transport of Enriched Uranium between Foreign Nations and the United States*, DOE/EIS-0309-SA-2, National Nuclear Security Administration, August 30.

DOE (U.S. Department of Energy), 2006b, Memorandum from C. M. Borgstrom, Director, Office of NEPA Policy and Compliance, to NEPA Community, "Need to Consider Intentional Destructive Acts in NEPA Documents," Washington, DC, December 1.

DOE (U.S. Department of Energy), 2009, *Supplement Analysis for the U.S. Disposition of Gap Material – Spent Nuclear Fuel*, DOE/EIS-0218-SA-4, National Nuclear Security Administration, Washington, DC, January 13.

DOE (U.S. Department of Energy), 2010, *Environmental Assessment for the U.S. Receipt and Storage of Gap Material – Plutonium and Finding of No Significant Impact*, DOE/EA-1771, National Nuclear Security Administration, Washington, DC, May. **OUO**

DOE (U.S. Department of Energy), 2012, *Stabilization, Packaging, and Storage of Plutonium-Bearing Materials*, DOE-STD-3013-2012, Washington, DC, March.

DOE (U.S. Department of Energy), 2015a, *Final Surplus Plutonium Disposition Supplemental Environmental Impact Statement*, DOE/EIS-0283-S2, Washington, DC, April.

DOE (U.S. Department of Energy), 2015b, *Environmental Assessment for Gap Material Plutonium—Transport, Receipt, and Processing and Finding of No Significant Impact*, DOE/EA-2024, December.

DOT (U.S. Department of Transportation), 2013a, Maritime Statistics, Vessel Calls at U.S. Ports by Vessel Type (Updated 3/28/2013), Maritime Administration (accessed June 20, 2013 at www.marad.dot.gov/library_landing_page/data_and_statistics/Data_and_Statistics.htm).

DOT (U.S. Department of Transportation), 2013b, Maritime Statistics, Cruise Summary Tables (updated 6/15/2013), Maritime Administration (accessed June 20, 2013 at www.marad.dot.gov/library_landing_page/data_and_statistics/Data_and_Statistics.htm).

DOT (U.S. Department of Transportation), 2018, Certificate of Compliance for Radioactive Material Package, Model 9975, Revision 5, Package Identification Number USA/9975/B(M)F-85, Washington, DC, August 31.

GAO (U.S. Government Accountability Office), 2019, *Surplus Plutonium Disposition, NNSA's Long-Term Plutonium Oxide Production Plans Are Uncertain*, GAO-20-166, Report to the Committee on Armed Services, U.S. Senate, Washington, DC, October.

IAEA (International Atomic Energy Agency, 2001, *Severity, Probability, and Risk of Accidents during Maritime Transport of Radioactive Material*, IAEA-TECDOC-1231, Vienna, Austria, July.

IAEA (International Atomic Energy Agency), 2018, *Regulations for the Safe Transport of Radioactive Material*, Specific Safety Requirements Number SSR-6 (Rev 1), Vienna, Austria.

INL (Idaho National Laboratory), 2019, *VTR Fuel Facility Plan*, INL/LTD-19-54001, S. C. Marschmann, C. E. Baily, C. Marsden, D. C. Crawford, May.

INL (Idaho National Laboratory), 2020, *Recommended Representative Isotopic Compositions for Potential VTR Pu Supplies*, INL/MIS-20-57910 Rev. 1, April.

NDA (Nuclear Decommissioning Authority), 2019, *2019 UK Radioactive Material Inventory*, December.

NNSA (National Nuclear Security Administration), 2013, Fact Sheet – RMRP: Removing Vulnerable Civilian Nuclear and Radiological Material (available at www.nnsa.energy.gov/mediaroom/factsheets/RMRP-remove), April 12.

Petretto, K., 2008, *Weak States Off-Shore – Piracy in Modern Times*, Discussion Paper, East African Human Security Forum, Hanns Seidel Foundation, Kenya, March.

PNTL (Pacific Nuclear Transport Limited), 2020, "40 Years of PNTL," Fact Sheet (accessed May 1, 2020 at www.pntl.co.uk/category/factsheets/).

SOLAS (International Convention for the Safety of Life at Sea), 1999, *International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes on Board Ships (INF Code),* London, May 27.

SRNL (Savannah River Nuclear Laboratory), 2020, Conceptual Assessment of VTR Add-on Processing Capability, SRNL-TR-2020-00171, Rev 2, Aiken, South Carolina, July 22.

WNTI (World Nuclear Transport Institute), 2007, *The INF Ship*, Fact Sheet No. 5, London, United Kingdom, August.

Appendix G
Scoping Comment Summary

APPENDIX G
SCOPING COMMENT SUMMARY

On August 5, 2019, the U.S. Department of Energy (DOE) published a Notice of Intent (NOI) in the *Federal Register* (84 FR 38021) to prepare a *Versatile Test Reactor Environmental Impact Statement* (VTR EIS) to evaluate the potential environmental impacts of constructing and operating a VTR capability. Publication of the NOI initiated a 30-day public scoping period.

During the scoping period, DOE received 45 comment documents[1] in which 173 comments[2] were identified. DOE reviewed the individual comments and those providing similar input were grouped together and treated as a single comment, concern, or issue. Analysis of written and oral public comments provided during the scoping period helped DOE further identify concerns and potential issues considered in the Draft VTR EIS. The scoping comments and DOE's resolutions are summarized below.

This scoping comment summary reflects DOE's resolution of scoping comments at a particular time, that is, as the Draft EIS was developed following the scoping period. As additional information becomes available and as DOE considers and responds to comments received about the Draft EIS, the resolutions presented in this appendix may continue to evolve. Nonetheless, it is DOE's intent that this appendix reflects the resolution of the scoping comments at the time the Draft EIS was prepared; it will not be updated.

National Environmental Policy Act (NEPA) Process

Comment Summary: One commenter stated that the VTR project is a broad-scope program and is not well defined; thus DOE should first conduct a Programmatic EIS, followed by tiered, site-wide EISs. The NOI is vague about the source of the plutonium needed to support VTR operation, the quantity of which could be dozens of metric tons over its lifetime. The range of environmental, safety, health, security, and cost impacts will differ greatly, depending on what source is used. The options need to be described in detail before a project-level assessment can be conducted.

Another commenter suggested that the scope of the VTR EIS should be limited given that the environmental impacts of Idaho National Laboratory (INL) operations have been described and fully documented for decades. The commenter believes the VTR EIS should rely on the extensive environmental analysis that is available for nuclear activities, including reactor operations at the INL Site, rather than conducting new, environmental analyses. Another commenter stated that NEPA requires all related projects and impacts be included in the VTR EIS.

DOE Response: DOE determined that a Programmatic EIS is not required. The VTR EIS was prepared in accordance with applicable Council on Environmental Quality and DOE NEPA regulations. Chapter 4 of this VTR EIS describes and analyzes the environmental impacts of options for location of the reactor fuel production capability, the VTR, the post-irradiation examination capability, and spent fuel storage. As discussed in Chapter 2, this EIS considers different sources of plutonium (domestic and foreign) for the reactor driver fuel and evaluates the appropriate corresponding environmental impacts. Chapter 2 also describes alternatives that were considered and dismissed from further analysis.

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[1] A comment document is a communication in the form of a letter, an electronic communication (email), a transcription of a recorded phone message, or an individual's comments in the transcript from a public meeting that contains comments from a sovereign nation, government agency, organization, or member of the public regarding the VTR EIS.

[2] A comment is a statement or question regarding the EIS content that conveys approval or disapproval of proposed actions, recommends changes, or seeks additional information.

The full suite of applicable impact analyses was included in the VTR EIS. Where applicable, the VTR EIS incorporates existing NEPA documentation by reference and refers to existing NEPA documents and other studies and reports for more detailed information.

Comment Summary: The National Park Service (NPS) requested to be a cooperating agency for this VTR EIS because there are areas under NPS jurisdiction within the area of potential effect. Potential impacts and mitigations for these resources need to be fully addressed in the VTR EIS.

DOE Response: As described in Chapter 2, under the Oak Ridge National Laboratory (ORNL) VTR Alternative, DOE would locate the VTR and associated facilities at the Melton Valley Site. The Melton Valley Site is over 4 miles away from the Y-12 National Security Complex (Y-12) where Building 9731, Pilot Plant (9731), and Building 9204-3 (Beta-3) are located, and over 1 mile from the Graphite Reactor at ORNL. Under the INL VTR Alternative, DOE would locate the VTR and associated facilities at the Materials and Fuels Complex (MFC) at the INL Site. The MFC is about 45 miles from the Craters of the Moon National Monument and Preserve. Therefore, it is unlikely that construction and operation of the VTR and associated facilities would have an impact on NPS administered locations. Environmental consequences, including impacts on cultural resources, aesthetics, noise and wildlife, and any needed mitigation measures, are described in the VTR EIS.

Comment Summary: Commenters requested that all documents cited in the Draft VTR EIS be publicly available.

DOE Response: To the extent practical, reference documents are available in the public reading rooms as announced in the Notice of Availability (NOA) for the Draft VTR EIS and on the project website. Certain copyrighted materials and sensitive information that could not be provided publicly may be available for review through coordination with the point of contact identified in the NOA.

Public Outreach

Comment Summary: Commenters asked if public scoping meetings would be held in addition to the webcasts. The commenters also stated there was not enough time for the public scoping comment period.

DOE Response: There were no scoping meetings in addition to the two webcast public scoping meetings (August 27 and 28, 2019). For those individuals who could not attend one of the scoping meetings, DOE provided other methods for submitting comments: (1) a link to "regulations.gov"; (2) email; (3) a toll-free phone line; and (4) the U.S. mail. The four presentations made by DOE during the webcast scoping meetings were available on the DOE project website after the scoping meetings through the end of the public scoping period. The length of the scoping period was in accordance with NEPA regulations. In addition, as required by NEPA regulations, a public comment period is planned for review of the Draft VTR EIS. The NOA describes the locations and dates of public hearings on the Draft VTR EIS.

Purpose and Need

Comment Summary: Commenters stated that a VTR is important for the United States to allow for crucial advanced technology and materials testing under fast-spectrum irradiation in order to help design the reactors of the future.

DOE Response: Comment noted.

Comment Summary: Commenters stated that the DOE mission-need statement fails to make the case that the VTR is needed. Commenters stated there are ways to simulate the range of neutron flux typical of a fast reactor in already existing test reactors. The mission-need statement also claims that reactor developers need a facility that can achieve at least 30 displacements per atom per year, although the reference it cites, a 2017 user needs assessment, only calls for a minimum of 20.

DOE Response: A VTR would foster experiments with much higher neutron energy and flux compared to the 35-plus research reactors currently operating at U.S. universities and national laboratories. Creating a fast neutron test environment is essential to the development of the next generation of reactor designs, many of which rely on fast neutrons to create the sustained chain reaction that generates heat.

These advanced technologies are very different from those in the existing commercial fleet of nuclear reactors operating in the United States that use thermal or slow neutrons to create a chain reaction to produce the heat to make electricity. The high neutron flux of a VTR would also be capable of accelerated materials testing to support thermal reactor needs.

Today, there is no fast-spectrum irradiation capability in the United States to support the advanced reactor research and development occurring at national laboratories and in the private sector. Without it, the United States will not be able to regain and sustain its leadership role in the development of the next generation of nuclear power reactors. Many developing countries are investing in nuclear power plants to help provide low-carbon, reliable electricity to their citizens. U.S. technology leadership in the area of advanced reactors is critically important from both economics (market share) and national security perspectives.

DOE's Nuclear Energy Advisory Committee (NEAC)[3] studied the issue and released a report in February 2017, recommending, *"*that DOE-NE proceed immediately with preconceptual design planning activities to support a new test reactor.*"* Multiple advanced reactor developers, including TerraPower, LLC, Westinghouse Electric Company, and Oklo Inc., submitted letters in support of the NEAC report. It was noted that the rate of approximately 6 displacements per atom per year possible at the best current experiment location, is too low to attain damage doses exceeding 100 displacements per atom[4] in a reasonable irradiation time.

In addition to the NEAC report, researchers from INL, Argonne National Laboratory and ORNL interviewed multiple domestic reactor vendors in 2016 to assess overall industry test reactor needs, including General Atomics, TerraPower, and Westinghouse. The report, *Versatile Irradiation Test Reactor User Needs Assessment* (referred to by the commenter) issued in January 2017, states that, *"*all survey responders indicated they would utilize irradiation services that a fast-spectrum reactor can provide with rapid accumulation of displacements per atom under prototypical conditions for qualification of fuel, qualification of fuel manufacturing processes, extension of the useful lifetime of cladding and structural materials under irradiation, study of corrosion behavior of materials and advanced coatings under irradiation, and demonstration of fuel performance.*"* As the commenter noted, 20 displacements per atom per year was the <u>minimum</u> identified in the user survey (i.e., greater than 20 displacements per atom per year would be highly desirable). DOE determined that a higher rate (e.g., 30 displacements per atom per year, or 50 percent better than the minimum) was desirable and would be achievable in the VTR.

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[3] NEAC was established in 1998 to provide independent advice to DOE's Office of Nuclear Energy on complex science and technical issues that arise in the planning, managing, and implementing of the Federal nuclear energy program. Committee members include representatives from universities, industry, foreign nations, and national laboratories.

[4] Note that for many of the advanced reactor designs, 100 or greater displacements per atom is typically the desired damage resistance value for advanced structural materials evaluation.

Nonproliferation

Comment Summary: One commenter stated that DOE should conduct a nonproliferation and security impacts assessment for the VTR program that addresses the VTR and its entire fuel cycle. The effectiveness of material accountancy and control measures at all associated fuel cycle facilities should be realistically analyzed with regard to the potential for theft and diversion of weapon-usable materials.

DOE Response: The VTR and support facilities would meet all laws, regulations, and requirements for material accountability and security, including measures to deter the potential for theft and diversion of weapons-usable materials. A nonproliferation and security impacts assessment is outside the scope of the VTR EIS.

Alternatives

Comment Summary: Commenters requested that the No Action Alternative be analyzed in detail. They also requested DOE consider the use of existing facilities, in particular, the use of the Advanced Test Reactor (ATR) or the High Flux Isotope Reactor (HFIR) to generate an adequate flux of fast neutrons. Other commenters stated that DOE should not consider the "No Use Alternative" (understood to mean the No Action Alternative).

DOE Response: This VTR EIS includes a No Action Alternative (described in Chapter 2 and considered and discussed in Chapter 4) as required by NEPA regulations. Existing test reactors, like ATR at the INL Site and HFIR at ORNL, are thermal neutron reactors. Modifications can be made to simulate fast neutron conditions and limited boosting of fast neutron fluxes in thermal reactors, but irradiation conditions (in terms of neutron flux and energy spectrum) are not sufficient to create data required for a formal fuels and materials development and qualification program for fast reactor designs. In order for new improved materials and fuels to be qualified, they would need to be tested under prototypic conditions. The existence and capabilities of these reactors were taken into consideration by the NEAC, which noted that although existing U.S. operational facilities "provide significant capability for testing fuels and materials in a thermal neutron spectrum, they provide only a very limited capacity for testing in a fast neutron spectrum." The absence of fast-neutron-spectrum testing capability was a key factor in DOE's decision to propose the VTR.

Comment Summary: Commenters stated that the NOI does not provide any indication as to why the INL Site and ORNL were the only reasonable alternative sites for the VTR. A commenter inquired if there is any possibility of the VTR being located at the Savannah River Site (SRS). They also asked what role the Savannah River National Laboratory (SRNL) could have in VTR reactor and/or fuel development. Another commenter suggested that non-DOE sites must also be considered, such as areas in New Mexico.

DOE Response: Through its internal scoping, DOE identified the INL Site and the Oak Ridge Reservation near ORNL as potential sites for the VTR based on such factors as existing supporting facilities (for post-irradiation examination), security requirements, experience, and work force. Alternatives considered and dismissed from detailed analysis are described in Chapter 2 of this EIS. SRS has been identified as a potential location for VTR reactor fuel production. SRNL is among the DOE national laboratories collaborating on the development of the VTR.

Comment Summary: A commenter stated that the VTR EIS should evaluate both thermal and fast reactor options. Another commenter stated that it would make more sense to build a new thermal neutron test reactor with the capability of generating fast neutrons, not the other way around.

DOE Response: DOE considered both thermal and fast reactor options. Modifications can be made to simulate fast neutron conditions and limited boosting of fast neutron fluxes in thermal reactors, but irradiation conditions (in terms of neutron flux and energy spectrum) are not sufficient to create data required for a formal fuels and materials development and qualification program for fast reactor designs. Therefore, the use of thermal reactors was considered and dismissed from detailed analysis.

Comment Summary: One commenter suggested the VTR EIS include an alternative for a lead-cooled fast reactor (LFR).

DOE Response: Other fast reactor designs, including restart of Fast Flux Test Facility (FFTF) and molten salt fast reactors, were included in the Analysis of Alternatives studies that led DOE to propose use of a sodium-cooled fast reactor technology-based VTR to meet the DOE mission needs. The sodium-cooled fast reactor concept for the VTR was chosen over other concepts because of the maturity of its technology. The VTR includes the flexibility (thus, the "versatile" in its name) to test materials and concepts for reactors besides sodium-cooled reactors. The VTR test capabilities include cartridge (closed) loops that would contain fuels or test materials isolated from the primary coolant and would be able to perform tests on different coolant types that include sodium, gas, molten salt, lead, and lead-bismuth eutectic. DOE plans to partner with universities and industry partners to develop a full range of test capabilities that include testing other fast reactor options. The other types of reactors considered by DOE are discussed in Chapter 2 of this EIS.

Comment Summary: Commenters stated that all the requirements for a new test reactor would be met by restarting FFTF at the DOE Hanford Site.

DOE Response: Construction of FFTF was completed in 1978, over 40 years ago, and FFTF operated successfully from 1982 to 1992. In a January 26, 2001 Record of Decision (ROD) (66 FR 7877), DOE decided to permanently deactivate FFTF. In a December 13, 2013 (78 FR 75913) ROD, DOE decided to decommission, dismantle, and entomb FFTF. The sodium coolant was drained from the FFTF reactor, although little additional progress on decommissioning has been made due to budget constraints and priorities. Restarting FFTF was considered, and included a walkdown of the facility by the VTR Project team and several former FFTF engineers and managers. FFTF was dismissed from detailed analysis, in part because FFTF had operated for 10 of its projected 20-year design life, and due to the technical uncertainty associated with a reactor and associated systems that have not operated in over 25 years (see Chapter 2).

Comment Summary: One commenter stated the VTR EIS must evaluate a reasonable range of reactor sizes because environmental impacts could be significantly different based on this fundamental issue. An amended NOI must be prepared and scoping extended so the public can provide input on the reactor size alternatives.

DOE Response: As described in the NOI, the initial evaluation of alternatives during the pre-conceptual design planning activity recommended the development of a sodium-cooled, fast neutron spectrum test reactor in the 300 megawatt thermal power level range. This design would provide a flexible environment for known and anticipated testing. The evaluation of alternatives in this VTR EIS is consistent with the conclusions of the test reactor options study (INL 2017) and the NEAC's February 2017 (NEAC 2017) recommendation.

Support and Opposition

Comment Summary: Some commenters expressed support and others expressed opposition to various aspects of the VTR project. Some commenters expressed support for the VTR to be located at the INL Site and others supported ORNL. Some commenters supported the No Action Alternative and strongly opposed locating the VTR at the INL Site.

DOE Response: As discussed in Chapter 2, DOE evaluated alternative locations for the VTR, including the INL Site and the Oak Ridge Reservation (near ORNL). In addition, this EIS includes consideration of a No Action Alternative. DOE intends to proceed with the NEPA process, consider all viable alternatives objectively, and announce its decision regarding a new VTR in a ROD issued no sooner than 30 days after the Final EIS is issued.

Technology

Comment Summary: One commenter stated that the VTR EIS should consider not only metal but also oxide and nitride fuels. New accident-tolerant fuels that can increase electricity output and have no chance of contributing to a design-basis, loss-of-coolant accident need to be fully supported in order to keep our existing commercial reactor fleet operational. Another commenter asked that DOE consider loop-type reactor designs.

DOE Response: Metallic fuel is planned for initial operation of VTR. Future operations could include other fuel forms and isotopic compositions. Operation of VTR with different fuel forms and compositions may require additional NEPA documentation.

DOE has considered both pool- and loop-type reactors. For a test reactor, pool-type designs offer several advantages over loop-type designs:

- Generally a smaller reactor for the same power level (better use of space);
- A simplified coolant boundary operating at low pressure (eliminates potential leak paths);
- Large thermal inertia (more tolerant of coolant transients);
- Improved ability to use passive, natural-circulation heat removal systems (better safety and more efficient); and
- Opportunity to store spent fuel in the primary vessel (improved safety).

DOE proposes to use the GE Hitachi PRISM reactor, a pool-type reactor, as the basis for VTR's design: that design will require several modifications, notably the elimination of electricity production and the accommodation for experimental locations within the core. Utilization of a well-developed reactor design that has undergone substantial review, including by the U.S. Nuclear Regulatory Commission (NRC) (NRC 1994), presents major advantages both in terms of technology readiness and the time required for detailed reactor development and implementation.

Comment Summary: A commenter asked under what conditions the VTR could be operated as a breeder reactor. They also inquired about how spent nuclear fuel (SNF) would be reprocessed, and if the SNF were not to be reprocessed, what methods of "conditioning the SNF for disposal" would be used, and where would it be disposed.

DOE Response: As indicated in the NOI and as stated in this VTR EIS (Chapters 1 and 2), there are no plans to operate VTR as a breeder reactor or reprocess SNF. The SNF would be treated to remove the sodium and safely stored on site until a repository becomes available.

Comment Summary: A commenter stated that DOE should consider how to use the energy produced by the reactor for additional research.

DOE Response: The VTR is proposed as a test reactor to provide a fast-neutron-spectrum testing capability to test advanced nuclear fuels and materials, including those for next-generation nuclear reactors. Making use of the energy produced is beyond VTR's current scope and purpose.

Reactor Driver Fuel and Control Rods

Comment Summary: Commenters expressed concerns regarding the lack of information about the VTR fuel. They asked if there is a difference between "start-up fuel" and fuel used for post start-up operation. They stated that the VTR EIS must include details of VTR fuel, including if it is to be made from reactor-grade plutonium, surplus weapons-usable plutonium, Zero Power Physics Reactor (ZPPR) fuel, high-assay low-enriched uranium (HALEU), thorium, or other materials. Where would the materials come from? Would U-233 be separated from irradiated fuel?

DOE Response: Metallic uranium-plutonium-zirconium driver fuel is planned for initial operation of VTR. DOE continues to evaluate the source of the plutonium to be used in driver fuel fabrication, which could include ZPPR fuel, reactor-grade plutonium, and surplus weapons-usable plutonium. The known potential fuel forms and compositions are described in this VTR EIS as is the potential plutonium transport from domestic locations. While HALEU may be used as a possible future driver fuel, DOE would use existing stores of enriched uranium as a source for this material. Future operations could include other fuel forms and isotopic compositions. Operation of VTR with different fuel forms and compositions may require additional NEPA analysis.

Comment Summary: Commenters expressed concerns regarding the lack of information about where the VTR fuel would be manufactured and noted that environmental impacts associated with fuel production and fabrication must be discussed. One commenter stated that the NOI mentions that the INL Site and SRS are where the reactor fuel could be fabricated. What facilities at the INL Site would be used? Can existing facilities at SRS, including the abandoned Mixed Oxide (MOX) Fuel Fabrication Facility, be used for fuel fabrication? If DOE is considering new facilities at SRS, would they be located in the K Area or elsewhere? Would the aging H-Canyon at SRS be considered for HALEU production, and if so, what are the associated risks and waste streams, and how would they be managed? If new facilities would be needed, please give details, including cost and construction and operation schedules. A commenter stated that an amended NOI must be prepared and scoping extended so the public can provide input on fuel fabrication alternatives in the VTR EIS.

DOE Response: The facilities that could be used for reactor fuel production are described in this VTR EIS in Chapter 2 and Appendix B. Reactor fuel production operations would be established in existing facilities/structures; new buildings would not be constructed. The impacts of modification and operation of these facilities are evaluated in this VTR EIS. SRS is being considered because it has a long history of fuel fabrication, has some of the feed materials onsite, and has facilities and personnel that can safely handle the materials. Because the National Nuclear Security Administration has another mission planned for the MOX Fuel Fabrication Facility, it is not considered a reasonable location for VTR reactor fuel production. The VTR EIS does not evaluate newly proposed means of producing HALEU, such as processing in H-Canyon.

Comment Summary: Commenters questioned how many staff would be employed for fuel fabrication and how the staff would be trained. How this expertise would be developed must be discussed in the VTR EIS.

DOE Response: This VTR EIS includes an estimate of staff needed to construct and operate each capability, including reactor fuel production. Whereas prior site experience and a knowledgeable and experienced workforce are factors considered in identifying reasonable alternatives and ultimately in making a decision on a location for the VTR, staff hiring, development, and training are administrative aspects of the activity that are outside the scope of the environmental impacts evaluated in the VTR EIS.

Comment Summary: A commenter was concerned that DOE asserted that the fabrication process for metal fuel is relatively simple. The commenter believed it should be compared to the major and costly effort that would be required for the dilute-and-dispose plutonium disposition program at SRS, which is arguably a far simpler process that also relies on existing facilities. Also, uranium-plutonium-zirconium fuel fabrication scale-up from EBR-II to VTR fuel dimensions and production rates must be demonstrated to reduce uncertainties.

DOE Response: This VTR EIS describes the VTR reactor fuel production process in Appendix B and evaluates the potential environmental impacts in Chapter 4. As indicated in the NOI, reactor fuel production at the INL Site or SRS are options evaluated in this EIS. As necessary to support deployment of the reactor fuel production process and to manage uncertainties, DOE would conduct proof-of-principal testing, demonstration, and scale-up activities. The dilute-and-dispose capability for surplus plutonium has a different end point and would not result in the production of reactor fuel.

Comment Summary: One commenter asked what type of control rods would be used and where they would be fabricated.

DOE Response: The preliminary design anticipates using clad boron-carbide absorber rods. DOE expects to be able to purchase control rods from a vendor as a commercial item, and as such, they do not require analysis in the EIS.

Spent Fuel Management and Disposition

Comment Summary: A commenter stated that no decisions have occurred regarding long-term SNF management and asked if the sites under consideration for this project would be capable of managing SNF long into the future. Another commenter requested the VTR EIS evaluate the impacts of managing the additional inventory of SNF on the ongoing sodium-bonded SNF pyroprocessing program at INL.

DOE Response: This VTR EIS addresses the environmental impacts associated with the treatment and temporary storage of the VTR SNF under all alternatives. This assessment includes the impacts associated with operation of existing facilities and the impacts associated with construction and operation of any new facilities. The SNF assemblies would be stored within the VTR reactor vessel until decay heat generation is reduced to a level allowing fuel transfer and treatment. When the decay heat reaches manageable levels, the SNF would be transferred to a fuel treatment facility (at the INL Site, it would be the Fuel Conditioning Facility), where the SNF would be treated using a simple melt-distill process. Use of this process would require the installation of a new distillation furnace. The intent is to prepare the VTR SNF for ultimate disposal only, the more complex electrometallurgical treatment used for EBR-II fuel (which recovers HALEU) is not required or planned. Because the treatment of VTR SNF would not utilize the same process/equipment used for EBR-II SNF, there should be no impact on that program. Following treatment, the SNF would be placed in dry storage casks and stored until shipment to a permanent repository. Dry cask storage of SNF until another facility (storage or disposal) becomes available could be accomplished at the sites considered for the VTR.

Comment Summary: Commenters requested the development of a permanent Federal repository for high-level radioactive waste (HLW), and stated that continuing to point to Yucca Mountain as the disposal solution is unacceptable. Commenters stated that safe permanent storage of the existing waste inventories should be the highest priority, and that DOE should evaluate the impact of orphaned HLW. A commenter requested an analysis of disposal of the SNF that the VTR would generate, and that all reasonable SNF disposition alternatives, including direct disposal, be evaluated.

DOE Response: The program for a geologic repository for SNF and HLW at Yucca Mountain, Nevada has been terminated. Notwithstanding the decision to terminate the Yucca Mountain Nuclear Waste Repository Program, DOE remains committed to meeting its obligations to manage and, ultimately, dispose of SNF and HLW. However, this commitment is beyond the scope of the VTR EIS. Existing SNF inventories are safely stored on site at operating and shut down nuclear facilities. The VTR SNF would be processed to remove sodium and stored on site until a consolidated storage facility or repository becomes available. The disposal of VTR SNF would be analyzed in the supplementary NEPA documentation prepared for the repository.

Comment Summary: A commenter stated the best way to dispose of SNF is to use it in generating new fuel in a fast breeder reactor, or to recycle it. One commenter questioned whether the VTR would be used to help develop reprocessing techniques.

DOE Response: If SNF is reprocessed and material recovered to send back to the reactor as nuclear fuel, it is referred to as a closed fuel cycle. If the fuel is used *"*once through*"* and not reprocessed, it is referred to as an open fuel cycle. VTR would not be operated as a breeder reactor and there are no plans to reprocess and/or "recycle" the VTR fuel in a closed fuel cycle. VTR will be operated with a "once through" fuel cycle and no reprocessing to recover and reuse uranium or plutonium would occur. VTR is being designed for the purpose of performing fuels and materials irradiation experiments that could yield information supporting evaluations of closed fuel cycles.

Environmental Impacts

Commenter Summary: One commenter believes that the project's potential effects on archaeological resources will need to be addressed through consultation with the State Historic Preservation Office and the Section 106 compliance process. One commenter stated that the VTR EIS needs to address visitor access to Manhattan Project National Historical Park facilities, including Building 9731, Pilot Plant (9731), and Building 9204-3 (Beta-3). Building 9731 is being considered for listing on the National Register of Historic Places. Building 9204-3 is currently inaccessible by park visitors due to significant mitigation and maintenance needs, as well as its location within a high security area.

DOE Response: The VTR EIS analyzes potential impacts on cultural resources, including archaeological and historic resources. Consistent with the National Historic Preservation Act and established relationships between DOE and the State Historic Preservation Offices, consultations would occur, as appropriate. It should be noted that the location being considered for the VTR near ORNL is not in the immediate vicinity of the two identified buildings, which are located over 4 miles away at Y-12.

Comment Summary: A commenter stated that the VTR EIS needs to address impacts on the night sky (dark sky), natural sounds, and wildlife at Craters of the Moon National Monument and Preserve.

DOE Response: Chapter 4 of this VTR EIS analyzes potential impacts on the night sky (aesthetics), natural sounds (noise), and wildlife (ecological resources). The VTR EIS also considers the impacts on nearby national parks and monuments within the region of influence (ROI).

Comment Summary: Commenters asked that the VTR EIS clearly describe the geology, depth to groundwater, direction of flow and speed of flow for the Snake River Plain Aquifer beneath the proposed facility and any storage site. They also recommend the VTR EIS include information on whether construction of the project would disturb a land area of one or more acres, contaminants of concern, impacted waters, and water bodies on the U.S. Environmental Protection Agency-approved 303(d) list that could be affected. Also included are how anti-degradation provisions of the Clean Water Act would be met, potential contamination of drinking water sources and measures that would be taken to protect drinking water, cumulative effects from this and other projects on hydrologic conditions, and whether specific permits would be needed.

DOE Response: Chapter 3 of this EIS describes water resources, and Chapter 4 analyzes potential impacts on surface and groundwater resources, including cumulative impacts, commensurate with the potential for impacts. This EIS also describes applicable environmental laws, regulations, permits, and agreements.

Comment Summary: One commenter stated the VTR EIS should describe the possible impact on the customs and culture for those living downstream of the VTR, and the impacts on habitat types, values, and functions associated with those waters.

DOE Response: This VTR EIS analyzes the potential for VTR and support facilities to impact downstream waters, as appropriate. Impacts on human health, ecological resources, and cultural practices are considered and any potential impacts identified in Chapter 4.

Comment Summary: A commenter stated that the VTR EIS should identify projected types and volumes of hazardous waste and expected management plans. Commenters recommended that the VTR EIS address potential direct, indirect, and cumulative impacts of both hazardous materials and wastes.

DOE Response: Chapter 4 of this VTR EIS identifies projected waste types (including radioactive and hazardous wastes) and their volumes. It also describes expected waste management processes (including storage and disposal). The VTR EIS addresses potential direct, indirect, and cumulative impacts of the management of hazardous materials and waste.

Comment Summary: One commenter recommended that the VTR EIS include the following items related to air quality impacts: a detailed discussion of ambient air conditions, data on emissions of criteria pollutants, pollutant data from mobile and stationary sources, an equipment emissions mitigation plan, health effects from air pollutants, and discussion of applicable Federal and State regulations. They also requested a discussion of mitigation measures to minimize impacts on air quality.

DOE Response: Chapter 3 of this VTR EIS describes existing ambient air quality within the applicable ROIs. Chapter 4 of this EIS analyzes potential impacts on air quality from mobile and stationary air emissions. Potential impacts from nonradiological air emissions are evaluated in relation to established regulatory standards. In addition, the potential health effects of radiological emissions are analyzed. The VTR EIS describes applicable environmental laws, regulations, permits, and agreements. Mitigation of impacts, if needed, is discussed.

Comment Summary: A commenter stated the proposed project may impact threatened, endangered, or candidate species listed under the Endangered Species Act (ESA), their habitats, as well as State-sensitive species. They recommend the VTR EIS identify potentially impacted species under ESA, and other sensitive species within the project area.

DOE Response: Potential impacts on ecological resources are analyzed in Chapter 4. The VTR EIS identifies threatened, endangered, candidate, and other sensitive species and their habitats within the ROI and evaluates potential impacts.

Comment Summary: One commenter requested the VTR EIS discuss the potential for seismic risk and approaches to evaluate, monitor, and manage this risk. This would include a seismic map, information on seismic design and construction standards and practices, and measures to avoid and mitigate the risks.

DOE Response: The VTR EIS includes a description of geology and soils, including seismicity and seismic risk. The human health effects from seismically induced accidents are discussed. Mitigation of impacts is discussed.

Comment Summary: A commenter recommended the VTR EIS include a discussion of reasonably foreseeable effects that changes in the climate may have on the proposed program. This could help inform the development of measures to improve the resilience of the program. If projected climate changes could notably exacerbate the environmental impacts of the program, commenters recommended these impacts be considered in the VTR EIS.

DOE Response: The VTR EIS considers climate change impacts in Chapter 4. The design and engineering of the VTR and support facilities considers operating under a range of extreme climate conditions.

Comment Summary: A commenter recommended the project be designed to include a mitigation monitoring program to ensure compliance with all mitigation measures and assess their effectiveness.

DOE Response: Mitigation of impacts is discussed. DOE would prepare a mitigation action plan (MAP) for any impacts requiring mitigation. The MAP would include mitigation monitoring.

Human Health and Safety

Comment Summary: Commenters stated that because radioactive materials may affect workers and the public, they recommend that the VTR EIS include information regarding hazardous materials releases, potential pathways of exposure, periods of exposure, and probable impacts from exposure. Commenters requested analysis of VTR emissions potential impacts on human health, including cancers; pulmonary, cardiovascular, and autoimmune diseases; and birth defects. The VTR EIS should also address whether radionuclide emissions would change substantially under any of the alternatives.

DOE Response: The potential impacts on human health of releases of radioactive materials from both routine emissions and accidents are evaluated in Chapter 4 of this VTR EIS. It includes information on facility emissions and potential pathways of exposure for workers and the public. The potential health effects of VTR emissions were analyzed using standard approaches for evaluation of the impacts of exposure to radiological materials. This EIS addresses emissions and associated health effects and compares them to identify substantial differences between the alternatives.

Comment Summary: A commenter requested DOE address the "fatal flaw" of plutonium and uranium moving through high-efficiency particulate air (HEPA) filters due to "alpha recoil."

DOE Response: The real-world performance of multiple stages of HEPA filters has been well demonstrated and experimental testing confirms the performance of HEPA filters for uranium and plutonium particles. The independent Defense Nuclear Facilities Safety Board (DNFSB) thoroughly evaluated the use of HEPA filters by DOE) and has issued multiple reports on the performance of HEPA filters within the DOE complex. HEPA filters used in support of the VTR activities would conform to the latest version of DOE Standard "Specifications for HEPA Filters Used by DOE Contractors," DOE-STD 3020-2015. Performance testing required by this standard for all HEPA filters credited for safety would ensure that the filters meet or exceed the performance requirements assumed in safety evaluations.

Accidents and Intentional Destructive Acts

Comment Summary: Commenters requested the VTR EIS consider the full range of accident scenarios that could result in large radiological releases, even if DOE considers the accidents incredible. Commenters requested the VTR EIS analyze core disassembly accidents and the risks of a sodium leak or sodium fire. They requested DOE fully evaluate the environmental impacts of these events. They recommend that the EIS describe measures that would be taken to ensure that the chances of an accident would be kept to a minimum and measures that would ensure that the workers would be protected. A commenter also requested that economic consequences be considered for severe reactor accidents.

DOE Response: In Chapter 4 and Appendix D, this VTR EIS describes and analyzes a suite of design-basis and beyond-design-basis accidents. The accident analysis for the EIS is based on the most current safety analysis contained in the safety basis documents, including the safety design report. The accidents consider applicable natural phenomena initiators, such as earthquakes, tornados, wildfires, flooding, volcanoes, and human initiators. Accident scenarios considered include core disassembly accidents and sodium leaks or fires. The EIS also analyzes the impacts of potential accidents on workers and public health and safety. A description of emergency response and post response cleanup in the event of an accident was included.

Comment Summary: One commenter stated that the VTR EIS should examine containment behavior and whether a leakage failure would lead to a catastrophic failure. They stated the behavior of the containment under elevated temperature and pressure, including the effect of aerosols within the containment atmosphere, has not been thoroughly investigated. They also mention that liner-anchorage-concrete interaction is significant in determining how liners tear in concrete containments.

DOE Response: The VTR is a pool-type test reactor that operates at relatively low pressures and is not subject to the types of accidents described by the commenter, which are typical of large light water reactors.

Comment Summary: Commenters stated the VTR and related facilities are subject to security breaches or terrorism from disgruntled employees, including cyber hacking. Commenters stated the VTR EIS must consider the full range of sabotage scenarios for the VTR that could result in radiological releases to the environment and the environmental impacts of the releases and must include an analysis of defenses against cyber-attacks.

DOE Response: The consequences of intentional destructive acts (IDAs) are described in the VTR EIS. The analysis of IDAs considers terrorism from disgruntled employees and cyber hacking. An analysis of physical or cyber vulnerabilities and defenses is a security function that would be performed independent of this EIS. Details of the IDA analysis are not available to the public for security reasons.

Environmental Justice and Native American Issues

Comment Summary: Commenters stated there should be coordination with Tribal Governments and communities, and recommended the VTR EIS describe the process and outcome of Government-to-Government consultation between DOE and each of the Tribal Governments and communities that could be affected by the project. A commenter also stated that the VTR EIS needs to address the potential for disproportionate adverse impacts on environmental justice populations near the VTR facilities.

DOE Response: DOE maintains Tribal outreach programs with the Native American Tribes surrounding applicable sites and routinely meets with interested Tribal Governments to discuss issues of mutual concern. In support of this VTR EIS, DOE will continue to hold discussions with Native American communities and Tribal governments.

The VTR EIS includes descriptions of minority and low-income populations near the candidate sites in Chapter 3. Consistent with environmental justice requirements, the potential for disproportionately high and adverse impacts on minority and/or low-income populations is addressed in Chapter 4.

Cumulative Impacts

Comment Summary: Commenters recommended the VTR EIS cumulative impact assessment consider the following: resources that are cumulatively impacted; appropriate geographic area and the time over which the effects have occurred and will occur; all past, present, and reasonably foreseeable future actions that have affected, are affecting, or would affect resources of concern, including those outside of DOE's jurisdiction; a benchmark or baseline; and scientifically defensible threshold levels.

DOE Response: This VTR EIS includes an analysis of cumulative impacts in Chapter 5, including the effects of the proposed action when added to other past, present, and reasonably foreseeable future projects in the ROI, including those outside of DOE’s jurisdiction.

Decontamination and Decommissioning

Comment Summary: Commenters requested the VTR EIS include discussion about ultimate decontamination and decommissioning of the facility after its useful life, including disposition of the fission products, SNF, and sodium coolant.

DOE Response: Chapter 4 of this VTR EIS includes a discussion of decontamination, decommissioning, and demolition of the VTR after its useful life.

Laws and Regulations

Comment Summary: Commenters requested that the Draft VTR EIS describe the framework under which a VTR would be regulated and recommended the VTR EIS include a list of all permits and authorizations that the project facilities already have and would need, including modification to any existing permit or authorization. They asked: Would the VTR be licensed by the NRC? If the NRC would not provide oversight of the reactor's design and operation, how would such oversight be accomplished? Would the DNFSB have an oversight role? Another commenter stated the VTR EIS should discuss ramifications of the 1995 Idaho Settlement Agreement.

DOE Response: DOE would authorize the VTR and provide oversight of construction and operations, like previous test reactors (e.g., ATR, HFIR, and Transient Reactor Test Facility [TREAT]). The VTR would not be licensed by NRC. Under the Atomic Energy Act (AEA) of 1954 and its amendments, and the AEA Energy Reauthorization Act of 1974, DOE has the authority to develop, construct, and operate its own reactors. Under this authority, DOE plans to conduct the safety review for the VTR and authorize its construction and operation. DOE facilities, such as the VTR, are generally exempt from NRC licensing in accordance with Section 110 of the Energy Reauthorization Act and Title 10, Code of Federal Regulations 50.11, Exceptions and Exemptions from Licensing Requirements. The VTR would not be a defense nuclear facility, and therefore, the DNFSB would not have an oversight role.

In Chapter 7, the VTR EIS addresses environmental laws, regulations, permits, and agreements. The 1995 Idaho Settlement Agreement is acknowledged in that chapter.

Comment Summary: A commenter stated that any nonradioactive wastes associated with construction and operation of the facilities must be handled in accordance with Federal and State solid and hazardous waste rules and regulations. A commenter recommended the Draft VTR EIS include discussion of specific hazardous and mixed waste management and monitoring practices, treatment methods, storage areas, and utilization of landfills for attaining compliance with State regulations.

DOE Response: Chapter 7 of this VTR EIS identifies applicable environmental laws, regulations, permits, and agreements related to waste management. Chapters 3 and 4 include descriptions of radioactive, hazardous, mixed, and nonhazardous waste management practices, including treatment, storage, and disposal.

Comment Summary: A commenter inquired whether there are any legal or regulatory constraints prohibiting the use of the VTR as a breeder reactor.

DOE Response: The VTR would be a test reactor. There is no legal constraint against using the VTR as a breeder reactor; however, there are no plans to use VTR as such. The VTR mission is to be a test reactor with its core configured so that it operates as a "burner" reactor, i.e., it would consume more fissile material than it would create. Accordingly, the DOE-approved safety basis would be developed based on configuring the VTR as a test reactor that does not include the capability to use it as a breeder reactor. Any proposal to reconfigure the reactor as a breeder would require a reanalysis of the design and safety basis, and would also include a re-evaluation of environmental impacts.

Cost and Schedule

Comment Summary: One commenter requested an estimate of the cost for the VTR's construction and startup. Another commenter stated that the schedule established for completion of the VTR is unrealistic, and the Draft VTR EIS must address the impacts of delay to the cost and schedule for the project. Would MOX "lessons learned" be applied to the VTR program?

DOE Response: Detailed cost estimates are not yet available. However, based on the current conceptual design and documentation submitted for Critical Decision 1 (CD-1, Approve Alternative Selection and Cost Range) (DOE 2020), the estimated cost range is between \$2.6 and \$5.8 billion. The range for completion of construction is estimated to be from fiscal year 2026 to fiscal year 2031. Based on the near-term schedule, the EIS should capture the likely environmental impacts of the alternatives for construction and operation of the VTR and supporting facilities. DOE does not plan to present cost and schedule information in the VTR EIS.

DOE always strives to learn from its past projects as well as those from the private sector. Specifically, VTR will begin construction after the appropriate level of final design has been completed as well as development of the supply chain, prototype testing of critical components, and completion of labor analysis studies.

Comment Summary: One commenter asked which private entities would use the VTR and how much would they pay toward construction and operation and management of waste? Would private entities be liable for negligence in using the VTR?

DOE Response: Once operational, the VTR will be designated as an Office of Nuclear Energy, Nuclear Science User Facilities (NSUF) partner facility. Through NSUF, access will be available to universities, DOE national laboratories, and industry through competitive peer-reviewed processes. In addition to access through NSUF, users can also gain access to the VTR on a pay-for-access basis. There is the potential for cost sharing with industry and other governments, but at this time, no such arrangements have been made. DOE would be the owner and operator of the VTR and would assume all risks and responsibilities associated with its operation. Requests for access will be evaluated for technical feasibility, safety, and capability of resources requested to perform the proposed work.

Out of Scope

Comment Summary: Commenters asked about renewable energy technologies and cost comparisons of those alternative energy sources. One commenter stated that buried waste at the INL Site must be addressed. A commenter requested that the VTR EIS consider employee expertise and whether sufficient human resources are available to support this project. One commenter stated the VTR EIS should describe any possible change to the psyche of people who live downstream and what impact it could have on local and regional economies.

DOE Response: The impacts and costs of alternative energy technologies, including renewable energy, is outside the scope of this VTR EIS. The impacts of existing buried waste at the INL Site and the cleanup of existing contaminated sites are outside the scope of this VTR EIS, although these activities will be considered as part of cumulative impacts. The availability of trained personnel, including personnel education and training, and the availability of funding for training, are administrative concerns that are outside the scope of this VTR EIS. DOE's analyses presented in this VTR EIS identify potential impacts that could occur as a result of the proposed action and alternatives on resource areas consistent with NEPA regulations. The results of the analyses provide decision-makers and the public, including people living downstream of the site, conservative estimates of potential impacts that could occur as a result of implementation of the proposed action and alternatives.

G.1 References

DOE (U.S. Department of Energy), 2020, Memorandum from M. W. Menezes, Chief Executive of Project Management, to R. Baranwal, Assistant Secretary for Nuclear Energy, Re: Approval of Critical Decision-1, *Approve Alternative Selection and Cost Range*, for the Versatile Test Reactor Project, Washington, DC, September 11.

INL (Idaho National Laboratory), 2017, *Advanced Demonstration and Test Reactor Options Study,* INL/EXT-16-37867, Rev. 3, ART Program, Idaho Falls, Idaho, January.

NEAC (Nuclear Energy Advisory Committee), 2017, *Assessment of Missions and Requirements for a New U.S. Test Reactor*, February.

NRC (U.S. Nuclear Regulatory Commission), 1994, *Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor*, NUREG-1368, Washington, DC, February.

Appendix H
Contractor Disclosure Statements

NEPA DISCLOSURE STATEMENT FOR PREPARATION OF A VERSATILE TEST REACTOR ENVIRONMENTAL IMPACT STATEMENT

CEQ regulations at 40 CFR 1506.5(c), which have been adopted by DOE (10 CFR 1021), require contractors who will prepare an EIS to execute a disclosure specifying that they have no financial or other interest in the outcome of the project. The term "financial interest or other interest in the outcome of the project," for the purposes of this disclosure, is defined in the March 23, 1981 guidance "Forty Most Asked Questions Concerning CEQ's National Environmental Policy Act Regulations," 46 FR 18026-18038 at Question 17a and b.

"Financial or other interest in the outcome of the project 'includes' any financial benefit such as a promise of future construction or design work in the project, as well as indirect benefits the contractor is aware of (e.g., if the project would aid proposals sponsored by the firm's other clients)," 46 FR 18026-18038 at 18031.

In accordance with these requirements, the offeror and any proposed subcontractors hereby certify as follows: (check either (a) or (b) to assure consideration of your proposal)

(a) ___X___ Offeror and any proposed subcontractor have no financial interest in the outcome of the project.

(b) _______ Offeror and any proposed subcontractor have the following financial or other interest in the outcome of the project and hereby agree to divest themselves of such interest prior to award of this contract.

Financial or Other Interests:

1.
2.
3.

Certified by:

Kelly C. Russell

Signature

Kelly C. Russell, Leidos

Name

September 30, 2020

Date

NEPA DISCLOSURE STATEMENT FOR PREPARATION OF A VERSATILE TEST REACTOR ENVIRONMENTAL IMPACT STATEMENT

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<u>Financial or Other Interests</u>:

1.
2.
3.

Certified by:

Frederick J. Carey, President
Potomac-Hudson Engineering, Inc.

Signature

____
Name

13-August-2020
____
Date