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PNNL-XXXXX, Draft (11/14/2022)

Development and Application of Risk Assessment Approach for Transportation Package Approval of a Transportable Nuclear Power Plant for Domestic Highway Shipment

November 2022

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U.S. DEPARTMENT OF ENERGY

Prepared for the U.S. Department of Energy
under Contract DE-AC05-76RL01830

PACIFIC NORTHWEST NATIONAL LABORATORY
operated by
BATTELLE
for the
UNITED STATES DEPARTMENT OF ENERGY
under Contract DE-AC05-76RL01830

Printed in the United States of America

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SUMMARY

The U.S. Department of Defense (DOD) Strategic Capabilities Office (SCO) has initiated a project, referred to as Project Pele, for the construction and demonstration of a prototype transportable microreactor or Transportable microreactor Nuclear Power Plant (TNPP). Under Project Pele a prototype transportable microreactor and associated reactor fuel will be fabricated at existing commercial facilities while startup and operation of the microreactor will be demonstrated at the U.S. Department of Energy's (DOE's) Idaho National Laboratory (INL) site. Additional future demonstration may be performed at a DOD reservation and, in that case, transportation of the irradiated TNPP would occur on public roads and highways. Moreover, DOD's potential use of transportable microreactors on military installations and potentially in field operations may be based to some extent on the experience gained from this demonstration project. This military use will necessitate shipments of microreactors, potentially containing irradiated nuclear fuel using transportation systems that are also utilized by the public (e.g., interstate highway system). Accordingly, such shipments would therefore likely be regulated by the U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Transportation (DOT). The transportation of TNPPs has never been licensed by the NRC and could be a challenge especially if the TNPP contains irradiated fuel.

Pacific Northwest National Laboratory (PNNL) has been tasked by SCO to address the regulatory challenges associated with safe transport of TNPPs containing irradiated nuclear fuel. In a previous study documented in PNNL-31867 (*Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages*), it was determined that the expected radioactive inventory in the irradiated fuel of a TNPP would likely require shipment in an NRC-approved Type B package (or spent nuclear fuel cask), but that a TNPP "package" is unlikely to meet NRC requirements in Part 71 of Title 10 of the *Code of Federal Requirements* (CFR) for a Type B package. It was therefore concluded that shipment of the TNPP package under existing regulations would likely require NRC approval using the 10 CFR 71.12 ("Specific exemptions") exemption process that relied on risk-informed decisionmaking supported by quantitative risk assessment.

However, the NRC transportation regulations in 10 CFR Part 71 do not currently include a risk-informed framework or definitive process for approval of transportation of radioactive materials. Rather, current regulations specify that packages used to transport radioactive materials meet deterministic performance standards. The performance standards define normal operating conditions and hypothetical accident conditions a package must be capable of withstanding without exceeding specified acceptance criteria. For a Type B package, or cask, being used to ship irradiated nuclear fuel, these acceptance criteria are defined to: (1) limit releases of radioactive material and radiation levels outside the package, and (2) assure that the used nuclear fuel will remain subcritical (that is, it will not undergo a self-sustaining nuclear reaction). Because of this gap and the fact that risk associated with not fully meeting the deterministic performance requirements in the current regulations may be acceptable, a risk-informed regulatory framework was proposed in PNNL-31867 for demonstrating that a TNPP package and associated shipment process and controls provides equivalent safety to that of a Type B package.

Following the framework report, at the request of SCO, PNNL developed a plan for the development and application of a risk assessment approach to support a risk-informed pathway for NRC and DOT approval of a one-time surface shipment of a microreactor transportation package. This plan, which is

documented in PNNL-33524 (*Plan for Development and Application of Risk Assessment Approach for Transportation Package Approval of an MNPP for Domestic Highway Shipment*), identifies the proposed content of a risk-informed exemption request to the NRC for the transport of an TNPP package and, specifically, the Project Pele microreactor.

As a follow-on to PNNL-33524, this report provides a demonstration implementation of this plan for a hypothetical one-time shipment of the Project Pele microreactor with irradiated fuel. This demonstration implementation is intended to be used as a guide or template for the development of a hypothetical risk-informed exemption request to the NRC by the Project Pele microreactor vendor for a one-time ground surface shipment by truck. Though this report only addresses the application of risk information for the TNPP package safety analysis focusing on 10 CFR Part 71 compliance, this same accident and risk information could also be used to support the Environmental Assessment (EA) or Environmental Impact Statement (EIS), required by 10 CFR Part 51 ("Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions") for exemptions.

Demonstration implementation of the plan includes the development of a TNPP transportation probabilistic risk assessment (PRA) methodology, proposed risk evaluation guidelines, technical information, data, and example analyses that provide a potential template for a microreactor vendor to follow when making a request to the NRC. It also addresses important supporting analyses to the PRA such as the treatment of key assumptions and sources of modeling uncertainty and the concept of defense-in-depth and safety margin. Key advantages of using the approach are: (1) increasing the likelihood of successfully obtaining regulatory transportation package approval, (2) informing the design on the relative risk significance of TNPP containment and shielding, and (3) informing the need for transportation compensatory measures as well as identification of appropriate measures. Note that although this TNPP transportation PRA methodology, technical information, data, and example analyses are being provided with the expectation that they could be used to support a request for a 10 CFR 71.12 exemption that will be submitted for approval of the Project Pele transportation package, the ultimate responsibility for the submittal of the transportation safety analysis report and the request for exemption to the NRC is that of the Applicant (i.e., the Project Pele microreactor vendor).

The TNPP PRA model discussed in this report is envisioned to be an in-process model that is updated as the Project Pele microreactor design matures and as refined information (e.g., release fractions) becomes available.

The first step in the development of the risk-informed approach was to develop proposed risk evaluation guidelines. The benefit of having risk evaluation guidelines is that if the risk assessment results derived from evaluating a TNPP transportation package can be found to be acceptable by comparing them to the risk evaluation guidelines, then a key basis for making a risk-informed decision has been satisfied. In addition, if the risk results are found to be unacceptable, then insights from the evaluation can potentially be used to identify design features or operational improvements that reduce the risk to an acceptable level. For this report, PNNL reviewed risk evaluation guidelines that have been developed or endorsed by the NRC for various types of applications for use in determining the acceptability of the estimated risk or risk significance of potential accident sequences that could occur during operation of licensed nuclear facilities. PNNL also reviewed risk evaluation guidelines that are used by DOE in assessing and managing the risk of operation of its nuclear facilities. Based on the results of these reviews, PNNL developed proposed risk evaluation guidelines for use in evaluating the acceptability of the risk from the shipment of a microreactor package containing irradiated fuel. Consistent with existing NRC and DOE guidelines, the proposed transportation guidelines were developed for two receptors that would potentially be exposed to radioactive materials released during a severe transportation accident: (1) a worker involved in the transportation of the TNPP package, and (2) a member of the public that is located close to or is involved in the accident, defined to be the maximally exposed offsite individual (MOI). The proposed guidelines are a composite of the reviewed NRC, DOE, and to some extent International Atomic Energy Agency (IAEA) guidelines, and therefore, are similar to and consistent with existing risk evaluation guidelines developed for purposes other than for transportation of radioactive materials. The proposed risk evaluation guidelines were also developed to be in alignment NRC nuclear safety goals and corresponding proposed Quantitative Health Objectives. The figure-of-merit in the proposed guidelines is total effective dose equivalent (TEDE).

The second step in the development of the risk-informed approach was to develop a TNPP package transportation PRA. The transportation PRA is based on information available from vendor (i.e., BWX Technologies, Inc. or BWXT) design material generated for Phase I of the Project Pele. The approach, data, and information presented in this report provides a demonstration of the PRA development process and provides useable information that serves to illustrate how a TNPP transportation PRA study could be performed to make the case that NRC safety goals are met (e.g., by showing that the potential risk from hypothetical transportation accidents meet the proposed risk evaluation guidelines). Further detailed design and safety analysis information that will be forthcoming in Phase II of the Project Pele is needed to better inform the TNPP transportation PRA. However, to the extent that the information was available in Phase I, it is reflected in the TNPP transportation PRA study presented in this report.

The PRA development process used standard methods acceptable to both DOE and NRC for assessing the safety of nuclear facilities. The process is summarized as follows:

1. Collection of the most current information available on the TNPP transportation package. For transportation purposes, the TNPP is separated into several modules, each of which is transported with its own truck/trailer. For the purposes of this demonstration implementation of the risk-informed approach, just the module containing the reactor system (including used nuclear fuel) and a portion of the primary cooling system, referred to as the Reactor Module, was evaluated. This module contains well over 99% of the radiological inventory. Information collected on this module included system design and configuration information, the estimated radionuclide inventory at various time periods following shutdown of the microreactor, and information on the process for preparing the module for shipment. For this demonstration implementation, it is assumed the reactor is shipped 90 days after shutdown of the microreactor that has operated for three years. The Reactor Module and its transportation configuration are referred to as the TNPP transportation package.

2. Identification of the TNPP transportation package safety functions. These functions are: (1) provide containment of radiological materials, (2) provide radiation shielding, and (3) maintain a criticality-safe configuration.

3. Identification and development of transportation accident scenarios. A standard hazard analysis process was used, which included: (1) identifying possible hazardous conditions that could occur during transportation, (2) postulating accident conditions and assigning likelihood and consequence bins to each accident, (3) screening accident conditions determined to have extremely low likelihood or consequences (e.g., graphite fire, aircraft impact), (4) identifying and assessing a comprehensive set of accident scenarios that are representative of the unscreened accident conditions, (5) grouping the accident scenarios by accident phenomena, and (6)

identifying and developing accident scenarios for each group for which detailed likelihood and consequence analysis is performed. A total of 31 accident scenarios were identified, which included both crash and non-crash scenarios. These scenarios were grouped into 12 accident scenarios, referred to as bounding representative accidents (or BRAs), for detailed analysis.

4. Development of the likelihood of each bounding representative accident. For postulated transportation accident scenarios involving a crash, data sources used to develop the likelihoods included state-level accident data for large trucks and geographic information system (GIS) information for the assumed transportation route, and route-specific Google-street views were used to supplement the likelihood estimation of certain accident scenarios. National accident data for large trucks was also used to supplement the likelihood estimation of certain accident scenarios where state-level data was not available. The assumed origin and destination of the shipment route has been made only for the purposes of analysis and to establish a credible process and pathway for development of the transportation PRA. These assumptions will be revised as necessary to reflect program decisions, objectives, and refinements in future. For postulated transportation accident scenarios that do not involve a crash, component failure data and simplified human reliability analysis was used to estimate potential failures that could result in a release of radioactive material during shipment of the TNPP transportation package.

5. Development of the consequence of each bounding representative accident. The consequence analysis was performed for both the worker and the public (MOI). Radiological dose pathways selected for inclusion in the analysis are those used by the IAEA for determining allowable quantity limits for certified transportation packages for radioactive materials, which are also used by the NRC in 10 CFR Part 71. Specifically, these pathways are direct external gamma and beta radiation doses, inhalation dose, and skin contamination dose from radioactive material released during an accident (doses from the submersion and ingestion pathways are not evaluated because they are negligible contributors to the total dose). Furthermore, the methodologies used by the IAEA to determine the allowable quantity limits for certified transportation packages were nominally used in the consequence assessment in this study for each pathway, with some refinement used to estimate the consequences to the MOI. The IAEA guidance for dose calculations locates an individual 1 m from the release point or source term who is interpreted in this study to be the worker. The IAEA guidance does not distinguish between a worker and the public in its dose calculations. Therefore, for this report, the MOI was assumed to be located 25 m from the release point, which is based on DOT isolation and protective action distance guidance for emergency response to transportation accidents involving high level radioactive material. However, for airborne releases, rather than use the IAEA guidance for estimating the source term, the traditional five-factor formula commonly used in DOE and NRC safety analyses was used for both the CW and the MOI to determine the radiological source term released as a result of the transportation accident. In this case, the factors used were based on NRC and/or DOE data for release of powder, which is conservative for the form of the radiological material contained in the TNPP transportation package, and on expert judgement. Where expert judgement is used, values were selected with an objective to be bounding. For all bounding representative accidents, the dose from inhalation is the largest contributor to the total radiological dose in the transportation PRA.

Note that the PRA does not currently include the contribution from a loss of shielding due to unavailability of sufficient design information on the TNPP transportation package. This gap is expected to be addressed in the next update of this report which is not expected to significantly change the change the risk insights from the TNPP transportation PRA.

The results of the PRA for each of the bounding representative accidents is shown in Figure ES-1 for the worker and in Figure ES-2 for the public. These results also show the proposed risk evaluation guidelines for comparison. As shown, the risks of all the bounding representative accidents except for BRA 3 are less than the risk evaluation guidelines. For BRA 3, which is a severe collision event with a heavy vehicle or unyielding object, the risk evaluation guidelines are slightly exceeded. More refined analysis and/or implementation of compensatory measures will be needed to reduce the risk from this accident. Two additional bounding representative accidents not shown are criticality events. One is a criticality event (i.e., BRA 10) due to control rod withdrawal caused by the impact energy of the accident. In this case, insufficient design information was available on the TNPP transportation package to evaluate this bounding accident for this study. Analysis of this event is expected to be included in the next update of this report. In the other event (BRA 9), the reactor is submerged in water as result of the accident to create a flooded criticality. In this case, the consequences were not determined because the estimated likelihood of this event was calculated to be less than 5E-07 per year which as shown in Figure ES-1 is acceptable regardless of the consequence. Therefore, BRA 9 was not shown as point in the graphic but would fall in the acceptable region.

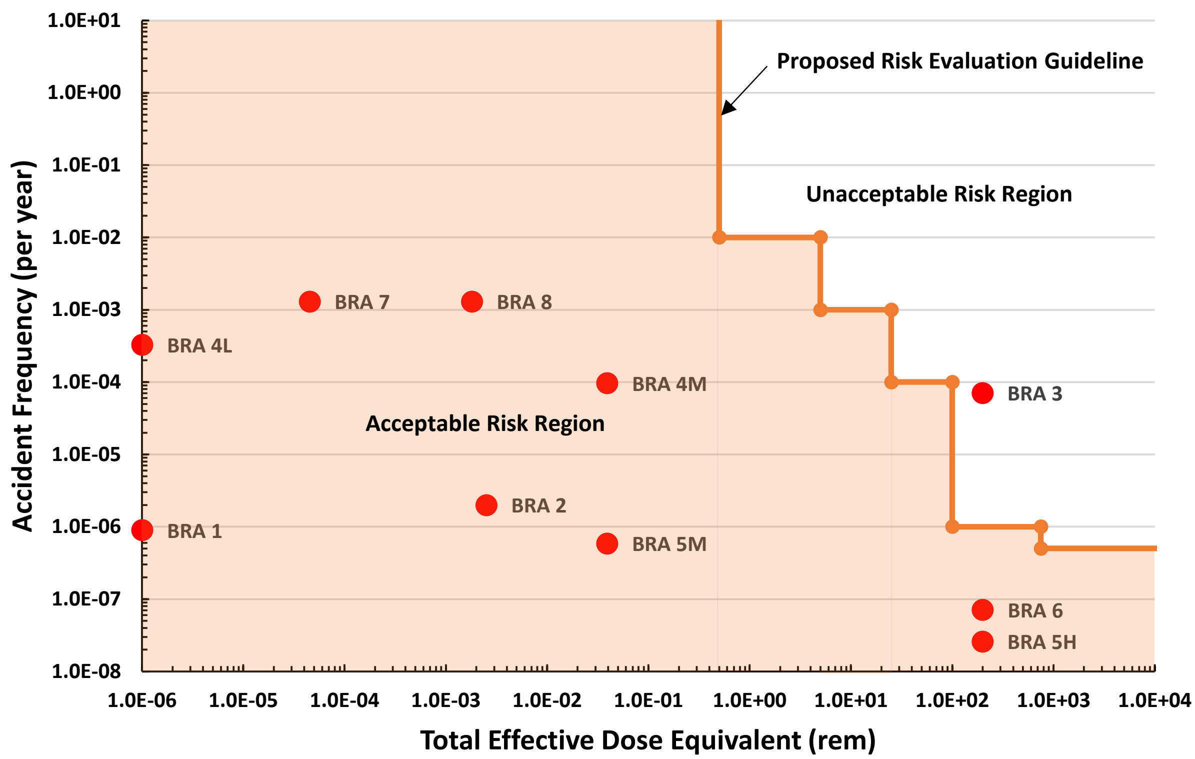

Figure ES-1. PRA Results for the Worker Compared to the Proposed Risk Evaluation Guideline

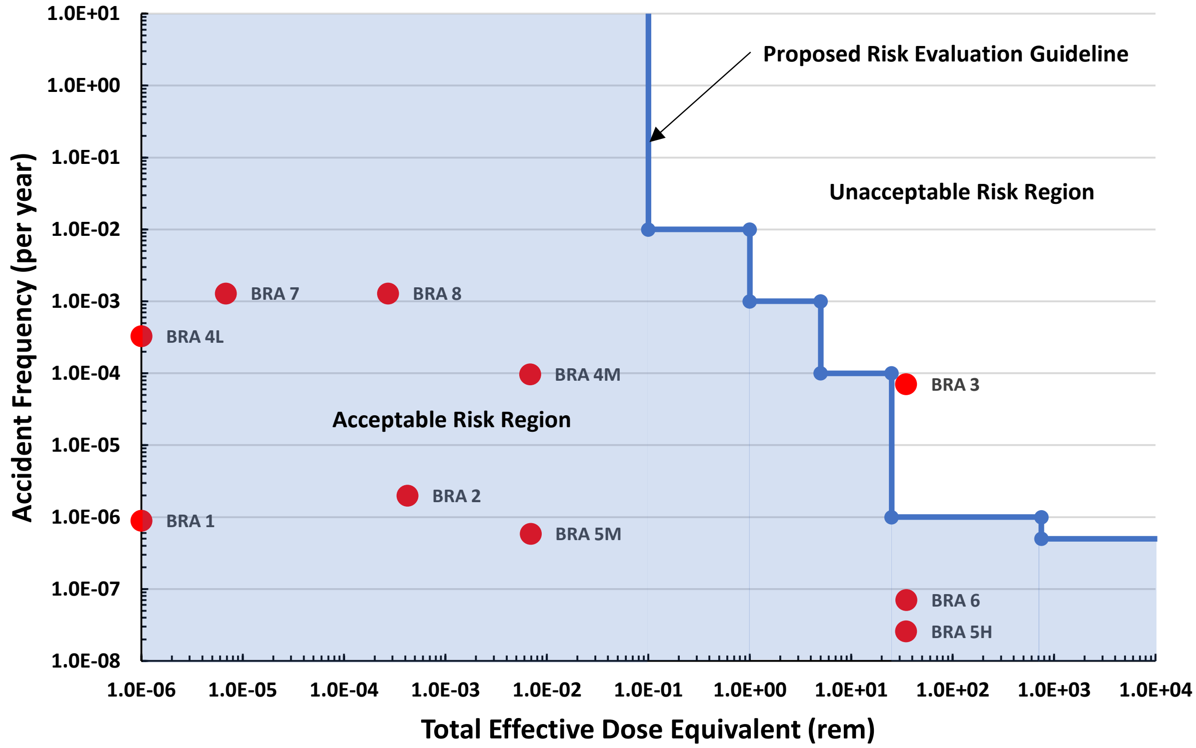

Figure ES-2. PRA Results for the Public Compared to the Proposed Risk Evaluation Guideline

A self-evaluation of the application defense-in-depth and safety margin philosophies to development of the TNPP transportation PRA and to its use for regulatory approval of the TNPP transportation package as part of this study. In general, the evaluation found defense-in-depth and safety margin philosophies were applied to development of the PRA consistent with NRC guidance for risk informed applications and the information available to perform this evaluation.

The next update of this report will include the performance of sensitivity studies that address the impact of key assumptions and sources of uncertainty and address the technical adequacy of the TNPP transportation package risk assessment. Proposed compensatory measures needed or suggested to reduce the risk associated with TNPP transportation will be developed to support the 10 CFR 71.12 exemption process. These measures will be developed after further design information is available to update the PRA and applicable sensitivity studies have been performed.

ACKNOWLEDGEMENTS

The authors gratefully acknowledge the U.S. Department of Defense Strategic Capabilities Office Project Pele Program Manager Dr. Jeff Waksman and the Idaho National Laboratory Manager, Special Reactor Concepts: Senior Technical Advisor for Microreactors Dr. Justin Coleman, for their support of this project.

The authors also acknowledge the contributions to this study by BWX Technologies, Inc., the vendor developing the Project Pele microreactor. Specifically, BWX Technologies, Inc. provided available design information and radionuclide inventories at different reactor shutdown cooling times for the Project Pele microreactor, which greatly enhanced the quality of the risk results and risk insights presented in this report.

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ACRONYMS

A&I	*Analysis & Information Online*
AAR	Association of American Railroads
AED	Aerodynamic Equivalent Diameter
AF	Attenuation Factors
AGR	advanced gas reactor
ALARA	as low as reasonably achievable
AOO	Anticipated Operational Occurrence
API	application programming interface
ARF	airborne release fraction
BDBE	Beyond Design Basis Event
BRA	bounding representative accident
BTU	British thermal unit
BWXT	BWX Technologies, Inc.
CCF	common cause factors
CDF	Core Damage Frequency
CFR	*Code of Federal Regulations*
CIRRPC	Committee on Interagency Radiation Research and Policy Coordination
CMV	commercial motor vehicle
CoC	Certificate of Compliance
CONEX	container express
CRDM	control rod drive mechanism
CRSS	Crash Report Sampling System (NHTSA)
CW	co-located worker
DBA	Design Basis Accident
DBE	Design Basis Event
DEM	Digital Elevation Model
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DR	damage ratio
EA	Environmental Assessment
EAB	Exclusion Area Boundary
EG	Evaluation Guideline
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
FAQ	frequently asked question
FARS	Fatality Analysis Reporting System
FEA	finite element analysis
FGR	Federal Guidance Report
FHE	first harmful event
FHWA	Federal Highway Administration
FIMA	fissions per initial metal atom
FIRST	Fatality and Injury Reporting System Tool
FMCSA	Federal Motor Carrier Safety Administration

ACRONYMS (continued)

FMCSR	Federal Motor Carrier Safety Regulations
FR	*Federal Register*
FSAR	final safety analysis report
GES	General Estimates System
GIS	geographic information system
GVW	gross vehicle weight
HAC	hypothetical accident conditions
HALEU	high-assay low-enriched uranium
HAZOP	Hazards and Operability Study
HEP	Human Error Probability
HM	hazardous material
HMIS	Health Monitoring Instrumentation System
HMR	Hazardous Materials Regulations
HRA	Human Reliability Analysis
HRCQ	highway route controlled quantity
HTGR	high-temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronic Engineers
INL	Idaho National Laboratory
ISA	Integrated Safety Analysis
ISCORS	Interagency Steering Committee on Radiation Standards
ISO	International Organization for Standardization
IHX	intermediate heat exchange (Module)
LBE	Licensing Basis Events
LCF	latent cancer fatality
LERF	Large Early Release Fraction
LPF	leak path factor
LRDM	low dispersible radioactive material
LSA	low specific activity
LWR	light water reactor
MAR	material at risk
MCMIS	Motor Carrier Management Information System
MHE	most harmful event
MOI	maximally exposed offsite individual
MRL	manufacturing readiness level
MTU	metric tons of uranium
MWe	megawatts of electric power
NBD	National Bridge Database
NCT	normal conditions of transport
NEI	Nuclear Energy Institute
NGA	National Governors Association
NGNP	Next Generation Nuclear Plant
NHD-HR	National Hydrography Dataset High Resolution

ACRONYMS (continued)

NHTSA	National Highway Traffic Safety Administration
NRC	U.S. Nuclear Regulatory Commission
OSM	Open Street Maps
PAG	Protective Action Guide
PBMR	Pebble Bed Modular Reactor
PIE	post-irradiation examination
PNNL	Pacific Northwest National Laboratory
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
PyC	pyrolytic carbon
QA	quality assurance
QHG	quantitative health guideline
QHO	quantitative health objective
RF	respirable fraction
RG	Regulatory Guide
RIDM	risk-informed decisionmaking
RIS	Regulatory Impact Summary
ROD	Record of Decision
RPV	Reactor Pressure Vessel
SAR	safety analysis report
SCO	Strategic Capabilities Office
SiC	silicon carbide
SNM	special nuclear material
SSC	structures, systems, and components
SSG	Specific Safety Guide (IAEA)
SSR	Specific Safety Requirements (IAEA)
SSURGO	Soil Survey Geographic (database)
STATSGO	State Soil Geographic (database)
TED	total effective dose
TEDE	total effective dose equivalent
TNPP	Transportable Nuclear Power Plant
TRISO	tri-structural isotropic (particle)
TRL	technology readiness level
UCO	uranium oxycarbide
USDA	United States Department of Agriculture
USGS	United States Geological Survey
WEBTRAGIS	Web-Based Transportation Routing Analysis Geographic Information System
WSMR	White Sands Missile Range

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1. INTRODUCTION

1.1 Background

The U.S. Department of Defense (DOD) Strategic Capabilities Office (SCO) has initiated a project, referred to as Project Pele, for the construction and demonstration of a prototype transportable microreactor. The Final Environmental Impact Statement (EIS) for this project was released by SCO in February 2022 (DOD 2022) and the associated Record of Decision (ROD) was issued in April 2022.[1] In the ROD SCO decided to implement the Proposed Action described in the Final EIS to fabricate a prototype transportable microreactor and reactor fuel at existing off-site commercial facilities and to demonstrate the microreactor at the U.S. Department of Energy's (DOE's) Idaho National Laboratory (INL) site. Moreover, additional future demonstration may be performed at a DOD reservation and, in that case, transportation of the irradiated TNPP would occur on public roads and highways. The joint effort between SCO and DOE, established by interagency agreement, would make use of DOE expertise, material, laboratories, and authority to demonstrate this transportable microreactor.

The U.S. Nuclear Regulatory Commission (NRC), consistent with its role as an independent safety and security regulator, is participating in this project to provide SCO with accurate, current information on the NRC's regulations and licensing processes in connection with construction and demonstration of a transportable microreactor. However, consistent with the non-commercial nature of the project, the prototype transportable microreactor is proceeding under authorization by the Secretary of Energy and does not require an NRC license. The DOD has future plans, which are dependent on the experience gained from this demonstration project, to use transportable microreactors on military installations and potentially in field operations. This will necessitate making shipments of microreactors, potentially containing irradiated nuclear fuel, using transportation systems that are also utilized by the public (e.g., interstate highway system). These shipments would therefore likely be regulated by the NRC and U.S. Department of Transportation (DOT).

Pacific Northwest National Laboratory (PNNL) has been tasked by SCO to address the regulatory challenges associated with safe transport of Transportable microreactor Nuclear Power Plants (TNPPs) containing irradiated nuclear fuel. In a predecessor study, PNNL-31867 (*Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages* [Coles et al. 2021)), PNNL developed a risk-informed regulatory framework (hereafter referred as the framework) for the licensing of the transportation of TNPPs in which irradiated nuclear fuel is an integral component of the TNPP transportation package. This framework report lays out viable regulatory pathways, including decision points for regulatory options and the supporting technical evaluations for those options in phases from near term to long term.

The preferred option for regulatory approval of microreactor transportation packages is to explicitly meet the deterministic requirements of 10 CFR Part 71 ("Packaging and Transportation of Radioactive Material") and be issued a Certificate of Compliance (CoC) by the NRC because then further review and approval of shipments by the NRC would not be necessary. However, a TNPP and its contents will likely not be able to comply with all the NRC regulatory requirements for a Type B or fissile material transportation package under 10 CFR Part 71 (e.g., the test requirements for hypothetical accident

[1] See https://www.cto.mil/pele_eis/.

conditions [HAC] in Section 71.73). The framework developed in PNNL-31867 lays out alternative risk-informed licensing options that are safe and feasible. Risk assessment methods, such as probabilistic risk assessment (PRA), can be used to show comparable safety to that provided by a Type B or fissile material package for surface transport. The framework included guidance on applicable regulations and discusses historical precedence in using risk information for licensing of a prototype TNPP one-time shipment that utilizes risk information to show comparable safety to that provided by a Type B or fissile material package.

The framework report (Coles et al. 2021), as discussed below in Section 2.0, identified that 10 CFR 71.12 ("Specific exemptions") is the most feasible regulatory option for transportation of the prototype TNPP but that it should be supported by a quantitative risk assessment. Though evaluation of the 10 CFR Part 71 requirements which require an exemption could theoretically be qualitative or semi-quantitative, there are significant challenges associated with using qualitative evaluation to demonstrate that transport of an microreactor can be performed at an acceptable level of risk. Concerns include the fact that transport of a microreactor will occur with irradiated fuel which will be a first-of-a-kind endeavor, design and modeling uncertainties, and the potential risk to the public if a transportation accident occurs.

Following the framework report, at the request of SCO, PNNL developed a plan for the development and application of a risk assessment approach to support a risk-informed pathway for NRC and DOT approval of a one-time surface shipment of a microreactor transportation package. This plan, which is documented in PNNL-33524 (*Plan for Development and Application of Risk Assessment Approach for Transportation Package Approval of an MNPP for Domestic Highway Shipment* [Maheras et al. 2021]), identifies the proposed content of a risk-informed exemption request to the NRC for the transport of an TNPP package and, specifically, the Project Pele microreactor.

1.2 Report Purpose and Objective

The purpose of this report is to provide a demonstration implementation of this plan (PNNL-33524) for a hypothetical shipment of the Project Pele microreactor. This demonstration implementation is intended to be used as a guide or template for the development of a hypothetical risk-informed exemption request to the NRC by the Project Pele microreactor vendor for a one-time ground surface shipment. This report only addresses the application of risk information in the TNPP package safety analysis, although this same risk information could also potentially be used in the Environmental Assessment (EA) or EIS, as applicable, required by 10 CFR Part 51 ("Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions") for exemptions. Hence, the determination of population risk for normal conditions of transport (NCT), or incident-free transport, is not addressed in this report.

Demonstration implementation of the plan (PNNL-33524) includes the development of a risk assessment methodology, risk evaluation guidelines, technical information, data, and example analyses that provide a potential template for a vendor to follow when making a request to the NRC. It also addresses important supporting analyses to the PRA such as the treatment of key assumptions and sources of modeling uncertainty and the concept of defense-in-depth and safety margin. Key advantages of using the approach are: (1) increasing the likelihood of successfully obtaining regulatory transportation package approval, (2) informing the design on the relative risk significance of TNPP containment and shielding, and (3) informing the need for transportation compensatory measures as well as identification of appropriate measures. Note that although this TNPP transportation PRA, methodology, technical information, data, and example analyses are being provided with the expectation that they could be used to support a request for a 10 CFR 71.12 exemption that will be submitted for approval of the Project Pele transportation package, the ultimate responsibility for the submittal of the transportation safety analysis report (SAR) and the request for exemption to the NRC is that of the Applicant (i.e., the Project Pele microreactor vendor).

The TNPP PRA model discussed in this report is envisioned to be an in-process model that is updated as the Project Pele microreactor design matures and refined information (e.g., release fractions) becomes available.

1.3 Description of Transportation Package

The microreactor transportation package evaluated in this report is assumed to meet the SCO-defined design goals and requirements for Project Pele and to be consistent with the microreactor concepts evaluated in the associated Final EIS for this project. Per the Final EIS (DOD 2022), the Project Pele prototype transportable nuclear microreactor would generate 1 to 5 megawatts of electric power (MWe) for a minimum of three years of full power operation. It is a tri-structural isotropic (TRISO) fueled high-temperature gas reactor utilizing high-assay low-enriched uranium (HALEU) oxycarbide (UCO). The design consists of multiple modules including: (1) a microreactor module (i.e., Reactor Module), (2) an intermediate heat exchange (IHX) module (i.e., IHX Module), (3) a control module (i.e., Control Module), and (4) a power conversion system module (i.e., Power Conversion Module). The Reactor Module consists of the transportable microreactor with constituent elements like the reactivity control system, portions of the Reactor Gas System (Primary Cooling system) loop, and portions of cooling water shielding system. The IHX Module contains the heat exchanger, the secondary cooling loop, and the inlet piping from the Primary Cooling system loop. The Control Module provides command and control of the TNPP system and contains the safety protection system, process control system, and electrical interconnects. The Power Conversion Module consists of a turbine generator, which converts the transportable microreactor thermal energy to electrical power that would be supplied to an electrical grid when deployed. Other containers may be used to transport interface or interconnecting piping (e.g., for the Primary Cooling system loop) and cabling between modules separately from the other four modules.

For surface transportation purposes, each of these modules would be contained in and integral with separate International Organization for Standardization (ISO)-compliant container express (CONEX) containers having dimensions of about 8 ft wide by 8 ft high by 20 ft long. A standard 20-ft general purpose ISO container is shown in Figure 1-1. Figure 1-2 shows a standard 20-ft ISO container loaded on an Army M872A4 semi-trailer with an Army M915A5 tractor truck.

In preparation for transport, interconnected piping and cabling between modules is disassembled and each module is prepared as a separate transport "package." The transport preparation activities are summarized as follows (BWXT 2022[2]):

[2] BWXT Final Design Report, pages 6-17 and 6-18.

Figure 1-1. Standard 20-ft General Purpose ISO Container

Figure 1-2. An Army M915A5 Tractor with a M872A4 Semi-Trailer Carrying a 20 ft ISO Container[3]

- Power Conversion Module – Disconnect the Power Conversion Module from its secondary pipes, coolant lines, and wiring. Install blank-off covers (cover pipe ends to keep any internal dust in place), coil and stow wiring. Load the module on rollback truck or trailer or use Rough Terrain Cargo Handler to lift onto transport.

[3] See http://www.military-today.com/trucks/m915a5.htm.

- IHX Module and Control Module – Disconnect the IHX Module from secondary and primary pipes (cover pipe ends to keep any internal dust in place). Load the module on rollback truck or trailer. Remove secondary pipes to packaging area (Laydown Yard) for loading onto a separate container. Initiate Reactor Module wireless parameter monitoring, then disconnect and collect Control Module cables, load the module on rollback truck or trailer.

- Reactor Module – Disconnect primary pipes and supports, install blank-off covers, close Grayloc® connectors, and cover pipe ends to keep any internal dust in place. Move primary and secondary pipes to the laydown area (for loading onto a separate container). Assemble Reactor Module trailer package.

Other transportation-related system requirements for the TNPP pertain to maximum weight limits, dose rate limits, and regulatory approval. The maximum weight limits are as follows:

- Reactor container – 42 tons.
- Maximum weight of any other individual containers – 26.5 tons.
- Maximum total weight of all containers – 70 tons.
- Maximum weight for reactor container and additional external shielding – 50 tons.
- Approximate weight of semi-trailer – 12 tons.

The TNPP containers will meet the NRC and DOT regulatory dose rate limits during shipment and are assumed to be transported via truck and trailer. The shipping package for the prototype TNPP Reactor Module will be designed in accordance with NRC requirements in 10 CFR Part 71.

Project Pele is currently in the build and demonstration phase in which the vendor is expected to deliver a full-scale TNPP, which will then undergo up to three years of demonstration testing. Demonstration testing is planned to consist of startup testing at a location on the INL site, transportation between test locations on the INL site, and testing at a second location at the INL site. While the Project Pele prototype microreactor is not currently planned to be shipped off the INL site, in anticipation of possible future shipments of microreactors on roadways accessible to the public and to aid in the develop of this risk-based licensing methodology, this report assumes a single off-site shipment route by domestic highway from the INL site to the White Sands Missile Range (WSMR) located in New Mexico. The assumed origin of INL and destination of WSMR has been made only for the purposes of analysis and to establish a credible process and pathway for development of the transportation PRA. These assumptions will be revised as necessary to reflect program decisions, objectives, and refinements in future.

The design information used in this report to develop the TNPP transportation PRA approach is based on information from vendor design material that was generated during Phase 1, or design phase, of Project Pele. As further detailed design and safety analysis information becomes available in later phases – such as disassembly (and possible reassembly) of the TNPP, and packaging and loading of the various TNPP modules for transport – it will better inform the TNPP transportation PRA. The purpose of this report is to provide information for the development and application of a PRA methodology for the domestic highway transport of the Project Pele prototype TNPP that would support a risk-informed pathway for NRC regulatory approval. Furthermore, this report is meant to be an in-process document that will be updated and revised based on further development and refinements of the Project Pele prototype TNPP design.

1.4 Risk Assessment Scope

Of the four TNPP modules, the Reactor Module and the IHX Module contain radioactive materials, a separate container may be used to transport interconnecting piping (e.g., for the Primary Cooling system loop) and cabling. Almost the entirety of the radioactive inventory generated during reactor operations (well over 99%) is contained within the Reactor Module, which includes the reactor core and associated used fuel. For the purposes of this report, it is assumed that only the Reactor Module will require a risk assessment to support NRC approval of the transportation package using the regulatory pathway described in Section 2.0, and so only the Reactor Module is specifically evaluated. The IHX Module and other containers are expected to contain sufficiently low levels of radioactivity as to not require shipment in a Type B package but, rather, can be shipped as a 10 CFR Part 71 Type A package or as a 49 CFR Part 173 ("Shippers – General Requirements for Shipments and Packagings") industrial (or strong-tight) package as applicable (BWXT 2022[4]). However, should any of these packages contain sufficiently high levels of radioactivity as to require shipment in a Type B package, the risk assessment methodologies and processes delineated in this report would also be applicable to the evaluation of these modules for NRC review and approval.

1.5 Summary of Risk Assessment Approach Sections

The remainder of this report is organized into the sections described below.

1.5.1 Definition of Regulatory Approach

The definition of the regulatory approach for licensing the prototype TNPP transportation package, or Reactor Module, is contained in Section 2. This section identifies the federal regulatory approaches that are available and why the 10 CFR Part 71 exemption process was identified to be the most feasible for the licensing transportation of the prototype TNPP. It also discusses why a risk-informed approach is needed to support this option.

1.5.2 Definition of Safety Goals and Risk Evaluation Guidelines

Section 3 discusses the development of proposed risk evaluation guidelines based on examination of risk thresholds and risk evaluation guidelines used for other nuclear applications and justifies how the proposed risk evaluation guidelines are consistent with the NRC's safety goals, current NRC guidance, and historical practice. This section discusses the importance of considering approaches to risk acceptance and engaging with applicable regulators at the early stages of planning for TNPP transportation, since any possible regulatory changes will take significant time. Even though a risk-informed approach may be achievable under the 10 CFR Part 71 exemption process for the prototype TNPP, it is likely that regulatory changes will be needed in the future when it is anticipated that the transportation of TNPPs will take place on a more frequent basis.

[4] BWXT Final Design Report, Table 7.4-1.

1.5.3 Transportation PRA Methodology, Data, and Results

Section 4 describes the proposed TNPP transportation PRA methodology and presents the application of the proposed PRA approach using Project Pele Phase 1 design information. This demonstration serves to clarify the approach and to provide useful data and assessment information that can be used in a more mature PRA effort. This section includes the overview of the risk assessment approach in Section 4.1. This approach involves a method to identify all credible TNPP accident scenarios that result in release of radioactive material to the environment or in direct radiation exposure to workers or the public combined with identification and accident analysis of a bounding set of representative accidents that are risk significant. Because the Project Pele prototype TNPP will meet many, but not all, of the 10 CFR Part 71 deterministic requirements, a full scope evaluation of all possible accident scenarios is not needed, particularly those accidents resulting in low radiological dose consequences. Section 4.2 provides characterization of the TNPP package radiological material inventory which includes the fuel and fission products and condensed fission gases released during normal operation that may be held-up or exist in other locations of the TNPP or package.

Section 4.3 discusses the identification and definition of possible transportation package safety functions, including removal of heat, prevention of criticality while configured for transport, minimization or prevention of release, and minimization or prevention of direct radiation exposure. The definition of potential transportation accident scenarios, including identification of accident scenarios that could lead to a release of radioactive material, loss of shielding, or criticality is discussed in Section 4.4.

Section 4.5 discusses the development of the likelihoods for the TNPP transportation accident scenarios. This section discusses the transportation accident rate data for large trucks. This section also discusses the collection and analysis of route-specific data on potential transportation hazards, including adjacency to and crossing over bodies of water, the frequency of unyielding objects and hard targets along the road, and identification of restricted heights. For the purposes of demonstrating the PRA approach, a highway (primarily interstate) transport route is assumed for the Project Pele Prototype TNPP from INL to WSMR in New Mexico. The assumed origin of INL and destination of WSMR has been made only for the purposes of analysis and to establish a credible process and pathway for development of the transportation PRA. These assumptions will be revised as necessary to reflect program decisions, objectives, and refinements in the future. This section also discusses the specific development of the accident frequency estimates for highway and non-impact related accident scenarios.

Section 4.6 discusses the transportation accident radiological dose consequences analysis, including definition of source terms (e.g., leak path factors or attenuation factors, damage ratios, airborne release fractions). Consequence analysis is based on determining the source term for the release, the mobility of that source term (i.e., particle size and behavior), and the corresponding radiological dose to a human receptor.

Section 4.7 includes the presentation of the TNPP transportation PRA baseline results which is a combination of the radiological dose consequences and the accident frequency for each defined bounding representative accident scenario. Given the degree of design PRA modeling uncertainty at this point, Section 4.8 provides the results from a set of PRA sensitivity studies that give important general risk insights and insights about the importance of key assumptions. Comparison of the PRA baseline and sensitivity risk results of bounding representative accident scenarios to the proposed risk evaluation guidelines are provided in Sections 4.7 and 4.8, respectively. The risk insights for the baseline PRA and sensitivity studies are discussed in Section 4.9.

1.5.4 Defense-in-Depth and Safety Margin Concerns, and Technical Adequacy of Transportation Risk Assessment

Section 5 defines the defense-in-depth philosophy and includes discussion of safety features/controls that are credited and not credited in the risk assessment. This section discusses potential compensatory measures that are credited in the TNPP transportation PRA or as a defense-in-depth measure. This section also describes the philosophy of incorporating safety margin into design and operation, and how both these philosophies work together with risk assessment and can even be demonstrated using a quantitative risk assessment approach.

Section 6 discusses the technical adequacy of the transportation risk assessment, including definition of the independent peer review process and results, and identification of applicable national standards. The regulating authorities need to have confidence that the information developed from a risk assessment is sound and reliable. Accordingly, the technical content needs to be complete, correct, and accurate, and produce insights with appropriate fidelity to support any decision contemplated.

Section 7 discusses the conclusions from the TNPP transportation risk assessment, including insights from comparison of risk assessment results to risk evaluation guidelines and identification of additional research, analysis needs, and supporting testing to be performed or finalized during Phase 2 of Project Pele.

The appendices in Section 8 include the TNPP inventory and development of the material at risk (MAR) and the TNPP transportation hazardous condition evaluation.

1.6 References

10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," *Code of Federal Regulations*. Accessed October 4, 2022, at https://www.govinfo.gov/content/pkg/CFR-2016-title10-vol2/pdf/CFR-2016-title10-vol2-part51.pdf.

10 CFR Part 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*. Accessed March 28, 2022, at https://www.nrc.gov/reading-rm/doc-collections/cfr/part071/index.html.

49 CFR Part 173, "Shippers – General Requirements for Shipments and Packagings," *Code of Federal Regulations*. Accessed October 4, 2022, at https://www.govinfo.gov/content/pkg/CFR-2010-title49-vol2/pdf/CFR-2010-title49-vol2-part173.pdf.

BWXT, 2022, *Project PELE, Phase IB – Final Design Report (FDR)*, ATL-RPT-110197, BWX Technologies, Inc., Lynchburg, Virginia, March 11, 2022. [Unclassified//DOD UCNI]
Appendix I – Essential Plans for Deployment.
Appendix II – Engineering Drawings.
Appendix III – Engineering Documentation and Analyses.
Appendix IV – Safety in Design Bases and Reports.

Coles, Garill, Steven Short, Steven Maheras, and Harold Adkins, 2021, *Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages*, PNNL-31867, Pacific Northwest National Laboratory, Richland, Washington, August 2021.

DOD, 2002, *Construction and Demonstration of a Prototype Mobile Microreactor Environmental Impact Statement*, February 2022. Available at https://www.cto.mil/pele_eis/.

Maheras, S. J., S. M. Short, G. A. Coles, H. E. Adkins, J. R. Phillips, P. P. Lowry, C. A. Condon, and K. Banerjee, 2021, *Plan for Development and Application of Risk Assessment Approach for Transportation Package Approval of an MNPP for Domestic Highway Shipment*, PNNL-33524, Pacific Northwest National Laboratory, Richland, Washington, December 28, 2021.

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2. DEFINITION OF REGULATORY APPROACH

This section discusses the regulatory pathway that was identified to be the most feasible for the licensing transportation of the prototype Transportable Nuclear Power Plant (TNPP) and why a risk-informed approach is needed to support this option. It also summarizes applicable background and federal regulations as needed to support later discussions of the proposed risk evaluation guidelines and TNPP transportation risk assessment approach.

2.1 Selected Regulatory Approach for Licensing Prototype TNPP Transport

The 10 CFR Part 71 ("Packaging and Transportation of Radioactive Material") exemption process was identified in an evaluation of potential regulatory approval options performed by the Pacific Northwest National Laboratory (PNNL) to be the most feasible approach for licensing transportation of the TNPP package in the near term (e.g., prior to possible future revisions to 10 CFR Part 71 to provide a non-exemption-based process for approval of an TNPP transportation package). Identification of possible regulatory options, evaluation of those options for both the demonstration and production stages of the Project Pele, and selection of the most feasible option for each stage is discussed in PNNL-31867 (*Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages* [Coles et al. 2021]). Though evaluation of which 10 CFR Part 71 requirements require an exemption could potentially be qualitative or semi-quantitative, there are significant challenges associated with demonstrating that transport of an TNPP can be performed at an acceptable level of risk. Challenges include that transport of a TNPP will occur with irradiated fuel which will be a first-of-a-kind endeavor, associated design and modeling uncertainties, and the potential risk to the public if a transportation accident occurs. Accordingly, for the near-term transportation of a TNPP, the 10 CFR 71.12 ("Specific exemptions") process should be supported by quantitative risk assessment. According to 10 CFR Part 71, four regulatory options are available for the approval of a TNPP transportation package in the United States:

1. Demonstration of compliance with environmental test conditions.
2. Demonstration of compliance with alternate environmental test conditions.
3. Request for special package authorization.
4. Request for Specific Exemptions.

These current regulatory pathways or options for obtaining U.S. Nuclear Regulatory Commission (NRC) approval of shipments involving Type B quantities of radioactive materials are discussed in PNNL-31867 (Coles et al. 2021). The preferred regulatory pathway was determined to be through the exemption process (10 CFR 71.12) because exemption(s): (1) can be applicable to multiple shipments (unlike the special package authorization approach under 10 CFR 71.41(d) ["Demonstration of compliance"]), (2) provide for greater flexibility in deviating from the deterministic requirements of 10 CFR Part 71 (compared to the alternative environmental and test conditions approach under 10 CFR 71.41(c)), and (3) have historical precedent (see PNNL-31867).

Compliance with all environmental and test conditions in 10 CFR 71.41(a) and all leak rate and shielding requirements in 10 CFR 71.51 ("Additional requirements for Type B packages") or 10 CFR 71.55 ("General requirements for fissile material packages") after hypothetical accident conditions (HAC) will likely prove challenging for TNPP transportation packages. As stated above, irradiated fuel will be shipped as an integrated component of the package (e.g., loaded in the TNPP). Accordingly, it seems infeasible and cost-prohibitive to acquire a Certificate of Compliance (CoC) for a TNPP package in the near term. A risk-informed approach is used to address the fact that a TNPP transportation package will likely not be able to comply with elements of the deterministic NRC requirements or the uncertainty with meeting the requirement(s).

If a design is unable to meet all the deterministic requirements of 10 CFR Part 71, the preferred option for requesting approval of the TNPP is to request exemptions from the specific requirements that are not practical to meet. Based on insights from past applications, use of an exemption will need to include the following, among the other standard contents of a transportation package approval request:

- Justification that meeting the requirements is "impractical," such as imposing infeasible physical constraints on the shipment;
- Preparation of an Environmental Assessment (EA);
- Obtainment of exemptions concurrently from both applicable NRC and U.S. Department of Transportation (DOT) regulations;
- Identification of compensatory measures such as administrative controls that protect the bases for the exemption by preventing or significantly reducing the likelihood of accident conditions that are outside of the analyzed configurations/conditions; and
- Demonstration that the risk to the public from the shipments is low and comparable to that of other activities regulated by the NRC.

As noted above, the requested exemption from NRC and DOT regulations will require an EA and need to: (1) justify that meeting the federal regulations is not practical (e.g., would impose infeasible restrictions on the design of an engineered containment package that makes it impractical to transport a TNPP), (2) identify administrative controls that protect the bases and assumptions of the risk-informed assessment, and (3) provide demonstration that the risk to the public is acceptably low.

2.2 References

10 CFR Part 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*. Accessed March 28, 2022 at https://www.nrc.gov/reading-rm/doc-collections/cfr/part071/index.html.

Coles, Garill, Steven Short, Steven Maheras, and Harold Adkins, 2021, *Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages*, PNNL-31867, Pacific Northwest National Laboratory, Richland, Washington, August 2021.

3. DEFINITION OF SAFETY GOALS AND RISK EVALUATION GUIDELINES

Regulatory risk evaluation guidelines do not exist for transportation of nuclear material as they do for nuclear power plants. The benefit of having risk acceptance guidelines is that if the risk assessment results derived from evaluating an activity such as a Transportable Nuclear Power Plant (TNPP) transportation package can be found to be acceptable by comparing them to the risk evaluation guidelines, then a key basis for making a risk-informed decision has been satisfied. In addition, if the risk results are found to be unacceptable, then insights from the evaluation can potentially be used to identify design features or operational improvements that reduce the risk to an acceptable level. This section discusses potential risk evaluation guideline approaches and presents proposed risk evaluation guidelines for TNPP transportation package risk that are consistent with the U.S. Nuclear Regulatory Commission's (NRC's) safety goal philosophy, guidance, and historical practice.

3.1 NRC Suggested Risk Evaluation Guidelines Based on QHGs

In general, impacts on the public from transport of nuclear material can occur in two different ways. They can occur as routine radiation exposure during normal operations or from an accident. For routine and chronic exposures, 10 CFR Part 20 ("Standards for Protection Against Radiation") provides regulatory limits and constraints that must be considered in decisionmaking. However, the focus of this section is on accident risk because the risk acceptance guidance for accidents that occur during the transport of radiological materials is not well-covered in the regulations.

For the accident risk impacts of this type of activity, NRC proposes guidance in a report titled *Risk-Informed Decisionmaking for Nuclear Material and Waste Applications* (NRC 2008) (hereafter referred to in this report as the RIDM report) for accepting the risk associated with transportation of nuclear material based on a risk assessment approach such as a Probabilistic Risk Assessment (PRA). The approach involves the use of quantitative health guidelines (QHGs) that are based on the same safety goals that the risk evaluation guidance for nuclear power plants is derived. However, the risk evaluation guidance presented in the RIDM report for the transportation of nuclear material has not been endorsed by NRC (as of yet), and there remains challenges to approving and applying the approach. The RIDM report itself cautions that development of risk evaluation guidelines based on QHGs needs discussion and is ultimately a policy decision. None-the-less, as a starting point to developing risk evaluation guidance for the transportation of a TNPP transportation package, the following is a summary of the proposed QHG approach.

The proposed quantitative health objectives (QHOs) are based on the 1986 NRC Safety Goal Policy statement published in the *Federal Register* (51 FR 30028) for nuclear power plants. NRC expressed this goal qualitatively as "...such a level of safety that individuals living or working near nuclear power plants should be able to go about their daily lives without special concern by virtue of their proximity to these plants." Per the RIDM report, this goal could be translated to the transportation of radioactive materials, as a level of safety such that "individual members of the public who live or work or find themselves in proximity to transported radioactive material should experience negligible additional risk by virtue of their proximity to that activity."

The following is the quantitative definition of the QHOs from the 1986 NRC Safety Goal Policy:

- "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 one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the United States 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 one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."

Based on these QHOs, the RIDM report proposes the following QHGs to define the threshold for negligible accident risk for use as risk evaluation guidelines for the risk associated with transportation of nuclear material:

- Public individual risk of acute fatality (QHG 1) is negligible if it is less than or equal to 5E-07 fatality per year.

- Public individual risk of a latent cancer fatality (LCF) (QHG 2) is negligible if it is less than or equal to 2E-06 fatality per year or 4 mrem per year.

- Public individual risk of serious injury (QHG 3) is negligible if it is less than or equal to 1E-06 fatality per year.

- Worker individual risk of acute fatality (QHG 4) is negligible if it is less than or equal to 1E-06 fatality per year.

- Worker individual risk of LCF (QHG 5) is negligible if it is less than or equal to 1E-05 fatality per year or 25 mrem per year.

- Worker individual risk of serious injury (QHG 6) is negligible if it is less than or equal to 5E-06 fatality per year.

These guidelines are expressed in terms of the risk to an individual member of the public or an individual worker. In essence, these threshold values are the allowed risk to the average individual within those two populations (i.e., public and worker). There are no specific guidelines in the NRC Safety Goal Policy statement for workers. However, based on considerations discussed in the RIDM report, the proposed criterion for workers was that the additional risk of prompt fatality from accidents involving acute exposure should be small in comparison to the same risk faced by United States workers in general but not as small as members of the public who are not formally trained in radiation protection. Accordingly, the RIDM report proposes that the term "small" be quantitatively defined as 2% of the background fatality risk faced by workers across all industries or equivalently 1% of the fatality risk in the higher-risk industries as shown in the quantitative criterion presented above for QHG 4. Similar rationale was used for LCF and serious injury (cancer illness) for quantitative criteria presented above for QHG 5 and QHG 6. For further explanation, see the RIDM report (NRC 2008) or the Summary in PNNL-31867 (*Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages* [Coles et al. 2021]).

Using terminology from the PRAs developed for nuclear power plants, a full-scope Level III PRA[5] is needed to determine the risk of transporting the TNPP packages in terms consistent with the QHGs (i.e., expected health impacts). This involves the following elements: (1) identifying possible accident scenarios that could potentially lead to release of radiological material from the microreactor package or lead to direct exposure to the contents, (2) calculating the likelihood of those accident sequences, (3) determining the physical consequences for possible accident sequences in terms of the extent to which the package will be breached or shielding will be lost, (4) calculating radiological consequences for possible accident sequences in terms of the quantity of radionuclides that is released to the environment, and (5) calculating the consequences of the accident sequences in terms of public and worker health impacts. The risk from those accident sequences to the average individual within the populations of interest needs to be calculated in terms of the health effects measured by the QHGs. The QHGs are presented as “expected values” which are values determined by multiplying each possible outcome by the likelihood each outcome will occur and then summing those values. Therefore, the total risk is determined by multiplying the likelihood and consequences for each accident scenario and summing the risk across scenarios. To understand the acceptability of the accident risk, the total risk results are compared to the QHGs for the two populations discussed above (i.e., the public and worker).

3.2 Development of Risk Evaluation Guidelines Surrogates for Safety Goal QHOs

There are three observations about the approach outlined in Section 3.1 which suggest there are advantages to adjusting the proposed RIDM approach using surrogate metrics:

1. It may not be necessary and would reduce calculational burden to express the release from the accident sequences in terms of rems to workers or the public without determination of health effects. Determination of health effects introduces complexities such as consideration of the varying population along a given transport route. Also, use of surrogates could be particularly helpful in this phase of TNPP design development given the number of sensitivity studies that will need to be performed to address important modeling uncertainties.

2. If the QHGs are expressed as pairs of acceptable likelihood and consequence values in which the consequence is expressed as radiological dose without combining the values, then comparisons can made of these radiological dose threshold limits to the radiological dose limits in relevant federal and international regulations and guidance. This comparison can be used to validate dose threshold limits derived from the QHGs. This substitute risk measure of pairs of likelihood and consequence values can be thought of as a surrogate to the proposed QHGs.

3. If the accident sequence results of a PRA were determined as pairs of likelihood and consequence values, then the PRA results will provide a greater level of information that can be useful for decision making or development of applicable design changes or compensatory measures.

This substitute risk measure of pairs of likelihood and consequence values can be thought of as a surrogate to the proposed QHGs. It is noted that even nuclear power plant PRAs, for which the PRA

[5] Level I PRA determines the Core Damage Frequency (CDF) and Large Early Release Fraction (LERF) and other release categories, Level II PRA determines the quantity and activity of the radioactive material released from the plant, and Level III determines the health consequences to the public.

technology is mature and well-accepted by NRC, are not typically taken to Level III to determine public health impacts. Rather, PRAs used to support risk-informed licensing decisions for stationary light-water reactors (LWRs) produce results in terms of Core Damage Frequency (CDF) and Large Early Release Fraction (LERF) because the risk evaluation guidelines established using these metrics are much more attainable and practical to use than QHGs. Accordingly, CDF and LERF are used as surrogate measures to the QHGs. NRC has issued guidance in Regulatory Guide (RG) 1.174, Revision 3 (*An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis* [NRC 2018]) that stipulates CDF and LERF levels at which a change in a plant's operating license would not be allowed using a risk informed approach. RG 1.174 states that it uses the NRC Safety Goal Policy statement and QHOs to define an acceptable level of risk based on "subsidiary objectives" derived from the safety goals and QHOs. RG 1.174 refers to CDF and LERF risk evaluation criteria (e.g., 1E-04 per year for total CDF and 1E-05 per year for total LERF) as "surrogates" based on the NRC Safety Goal Policy statement and QHOs. In support of these surrogates for the current fleet of light-water nuclear power plants, NRC has demonstrated that these are acceptable metrics for the latent and early QHOs using calculations presented in a NRC memo entitled *Transmittal of Technical Work to Support Possible Rulemaking on a Risk-Informed Alternative to 10 CFR 50.46/GDC 35*, Appendix C, "Quantitative Guidelines from the Framework for Risk-Informing 10 CFR Part 50" (Thadani 2002).

The development of proposed surrogate measures to the QHGs proposed in the RIDM report (NRC 2008) is addressed in Section 3.2.1 through Section 3.2.4. First the selection of dose threshold limits is explored by examining comparable dose limits stipulated or referenced by federal and international regulations and associated guidance. Then the selected dose threshold limits are paired with applicable likelihood limits based on this examination. These limits are then tested and refined to demonstrate that they are equivalent or more conservative than the QHGs proposed in the RIDM report.

Section 3.2.1 discusses the risk evaluation guideline concepts used by the U.S. Department of Energy (DOE) for safety analysis of nuclear facilities; Section 3.2.2 discusses performance criteria for Integrated Safety Analysis (ISA) of nuclear fuel cycle facilities; Section 3.2.3 discusses the referenced dose limit used in the Q system for radiological material package requirements; Section 3.2.4 discusses NRC risk evaluation guidelines used to identify Licensing Basis Events (LBEs) in licensing advanced non-LWRs; and Section 3.2.5 discusses the selection of pairs of radiological dose and likelihood limits and the comparison of those limits to the QHGs proposed in the RIDM report.

3.2.1 Risk Evaluation Guidelines Used by DOE for Nuclear Safety Basis Development

The DOE uses the concept of risk evaluation threshold values to support the nuclear safety basis for non-reactor nuclear facilities but are not based on QHOs. Rather than requiring calculation of the risk of health effects to the public in terms of latent cancers and fatalities, the maximum radiological (or toxic) dose to the nearest member of the public as well as the onsite worker are calculated and then evaluated according to accepted risk evaluation guidelines. The risk assessment approach used to support the allocation of nuclear safety basis controls at a DOE nuclear facility is typically not a PRA but rather a qualitative or semi-quantitative risk informed hazard analysis supported by accident analysis, and if needed, by event and fault trees modeling (like the event and fault trees modeling used in PRA). The DOE guidance on the development of a nuclear safety basis essentially stipulates estimating the likelihood and consequence of identified accident scenarios. DOE refers to its process as "risk ranking" in DOE-STD-3009-2014 (*DOE Standard – Preparation of Nonreactor Nuclear Facility Documented Safety*

Analysis [DOE 2014]). Their concept of "risk ranking" is based on characterizing the risk of an activity or facility in terms of the consequence and likelihood of possible accident sequences. This guidance is used for cited nuclear facilities rather than transport of radiological material but is examined in this report because it is a widely used and accepted approach in the DOE complex.

DOE-STD-3009-2014 provides guidance on using "risk ranking" to support selection of Design Basis Accidents (DBAs) and to identify and evaluate the effectiveness of needed risk controls for a facility. The standard states that if the unmitigated off-site release consequence of an accident exceeds the "Evaluation Guideline (EG)" of 25 rem total effective dose (TED) per year, then controls shall be applied to prevent the accident or mitigate its consequences to below the EG. If unmitigated offsite doses between 5 and 25 rem per year were calculated (i.e., challenging the EG), then controls should be considered. The DOE-STD-3009-2014 standard states that "a prompt fatality would not occur if the whole body absorbed dose received in a few hours is less than 100 rads, therefore, the selection of 25 rem value from a 50-year dose commitment provides protection from acute radiation risk." The standard also states that in the United States, the radiological dose from natural background averages is about 0.36 rem per year and about 25 rem over a lifetime. However, the stipulation in DOE-STD-3009-2014 standard is that 25 rem is not expected or exceeded. It is noted that this dose limit is cited in other regulations. The NRC siting guidelines from 10 CFR Part 100 ("Reactor Site Criteria"), Section 100.11 ("Determination of exclusion area, low population zone, and population center distance"), establish an exclusion zone around a commercial nuclear plant to prevent a total radiation dose to the whole body in excess of 25 rem and the associated acute health risk. Also, 10 CFR 50.34 ("Contents of applications; technical information") regarding engineered safety features for stationary nuclear power reactors requires the applicant for a construction permit to perform a safety assessment that shows that the postulated fission product release from a major accident would not result in a radiation dose in excess of 25 rem total effective dose equivalent (TEDE[6]) to an individual located on the boundary of the exclusion area for a period of 2 hours following the onset of the release, or on the outer boundary of the low population zone for the entire duration of the passage of the plume resulting from the release.

DOE-STD-3009-2014 standard provides guidance on defining consequence and likelihood categories that can be used for rank ranking. DOE-STD-3009-2014 consequence-level categories are defined in terms of radiological (and chemical) dose to an individual receptor. Accident event likelihood intervals are defined for categories ranging from "Anticipated" to "Beyond Extremely Unlikely." The DOE-STD-3009-2014 standard establishes these measures for a member of the public, referred to as the maximally exposed offsite individual (MOI) and for a co-located worker (CW). The MOI is an adult located at the point of maximum exposure on the DOE facility site boundary (or located at some farther distance where an elevated or buoyant radioactive plume is expected to cause the highest exposure). However, because there is no "site boundary" associated with the shipment of an TNPP package, the MOI for the purposes of this assessment is the maximally exposed member of the public. The CW is a worker not necessarily involved in the activity where the release occurs and is assumed to be located 100 m from the facility perimeter or release point. Consideration of the CW may have an analogous application for a TNPP transportation accident given there are workers, such as truck drivers, involved in transport operations. Hence, for the purposes of this assessment, the CW is co-located with the TNPP package during its shipment.

[6] For the purposes of this report, the NRC and DOE terminologies for expressing the sum of internal and external exposures to an individual as the total effective dose equivalent, or TEDE, and total effective dose, or TED, respectively, are equivalent.

DOE-STD-3009-2014 establishes a TED of 25 rem as the nuclear safety limit for the MOI and a TED of 100 rem as the nuclear safety limit for the CW and defines the following consequence and likelihood categories for risk ranking:

- High consequences for the MOI to be a TED >25 rem.
- Moderate consequences for the MOI to be a TED <25 rem but ≥5 rem.
- Low consequences for the MOI to be a TED <5 rem.
- High consequences for the CW to be a TED >100 rem.
- Moderate consequences for the CW to be a TED <100 rem but ≥25 rem.
- Low consequences for the CW to be a TED <25 rem.

DOE-STD-3009-2014 classifies likelihood categories to be Beyond Extremely Unlikely, Extremely Unlikely, Unlikely, and Anticipated and defines the following consequence and likelihood categories for risk ranking:

- Beyond Extremely Unlikely accidents as having a likelihood <1E-06 per year.
- Extremely Unlikely accidents having a likelihood of between 1E-04 and 1E-06 per year.
- Unlikely accidents as having likelihood of between 1E-02 and 1E-04 per year.
- Anticipated accidents as having likelihood of greater than 1E-02 per year.

Risk evaluation guidance using surrogates for the QHGs for evaluating the TNPP transport risk could be informed using these consequence-level and likelihood category definitions along with the guidance in DOE-STD-3009-2014. A hypothetical risk evaluation scheme using a graded risk approach that is consistent with the guidance from the DOE-STD-3009-2014 standard is presented in Table 3-1. Given that the TNPP packages will be designed to be robust, the most important part of this risk evaluation scheme will likely be the lower likelihood, higher consequence criteria, however, higher likelihood, lower consequence criteria is also included. Table 3-1 reflects this by not including proposed risk evaluation criteria for Anticipated accidents. It is assumed that these accidents are mitigated by the TNPP transport package design that meets DOE annual exposure limits of 0.1 rem to the public (MOI) from normal operations per DOE Order 458.1 (*Radiation Protection of the Public and the Environment* [DOE 2011]) and 5 rem to the worker (CW) from normal operations per 10 CFR Part 835 ("Occupational Radiation Protection*")*.

This investigation of potential risk evaluation guidelines concepts based on DOE guidance for nuclear facilities suggests the following:

- A radiological dose of greater than 25 rem to the public and 100 rem to workers is acceptable, if the likelihood of the accident that produces this consequence is 1E-06 per year or less; and is unacceptable if the likelihood of the accident is more than 1E-06 per year.

- A radiological dose of less than or equal to 25 rem to the public and less than or equal to 100 rem to workers is acceptable, if the likelihood of the accident that produces this consequence is less than 1E-04 and greater than 1E-06 per year; and is unacceptable if the radiological dose is greater than 5 rem to the public or greater than 100 rem to workers if the likelihood of the accident is more than 1E-04 per year.

Table 3-1. Hypothetical Radiological Dose Evaluation Guidelines Based on DOE-STD-3009-2014

Annual Accident Frequency (per year)(1)	Radiological Dose Consequence to the MOI(2)	Radiological Dose Consequence to the CW(2)	Risk Acceptability
≤1E-06	>25 rem TED	>100 rem TED	Acceptable
>1E-06	>25 rem TED	>100 rem TED	Unacceptable
≤1E-04 and >1E-06	≤25 TED	≤ 100 rem TED	Acceptable
>1E-04	>5 rem TED	>25 rem TED	Unacceptable
≤1E-02 and >1E-04	≤5 rem TED	≤25 rem TED	Acceptable

(1) The radiological dose consequences are presented as a TED, which is based on integrated committed dose to all receptor organs thereby accounting for external exposures as well as a 50-year committed effective dose.
(2) If the accident frequency is <1E-06 per year, the risk of the accident scenario is generally acceptable regardless of its radiological dose consequence. However, further analysis may be warranted if the consequences are expected to be exceptionally high (e.g., much greater than 25 rem TED to the MOI).

- A radiological dose of less than or equal to 5 rem to the public and less than or equal to 25 rem to the worker is acceptable, if the likelihood of the accident that produces this consequence is greater than 1E-04 per year and less than or equal to 1E-02 per year.

These regions of acceptable and unacceptable risk for MOI and the CW are shown graphically in Figure 3-1 and Figure 3-2, respectively.

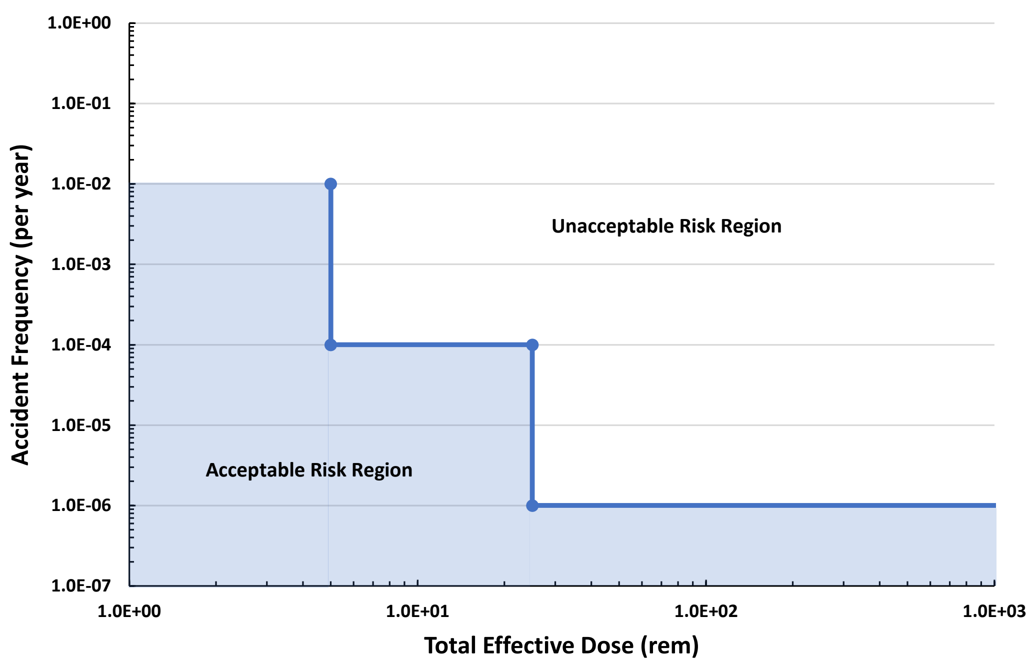

Figure 3-1. Frequency Consequence Chart for MOI Based on DOE-STD-3009-2014

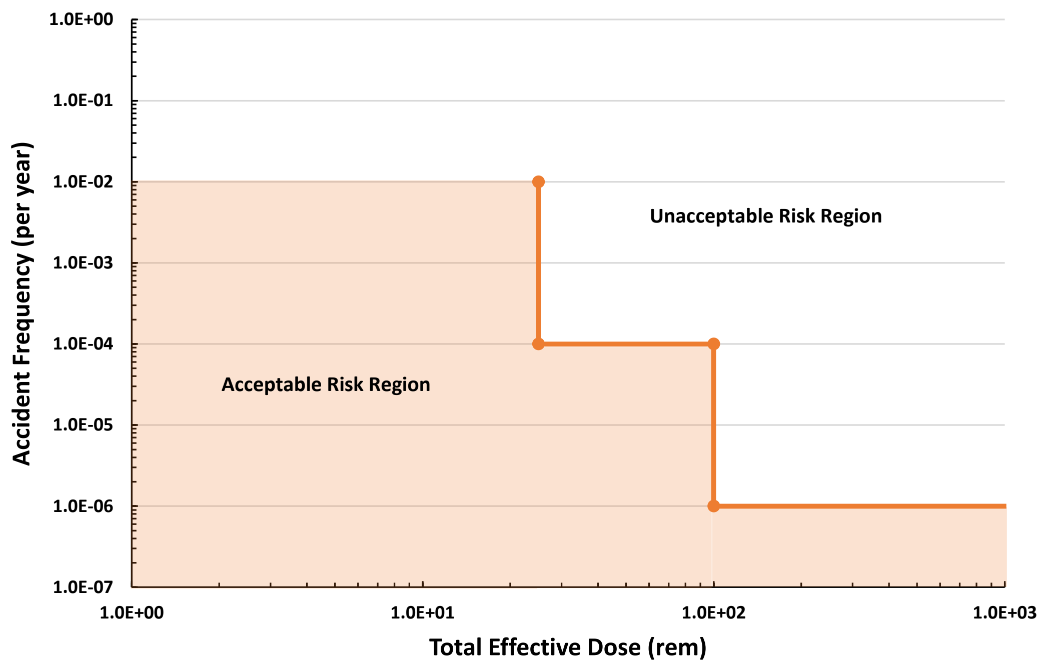

Figure 3-2. Frequency Consequence Chart for CW Based on DOE-STD-3009-2014

3.2.2 NRC Performance Criteria for Integrated Safety Analyses of Nuclear Fuel Cycle Facilities

NRC requirements for applications for licenses to possess and use more than a critical mass of special nuclear material (SNM), which includes certain nuclear fuel cycle facilities, is provided in 10 CFR Part 70 ("Domestic Licensing of Special Nuclear Material"). Subpart H of 10 CFR Part 70 identifies risk-informed performance requirements and requires applicants and existing licensees to conduct an ISA. An ISA, as defined in 10 CFR Part 70, is as follows:

> A systematic analysis to identify facility and external hazards and their potential for initiating accident sequences, the potential accident sequences, their likelihood and consequences, and the items relied on for safety. As used here, integrated means joint consideration of, and protection from, all relevant hazards, including radiological, nuclear criticality, fire, and chemical. However, with respect to compliance with the regulations of this part, the NRC requirement is limited to consideration of the effects of all relevant hazards on radiological safety, prevention of nuclear criticality accidents, or chemical hazards directly associated with NRC licensed radioactive material. An ISA can be performed process by process, but all processes must be integrated, and process interactions considered.

This guidance is used for nuclear nonreactor facilities (e.g., nuclear fuel cycle facilities) licensed by NRC rather than transport of radiological material but is examined in this report because it is an NRC accepted approach.

In essence, ISA is a systematic examination of a facility's processes, equipment, structures, and personnel activities to ensure that all relevant plant and external hazards that could result in unacceptable consequences have been adequately evaluated and appropriate protective measures have been identified. Like PRA, an ISA includes a comprehensive identification of potential accident sequences or events that would result in unacceptable consequences. However, unlike PRA, ISA is not typically performed using extensive fault and event trees analysis. Rather, an ISA is generally qualitative or semi-quantitative as opposed to fully quantitative as in a PRA (e.g., likelihood and consequences are estimated). This methodology, adapted from the chemical process industry, provides for flexibility in the scope and detail of the analysis, depending on the magnitude of the hazards and the nature of the system. Guidance on use of ISA for NRC fuel cycle applications is provided in NUREG-1513 (*Integrated Safety Analysis Guidance Document* [NRC 2001]). This method has been used by the NRC to address the safety in fuel fabrication facilities and in spent fuel storage facilities. The ISA methodology is very similar to the DOE-STD-3009-2014 method for developing the safety basis for non-reactor nuclear facilities discussed in the previous section.

Section 61 of 10 CFR Part 70 ("Performance requirements") defines the performance requirements that must be shown to be met via an ISA. The relevant performance requirements for this assessment define limits in terms of TEDE to the public and to the worker. The risk of high consequence events is to be limited using nuclear safety controls. High and intermediate consequence events are defined as follows:

- High-consequence events that result in an acute worker dose of 100 rem or greater TEDE.
- Intermediate-consequence events that result in an acute worker dose of 25 rem or greater TEDE and which are not high-consequence events.
- High-consequence events that result in an acute dose of 25 rem or greater TEDE to any individual located outside the controlled area.
- Intermediate-consequence events that result in an acute dose of 5 rem or greater TEDE to any individual located outside the controlled area and which are not high-consequence events.

The regulation in 10 CFR 70.61 also specifies the permissible likelihood of occurrence of accident sequences of different consequences. Specifically, high-consequence accident sequences must be "highly unlikely" and intermediate-consequence accident sequences must be "unlikely."
While "highly unlikely" and "unlikely" are not defined in 10 CFR Part 70, NUREG-1520, Revision 2, (*Standard Review Plan for Fuel Cycle Facilities License Applications* [NRC 2015]) provides guidance on defining these likelihood categories. This report specifies that applicants to the NRC may choose to provide quantitative definitions of these terms, and then provides one example of quantitative guidelines that are acceptable to show compliance with 10 CFR 70.61. These guidelines are as follows:

- Unlikely events, as applied to individual accident sequences identified in the ISA, have a likelihood of less than 1E-04 per event, per year.
- Highly Unlikely events, as applied to individual accident sequences identified in the ISA, have a likelihood of less than 1E-05 per event, per year.

These quantitative guidelines are used to define the largest likelihood values that would be acceptable limits. Definitions based on lower limits are also acceptable.

Risk evaluation guidance using surrogates for the QHGs for evaluating the TNPP transport risk could be informed using these consequence-level and likelihood category definitions along with the guidance in NUREG-1520. A hypothetical risk evaluation scheme using a graded risk approach that is consistent with the guidance in NUREG-1520 is presented in Table 3-2.

Table 3-2. Hypothetical Radiological Dose Evaluation Guidelines
Based on 10 CFR Part 70 and NUREG-1520

Annual Accident Frequency (per event, per year)	Radiological Dose Consequence to the Offsite Public(1)	Radiological Dose Consequence to the Worker(1)	Risk Acceptability
<1E-05	≥25 rem TEDE	≥100 rem TEDE	Acceptable
≥1E-05	≥25 rem TEDE	≥100 rem TEDE	Unacceptable
<1E-04 and ≥1E-05	≥5 and <25 rem TEDE	≥25 and < 100 rem TEDE	Acceptable
≥1E-04	≥5 rem TEDE	≥25 rem TEDE	Unacceptable
≥1E-04	<5 rem TEDE	<25 rem TEDE	Acceptable

(1) The radiological dose consequences are presented as a TEDE, which is based on integrated committed dose to all receptor organs thereby accounting for external exposures as well as a 50-year committed effective dose equivalent.

This investigation of potential risk evaluation guidelines concepts based on NRC requirements and guidance for nuclear facilities licensed to possess and use more than a critical mass of SNM suggests the following:

- A radiological dose of 25 rem or greater to the public and 100 rem or greater to workers is acceptable, if the likelihood of the accident that produces this consequence is less than 1E-05 per year per event; and is unacceptable if the likelihood of the accident is 1E-05 per year or greater per event.

- A radiological dose of 5 rem or greater but less than 25 rem to the public and 25 rem or greater and less than 100 rem to workers is acceptable, if the likelihood of the accident that produces this consequence is less than 1E-04 and greater than or equal to 1E-05 per year per event; and is unacceptable if the radiological dose is 5 rem or greater to the public or 25 rem or greater to workers if the likelihood of the accident is 1E-04 per year or greater per event.

- A radiological dose of less than 5 rem to the public and less than 25 rem to workers is acceptable if the likelihood of the accident that produces this consequence is greater than 1E-04 per year per event.

These regions of acceptable and unacceptable risk for offsite public and the worker are shown graphically in Figure 3-3 and Figure 3-4, respectively.

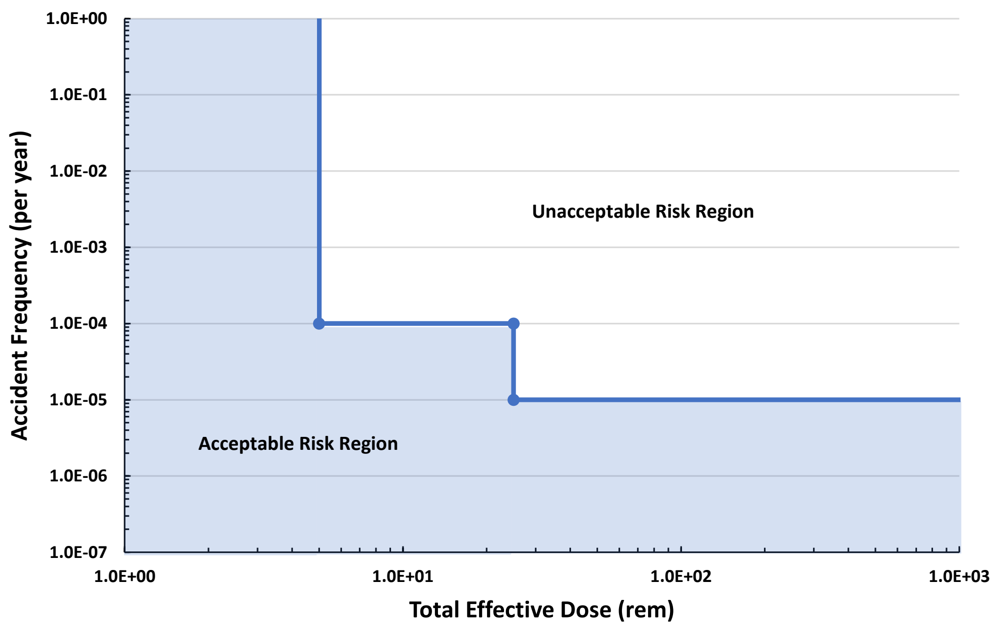

Figure 3-3. Frequency Consequence Chart for Offsite Public
Based on 10 CFR Part 70 and NUREG-1520

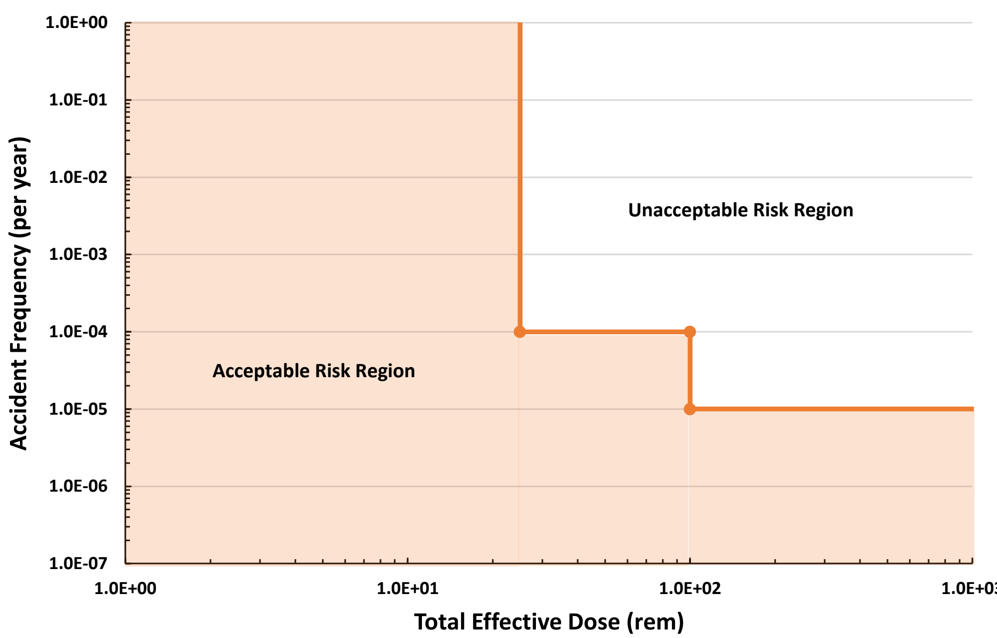

Figure 3-4. Frequency Consequence Chart for Worker
Based on 10 CFR Part 70 and NUREG-1520

3.2.3 Risk Reference Used in Developing the IAEA Q System

The Q system was developed by United Kingdom researchers (MacDonald and Goldfinch 1983) for the International Atomic Energy Agency (IAEA) to support regulation of transport of radioactive materials. The Q system defines the "quantity" limits, in terms of so-called A_1 and A_2 values for radionuclides that are allowed in a Type A package (IAEA 2014, IAEA 2018). These limits are also used for several other purposes in the Transport Regulations (*Regulations for the Safe Transport of Radioactive Material*, IAEA Specific Safety Requirements No. SSR-6 [IAEA 2018]), such as in specifying package activity leakage limits for other packages (e.g., Type B(U), Type B(M), or Type C packages). The content limits are set to ensure that the radiological consequences of severe damage to a Type A package are acceptable and design approval by the competent authority is not required, except for packages containing fissile material. The more robust Type B(U) or Type B(M) packages require testing that takes into account a large range of accidents which expose packages to severe dynamic forces. The use of consequence and likelihood pairs in the Q system as risk acceptance criteria is not obvious, but this guidance was, none-the-less, examined because it pertains directly to transport of radiological material.

Under the Q system, a series of exposure pathways is considered, each of which might lead to persons in the vicinity of a Type A package involved in a severe transport accident receiving external or internal radiation exposure. The effective dose to a person exposed in the vicinity of a transport package following an accident was set to not exceed 50 mSv (and not exceed specified organ and lens of the eye limits). This value of 50 mSv or 5 rem was essentially the annual dose limit for radiation workers and is also the occupational dose limit per year for general employees in the United States per 10 CFR 835.202 ("Occupation dose limits for general employees") of 10 CFR Part 835. For calculating radiological dose, IAEA Specific Safety Guide No. SSG-26 (*Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material* [IAEA 2014]) states that a human receptor is assumed to be 1 m from the damaged package and to remain at this location for 30 minutes. This is stated as being a "cautious judgment" of the incidental exposure of persons initially present at the scene of an accident. This assumption does not impact the allowed reference dose of 5 rem that was selected but does imply that the IAEA regulators thought that the receptor could be in very close proximity to the damaged package.

The table of A_1 and A_2 values provided in Appendix A of 10 CFR Part 71 ("Packaging and Transportation of Radioactive Material") presents the allowed activity (i.e., Terabecquerel or Curies) and specific activity (i.e., Terabecquerel or Curies) per gram for each radionuclide which correlates to the quantity limit of each radionuclide allowed before more robust packaging is required. As such, the allowed A_2 activity is exceeded for a given radionuclide if the A_2 quantity limit is exceeded. In practice, there will be multiple radionuclides present, therefore, 10 CFR Part 71, Appendix A presents a sum-of fractions approach to determining whether the A_2 values are exceeded.

The analysis of accidents that could damage a package uses the reference dose of 5 rem to judge when a Type A package is insufficient to limit the transportation risk of the package. Using the more robust Type B package over a Type A package provides a high level of confidence that 5 rem is not exceeded if the package is damaged. This implied consequence limit of 5 rem and the fact that release from a damaged Type B package is highly unlikely, suggests the following:

- A radiological dose of 5 rem is acceptable, if the likelihood of the accident that produces this consequence is highly unlikely or less; and is unacceptable if the likelihood of the accident is more than highly unlikely.

3.2.4 NRC Endorsed Risk-Informed Methodology in Support of Licensing Advanced Reactor Design

The nuclear industry has produced guidance for a risk-informed performance-based and technology-inclusive process to inform licensing of advanced non-LWR designs. This involves a risk informed approach for selection of LBEs; safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth adequacy as described in Nuclear Energy Institute (NEI) 18-04, Revision 1 (*Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development* [NEI 2019]). The approach uses a set of frequency-consequence criteria like the likelihood-consequence criteria being proposed and discussed in the preceding section.

The approach presented in NEI 18-04 was developed because CDF and LERF measures may not be applicable to non-LWRs. The phenomena associated with core damage and substantial release of radiological material can be significantly different for advanced non-LWRs, and therefore, the concept of CDF and LERF is not be necessarily comparable.

NRC endorsed the methods described in NEI 18-04 for informing the licensing basis and content of applications for permits, licenses, certifications, and approvals for non-LWRs in RG 1.233, Revision 0 (*Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors* [NRC 2020]). None-the-less, the guidance document presents the frequency-consequence evaluation plot shown in Figure 3-3 in which accidents whose likelihood and consequence fall above the blue line are considered to represent unacceptable risk. Accordingly, this risk must be addressed in the safety basis to ensure that it is controlled below the blue line. The y-axis is the event frequency per year and the x-axis is the 30-day total effective dose equivalent (rem) at the Exclusion Area Boundary (EAB).

NEI 18-04 emphasizes that the frequency-consequence target line shown in Figure 3-5 is not to be considered as a demarcation of acceptable and unacceptable results. Rather, it "provides a general reference to assess events, SSCs, and programmatic controls in terms of sensitivities and available margins." This point is further emphasized by the NRC staff in RG 1.233:

> The staff emphasizes the cautions in NEI 18-04 that the F-C [frequency-consequence] target figure does not depict acceptance criteria or actual regulatory limits. The anchor points used for the F-C target figure are expressed in different units, timescales, and distances than those used in NRC regulations to provide common measures for the evaluations included in the methodology.[7] The F-C target provides a reasonable approach for use within a broader, integrated approach to determine risk significance, support SSC classification, and confirm the adequacy of defense-in-depth.

[7] An example provided in RG 1.233 is the anchor point at an event sequence frequency of 5E-07 per plant year and TEDE at the EAB of 750 rem for the 30-day period following the onset of a potential release. This anchor point is used to define a sliding F-C target in the region of potential low frequency, high consequence scenarios for use in assessing the importance of SSCs and other measures to provide defense-in-depth. A traditional measure used to assess risk in the low frequency, high consequence domain is the NRC's safety goals. However, the anchor point is not intended to directly represent the QHOs for either early or latent health effects. The methodology described in NEI 18-04 includes a separate assessment of a design against the QHOs for the integrated risks over all the LBEs.

The methodology proposed in this report includes this broader integrated approach by considering safety margins, defense-in-depth, and the results of sensitivity studies.

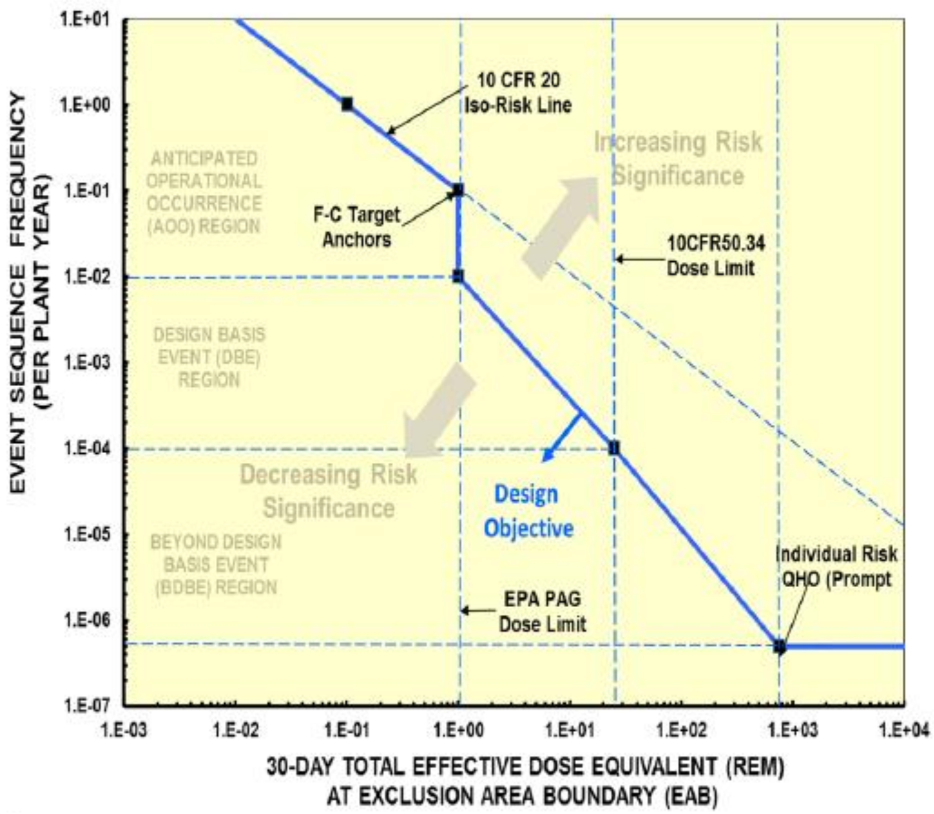

Figure 3-5. Frequency-Consequence Targets from NEI 18-04, Revision 1 (NEI 2019)

The events of interest are the Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), Beyond Design Basis Events (BDBEs), and DBAs which as a collection are referred to LBEs. AOOs are anticipated events expected to occur one or more times during the life of a nuclear power plant. Event sequences with mean frequencies of 1E-02 per year and greater are classified as AOOs. DBEs are infrequent event sequences that are not expected to occur in the life of a nuclear power plant but are less likely than AOOs. Event sequences with mean frequencies of 1E-04 per year to 1E-02 per year are classified as DBEs. BDBEs are rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more Reactor Modules but are less likely than a DBE. Event sequences with mean frequencies of 5E-07 per year to 1E-04 per year are classified as BDBEs.[8] DBAs are postulated event sequences used to set design criteria and performance objectives for the design of safety related SSCs. DBAs are derived from the DBEs by prescriptively assuming that only safety related SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits (i.e., dose at the EAB would not exceed of 25 rem TEDE).

[8] Event sequences with upper 95th percentile frequencies less than 5×10^{-7} per year are retained in the PRA results and used to confirm there are no cliff edge effects. They may also be taken into account in the risk-informed, performance-based defense-in-depth evaluation.

For low-frequency AOOs (i.e., events frequencies between 1E-01 and 1E-02 per year), Figure 3-5 shows that the radiological dose consequence should not exceed 1.0 rem which corresponds to the U.S. Environmental Protection Agency (EPA) Protective Action Guide (PAG) limit to avoid the need for offsite emergency response for any AOO. For high-frequency AOOs (i.e., events having a frequency greater than 1E-01 per year), the figure shows that the radiological dose consequences are based on the iso-risk profile defined by the annual exposure limits of 10 CFR Part 20 ("Standards for Protection Against Radiation"; i.e., 100 mrem per year).

For DBEs (i.e., events frequencies between 1E-02 and 1E-04 per year), Figure 3-5 shows that the allowed radiological dose consequence range from 1 rem at 1E-02 per year to 25 rem at 1E-04 per year. The guidance states that this aligns with the dose calculated at the EAB for the 30-day period following onset of the release and aligns the lowest frequency DBEs to the limits in 10 CFR 50.34 (dose at the EAB would not exceed of 25 rem TEDE).

For BDBEs (i.e., events frequencies between 1E-04 and 5E-07 per year), Figure 3-5 shows that the allowed radiological dose consequence range from 25 rem to 750 rem. The guidance states that these criteria ensure that the QHO for early health effects is not exceeded for individual BDBEs.

A hypothetical risk evaluation scheme using a risk matrix approach based on a conservative interpretation of the guidance in NEI 18-04 is presented in Table 3-3 as an illustration of a surrogate approach to the QHGs. It is considered conservative because it uses a stair-step risk acceptance line that if it were plotted in Figure 3-5 would meet the diagonal lines shown in the plot at the top of the step but fall below the diagonal line at the bottom of the step. The dose limits shown in Figure 3-5 were not applied to workers because that level of differentiation was not made in the NEI 18-04 guidance. It is expected that after controls are applied that LBEs will not release any radioactive material.

Table 3-3. Hypothetical Radiological Dose Evaluation Guidelines Based on NEI 18-04

Annual Accident Frequency (per year)(1)	Radiological Dose Consequence to the Offsite Public(2)	Radiological Dose Consequence to the Worker(2)	Risk Acceptability
≤5E-07(3)	>25 rem TEDE(3)	NA	Acceptable
>5E-07	>25 rem TEDE	NA	Unacceptable
≤1E-04 and >5E-07	≤25 rem TEDE	NA	Acceptable
>1E-04	>25 rem TEDE	NA	Unacceptable
≤1E-02 and >1E-04	≤1 rem TEDE	NA	Acceptable
>1E-02	>1 rem TEDE	NA	Unacceptable
>1E-02	≤100 mrem TEDE	NA	Acceptable

(1) Determination of the accident frequency should account for multiple shipments per year, if applicable.
(2) The radiological dose consequences are presented as a TEDE, which is based on integrated committed dose to all organs thereby accounting for direct exposure as well the 50-year committed effective dose equivalent.
(3) If the accident frequency is <5E-07 per year, the risk of the accident scenario is generally acceptable regardless of its radiological dose consequence. Event sequences with frequencies less than 5E-07 per year are retained in the PRA results and used to confirm there are no cliff edge effects. They may also be taken into account in the risk-informed, performance-based evaluation of defense-in-depth.

This investigation of potential risk evaluation guidelines concepts based on NRC guidance for licensing non-LWRs suggests the following:

- A radiological dose of greater than 25 rem to the public is acceptable, if the likelihood of the accident that produces this consequence is 5E-07 per year or less; and is unacceptable if the likelihood of the accident is more than 5E-07 per year.

- A radiological dose of less than or equal to 25 rem to the public is acceptable, if the likelihood of the accident that produces this consequence is 1E-04 per year or less and greater than 5E-07 per year; and a radiological dose of greater than 25 rem is unacceptable if the likelihood of the accident is more than 1E-04 per year.

- A radiological dose of less than or equal to 1 rem to the public is acceptable, if the likelihood of the accident that produces this consequence is greater than 1E-04 per year but less than or equal to 1E-02 per year; and a radiological dose of greater than 1 rem is unacceptable if the likelihood of the accident is more than 1E-02 per year.

- A radiological dose of 100 mrem to the public is acceptable for if the likelihood of the accident that produces this consequence is greater than 1E-02 per year.

These regions of acceptable and unacceptable risk are shown graphically in Figure 3-6.

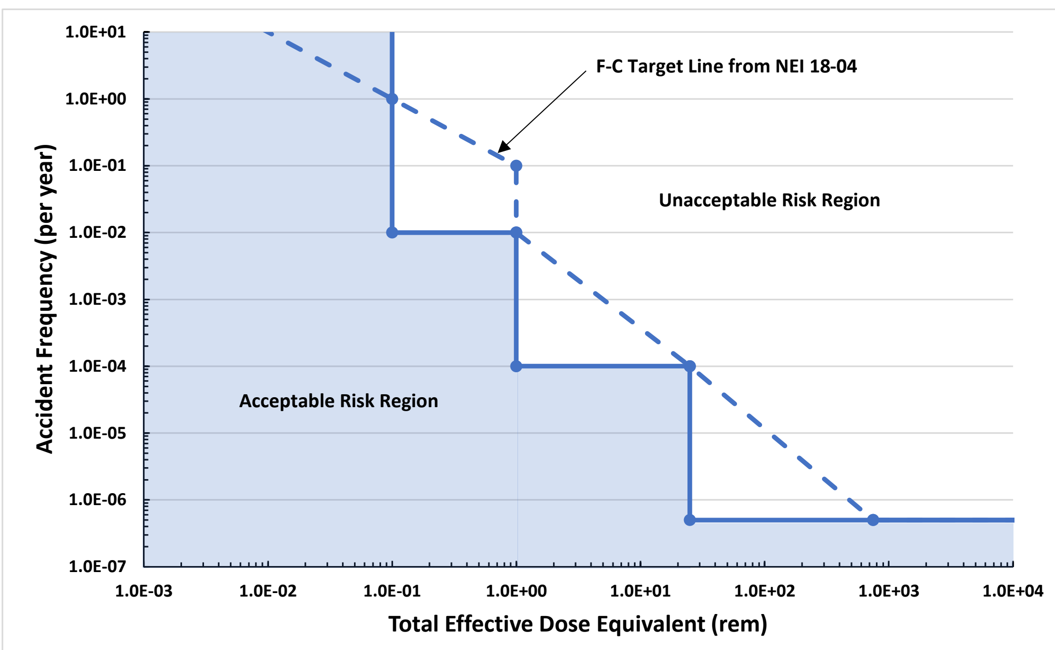

Figure 3-6. Frequency Consequence Chart for the Offsite Public Based on NEI 18-04

3.2.5 Selection of Dose and Likelihood Limits as Surrogates to the Safety Goal QHOs

This section discusses the selection of pairs of radiological dose and likelihood limits and the comparison of those limits to the QHGs proposed in the RIDM report (NRC 2008).

Table 3-4 summarizes risk limits from the DOE-STD-3009-2014 "risk ranking" criteria process for nuclear facilities, NRC performance criteria for nuclear fuel facilities, the Q system, and risk-informed licensing of advanced non-LWRs. These pairs of consequence and likelihood criteria are used to help develop risk evaluation guidelines surrogate measures to the NRC Safety Goal Policy QHOs. These likelihood-consequence pairs based on the four sources are consistent and complimentary.

Table 3-4. Summary of Relevant Risk Limits from Other Applications

Dose Limit	DOE Risk Ranking of Accident Risk (DOE-STD-3009-2014)	Performance Criteria for ISA of Nuclear Fuel Facilities (NUREG-1520)	Q System Reference Dose	NRC Risk-Informed Licensing of Non-LWRs
750 rem	Not Applicable	Not Applicable	Not Applicable	A radiological dose of greater than 750 rem is acceptable if the likelihood of the accident is less than 5E-07 per year.
25 rem	A radiological dose of 25 rem to the public and 100 rem to workers is acceptable if the likelihood of the accident is more than 1E-06 per year but less than 1E-04 per year.	A radiological dose of 25 rem to the public and 100 rem to workers is acceptable if the likelihood of the accident is more than 1E-05 per year but less than 1E-04 per year.	Not Applicable	A radiological dose of 25 rem is acceptable if the likelihood of the accident is more than 5E-07 per year and less than 1E-04 per year.
5 rem	A radiological dose of 5 rem to the public and 25 rem to workers is acceptable if the likelihood of the accident is more than 1E-04 per year.	A radiological dose of 5 rem to the public and 25 rem to workers is acceptable if the likelihood of the accident is more than 1E-04 per year.	A radiological dose of 5 rem is acceptable if the likelihood of the accident is highly unlikely.	Not Applicable
1 rem	Not Applicable	Not Applicable	Not Applicable	A radiological dose of 1 rem to the public is acceptable if the likelihood of the accident is more than 1E-04 per year and less than 1E-02 per year.
100 mrem	Not Applicable	Not Applicable	Not Applicable	A radiological dose of 100 mrem to the public is acceptable if the likelihood of the accident is more than 1E-02 per year.

A selected conservative combining of the criteria above that could be used to produce a risk evaluation guideline scheme is described below.

- A radiological dose of 750 rem or greater to the public is acceptable, if the likelihood of the accident that produces this consequence is 5E-07 per year or less; and is unacceptable if the likelihood of the accident is more than 5E-07 per year. Though not cited for workers in the applications examined, this limit could be also considered applicable to workers. Though the radiological dose limit is very high and could lead to acute death if received as the result of a single event, this consequence is only considered acceptable if the frequency of such an event is 5E-07 per year or less and in consideration of the overall integrated assessment.

- A radiological dose of 25 rem or greater to the public and 100 rem or greater to workers is acceptable, if the likelihood of the accident that produces this consequence is 1E-06 per year or less; and is unacceptable if the likelihood of the accident is more than 1E-06 per year.

- A radiological dose of 5 rem or greater to the public and 25 rem or greater to workers is acceptable, if the likelihood of the accident that produces this consequence is 1E-04 per year or less; and is unacceptable if the likelihood of the accident is more than 1E-04 per year.

- A radiological dose of 1 rem or greater to the public and 5 rem or greater to workers is acceptable, if the likelihood of the accident that produces this consequence is 1E-03 per year or less; and is unacceptable if the likelihood of the accident is more than 1E-03 per year. At an accident frequency of 1E-03 per year a dose limit for the worker is not cited in the applications examined. However, as discussed earlier, 10 CFR Part 835 defines an occupational dose limit of 5 rem per year TED for general employees. While this anchor point has different timescales and distances than those used for DOE safety analyses, it is used here to provide a common measure for the evaluations included in the methodology.

- A radiological dose of 100 mrem or greater to the public and 500 mrem or greater to workers is acceptable if the likelihood of the accident that produces this consequence is 1E-02 per year or less; and is unacceptable if the likelihood of the accident is more than 1E-02 per year. At accident frequencies less than 1E-02 per year a dose limit for the worker is not cited in the applications examined. However, DOE guidance specifies that facility-specific annual administrative control levels should be established. An administrative control level for radiological workers of 500 mrem per year is common at most DOE sites, such as the Hanford Site (DOE 2016). Again, while this anchor point has different timescales and distances than those used for DOE safety analyses, it is used here to provide a common measure for the evaluations included in the methodology.

It remains to be shown that the risk associated with selected radiological dose and likelihood limits above are encompassed by the QHGs. If these limits cannot be shown to be encompassed by the QHGs, then they might not serve as justifiable surrogates to QHGs. To this end, a simplified approach for converting radiological dose to health effects is described below and then used to check to see whether the selected radiological dose and likelihood limits appear to be encompassed by the QHGs.

A DOE memorandum from the Office of Environmental Policy and Guidance dated August 9, 2002 (Lawrence 2002) provides guidance on calculating radiation risk estimates from dose using a technical report attached to the memorandum by the Interagency Steering Committee on Radiation Standards

(ISCORS[9]), *A Method for Estimating Radiation Risk from TEDE* (ISCORS Technical Report No. 1), dated July 2002. The memorandum states that exposure-to-risk estimates are from a tabulation in a September 1999 report, *Cancer Risk Coefficients for Environmental Exposure to Radionuclides – Federal Guidance Report No. 13* (EPA 1999).

The ISCORS report attached to the memorandum is stated to supersede the 1992 Committee on Interagency Radiation Research and Policy Coordination (CIRRPC) guidance and recommends that agencies use a conversion factor of 6E-04 fatal cancers per TEDE (rem) for mortality and 8E-04 cancers per rem for morbidity when making qualitative or semi-quantitative estimates of risk from radiation exposure to members of the public. The TEDE-to-risk factor provided by ISCORS in Technical Report No. 1 is based upon a static population with characteristics consistent with the United States population. The memorandum states that there are no separate ISCORS recommendations for workers. However, it recommends that for workers (adults), a risk of fatal cancer of 5E-04 per rem and a morbidity risk of 7E-04 per rem may be used. A more precise conversion could be made if the exact radionuclide inventory of material generating the dose were known.

The QHGs discussed in Section 3.1 are presented in Table 3-5 but organized by receptors and levels of health risk. It shows the acceptance criteria for the public and the worker for three levels of health concern: (1) acute fatality, (2) LCF, and (3) serious injury which in the context of radiological risk is interpreted to mean illness from cancer. Table 3-5 shows that the risk acceptance threshold (in terms of expected value) is lower for acute fatality compared to LCF or cancer illness indicating less tolerance for this type of risk. The selected pairs of radiological consequence and likelihood limits identified above as candidate surrogates to the QHGs do not address levels of health concern. However, this is addressed in the assessment performed below to see how these selected likelihood-dose limits compare to the QHGs and whether they are encompassed by the QHGs (i.e., conservative compared to the QHGs).

Table 3-5. NRC Proposed QHGs from Interpretation of Safety Policy Statement (NRC 2008)

Receptor	Acute Fatality	Latent Cancer Fatality	Serious Injury (Cancer Illness)
Public	QHG-1 - Public individual risk of acute fatality is negligible if it is less than or equal to 5E-07 fatality per year.	QHG-2 - Public individual risk of a LCF is negligible if it is less than or equal to 2E-06 fatality per year or 4 mrem per year	QHC-3 - Public individual risk of serious injury is negligible if it is less than or equal to 1E-06 injury per year.
Worker	QHG-4 - Worker individual risk of acute fatality is negligible if it is less than or equal to 1E-06 fatality per year.	QHG-5 - Worker individual risk of LCF is negligible if it is less than or equal to 1E-05 fatality per year or 25 mrem per year.	QHG-6 - Worker individual risk of serious injury is negligible if it is less than or equal to 5E-06 injury per year.

[9] The Interagency Steering Committee on Radiation Standards (ISCORS) is comprised of eight Federal agencies, three Federal observer agencies and two state observer agencies to facilitate consensus on acceptable levels of radiation risk to the public and workers and promote consistent risk approaches in setting and implementing standards for protection from ionizing radiation. Available at https://www.iscors.org.

Table 3-6 presents an evaluation of the selected likelihood-dose limits based on the evaluation described up to this point to see how they compare to the QHGs and whether some adjustment of the selected likelihood-dose limits is warranted. Table 3-6 shows the conversion of each surrogate radiological dose-likelihood limit to health effect using the conversion factors cited above from the DOE memorandum (Lawrence 2002). As discussed above, two sets of conversion factors are provided in the memorandum. One set is for radiological dose to the public and consists of a conversion factor for mortality (i.e., 6E-04 fatalities per rem) and another conversion factor for morbidity (i.e., 8E-04 injuries per rem). The other set is for radiological dose to the worker, and also consists of a conversion factor for mortality (i.e., 5E-04 fatalities per rem) and another conversion factor for morbidity (i.e., 7E-04 injuries per rem). The conversion factors are used to convert the selected likelihood-dose limits for the public and worker to expected fatalities and injuries. Then the calculated expected fatalities and injuries are compared to the applicable QHGs. Specifically, they are compared to QHG-1 for acute fatality to a member of the public (QHG-2 for LCF in the public might also be applicable but QHG-1 is used because it is a lower threshold, and therefore, more conservative); QHG-3 on serious injury (cancer illness) to a member of the public; QHG-4 on acute fatality to workers (QHG-5 for LCF in workers might also be applicable but QHG-4 is used because of its lower threshold, and therefore, more conservative); and QHG-6 on serious injury (cancer illness) to workers.

The selected likelihood-dose limits for the public and workers are shown in the first two columns of Table 3-6. The dose-to-health effect conversion factors and resulting expected fatalities and injuries are shown in the third and fifth columns. The fourth and sixth column indicate whether the selected likelihood-dose limits are bounded by the QHGs.

Table 3-6. Comparison of Selected Dose-Consequence Limit Surrogates to the Limiting QHGs

Dose Limit (rem)	**Frequency (per year)**	**Risk of Fatality**	**QHG for Acute Fatality**	**Risk of Injury**	**QHG for Serious Injury**
Offsite Public		Conversion 6×10^{-4} fatality/rem	QHG-1 5×10^{-7} fatality/rem	Conversion 8×10^{-4} injury/rem	QHG-3 1×10^{-6} injury/rem
750	5E-07	2.3E-07	OK	3.0E-07	OK
25	1E-06	1.5E-08	OK	2.0E-08	OK
5	1E-04	3.0E-07	OK	4.0E-07	OK
1	1E-03	6.0E-07	OK(1)	8.0E-07	OK
100 mrem	1E-02	6.0E-07	Not bounded	8.0E-07	OK(2)
Worker		Conversion 5E-04 fatality/rem	QHG-4 1E-06 fatality/rem	Conversion 7E-04 injury/rem	QHG-6 5E-06 injury/rem
750	5E-07	2.3E-07	OK	3.0E-07	OK
100	1E-06	5.0E-08	OK	7.0E-08	OK
25	1E-04	1.3E-06	OK(1)	1.8E-08	OK
5	1E-03	2.5E-06	Not bounded	3.5E-06	OK
500 mrem	1E-02	5.0E-06	Not bounded	6.0E-06	Not bounded

(1) Within the margin of error.
(2) However, is not bounded for accident frequencies that are somewhat greater than 1E-02 per year (i.e., greater than 1.4E-02).

Based on this evaluation, some of the lower consequence but higher likelihood limits are not bounded by QHGs. These are indicated in red font. It is worth noting, that a TNPP package will be designed to remain intact for most hazards and initiating events that can cause accidents particularly if the event is not highly unlikely, such as anticipated events (e.g., dropping the package while loading it onto the truck, failure of a seal caused by vibration during transport, loss of containment due to the environmental factors such as cold weather). Therefore, the high-consequence low-likelihood part of this risk evaluation scheme is likely to be more important than the low-consequence higher likelihood part of this scheme. Furthermore, the methodology proposed in this report includes this broader integrated approach by considering safety margins, defense-in-depth, and the results of sensitivity studies.

3.3 Proposed Surrogate Risk Evaluation Guidelines Based on the Safety Goal QHOs

This section proposes risk evaluation guidelines for evaluating the risk associated with the transportation of a demonstration-phase TNPP transportation package based on concepts previously discussed. Section 3.1 discusses establishing the risk evaluation guidance on QHGs as proposed by NRC in a report on risk-informed decisionmaking for activities that include transportation of nuclear material (i.e., the RIDM report [NRC 2008]). Section 3.2 describes the concept of using surrogate risk measures equivalent to or bounded by the QHGs that are more practical to use and allow helpful comparisons to other radiological risk guidance. Section 3.2.1 through Section 3.2.4 discuss risk evaluation guidance concepts that have been established for other nuclear applications using radiological dose and likelihood limits. Section 3.2.5 assesses how those radiological dose and likelihood limits and their ties related to federal and international guidance could be combined to establish risk evaluation guidelines that are consistent with or bounded by the QHGs.

Table 3-7 presents the proposed risk evaluation guidelines in term of likelihood and radiological dose consequences, which are reflective of the development results described in Section 3.2.5 and which are compared against the applicable proposed QHGs in Table 3-6. The regions of acceptable and unacceptable risk from Table 3-7 are shown graphically in Figure 3-7 and Figure 3-8 for the offsite public and worker, respectively.

Based on the challenges (e.g., lack of applicable data) and uncertainties (e.g., evolving design) associated with producing a detailed event-based PRA for transportation of an TNPP, it is envisioned that the results of a PRA will be representative bounding accidents (as is the case in this report). Therefore, justification can potentially be made that the risk results are sufficiently conservative that the results for each bounding representative accident could be compared separately to the risk evaluation guidelines (e.g., summation of the risk results from multiple bounding accidents produces grossly conservative and unrealistic results). However, to produce more confidence that the risk acceptance guidelines are met or if the PRA results are more granular, the dose consequences for each bounding accident or accident sequence within each frequency interval should be added together. The total dose for each frequency interval would then be compared against the risk evaluation guidelines. As was shown in Table 3-7, meeting the risk evaluation guidelines generally provides confirmation that the QHGs are met. The potential exception is for lower consequence but higher likelihood accidents that are not bounded by QHGs. However, as discussed earlier, a TNPP package will be designed to remain intact for most hazards and initiating events that can cause accidents particularly if the event is not highly unlikely, such as anticipated events (e.g., dropping the package while loading it onto the truck, failure of a seal caused by vibration during transport, loss of containment due to the environmental factors such as cold weather).

Table 3-7. Proposed Radiological Risk Evaluation Guidelines

Annual Accident Frequency (per year)[(1)]	Radiological Dose Consequence to the Offsite Public[(2)]	Radiological Dose Consequence to the Worker[(2)]	Risk Acceptability
≤5E-07[(3)]	≥750 rem TEDE[(3)]	≥750 rem TEDE[(3)]	Acceptable
>5E-07	>750 rem TEDE	>750 and TEDE	Unacceptable
≤1E-06 and >5E-07	≥25 and <750 rem TEDE	≥100 and <750 rem TEDE	Acceptable
>1E-06	>25 rem TEDE	>100 rem TEDE	Unacceptable
≤1E-04 and >1E-06	≥5 and <25 rem TEDE	≥25 and <100 rem TEDE	Acceptable
>1E-04	>5 rem TEDE	≥25 rem TEDE	Unacceptable
≤1E-03 and >1E-04	≥1 and <5 rem TEDE	≥5 and < 25 rem TEDE	Acceptable
>1E-03	>1 rem TEDE	>5 rem TEDE	Unacceptable
≤1E-02 and >1E-03	≥100 mrem and <1 rem TEDE	≥500 mrem <5 rem TEDE	Acceptable
>1E-02	≥100 mrem TEDE	≥500 mrem TEDE	Unacceptable
>1E-02	<100 mrem TEDE	<500 mrem TEDE	Acceptable

(1) Determination of the accident frequency should account for multiple shipments per year, if applicable.
(2) The radiological dose consequences are presented as a TEDE, which is based on integrated committed dose to all organs thereby accounting for direct exposure as well the 50-Year committed effective dose equivalent.
(3) If the accident frequency is <5E-07 per year, then the risk of the accident scenario is generally acceptable regardless of its radiological dose consequence. Bounding accidents with frequencies less than 5E-07 per year are evaluated (e.g., using sensitivity studies), to confirm there are no cliff edge effects. They may also be taken into account in the risk-informed, performance-based evaluation of defense-in-depth.

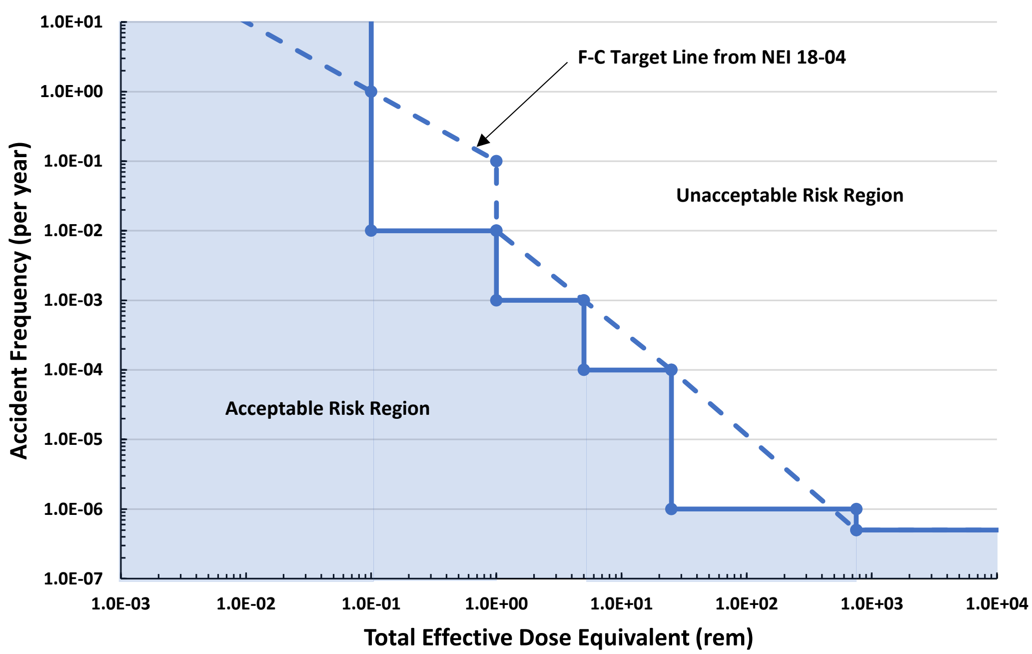

Figure 3-7. Proposed Offsite Public Risk Evaluation Guidelines Chart for Transport of a TNPP Package

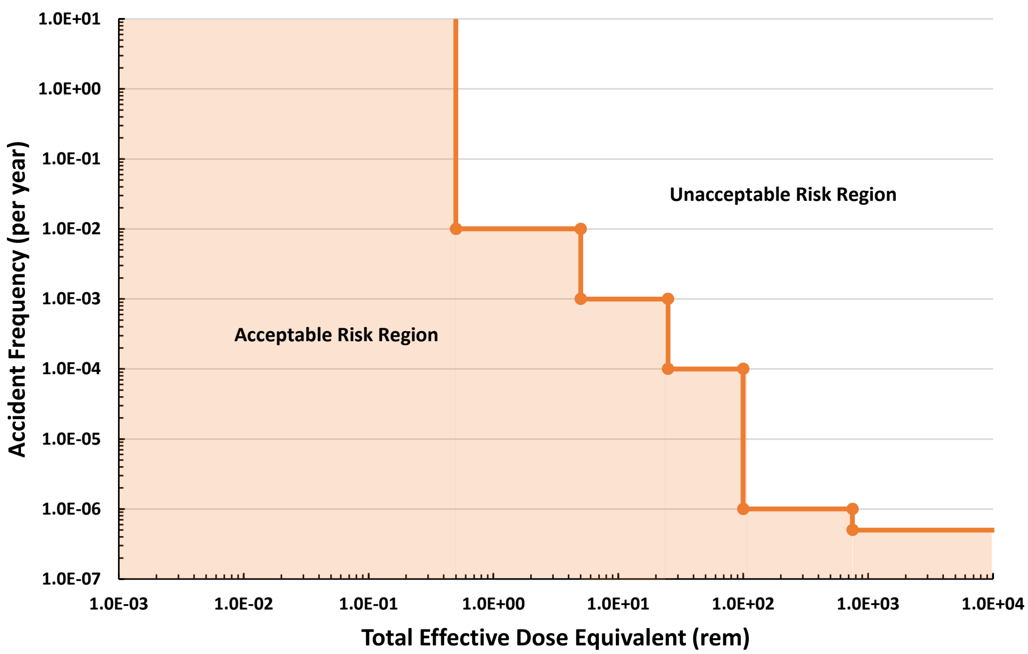

Figure 3-8. Proposed Worker Risk Evaluation Guidelines
Chart for Transport of a TNPP Package

Therefore, the high-consequence low-likelihood part of this risk evaluation scheme is likely to be more important than the low-consequence higher likelihood part of this scheme. Furthermore, the methodology proposed in this report includes a broader integrated approach by considering safety margins, defense-in-depth, and the results of sensitivity studies as discussed in Sections 4.8 and Section 5. These are discussed later in this report.

Transportation of a demonstration-phase TNPP package is envisioned as a single shipment. Therefore, application of the proposed risk evaluation guidelines is straightforward. If there are multiple shipments of the same TNPP in the same year, then application of the guidelines is still applicable because the risk criteria is provided on a per-year basis. However, the aggregate risk of transport over the year would need to be compared to the risk evaluation guidelines.

Future challenges may need to be addressed during the production phase of TNPP deployment. For example, if the TNPP is transported at different times by different entities, then a mechanism may need to be developed to track the accumulation of the associated risk over a year and share it among the entities responsible for the transports. A more difficult challenge arises if multiple transports of different TNPPs over the same or overlapping routes can affect the same or overlapping members of the population. This is a different situation than a member of the public that lives near a nuclear power plant that could potentially be exposed to the release of radiological material from an accident that occurs at the plant. The population at risk is different for each nuclear plant site because of the considerable differences between plants. The concept of not unduly adding to the general background risk in the United States is addressed using the proposed risk evaluation guidelines but it does not consider the cumulative impact to the same population from the transport of different TNPPs. Additional guidelines may be needed to track the aggregate risk to specific populations. The application of the proposed risk evaluation guidelines should be readdressed, and the challenges associated with shipment of multiple TNPPs considered when that situation becomes realistic. However, for transportation of a demonstration-phase TNPP package under the 10 CFR Part 71 exemption process, application of the proposed risk evaluation guidelines is straightforward.

The proposed risk evaluation guidelines presented in the section in terms of likelihood and radiological dose consequences achieves three objectives:

1. It provides practical quantitative guidance to use in evaluating the risk acceptance of transport of the demonstration-phase package under the 10 CFR Part 71 exemption process supported by quantitative risk evaluation.

2. It provides the foundation for a risk-informed methodology that could be applied to the production phase TNPP packages under the 10 CFR Part 71 exemption process or other licensing options.

3. It provides the foundation for a risk-informed methodology that could potentially be used to inform NRC decisionmaking or NRC guidance on risk-informing the licensing of TNPP transport.

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4. TNPP TRANSPORTATION PRA METHODOLOGY, DATA, AND RESULTS

This section presents a Transportable Nuclear Power Plant (TNPP) transportation Probabilistic Risk Assessment (PRA) study based on information gleaned from vendor design material generated for Phase I of the Project Pele. The approach, data, and information presented in this section provides useable information and serves to illustrate how a TNPP transportation PRA study could be performed consistent with the safety goals and risk evaluation guidelines discussed and presented in Section 3.0 of this report. Further detailed design and safety analysis information that will be forthcoming in Phase II of the Project Pele is needed to better inform the TNPP transportation PRA. Key information includes plans and design information associated with the dismantlement of the TNPP, packaging and loading of the various TNPP modules for transport, and transport of the TNPP packages. However, to the extent that the information was available in Phase I, it is reflected in the TNPP transportation PRA study presented in this report.

This section includes an overview of the risk assessment approach, characterization of the TNPP package radiological material inventory, identification of the TNPP package safety functions, development of TNPP transportation accident scenarios, development of the likelihoods of TNPP transportation accidents, development of the consequences for TNPP transportation accidents, and risk summary and comparison to the risk evaluation guidelines proposed in Section 3.0 of this report. The information includes data that was compiled and analyzed to support the risk determination.

4.1 Overview of Risk Assessment Approach

This section provides an overview of the risk assessment approach used to determine the level of risk associated with transportation of the prototype TNPP package (i.e., Reactor Module for this report) and whether that risk is acceptable for licensing. The development of a TNPP transportation risk assessment is a novel endeavor, because until now there has not been a need to develop a risk-informed licensing bases for a transportable microreactor, and therefore, a technical basis has not been thoroughly investigated. The licensing of transportation of a TNPP package does not fit cleanly into the existing licensing categories for transportation of nuclear material in approved containers, casks, or packages or for operation of a stationary nuclear power plant. However, the risk assessments performed for the transportation of radiological material in approved containers is commonly performed and provides some insight. Likewise, the use of PRA for risk-informed applications associated with amending the operating license of light water reactors (LWRs) in the United States has become common and the development and review of the PRA models that support such applications is now very mature. However, it is unlikely that the TNPP package will be able to meet the requirements for a Type B package for transporting material with high levels of radioactivity, and hazards exist associated with TNPP transport that are not encompassed by the typical safety basis or the PRA of a stationary nuclear power plant.

Accordingly, the reasons for performing a TNPP package risk assessment are significant and include: (1) demonstration that the risk associated with transportation activity is acceptably low and can be used to support a 10 CFR Part 71 (“Packaging and Transportation of Radioactive Material”) exemption for this first-of-a-kind undertaking, (2) identification of design features and administrative controls that must be protected to ensure an acceptable level of risk and identification of compensatory measures that should be performed during the transport, and (3) identification of the possible trade-off between the design and risk (e.g., the trade-off between reducing the weight and size associated with containment features and the risk associated with breaching the containment in a vehicle accident).

The U.S. Nuclear Regulatory Commission (NRC) paper SECY-99-100, *Framework for Risk-Informed Regulation in the Office of Nuclear Material and Safety and Safeguards* (NRC 1999), describes the results of an effort to scope the development of a framework for applying risk assessment methods to the regulation of nuclear material uses and waste disposal and makes recommendations to the NRC Commission for how to proceed. This paper and the proposed guidance in the RIDM report (NRC 2008) indicates that for transportation of nuclear material the most appropriate risk assessment method is either a PRA or an Integrated Safety Analysis (ISA). As the RIDM report explains, ISAs are normally qualitative or semi-quantitative assessments, and therefore, are not as effective in producing the quantitatively derived benefits discussed above such as demonstrating that the risk of transport meets accepted safety goals and quantitative risk evaluation guidelines. PNNL-31867 (*Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages* [Coles et al. 2021]) describes historical examples of using risk information and insights to develop the technical basis for regulatory approval of transportation packages. However, none of these past cases were as technically challenging as transport of a TNPP. In fact, past risk informed approvals primarily consisted of showing that transportation accidents leading to radiological consequence of any significance were incredible, especially in consideration of stipulated compensatory measures. Accordingly, the quantitative risk assessment approach presented in this report is a PRA performed to be consistent with the units of measure used in risk evaluation guidelines presented in Section 3.0.

The term “risk” is defined by Kaplan and Garrick (1981), who are two pioneers in PRA especially as it pertains to high-risk highly engineered systems such as nuclear power plants, as a risk triplet that defines the set, $<s_i, f_i, x_i>$, in which S_i, represents the i^{th} scenario (sequence or progression), f_i is the associated frequency, and x_i is the resulting consequence. In simple language, risk is the determination of: (1) What can go wrong?, (2) How likely is it?, and (3) What are the consequences? PRA modeling of accident scenarios typically involves two types of logic analyses: fault and event tree analyses. Fault-tree analysis is a deductive process used for determining combinations of system failures and human errors that could result in the occurrence of defined undesired events. Event-tree analysis, by comparison, uses inductive logic to define possible accident sequences starting with specific initiating events and then mapping possible subsequent events that lead to different outcomes. For the most part, complex system analysis (e.g., failure of the control rods to SCRAM or Emergency Diesel Generators to start) using fault trees is not required or beneficial in TNPP transportation PRA. The failures that are considered in the TNPP transportation accident scenarios are primarily the result of the initiating event itself as opposed to subsequent random failures. Also, even though event tree models have been used in past transportation risk assessments, the benefit of their use for this application and phase of project development is seen as limited. Accordingly, the focus of this PRA is on identification of accident scenarios, development of the likelihood of those scenarios, and development of the consequences of those scenarios. The non-use of fault trees and events is discussed later in more detail in Section 4.4.1.

That said, preparatory steps are needed to support the process of accident scenario, likelihood, and consequence development. The primary hazard of concern in all TNPP transportation accidents is the TNPP package radiological material inventory which needs to be characterized in detail to support accident scenario development and even more importantly to perform the consequence analysis.

Additionally, another key preparatory step is identification of the safety functions that must be performed during transport by the TNPP and its package. This information is key to supporting the postulation of undesired events and outcomes should events occur that defeat or degrade one of these safety functions.

Concerning modeling assumptions in general, the TNPP transportation PRA presented in this report is based on information available from the reactor vendor at the end of Phase I. To develop the TNPP transportation PRA, several assumptions had to be made which are identified in applicable discussions in Section 4.0. For example, selection of the Idaho National Laboratory (INL) site as the origination site and White Sands Missile Range (WSMR) as the destination of the TNPP transport is an assumption that is identified in Section 4.5. Though assumptions may be used at this phase of the project, the PRA presented in this report is meant to be updated and revised based on Phase II Project Pele prototype TNPP design information and refinement. Moreover, the results of sensitivity studies will be presented to show the impact of important assumptions on the risk estimates which still may be needed after Phase II.

The following primary elements of the TNPP transportation PRA development are discussed in this section:

- Characterization of the TNPP package radiological material inventory (Section 4.2).
- Identification of the TNPP package safety functions (Section 4.3).
- Identification and development of the TNPP transportation accident scenarios (Section 4.4).
- Development of the TNPP transportation accident scenario likelihoods (Section 4.5).
- Development of the TNPP transportation accident scenario consequences (Section 4.6).

The outcome of a PRA is a list of undesired event accident sequences that reflect the system's response to the range of initiating events that can be expected. Because the likelihood and consequence of each accident sequence is estimated, a measure of the overall risk from the activity can be determined and compared to the risk acceptance guidelines, such as those proposed in Section 3.0 of this report. The PRA model can also be used to perform sensitivity studies of the impact of key sources of modeling uncertainty on the calculated risk. Accordingly, the following primarily results of the TNPP transportation PRA are discussed in this section:

- Presentation of the TNPP transportation baseline PRA results and comparison to risk evaluation guidelines (Section 4.7).

- Definition of TNPP transportation PRA sensitivity studies and presentation of results (Section 4.8).

- Presentation of risk insights for baseline and sensitivity studies (Section 4.9).

4.2 Characterization of TNPP Package Radiological Material Inventory

This section presents discussion of the Project Pele prototype TNPP radionuclide inventory during transportation. This radionuclide inventory is then used in the TNPP transportation PRA to: (1) define the material at risk in a transportation event that could become the source term in an accident leading to release of radiological material, and (2) define the potential sources of direct radiation exposure in a transportation accident.

Section 4.2.1 discusses the basis for the estimated radiological inventory possible during transport. This includes discussion of the results of efforts to estimate diffusion from tri-structural isotropic (TRISO) fuel based on published reports and the impact of manufacturing specifications on these estimates. Section 4.2.2 provides the specific estimated radiological inventory that is used in the TNPP transportation PRA.

4.2.1 Bases for Estimated Radiological Inventory

The radiological inventory present during the transportation of a previously operated TNPP is a function of the reactor design (e.g., power level, core configuration, materials of construction, coolant), its operation (e.g., equivalent full-power days), time period between reactor shutdown and its transport, and the configuration of the TNPP during transportation (e.g., one or multiple modules or transportation packages). For the purposes of this report, the estimated radiological inventory possible during transport is that developed for the Project Pele prototype TNPP. The inventory was developed using the ORIGEN2 computer code system and assumed a reactor operating time period of 3 years, an initial core loading of 0.18 metric tons of uranium (MTU), and a uranium enrichment of 19.5 wt% (BWXT 2022[10]). This inventory is only that associated with burnup of the fuel and does not include the inventory due to the activation of the reactor materials of construction or the coolant.

4.2.2 Estimated Radiological Inventory

The estimated radiological inventory (BWXT 2022[11]) was developed for multiple cooling time periods ranging between time zero (at reactor shutdown) and two years after reactor shutdown. Included in the inventory was a 90 day cooling case that is assumed to be the start of transportation in the analyses performed in this report. The inventory provided contained over 1000 individual radionuclides. For the purposes of this report, these radionuclides were screened to identify those that were judged to be potentially significant to risks associated with transportation accidents involving a potential release of radioactive material from the Reactor Module.

The screening of radionuclides was conducted in two phases, the first on total curies (radionuclides with greater than 10 microcuries) and the second based on the A_2 values provided in Table A-1 of 10 CFR Part 71 (radionuclides present in greater than 0.1% of their A_2 value). The A_2 value, as defined in 10 CFR 71.4 (“Definitions”) is the maximum activity of radioactive material (with some exceptions) permitted in a Type A package. The A_2 values are radionuclide specific and are normalized based on radiological hazard during transport. The A_2 values take into account how the human body absorbs each radionuclide. A_2 values are derived so there is reasonable assurance that a person exposed within the vicinity of a transportation accident will not exceed the annual dose limit for radiation workers (Regulatory Impact Summary [RIS] 2013-04, “Content Specification and Shielding Evaluations for Type B Transportation Packages” [NRC 2013a]). The derivation of the A_2 values is based on the Q system approach, which is described in Appendix I to the International Atomic Energy Agency (IAEA) Specific

[10] BWXT Final Design Report, Table 2.3.1.1.3-1.
[11] BWXT spreadsheet “B1.34-NuclideConcentrations(Ci)-Fuel.xlsx” provided on August 11, 2022.

Safety Guide No. SSG-26 (*Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2012 Edition)* [IAEA 2014]). The A_2 values represent quantities of radionuclides that are tied to specific dosimetric consequences:

> The dosimetric basis of the A_2 system relied upon a number of somewhat pragmatic assumptions. An intake of 10^{-6} A_2, leading to half the annual limit on intake for a radiation worker, was assumed in the derivation of A_2 as a result of a 'median' accident. The median accident was defined arbitrarily as one which leads to complete loss of shielding and to a release of 10^{-3} of the package contents in such a manner that 10^{-3} of this released material was subsequently taken in by a bystander. [See Appendix I, page 272, IAEA 2014 for more details.]

A_2 values were used as criteria for screening radionuclides for inclusion in the consequence assessment because they weight the consequence of the radionuclides by the potential dose consequence. This screening methodology allows for a targeted dosimetry approach weighted by consequence. In the first screening phase, radionuclides for which there was not an A_2 value were included in the screened-in radionuclide list if its activity was greater than or equal to 2.4E-03 times 0.001 or 2.4E-06 Ci. This approach is consistent with 10 CFR 50.71 ("Maintenance of records, making of reports") which specifies that an A_2 value from Table A-3 of this regulation may be used if an A_2 value for the radionuclide is not provided in Table A-1 of this regulation. The value of 2.4E-03 is the smallest value from Table A-3. Furthermore, for a radioactive material composed of a mixture of radionuclides, per 10 CFR Part 71, the A_2 value for the material is determined as the sum-of-the-fractions of the radionuclide-specific A_2 values. A factor of 0.001 was used to account for the TNPP core inventory being a mixture of over 1000 radionuclides. This is conservative because the remaining number of radionuclides present in the mixture after 90 days of cooling is less than 400. For the second screening, radionuclides for which there was an A_2 value were included in the screened-in radionuclide list if its activity was greater than or equal to its A_2 value times 0.001.

The screening analysis resulted in the identification of 112 nuclides for a 90-day cooling period that are included in the consequence analysis. Appendix 8.1 provides the radionuclides and quantities. The consequence analysis only includes the screened-in radionuclides that could be released in an accident scenario because including all the radionuclides in the analysis is a calculational burden and, furthermore, will have negligible impact on the dose results because over 99.99999% of the radionuclide inventory at 90-days is included in the consequence analysis.

4.2.3 Release Mechanisms from Uranium Oxycarbide TRISO Fuel and Fuel Compacts

The Project Pele reactor core is contained within the Reactor Module and housed within the pressure boundary created by the reactor vessel and Primary Cooling system. The core is comprised of fuel assemblies in coolant channels, moderator blocks, and control rods. The fuel assemblies consist of cylindrical TRISO fuel compacts stacked inside graphite fuel sleeves having graphite end plugs. The fuel sleeves have spacer nubs that center the fuel assembly within a coolant channel. The fuel compact is composed of uranium oxycarbide (UCO)[12] TRISO particles contained within a graphite matrix. The TRISO fuel is designed to ensure that its characteristics are bounded by the key parameters of advanced gas-cooled reactor (AGR) testing of TRISO fuel. The overall design of the core ensures that the operating

[12] The Project Pele microreactor UCO is a blend of HALEU uranium dioxide (UO_2) and uranium dicarbide (UC_2).

and passive cooling scenarios do not allow the fuel to exceed its expected maximum allowable temperatures, which ensures the TRISO fuel will maintain its structural integrity necessary for retention of fission products (BWXT 2022[13]). However, the current BWX Technologies, Inc. (BWXT) fuel compact design and fabrication is sufficiently different from the AGR fuel compact design that all its characteristics may not be bound by the AGR testing. For instance, the larger size of the BWXT compact may impact heating times and final heat treatments needed to assure the proper thermal conductivity of the compact matrix. In addition, higher force/pressure than that used in the AGR program may be required to get acceptable matrix density. One significant concern is that compacting at higher pressure could damage more particles. Sensitivity studies on the impact of damaged TRISO particles and compact matrix role in radionuclide release and retention will be performed to appropriately bound consequence results. Post-irradiation examination (PIE) of the TNPP fuel compacts is planned for the Project Pele demonstration unit to confirm the applicability of the AGR fuel compact qualification to the TNPP fuel compact design. A cross-sectional view of a TRISO fuel particle is shown in Figure 4-1. An example of AGR fuel compacts, which have a smaller diameter than the Project Pele TNPP fuel compacts, is shown in Figure 4-2.

The high-assay low-enriched uranium (HALEU) oxycarbide TRISO fuel particles and fuel compacts release fractions of certain fission products[14] during normal operations and anticipated operational occurrences (AOO) of TRISO fueled reactors. Furthermore, additional fractions of fission products may be released from the TRISO fuel particles and fuel compacts due to loads and/or conditions experienced as a result of a transportation accident during shipment of the TNPP. This section provides an overview of the available information on these potential release mechanisms.

4.2.3.1 Release Mechanisms during Normal Operations and AOOs

The most prominent releases are isotopes of silver and a few other metal species that are well known to diffuse through and escape TRISO particles within the temperature range that the fuel compacts experience in normal operation, which is up to 1400 °C for the TNPP (BWXT 2022[15]). This involves the diffusion of fission product species through the specific pyrolytic carbon (PyC) coating layers (porous PyC buffer, high density inner and outer PyC layers) and the silicon carbide (SiC) layer that is sandwiched between the two high density PyC layers within each TRISO fuel particle. In addition to these diffusing metal isotope species, noble gas isotopes and certain more volatile fission product species also diffuse through these layers at largely varying, but much lower rates.

Various physical mechanisms for the diffusion are involved and are strongly dependent on the specific microstructure and defects of the UCO fuel kernel and of the four coating layers. These complex diffusion mechanisms are most often modeled as simple Fickian diffusion with strongly temperature dependent effective diffusion coefficients and the assumption of negligible counter diffusion effects (which is a reasonable assumption at low release rates into a medium that is far from saturation by competing species). The most common model for the coefficients varies exponentially with temperature. As a result, different fission product species diffuse at widely different rates, and most do not readily diffuse within the normal operations and AOO envelopes by design.

[13] BWXT Final Design Report, Section 2.3.1.1.3.
[14] The term "fission product" is used broadly to include isotopes that are produced as a result of fission processes (direct fission products and isotopes that result from the radioactive decay of direct fission products) and isotopes resulting from neutron activation of fission products.
[15] BWXT Final Design Report, Table 4.5.1.2-3.

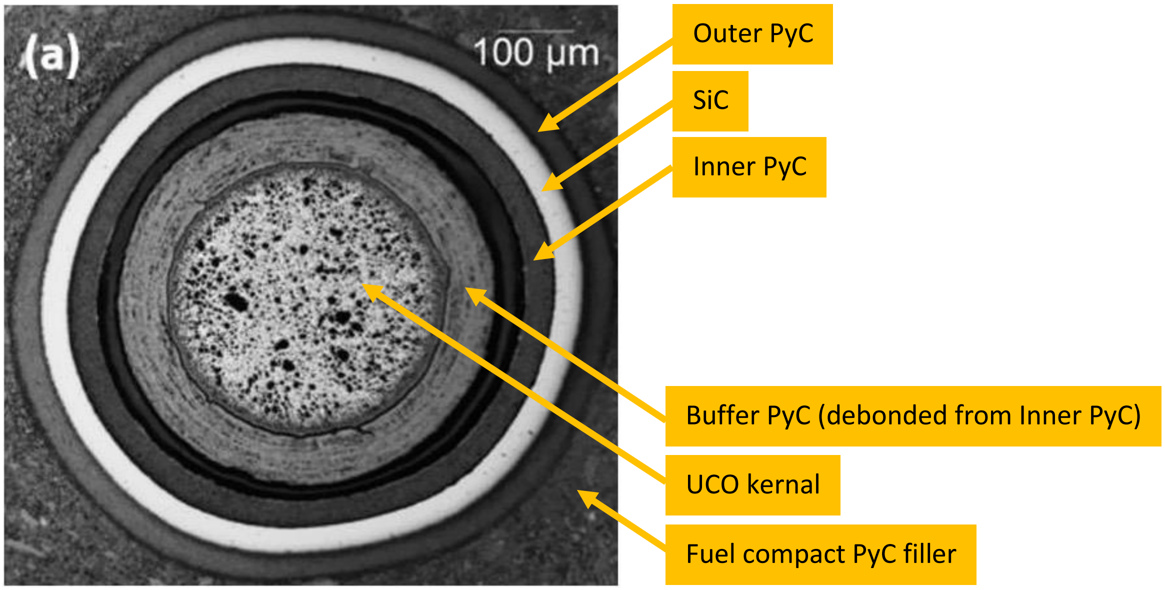


Figure 4-1. Cross Section of Irradiated UCO TRISO Fuel Particle (EPRI 2020)

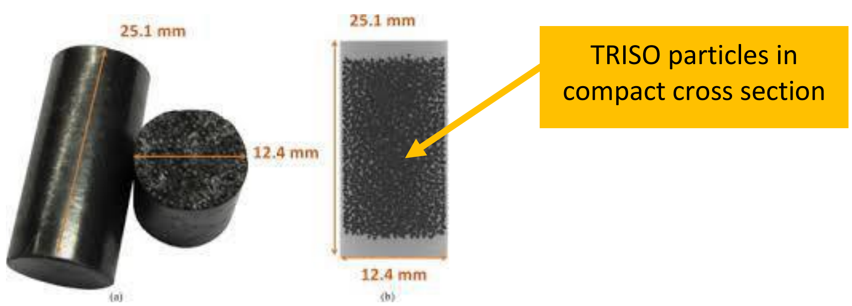


Figure 4-2. AGR TRISO Fuel Compacts (Harp et al. 2014)

In design basis events (DBE) and beyond design basis events (BDBE), significant heat soak circumstances may occur where fuel compact temperatures are expected to rise from roughly 1200 °C up to roughly 1400 °C to 1600 °C. At these elevated temperatures, fission product releases increase since diffusion rates increase. However, transportation of an TNPP that has experienced a DBE or BDBE is beyond the scope of this assessment.

Releases from the TRISO fuel particles also occurs because of certain observable manufacturing defects (e.g., a SiC layer is incomplete or missing) and microscopic defects that result in a few certain well known microscopic failure mechanisms (e.g., palladium corrosion attack of the SiC layer, incomplete debonding of a certain PyC layer at higher burnup that causes subsequent failure of the SiC layer). These microscopic mechanisms occur stochastically in a small fraction of particles at high temperature and higher burnups – this fraction of failures can only be determined through PIE of individual irradiated TRISO fuel particles as part of a fuel qualification program.

The observable manufacturing defects and to some degree the microscopic defects can be significantly reduced by proper materials selection and manufacturing techniques pursuant to meeting a required fuel specification under a TRISO fuel manufacturing quality assurance (QA) program.[16] As a result, releases from the TRISO fuel particles during reactor operation are largely restricted by these defect mechanisms. Reactors and fuel elements are typically designed to minimize these gradual releases and to capture the fugitive fission products within the reactor core graphitic materials and surfaces within the primary cooling circuit boundary to a large extent. UCO TRISO fuel is the most accident tolerant fuel that has been developed to date that also has high technology readiness level (TRL) and manufacturing readiness level (MRL). Topical Report EPRI-AR-1(NP)-A (*Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance* [EPRI 2020]) provides the results of substantive research efforts that support this conclusion and demonstrates the performance of UCO TRISO fuel particles over a range of normal reactor operating and off-normal accident conditions. Furthermore, the NRC final safety evaluation of the EPRI report concludes that the data in the report can be used to support safety analyses referencing the unique design features of the TRISO fuel particle, subject to the performance thresholds of the AGR tests discussed in the report and the specified Limitations and Conditions provided in Section 4.0 of the safety evaluation report (NRC 2021).

As discussed later in this report, the research results presented in the EPRI report bound the fission product release fractions and fuel failure fractions used in the risk analysis to estimate releases during reactor operations that could be retained within the TNPP transportation package during transport and therefore available for release in the event of a severe transportation accident. However, as previously discussed, the current Project Pele TNPP fuel compact design is sufficiently different from the AGR fuel compact design that its characteristics may not be bounded by the AGR testing. PIE of the TNPP fuel compacts is planned to confirm the applicability of the AGR fuel compact qualification to the TNPP fuel compact.

4.2.3.2 Release Mechanisms as a Result of Transportation Accidents

In typical fixed site nuclear power plant deployments in the future, a TRISO fueled nuclear power plant will meet design requirements for external hazards comparable to other fixed site nuclear power plants. However, the TNPP concept introduces a broader range of external hazards to the reactor because of transportation which: (1) could expose the reactor to different or more severe hazards than seen at the operating location; (2) introduces tradeoffs in reactor design that are required to meet transportability considerations (e.g., maximum weight of a shippable unit), and (3) the requirement to be able to assemble and disassemble the TNPP to deploy and transport the unit could introduce human and mechanical faults not normally present in a fixed facility. Hazards associated with disassembly and re-assembly of the TNPP are important activities to evaluate for site-based licensing but are not included within the scope of licensing for transportation. In this report, the focus is on normal (non-emergency) transportation of the Reactor Module within the conterminous United States, with NRC approval of the TNPP transportation package and U.S. Department of Transportation (DOT) regulation of its shipment during transit.

A principal difference posed by the user requirement of transportability are hazards to the reactor, its irradiated fuel, and contaminated systems and components (e.g., primary cooling system) resulting from accidents during transport of the TNPP. Transportation accidents could involve kinetic hazards (such as impacts with vehicles or other objects), fire hazards, random failures, or human errors (e.g., human

[16] See IAEA-TECDOC-CD-1674, page 129 (IAEA 2012); Topical Report EPRI-AR-1(NP)-A, page 5-7 (EPRI 2020).

error in preparing the TNPP package for transport), different exposure scenarios to natural hazards (such as earthquakes and tornados while in transport), and potential submersion in water.

There is currently limited information available on the performance of the TNPP transportation package or its contents during its transport. Preliminary analysis of the TNPP transportation package has been completed for selected normal conditions of transport (NCT) and hypothetical accident conditions (HAC) based on the requirements of 10 CFR Part 71. Specifically, shock, vibration, free drop, and penetration assessments have been performed using finite element analysis (BWXT 2022[17]). However, these preliminary assessments did not assess the potential for damage to the irradiated TRISO fuel particles/compacts.

With respect to the TRISO fuel and compacts, plus other closely associated core components, the transfer of impact loads and vibration spectra to those structures is important. Impact load is critical to understanding the possible reconfiguration of fuel and near fuel materials in addition to potential failures of fission product barriers and production of fine particulates that represent a potential source of released material. Because no transportation assessments have yet been performed on the performance of the TRISO fuel and compacts, engineering judgements were made in this report regarding the consequences of dynamic loads and vibration on these components during anticipated transportation and bounding accident conditions.

Reconfiguration of core components from kinetic accidents and/or immersion in water may raise questions about maintenance of subcriticality of the fuel. The current prototype TNPP design does not provide assurance of subcriticality in flooded conditions. Further design development and testing is planned to be performed to provide assurance of subcriticality in flooded conditions for production units of the TNPP (BWXT 2022[18]). Though a brief criticality pulse may not mechanically destroy the TRISO fuel barriers if the temperature of the UCO fuel kernel remains below its melting point (IAEA-TECDOC-CD-1674 [*Advances in High Temperature Gas Cooled Reactor Fuel Technology*], page 464 [IAEA 2012]), sustained high temperature is a mechanism for enhanced fission product diffusion and potential release of material. A bounding assumption of any criticality accident would be to assume broader SiC barrier failures have occurred from thermal/mechanical shock and potential kernel melt until demonstrated otherwise. A broad failure of SiC barriers could result in a significant increase in release of fission products and gases for an inadvertent criticality accident. However, diffusion of this released material through other near fuel graphitic structures could limit release depending on time at temperature and potential mechanical damage to fuel and near fuel structures.

At higher burnup there is a weakening of fuel material strength and considerable fission gas pressure behind the SiC barrier. The SiC layer acts as a pressure vessel for gaseous and volatile fission products in a TRISO particle. This is a complex fuel performance problem since layer interactions typically initially unload the SiC layer as burnup occurs. At some point in a mid-burnup range, that turns the other way and the SiC barrier begins to load until eventually a barrier failure occurs at very high burnup. The SiC barrier is essentially a pressure vessel that can fail given sufficient fission gas pressure combined with reductions in material strength. The AGR qualification studies demonstrate burnup as high as 19% fissions per initial metal atom (FIMA) with no obvious broad failures of SiC layers. At the expected burnups in the current TNPP demonstration design, which will be 10% FIMA or below, gas pressure

[17] BWXT Final Design Report, Appendix III.44, “BWXT Reactor Design Preliminary Transportation and Severe Accident Analyses,” Executive Summary.
[18] BWXT Final Design Report, Section 7.5.1.

should not fail the SiC layers. If insulted by a sudden and potent thermal or mechanical shock, when in a condition of high burnup, some SiC barriers might fail indicating some enhancement of release of gaseous radioactive material. More likely of concern is vibration spectra and consequence that is critical to understand both fatigue tolerance of the fuel compacts and near fuel materials and tribology that could generate fine particulates that represent a source of released radioactive material.

Similarly, with fire hazards, the irradiated thermal material properties are important to understand maximum temperatures that may occur in the fuel and core materials that hold up diffusing fission product species. Time at high temperature is one of the common sources of release of radionuclides from TRISO fuels. In this case, the combination of fire applying heat to the exterior of the reactor and decay heat from the core would determine the outcome. Since the TRISO fuel typically performs very well up to 1400 °C, remaining decay heat is a significant factor in a consequence analysis. The codified regulatory pool fire test is set at 800 °C – conservative for a liquid fossil fuel pool fire. In the absence of sufficient decay heat, the fire hazard would seem to be related to potential enhancement of barrier failures other than TRISO fuel and near fuel core component barriers. For example, failures of seals on the reactor containment boundary could release plated out radioactive material in the system that was released from the TRISO fuel during reactor operations.

Because the temperature of the graphite needed to produce a self-sustaining graphite fire is not expected to be able to be reached during plausible transportation accidents involving limited air ingress though a failed seal (considering the contribution from both decay heat and an 800 °C fire for 30 minutes), a so-called "graphite fire" is not plausible for conditions during bounding transportation accidents. Rather, a process of very slow surface oxidation might occur that proceeds depending strongly on temperature of the graphite (strongly implying high available decay heat or high injected air temperature both of which seem implausible). Extended time at temperature, sufficient oxygen, and possibly water vapor adjacent to graphitic materials must be maintained for over 100 hours at well over 1000 °C simply to oxidize the fuel sleeves and compact graphite (Moormann 2011). However, a small amount of oxidation (if any occurred) could produce gaseous reaction products and heat could potentially release aerosols that contain radioactive material that might have been previously plated in the reactor containment boundary (especially near a failed seal where hot moist air might enter), but a self-sustaining fire, based on the reactor's graphitic materials acting as an oxidizing fuel, seems implausible and the TRISO fuel particles themselves resist air oxidation caused releases up to 1400 °C for well over 100 hours at that temperature. Hence, any increase in releases from a transportation accident involving a fire would be principally associated with materials that had previously plated in the reactor containment boundary.

Moreover, storage of significant Wigner energy in these graphitic structures in this reactor should not pose a significant risk in transportation since the core graphite operates at a high temperature and is therefore annealed. Other heat sources would have to raise the temperature of the graphite above its operating temperature. Thus, any stored energy from irradiation of the graphite should not significantly contribute to any fire hazard during plausible transportation accidents (NUREG/CR-4981, *A Safety Assessment of the Use of Graphite in Nuclear Reactors Licensed by the U.S. NRC* [Schweitzer et al. 1987]).

4.2.3.3 Information Needs in Support of Transportation Risk Analysis

With respect to transportation risk analysis, the likely inability to meet the codified regulations for a Type B package, for the Project Pele TNPP Reactor Module containing irradiated fuel, indicates that mechanical data for fresh and irradiated TRISO fuel and compacts, and other closely associated core components, is essential if it is determined that conservatism should be reduced. Mechanical performance of the reactor core during bounding accidents and transit at speed over rough roadways involving impact loads and vibration spectra transferred to those structures must be evaluated. It is also important to understand possible reconfiguration of the core that could present a criticality hazard in addition to potential failures of TRISO fuel fission product barriers and production of fine particulates that could be a source of radioactive material released into the primary pressure boundary.

To make these risk judgements requires knowledge of fresh and irradiated material mechanical property measurements and improvements in risk modeling parameters. These properties are generally determined as part of the efforts underlying fuel qualification and reactor design certification. The TRISO fuel particles and compacts are complex composite materials which are then irradiated. As such, reliable prediction of involved mechanical behaviors is limited apart from certain bulk measurements including large ensembles of TRISO particles and compacts as systems.

The AGR-1 and AGR-2 qualification studies demonstrate burnup as high as 19% FIMA with no obvious broad failures of SiC layers. At the expected lower burnups in the Project Pele TNPP demonstration design, which may be ~10% FIMA or below, gas pressure is not expected to fail the SiC layers. However, if additionally insulted by potent mechanical impacts, some SiC barriers might fail indicating enhancement of expected release fractions. Another concern is vibration spectra and consequence that is critical to understand fatigue tolerance of the fuel compacts and near fuel materials and tribology that could release fine particles that carry radioactive materials.

The material properties of TRISO fuel, compacts, and close core materials, particularly strength of materials, are very relevant to potential release of radioactive material resulting from a transportation accident. Mechanical impact resulting from an accident is expected to have more impact to these materials than a fire given the properties of these materials. The measured ability of TRISO fuel, compacts, and close core structures to tolerate impact loads and vibration spectra and how that may influence the potential to release radioactive material during a bounding transportation accident has not been examined in detail in the literature to date. This is not surprising since the emphasis has been on larger stationary TRISO fueled nuclear power plants, and the fresh and spent fuels from these reactors were and would be transported in containers that meet the codified regulatory requirements for Type B packages.

The TRISO fuel particles themselves are composite structures composed of layers as shown in Figure 4-1. The UCO kernel and SiC layer are ceramic materials and the PyC layers are anisotropic graphite materials, all of which tend to display a range of brittle failure characteristics and reasonably high failure strengths. Moreover, there is an inner porous PyC layer deposited on the UCO kernel surface that acts as a buffer to allow some expansion of the UCO kernel under irradiation, in addition to providing a gas plenum that contains fission gases as they evolve during burnup.

The mechanical strength of the porous PyC layer is considerably less than the UCO kernel and all the other structural layers that make up the TRISO particle. This reduced strength of the porous layer allows it to sacrificially fail while still retaining some integrity as a “spacer” within the particle, retaining the UCO kernel, roughly centered in the TRISO particle. Moreover, the comparative weakness of the porous PyC layer allows it to delaminate from the inner high density PyC layer which helps protect it from anisotropic mechanical stresses and certain fission product related corrosive attacks from the UCO kernel. Such corrosive attacks can harm the SiC layer that acts as the high-pressure containment vessel of the TRISO particle.

The TRISO particle geometry and degree of bonding between the deposited layers and their behavior under irradiation results in the overall observed composite properties of an individual particle and the ensemble of all the particles taken together determines the composite properties of the compact fuel form in the Project Pele TNPP fuel assemblies (compacts, fuel sleeve, end plugs). This is the case for nearly all the ensemble properties and particularly true for the thermal and mechanical properties. For the mechanical properties that matter most under transportation accident conditions, evaluation of the compact fuel form by finite element modeling is rather complex. Such semi first-principles modeling efforts have not shown high predictive fidelity to any given irradiated TRISO or compact measurements on average. High fidelity modeling tends to be more useful to understand specific failure mechanisms identified through PIE of individual TRISO particles, etc.

There are many reasons for this having to do with the statistical nature of the manufacture of the TRISO particle fuel and manufacture of the compacts as composite structures. As a result, repetitive measurements on irradiated batches of particle fuel and compacts to determine upper and lower bounds and averages that are consistent with a set approach to manufacturing specifications are the standard practice for qualification in a particular specified use of the fuel form.

Some direct measurements of hardness of the containment layers inside the TRISO particles have been made. These measurements were made on unirradiated materials, so they represent the case without radiation damage. Irradiated materials are expected to be somewhat weaker. However, this is a fuel that operates at very high temperature and so some limited annealing may affect the impact of radiation damage, and especially for the SiC pressure boundary since that material begins to decompose at about 500 °C beyond the fuel operating temperature of up to a maximum of about 1400 °C. The PyC materials and UCO kernel do not decompose until far higher temperatures than SiC.

It is possible that the greater concern is certain fission product corrosion attacks that appear to be exacerbated by stress concentration in cracks and defects in the SiC layer. Tiny inclusions of palladium and uranium are indicated as sources of this corrosion cracking inside a TRISO particle. Therefore, to understand the strength of the material following irradiation, it must be actual fuel that is burned up since the corrosion processes and attacks will not exist in an unirradiated surrogate used to ease performance of measurements.

In any event, some very precise and stringent preparations and measurements on actual unirradiated TRISO particle materials have been made (e.g., Byun et al. 2008, Hosemann et al. 2013). To perform these measurements, individual particles are abraded to expose a cross section and then indentation hardness measurements are made. The hardness measurements demonstrate that the SiC layer has over 10 times the hardness of the typical high density PyC material in the adjacent layers. The high density PyC layer material is ~3 GPa whereas the SiC material is ~40 GPa hardness. From these hardness measurements, it is clear that the SiC layer is a rather dominant element in the overall TRISO particle strength. The hardness of the porous PyC buffer layer is not given in those reports since it does not contribute much strength to the composite structure, but rather makes space for gases and acts as a geometric spacer for the UCO kernel.

Some of the abraded TRISO particles are further treated to release the SiC hemi-shell by burning out the graphitic layers using oxygen. The released SiC hemi-shell is then crush tested in a delicate apparatus with a very small end effector to test a single ~0.5 mm diameter hemi-shell. To understand the crushing process mechanically and the involved fracture stress and the local fracture stress (from stress concentration), a detailed finite element model is used to estimate the stress fields during the static crushing process. The SiC layers tested this way have, over a broad set of different manufacturing arrangements, produced fracture stress ranging between ~200 MPa and ~1000 MPa. Local fracture stress is roughly twice the magnitude. In British units, this corresponds to a range between ~29,000 psi and ~145,000 psi. These tiny SiC layers inside a TRISO particle are quite strong.

To produce measurements on TRISO particles that account for weakening from various irradiation processes might involve separation of irradiated particles from the compact host material, burning away of the remaining graphite using oxygen to produce free irradiated TRISO particles (the outer layer of PyC could be neglected mechanically). Then crushing of the freed particles in a similar apparatus as described briefly above (a very delicate piece of equipment). This sophisticated measurement would have to be done in a hot cell on many liberated irradiated individual TRISO particles. Fracture of the particle could be detected by sniffing for a burst of fission gas release. A finite element model could be combined with the measurements to back out a better understanding of the irradiated mechanical strength.

Moreover, crushing tests of whole irradiated compacts would be very useful to understand the strength of the fuel form under various impact load circumstances. Such crushing tests might be driven beyond initial fracture of the compact to determine if there is evidence that individual TRISO fuel particles are failing mechanically, until the compact disassembles. If similar mechanical data are available for other materials in the irradiated core, then reasonably conservative models could be used to understand (and potentially rule out) certain consequences of bounding accident scenarios.

For example, if pressures exerted on TRISO particles are reasonably expected to be far below known static fracture stress, then expectations of release fractions may not include fission products still contained inside the TRISO particle SiC layers (most of the total inventory). If that is proved to be the case, then the SiC layers may partly stand in for containment that would be typically provided by a Type B certified package. But that would have to be measured to determine the statistical spread of PIE data to robustly underpin the argument and quantify the expected limits and determine safe margins.

Similarly, if models, informed by measurement data, show that expected impact loads would not fracture and further disassemble compacts, then certain adsorbed radioactive material in the compacts may also be unavailable for release during bounding transportation accidents. Moreover, if the compacts fail at a much lower load than the TRISO particles (very likely), then the compacts are analogous to some degree of an impact limiting device since they absorb some energy prior to communication of the load to the TRISO particles.

Understanding the relative strength of irradiated core structural components (graphitic and otherwise), TRISO compacts, and TRISO particles in comparison to expected impact loads under various bounding accidents is essential to judging the degree of conservatism in risk assessment release models. To reduce conservatism (if necessary), delineating these failure behaviors via PIE of these components would enable higher fidelity risk modeling. While mechanical testing of unirradiated Project Pele TNPP core structure components, TRISO compacts, and TRISO particles are a starting point, PIE at the upper end of the expected burnup range is essential to understand the envelope of mechanical properties ultimately.

Other issues are the effects of tribology and fatigue. Here, the specific design of the fuel elements and how relative motions at contacting surfaces or oscillating loading within the fuel elements and near fuel structural components might occur are important to understand the potential for either formation of material cracking or other means of formation of released fines or larger particulates that may escape into a coolant gas channel and become mobile during operation (hydraulic chatter) or during mobility and transportation (e.g., vibration and impacts from rough roads, that may be transferred to the core structures over miles of travel).

Initially, some of the uncertainty of these irradiated mechanical properties might be derived from testing on previously irradiated materials from the AGR qualification series experiments. Consideration might be given to devising mechanical tests of these existing irradiated materials to augment unirradiated and irradiated mechanical data that may already exist from the AGR program. The compact design of the Project Pele TNPP is sufficiently different from AGR compact designs such that its behavior may vary somewhat (Project Pele TNPP compact diameters are considerably larger and may prove weaker than AGR compacts – unirradiated testing may reveal this one way or the other), but TRISO level behavior should be very comparable. Arguably, some of the AGR TRISO has considerably higher burnup so it may prove weaker in crush testing and thus prove to be conservative with respect to Project Pele TNPP TRISO fuel.

In this fuel type, a concern would seem to be diffused fission products that may be held up in graphitic materials that may be released into the primary pressure boundary as part of generated fines and larger particulates. It may be that this release pathway is of greater concern regarding occupational doses associated with mobility operations and less with respect to reactor operations or during normal transportation conditions. The Project Pele TNPP design requires opening of the primary pressure boundary to prepare for transportation of TNPP modules or to unload and assemble them for operation at a site. The circulation and accumulation of graphite “dust” has been noted in high-temperature gas-cooled reactor (HTGR) designs since early in the development of the technology (e.g., Van Howe and Raudenbush 1978 discuss issues during the startup and initial operation of the Fort St. Vrain HTGR).

However, generation of fines and other particulates increases the radioactive material inventory that is contained in the primary pressure boundary in the form of deposited and suspended particulate matter and possibly in the form of plated out releases of condensable radioactive material that is fixed to that particulate matter. A portion of this material will be more mobile, within the so-called “circulating inventory”, and thus more readily releasable in a transportation accident as opposed to certain plate out and fines that are more tightly bound to interior surfaces within the primary pressure boundary.

Another common issue that may be more challenging to address in this application is the releasable tritium inventory. The proposed reactor core and reactor gas system design is different from previous HTGR designs. In addition, the Project Pele TNPP design does not have a steam generator and steam cycle in its secondary power generation circuit. Water is a significant tritium sorbing medium in certain HTGR designs. In comparison, the Project Pele TNPP has an open single pass hot air Brayton cycle that uses coarse filtered ambient air as a working fluid. So, humid air, exiting a turbine into the environment, would presumably carry away tritium that permeates through the intermediate heat exchanger, through other secondary circuit seals, etc. This would be in addition to tritium that escapes from the primary pressure boundary into the environment directly and from activation external to the TNPP in nearby materials (e.g., temporary shielding that includes lots of water and various minerals that participate in holdup of tritium).

The Project Pele core also substitutes beryllium in the place of graphite as a moderator to considerable extent. Graphite is known to sorb considerable tritium. The effect of irradiation on beryllium material properties (mechanical strength and thermal conductivity in particular –

through irradiation induced swelling and cracking) is not as mature as for graphite. So, the efficacy of the beryllium moderator to sorb tritium may also be reduced. The tritium release characteristics of the Project Pele TNPP likely do not pose a transportation safety concern since the unit is quite small and full power operation days are limited, but the "open" nature of the TNPP design may imply that consideration of occupational dose management and environmental impact issues may indicate that monitoring of tritium level is important to inform future TNPP variant design and operations.

Large commercial HTGRs that intend to use steam cycle or direct helium Brayton cycle do not have the broader tritium path to the environment via the single pass power cycle. Helium cleanup systems have typically included particle filtration and various molecular sieves and getters to collect condensable metallic and volatile fission products, including hydrogen getters that would collect tritium, thereby reducing the inventory available to escape into the environment (General Atomics 2008). Previous test and demonstration HTGRs have typically included helium cleanup systems (e.g., Van Howe and Raudenbush 1978, Verfondern et al. 2001).

Since no fractional flow rate particulate and gas clean up system is currently envisioned in the Project Pele TNPP's reactor gas system design (other than through gas withdrawal plus fresh coolant from the coolant makeup generator), understanding of generation of fines and larger particulate, plate out, and their carried radionuclide inventories are potentially a more significant part of the transportation risk assessment modeling since respirable dose tends to be dominant in certain assessments. To that end, sampling of fines, any larger particulates, and plate out on the interior surfaces of primary pressure boundary coolant ducts near their joints (when disassembled) is advisable to help inform estimates of the amount and the radioactive inventory contained. Engineered removeable or fixed coupons and/or swipe and scrape sampling procedures might be used to collect information for this purpose.

Moreover, consideration should be given to grab sample collection and examination of fines and gases collected during gas withdrawals from the primary circuit reactor gas system. A filter and a large gas withdrawal collection bottle is envisioned in the design, but it is unknown whether samples can be practically taken from the current envisioned withdrawal system and then evaluated later in a laboratory setting. Evidence of any coolant related oxidation chemistry in the primary pressure boundary should be evaluated (for example, coolant formation and related chemistry). It is plausible that these issues are not significant to transportation safety, but it is advisable to rule them out by collecting and examining samples. Collected information from these filter and grab samples may also lead to learning that advises follow-on TNPP variant designs.

4.2.4 Sources of Radiation Exposure in a Transportation Accident

The radioactive material inventory identified and described in Section 4.2.2 is a source of radiological dose to the worker and the public in the event of a transportation accident. This radiological dose is from three sources: (1) internal exposure to material that is released due to an accident, (2) external exposure to material that is released due to an accident and so is likely unshielded, and (3) external exposure to material that is not released due to an accident, but which may or may not be shielded following the accident. This section discusses the development of each of these contributors of potential radiation exposure due to a transportation accident involving the Reactor Module.

The first two sources of radiological dose are due to the material that is released during/following the accident. The material released is referred to as the source term. The source term is directly related to the quantity of radiological material that is available to be released in an accident, which is referred to as the material at risk (MAR). The development of the MAR is described in Section 4.2.4.1.

The third source of radiological dose is due to the inventory of material that is not released during the accident. Depending on the severity of the accident this material may or may not be shielded. Section 4.2.4.2 discusses the development of this inventory and summarizes the shielding design for the Reactor Module.

4.2.4.1 Development of Material at Risk for a Release

In general, MAR is the quantity of radiological material that is available to be released in an accident. For the prototype TNPP, the MAR includes fission products and actinides produced from irradiation of the HALEU fuel and radionuclides produced from neutron activation of systems, structures, and components (SSCs) and other materials present in the Reactor Module. Depending on the magnitude and scope of the physical stress imposed on the Reactor Module contents because of the accident, the MAR may range from the total inventory in the module, as defined in Section 4.2.2, to a subset of the inventory in the module. The primary contributors to the MAR are as follows:

1. Radionuclides contained within the TRISO fuel particles,

2. Radionuclides that were released from the TRISO fuel particles during normal reactor operations due to defects in the SiC coating,

3. Radionuclides that were released from the TRISO fuel particles during normal reactor operations due to in-service failure of the particles,

4. Radionuclides that were released from the TRISO fuel particles due to diffusion through the SiC coating, and

5. Radionuclides that were produced outside of the TRISO fuel particles due to irradiation of heavy metal contamination.

Other secondary sources of potential MAR in a transportation accident include:

6. Radionuclides produced from neutron activation of the Reactor Module SSCs, such as the reactor pressure vessel, shield tank, graphite moderator, etc.

7. Radionuclides produced from neutron activation of any residual coolant

8. Radionuclides produced from neutron activation of any residual water in the Shield Tank and other components

9. Contamination located on inside surfaces of the Reactor Module, including the exterior surfaces of the reactor pressure vessel and Primary Cooling system, and

10. Contamination located on the exterior surfaces of the Reactor Module.

However, these secondary sources of MAR are expected to be negligible contributors because: (1) neutron activation products in SSCs are contained within the metal/material-of-construction matrix and so are not releasable, (2) the coolant and water are removed from the Reactor Module prior to shipment leaving minor residual quantities, and (3) contamination levels will be maintained low (in trace amounts) by standard operator procedures to maintain worker dose as low as reasonably achievable (ALARA) and to meet NRC/DOT transportation requirements.

The development of the MAR for use in the analysis of a subsequent release due to a transportation accident is primarily based on TRISO fuel fabrication and core material radionuclide retention characteristics. The approach used in this report to develop the MAR that is released from the TRISO during normal reactor operations is based on the approach utilized in INL/EXT-11-24034 (*Scoping Analysis of Source Term and Functional Containment Attenuation Factors* [INL 2012]). To implement this approach, the radionuclides identified in Appendix 8.1 were first binned into the set of fission product classes identified in INL/EXT-11-24034 to facilitate the application of radionuclide-specific release fractions. This binning is shown in Table 4-1.

Table 4-1. Fission Product Classification

INL/EXT-11-24034 Fission Product Class	**TNPP Radionuclides Represented (Appendix 8.1)**
Noble Gases (Xe-133, Kr-85, Kr-88)	Kr-85, Xe-131m, Xe-133, Xe-133m, Xe-135
I, Br, Se, Te	Br-82, I-130, I-131, I-132, I-133, I-135 Se-79, Te-123m, Te-125m, Te-127, Te-127m, Te-129, Te-129m, Te-131, Te-131m, Te-132
Cs, Rb	Cs-132, Cs-134, Cs-135, Cs-136, Cs-137, Rb-86
Sr, Ba, Eu	Ba-136m, Ba-137m, Ba-140, Eu-152, Eu-154, Eu-155, Eu-156, Eu-157, Ga-72, Gd-153, Gd-159, Sr-89, Sr-90, Sr-91, Tb-160, Tb-161
Ag, Pd	Ag-109m, Ag-110, Ag-110m, Ag-111, Ag-112, Ge-77, In-115m, Pd-109, Pd-112, Sn-117m, Sn-119m, Sn-121, Sn-121m, Sn-123, Sn-125, Sn-126
Sb	As-77, Cd-113m, Cd-115, Cd-115m, Sb-122, Sb-124, Sb-125, Sb-126, Sb-127, Zn-72
Mo, Ru, Rh, Tc	Mo-99, Nb-95, Nb-95m, Nb-96, Nb-97, Nb-97m, Rh-102, Rh-102m, Rh-103m, Rh-105, Rh-106, Ru-103, Ru-106, Tc-99, Tc-99m
La, Ce	Ce-139, Ce-141, Ce-143, Ce-144, La-140, Nd-147, Pm-147, Pm-148, Pm-148m, Pm-149, Pm-151, Pr-142, Pr-143, Pr-144, Pr-144m, Sm-151, Sm-153, Y-89m, Y-90, Y-91, Y-91m, Y-93, Zr-95, Zr-97
Pu, actinides	Am-241, Am-242, Am-242m, Am-243, Cm-242, Cm-243, Cm-244, Np-237, Np-238, Np-239, Pa-233, Pu-236, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Th-234, U-236, U-237
Hydrogen (H-3)[1]	H-3

(1) Not included in INL/EXT-11-24034.

Fission products and gases are created in the kernel of the TRISO particles and are a result of fission. Actinides are created in the kernel of the TRISO particles and are a result of neutron irradiation of the HALEU. Most of the fission products, actinides, and gases produced are retained within the TRISO particles. However, during normal operations a very small percentage of the fission products, actinides, and gases may transport to the kernel surface, through the particle coatings, which may or may not be intact, and be released. Once released from the TRISO boundary they are either captured in the fuel compact matrix or the core structure or released into the primary cooling system. Once in the primary cooling system, condensable fission products and actinides plate out on cooler metallic surfaces and or interact with dust in the system while noble gases are assumed to remain in circulation.

As previously discussed, for HTGRs, the major sources of fission products and actinides outside of the TRISO fuel during normal operations result from:

1. Heavy metal contamination in the outer graphite layer of the fuel particles and potentially in the compact graphite.

2. Particles with SiC coating defects.

3. Incremental in-service fuel failures that occur under normal operation.

4. Diffusive release through the fuel particle SiC and other coatings.

The values for the fuel related release parameters, based on assumed 50% and 95% confidence levels, are shown in Table 4-2. The in-service failure values in Table 4-2 are for a prismatic HTGR having a reactor outlet temperature of 900 °C. This is conservative for the prototype TNPP, which has a reactor outlet temperature of 760 °C or 1033 °K (BWXT 2022[19]). These values are about a factor of five higher than allowable 95% confidence levels specified in the fuel performance requirements for historical HTGR designs under normal operations (EPRI 2020), and thus conservative for the risk analysis.

The values selected for heavy metal contamination and SiC defects reflect fuel fabrication (fuel quality) experience in the United States. The heavy metal contamination fractions are the same as those used historically by United States LWR designers in their design assessments. These fractions are about a factor of five higher than allowable 95% confidence levels specified in the fuel performance requirements for historical HTGR designs (EPRI 2020), and thus conservative for the risk analysis. The SiC coating defect fractions are about four times lower than the fuel manufacture defect specification for the maximum allowable particle defect fractions calculated at a 95% confidence level for historical HTGR designs (INL 2010, EPRI 2020). While this assumption is not conservative for the risk analysis, it is reflective of best estimates based on experimental data on TRISO fuel performance (EPRI 2020).

The fuel failure fractions presented in Table 4-2 were assumed to be normally distributed.

[19] BWXT Final Design Report, Table 3.2.2-1.

Table 4-2. TRISO Fuel Fabrication and Failure Parameters – Normal Operations(1)

Fission Product Class	Fabrication				Operations	
	Fraction Heavy Metal Contamination		Fraction SiC Coating Defects		In-Service Failures	
Confidence Limit	50%	95%(2)	50%	95%	50%	95%
Noble Gases	1E-05	1E-04	NA	NA	1.4E-05	7E-05
I, Br, Se, Te	1E-05	1E-04	NA	NA	1.4E-05	7E-05
Cs, Rb	1E-05	1E-04	1E-05	3E-05	2.1E-04	1.05E-03
Sr, Ba, Eu	1E-05	1E-04	1E-05	3E-05	2.1E-04	1.05E-03
Ag, Pd	1E-05	1E-04	1E-05	3E-05	2.1E-04	1.05E-03
Sb	1E-05	1E-04	1E-05	3E-05	2.1E-04	1.05E-03
Mo, Ru, Rh, Tc	1E-05	1E-04	1E-05	3E-05	2.1E-04	1.05E-03
La, Ce	1E-05	1E-04	1E-05	3E-05	2.1E-04	1.05E-03
Pu, actinides	1E-05	1E-04	1E-05	3E-05	2.1E-04	1.05E-03
Hydrogen (H-3)(3)	1E-05	1E-04	1E-05	3E-05	1.4E-05	7E-05

(1) Except as noted values are from INL/EXT-11-24034 (INL 2012).
(2) Heavy Metal Contamination 95% values are from INL/MIS-21-62587, *Microreactor TRISO Fuel Specification* (INL 2021).
(3) Fabrication and operations factors for Hydrogen (H-3) are assumed to be the same as for Noble Gases.

The second key element with respect to radionuclide release is the attenuation of the release. Attenuation factors (AF) represent the capability of the identified barrier to retain fission products and actinides released due to the mechanisms discussed above. The AFs used in this report were estimated by an expert panel based on HTGR fuel testing results and radionuclide transport predictions developed for previous HTGR designs such as the modular HTGR and Pebble Bed Modular Reactor (PBMR) (INL 2012). It is important to note that in developing the AF values, the experts accounted for the plant configuration and the design service conditions for the conceptual HTGR designs developed for the Next Generation Nuclear Plant (NGNP) project. No attempt was made in this report to update the AF values to account for the TNPP fuel element and reactor design and plant configuration. This is a source of uncertainty for this report.

The AFs during normal operations, based on assumed 50% and 95% confidence levels, are shown in Table 4-3. Again, the AFs in Table 4-3 are conservative from the perspective that they are for a prismatic HTGR having a reactor outlet temperature of 900 °C. The AFs are assumed to have a lognormal distribution.

The heavy metal contamination AFs reflect attenuation of the products of irradiation of heavy metal contamination on the outer surface of the TRISO fuel particles into the surrounding graphite moderator. The kernel AFs reflect attenuation of fission products and actinides in TRISO fuel particles having failed coatings (either as a result of fabrication or in-service failures) through the UCO kernel and into the surrounding graphite moderator. The diffusive release thru coating AFs reflects attenuation of fission products and actinides through the intact TRISO fuel particle kernel and coatings due to diffusion and into the surrounding graphite moderator. The graphite AFs reflect the attenuation of fission products and actinides released from the TRISO fuel particle, due to the previously described mechanisms, through the graphite moderator into the Primary Cooling system.

It is noteworthy that some of the AF results presented in Table 4-3 are not, on face value, intuitive, but are nevertheless reflective of experimental and past operational results. For example, the AFs are higher for the noble gases and I, Br, Se, and Te classes than for some of the other classes, such as the Cs, Rb class, because the PyC layer in the TRISO fuel particle is more effective at retaining noble gasses (Kr, Xe), halogens (I, Br), and tellurium than metals classes such as alkali metals (Cs) and noble metals (Ag). A similar dichotomy is shown for heavy metal contamination to account for the contribution to releases from each in the experimental data.

Table 4-3. Normal Operations Attenuation Factors(1)

Fission Product Class	**Heavy Metal Contamination**		**Kernel**		**Diffusive Release thru coating**		**Graphite**	
Confidence Limit	50%	95%	50%	95%	50%	95%	50%	95%
Noble Gases	5	1.5	25	8.33	5E+07	5E+06	1	1
I, Br, Se, Te	5	1.5	25	8.33	5E+07	5E+06	1	1
Cs, Rb	1	1	1.2	1	1E+07	1E+05	2	1
Sr, Ba, Eu	1	1	3	1	500	100	100	30
Ag, Pd	1	1	1	1	200	40	1	1
Sb	1	1	1	1	5E+07	5E+05	5	1
Mo, Ru, Rh, Tc	1	1	250	15	1E+07	1E+06	200	60
La, Ce	1	1	250	15	1E+07	1E+06	200	60
Pu, actinides	1	1	500	50	1E+07	1E+06	5E+03	500
Hydrogen (H-3)(2)	5	1.5	25	8.33	5E+07	5E+06	1	1

(1) Except as noted, Attenuation Factors are from INL/EXT-11-24034 (INL 2012).
(2) Fabrication and operations factors for Hydrogen (H-3) are assumed to be the same as for Noble Gases.

The TRISO fuel particle fabrication and failure parameters and the AFs are used to develop the TRISO fuel release fractions from normal reactor operations. For the purposes of the risk analysis, material released from the TRISO fuel during normal reactor operations are assumed to be contained within the reactor core structure (i.e., fuel compacts) or in the reactor containment boundary. Released material inventory is then available for release during transportation accidents. Any fuel inventory not released continues to be retained within the intact TRISO fuel particles, although additional releases are possible from TRISO fuel particles that are intact after completion of normal operations due to a severe transportation accident. Releases due to severe transportation accidents are addressed in Section 4.6.

Thus, for normal operations, for each individual radionuclide, the MAR potentially available for release in a severe transportation accident is estimated for each of the three locations as follows:

Inventory of radionuclide i released into the reactor core structure or graphite fuel compacts (CS)

$$Release_{i,CS} = Inv_i \times [\frac{RP_{i,HMC}}{AF_{i,HMC}} + \frac{RP_{i,FD} + RP_{i,ISF}}{AF_{i,K}} + \frac{1}{AF_{i,DIF}}]$$

where:

Inv$_i$ = the total inventory of radionuclide i in the reactor fuel (Appendix 8.1, Table 8.1-1)

$RP_{i,HMC}$ = release parameter (RP) or fraction of the radionuclide i inventory that is heavy metal contamination (HMC) (Table 4-2)

$RP_{i,FD}$ = RP or fraction of the radionuclide i inventory that is in defective TRISO fuel particles (FD) (Table 4-2)

$RP_{i,ISF}$ = RP or fraction of the radionuclide i inventory that is in TRISO fuel particles that failed in-service (ISF) (Table 4-2)

$AF_{i,HMC}$ = HMC attenuation factor for radionuclide i (Table 4-3)

$AF_{i,K}$ = fuel kernel (K) attenuation factor for radionuclide i (Table 4-3)

$AF_{i,DIF}$ = diffusive release through TRISO coating (DIF) attenuation factor for radionuclide i (Table 4-3)

MAR for radionuclide i that is released into the Primary Cooling System

$$MAR_{i,PB} = \frac{Release_{i,CS}}{AF_{i,G}}$$

where:

$AF_{i,G}$ = graphite attenuation factor for radionuclide i (Table 4-3)

MAR for radionuclide i that is released into the reactor core structure (CS)

$$MAR_{i,CS} = Release_{i,CS} - MAR_{i,PB}$$

MAR for radionuclide i that is retained in intact TRISO fuel particles

$$MAR_{i,TRISO} = Inv_i - MAR_{i,CS} - MAR_{i,PB}$$

Noble gases and hydrogen released from the TRISO fuel were assumed to remain in the coolant during reactor operation. Prior to transportation, this system is depressurized. As such, these gases do not contribute to public or worker exposure in a subsequent transportation accident. All other fission products released to the coolant boundary were assumed to plate out and are included in the MAR available for release.

The process for developing each of the three categories of MAR was to perform a Monte Carlo analysis of 100,000 trials for one of the radionuclides in each of the fission product classes defined in Table 4-1. The mean and 95th percentile inventory values were extracted for each of the three MAR categories for each radionuclide. Mean and 95th percentile release fractions were then developed for each radionuclide, or release category, for releases into the reactor core structure and into the primary cooling system pressure boundary by dividing the applicable Monte Carlo-estimated released inventory by the total original core inventory. The resultant release fractions are provided in Table 4-4. These releases fractions were multiplied by the total inventory of the applicable radionuclides in each release category to obtain the inventory of each radionuclide released to each location.

Table 4-4. Release Fractions from Normal Operations

Fission Product Class	Reactor Core Structure		Reactor Cooling System Pressure Boundary	
Confidence Limit	Mean	95%	Mean	95%
Noble Gases	0	0	8.2E-06	3.2E-05
I, Br, Se, Te	0	0	8.0E-06	3.2E-05
Cs, Rb	1.5E-04	5.1E-04	1.6E-04	5.5E-04
Sr, Ba, Eu	3.4E-03	9.9E-03	1.9E-05	6.7E-05
Ag, Pd	0	0	8.4E-03	2.5E-02
Sb	2.6E-04	8.6E-04	1.0E-04	4.5E-04
Mo, Ru, Rh, Tc	3.3E-05	1.1E-04	2.1E-07	8.7E-07
La, Ce	3.3E-05	1.1E-04	2.1E-07	8.7E-07
Pu, actinides	2.9E-05	1.0E-04	1.5E-08	6.6E-08
Hydrogen (H-3)	0	0	8.0E-06	3.2E-05

4.2.4.2 Unreleased Inventory

The primary and secondary sources of MAR discussed in the previous section could also be the source of direct radiation exposure in a transportation accident. Direct radiation exposure is addressed as a separate radiological dose pathway to workers and the public. This section discusses the TNPP radionuclide inventory that could be the source of direct radiation exposure if: (1) the radiological material becomes unshielded (or partly unshielded) in a transportation accident scenario in which the TNPP package shielding is damaged, or (2) an undesired situation develops resulting in longer worker exposure time to radiation during the transport than planned. The sources of potential direct exposure overlap with those discussed in the previous section and consist of:

1. Radioactive material contained within irradiated intact TRISO fuel particles,

2. Radioactive material released from the TRISO fuel particles and from heavy metal contamination during normal reactor operations and is held up in the reactor core structures and primary cooling system, and

3. Radioactivity resulting from neutron activation of Reactor Module components during normal reactor operations such as (BWXT 2022):

    - Fuel compacts (graphite) and graphite moderator blocks,

    - Reactor core support structures,

    - Control rods and control rod drive mechanisms (CRDM),

    - The reactor pressure vessel (RPV) and shield tank,

    - Saddle mount for cooling water system or shield tank,

- The Reactor Module structural components, including CONEX box and external lead shielding,
- Copper wiring for instrumentation and control, and
- Lead, steel, and tungsten shielding that are integral to the reactor system.

4.3 Identification of TNPP Package Safety Functions

This section discusses the identification of transportation TNPP package safety functions. The vendor transportation plan includes identification and discussion of safety functions associated with the transport of irradiated fuel that will be evaluated in detail in the next phase of the project. These safety functions consist of containment of radiological material, radiation shielding, and maintaining criticality safety, which are typical evaluation topics in a Safety Analysis Report (SAR). However, the vendor transportation plan also discusses the need for passive heat removal while the reactor is in shutdown mode during transportation. These safety functions are discussed in this section.

Regarding the containment of radiological materials safety function, the vendor discussed in their final Phase I design documentation the testing and modeling that will be performed in Phase II to demonstrate transportation safety during NCT and HAC. Dynamic finite element analysis (FEA) will be performed to determine how transportation accident loads may transfer through the package and reactor core. Data from physical testing will be generated to support these assessments.

Currently there is little available information on the performance of the transportation package, or Reactor Module, or its contents during its transport. Preliminary analysis of the Reactor Module has been completed for selected NCT and HAC based on the requirements of 10 CFR Part 71. Specifically, shock, vibration, free drop, and penetration assessments have been performed using FEA. These preliminary assessments did not assess the potential for damage to the TRISO fuel particles and compacts. Based on vendor reports the following were evaluated (BWXT 2022[20]):

- Normal Conditions of Transport – A shock load evaluation and a vibration load evaluation were performed using U.S. Department of Defense (DOD) Military Specification MIL-STD-810H (DOD 2019). Very little damage resulted from these two types of over-the-road vibrations. There was some minor yielding in the tube sheet that supports the control rod tubes and saddles but no plasticity in the reactor vessel, shielding, bracing, or container.
- Hypothetical Accident Conditions – Based on 10 CFR 71.73, a puncture evaluation (assuming a 6-in. diameter steel cylinder) and a free drop evaluation (assuming 30 ft at any angle) were performed. For the puncture evaluation (which did not include the reactor internals and internal support structures) at the critical puncture angle, the container, shielding, and pressure vessel were able to prevent perforation or cracking of the pressure vessel boundary, leaving instead a significant dent in the vessel steel (however, additional puncture analysis is proposed). The free drop evaluation resulted in considerable damage and significant acceleration of internals. Damage to the shielding layers is expected in all angles of drop and fracture of these

[20] BWXT Final Design Report, Appendix III.44, “BWXT Reactor Design Preliminary Transportation and Severe Accident Analyses,” Executive Summary.

shielding layers may result in increased measured radiation outside the package. Breach of the containment boundary is possible in both end-drop conditions as well as oblique corner drops.

The Hazardous Condition Evaluation (Appendix 8.2) postulates accident conditions from a broad range of possible hazards which includes impacts from highway accidents as well loss of safety function caused by other hazards such as road vibration and weather events. The Hazardous Condition Evaluation is the primary element of the hazard analysis, and therefore, the term "hazard analysis" is used to refer to this primary element of the assessment or the whole process. As described in Section 4.4.2, identification of hazards, which is the starting point of the Hazardous Condition Evaluation, came from vendor's design information plus expert knowledge of additional hazards specific to transportation. Screening of hazardous condition qualitatively judged to be of low risk is also part of the hazard analysis process.

Meeting the HAC test conditions prescribed in 10 CFR Part 71 is likely not feasible; therefore, the TNPP transportation PRA and associated risk information will be used to support the 10 CFR 71.12 process. Based on the current vendor design information, it is not known what deterministic requirements may not be met. Therefore, it is necessary to make assumptions in the TNPP transportation PRA about the fragility of the design to accident phenomena. These assumptions are documented and their impact on the estimated risk associated TNPP transportation is investigated using sensitivity studies that explore the impact of different levels of conservatism. Different forms of radiological material containment exist starting with the TRISO fuel itself which is considered the first level of containment. The fuel is encased in ceramic material that is hardened against high temperature as described in Section 4.2. This barrier must be breached before radioactive material within the TRISO fuel is released in a transportation accident. The vulnerability of the TRISO fuel to physical phenomena that could occur during a transportation accident needs to be evaluated. During the accident development stage of the PRA using hazard analysis, the potential for release of radiological material from the TRISO fuel is assessed against physical phenomena that could occur during the TNPP transportation accident such as mechanical impact.

Another form of containment for the transportation package is the reactor vessel; including connected systems such as the control rod drive system and associated piping such as the primary cooling system for which there will be portions that remain connected to the reactor vessel for transportation (this is defined as the reactor containment boundary). Again, as described in Section 4.2 for an irradiated core, there are fission products from the TRISO fuel that have diffused during reactor operation into the core structure and material that has plated-out in the reactor containment boundary. Radioactive material from these locations could be released into the air if corresponding containment features are breached. To evaluate the containment function afforded by the reactor containment boundary, it needs to be assessed against the possible physical phenomena that could be encountered in an TNPP transportation accident scenario. These features of the reactor vessel system that should be assessed include seals, lids, welds, cover plates, valves such as relief valves, drains, joints and connections, and mechanisms such as closure devices used to maintain containment when the primary cooling system is disconnected. Material properties of all these components are an important consideration. Tests are used to establish that the normal leak rate meets the requirements of 10 CFR 71.51 for NCT.

Another form of containment is the transportation package or module itself because it includes the CONEX box and may contain a portion of material released from the reactor containment boundary. In addition, surface contamination may be present outside the reactor vessel and primary cooling system. This is likely to be low level surface contamination that could become released in a transportation accident.

Regarding the radiation shielding safety function, shielding internal and external to the reactor module is expected to be used during transport. These features include built-in and supplemental bolt-on tungsten shielding. Tungsten shielding is used during transportation because normal shielding water used during operation would be too heavy for transport. To evaluate the shielding function that the shielding elements afford, they need to be assessed against the possible physical phenomena that could be encountered in an TNPP transportation accident scenario. These physical phenomena, such as mechanical impact and fire, are likely to be the same phenomena that can damage the package and cause release of radiological material. Tests and analyses are used to establish that the level of radiation from the TNPP package at the surface of the package meets the requirements of 10 CFR 71.47 ("External radiation standards for all packages") and 49 CFR 173.441 ("Radiation level limitations and exclusive use provisions") for NCT. The PRA will be used to determine the risk associated with conditions for which the TNPP package shielding function does not meet the requirements for HAC specified in 10 CFR 71.51.

Regarding criticality safety, the vendor design addresses this possibility and has proposed transportation poison rods to preclude a control rod withdrawal criticality accident. However, transportation poison rods are not proposed for the current demonstration unit. Concerning an addition of moderator scenario that results in criticality, the vendor stated that the prototype design will not preclude criticality during a water immersion inundation event. Though both types of criticality scenarios may have very low likelihoods, criticality scenarios are included in the accident development process by postulating these events in the hazard analysis. Given applicable 10 CFR Part 71 requirements are not met, the risk of criticality accidents need to be evaluated in the PRA. Highway accidents that involve significant impact with moving or fixed objects could cause a control rod withdrawal event if the mechanisms that keeps the control rods inserted fails. Highway accidents that result in the TNPP transportation package being submerged in a body of water are included in the PRA.

The PRA will be used to determine the risk associated with conditions for which the TNPP package does not meet the requirements for HAC specified in 10 CFR 71.51.

The vendor design documents address passive heat removal of decay during transportation for reasons other than as a required safety function. For example, it is indicated that passive cooling will be required during transport to "ensure that critical electronics and systems can properly function." However, this does not imply a nuclear safety function. Remote parameter monitoring of these systems is expected to be implemented to provide real-time health diagnostics to allow timely response to be made for abnormal conditions that may occur during transport. The proposed parameter monitoring system (i.e., Health Monitoring Instrumentation System [HMIS]) is expected to monitor such parameters as airborne and direct radiation, reactor containment boundary pressure and temperature, control rod position, and shock and vibration. It is expected that decay time has a significant impact on the decay heat that is possible during TNPP transportation with irradiated fuel. Assumptions for the TNPP transportation PRA are made about the maximum possible residual heat load that could occur during transport based on the time since shutdown.

However, loss of passive heat transfer of decay heat during transportation of the TNPP package does not appear to lead directly to a transportation accident but rather is expected to potentially cause degradation of the reactor caused by damage to materials due to exceeding their maximum allowable use threshold. Such increase in temperature could be detected by a HMIS if included. Failure of passive heat transfer of decay heat caused by human error or other failures may also lead to other effects such as increasing the pressure inside the reactor vessel which are addressed in the hazard analysis for its impact on the containment safety function.

The conclusions based on examination of safety function required during transportation are that they are the same as for traditional transportation packages of high-level radioactive material (i.e., containment of radiological material, shielding from radiological material, and maintaining criticality safety). These safety functions were considered during identification and development of accident scenarios as part of the hazard analysis. Additionally, it is noted that loss of passive heat transfer of decay heat during transportation of the TNPP package could lead to possible degradation of the reactor caused by damage to materials that exceed their maximum allowable use threshold, though it is not expected to impact the reactor containment boundary safety function. Even though this degradation is not the source of a transportation accident it could have a safety related consequence if the damage went undetected after the reactor is reassembled and operated.

4.4 Identification and Development of TNPP Package Transportation Accident Scenarios

This section describes the identification and development of TNPP package transportation accident scenarios. It discusses the approach for identifying and defining accident scenarios that could lead to a release of radioactive material, loss of shielding, or criticality. As described in Section 4.1, a PRA is typically founded on a comprehensive identification of what can go wrong. In principle, a PRA would consider all credible accident scenarios that result in release of radioactive material to the environment or in direct radiation exposure to workers or the public. In practice, high-likelihood low-consequence accident scenarios are not expected to be as important as low-likelihood high-consequence accident scenarios because packages and containers for transporting radiological material are designed to be very robust, even those that do not meet Type B packaging requirements. This section provides a discussion of the general approach to identifying TNPP transportation package accident sequences in Section 4.4.1, use of hazard analysis to identify hazardous conditions and make qualitative estimates of their risk that are then used to develop TNPP transportation accident scenarios in Section 4.4.2, and determination of the TNPP transportation accident scenarios to include in the PRA in Section 4.4.3.

4.4.1 Approach to Development of Accidents Scenarios

In general, development of accident sequences for a PRA consists of three major elements: (1) identifying the accident sequence initiating events, (2) developing system response models that define how the item of interest responds to the initiating event which can include consideration of design features meant to prevent or mitigate the consequences of an accident and administrative controls, and then (3) defining the sequence of events that leads to undesired outcomes as described in the RIDM report (NRC 2008).

An initiating event can be a system upset or failure, a human error, or an external event (e.g., an event outside the system or activity of interest like a natural phenomenon event). Identification of initiating events requires a systematic search across the range of events that can affect the system of interest. There are multiple methods for identifying initiating events for PRA including inductive and deductive approaches and searching through event data. Inductive approaches (i.e., bottom-up) include use of a hazard analysis or hazard identification checklist and are particularly useful when an understanding of the broad range of possibilities is needed (Coles et al. 2021). Deductive approaches (i.e., top-down) include use of a Master Logic Diagram that defines a top event (e.g., reactor core damage) and delineates all the ways in which the top event can occur. For systems and activities for which event data may be incomplete, it is common to identify possible accident scenarios using a hazard analysis to identify potential hazardous conditions that can lead to undesired outcomes (NRC 2008).

Previous transportation risk assessments have defined transportation accidents with the aid of an event tree such as the event trees developed for transportation risk assessment studies performed by the NRC like the study presented in NUREG-2125 (*Spent Fuel Transportation Risk Assessment* [NRC 2014]). The event trees presented in NUREG-2125 for transportation of such packages as spent nuclear fuel casks consist mostly of accident sequences associated with various kinds of high-energy highway vehicle accidents such as collisions with moving vehicles (e.g., cars, trucks, and trains) or fixed objects (e.g., buildings, trees, bridge abutments, interstate structures, or the ground after a fall to a lower elevation), and non-collision accidents (e.g., rollover or jack-knife). A fire or explosion could happen randomly while the transport vehicles are in motion or stationary, or could happen as a result of a highway vehicle accident. Accordingly, this kind of event tree is typically constructed using transportation accident data and geographic information system (GIS) data.

The event trees like those shown in NUREG-2125 are mostly useful for sorting out the different kinds of highway vehicle accidents that can occur during transport opposed to defining the course of accident scenarios based on the success or failure of different nodes that correspond to various prevention and mitigation systems functions (e.g., the course of an accident after a large pipe break at a nuclear powerplant). It is noteworthy that the NRC transportation risk assessment reports that describe the use of event trees do not refer to these models as PRA models. For the TNPP transportation PRA, event trees are not explicitly developed for the TNPP transportation PRA presented in this report because they are viewed as having limited value in this application. Rather, the accident scenarios are defined independent of each other and in enough detail so that adequate likelihood and consequence analyses can be defined and performed. None-the-less, the results of the accident analysis identification process were compared to the transportation events trees like the ones shown in NUREG-2125 as a way to review the comprehensiveness of the accident scenarios identification and development.

Also, as discussed in Section 4.1, the use of fault trees in the TNPP transportation PRA for accident sequence development is seen as having limited value at this stage of Project Pele. For the most part, complex system analysis (e.g., failure of the control rods to SCRAM or Emergency Diesel Generators to start) using fault trees is not required or beneficial at this point in the TNPP transportation PRA. Fault trees are a useful way to understand the combinations of random failures that could happen subsequent to an initiating event and result in an undesired outcome. However, in this TNPP transportation PRA, the failures that occur during accident scenarios are primarily the result of the initiating event itself as opposed to subsequent random failures. Therefore, like event tree development, the development of fault trees is not included at this phase of project. That said, it is possible that at a later stage (e.g., Phase II) more detailed modeling could be beneficial. For example, if a system such as a HMIS were identified to be a key mitigating system, then it might be important to model the system using a fault tree to gain a more accurate understanding of the risk associated with certain accident scenarios. Accordingly, hazard analysis is used as a systemic way to identify and define TNPP transportation accidents that are important contributors to risk and need to be evaluated in detail.

The accident scenarios defined for TNPP transport are not complex in terms of requiring models of multiple active interdependent systems like the nuclear power plant safe shutdown systems. However, the TNPP package transportation PRA is a first-of-its-kind endeavor, and the associated hazards are apt to be different from transportation of nuclear material in approved containers, casks, or packages or cited stationary reactors. Therefore, an approach is needed that explores a broad range of possibilities and does not depend on specific accident event data. It is judged that hazards identification and assessment meets those criteria as described in the RIDM report and PNNL-31867 (Coles et al. 2021). Therefore, identification and assessment of possible TNPP transportation hazardous conditions is presented in Section 4.4.2.

In principle, a PRA would consider all credible accident scenarios that result in the release of radioactive material to the environment or in direct radiation exposure to workers or the public. In practice, however, low-likelihood high-consequence accident scenarios will likely dominate the risk because packages and containers for transporting radiological material are designed to be very robust. If such accident scenarios can be shown to be insignificant contributors to risk, then there is a basis for not calculating the risk of all possible accident scenarios. Other rationale for not being overly comprehensive is that the design and transportation details and certain PRA modeling input information needed to perform a detailed comprehensive evaluation are not yet complete necessitating use of a significant number of PRA modeling assumptions. Therefore, a useful strategy for this initial stage of the Project Pele (and perhaps also applicable for later PRAs) is to develop representative and bounding accidents to reduce the number of accidents that need to be quantified which facilitates exploration of the impact of different sources of modeling uncertainty on the risk estimates. Accordingly, bounding representative accident scenarios are defined in a way to be representative of a group of accident scenarios that are similar, but bounds the risk (i.e., has the highest risk) of all variations of the accident scenarios in the group. The description of defining bounding representative accidents based on identification of and description of the full set of risk important accident scenarios is presented in Section 4.4.3.2

4.4.2 Identification and Assessment of TNPP Transportation Hazardous Conditions

This section describes how identification and assessment of hazardous conditions during TNPP transportation was used as a way to systemically identify TNPP transportation accidents that are important contributors to risk. It describes the use of hazard analysis sessions using subject matter experts in the process familiar with TNPP vendor designs to generate a comprehensive listing of hazardous conditions of concern. It also describes the information captured on the hazardous condition worksheets by the experts which was used generate TNPP transportation accident scenarios.

The RIDM report (NRC 2008) explains that hazard analysis, in addition to being an alternative approach to PRA (i.e., an ISA approach as discussed in Section 4.1), can also be used to support a PRA as mentioned in Section 4.4.1. Appendix D of the RIDM report discusses the use of hazard evaluation methods such as a Hazards and Operability Study (HAZOP), which is commonly used in the chemical industry, to help identify and construct potential accident event sequences. Additionally, DOE-STD-3009-2014 (*DOE Standard – Preparation of Nonreactor Nuclear Facility Documented Safety Analysis* [DOE 2014]) provides useful guidance on non-quantitative risk characterization based on assessment of postulated hazardous conditions. This general approach was applied to identify hazardous conditions for TNPP transportation. The U.S. Department of Energy (DOE) standard defines the term "hazard analysis" as:

> "The identification of materials, systems, processes, and plant characteristics that can produce undesirable consequences (hazard identification), followed by the assessment of hazardous situations associated with a process or activity (hazard evaluation). Qualitative techniques are

> usually employed to pinpoint weaknesses in design or operation of the facility that could lead to accidents. The hazard evaluation includes an examination of the complete spectrum of potential accidents that could expose members of the public, onsite workers, facility workers, and the environment to radioactive and other hazardous materials."

Many of the hazard analysis approaches referred to above can be used to make qualitative or semi-quantitative estimates of the risk to assess hazardous conditions by assigning those conditions to likelihood and consequence severity categories.

A series of expert panel sessions were held over the course of few weeks in late February and early March 2022 to identify and assess hazardous conditions associated with TNPP transport. The session participants were experts in PRA (i.e., nuclear power plant PRA and transportation of nuclear material risk assessment), hazard analysis, nuclear safety analysis, and nuclear material packaging safety who made themselves familiar with TNPP vendor designs. The session experts filled out a hazardous condition worksheet to generate a comprehensive listing of postulated hazardous conditions of concern and evaluate their risk. The conditions of concern were those that could defeat the safety function of the TNPP transportation package identified in Section 4.3 of this report and pertain to maintaining criticality safety, maintaining radiation shielding, ensuring containment of radiological material, and passive heat removal during transport. The Hazardous Condition Evaluation Worksheet used to capture the hazardous analysis results are discussed in Section 4.4.2.1. The assumptions made about the microreactor design and transport to support the hazardous condition evaluations are discussed in Section 4.4.2.2. Applicable hazards are used a starting point to generate the hazardous conditions postulated in the worksheet. These hazards include those identified in the vendor design documents for a stationary reactor plus expert knowledge of additional hazards that are specific to transportation.

4.4.2.1 Hazardous Condition Evaluation Worksheets

The worksheets were filled out by first considering the hazards identified in the Project Pele vendor Phase I design reports for stationary operation of the TNPP that could potentially also pertain to transport of the TNPP.

In addition, hazards exclusively associated with transportation were added based on the description of transport of the TNPP package provided in the vendor's Phase I reports and detailed knowledge of transportation risk based on having performed previous transportation risk assessments. Regarding the TNPP design and transport process, the hazard analysis team relied on information in the Project Pele vendor Phase I design reports as clarified in some cases by the vendor. A listing of the primary assumptions used in the hazard analysis is presented in Section 4.4.2.2.

This process considered hazards such as the kinetic energy associated with moving vehicles and thermal energy associated fires such as diesel fuel fire. The process also considered hazardous conditions that could occur for a stationary reactor but created different hazardous conditions for a TNPP in transport. This included loss of confinement of the TNPP package, hazards associated with natural phenomenon like severe weather, and human errors in preparing for transport that could lead to failure or degradation of the TNPP package. These worksheets were produced for following hazard categories as presented in Appendix 8.2:

- Table 8.2-1 – Fire Hazard Events.
- Table 8.2-2 – Explosion Events.

- Table 8.2-3 – Kinetic Energy Events.
- Table 8.2-4 – Potential Energy Events.
- Table 8.2-5 – Loss of Containment Events.
- Table 8.2-6 – Direct Radiological Exposure Hazard Events.
- Table 8.2-7 – Criticality Events.
- Table 8.2-8 – Man-Made External Events.
- Table 8.2-9 – Natural Phenomena Hazards.

The hazard analysis does not include consideration of hazardous conditions that occur uniquely during dismantlement of the TNPP, loading it onto the transport trailers, unloading it from the transport trailers, or reassembling the TNPP modules, except to the extent to which latent errors or failures occur that do not manifest themselves until transport of the TNPP package. While these activities might have an important contribution to overall risk of reactor operations, they are not considered to be within the scope of the TNPP PRA which provides a risk-informed basis for on-the-road transportation.

The first column on the left side of the worksheets for a given hazard category (e.g., Fire Hazard Events) is labeled Event Class which is a subdivision of the hazard category. For example, the Events Classes for the Fire Hazard Events category are General Fire, Diesel Fuel Fire, Oil and Grease Fire, and Graphite Fire. The second column is labeled the Initiating Event Category which describes how the hazardous condition came into being (i.e., how it was initiated). For example, the first Initiating Event Category in the Fire Hazard Events worksheet which is under General Fire is "Ignition of flammable materials in a transport container (e.g., associated with the module, the overpack, or system components)."

The third column is labeled the Hazardous Event Summary and is a description of the hazardous condition. Given the safety functions that must be preserved during transport as discussed in Section 4.3 (i.e., containment, shielding, and prevention of criticality), the Hazardous Event Summary always concerns: (1) a release of radiological material to the environment, (2) direct radiation exposure (or an increase worker radiation exposure), or (3) a criticality which potentially involves both direct radiation and release of radiological material. In terms of the PRA, the Hazardous Event Summary is essentially a description of the accident scenarios. The fourth column is a Initiator Likelihood that the hazardous condition occurs as defined in the Hazardous Event Summary. The Initiator Frequency designations are common ranges used in hazard analysis as shown in the following:

- Anticipated (Frequency ≥ 1E-02).
- Unlikely (1E-02 > Frequency ≥ 1E-04).
- Extremely Unlikely (1E-04 > Frequency ≥ 1E-06).
- Beyond Extremely Unlikely (1E-06 > Frequency).

The fifth column is a qualitative Consequence Description (i.e., Physical Consequences) of the outcome of the hazardous condition defined in the Hazardous Event Summary in terms of damage that impacts radiological inventory of the TNPP package. The sixth column (i.e., Qualitative Risk Characterization) is a qualitative characterization of risk as High, Moderate, or Low to the workers involved in the transport and to the public. Included in this column is identification of MAR potentially released or part of the radiological inventory of the TNPP package that becomes unshielded and could cause direct exposure to a worker or the public. As described in Section 4.2 of this report, the following contributors to the MAR are selected as applicable for each hazardous condition (i.e., accident scenario):

1. Nongaseous fission products contained within the TRISO fuel or heavy metal contamination within the compacts that subsequently damaged in an accident,

2. Fission gases contained within the TRISO fuel or heavy metal contamination within the compacts that are subsequently damaged in an accident,

3. Fission products from the TRISO fuel that has diffused and is held up in the core structures,

4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary (i.e., reactor pressure vessel or primary cooling system),

5. Contamination outside the reactor.

The seventh column of the worksheets (i.e., Preventive SSCs) identifies SSCs that could prevent the hazardous condition (i.e., accident scenario) and the last column of the worksheets (i.e., Mitigative SSCs) identifies SSCs that could mitigate the risk from the hazardous condition (i.e., accident scenario).

Thus, the hazard analysis worksheets provide: (1) identification of the hazardous conditions that could occur during transportation of the TNPP package, (2) a semi-quantitative judgment of the likelihood of each hazardous condition, (2) a qualitative description of the consequences of each hazardous condition, (4) a qualitative description of the risk associated with each identified hazardous condition, and (5) identification of preventative and mitigative features and systems to reduce the risk associated with the hazardous condition. For the sake of aligning with the PRA, the hazardous conditions were formulated in a way to describe accident scenarios. For example, hazardous conditions that involve a release from the TNPP package were formulated as a release of radiological material from the TNPP package (or some specific part of package) to the environment caused by a named hazard due to specific conditions created by or associated with the hazard.

The hazardous conditions that were postulated and evaluated in the process described above are presented in Appendix 8.2 of this report after removing hazardous conditions that were deemed not to be applicable to transport (i.e., only applicable to stationary operation of the reactor).

4.4.2.2 Hazardous Condition Evaluation Assumptions

As stated in Section 4.4.2.1, the basis for the hazard analysis was the TNPP design and transportation process information from the Project Pele vendor Phase I design reports as clarified in some cases by the vendor. A listing of the primary assumptions used in the hazard analysis is presented below. Specific assumptions about other aspects of the PRA such as factors important to estimating the accident likelihoods and factors important to estimating the radiological consequence from a transportation accident are identified in the sections of the report that address those analyses in detail (i.e., Sections 4.5.4 and 4.6.4).

1. The hazard analysis focuses on the Reactor Module because it contains the reactor, the fuel, portions of the primary cooling system and nearly all of the radiological material inventory. There may be portions of the primary cooling system with radioactive material, including the intermediate heat exchange (IHX) and piping connecting the reactor to the IHX, that are transported in a separate module or containers, but it is assumed that these pieces will be shipped as Low Specific Activity (LSA) or Surface Contaminated Object (SCO). Also, the

radiological contamination or activated material that might exist in the other modules is assumed to be very low, and therefore is not explicitly addressed.

2. It is assumed that the TNPP transportation package also includes spent fuel after a specified period of decay as described in the consequence analysis presented in Section 4.4.2.

3. It was assumed that there is no gas clean-up system in the design, so its contribution to radioactive transportation inventory is not considered, neither for removal of fission products released during normal operations nor as a source of radiological MAR.

4. It is assumed that submersion of the reactor vessel into a body of water could hypothetically lead to a criticality based on the available design information.

5. No credit is taken for a HMIS given that one has not yet been defined, though such a system could reduce the risk from of certain kinds of accidents.

6. It is assumed that loss of passive heat transfer from the reactor to the environment could lead to pressurization of the reactor containment boundary but decay heat by itself would not lead to failure of a containment seal or device.

7. It is assumed that there is only enough combustible material inside the transport container in the form of cable and wire jacket and insulation to lead to a small fire.

8. It is assumed that no (or minimal) other flammable material, other than cable and wire jacket and insulation and minimal quantities of grease and oil, exist in the transport container. It is assumed that no significant quantity of plastic wrapping or flammable packing material is used in the transport container.

9. It is assumed there will be energized electrical components in the Reactor Module during transport associated with parameter monitoring, lighting, and ventilation.

10. It is assumed that the quantity of diesel fuel in the transport vehicle is about 300 gallons.

11. It is assumed for the hazard analysis that there is no prohibition about transporting during inclement weather (e.g., extreme, wind, rain, or temperature related scenarios were included). This assumption was reconsidered for the accident analysis.

12. It is assumed that extreme weather events that can contribute to the occurrence of highway accidents that damage the TNPP package are included in the large truck data, and therefore, do not need to be separately considered in separate scenarios.

13. It was assumed that there would be no specific control of passing or oncoming vehicles (i.e., collision with other vehicles was assumed possible) in development if the likelihood estimates.

14. Hazardous conditions qualitatively evaluated to be low risk were not carried forward for detailed accident analysis. Low risk scenarios were screened out because the likelihood was determined to be "Beyond Extremely Unlikely" or the consequences were determined not to significantly impact any of the TNPP radiological inventory contributors listed in Section 4.4.2.

4.4.3 Development and Identification of Accident Scenarios for Detailed Analysis

The hazardous conditions discussed in the previous section that are identified and assessed in Appendix 8.2 of this report were used as the basis to define the accident scenarios for the TNPP transportation PRA. Section 4.4.3.1 provides an identification and description of the TNPP transportation accident scenarios defined from the hazardous conditions, including discussion of parameters and factors that are important to accident sequence likelihood and consequence analysis. Section 4.4.3.2 discusses development of bounding representative accident scenarios for detailed accident analysis.

4.4.3.1 Identification and Description of the Full Set of Important Accident Scenarios

As described in Section 4.4.2, the hazardous conditions listed in Appendix 8.2 were specifically formulated to contain the information needed to define accident scenarios. Accordingly, the hazardous conditions identified in Appendix 8.2 are essentially, with some limited adjustment, the accident scenarios. The primary adjustment was to combine hazardous conditions that involve the same accident phenomena and produce the same kind of accident and accident consequences. The primary example of this is that weather related events (e.g., ice and snow events) that could cause highway accidents were considered encompassed by the highway accidents, because the highway accidents consider all root causes of the accident whether they are human, mechanical, or weather related (e.g., the likelihood of these accidents include the contribution from all root causes). Ultimately, it may be discovered that certain accidents should be further subdivided because variations of the accident may produce a different radiological consequence and have a different likelihood that are important to the conclusions of this report.

In a PRA, the typical way to organize accident scenarios is by initiating event categories, but other factors can also play an important role such as accident phenomena and resulting radiological dose pathways to a worker or the public. Table 4-5 presents a condensed summary of TNPP accident scenarios in which conditions estimated to be of low risk were screened out because the likelihood was determined to be “Beyond Extremely Unlikely” or the consequences were determined not to significantly impact any of the TNPP radiological inventory contributors listed in Section 4.4.2. The low-risk accident scenarios based on the criteria used above are judged not to have a meaningful impact on the estimated risk. Therefore, the screening of low-risk scenarios should not change the conclusions derived from the TNPP PRA results associated with the goals of performing the PRA discussed at the front part of Section 4.1. As described in Section 4.4.2, the hazardous conditions listed in Appendix 8.2 were specifically formulated to contain the information needed to define accident scenarios. Accordingly, the hazardous conditions identified in Appendix 8.2 and condensed in Table 4-5 are considered to be accident scenarios. However, a more complete description that includes discussion of factors important to an accident analysis is provided after Table 4-5.

Many of the accidents presented in Table 4-5 are highway accidents of the type typically considered in a transportation risk assessment involving a qualified package. These include high energy events that involve collisions with other types of vehicles; collision with a fixed object; a drop from an elevated surface like a bridge, embankment, or overpass; or a roll-over. These same highway accidents could also involve a fire that occurs as a result of the accident. This second set is distinguished from the first set that involves impact only because fire introduces an additional release mechanism beyond damage to the package caused by mechanical impact. Most of the accidents in these first two sets could involve all of the TNPP radiological inventory contributors listed in Section 4.4.2.

Another type of high-energy accidents are the fires-only accidents that do not involve mechanical impact and do not necessarily occur on the highway (e.g., they could occur at a gas station). These fires are considered separately from fires that occur as part of a collision because: (1) there is only one damage and release mechanism, and (2) a general fire or diesel pool fire involving the quantities of fuel carried on a transport vehicle likely cannot get hot enough to damage the TRISO fuel. Thermal testing of TRSIO fuel suggests that the fuel remains intact with very low radiological material release at temperatures up to 1600 °C to 1700 °C per INL/EXT-16-40784 (*A Summary of the Results from the DOE Advanced Gas Reactor (AGR) Fuel Development and Qualification Program* [Petti et al. 2017]). These are much higher temperatures than testing predicts for large-scale diesel pool fires (Tiwari 2019).

Another high energy accident to discuss along with these three sets is a tornado or high wind event that lifts or rolls the transport vehicle whether it is parked or moving. The physical phenomena (e.g., delta-pressure or airborne dispersion) that accompany this kind of external event can be an important consideration in the accident analysis (e.g., because it causes more damage or more release of radiological material dispersion than an event without wind).

Low energy accidents could also be important contributors to risk because they may occur at a higher frequency than a highway-related accident, even though they may result in lower levels of radiological consequence to workers or the public. Given the reactor and IHX will be separated into two different modules, a device will need to be temporarily installed at the points where these systems are separated to provide containment and are locations in the package that might be vulnerable to failure. One set of containment concerns loss of the reactor containment boundary containment when it is not pressurized, while another set concerns loss of reactor containment boundary containment when it is pressurized. A third set of containment concerns loss of containment from system elements which are not part of or contained by the reactor containment boundary.

The first set of low energy accidents concern a non-pressurized release from the reactor containment boundary for one of the following reasons: (1) random containment failure (e.g., failure of a seal, connection, or joint), (2) vibration and shock from over-the-road travel, (3) human error in packaging the system, (4) human error during TNPP disassembly leading to undetected latent failures in containment, and (5) extreme cold that fails containment.

The second set of low energy accidents concern a pressurized reactor containment boundary due to reactor decay heat causing pressure that is released due to the following reasons: (1) high ambient air temperature that in combination with the residual decay heat pressurizes the reactor containment boundary, and (2) impact on vents or the heat transfer pathway that decreases heat removal to the extent the reactor containment boundary pressurizes. These accidents are postulated because low level pressurization of the reactor containment boundary during transportation appears credible. For transportation, the gas that cools the reactor has been discharged (BWXT 2022[21]) and the Shield Tank that shields radiation, removes decay heat, and is a source of component cooling water has been drained of water to satisfy transportation weight limitations (BWXT 2022[22]). During shipment, the Shield Tank functions as an impact limiter to the reactor vessel (BWXT 2022[23]). A thermal analysis by the vendor shows that after active cooling was stopped, the decay heat resulted in reheating the fuel to a maximum temperature of 895 °K after about 4 days (or 9 days after reactor shutdown), which slowly

[21] BWXT Final Design Report, page 7-17.
[22] BWXT Final Design Report, page 3-9.
[23] BWXT Final Design Report, page 7-3.

decreased thereafter (BWXT 2022[24]). The decay heat generation about 12 days after reactor shutdown is estimated to be 20.2 kW (BWXT 2022[25]). Pacific Northwest National Laboratory (PNNL) estimates the decay heat generation will be about half this at 90 days after reactor shutdown. Given the reactor coolant system is depressurized when it is prepared for shipment (i.e., it is assumed to be at an ambient pressure of 0.1 MPa), PNNL estimates the decay heat generation would pressurize the reactor coolant system to a maximum of about 0.3 MPa during shipment based on the ideal gas law. Thus, some degree of pressurization is possible and provides a mechanism to discharge radioactive material from the reactor containment boundary.

The third set of low energy accidents concern loss of containment from other parts of the package besides the reactor containment boundary caused by: (1) pressurization due to radiolysis of hydrogenous material (e.g., Shield Tank not fully drained) and possible hydrogen accumulation and ignition, (2) pressurization caused by loss of ventilation or high ambient air temperatures, (3) containment failure caused by random or vibration caused failures, and (4) containment failure due to a hail storm that causes general severe vibration. For this third set, the radiological material available for release will primarily be only loose contamination.

There are three unique accident groups that produce consequences to the worker and public through different radiological dose pathways from the radiological release accident scenarios described above. The radiological risk to workers and the public from release of radiological material is primarily from inhalation of airborne radioactive material. The first set concerns direct exposure of a worker to radiation due to the loss of shielding as the result of a highway accident (e.g., collisions and drop from a bridge) described above. However, it is important to keep this set separate from the highway accidents described earlier to highlight that dose consequences to the worker can occur through a completely difference dose pathway if loss of shielding occurs during the accident. The second set concerns additional exposure of the worker to normal direct radiation caused by an increase in exposure time due to: (1) mechanical breakdown of the transport truck or trailer, (2) technical problems with the package that requires worker attention, and (c) adverse weather that delays transport. The third set concerns a criticality event caused by a highway accident (e.g., collision or drop into a body of water) that can result in both large direct radiation to the worker and release of radiological material to the environment due to: (1) addition of a moderator and potential change in core geometry, or (2) fast control rod withdrawal. Again, it is important to keep this set separate from the highway accidents described earlier to highlight that the accident phenomena of a criticality event and the resulting potential dose consequences are different than those associated with a release of radioactive material.

Below Table 4-5 presents a condensed summary of TNPP accident scenarios identified by the hazard analysis process after hazardous conditions estimated to be of low risk were screened out as described above based on the qualitative estimates of accident likelihood and consequence. As discussed in Section 4.4.2.1, the accident likelihood categories assigned by the hazard analysis team in the second to last column are based on the following definitions:

- Anticipated (Frequency ≥ 1E-02).
- Unlikely (1E-02 > Frequency ≥ 1E-04).
- Extremely Unlikely (1E-04 > Frequency ≥ 1E-06).
- Beyond Extremely Unlikely (1E-06 > Frequency).

[24] BWXT Final Design Report, Appendix III.45, Section 7.3.3.
[25] BWXT Final Design Report, Appendix III.45, page 17.

The accident consequence groups that were assigned by the hazard analysis team are presented in the last column of Table 4-5 and are defined using the MAR contributors described in Section 4.4.2.1. The consequence groups indicate which MAR contributors could possibly be released or become unshielded in a TNPP transportation accident and provide a qualitative sense of the potential magnitude of the radiological risk to a dose receptor.

Consequence Group A (Very High) indicates that all MAR contributors could potentially be partially released or unshielded as listed in the following:

- Nongaseous fission products contained within the TRISO fuel or heavy metal contamination within the compacts that are subsequently damaged in an accident,
- Fission gases contained within the TRISO fuel or heavy metal contamination within the compacts that are subsequently damaged in an accident,
- Fission Products from the TRSIO fuel that has diffused and is held up in the core structures,
- Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary,
- Contamination outside the reactor.

Consequence Group B (High) indicates that the following MAR contributors could potentially be partially released or unshielded:

- Fission Products from the TRSIO fuel that has diffused and is held up in the core structures,
- Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary,
- Contamination outside the reactor.

Consequence Group C (Moderate) indicates that the following MAR contributors could potentially be partially released or unshielded:

- Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary,
- Contamination outside the reactor.

Consequence Group D (Low) indicates that just the following could potentially be partially released or unshielded:

- Contamination outside the reactor.

Consequence Group E (non-criticality direct radiation exposure) pertains to potential direct exposure to: (1) existing TRISO fuel, (2) fission products held up in the compact and other core structures, (3) radiological material condensed or plated out in the reactor coolant boundary, and (4) activated reactor system components such as the control rods and motors, the Reactor Pressure Vessel, copper wires, and tungsten shielding.

Consequence Group F (criticality event) pertains to direct radiation exposure and radiological material released up to the MAR defined for Consequence Group A in a criticality event.

Detailed quantitative determination of accident likelihood and consequence is presented in Section 4.5 and Section 4.6, respectively, of this report.

Table 4-5. Identification of TNPP Accident Scenarios After Low-Risk Conditions are Screened Out (4 sheets total)

Accident Class	Accident Scenario	Qualitative Likelihood	Qualitative Consequence(4)
	Release of radiological material from damaged TNPP and/or package caused by:	**Per year range**	**Group**
1. Collision with vehicle(1)			
(a) Light	Collision with a light vehicle	Unlikely	B
(b) Heavy	Collision with a heavy vehicle	Unlikely	A
2. Collision (non-vehicle)(1)			
(a) Fixed object	Collision with fixed object (e.g., wall, road or bridge structures, embankment, overpass structure)	Unlikely	A
(b) Drop	Drop to a lower elevation surface (e.g., drop off bridge, embankment, overpass)	Unlikely	A
3. Non-collision road accident(2)			
(a) Roll-over	Roll-over with no collision with an object or vehicle	Anticipated	A
(b) Jackknife	Jackknife with no collision with an object or vehicle	Anticipated	B
4. Collision and subsequent fire(1)			
(a) Collision with vehicle or fixed object, or rollover and fire	Collision of transport vehicle with TNPP package with a vehicle in motion (e.g., truck, bus, car, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or a non-collision accident (e.g., rollover) and subsequent diesel fuel fire	Unlikely	A
(b) Collision with a tanker with flammable material and fire	Collision of the transport vehicle with TNPP package with a vehicle with a large amount of combustible or explosive material (e.g., a gasoline tanker, transport of flammable chemicals)	Extremely Unlikely	A
(c) Drop accident and fire	Drop from an elevated surface (e.g., bridge, embankment, overpass) and subsequent fire	Unlikely	A
5. Tornado or high wind event(3)			
(a) Mechanical impacts and delta-pressure	Impacts with moving and fixed objects, rollovers, drops, and delta-pressure impacts	Unlikely	A

Table 4-5. Identification of TNPP Accident Scenarios After Low-Risk Conditions are Screened Out (4 sheets total)

Accident Class	Accident Scenario Release of radiological material from damaged TNPP and/or package caused by:	Qualitative Likelihood Per year range	Qualitative Consequence[(4)] Group
6. Fire only event			
(a) General	General fire in the transport container (e.g., associated with module, overpack, or TNPP system components)	Anticipated	B
(b) Diesel fuel	Diesel fuel fire associated with transport vehicle	Anticipated	B
(c) Oil and grease	Oil or grease fire in a transport container (e.g., associated with module, overpack, or TNPP system components)	Anticipated	B
7. Loss of pressurized reactor containment boundary[(4)]			
(a) Random failure	Loss of pressurized reactor containment boundary caused random containment failure (e.g., seal, connection, or joint failure)	Unlikely	C
(b) Vibration and shock	Loss of pressurized reactor containment boundary caused by vibration and shock (e.g., from over the road travel, braking, wind, engine vibration)	Anticipated	B
(c) Human error preparing package	Loss of pressurized reactor containment boundary caused by procedural failures or human errors in preparing TNPP package for transport (e.g., sealing the reactor containment boundary)	Anticipated	C
(d) Human error in dismantlement	Loss of pressurized reactor containment boundary caused by procedural failures or human error during plant disassembly leads to undetected latent failures in containment elements (e.g., sealing the reactor containment boundary)	Anticipated	C
(e) Extreme cold	Loss of pressurized reactor containment boundary caused by extreme cold environmental temperature (e.g., beyond design limits of a containment feature during transport)	Anticipated	C
8. Loss of pressurized reactor containment boundary[4)]			
(a) Mechanical impact on vents or heat transfer pathway and containment failure	Loss of pressurized reactor containment boundary caused by residual heat buildup from loss of heat transfer due to minor impacts involving TNPP package (e.g., damage of vents or impacts on heat transfer pathway) that could occur from movement of the package or other objects in the transport container in combination with failure of reactor containment boundary caused by random failure, human error, vibration or extreme cold.	Anticipated	C

Table 4-5. Identification of TNPP Accident Scenarios After Low-Risk Conditions are Screened Out (4 sheets total)

Accident Class	Accident Scenario	Qualitative Likelihood	Qualitative Consequence[(4)]
	Release of radiological material from damaged TNPP and/or package caused by:	**Per year range**	**Group**
(b) High ambient air temperature and containment failure	Loss of pressurized reactor containment boundary caused by residual heat buildup and/or excessively high ambient air temperatures in combination of with failure of reactor containment boundary caused by random failure, human error, or vibration.	Anticipated	C
9. Loss of general package containment[(4)] (not in reactor containment boundary)			
(a) Radiolysis and possible hydrogen accumulation	Pressurization in TNPP package due to radiolysis of hydrogenous material (e.g., moisture, bound water, plastics, Shield Tank not fully drained) and possible hydrogen accumulation and ignition	Anticipated	D
(b) Loss of ventilation or high air temperatures	Pressurization in TNPP package due to loss of ventilation or high ambient air temperature during transport	Anticipated	D
(c) Random, vibration or human	Failure of TNPP package containment due to random or vibration caused failure (e.g., of a seal) or human error during transport	Anticipated	D
(d) Severe hailstorm	Failure of TNPP package containment from a severe hailstorm that causes significant vibration of the transport vehicle, container and TNPP package	Anticipated	D
10. Loss of shielding (non-criticality)			
(a) Drop of vehicle	Direct radiation exposure caused by loss of shielding (e.g., bolt-in shielding) due to drop of the transport vehicle with TNPP package off a bridge, embankment, or elevated surface (e.g., overpass)	Unlikely	E
(b) Vehicle collision	Direct radiation exposure caused by loss of shielding (e.g., bolt-in shielding) from damage due to collision of transport vehicle with TNPP package with a vehicle in motion (e.g., truck, bus, car, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (e.g., rollover) during transport	Unlikely	E
11. Increase in exposure time			
(a) Mechanical breakdown	Increase in worker exposure time due to breakdown of transport truck or trailer (e.g., engine, transmission or axile failure) that delays transport	Anticipated	E

Table 4-5. Identification of TNPP Accident Scenarios After Low-Risk Conditions are Screened Out (4 sheets total)

Accident Class	Accident Scenario	Qualitative Likelihood	Qualitative Consequence(4)
	Release of radiological material from damaged TNPP and/or package caused by:	Per year range	Group
(b) Technical problems with package	Increase in worker exposure time caused by breakdown or technical issues associated with TNPP, TNPP package, or overpack and shielding that requires resolution due to unanticipated random failures or operator errors that delays transport	Anticipated	E
(c) Adverse weather	Increase in worker exposure time to radiation from TNPP package caused by adverse weather that delays transport	Anticipated	E
12. Criticality			
(a) Addition of moderator and change in core geometry	Direct exposure and release of radiological material immersion of the transport vehicle with TNPP into a body of water (e.g., fall off a bridge or over an embankment into body of water including standing water from rain or flooding) and possible changes core geometry	Extremely Unlikely	F
(b) Control rod withdrawal	Direct exposure and release of radiological material fast control rod bank withdrawal at cold conditions during transport due to collision with a vehicle in motion (e.g., car, truck, bus, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (rollover).	Extremely Unlikely	F

(1) High wind, rain, snow, or ice can cause an accident and create special conditions that can impact radioactive material dispersion and transport.
(2) Rollovers and jackknifes were identified together as non-collision road accidents but are separated here into different accidents, because rollovers can result in hard impact with the road surface. Whereas jackknife events do not result in external impact collision, and therefore, lead to lesser consequences than postulated in the hazard analysis.
(3) Tornados and high wind can cause the accident and impact radioactive material dispersion and transport.
(4) In many cases, the presence of parameter monitoring system that measures such parameters as radiation, pressure, and temperature is a mitigation feature that could impact both the likelihood and consequence estimates of these sequences.

The following sections (Section 4.4.3.1.1 through Section 4.4.3.1.30) contain further descriptions of each TNPP transportation accident scenario presented in Table 4-5 including discussion of factors important to a detailed accident analysis of the accident. These discussions include general statements about how the likelihood and consequence of these accidents should be determined. The actual accident and likelihood determinations are made for bounding representative accidents and are discussed in Section 4.5 and Section 4.6, respectively.

4.4.3.1.1 Accident 1(a) – Collision with a Light Vehicle

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to collision of the transport vehicle with the TNPP package with a light vehicle in motion (e.g., car, light truck). Like the other highway accidents, there is a strong possibility of mechanical damage to the truck and transport container, and the possibility of damage to the TNPP package itself. Unlike the other highway accidents, not all available MAR contributors are assumed to be partially released. It is assumed that the TRISO fuel is not damaged, and therefore no gaseous or non-gaseous release from the TRISO fuel is assumed to occur.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set lower than values used for collision with a light vehicle. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on the accidents involving light vehicles. If the radiological dose consequences are the same for collision with a heavy vehicle then these accidents could be combined and the likelihood of the accident adjusted accordingly.

4.4.3.1.2 Accident 1(b) – Collision with a Heavy Vehicle

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to collision of the transport vehicle with a heavy vehicle in motion (e.g., truck or train). Like the other highway accidents, there is a strong possibility of mechanical damage to the transport container and the possibility of damage to the TNPP package itself. Accordingly, some fraction of all available MAR is assumed to be released, but unlike a collision with a light vehicle it is assumed that some fraction of the TRISO fuel is damaged, and therefore, a gaseous and non-gaseous release from the TRISO fuel is assumed to occur.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set higher than values used for collision with a heavy vehicle. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on the accidents involving heavy vehicles, particularly if the radiological dose consequences are significantly more than the dose consequences from collision with a light vehicle.

4.4.3.1.3 Accident 2(a) – Collision with a Fixed Object

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to collision of the transport vehicle with the TNPP package with a fixed object (e.g., wall, road or bridge structures, embankment, overpass structure). This type of collision could be of particular concern if the vendor uses a transport container that is higher than normal, as seems possible from the vendor design information. Like the other highway accidents, there is a strong possibility of mechanical damage to the transport container and the possibility of damage to the TNPP package itself. Accordingly, some fraction of all available MAR is assumed to be released including some fraction of the TRISO fuel.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set consistence with collision with the fixed object that can create the most damage to the TNPP package. If a worst-case collision with an object is rare and the consequences are high, then consideration should be given to analyzing collision with the object as a separate scenario if the scenario results in high radiological dose consequences. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on the accidents involving collision with fixed objects (e.g., wall, road or bridge structures, embankment, overpass structure). As stated above, if the scenario results in high radiological dose consequences because of collision with one type of object, then consideration should be given to analyzing collision with the object as a separate scenario with a lower accident frequency.

4.4.3.1.4 Accident 2(b) – Drop to a Lower Elevation Surface

This accident concerns release of radiological material from the TNPP package to the environment caused by drop of the transport vehicle with the TNPP package off a bridge, embankment, or elevated surface (e.g., overpass). Like the other highway accidents, there is a strong possibility of mechanical damage to the transport container and the possibility of damage to the TNPP package itself. Accordingly, some fraction of all available MAR is assumed to be released including some fraction of the TRISO fuel. Unlike other highway accidents, there is the possibility that the TNPP package is dropped into a body of water sufficient to flood the core. This version of the accident is identified in Table 4-5 as Accident 12(a) which is a criticality event and should be treated separately because it is less likely and because it involves calculating dose to the worker and public through a different radiological dose pathway.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set consistent with the drop that can create the most damage to the TNPP package. If such a worst-case accident is rare and the consequences are high, then consideration should be given to analyzing the accident as a separate scenario if the scenario results in high radiological dose consequences. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on the accidents involving drops from the roadway to a lower surface such as a drop off a bridge, embankment, or overpass. As stated above, if the scenario results in high radiological dose consequences and is limited to certain features, then consideration should be given to analyzing those accidents separately with a lower accident frequency.

4.4.3.1.5 Accident 3(a) – Rollover

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to a non-collision rollover accident involving the transport vehicle with the TNPP package. Though this accident technically does not result in a collision with another vehicle or object or a drop, it does involve hard impact with the ground which is likely to be the asphalt or concrete roadway and shoulder. Like the other highway accidents, there is a strong possibility of mechanical damage to the transport container and the possibility of damage to the TNPP package itself. Accordingly, some fraction of all available MAR is assumed to be released including some fraction of the TRISO fuel. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set consistent with the non-collision accident that can create the most damage to the TNPP package. If the worst accident is rare and the consequences are high, then consideration should be given to analyzing the accident as a separate scenario if the scenario results in high radiological dose consequences. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on non-collision rollover accidents stated above. If the scenario results in high radiological dose consequences and is limited to certain kinds of rollover accidents, then consideration should be given to analyzing those accidents separately with a lower accident frequency.

4.4.3.1.6 Accident 3(b) – Jackknife

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to a non-collision jackknife accident involving the transport vehicle with the TNPP package. This accident does not result in a collision with another vehicle or object or a drop, but could involve violent swinging of the transport container trailer and contents. This might lead to some impact internal to the container for objects (e.g., tools) that become unrestrained, but such impact is not expected to damage the package. Accordingly, there is some possibility of mechanical damage to the container contents and the possibility of damage to the TNPP package itself. Accordingly, some small fraction of all available MAR is assumed to be released, but not the TRISO fuel itself. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set consistent with this non-collision accident. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on non-collision jackknife accidents as stated above.

4.4.3.1.7 Accident 4(a) – Collision with a Vehicle, Fixed Object, or Rollover and Subsequent Fire

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to collision of the transport vehicle with the TNPP package with a vehicle in motion (e.g., truck, bus, car, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or a non-collision accident (e.g., rollover) and subsequent diesel fuel fire. Like most highway accidents, there is a strong possibility of mechanical damage to the transport container and the possibility of damage to the TNPP package itself. In addition to mechanical damage caused by impact, the fire can create thermal stress for material such as metal, so that activated material or material that contains held-up or plated-on radioactive material can be made airborne from sloughing of oxide from the oxidizing mass. Moreover, the fire can create convective current that causes the material to be airborne as described in DOE-HDBK-3010-94 (*Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* [DOE 2013]). Accordingly, some fraction of all available MAR is assumed to be released including some fraction of the TRISO fuel.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to consider both the mechanical impact and fire phenomena and should be set consistent with the situation for this scenario. Fire can cause additional damage to the package (e.g., fail containment seals not already failed) and provide an additional airborne release mechanism in addition to the impacts caused by mechanical impact. Therefore, the damage ratio, airborne release fraction, and respirable fraction used may be the sum of the ratio and fractions used for mechanical impact and fire. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on highway accidents that involve fire. As with all accident scenarios it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency.

4.4.3.1.8 Accident 4(b) – Collision with Tanker and Subsequent Fire

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to collision of the transport vehicle with the TNPP package with a vehicle with a large amount of combustible or explosive material (e.g., a gasoline tanker, transport of flammable chemicals) and subsequent and possible explosion. Like most highway accidents, there is a strong possibility of mechanical damage to the transport container and the possibility of damage to the TNPP package itself. In addition to mechanical damage caused by impact, fire can: (1) create thermal stress for material such as metal so that activated material or material that contains held-up or plated-on radioactive material can be made airborne from sloughing of oxide from the oxidizing mass, and (2) create a convective current that causes the material to be airborne. An additional consideration is the fact that the tanker could contain explosive material which may cause greater mechanical impact but perhaps less thermal damage. Accordingly, some fraction of all available MAR is assumed to be released including possibly some fraction of the TRISO fuel. The significance of this accident is that it likely produces the highest dose consequences of any highway accident because it is an impact with a heavy vehicle in combination with a large and long-lasting fire fueled by a considerable quantity of flammable material (e.g., a gasoline tanker).

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to consider both the mechanical impact and fire phenomena and should be set consistent with the worst-case situation. Fire can cause additional damage to the package (e.g., fail containment seals not already failed) and provide an additional airborne release mechanism in addition to the impacts caused by mechanical impact. Therefore, the damage ratio, airborne release fraction, and respirable fraction used may be the sum of the ratio and fractions used for mechanical impact and fire. An additional consideration is the fact that the tanker could contain explosive material which may cause greater mechanical impact, but perhaps less thermal damage. This variation will require a different damage ratio, airborne release fraction, and respirable fraction from fire events and could be addressed separately and be determined to be bounded by the consequences of a large fire. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses. For this accident, a maximum fire based on the combination of diesel fuel from the transport vehicle and the truck pulling the tanker in combination with the flammable material in the tanker needs to be assessed to determine the additional consequence of this accident compared to Accident 4(a).

The likelihood of this accident should be based on highway accidents rather than this specific worst-case accident. As discussed above, there may be a need to divide the likelihood of accidents involving a tanker carrying flammable or explosive material between those that involve explosions and those that involve a large fire.

4.4.3.1.9 Accident 4(c) – Drop to a Lower Elevation Surface and Subsequent Fire

This accident concerns release of radiological material from the TNPP package to the environment caused by drop of the transport vehicle with the TNPP package off a bridge, embankment, or elevated surface (e.g., overpass) and subsequent diesel fuel fire. Like most highway accidents, there is a strong possibility of mechanical damage to the transport container and the possibility of damage to the TNPP package itself. Again, in addition to mechanical damage caused by impact, fire can: (1) create thermal stress for material such as metal, so that activated material or material that contains held-up or plated-on radioactive material can be made airborne from sloughing of oxide from the oxidizing mass, and (2) create a convective current that causes the material to be airborne. Accordingly, some fraction of all available MAR is assumed to be released including possibly some fraction of the TRISO fuel. The significance of this accident compared to the other two accidents that involve impact and fire is that it may be more difficult and take longer to implement emergency response. For example, if the transport vehicle dropped off of a bridge into ravine that is difficult to access, then the fire may be free to burn longer.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to consider both the mechanical impact and fire phenomena. Fire can cause additional damage to the package (e.g., fail containment seals not already failed) and provide an additional airborne release mechanism in addition to the impacts caused by mechanical impact. Therefore, the damage ratio, airborne release fraction, and respirable fraction used may be the sum of the ratio and fractions used for mechanical impact and fire. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses. As stated above, if the transport vehicle drops into an area difficult to access such as a ravine, then it may be more difficult and take longer to implement emergency response. This in turn would likely impact duration assumed in the consequence analysis.

The likelihood of this accident should be based on a highway accident that involves a drop to a lower elevation surface (e.g., off a bridge or overpass) and subsequent fire. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. For example, it may be extremely unlikely that the accident involves a drop into an inaccessible area.

4.4.3.1.10 Accident 5(a) – Tornado or High Wind Event

This accident concerns release and dispersion of radiological material from the TNPP package to the environment caused by damage to the TNPP and package from a tornado or high wind event during transport leading to severe impacts (e.g., impacts with moving and fixed objects, rollovers, and drops) and delta pressure impacts. Like most highway accidents, there is a strong possibility of mechanical damage to the container and the possibility of damage to the TNPP package itself. Accordingly, some fraction of all available MAR is assumed to be released including some fraction of the TRISO fuel. In addition to mechanical damage caused by impact, the delta pressure caused by a tornado could cause pressurized release from the reactor core or reactor containment boundary.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to consider both the mechanical impact and potential delta-pressure phenomena and should be set consistent with the situation for this scenario. Dispersion of radiological material by the wind is actually a factor that could dilute or decrease radiological dose to the worker and public. However, this dispersion would be difficult to model, and in any event, should not be credited as a positive factor given that it would be hard to predict. It would be safely conservative to assume damage to the TNPP package but not credit dispersion.

The likelihood of this accident should be based on the frequency of a tornado or high wind event along the route. Given that the frequency is likely variable along the route, the highest frequency along the route could be used to be conservative or the route could be parsed into sections and multiple accidents postulated. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences. So, in this case, in addition to possibly parsing tornado events by sections of the route, the wind events could be separated from the tornado events, particularly if the tornado events result in significantly higher radiological consequences.

4.4.3.1.11 Accident 6(a) – General Fire Only Event

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to general fire in the transport container (e.g., associated with the module, the overpack, or system components). Fire scenarios are differentiated from impact scenarios that result in fire, in that it is a “fire only” event and could occur on the highway, while parked, or during refueling. In this scenario, this fire originates in the transport container in or around the TNPP package. Accordingly, some fraction of the available MAR is assumed to be released except the TRISO fuel itself. Thermal testing of TRSIO fuel suggests that the fuel remains intact with very low radiological material release at temperatures up to 1600 °C to 1700 °C per INL/EXT-16-40784 (Petti et al. 2017). These are much higher temperatures than testing predicts for non-fuel general fire.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to consider the fire impacts consistence with the situation for this scenario. Fire could damage the packaging and containment features such as the seals on the Primary Cooling system piping and, given the fire originates in or directly around the TNPP package, that damage could be greater than a fire that originates from outside the transport container (e.g., a diesel fuel fire). If the fire is big enough, it could: (1) create thermal stress for material such as metal, so that activated material or material that contains held-up or plated-on radioactive material can be made airborne from sloughing of oxide from the oxidizing mass, and (2) create a convective current that causes the material to be airborne as discussed in Section 4.4.3.1.7. However, the fire is expected to be much smaller

The likelihood of this accident should not be based on truck fires given the unusual load, but rather a general fire for a comparable situation. The likelihood might be bounded by the fire ignition frequency of the area of a nuclear power plant without large operating pumps or heavy switchgear. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency.

4.4.3.1.12 Accident 6(b) – Diesel Fuel Fire Only Event

This accident concerns release of radiological material from the TNPP package to the environment caused by damage due to ignition of a spill or leaked diesel fuel from the transport vehicle that propagates to the package. As stated above, fire scenarios are differentiated from impact scenarios that result in fire, in that it is a "fire only" event and could occur on the highway, while parked, or during refueling. In this scenario, this fire originates outside the transport vehicle. Accordingly, some fraction of the available MAR is assumed to be released except the TRISO fuel itself. As stated above, thermal testing of TRSIO fuel suggests that the fuel remains intact with very low radiological material release at temperatures up to 1600 °C to 1700 °C (Petti et al. 2017). These are much higher temperatures than testing predicts for large-scale diesel pool fires according to testing by such sources as *Journal of Physics* (Tiwari 2019).

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to consider the fire impacts consistence with the situation for this scenario. Fire could damage the packaging and containment features such as the seals on the Primary Cooling system piping. Given that the fire originates from outside the transport container, the damage inside the transport containers may be less than for fires that originate inside the transport container including the TNPP package itself. Also, as explained in Section 4.4.3.1.7, fire can: (1) create thermal stress for material such as metal, so that activated material or material that contains held-up or plated-on radioactive material can be made airborne from sloughing of oxide from the oxidizing mass, and (2) create a convective current that causes the material to be airborne.

The likelihood of this accident should be based on truck diesel fuel fires. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. In this case, it may be difficult to separate the fires that occur from impact from fires that occur without impact. Therefore, in this case it might be practical to conservatively assume that the risk of this accident is bounded by the consequences associated with an accident that involves impact but the likelihood of any truck fire is applied.

4.4.3.1.13 Accident 6(c) – Oil and Grease Fire

This accident concerns release of radiological material from the TNPP package caused by ignition of grease/oil in a transport container (e.g., associated with the module, the overpack, or system components). This accident could be considered a subset of Accident 6(a) which is a general fire that originates inside the transport container including the package as described in Section 4.4.3.1.11. However, if it is determined that the TNPP itself or the package has more than minimal quantities of oil (or grease) then the fire might produce more damage than a general fire in which combustibles are somewhat limited. As stated earlier, fire scenarios are differentiated from impact scenarios that result in fire, in that it is a "fire only" event and could occur on the highway, while parked, or during refueling. In this scenario, this fire originates in the transport container in or around the TNPP package. Accordingly, some fraction of the available MAR is assumed to be released except the TRISO fuel itself. Thermal testing of TRSIO fuel suggests that the fuel remains intact with very low radiological material release at temperatures up to 1600 °C to 1700 °C per INL/EXT-16-40784 (Petti et al. 2017). These are much higher temperatures than testing predicts for this type of fire.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to consider the fire impacts consistence with the situation for this scenario. Fire could damage the packaging and containment features such as the seals on the Primary Cooling system piping and, given the fire originates in or directly around the TNPP package, that damage could be greater than a fire that originates from outside the transport container (e.g., a diesel fuel fire). Moreover, if it is determined that the TNPP itself or package has more than minimal quantities of oil (or grease) then the fire might produce even greater damage.

The likelihood of this accident should not be based on truck fires given the unique load, but rather a general fire for a comparable situation. The likelihood might be bounded by the fire ignition frequency of the area of a nuclear power plant with oil and grease. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. However, in this case, it could be beneficial to combine this scenario with other loss of non-pressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

4.4.3.1.14 Accident 7(a) – Loss of Reactor Containment Boundary (Non-Pressurized) Caused by Random Failure

This accident concerns release of radiological material to the environment from the reactor containment boundary caused by containment failure (e.g., seal, connection, or joint failure). Given the reactor and heat exchanger will likely be separated into two modules, a containment feature is needed at the points where these systems are separated and could be vulnerable to failure. Some fraction of the available MAR is assumed to be released except the TRISO fuel itself and radioactive material that has diffused and is held up in the compact and other core structures. This primarily consists of radioactive material that has plated-out in the reactor containment boundary.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect that containment has been breached and a non-pressurized condition exists, so there is limited motive force to discharge radiological material from the package. It should reflect the fact that road vibration and shock may have loosened radioactive material from the surfaces inside the reactor containment boundary.

The likelihood of this accident should be based on the likelihood of random containment failures (e.g., seal, connection, or joint failure). The failure probability can be better estimated once the details of the containment features are fully known. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. However, in this case, it could be beneficial to combine this scenario with other loss of non-pressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

4.4.3.1.15 Accident 7(b) – Loss of Reactor Containment Boundary (Non-Pressurized) Caused by Vibration or Shock

This accident concerns release of radiological material from the TNPP package to the environment caused by failure of the reactor containment boundary due to vibration and/or shock during transport (e.g., caused by over the road travel, braking, wind, engine vibration) that loosens, degrades, or fails component material, seals, and connections. As stated above, the reactor and heat exchanger will likely be separated into two modules; therefore, a containment feature is needed at the points where these systems are separated which could be vulnerable to failure. Some fraction of the available MAR is assumed to be released except the TRISO fuel itself and radioactive material that has diffused and is held up in the compact and other core structures. This primarily consists of radioactive material that has plated-out in the reactor containment boundary. Vibration and shock from road travel could also contribute to loosening radioactive material plated-out in the reactor containment boundary and surface material diffused onto the compact and other core structures. In this scenario, the reactor containment boundary is assumed not to be pressurized (e.g., there may not be enough decay heat to pressurize sealed systems depending on the heat load and passive cooling rate).

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect that a non-pressurized containment has been breached but that vibration and shock may have loosened radioactive material inside the reactor containment boundary such as the core structure.

The likelihood of this accident should be based on the likelihood of containment failures (e.g., component material, seal, connections, or joint failure) that occur from vibration and shock. Applicable failure rates may not be easy to find or develop, so estimation could be based on the high-end of the failure probability distribution for random failures. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. In other cases, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set. However, in this case, it could be beneficial to combine this scenario with other loss of non-pressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

4.4.3.1.16 Accident 7(c) – Loss of Reactor Containment Boundary (Non-Pressurized) Caused by Human Error Preparing the Package

This accident concerns release of radiological material from the TNPP package caused by procedural failures or human errors in preparing the TNPP package for transport (e.g., sealing the reactor containment boundary). Some fraction of the available MAR is assumed to be released except the TRISO fuel itself and radioactive material that has diffused and is held up in the compact and other core structures. This primarily consists of radioactive material that has plated-out in the reactor containment boundary. In this scenario, the reactor containment boundary is assumed not to be pressurized (e.g., failure to achieve a pressure tight boundary).

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect that a non-pressurized containment has been breached, so there is limited motive force to discharge radiological material from the package. It should reflect the fact that road vibration and shock may have loosened radioactive material from the surfaces inside the reactor containment boundary.

The likelihood of this accident should be based on human error associated with preparing the TNPP package for transport (e.g., sealing the Primary Cooling system, IHX Module, and any separated Primary Cooling piping). Estimates might be made using guidance for nuclear power plant operator actions from a Human Reliability Analysis (HRA) methodology such as NUREG/CR-6883 (*The SPAR-H Human Reliability Analysis Method* [Gertman et al. 2005]). This guidance states, for example, that the base probability of an execution error that does not require diagnosis and for which all Performance Shaping Factors (PSFs) are nominal is 1E-03. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. In other cases, it may be advantageous to combine certain accidents using the highest likelihood and consequence of the scenarios in the set. However, in this case, it could be beneficial to combine this scenario with other loss of non-pressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

4.4.3.1.17 Accident 7(d) – Loss of Reactor Containment Boundary (Non-Pressurized) Caused by Human Error in Dismantlement

This accident concerns release of radiological material from the TNPP package caused by procedural failures and human error during plant disassembly that leads to undetected latent failures in containment elements (e.g., sealing the Primary Cooling system, IHX Module, and any separated Primary Cooling piping). Some fraction of the available MAR is assumed to be released except the TRISO fuel itself and radioactive material that has diffused and is held up in the compact and other core structures. This primarily consists of radioactive material that has plated-out in the reactor containment boundary. In this scenario, the reactor containment boundary is assumed not to be pressurized (e.g., there may not be enough decay heat to pressurize sealed systems depending on the heat load and passive cooling rate).

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect that a non-pressurized containment has been breached, so there is limited motive force to discharge radiological material from the package. It should reflect the fact that road vibration and shock may have loosened radioactive material from the surfaces inside the reactor containment boundary.

The likelihood of this accident should be based on procedural failures and human error during plant disassembly that leads to undetected latent failures in containment elements (e.g., sealing the Primary Cooling system, IHX Module, and any separated Primary Cooling piping). As described above in more detail, estimates might be made using guidance for nuclear power plant operator actions from a HRA methodology (Gertman et al. 2005). As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. In other cases, it may be advantageous to combine certain accidents using the highest likelihood and consequence of the scenarios in the set. However, in this case, it could be beneficial to combine this scenario with other loss of non-pressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

4.4.3.1.18 Accident 7(e) – Loss of Reactor Containment Boundary (Non-Pressurized) Caused by Extreme Cold

This accident concerns release of radiological material from the TNPP package to the environment from the failure of the TNPP packaging seal and reactor containment boundary containment due to extreme cold environmental temperature (e.g., beyond design limits of the containment feature during transport). As stated above, the reactor and heat exchanger will likely be separated into two modules, and so a containment feature is needed at the points where these systems are separated which could be vulnerable to failure. Some fraction of the available MAR is assumed to be released except the TRISO fuel itself and radioactive material that has diffused and is held up in the compact and other core structures. This primarily consists of radioactive material that has plated-out in the reactor containment boundary. In this scenario, the reactor containment boundary is assumed not to be pressurized (e.g., there may not be enough decay heat to pressurize sealed systems depending on the heat load and passive cooling rate).

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect that a non-pressurized containment has been breached, so there is limited motive force to discharge radiological material from the package. It should reflect the fact that road vibration and shock may have loosened radioactive material from the surfaces inside the reactor containment boundary.

The likelihood of this accident should be based on the likelihood of containment failures (e.g., component material, seals, joints, and connections) that occur from extreme cold. Applicable failure rates may not be easy to find or develop, so estimation could be based on the high-end of the failure probability distribution for random failures. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. In other cases, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set. So, for example, it could be beneficial to combine this scenario with other loss of non-pressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

4.4.3.1.19 Accident 8(a) – Loss of Pressurized Reactor Containment Boundary Caused by Mechanical Impact on Heat Transfer System

This accident concerns release of radiological material to the environment from a pressurized reactor containment boundary caused by residual heat buildup due to loss of heat transfer from mechanical impacts involving the TNPP package (e.g., damage of vents or impacts on heat transfer pathway) in combination with failure of reactor containment boundary caused by random failure, human error, vibration, or extreme cold. (The possibility of reactor containment boundary pressurization is discussed in Section 4.4.3.1.) Some fraction of the available MAR is assumed to be released except the TRISO fuel itself and radioactive material that has diffused and is held up in the compact and other core structures. This primarily consists of radioactive material that has plated-out in the reactor containment boundary.

The pressurized condition will provide a mechanism for discharge of some portion of the radioactive material.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect containment breached and a pressurized condition exists that provides a mechanism to discharge radioactive material consistent with the situation for this scenario.

The likelihood of this accident should be based on the likelihood of damage of vents or impacts on the heat transfer pathway that could occur from the package or other objects moving within the transfer container in combination with reactor containment boundary failures like those discussed in Sections 4.4.3.1.14 through 4.4.3.1.18. Prevention and mitigation systems might include constraints and a parameter monitoring system, and therefore, component or system failure rates can be used to estimate their failure probabilities. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. However, in this case, it could be beneficial to combine this scenario with other loss of pressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

4.4.3.1.20 Accident 8(b) – Loss of Reactor Containment Boundary (Pressurized) Caused by High Ambient Temperature

This accident concerns release of radiological material to the environment from the pressurized reactor containment boundary caused from residual heat buildup and excessively high ambient air temperatures in combination with failure of the reactor containment boundary caused by random failure, human error, vibration, or extreme cold. (The possibility of reactor containment boundary pressurization is discussed in Section 4.4.3.1.) Some fraction of the available MAR is assumed to be released except the TRISO fuel itself and radioactive material that has diffused and is held up in the compact and other core structures. This primarily consists of radioactive material that has plated-out in the reactor containment boundary. The pressurized condition will provide a mechanism for discharge of some portion of the radioactive material.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect containment breached and a pressurized condition exists that provides a mechanism to discharge radioactive material consistent with the situation for this scenario.

Given that some level of pressurization will exist from decay heat, the likelihood of this accident should be based on the likelihood of very high ambient air temperatures in combination with reactor containment boundary failures like those discussed in Sections 4.4.3.1.14 through 4.4.3.1.18. Prevention and mitigation systems might include vents and a HMIS, and therefore, component or system failure rates can be used to estimate their failure probabilities. As with all accident scenarios, it may be necessary to separate a high-consequence accident from those that produce lesser consequences particularly if the high-consequence accident is a small contribution to the accident frequency. However, in this case, it could be beneficial to combine this scenario with Accidents 6(a) and 6(c), particularly if the likelihood and radiological consequences of the accidents are about the same.

4.4.3.1.21 Accident 9(a) – Loss of General Package Containment from Radiolysis

This accident concerns release of radiological material (e.g., activation products or contamination) in escaped air or gas from the TNPP package to the environment caused by pressurization due to radiolysis of hydrogenous material (e.g., moisture, bound water, plastics, Shield Tank not fully drained) including possible hydrogen accumulation and ignition. This primarily concerns contamination outside the TNPP itself but inside the TNPP package. The pressurized condition will provide a mechanism for discharge of the radioactive material.

Damage ratios, airborne release fractions, and respirable fractions may not apply to this accident scenario because it primarily concerns discharge of radioactive contamination from inside the package to which the worker can be exposed. Estimates of radiological contamination can be made from radiological activities during shutdown at nuclear power plants.

The likelihood of this accident scenario is based on whether radiolysis of hydrogenous material (e.g., moisture, bound water, plastics, Shield Tank not fully drained) can occur and whether it can contribute to pressurized discharge of radioactive material. As with all scenarios, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set. So, for example, loss of general package containment from accidents might be combined.

4.4.3.1.22 Accident 9(b) – Loss of General Package Containment from High Temperature

This accident concerns release of radiological material (e.g., contamination) in escaped air from the TNPP package to the environment caused by pressurization in the TNPP package due to loss of ventilation or high ambient air temperature during transport. This primarily concerns contamination outside the TNPP itself but inside the TNPP package. The pressurized condition will provide a mechanism for discharge of the radioactive material.

Damage ratios, airborne release fractions, and respirable fractions may not apply to this accident scenario because it primarily concerns discharge of radioactive contamination from inside the package but outside the containment systems to which the worker can be exposed. Estimates of radiological contamination can be made from radiological activities during shutdown at nuclear power plants.

The likelihood of this accident scenario is based on the failure probability of adequate cooling or ventilation which can be based on the dominate applicable component failure rates. As with all scenarios, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set. So, for example, loss of general package containment from accidents might be combined.

4.4.3.1.23 Accident 9(c) – Loss of General Package Containment from Random Failures

This accident concerns release of radiological material (e.g., contamination) in escaped air from the TNPP package to the environment caused by failure of containment due to random or vibration caused failure (e.g., of a seal) or human error during transport. This primarily concerns contamination outside the TNPP itself but inside the TNPP package. There is no pressurized condition to foster discharge of the radioactive material.

Damage ratios, airborne release fractions, and respirable fractions may not apply to this accident scenario because it primarily concerns discharge of radioactive contamination from inside the package but outside containment systems to which the worker can be exposed. Estimates of radiological contamination can be made from radiological activities during shutdown at nuclear power plants.

The likelihood of this accident scenario is based on the estimated random failure probability of the package containment. As with all scenarios, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set. So, for example, loss of general package containment from accidents might be combined.

4.4.3.1.24 Accident 9(d) – Loss of General Package Containment from Hailstorm

This accident concerns release of radiological material from the TNPP to the environment caused by failure of the package from a severe hailstorm that causes significant vibration of the transport vehicle, container, and TNPP package. This primarily concerns contamination outside the TNPP itself but inside the TNPP package. There is no pressurized condition to foster discharge of the radioactive material.

Damage ratios, airborne release fractions, and respirable fractions may not apply to this accident scenario because it primarily concerns discharge of radioactive contamination from inside the package but outside containment systems to which the worker can be exposed. Estimates of radiological contamination can be made from radiological activities during shutdown at nuclear power plants.

The likelihood of this accident scenario is based on the estimated likelihood of a hailstorm during transport which, if it happens, may be hard to evade. As with all scenarios, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set. So, for example, loss of general package containment from accidents might be combined.

4.4.3.1.25 Accident 10(a) – Loss of Shielding from Drop of Vehicle to a Lower Surface

This accident concerns exposure of the worker to direct radiation from loss of shielding (e.g., bolt-on shielding and cable mesh) due to drop of the transport vehicle with the TNPP package off a bridge, embankment, or elevated surface (e.g., overpass). There is a potential for direct exposure to the worker from existing TRISO fuel, fission products held up in the compact and other core structures and the reactor containment boundary and activated reactor system components such as the control rods and motors, RPV, copper wires, and tungsten shielding.

The damage ratios, airborne release fractions, and respirable fractions used in the consequence analysis for release of radiological material into the environment are not germane to the analysis of direct radiation exposure. However, this radiological dose pathway (direct radiation from loss of shielding) should be considered in combination with the radiological material release pathways determined for Accident 2(b) which is a transfer container drop event. Regarding Accident 4(c), which is transfer container drop event and sequent fire, the fire might initially prevent workers from getting close to the TNPP package to receive direct radiation exposure, but exposure might occur after the fire is extinguished. Therefore, Accidents 10(a) and 4(c) should also be considered together so that all applicable dose pathways are considered.

The likelihood of this accident should be based on the accidents involving drops from the roadway to a lower surface such as a drop off a bridge, embankment, or overpass encompassing those that subsequently lead to fire. As stated above, if the scenario results in high radiological dose consequences and is limited to certain features, then consideration should be given to analyzing those accidents separately with a lower accident frequency.

4.4.3.1.26 Accident 10(b) – Loss of Shielding from Impact Caused by Vehicle Collision

This accident concerns exposure of the worker to direct radiation from loss of shielding (e.g., bolt-in shielding and cable mesh) from damage due to collision of the transport vehicle with the TNPP package with a vehicle in motion (e.g., truck, bus, car, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (e.g., rollover) during transport. There is a potential for direct exposure to the worker from existing TRISO fuel, fission products held up in the compact and other core structures and the reactor containment boundary and activated reactor system components such as the control rods and motors, RPV, copper wires, and tungsten shielding.

The damage ratios, airborne release fractions, and respirable fractions used in the consequence analysis for release of radiological material into the environment are not germane to the analysis of direct radiation exposure. However, this radiological dose pathway (direct radiation from loss of shielding) should be considered in combination with the radiological material release pathways determined for Accidents 1(a), 1(b), 2(a), 2(b), 3(a) and 3(b) which are road accidents. Regarding Accidents 4(a), 4(b), and 4(c) which are collisions and sequent fire, the fire might initially prevent workers from getting close to the TNPP package to receive direct radiation exposure, but exposure might occur after the fire is extinguished.

The likelihood of this accident should be based on the accidents involving collision of the transport vehicle with TNPP package with a vehicle in motion (e.g., truck, bus, car, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (e.g., rollover) during transport. As stated above, if the scenario results in high radiological dose consequences and is limited to certain features, then consideration should be given to analyzing those accidents separately with a lower accident frequency.

4.4.3.1.27 Accident 11(a) – Increase in Exposure Time to Normal Radiation Caused by Mechanical Breakdown

This accident concerns increased exposure of the worker to normal levels of radiation at the transport container caused by breakdown of the transport truck or trailer (e.g., engine, transmission, or axle failure) that delays transport. Although, applicable occupational controls for radiation exposure will be applied, emergency situations that are critical to resolve might lead to an unintentional undesired increase in dose to the radiation levels that normally exist near the transport containers during transport.

The damage ratios, airborne release fractions, and respirable fractions used in the consequence analysis for release of radiological material into the environment are not germane to the analysis of direct radiation exposure for this situation.

The likelihood of this accident could be based on the statistics or truck breakdowns on the highway. As stated above, if the scenario results in high radiological dose consequences and is limited to certain features, then consideration should be given to analyzing those accidents separately with a lower accident frequency. Conversely, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set.

4.4.3.1.28 Accident 11(b) – Increase in Exposure Time to Normal Radiation Caused by Technical Problems with Package

This accident concerns increased exposure of the worker to normal levels of radiation at the transport container caused by breakdown or technical issues associated with the TNPP, the TNPP package, or the overpack and shielding that require resolution due to unanticipated random failures or operator errors that delay transport. Although applicable occupational controls for radiation exposure will be applied, emergency situations that are critical to resolve might lead to an unintentional undesired increase in dose to the radiation levels that normally exist near the transport containers during transport.

The damage ratios, airborne release fractions, and respirable fractions used in the consequence analysis for release of radiological material into the environment are not germane to the analysis of direct radiation exposure for this situation.

The likelihood of this accident could be based on the estimated frequency of breakdowns, random package containment errors, or human errors (or the highest frequency of the various contributors). Random failure might be estimated using the same approach as used for Accident 7(a) and human error might be estimated using the same approach as used for Accident 7(c) or 7(b). As stated above, if the scenario results in high radiological dose consequences and is limited to certain features, then consideration should be given to analyzing those accidents separately with a lower accident frequency. Conversely, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set.

4.4.3.1.29 Accident 11(c) – Increase in Exposure Time to Normal Radiation Caused by Adverse Weather Delays

This accident concerns increased exposure of the worker to normal levels of radiation at the transport container caused by adverse weather that delays transport. Although, applicable occupational controls for radiation exposure will be applied, emergency situations that are critical to resolve might lead to an unintentional undesired increase in dose to the radiation levels that normally exist near the transport containers during transport.

The damage ratios, airborne release fractions, and respirable fractions used in the consequence analysis for release of radiological material into the environment are not germane to the analysis of direct radiation exposure for this situation.

The likelihood of this accident could be based on the estimated frequency of severe weather along the route significant enough to delay transport. As stated above, if the scenario results in high radiological dose consequences and is limited to certain features, then consideration should be given to analyzing those accidents separately with a lower accident frequency. Conversely, it may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set.

4.4.3.1.30 Accident 12(a) – Criticality Accident Caused by Addition of Moderator and Change in Core Geometry

This accident concerns exposure of the worker to direct radiation from a criticality and release of radioactive material caused by a criticality event due to the immersion of the transport vehicle with the TNPP into a body of water (e.g., fall off a bridge or over an embankment into a body of water including standing water from rain or flooding) and possible changes to core geometry. There is a potential for the direct exposure of workers to high levels of radiation from a criticality event involving the existing TRISO fuel. The level of radiation during a criticality event could be significantly higher than the transportation shielding is designed to mitigate. Additionally, the shielding could be become significantly degraded from impacts that occur during the accident. In addition to direct radiation exposure, some fraction of all available MAR might be released including some fraction of the TRISO fuel.

For the radiological release portion of the consequence analysis, the damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set consistent with the drop that can create the most damage to the TNPP package. For the direct radiation portion of the consequence (non-release), the loss of shielding should be consistent with the drop accident that can create the most damage to the TNPP package. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on the accidents involving drops from the roadway to a lower surface such as a drop off a bridge, embankment, or overpass into a body of water deep enough to submerge the TNPP (the barrel of the reactor body will be on the order of 5 ft in diameter and 7 ft long). Therefore, a drop event would have to occur near a body of water like a river or lake though even a borrow pit full of water could be enough to submerge the reactor. The accident frequency, therefore, is the frequency of the drop event times the conditional probability that it ends up in a body water deep enough to submerge.

4.4.3.1.31 Accident 12(b) – Criticality Accident Caused by Control Rod Withdrawal

This accident concerns exposure of the worker to direct radiation from a criticality and release of radioactive material caused by a criticality event due to fast control rod bank withdrawal at cold conditions during transport due to collision with a vehicle in motion (e.g., car, truck, bus, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (rollover) during transport which causes loss of or degraded shielding. There is a potential for the direct exposure of workers to high levels of radiation from a criticality event involving the existing TRISO fuel. The level of radiation during a criticality event will be significantly higher than the transportation shielding is designed to mitigate. Additionally, the shielding could be become significantly degraded from impacts that occur during the accident. In addition to direct radiation exposure, some fraction of all available MAR might be released including some fraction of the TRISO fuel.

For the radiological release portion of the consequence analysis, the damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set consistent with the highway accident that can create the most damage to the TNPP package. For the direct radiation portion of the consequence (non-release), the loss of shielding should be consistent with the highway accident that can create the most damage to the TNPP package. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) should be considered for its impact on any of the radiological dose pathway consequence analyses.

The likelihood of this accident should be based on the highway accidents listed above that can potentially cause the control rod withdrawal event. The conditional probability that a highway accident occurs that causes a control rod withdrawal event will likely be difficult to estimate. Given that a criticality event could produce significant direct radiation exposure to the worker, some bounding estimate of the likelihood is needed.

4.4.3.2 Development of Bounding Representative Accident Scenarios

As described at the end of Section 4.4.1, there is sufficient rational and a practical advantage to only performing detailed accident analysis on bounding representative accident scenarios. The practical advantage is that reducing the number of accidents that must be evaluated in detail not only reduces the number of baseline calculations but also the number of corresponding sensitivity studies that must be performed. Sensitivity studies are an important way to investigate the impact of sources of modeling uncertainty on TNPP transportation risk and will be used in this report to determine the level of impact that certain PRA assumptions have on the risk results.

The thirty-one accident scenarios discussed in detail in Section 4.4.3.1 and identified in Table 4-5 have been organized into groups of accidents that have similar characteristics for the purpose of defining bounding representative accidents. Accordingly, characteristics are described for the group based on the physical accident phenomena, likelihood, and potential radiological dose consequences of the accidents within a given group. This description provides the definition of a bounding representative accident. The definition is intended to encompass all accidents in the group that constitutes the bounding representative accident. For example, the consequences of the bounding representative accident should be at least as great as consequences from any of the individual accidents in the group. The likelihood of the bounding representative accident is the sum of frequencies of all the accidents defined to be part of the bounding representative accident. When a bounding representative accident was too conservative, then the group was subdivided to remove some conservatism.

Based on accident phenomena, the TNPP transportation accidents can be organized into the following classes for discussion: (1) accidents that involve fire only, (2) road accidents that involve high energy impact that could cause release of radiological material or loss of shielding, (3) road accidents that involve high energy impact and fire, (4) release of radioactive material from a pressurized reactor containment boundary, (5) release of radioactive material from a non-pressurized reactor containment boundary, (6) release of radioactive material from a non-reactor containment boundary element of the package, (7) unplanned increase in exposure time to radiation, and (8) a criticality event. The following sections describe the development of bounding representative accidents for these eight classes of accidents. The final discussion in Section 4.4.3.2.9 lists and defines the resulting bounding representative accidents.

Accident frequency development for these accidents is discussed in Section 4.5 and summarized for each of the bounding representative accidents in Table 4-20. Radiological dose consequence for these accidents is presented in Section 4.6 and summarized for the bounding representative accidents in Table 4-26.

4.4.3.2.1 Fire Only Accidents

This section describes development of bounding representative accidents for the fire-only accidents. There are six separate fire accidents in which three pertain to impacts from road accidents that involve subsequent fire. These three accidents are not addressed in this section because they involve impact and fire. The fire accidents involving fire-only are of different sizes and origins, and could happen anytime during transport including during refueling (i.e., Accidents 6(a), 6(b), and 6(c)). This set of events includes fires that originate internal to the transport container (i.e., Accidents 6(a) and 6(c)). This includes a general fire such as a cable fire ignited by an electrical fault and an oil or grease fire which is expected to be present in limited quantity in the transportation container. Though these fires could potentially impact the package directly because they are internal to the transport container, they are apt to be small given the lack of flammable material in the container. The other fire (i.e., Accident 6(b)) is a diesel fuel fire and that originates outside of the transport container. This fire is likely to be much bigger but must propagate into the transport container to cause damage to the TNPP transportation package. If the internal and external fires-only events are grouped together to form a bounding representative accident, the consequences of the bounding case could be overly conservative. A bounding case accident scenario that encompasses all three fires could assume the worst-case conditions of the three cases (i.e., that the fire source is diesel fuel and that it originates inside the transport container). Additionally, the likelihood of spurious fire in the transport container not related to a diesel fuel fire or not ignited by engine heat is very unlikely compared to a diesel fuel fire that originates outside the transport container which is more likely.

Therefore, the three fire-only accident scenarios are divided into two cases: Bounding Representative Accident (BRA) 1 and BRA 2. The first, BRA 1, is a fire that originates inside the transport container. It is a general fire that originates from such sources as an electrical cable fault, propagates to the package, and ignites combustible material associated with the package. It includes an oil or grease fire that is ignited from a hot surface or electrical fault. All MAR (i.e., the TRISO fuel itself, radiological material diffused into the core during operation, radiological material that has condensed or plated-out in the reactor containment boundary) is protected from the direct effects of a fire by the shielding vessel or the reactor pressure vessel and coolant boundary. Due to the limited size of the fire, failure of the reactor containment boundary and release of materials is not postulated for this event.

The likelihood of BRA 1 is not based on truck fires given the unusual load, but rather a general fire for a comparable situation.

BRA 2 is a diesel fuel fire that originates outside the transport container and propagates into the transport container and ignites combustible material in the transport container which damages the package. The quantity of diesel fuel assumed is limited to the maximum possible fuel in transporter fuel tanks (e.g., 300 gallons). Fires that involve a collision and ensuing fire including those that involve a greater quantity of diesel fuel are considered in other accidents such as impact with another truck and a tanker and subsequent fire.

The likelihood of BRA 2 is based on truck accident data for fire-only events.

4.4.3.2.2 Road Impact and Loss of Shielding Accidents

This section describes development of bounding representative accidents for the impact-only accidents (no fire) that occur on the road. These accidents can result in release of radioactive material and loss of shielding resulting in direct radiation exposure. There are seven separate road impact accidents (Accidents 1(a), 1(b), 2(a), 2(b), 3(a), 3(b), and 4(a) associated with collision with another vehicle, collision with fixed objects, drops to a lower elevation, non-vehicle accidents, and a high wind event) that could cause impacts that damage the TNPP package. Two of these accidents that are referred as non-collision accidents (i.e., Accidents 3(a) and 3(b)) do not technically involve a collision because they do not involve collision with another vehicle or object and do not involve a drop to lower elevation. However, a rollover does involve hard impact with the ground which is likely to be the asphalt or concrete roadway and shoulder. A jackknife could involve violent swinging of the trailer and contents which could lead to some impact internal to the container for objects that become unrestrained (e.g., tools), but such impacts are not expected to damage the package. The high wind event (i.e., Accident 4(a)) can lift or move the transport container causing impact. The degree of damage to the TNPP package is hard to estimate because no tests and only preliminary analysis has been performed so far. Accordingly, it is hard to differentiate road accidents that involve impact from each other in terms of potential damage to help define bounding representative accidents.

It is assumed, however, that impact with heavy vehicles and solid unyielding objects (e.g., concrete abutment), impacts with hard rock, drops to a lower elevation, and rollovers would create significant forces on the TNPP package. These forces create damage to the TNPP package and its shielding which results in release of radiological material and increased direct radiation. Conversely, impact with light vehicles or objects, and impacts do not create much force (e.g., impact with signs), and jackknifes are not expected to cause much damage to the TNPP package.

Based on the discussion above, BRA 3 includes impact with heavy vehicles and solid unyielding objects (e.g., concrete abutment or a rock embankment), and falls to a lower elevation (e.g., drop from a bridge), and rollovers which can result in hard impact of the asphalt or concrete roadway. It is assumed that this bounding accident results in damage to the TNPP package and shielding which results in release of radioactive material and direct radiation exposure. It is assumed that the high wind accident creates a level of TNPP package damage like the other accidents in this group. The potentially positive effect of diluting the concentration of radiological material that is released is not be credited. Determination of the damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this bounding representative accident considers the mechanical impacts consistent with the worst-case situation for this bounding representative accident scenario.

The likelihood of BRA 3 is based on the sum of the accident frequencies for the accident scenario assigned to this group. Though this scenario was postulated in the hazard analysis, an accident involving high wind that leads to a consequence of this severity is assumed to be very unlikely for BRA 3 because a transport would not deliberately be allowed during extremely inclement weather. Therefore, the likelihood of BRA 3 is assumed to be dominated by highway accidents that lead to severe impact.

Based on the discussion above, BRA 4 includes impact with light vehicles or objects that do not create much force when impacted (e.g., signs), jackknifes that do not involve impact, and impacts with an yielding object (e.g., a road sign or soil/clay embankment). BRA 4 is further broken down into 4M (medium) and 4L (light). BRA 4M accidents are less than a hard impact highway accident that results in release of some radiological material and loss shielding. These medium impact accidents are defined as a severe collision with a light vehicle. It is assumed that this bounding representative accident results in some degree of damage to the TNPP package and shielding which results in release of radioactive material and direct radiation exposure, but less damage than BRA 3. The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this bounding representative accident consider the mechanical impacts consistence with the worst-case situation. The likelihood of BRA 4M is based on severe collisions with a light vehicle (i.e., one that results in fatality and or injury).

BRA 4L accidents result in no release of radiological material or loss of shielding. These light impact accidents are defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment) or impact that is not severe with a light vehicle (e.g., results in property damage only). Accordingly, the likelihood of BRA 4L is based these kinds of accidents. A precise definition of yielding objects is discussed in the frequency estimation in Section 4.3.3.1 as presented in Table 4-16.

4.4.3.2.3 Road Impact and Subsequent Fire Accidents

This section describes development of bounding representative accidents for the road impact-only accidents that lead to fire. There are three accidents of this type (i.e., Accidents 4(a), 4(b), and 4(c)) which consist of collision with a vehicle or fixed object or rollover and fire, collision with a tanker carrying flammable material and fire, and an accident that involves a drop to a lower surface (e.g., a drop from a bridge or overpass) and fire. Accordingly, it is assumed mechanical damage caused by impact, as discussed for accidents addressed in Section 4.4.3.2.2, and fire which can: (1) create thermal stress for material such as metal, so that activated material or material that contains held-up or plated-on radioactive material can be made airborne from sloughing of oxide from the oxidizing mass, and (2) create a convective current that causes the material to be airborne. If the collision is with another large truck, there could potentially be a maximum of 600 gallons of diesel fuel involved if the maximum fuel capacity of both trucks is assumed to be 300 gallons. Therefore, determination of the damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this bounding representative accident consider these release mechanisms.

Collision with a large tanker carrying flammable material has the potential to produce the greatest radiological consequence because the fire can be larger than other fires in this group given the possible quantity of flammable material that might be spilled from the tanker in combination with the fact that a tanker is a heavy vehicle with the potential to create strong forces in a collision. An additional consideration is the fact that the tanker could contain explosive material which may cause greater mechanical impact. Accordingly, BRA 5 is defined as Accidents 3(a) and 3(b) which are essentially all road impact accidents that result in fire except collision with a tanker carrying flammable material. BRA 6 is then defined as collision with a tanker carrying flammable material and subsequent fire.

The likelihood of BRA 5 is based on the sum of the accident frequencies for the scenarios assigned to this group and the likelihood of BRA 6 is based on the accident frequency for collision with a tanker carrying flammable material.

BRA 5 is further broken down into 5H (hard) and 5M (medium). BRA 5H accidents are hard impact highway accidents (i.e., equivalent to the impacts defined by BRA 3) that result in fire with exception of collision with a tanker carrying flammable material. BRA 5M accidents are medium impact highway accidents (i.e., severe collision with a light vehicle that leads to a fatality or injury) that results in fire.

4.4.3.2.4 Loss of Non-Pressurized Reactor Containment Boundary

This section describes development of the bounding representative accidents for loss of package containment events specifically for a non-pressurized release from the reactor containment boundary but not associated with a road impact accident. There are five accidents of this type (i.e., Accidents 7(a), 7(b), 7(c), 7(d) and 7(e)) which consist of breach of the reactor containment boundary for the following reasons: (1) random containment failure (e.g., failure of a seal, connection, or joint), (2) vibration and shock from over-the-road travel, (3) human error in packaging the reactor containment boundary, (4) human error during TNPP disassembly leading to undetected latent failures in containment, and (5) extreme cold that fails containment. These accidents lead to about the same radiological consequences in that there is no motive force to drive material out of the containment except for possible small differences in pressure and temperatures inside and outside the sealed elements. Vibration and shock, which is a cause for one the accident scenarios in this group, is also a factor in the other scenarios of this group even though it does not cause the breach. Table 4-5 indicates that vibration and shock could loosen surface material held up in the compact or other core structures adding to radiological material that might be released (i.e., the shock and vibration accident scenarios is assigned to Consequence Category B). Accordingly, some fraction of the available MAR is assumed to be released except the TRISO fuel itself, but radioactive material that has diffused and is held up in the compact and other core structures might be loosened by vibration and shock and also be released. The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident needs to reflect that a non-pressurized containment has been breached but that road travel vibration and shock may have loosened radioactive material in the reactor containment boundary. Accordingly, BRA 7 is defined as five of these accidents (i.e., Accidents 7(a), 7(b), 7(c), 7(d) and 7(e)).

The likelihood of BRA 7 is based on the sum of the accident frequencies for the accident scenarios assigned to this group.

4.4.3.2.5 Loss of Pressurized Reactor Containment Boundary

This section describes development of bounding representative accidents for loss of package containment events that are not associated with a road accident and specifically involved pressurized release from the reactor containment boundary. There are two accidents of this type (i.e., Accidents 8(a) and 8(b)) which consist of breach of the reactor containment boundary: (1) impact on vents or the heat transfer pathway that decreases heat removal in combination with a containment failure, and (2) high ambient air temperature and residual decay heat in combination with containment failures. These accidents are considered to lead to similar radiological consequences in that motive force, but non-continuous force, exists to drive material out of the containment. As soon as the pressure inside and outside the contained elements equalize (which could happen quickly), then the motive force dissipated. The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident needs to reflect that containment is pressurized and that normal road travel vibration and shock may have loosened radioactive material in the reactor containment boundary. Accordingly, even though Table 4-5 indicates that the source term does not include radioactive material that has diffused and is held up in the compact and other core structures (i.e., is not assigned to Consequence Category B), it should be assumed for these accidents that vibration and shock from road travel may have loosened radioactive material from surfaces inside the reactor containment boundary such as the core structure. The two accident scenarios discussed define BRA 8.

The likelihood of BRA 8 is based on the sum of the accident frequencies for the accident scenario assigned to this group.

4.4.3.2.6 Loss of Non-Pressurized Non-Reactor Containment Boundary

This section describes accident involving loss of package containment events for non-pressurized non-reactor containment boundary accident scenarios. There are four accidents of this type (i.e., Accidents 9(a), 9(b), 9(c), and 9(d)) which pertain to containment breach of other parts of the TNPP package besides the reactor containment boundary. As shown in Table 4-5, these accidents have been assigned to Consequence Category D and only involve release of contamination from the package outside the reactor. Given that these accidents only result in release of contamination from package elements that have been handled during disassembly and loading of the TNPP package, the management of the risk from these scenarios can be considered covered by normal radiation safety practices. These scenarios should be provided as input to development of radiation safety controls, but do not define a bounding representative accident for which detailed likelihood and consequences will be developed.

4.4.3.2.7 Unplanned Exposure to Radiation

This section describes unplanned increase in exposure time to radiation. This set of events consist of Accidents 11(a), 11(b), and 11(b) in which technical or logistical difficulties result in a lengthened transport time and an increased exposure of workers to radiation caused by: (1) mechanical breakdown of the truck, trailer, or transport container, (2) technical problems with the TNPP package that requires resolution due to unanticipated failure or errors, and (3) adverse weather that stalls or delays transport. Given that these accidents only result in increased routine exposure (though unanticipated), the management of the risk from these scenarios can be considered covered by normal radiation safety practices. However, these scenarios should be provided as input to development of those controls, but do not define a bounding representative accident for which detailed likelihood and consequences will developed.

4.4.3.2.8 Criticality Accidents

This section describes development of bounding representative accidents for criticality accidents that happen during transport. There are two accidents of this type (i.e., Accidents 12(a), and 12(b)) one of which consists of the addition of a moderator and a change in core geometry, and the other which consist of control rod withdrawal. Both require a road accident to initiate the accident. Accident 12(a) consists of a drop into a body water (e.g., from a bridge) and enough impact to cause a change in core geometry. Accident 12(b) consist of inadvertent control rod bank withdrawal at cold conditions caused by a road accident.

BRA 9 is defined as the addition of a moderator and a change in core geometry caused by drop into a body of water (i.e., Accident 12(a)). This accident requires a highly unlikely set of circumstances because there are only limited sections of road where such a drop into a body of water is sufficient enough to immerse the TNPP. The frequency of this accident can be determined the estimating the likelihood a road accident over those limited sections of highway or by looking at the data for accidents that result in a submerged vehicle.

BRA 10 is defined as a control rod withdrawal event caused by a road accident which could be collision with another vehicle or fixed object, a rollover, or a drop to a lower surface (i.e., Accident 12(b)). The likelihood of this accident scenario is based on the accident frequency and the conditional probably that control rod withdrawal occurs as a result of the accident.

4.4.3.2.9 Summary of Bounding Representative Accidents

Table 4-6 provides a summary definition of the bounding representative accidents discussed in Sections 4.4.3.2.1 through 4.4.3.2.8.

Table 4-6. Bounding Representative Accident Definitions

ID	Descriptions
BRA 1	Fire-only event that originates inside the transport container.
BRA 2	Diesel fuel fire-only event that originates outside the transport container and propagates into the transport container and ignites combustible material in the transport container which damages the package.
BRA 3	Hard impact highway accident that leads to release of radioactive material and loss of shielding. Includes impact with heavy vehicles and unyielding objects (e.g., concrete abutments or rock embankments), drops to lower elevation, or rollovers.
BRA 4M BRA 4L	Less than a hard impact highway accident that results in release of some radiological material and loss shielding. Medium impact that involves a severe collision with a light vehicle. Less than a hard impact highway accident that results in no release of radiological material or loss of shielding. Light impact such as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment) or impact that is not severe with a light vehicle.
BRA 5H BRA 5M	Hard impact highway accidents (i.e., equivalent to the impacts defined by BRA 3) that result in fire with exception of collision with a tanker carrying flammable material. Medium impact highway accidents (i.e., severe collision with a light vehicle) that results in fire.
BRA 6	Collision with a tanker carrying flammable material that leads to fire.
BRA 7	Loss of non-pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.
BRA 8	Loss of pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.
BRA 9	Addition of moderator and a change in core geometry caused by a drop into body of water that results in criticality.
BRA 10	Control rod withdrawal caused by impact from a road accident that results in criticality.

4.5 Development of Likelihoods for TNPP Transportation Accident Scenarios

This section describes the development of the likelihood of TNPP transportation accidents and provides the bases for those estimates. Section 4.5.1 discusses collection, analysis, and characterization of route specific hazards. Section 4.5.2 discusses collection and analysis of large truck accident data. Section 4.5.3 discusses development of the likelihoods for accidents that can occur during TNPP transportation. Section 4.5.4 provides a listing of the primary assumptions that were made as part the accident likelihood development. Section 4.5.5 provides the estimation of the frequencies for each bounding representative accident.

4.5.1 Characterization of Route Specific Spatially Derived Hazards

The assumed route from INL to WSMR passes through parts of Idaho, Utah, Wyoming, Colorado, and New Mexico (Figure 4-3). The route traverses a diverse geography and population from very rural Wyoming to metro Denver. In the metro Denver area, a bypass route to the east (on Colorado E-470) was included in the analysis.

4.5.1.1 Background

There are many environments with different risks that will be encountered as the package is transported between locations. Although there are very few very large truck accidents from which to infer relative risk of each of the environmental hazards, these hazards are investigated and characterized. Based on past transportation studies they can include:

- Soil types. Following the work of Mills et al. (2006), the relative hardness was quantified for the assumed routes and is discussed in Section 4.5.1.2.

- Bridges. Presence of underpasses and overpasses were enumerated and is discussed in Section 4.5.1.3.

- Rivers and Waterbodies. Bodies of water sufficient to submerge the reactor vessel near the assumed route was investigated and is discussed in Section 4.5.1.4.

    - River, stream, and waterbody crossings.
    - Length of route adjacent to rivers, stream, and waterbodies.

- Drop offs. Portions of the assumed route where a vehicle could drop to a lower elevation was investigated and is discussed in Section 4.5.1.5.

- Population density. Population density data though not used in the TNPP transportation PRA is typically used in Environmental Impact Statements (EIS) and is discussed in Section 4.5.1.6

In the sections below, these hazards are discussed in general for the assumed route. Publicly available data were used with the provided route information for this report.

Figure 4-3. Route from INL to WSMR, including a Bypass Route to the East of the Denver Metro on Colorado E-470

4.5.1.2 Soil Types

The following definitions will be used for the identification of Map Unit subcomponents that will behave like "Hard Rock", "Soft Rock", "Rocky Soil", or "Other Soils, Clay, Silt". Previous work used the 1:250,000 scale 1996 State Soil Geographic (STATSGO, USDA-SCS 1993) data, however those data have been superseded by the more well resolved Soil Survey Geographic (SSURGO) data (USDA-NRCS 2005). The SURRGO data is 1:24,000 scale and uses a different data model than the earlier STATSGO data.

Following the work of Mills et al. (2006), the new soils data model was used to classify the data into the four categories. Multiple tables were needed: the "chorizons" (soil horizons which include information about the existence and fraction of rocky soil) and corestrictions (information about depth to bedrock and cementation). These data have a one-to-many correspondence for the map units. The most conservative hardness category was used for each map unit.

The previous STATSGO data model included a hardness category that specified if the bedrock was removable with a backhoe or only by blasting. As those data are not included in the current data model, the cementation information from the corestrictions table was used. That table classifies hardness as (from hardest to least): Indurated, Very Strongly Cemented, Strongly Cemented, Moderately Cemented, Weakly Cemented, and Noncemented.

The following hierarchy was used to determine the category for each Map Unit; the most conservative hardness for each Map Unit was used:

- A Map Unit subcomponent was defined to be "Hard Rock" whenever the average depth to the bedrock that lies below the subcomponent surface was on average ≤ 2 ft and the bedrock was "moderately cemented" to "indurated".

- If the Map Unit subcomponent was not defined to be "Hard Rock", then it would be defined to be "Soft Rock" if the average depth to the bedrock was, on average, ≤ 2 ft and the bedrock was "weakly-cemented" or "non-cemented".

- If the Map Unit subcomponent was not "Hard Rock" or "Soft Rock", then it was defined as "Rocky Soil" when the mass percent of rocks in the rocky soil layers in the top 3 ft of the soil was ≥ 25%, the average diameter of these rocks is ≥ 3 in., and the sum of the thicknesses of these layers is ≥ 2 ft.

- If the Map Unit subcomponent was not "Hard Rock", "Soft Rock", or "Rocky Soil", then it would be defined to be "Other Soils, Clay, or Silt, or water".

GIS methods of analysis used ArcMap™ software by Esri to overlay transportation routes from INL to WSMR onto the state-level SSURGO-derived soils categories and to determine the wayside-surface occurrence frequencies on a state-by-state basis. Tables 4-7 and 4-8 summarize the wayside surface types for each state for the two potential routes.

The assumed highway transport route for the Project Pele Prototype TNPP was from the INL to WSMR in New Mexico. The highway transport route for assessment of soil types in this section and bridges in Section 4.5.1.3 was estimated using the Web-Based Transportation Routing Analysis Geographic Information System (WebTRAGIS) computer code (Peterson 2018). Figure 4-4 through Figure 4-8 shows a potential highway route from INL to WSMR generated using WebTRAGIS based on the highway route-controlled quantity (HRCQ) routing requirements in 49 CFR 397.101 ("Requirements for motor carriers and drivers"). Figure 4-7 illustrates a potential sensitivity case where the E-470 beltway is used to bypass the center of Denver, Colorado.

The assumed destination of WSMR has been made for the purposes of analysis in the transportation PRA as well as demonstration of process and can be later altered, if necessary, by the Project Pele vendor to reflect program refinements prior to submittal of the transportation SAR and the request for exemption to the NRC.

Table 4-7. Surface Occurrence Fractions for Wayside Surfaces – INL to WSMR via Denver

State Traversed	Surface Type			
	Hard Rock	Soft Rock	Rocky Soil	Other
Idaho	0.228	0.000	0.232	0.541
Utah	0.079	0.000	0.309	0.612
Wyoming	0.073	0.162	0.000	0.764
Colorado	0.114	0.066	0.024	0.796
New Mexico	0.116	0.034	0.035	0.815
Route Average(1)	0.111	0.068	0.067	0.753

(1) Distance-weighted average values.

Table 4-8. Surface Occurrence Fractions for Wayside Surfaces – INL to WSMR via Denver, Colorado E-470 Bypass

State Traversed	Surface Type			
	Hard Rock	Soft Rock	Rocky Soil	Other
Idaho	0.228	0.000	0.232	0.541
Utah	0.079	0.000	0.309	0.612
Wyoming	0.073	0.162	0.000	0.764
Colorado	0.110	0.068	0.023	0.799
New Mexico	0.116	0.034	0.035	0.815
Route Average(1)	0.111	0.069	0.066	0.754

(1) Distance-weighted average values.

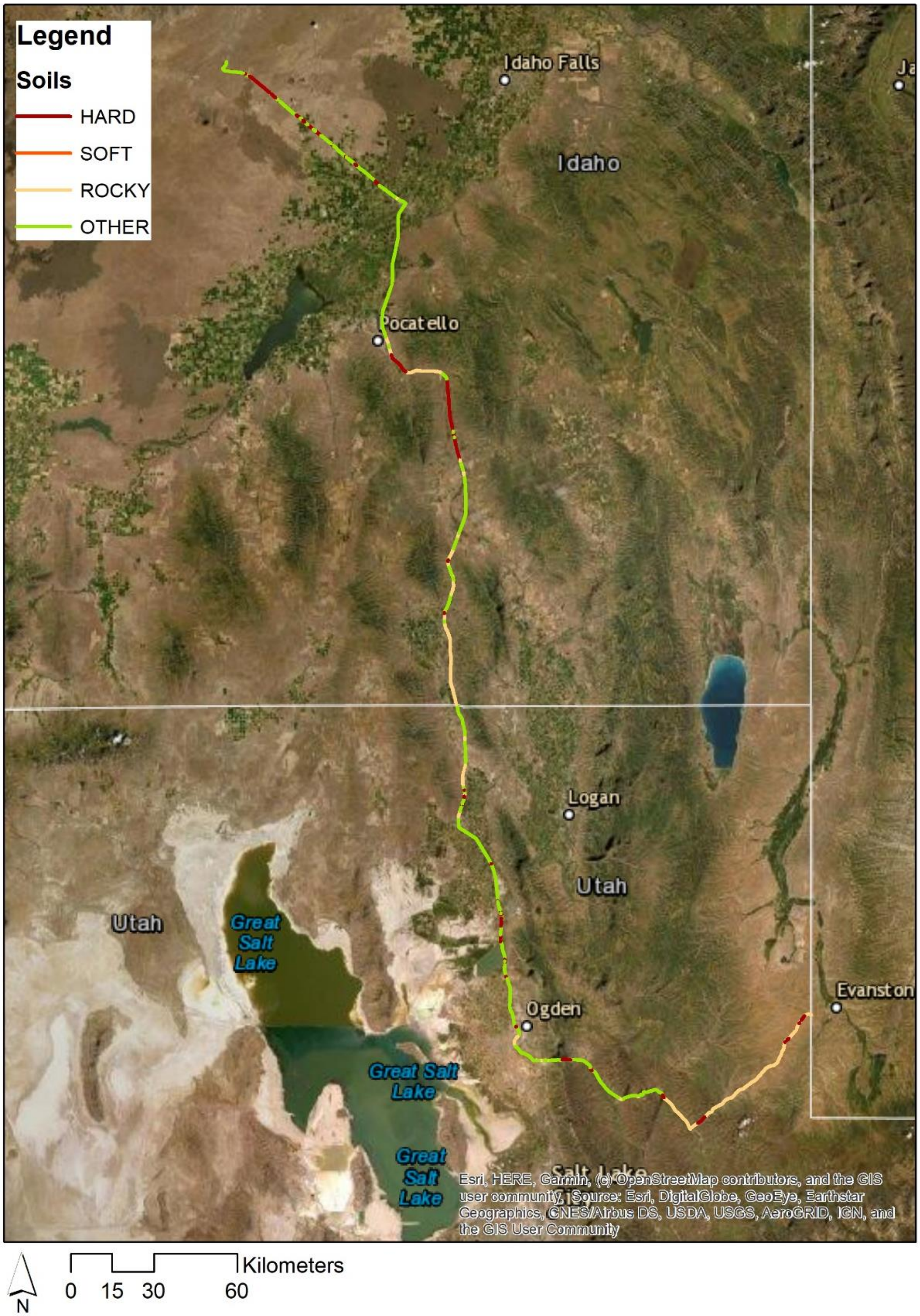


Figure 4-4. Potential Route from INL to WSMR with
Wayside Geology Classification: Idaho and Utah

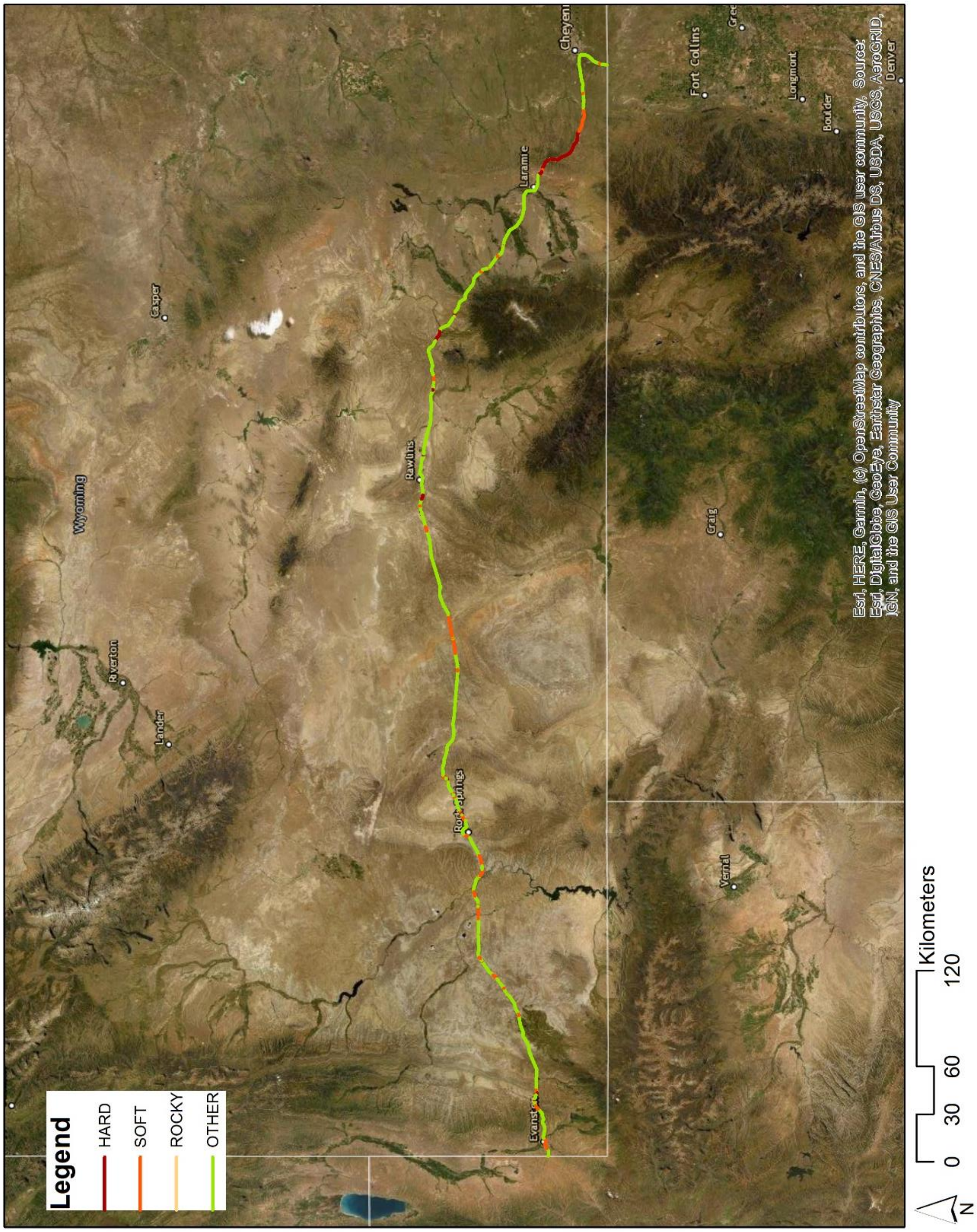


Figure 4-5. Potential Route from INL to WSMR with Wayside Geology Classification: Wyoming

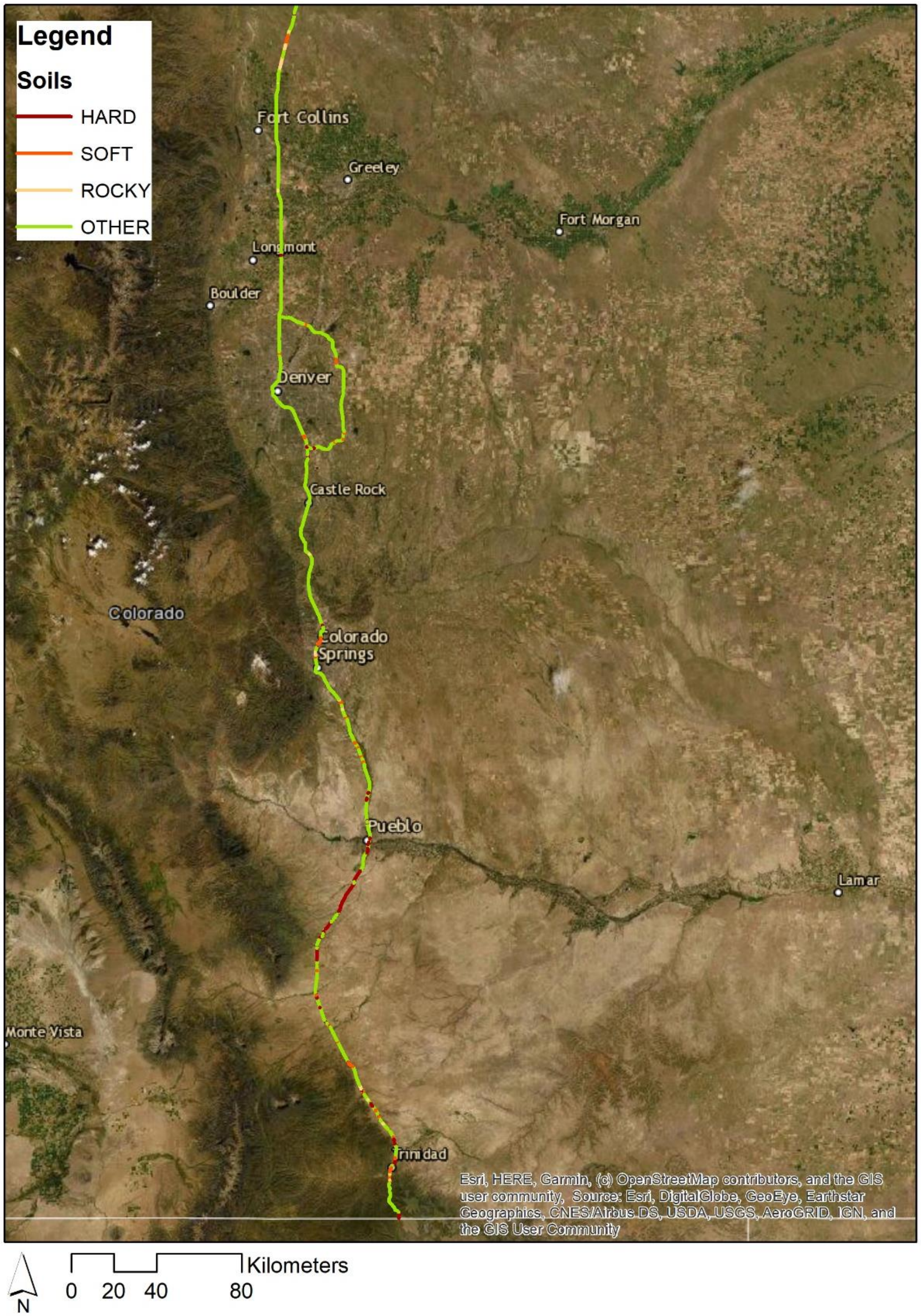


Figure 4-6. Potential Route from INL to WSMR with
Wayside Geology Classification: Colorado

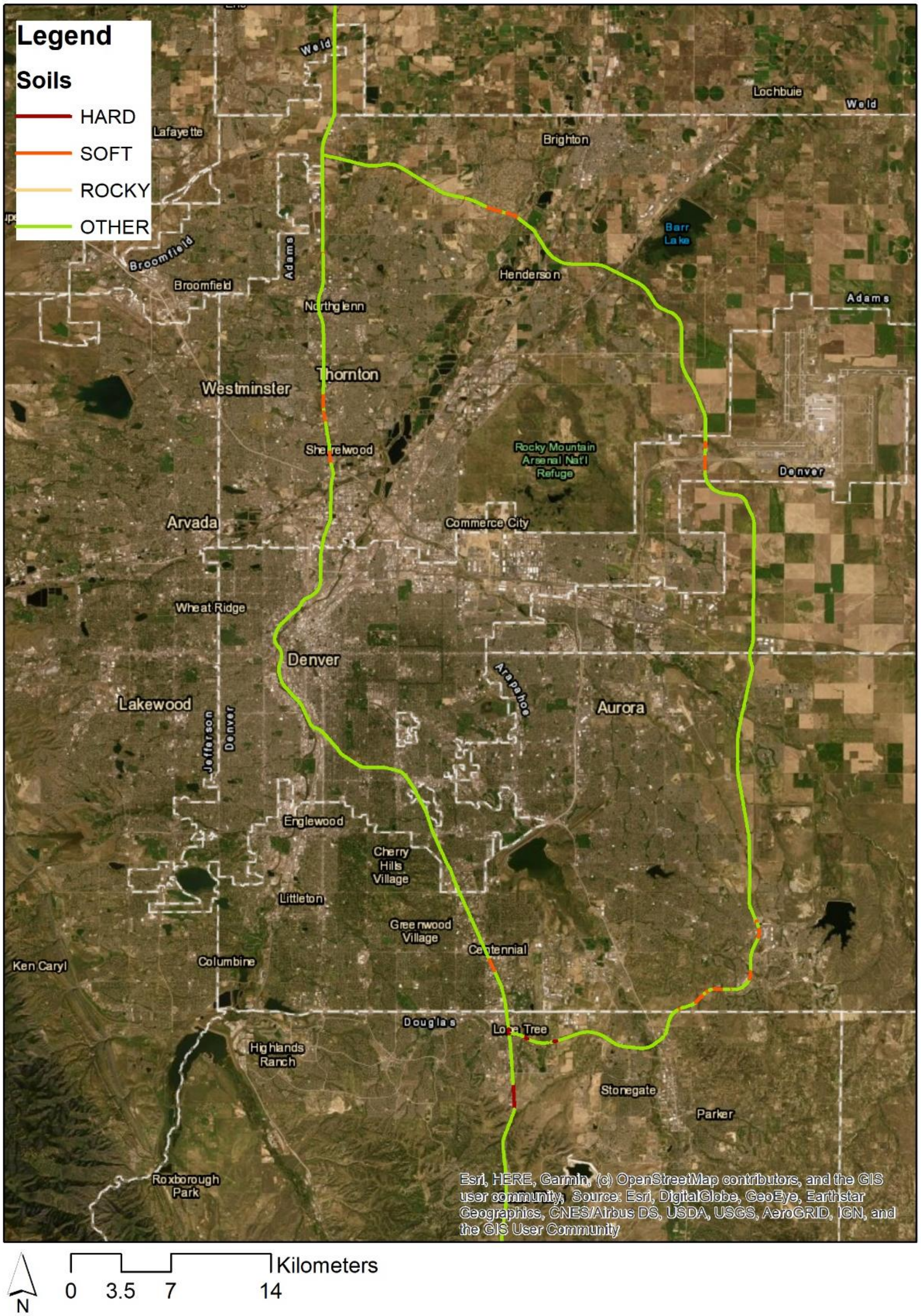


Figure 4-7. Potential Routes from INL to WSMR with Wayside Geology Classification: Denver Metro (The Eastern Route uses Colorado E-470 to Bypass the Denver Metro and the Mousetrap)

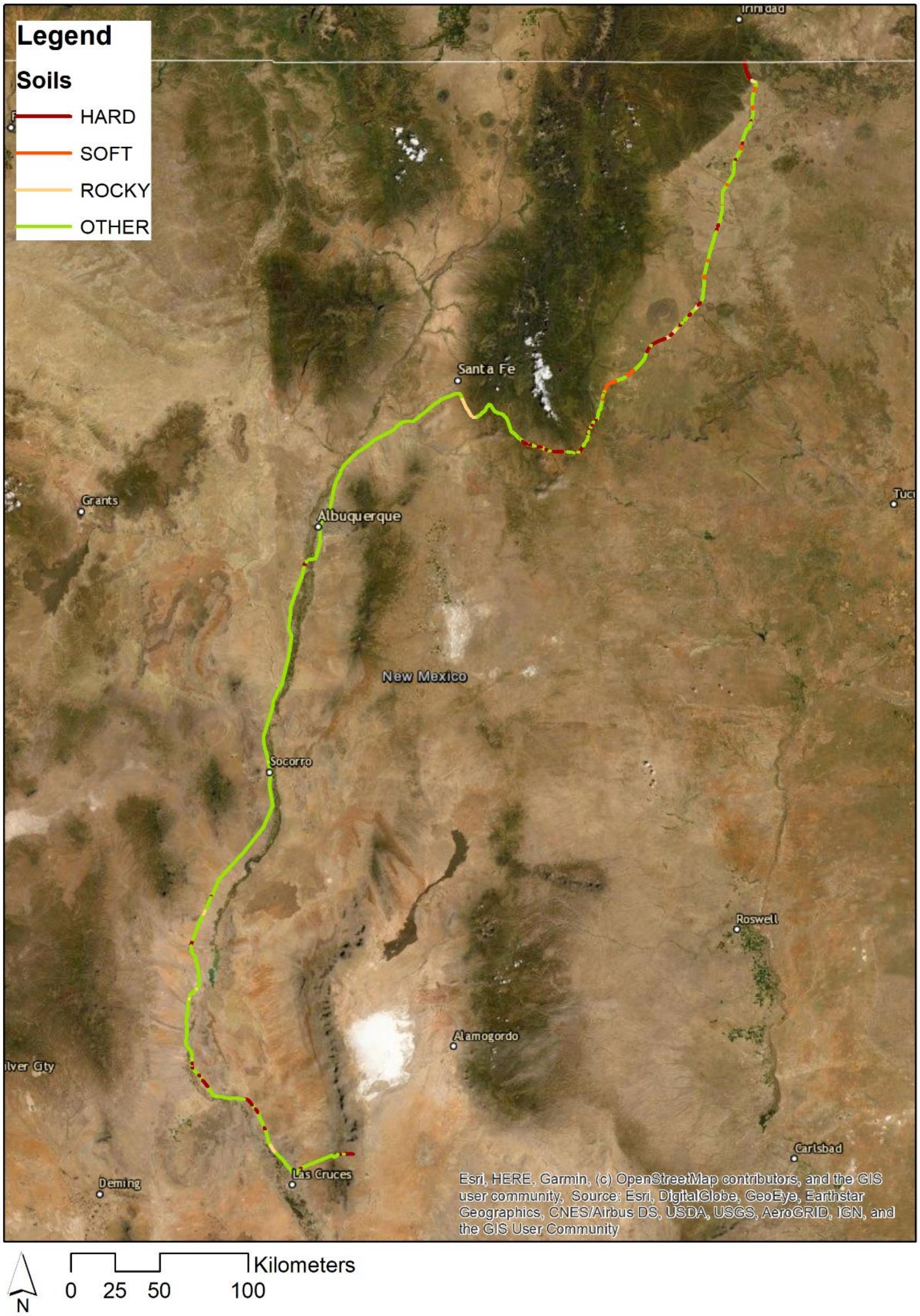


Figure 4-8. Potential Route from INL to WSMR with Wayside Geology Classification: New Mexico

4.5.1.3 Bridges

The National Bridge Database (NBD)[26] was used to assess the bridges along the transportation route. There are two types of bridges: overpasses and underpasses. The overpasses can be over roads, rivers, railroads or other features, while the underpasses are typically under roads or railroads. The overpasses present "fall" hazards, while the underpasses have substantial support structures that could present collision hazards. Only the major river crossings are typically included in the NBD. The NBD has inconsistencies in location and variations in coding from state-to state. Table 4-9 has state-by-state information on the count of underpasses and overpasses, and the minimum drop from the overpasses (maximum drop is not in that database).

Table 4-9. National Bridge Inventory Used to Determine the Number over Overpasses and Underpasses Along the Assumed Route

State	Total Under/Overpass	Route Underpass	Route on Overpass	Overpass Minimum Drop		
				0-5 m	5-9 m	>9 m
Idaho	59	14	45	25	20	0
Utah	92	18	74	42	32	0
Wyoming	175	15	160	112	48	0
Colorado outside Denver	241	79	162	118	43	1
Colorado - Denver East	62	25	37	16	20	1
Colorado - Denver West	71	43	28	15	13	0
New Mexico	358	72	286	227	59	0

4.5.1.4 Rivers and Waterbodies

In areas with significant topography, such as mountainous areas, the most efficient passage is predominantly along the river valleys which is where the highways are constructed. Between INL and WSMR, there are several mountainous areas to have to be traversed. In these areas, the interstate crosses a few major rivers and streams, and there are several locations where rivers and stream run parallel to the interstates (e.g., Figure 4-9).

Bodies of water with sufficient depth to submerge the reactor vessel diameter have the potential to initiate a flooded reactor criticality event. (The reactor vessel is about 5 ft in diameter not counting the empty water shield which could be ruptured in an accident.) Accordingly, rivers and streams (there are no other bodies of water) within 50 m of the highway are considered hazards if there is enough downward slope from the roadbed to the water so that an accident could potentially result in the transportation package ending up in the water. Accordingly, stream and river crossings that cross the route or are adjacent to the route were investigated.

[26] Available at https://www.fhwa.dot.gov/bridge/nbi.cfm.

Figure 4-9. I-84 Along the Weber River from Google Maps
(Note the River Directly Adjacent to the Interstate)

The evaluation described in Sections 4.5.1.2 and 4.5.1.3 used an assumed route at a level of resolution that did not consider the difference between northbound and southbound lanes including on and off ramps or the width of the median between lanes. This approach is insufficient to assess the proximity to bodies of water to route travelled by the TNPP package because precision is needed to identify whether the body of water presents a hazard. Accordingly, the southbound route was redefined at a higher level of resolution and included consideration of on and off ramps and loops, so that a detailed dataset could be created of route segments where a water hazard exists. The route data was extracted from Open Street Maps (OSM) (2022) using QGIS 3.2[27] a Java-based OSM data query tool (Stadtherr et al. 2022). This data proved to be more spatially accurate than the data used generate the hazard results described in Section 4.5.1.2 on soil types, and Section 4.5.1.3 on bridges. The route was split into 100 ft segments for its entire length except for where the Open Street Map source data uses shorter segments. The latitude and longitude of each segment was used to determine the true geographical length and to correct segments where auto-parsing of the route into 100-ft segments and the segments used in OSM were not in sync. The segmentation allowed filtering of the portion of the route where the hazards exist.

[27] formerly Quantum GIS

Stream data was extracted from the National Hydrography Dataset High Resolution (NHD-HR) (USGS 2022a), which is the best available GIS hydro-data for the United States. The streams dataset from NHD-HR is based on high resolution 3D digital elevation modeling and contains attributes for flow rate in cubic feet per second (ft^3/sec). This is calculated by adjusting the natural flow of water with the measured flow at stream gages scattered throughout the network. This dataset is considered to have the available “best flow and velocity estimates” (USGS 2022a, page 62).

Investigation for the presence of water in streams at various flow rates using images was used to determine that a flow rate of 3 ft^3/sec appeared to be the threshold flow for streams with enough volume to conceivably submerge a reactor vessel as shown in Figure 4-10. Most streams below a flow rate of 3 ft^3/sec showed very little to no visible water in the imagery investigation; therefore, “flow value 3” was used as a surrogate for the water depth of concern. This approach was used because no depth values are provided in the NHD-HR dataset. Streams with flow values 3 or greater, within 50 m of the route and not part of an underground pipe network, were queried using ArcGIS 10.8, a cloud-based mapping and analysis tool.[28] This included perennial and intermittent streams, canals, and artificial paths which is the designation given by the United States Geological Survey (USGS) to virtual lines running through large bodies of water such as lakes and wide rivers (USGS 2022a, page 65). Locations where the transportation route physically crosses rivers and streams, and where rivers and streams run adjacent to the route within 50 m were investigated.

Locations where streams run adjacent to the transportation route within 50 m of the road were further filtered based on slope. If the downhill slope of the adjacent land was greater than 1:4 to an adjacent stream, then it was conservatively assumed to be steep enough to cause the truck to roll or slide 50 m, given it left the road. Slope data was derived from a Digital Elevation Model (DEM) available for the entire United States (a DEM model is a representation of the topographic surface of the earth excluding surface objects like trees and buildings). This DEM is 8.3 x 8.3-m (often referred to as 10 m) resolution and was clipped to within 100 m of the route to remove edge effects at the 50-m threshold used for stream qualification. Next, elevation data was used to derive a slope dataset for 8.3 m by 8.3 m sections away from the road by querying whether the slope was 25% or greater between sections. The slope data was then used to screen adjacent streams less than 50 m with a slope greater than or equal to 1:4. These locations were then visually investigated to ensure that the sloped area was, in fact, between the roadway and the downhill stream as illustrated in Figure 4-11. Route segments closer than 8.3 m to the adjacent stream were added back to the qualifying segments to make up for the slope analysis resolution considering adjacent land relatively flat ground or not finding any slope data immediately next to the road.

[28] See https://doc.arcgis.com/en/arcgis-online/get-started/what-is-agol.htm.

Figure 4-10. Stream Images at Various Flow Rates Used to Determine the Minimum Threshold for Qualifying Streams in the Analysis

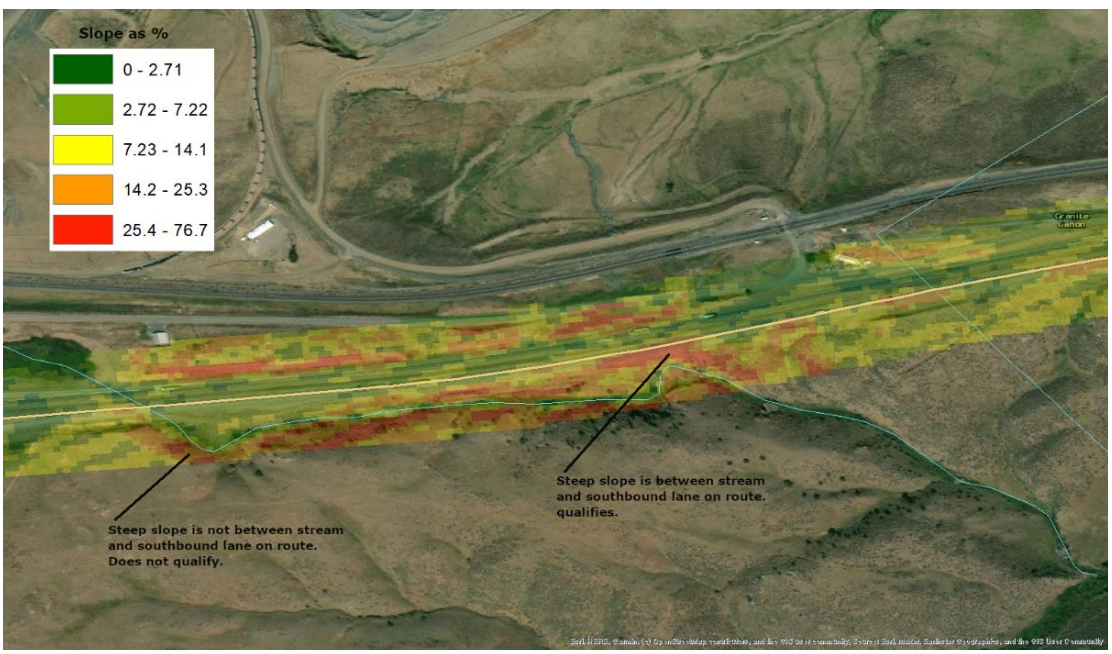


Figure 4-11. Slope Adjacent to Roadside Checked Manually to Ensure they are Downhill to the Stream

To visually assess the selected sites of concern, Google Street View images were downloaded for each of the qualifying route segments for both crossings and adjacency. Latitude and longitude values were assigned to the center of each route segment line using the Add Geometry tool in ArcGIS 10.8. These locations were input to an internal PNNL proprietary tool (Eshun et al. 2022) that generates images in any 360° direction from Google Street Map view application programming interface (API). Visual inspection using Google Street View images was used to qualify selected segments. The inspection revealed that many of the streams were dry at the time the Google Street View image was taken, suggesting that the overall assessment could be conservative or that timing of transport is an important consideration. These Google Street View images were not used to make refinements in the assessment but could be as needed. The plot showing where along the route that these views were taken is presented at the end of this section in Figures 4-12 and 4-13. The image ID numbers in the image name correspond to the route segment identification number shown in Figures 4-12 and 4-13.

This assessment of river and stream crossings and adjacency (i.e., adjacency refers to portions of the route where the assumed route runs along a river or stream) resulted in the final route segment counts and length totals listed in Table 4-10.

Table 4-10. River and Stream Crossing and Adjacency
(Total route is 1,289.2 miles)

	Segment count	**Total feet**
Route Through Denver		
Adjacent	156	14,393
Crossing	145	14,130
Total Route	301	28,523 (5.4 miles)
Route Bypassing Denver		
Adjacent	154	14,193
Crossing	147	14,419
Total Route	301	28,612 (5.4 miles)

This GIS analysis has some notable limitations. First, the NHD-HR stream dataset is based on best available location data of the stream/waterbody network derived from best available 3D elevation, lidar sourced, output for the United States. However, stream dynamics (e.g., change in the stream path since data collection) and computational error means that no guarantee can be made of the exact stream location without physical confirmation. Secondly, the stream flow minimum threshold of 3 ft^3/sec is based on visual investigation of water presence in streams and is only a rough surrogate for real depth measurements. Lastly, the resolution of the slope data (8.3 m) is likely to result in an exclusion of route-adjacent streams that are within 10 m of the road because the slope might appear flat in that area.

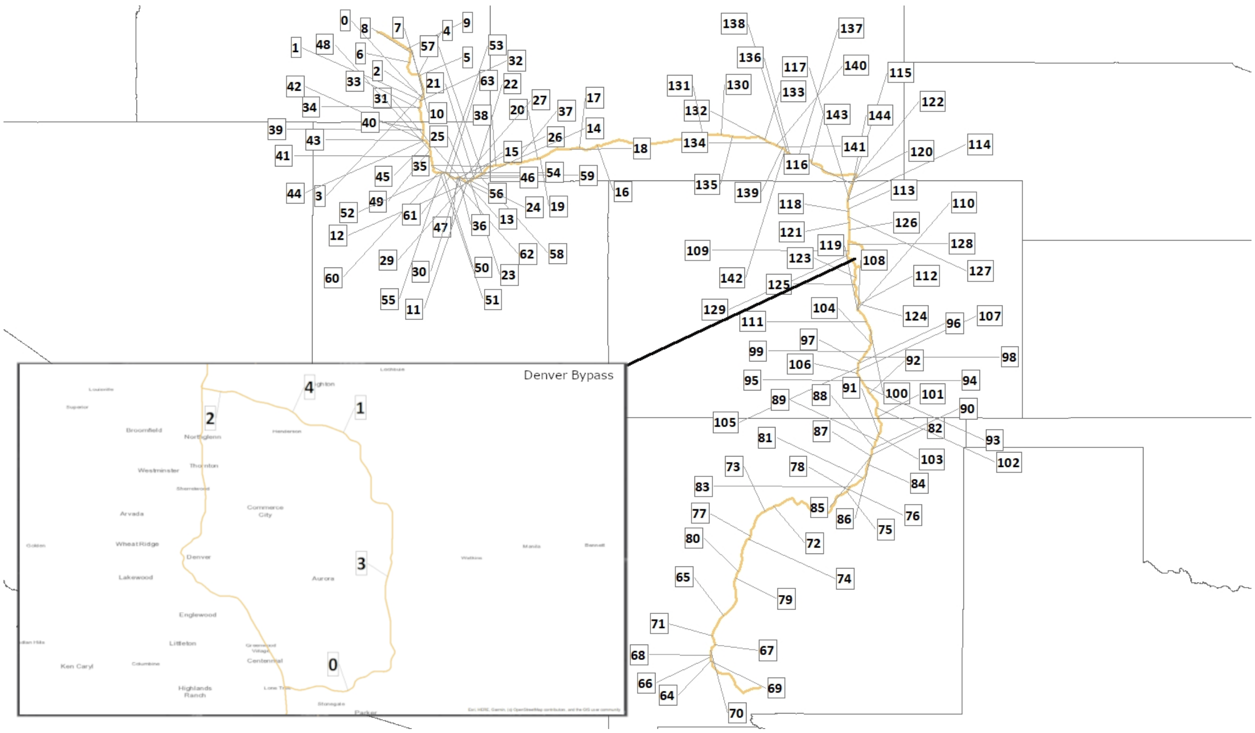


Figure 4-12. Road Segments Crossing Streams with a Flow Rate Greater than 3 ft^3/sec
(The ID Number Corresponds to the Google Street View Image)

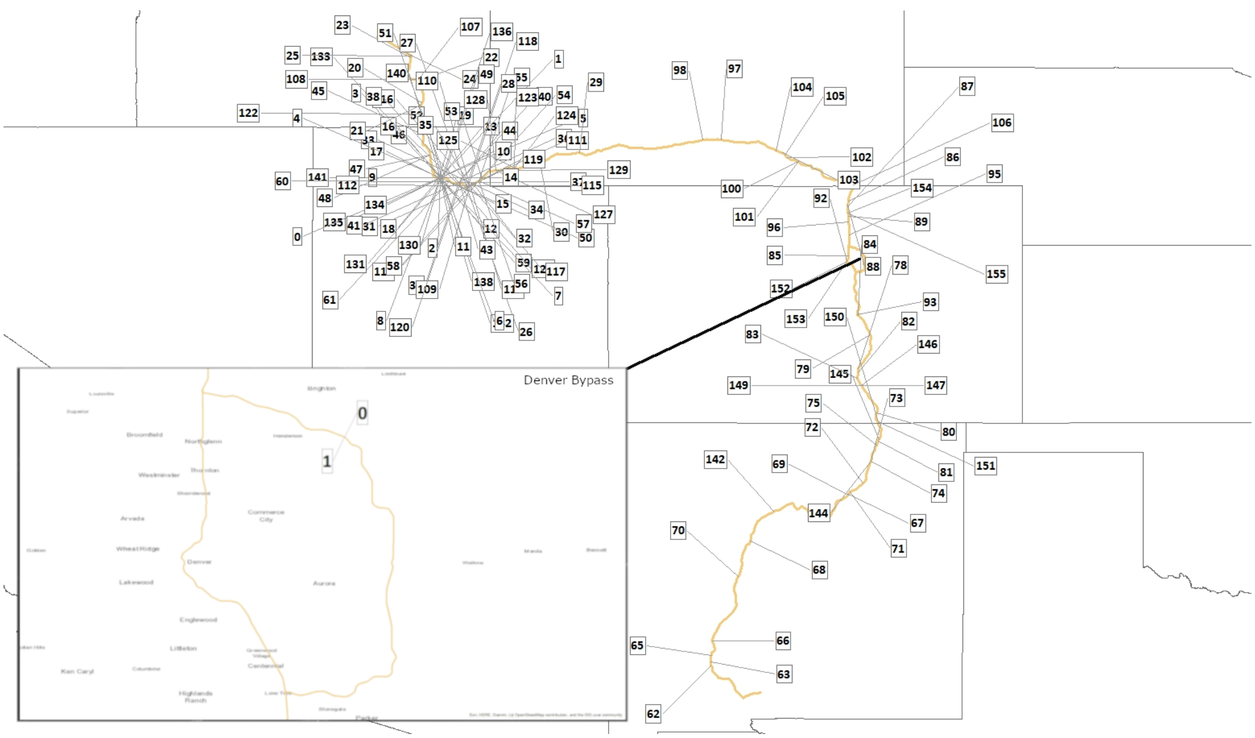


Figure 4-13. Road Segments that Run Adjacent within 50 m of a Stream
that has a Flow Rate Greater Than 3 ft^3/sec
(The ID number corresponds to the Google Street View image)

4.5.1.5 Drop Off to a Lower Elevation

Another specific hazard of the route are locations where there is a drop to a lower elevation just off the roadway. If a truck has an accident in these locations (e.g., on bridge or overpass, or near a steep embankment) and leaves the road, then significantly more damage to the TNPP Package could occur if the vehicle drops to a lower elevation.

As discussed in Section 4.5.1.4, the evaluations described in Sections 4.5.1.2 and 4.5.1.3 of the report used an assumed route at a level of resolution that did not consider the difference between northbound and southbound lanes including on and off ramps or the width of the median between lanes. Therefore, the southbound route was redefined at a higher level of resolution and included consideration of on and off ramps and loops, so that a detailed dataset could be created of route segments where drop offs of concern exist. Also, like investigation of bodies of water in Section 4.5.1.4, the route data was extracted from Open Street Maps (2022) using QGIS 3.2 Java-based OSM data query tool (Stadtherr et al. 2022). As described in Section 4.5.1.4 for investigation of bodies of water sufficient to submerge the reactor vessel, the route, in this case, was also split into 100 ft segments for its entire length except for where the Open Street Map source data uses shorter segments. The latitude and longitude of each segment were used to determine the true geographical length to correct segments where auto-parsing of the route into 100-ft segments used in OSM were not in sync. This segmentation allowed filtering of route segments based on the hazard analysis needs at a resolution within 100 ft.

Drop-off locations were decided based on slope percentage. If the slope of the adjacent land was greater than 33% grade within 25 m of the road, it was assumed to be sufficiently steep that a truck would drop (or roll or slide) to a lower elevation if it left the road. Slope data was derived from a Digital Elevation Model (DEM) available for the entire United States from the USGS. A DEM is a representation of the topographic surface of the earth excluding trees, building and other surface objects (USGS 2022b). This DEM had 8.3 x 8.3-m (often referred to as 10 m) resolution and was clipped to within 100 m of the route to remove edge effects beyond 25-m. Next, the elevation data was used to derive a slope dataset for 8.3-m by 8.3-m sections away from the road by querying whether the slope was 33% or greater between sections. The slope data was screened to include only areas greater than 33%. Then, the route segments within 25 m of sloped areas greater than 33% grade were examined to determine the direction of slope to ensure the slope was down and away from the route and not down and toward the route as illustrated in Figure 4-14.

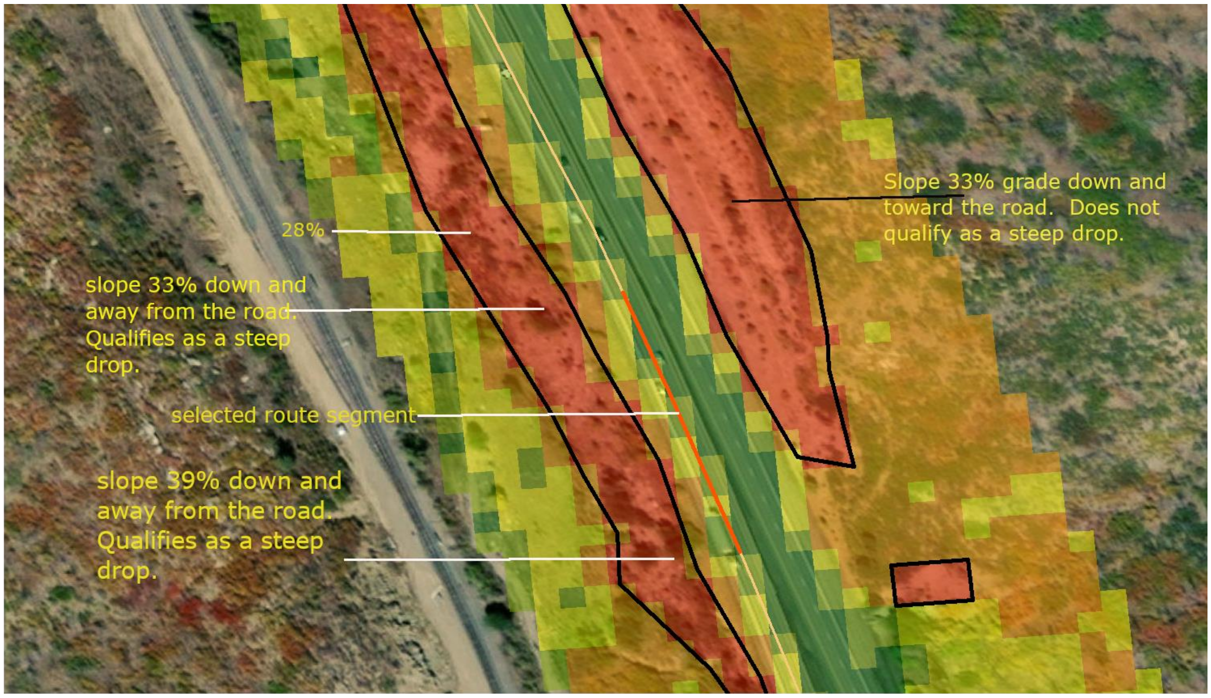


Figure 4-14. Route with a Qualifying and Non-Qualifying Slope of 33% Grade or More to the Sides of the Route Based on its Direction of Slope

Lastly, street view images were generated to visually confirm that the drop-off locations are real and no barriers existed between the route segment that would exclude a drop of the transportation package to a lower elevation. Google Street View images were downloaded for each of the qualifying route segments to visually assess selected sites of concern. Latitude and longitude values were assigned to the center of each route segment line using the Add Geometry tool in ArcGIS 10.8. These central coordinates were used in an internal PNNL proprietary tool (Eshun et al. 2022) that generates images in any 360° direction from Google Street View API. The road segment identification number are shown in Figure 4-15 and 4-16. These are the same locations of Google Street View images shown in Figures 4-17, 4-18, 4-19, and 4-20.

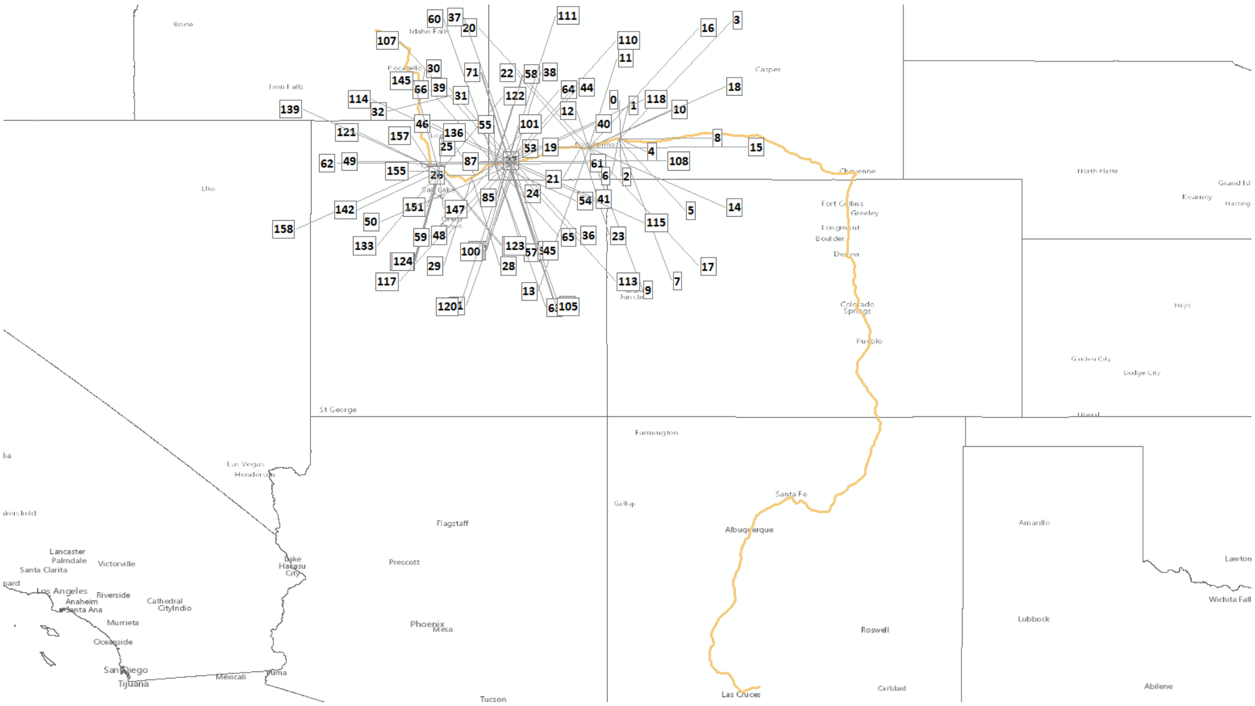


Figure 4-15. Road Segments 1-159 where a Drop-Off was Identified within 25 m of the Route with an Immediate Slope of at least 33% Grade

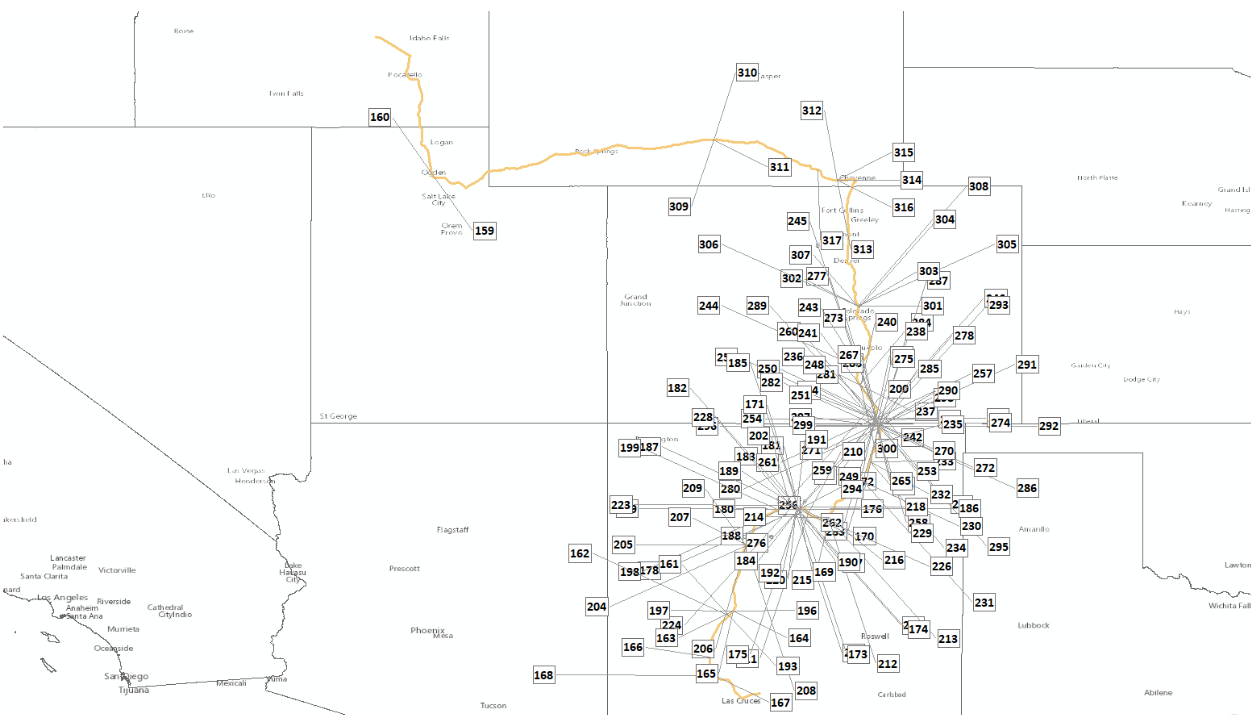


Figure 4-16. Road Segments 160-318 where a Drop-Off was Identified within 25 m of the Route with an Immediate Slope of at least 33% Grade

The visual analysis confirmed the GIS analysis captured various types of steep slopes from the road. In some cases, the potential for a drop event was easy to confirm because the location was obviously a bridge as shown in Figure 4-17. In other cases, the general slope appears to be up and away from the road but contains a steep drop (e.g., into ditch) near the road before the slope goes up like shown in Figure 4-18. The analysis also picked up steep drops associated with generally gentle terrain where there was hole or hollow next to the road as shown in Figure 4-19. Lastly, drops were typically found at the end of bridge guardrails where the earth berm created for the bridge is still steep as shown in Figure 4-20A, where the road is built up to a higher elevation than a rail line valley below as shown in Figure 4-20B and 4-20C, and where a drainage canal digs out the earth creating a steep drop along the side of the road as shown in Figure 4-20D.

Figure 4-17. Bridge Drop-Off from Route picked out by the GIS Analysis

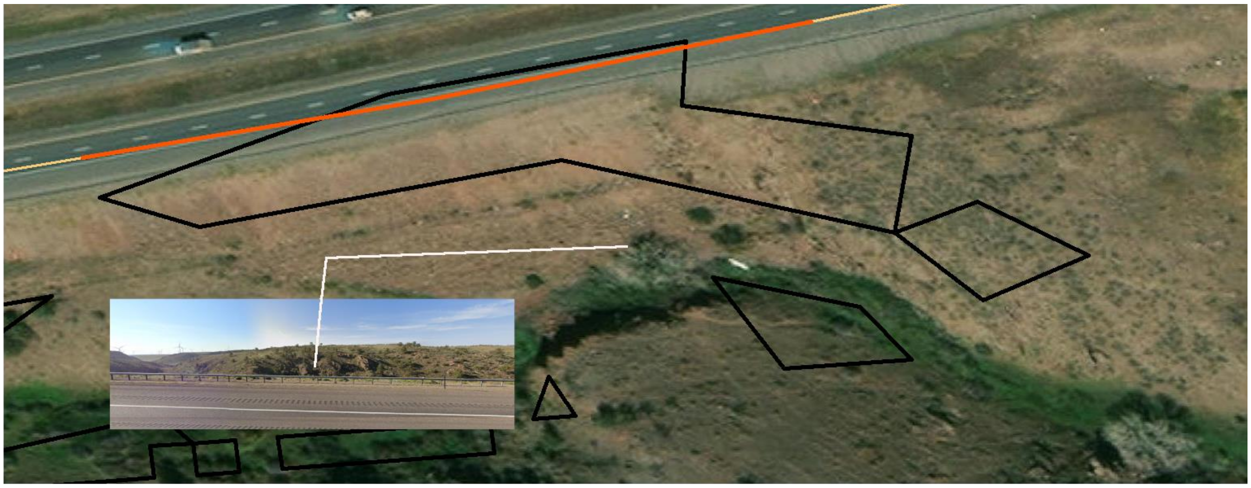


Figure 4-18. Large Ditch where the Slope appears both Down and Away as well as Down and Toward the Route

Figure 4-19. Relatively Gentle Slopes along the Route with a Deep Hole Emerging next to the Road

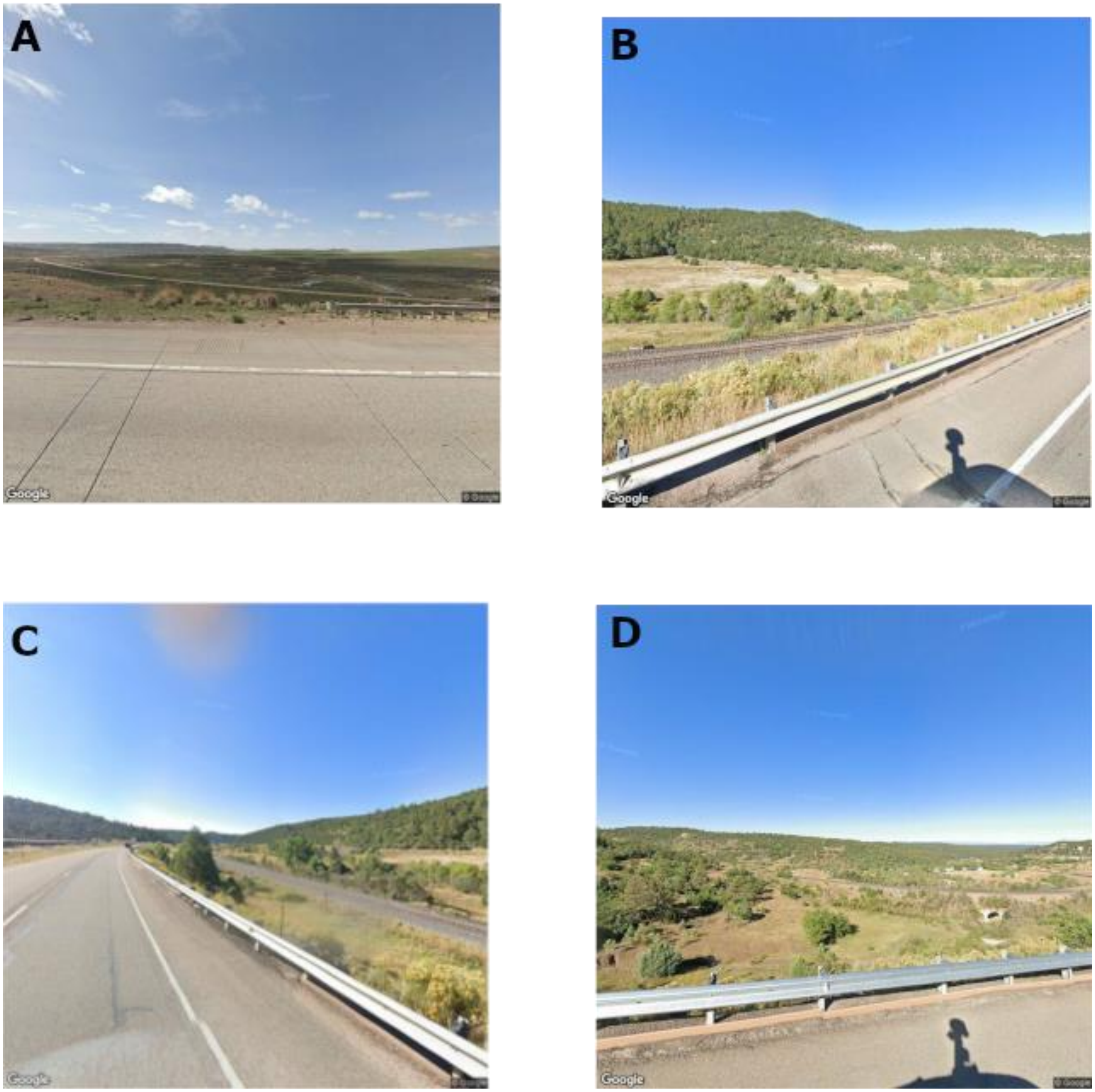


Figure 4-20. Various Steep Drop Scenarios

The GIS analysis identified 318 segments of the route that were considered sufficiently steep that a truck would drop (or roll or slide) to a lower elevation if it left the road as a result of an accident. The total length of the assumed route where this hazard exists translates to 31,800 ft of the route or 5.9 miles.

The primary limitations of this GIS analysis concern resolution of the data, features along the road that may prevent a drop to a lower elevation, and assumptions about what constitutes a “steep slope.” The resolution of the slope data (8.3 by 8.3-m) makes it likely that route-adjacent drop-offs that are within 10 m of the road may not be identified if the drop in elevation is not reflected in second 8.3-m section from the road. Secondly, any barrier that protects the truck from getting to the sloped ground such as a wall or ground feature (e.g., rock outcrop) less than 8.3 m wide may be difficult to identify from a top-down view (e.g., using satellite imagery) and would not show up as a higher elevation. Finally, the 33% grade criteria define “steep drops” based on visual investigation of known steep drops along the route and may potentially be too shallow to cause significant damage to the TNPP transport package. More data would be needed to accurately estimate criteria used to define a “steep slope.”

4.5.1.6 Population Density Information

A TNPP containing its irradiated fuel would contain a HRCQ of radioactive material as defined in 49 CFR 172.403 (“Class 7 (radioactive) material”) and would be subject to the highway routing requirements in 49 CFR 397.101. These requirements include ensuring that the motor vehicle is operated on routes that minimize radiological risk. The determination of radiological risk is required to consider available information on accident rates, transit time, population density and activities, and the time of day and the day of week during which transportation will occur. In general, these requirements are met by using an interstate highway, an interstate bypass or beltway around a city, and a state-designated preferred route.

The total distance for an HRCQ route from INL to WSMR generated using the WebTRAGIS computer code (Peterson 2018) is 1398.5 miles. Figure 4-3 illustrates this route and Figures 4-21 through 4-23 illustrate the spatial distribution of the population density along this route.

This route transits the Denver metropolitan area using I-25 and passes through the intersection of I-25 and I-70, an area colloquially known as the “Mousetrap.” The total population within 800 m of this route was estimated to be 1,660,000 people. If the Colorado E-470 beltway to the east of Denver were used to bypass the Mousetrap (see Figure 4-7), the total population within 800 m of the route would be reduced to 1,650,000, a reduction of about 1%. In addition, avoidance of the Mousetrap could reduce the potential for a transportation accident in the Denver metropolitan area, which has a population of about 3,000,000 people. Although the E-470 beltway is not a HRCQ route, it is recommended that the use of the E-470 beltway around Denver be discussed with the State of Colorado.

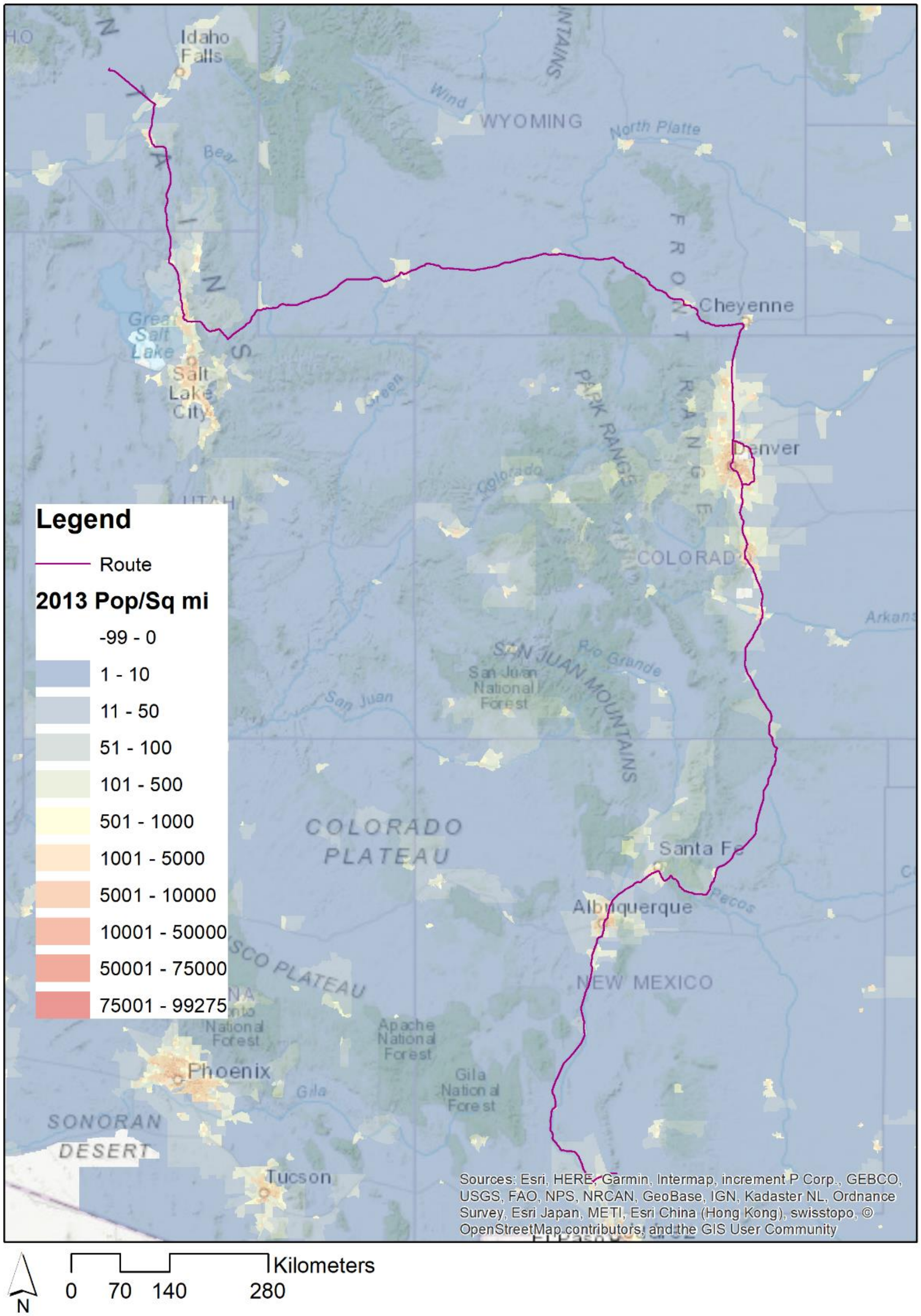


Figure 4-21. Population Density for the Entire Route

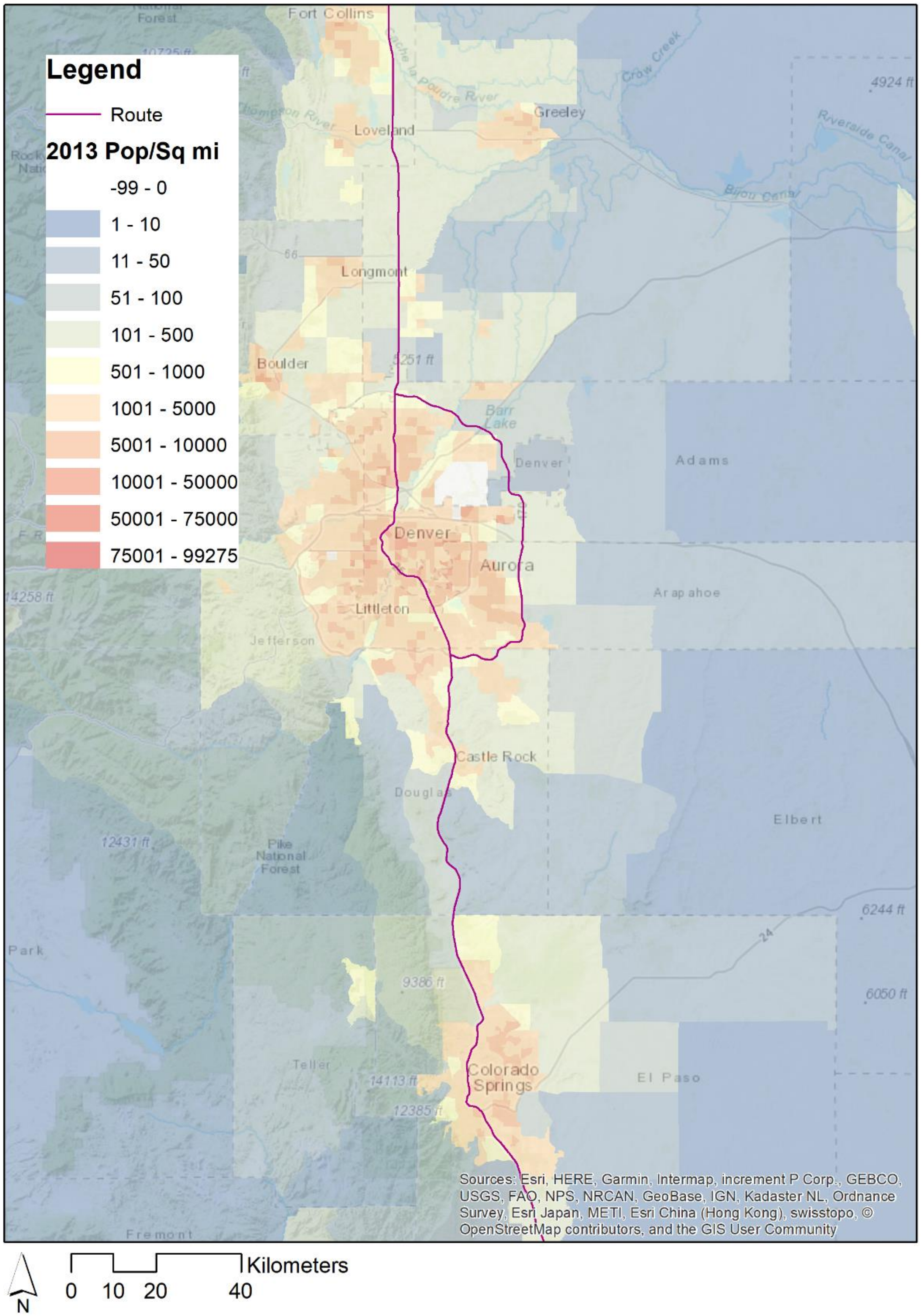


Figure 4-22. Population Density Along the Route for the Colorado Front Range including a Denver Bypass Route (Colorado E-470) to the East of the Metro Area

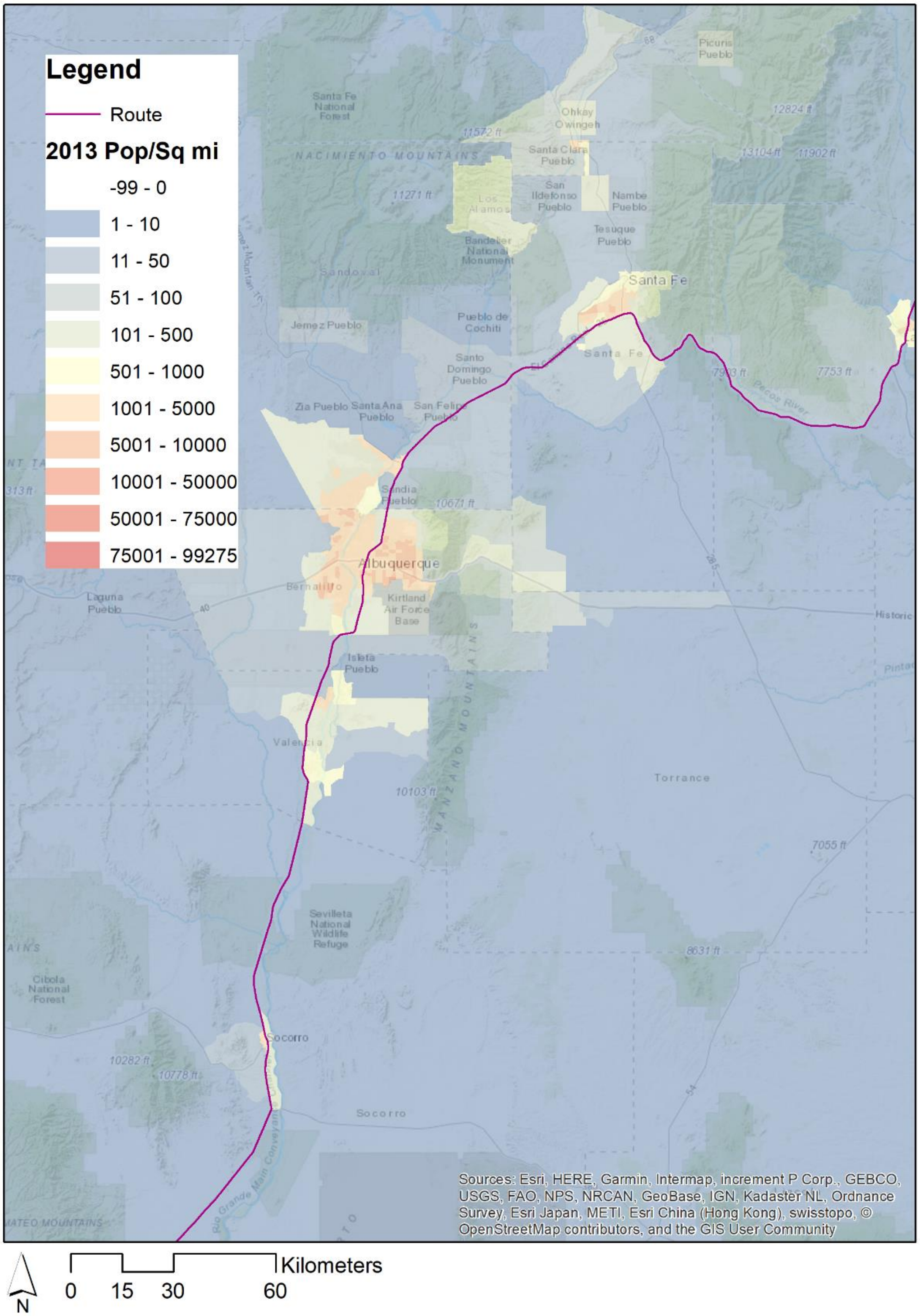


Figure 4-23. Population Density Along the Route in New Mexico, Including the Greater Albuquerque and Santa Fe Regions

4.5.2 Transportation Accident Rate Data Collection for Very Large Trucks

Accident occurrence data discussed in this section are based on "very large trucks" which are those categorized as greater than 26,000 lb gross vehicle weight (GVW) and include combination trucks. Statistics for this type of truck are of interest for this analysis because they are most indicative of accident rates for the type of truck that could be used for transport of a TNPP.

Combination truck categories include a truck tractor not pulling a trailer; a tractor pulling at least one full or semi-trailer (data are available for one, two or three trailers); or a single-unit truck pulling at least one trailer. Separate mileage data are not available for >26,000 lb GVW trucks but are readily available for combination trucks which is the major subset of very large trucks. However, mileage of some very large single-unit trucks (>26,000 lb GVW) may be omitted.

Accident rates were determined for very large truck travel from Idaho, with a presumed starting point of the INL, to New Mexico, ending at the WSMR. Traveling primarily on interstate highways, the route would transit five western states – Idaho, Utah, Wyoming, Colorado, and New Mexico – a total distance of 1,289 miles.

Accident data and mileage statistics were evaluated for the three years 2017-2019 for the five states and nationwide where state-specific data were not available. Key information sources for number of accidents and vehicles involved annually were the Motor Carrier Management Information System (MCMIS) for large truck "all accident" (crash[29]) data, and the Fatality Analysis Report System (FARS) mainly for fatal accident data but also for detailed nationwide injury only and property damage only data. New data platforms were established in 2016 for these data sources which are used in this report. Accident data were compiled starting with data for 2017 to allow for any database transition issues to be addressed. MCMIS data are considered preliminary for 22 months to allow for changes. At the time the data queries were made, 2019 was the most recent year of final data. Also, ending data collection in 2019 avoided the effect of the COVID-19 pandemic in 2020. The Federal Highway Administration (FHWA) was the source for mileage data used to determine accident rates per mile for 2017-2019.

Section 4.5.2.1 discusses very large truck mileage for the route divided into interstate and all state highways miles by each of the five states the route traverses. Section 4.5.2.2 discusses accident events and rates for very large trucks on all state highways of the route. Section 4.5.2.3 discusses accident rates for very large trucks on interstate highways for the route.

4.5.2.1 Very Large Truck Mileage for the Route

Detailed information on annual miles traveled by all vehicles, including categories of combination trucks and single-unit trucks, are available from the FHWA for each individual year.[30] Two annual summary tables are available from FHWA and can be used to determine state-specific mileage for combination trucks. The first is Table VM-2, "Vehicle-miles of travel, by functional system." The second is Table VM-4, "Distribution of annual vehicle distance traveled."

[29] Crashes include all fatal and non-fatal involvement.

[30] For example, https://www.fhwa.dot.gov/policyinformation/statistics/2019/.

Table VM-2 breaks down functional system travel into two major categories of "rural" and "urban" travel. In each of these are seven sub-categories, with the one of interest being interstate systems. The other sub-categories are other freeways and expressways, other principal arterial, minor arterial, major collector, minor collector, and local.

Table VM-4 provides the percentage of annual travel by vehicle type, also separated into separate "rural" and "urban" files. For both "rural" and "urban" there are three functional system travel categories – interstate system, other arterials, and other – combining some of the sub-categories of Table VM-2. Within these categories vehicle type percentages are presented for motorcycles, passenger cars, light trucks, buses, single-unit trucks, and combination trucks. The main category of interest is combination trucks, to be primarily consistent with very large trucks. The category of single-unit trucks would be included to address all large trucks (>10,000 lb GVW). As noted earlier, some single-unit trucks maybe be very large trucks but their mileage not included with combination trucks.

Very large trucks traveled nearly 29 billion miles on all highways in the five states from 2017-2019. Of these, 15.9 billion miles or 55% were traveled on interstate highways which are characteristic of the possible TNPP transport route. Annual and total mileage in each of the five states are shown in Tables 4-11 and 4-12.

Table 4-11. Very Large Truck Interstate Mileage 2017 to 2019(1)

State	2017 (Miles × 1E+06)	2018 (Miles × 1E+06)	2019 (Miles × 1E+06)	2017-2019 Total (Miles × 1E+06)	Percentage of 5 States
Colorado	1,117	1,131	1,146	3,393	21.3%
Idaho	652	665	678	1,996	12.5%
New Mexico	1,690	830	647	3,166	19.9%
Utah	1,491	1,529	1,546	4,566	28.7%
Wyoming	883	921	984	2,788	17.5%
TOTAL	5,833	5,075	5,001	15,909	100%

(1) Source: FHWA, Table VM2 and Table VM4. Data for combination trucks.

Table 4-12. Very Large Truck All State Highways Mileage 2017 to 2019(1)

State	2017 (Miles × 1E+06)	2018 (Miles × 1E+06)	2019 (Miles × 1E+06)	2017-2019 Total (Miles × 1E+06)	Percentage of 5 States
Colorado	2,123	2,127	2,181	6,431	22.2%
Idaho	1,099	1,110	1,140	3,348	11.%
New Mexico	2,487	1,645	1,253	5,385	18.6%
Utah	3,113	3,148	3,031	9,291	32.1%
Wyoming	1,437	1,536	1,551	4,524	15.6%
TOTAL	10,258	9,566	9,156	28,980	100%

(1) Source: FHWA, Table VM2 and Table VM4. Data for combination trucks.

4.5.2.2 Accident Events and Rates for Very Large Trucks on all State Highways of the Route

The basis for accident rates of very large trucks is the state-specific number of "all crash events" which occurred in the United States during calendar years 2017, 2018, and 2019. "All Crashes" are defined to include "fatal and non-fatal crash involvements." These data were accessed using the DOT, Federal Motor Carrier Safety Administration (FMCSA) online data query tool, *Analysis and Information Online (A&I)*.[31] The A&I crash statistics module provides users an ability to view crash data reports from either MCMIS or FARS:

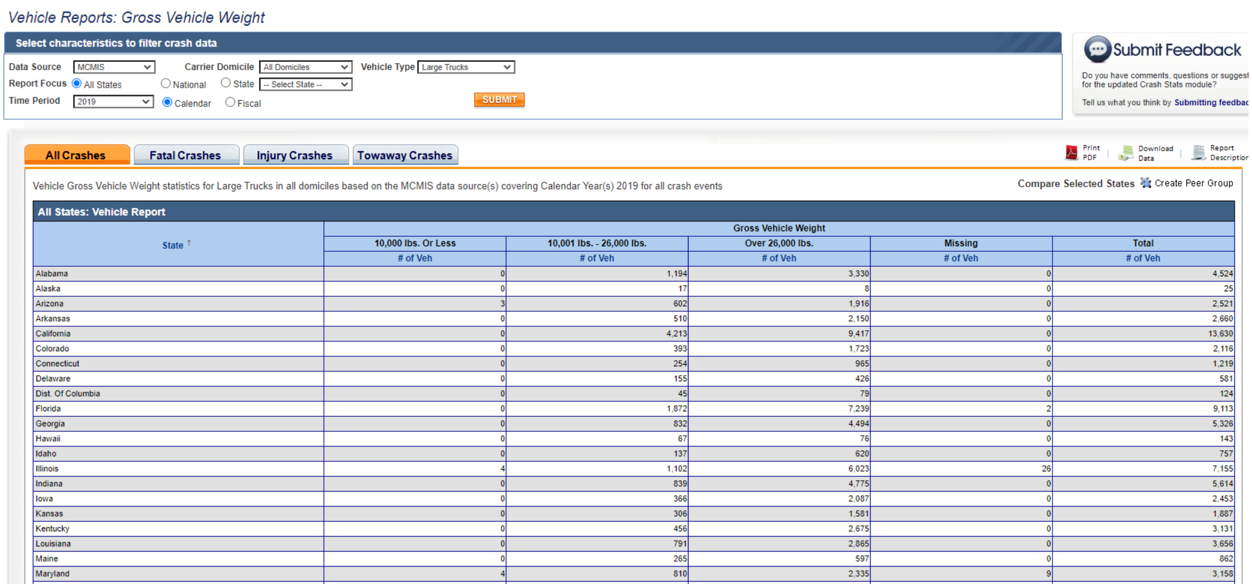
1. MCMIS includes crashes involving large trucks and buses (commercial motor vehicles) that are reported by states to the FMCSA through the SAFETYNET computer reporting system. It includes data elements collected on trucks and buses that meet the National Governors Association (NGA) recommended crash threshold. The FMCSA operates and maintains the MCMIS.

2. FARS is a census of crashes involving any motor vehicle on a trafficway, but only includes fatal crashes. FARS is maintained by the National Highway Traffic Safety Administration (NHTSA).

The primary data source for "all crash" data is the MCMIS.[32] The MCMIS crash reporting system data are based on state police crash reports electronically transmitted from the states to the FMCSA. Each crash file may contain multiple records for a crash. Separate reports are entered for each commercial motor vehicle involved in a crash. The MCMIS contains information on the safety fitness of commercial motor carriers (trucks and bus) and hazardous material (HM) shippers subject to the Federal Motor Carrier Safety Regulations (FMCSR) and the Hazardous Materials Regulations (HMR).

Crash statistics data can be filtered for large trucks using the A&I Crash Query Tool query tool. The crash statistics were accessed during the early months of 2022. There are also published crash reports for commercial motor vehicles (CMV) that provide additional information,[33] such as the *FMCSA Pocket Guide to Large Truck and Bus Statistics*.[34] The data query used readily available summary reports from the A&I Crash Query Tool. A snip of a summary report is shown in Figure 4-24, the "Vehicle Gross Vehicle Weight statistics for Large Trucks in all domiciles based on the MCMIS data source(s) covering Calendar Year(s) 2019 for all crash events." This query tab is for "All Crashes." Years 2017 and 2018 were done similarly. Note that column 4 provides state-specific data for "Gross Vehicle Weight – Over 26,000 lbs."

[31] Available at https://ai.fmcsa.dot.gov/default.aspx.
[32] MCMIS data are considered preliminary for 22 months to allow for changes. Therefore, 2019 was the most recent year of final data and the three years 2017-2019, which were also without the effect of the COVID-19 pandemic in 2020, were used as the basis for accident rates.
[33] Available at https://ai.fmcsa.dot.gov/CrashStatistics/CrashProfile.aspx.
[34] Available at https://www.fmcsa.dot.gov/safety/data-and-statistics/commercial-motor-vehicle-facts.



Vehicle Gross Vehicle Weight statistics for Large Trucks in all domiciles based on the MCMIS data source(s) covering Calendar Year(s) 2019 for all crash events

All States: Vehicle Report

State	Gross Vehicle Weight: 10,000 lbs. Or Less # of Veh	10,001 lbs. - 26,000 lbs. # of Veh	Over 26,000 lbs. # of Veh	Missing # of Veh	Total # of Veh
Alabama	0	1,194	3,330	0	4,524
Alaska	0	17	8	0	25
Arizona	3	602	1,916	0	2,521
Arkansas	0	510	2,150	0	2,660
California	0	4,213	9,417	0	13,630
Colorado	0	393	1,723	0	2,116
Connecticut	0	254	965	0	1,219
Delaware	0	155	426	0	581
Dist. Of Columbia	0	45	79	0	124
Florida	0	1,872	7,239	2	9,113
Georgia	0	832	4,494	0	5,326
Hawaii	0	67	76	0	143
Idaho	0	137	620	0	757
Illinois	4	1,102	6,023	26	7,155
Indiana	0	839	4,775	0	5,614
Iowa	0	366	2,087	0	2,453
Kansas	0	306	1,581	0	1,887
Kentucky	0	456	2,675	0	3,131
Louisiana	0	791	2,865	0	3,656
Maine	0	265	597	0	862
Maryland	4	810	2,335	9	3,158

Figure 4-24. FMCSA A&I Data Query Tool Showing State Accidents by GVW for Large Trucks, 2019

Table 4-13 shows the number of all crashes involving very large trucks on all state highways in the five western states of interest for 2017-2019, a total of 16,207 crash events both fatal and non-fatal. The accident rate on all state highways is determined using all state highways mileages in Table 4-12, a total of about 29 billion miles. For travel on all state highways the accident rates range from 3.97E-07 per mile in Utah to 7.27E-07 per mile in Colorado. Overall, for the five states the accident rate is 5.59E-07 per mile.

Table 4-13. Very Large Trucks Crashes and Accident Rate for all State Highways(1)

State	2017 Events	2018 Events	2019 Events	2017-2019 Total Events	All Highway Accident Rate per Mile
Colorado	1,383	1,571	1,723	4,677	7.27E-07
Idaho	742	614	620	1,976	5.90E-07
New Mexico	892	946	969	2,807	5.21E-07
Utah	1,224	1,313	1,155	3,692	3.97E-07
Wyoming	983	935	1,137	3,055	6.75E-07
ALL 5 STATES	5,224	5,379	5,604	16,207	5.59E-07

(1) Source: MCMIS, All States: Vehicle, GVW over 26,000 lb. All crash events so rate uses all state highways mileage.

These accident rates use crash events for >26,000 lb GVW trucks and state highway mileage for combination trucks. Events involving very large single-unit trucks are included, but mileage for very large single-unit trucks is not included. Therefore, these accident rates are likely conservative (tending to overestimate) compared to actual accident rates for all very large trucks.

4.5.2.3 Accident Rates for Very Large Trucks on Interstate Highways for the Route

The accident rates in Section 4.5.2.2 were determined for all accidents on state highways based on the available information in MCMIS and queried using the A&I tool. However, nearly the entire assumed route through the five western states is on interstate highways – limited access multi-lane highways that certainly have different miles traveled and likely have different accident rates per mile. Interstate only data are not available in MCMIS; however more detailed information is available for accidents involving a fatality. The relative occurrence of fatal accidents on interstate and non-interstate highways was assumed to be representative of the relative occurrence of non-fatal accidents on these highways and is used to estimate the all-accident rate on the interstate only. Detailed fatal crash data were queried to provide this information.

Fatal crash data are available from the NHTSA using the Fatality and Injury Reporting System Tool (FIRST). This query tool allows a user to construct customized queries from the FARS and also from the General Estimates System (GES) / Crash Report Sampling System (CRSS).[35]

Data were queried for fatal accidents on interstate and non-interstate highways involving very large trucks (>26,000 lb GVW) for 2017-2019. Very large trucks involved in fatal accidents totaled 240, while they were involved in 626 fatal accidents that occurred on all highways in the five states over the three years. Summarized results of the query (for the most harmful event [MHE]) are shown in Table 4-14.

Table 4-14. Very Large Truck Fatal Accidents 2017-2019 by Type of Highway

State	Interstate	Non-Interstate	Unknown	Total	Percentage Interstate of Total
Colorado	50	140	—	190	26.3%
Idaho	32	64	—	96	33.3%
New Mexico	91	91	2	184	49.5%
Utah	40	58	—	88	34.1%
Wyoming	37	31	—	68	54.4%
ALL 5 STATES	240	384	2	626	38.3%

The adjustment to estimate the all-accident rate on interstate highways for very large trucks uses the data presented earlier in this section. The number of crash events on all state highways in Table 4-13 is multiplied by the percentage of fatal accidents occurring on interstate highways in Table 4-14; overall, this assumes the number of all types of very large truck accidents on interstate highways is 38.3% of accidents occurring on all highways. This adjusted "number of crash events on interstate highways" is then divided by the very large truck mileage presented in Table 4-11 which totaled 15.9 billion miles during 2017-2019.

The estimated very large truck interstate accident rates and the data used to determine these rates are presented in Table 4-15. For additional information, the percentage of interstate miles of total state highways is also presented, showing how the 29 billion miles on all highways compares to 15.9 billion

[35] FIRST is available at https://cdan.dot.gov/query.

miles on interstate highway. In general, the all highway all-accident rates in Table 4-13 are reduced by about a factor of two in estimating the interstate highway all-accident rates in Table 4-15.

Table 4-15. Determination of Very Large Truck Interstate All-Accident Rates 2017-2019 using Fatal Accident Comparison[(1),(2)]

State	Total Events	% Fatal Accidents on Interstates	Interstate Miles × 10^6	*% Interstate Miles of all Highways Miles (for information only)*	Interstate All-Accident Rate per mile per year
Colorado	4,677	26.3%	3,393	*52.8%*	3.63E-07
Idaho	1,976	33.3%	1,996	*59.6%*	3.30E-07
New Mexico	2,807	49.5%	3,166	*58.8%*	4.39E-07
Utah	3,692	34.1%	4,566	*49.1%*	2.76E-07
Wyoming	3,055	54.4%	2,788	*61.6%*	5.96E-07
ALL 5 STATES	16,207	38.3%	15,909	*54.9%*	**3.90E-07**

(1) Total events from Table 4-13; % fatal accidents on interstates is from Table 4-14; interstate miles from Table 4-11; and % interstates miles of all highways for information only.
(2) The relative occurrence of fatal accidents on interstate and non-interstate highways was assumed to be representative of the relative occurrence of non-fatal accidents on these highways and is used to estimate the frequency for all accidents on the interstate only.

As for the all-highway rates, these estimated interstate accident rates are likely conservative (tending to overestimate) because mileage for very large single-unit trucks has not been included.

4.5.3 Development of the Likelihoods for TNPP Transportation Accidents

This section discusses development of the likelihoods for accidents that can occur during TNPP transportation and estimation of the frequencies for each bounding representative accident. Section 4.5.3.1 discusses development of the likelihood for TNPP transportation accidents that involve highway accidents including non-collisions (e.g., jackknifes) as well as collisions that involve external impact. Section 4.5.3.2 discusses development of the likelihood for TNPP transportation accidents that result in loss of containment events that do not involve external impacts to the transportation package.

4.5.3.1 Development of the Likelihoods for Highway Accidents

The section discusses development of accident frequencies for TNPP transportation highway accidents. The development of these accident frequencies is based on route specific data in combination with accident data from a nationwide accident (i.e., crash) database. This includes collisions that result in external impact to the TNPP package and non-collisions (e.g., jackknifes) that may induce minor impacts internal to the transportation package if an object becomes unrestrained.

The basis for developing the route specific accident likelihoods is the very large truck interstate accident data for the five states described in Section 4.5.2. However, these state-specific data only have sufficient resolution to determine frequencies for those classified as fatal accidents. Developing likelihood estimates for all accidents (i.e., fatal, injury-only, and property damage-only accidents) requires using national accident data statistics. The national accident datasets have a greater degree of resolution about the types of accidents that have occurred and are used to determine the percentage of accidents that can be attributed to certain accident types. These percentages are used in combination with the total accident frequency calculated for the five-state route from INL in Idaho to WSMR in New Mexico to determine an accident frequency of different accident types. Development of these accident frequencies is discussed in Section 4.5.3.1.1.

For certain kinds of accidents, GIS data for route-specific road hazards not specified in the accident data are used in combination with the total five state accident frequency to determine the accident frequencies. These accidents involve submersion into a body of water which could cause a criticality which is discussed in Section 4.5.3.1.2 and a drop to a lower elevation which is discussed in Section 4.5.3.1.3.

4.5.3.1.1 Highway Accident Frequencies from Accident Data

Nationwide all-accident data on interstate highway systems are available involving large trucks – single-unit and combination trucks greater than 10,000 lb GVW. The majority of these trucks are very heavy trucks, greater than 26,000 lb GVW. These nationwide large truck data are assumed to be representative of very large truck accidents along the five-state route. Nationwide large truck crash types and numbers of events from the NHTSA dataset for 2017-2019 are shown in Table 4-16 for all fatal and non-fatal accident types. These data are used to develop likelihoods for specific accident types in the following discussions. The dataset contains 40 different accident types (i.e., crash event descriptions) divided into four categories: unique (4), non-yielding (6), yielding (27), and “split” (3). Split categories reflect accident types for which additional resolution is needed as described below.

Three accident types are not included in Table 4-16 because they were determined not to be relevant to TNPP transportation based on their description.

Event data for the three “Split” categories are shown in Table 4-17. The first entry in Table 4-17 pertains to accidents involving the “motor vehicle in transport” which is 84% of the total events. This group consists of accidents involving a large truck in motion on an interstate that impacts any other motor vehicle on the roadway, including stalled, disabled, or abandoned vehicles. For the purposes of the PRA, these accidents are split between collisions with heavy or light vehicles. A “heavy vehicle” crash is considered to be a collision between a large truck and a combination truck or bus. The split between “heavy” and “light” collisions is determined using the 2017-2019 nationwide data on the fraction of miles for combination trucks plus buses compared to all vehicle miles during that period. Note that fraction of miles is 12.2% of all nationwide miles. This percentage does not include the mileage of very large single-unit trucks because that breakout is not available as part of single-unit truck mileage. However, accident events for very large single-unit trucks are also not included in the total accident rate per mile. Light vehicle collisions are considered to make up the remainder of the “motor vehicle in transport” events (i.e., 87.8%).

The next two entries in Table 4-17 pertain to the number of crashes with embankments or the ground and for the purposes of the PRA are split into non-yielding or yielding collisions. The percentage of non-yielding events for collision with the embankment and the ground are determined using the percentage of “hard rock” that exists along the transportation route as shown in Tables 4-6 and 4-7. The fraction of hard rock wayside surface value is 11.1% of the proposed transport route mileage. The remainder of the wayside surfaces are considered yielding.

Table 4-16. Nationwide Large Truck Interstate Accident Events from 2017-2019 Including Those Resulting in Fatality, Injury Only, and Property Damage Only Events(1)

Large Truck Crash Description	Category	Number of Events	Percent of Total
Rollover/Overturn	Unique	14,607	2.12%
Jackknife (Harmful to this Vehicle)	Unique	3,704	0.54%
Fire/Explosion	Unique	2,340	0.34%
Immersion or Partial Immersion	Unique	7	0.0010%
Concrete Traffic Barrier	Non-yielding	5,703	0.83%
Bridge Overhead Structure	Non-yielding	937	0.14%
Other Fixed Object	Non-yielding	686	0.10%
Bridge Rail (Includes Parapet)	Non-yielding	271	0.04%
Bridge Pier or Support	Non-yielding	172	0.03%
Unknown Fixed Object	Non-yielding	1	0.000%
Motor Vehicle in Transport	Split: heavy/light	581,859	84.59%
Embankment	Split: hard/other	1,794	0.26%
Ground	Split: hard/other	128	0.02%
Motor Vehicle In-Transport Strikes or is Struck by Cargo, Persons or Objects Set-in-Motion from/by Another Motor Vehicle In Transport	Yielding	37,077	5.39%
Guardrail Face	Yielding	7,598	1.10%
Parked Motor Vehicle (Not in Transport)	Yielding	5,005	0.73%
Other Object (Not Fixed)	Yielding	4,251	0.62%
Live Animal	Yielding	4,082	0.59%
Cable Barrier	Yielding	3,013	0.44%
Ditch	Yielding	2,673	0.39%
Tree (Standing Only)	Yielding	2,257	0.33%
Fence	Yielding	1,522	0.22%
Utility Pole/Light Support	Yielding	964	0.14%
Traffic Sign Support	Yielding	886	0.13%
Wall	Yielding	803	0.12%
Pedestrian	Yielding	763	0.11%
Post, Pole, or Other Supports	Yielding	718	0.10%
Object that had Fallen from Motor Vehicle In-Transport	Yielding	674	0.10%
Guardrail End	Yielding	669	0.10%
Impact Attenuator/Crash Cushion	Yielding	543	0.08%
Other Traffic Barrier	Yielding	493	0.07%
Other Non-Collision	Yielding	456	0.07%
Working Motor Vehicle	Yielding	304	0.04%
Curb	Yielding	300	0.04%
Unknown Object (Not Fixed)	Yielding	258	0.04%
Culvert	Yielding	256	0.04%
Pedalcyclist	Yielding	61	0.01%
Building	Yielding	51	0.01%
Non-Motorist on Personal Conveyance	Yielding	1	0.000%
Boulder	Yielding	1	0.000%
Total Occurrences, 2017-2019	—	687,888	—

(1) Source: NHTSA, FIRST data query. Three crash types are not included: cargo/equipment loss or shift (982), fell/jumped from vehicle (7), and reported as unknown (20).

Table 4-17. Accident Category Splits for Large Truck Crash Events on Interstate Highways

Large Truck Crash Description	Events	% Total	Heavy Vehicle Collision	Light Vehicle Collision	Basis for Heavy Vehicle
Motor Vehicle In Transport	581,859	84.59%	70,971	510,888	12.2%, combo trucks + buses miles, nationwide(1)
—	—	—	Non-yielding	Yielding	Basis for Non-yielding
Embankment	1,794	0.26%	199	1,595	11.1% hard rock along route(2)
Ground	128	0.02%	14	114	11.1% hard rock along route(2)

(1) Combination trucks and buses are considered heavy vehicles (>26,000 lb GVW). Does not include large single-unit trucks, where information is not available.
(2) Based on information provided in Table 4-6 and Table 4-7.

Using the data described above, the large truck accident events are organized into the eight categories shown in Table 4-18 and the percentage of accidents that occur in each accident category is determined (e.g., collisions with light vehicles is 74.3% of all crashes).

The nationwide interstate accident rate per mile for large trucks is determined by dividing the number of events in each category by the nationwide mileage of large trucks on nationwide interstate highways and is shown in Table 4-18 only for comparison to the interstate accident rates for the TNPP transport route. The overall accident rate for large trucks on interstates nationwide is 1.81E-06 per mile and the accident rates for all eight accident categories sum to this overall rate. It is important to note this is the nationwide interstate all-accident rate for large trucks, which are combination trucks and all single-unit trucks greater than 10,000 lb GVW, and not for very heavy trucks greater than 26,000 lb GVW.

Table 4-18. Nationwide Large Truck Interstate Accident Likelihoods for Key Accident Categories(1)

Accident Category	Number of Events	Percentage of Total	Accident Rate per Mile
Light vehicle collision	510,888	74.3%	1.35E-06
Heavy vehicle collision	70,971	10.3%	1.87E-07
Yielding impacts	77,388	11.3%	2.04E-07
Non-yielding impacts	7,983	1.2%	2.10E-08
Rollover/overturn	14,607	2.1%	3.85E-08
Jackknife	3,704	0.34%	9.76E-09
Fire/explosion	2,340	0.54%	6.17E-09
Immersion/partial immersion	7	0.0010%	1.85E-11
Total	687,888	100.0%	1.81E-06

(1) Accident categories based on information in Table 4-16 and Table 4-17.

The frequencies for categories of very large truck accidents on the interstate highways of the TNPP transport route are determined starting with the very large truck accident rate of 3.90E-07 per mile developed in Section 4.5.2.3 and presented in Table 4-15. To obtain each accident category identified in Table 4-18, the total accident frequency is multiplied by the percent contribution the category makes to the total number of accidents. These nationwide accident proportions are assumed to be applicable for very large trucks in the five western states. The estimated accident frequencies for each of the eight categories for very large trucks in the five states are shown in Table 4-19 for the entire 1,289-mile transport route.

Table 4-19. Very Large Truck 5-State Interstate Accident Likelihoods(1)

Accident Category	Accident Rate per Mile	Percentage of Total	Accident Frequency for a 1,289-mile route
Light vehicle collision	2.90E-07	74.3%	3.74E-04
Heavy vehicle collision	4.03E-08	10.3%	5.19E-05
Yielding impacts	4.39E-08	11.3%	5.66E-05
Non-yielding impacts	4.53E-09	1.2%	5.84E-06
Rollover/overturn	8.29E-09	2.1%	1.07E-05
Jackknife	1.33E-09	0.34%	1.71E-06
Fire/explosion	2.10E-09	0.54%	2.71E-06
Immersion/partial immersion	3.97E-12	0.0010%	5.12E-09
Total	3.90E-07	100.0%	5.03E-04

(1) Accident rates based on the very large truck total accident frequency from Table 4-15 and percentage information from Table 4-18.

Additional queries of the MCMIS database were performed support development of frequencies for bounding representative accidents identified in Table 4-5 and discussed in Section 4.4.3.2.9. Given bounding accidents are representative of a subset of the TNPP transportation accidents identified by the hazard analysis presented in Table 4-4 and discussed in Section 4.4.3.1, the frequencies of contributors were determined and summed together to calculate the accident frequency for each bounding representative accident. To determine the frequency of certain bounding representative accidents, frequencies for applicable accident categories in Table 4-19 were required in combination with additional breakout of the accident data. This includes breakout of accident frequencies based on the cause of fire/explosion accidents and the level of impact of collisions (i.e., hard, medium, light). Nationwide percentages were determined for each these breakouts and then applied to the five-state accident frequencies to estimate the cited accident frequencies.

Accidents that involve fire can be initiated by a non-fire event or the fire itself can be the initiating event (i.e., a fire-only event). The MCMIS database was queried for all nationwide large truck accidents on interstate highways, including fatalities, injury only accidents, and property damage only accidents, from 2017-2019. Fire events were identified by querying for the MHE and the first harmful event (FHE). As shown in Table 4-18, a total of 2,340 fire/explosion accidents occurred involving large trucks during this period. Of these accidents, FHE was specified as some other type of accident for 25.4% of the events where fire/explosion was the MHE. More common were fire-only events where fire/explosion was both the FHE and MHE. Also, if no FHE was specified in the data (i.e., only an MHE was specified), the accident was assumed to be only a fire/explosion event. Based on these criteria, fire-only events occurred in 74.6% of the fire/explosion events. These percentages are used to develop accident frequencies for BRA 2, BRA 5H, BRA 5M, and BRA 6, all of which involve fire.

Additional fire/explosion information was needed for BRA 6, which involves collision with a tanker truck carrying flammable liquids. The accident databases do not breakout accidents involving tanker trucks from other large trucks. However, there is available information on miles driven by tanker trucks for each state in the U.S Census Bureau *2002 Economic Census, Vehicle Inventory and Use Survey* (U.S. Census Bureau 2004a, 2004b, 2004c, 2004d, 2004e). The percentage of flammable liquid tankers was estimated for the five states on the assumed route by using total miles traveled by tanker truck-tractors (combination trucks) divided by the total number of miles by all heavy-heavy trucks (>26,000 lb) in 2002. Tanker trucks were estimated to comprise 10.3% of very heavy trucks. Use of this percentage leads to a somewhat conservative estimation of the frequency of collisions with a tanker carrying flammable liquids because it includes all tankers. These percentages are used to develop the accident frequencies for BRA 6.

The BRA 4M and BRA 4L are "less than hard" impacts further differentiated as medium and light impacts as defined in Table 4-5. The MCMIS dataset does not support breakout of data into these categories, so estimates were based on further evaluating the accidents categorized as "Motor Vehicle in Transport," shown in Table 4-17. For this accident category, accidents that resulted in fatality or injury only were assumed to be the result of medium impact; this was 26.0% of the 581,859 accident events considered. Light impacts were assumed for property damage only accidents, accounting for 76.0% of the transport accidents. These percentages are used to develop the accident frequencies for BRA 4M and BRA 4L. These bounding representative accidents are limited to collisions with light vehicles.

To estimate the final accident frequencies for BRA 5H and BRA 5M, further breakout by hard and medium impacts was required. Hard impacts are heavy vehicle collisions, non-yielding impacts, rollovers/overturns, and drops. Medium impacts are other crashes, including light vehicle collisions, yielding impacts, and jackknifes.

4.5.3.1.2 Frequency of Highway Accidents that Could Result in a Criticality Event

The accidents for which frequencies need to be developed to support estimation of applicable bounding representative accidents that involve potential criticality include:

- Drop into a body of water that submerges the reactor vessel resulting in criticality.
- Control rod withdrawal caused by impact during road accident that results in criticality.

The first accident is a drop into a body of water that submerges the reactor vessel which is about 5 ft in diameter without considering the empty shield tank which surrounds it which could be ruptured in the accident. This accident is a subset of accidents that result in a drop to a lower elevation; therefore, the likelihood of this criticality accident is subset of the likelihood that the transport vehicle drops to a lower elevation. Two approaches were taken in developing a frequency for this accident which have different advantages and yielded somewhat different results. Therefore, both approaches are presented, one using national truck accident data and the other using GIS data combined with the five state truck data for the assumed route.

In one approach, the GIS hazard data and analysis presented in Section 4.5.1.4 and truck accident frequency data and analysis presented in Section 4.5.2 were used to estimate a frequency for this accident. In Table 4-9 of Section 4.5.1.4, the results of the GIS analysis show that there are 301, 100 ft long segments of the route where, if an accident occurred and the transport vehicle left the road, it could end up in a body of water deep enough to submerge the reactor vessel. These segments represent 28, 612 ft of the route that bypasses Denver for a total of about 5.4 miles (whether the Denver bypass is used or not). This distance is about 0.42% of the 1,289-mile route. The overall accident frequency for the route is estimated by multiplying the total (fatal and non-fatal) accident rate for the five states along the route by total miles of the route (i.e., 3.9E-07 per mile per year x 1,289 miles) which yields a frequency of 5.03E-04 per year. Given that only 0.42% of the route can lead to the submersion accident and that the accidents are randomly distributed, the estimated frequency of the submersion accident is 2.11E-06 per year assuming one shipment in a year. This estimation is likely conservative because it is assumed that an accident that occurs in a 100-ft segment of the route near a sufficiently deep body of water where there is sufficient slope that the reactor vessel will always slide or roll into the body of water.

In the other approach for the flooded criticality accident, the proportion of "immersion/partial immersion" events to total number of large truck interstate accidents nationwide was developed as presented in Table 4-18. This ratio was then multiplied by the route specific five state interstate accident rate times the number of miles in the route in Table 4-19 to get an accident frequency of 5.12E-09 per year assuming one shipment in a year. However, the national dataset indicates that there were only seven immersion events in the 2017 to 2019 timeframe designated as the MHE and each involved a fatality. Submersion events designated as FHEs included collision with a motor vehicle in transport and collision with a guardrail face. It is not known whether there are other immersion events sufficient to submerge the reactor vessel that are not included in this count, and the estimated accident frequency may be non-conservative. Therefore, both the GIS estimate of 2.11E-06 per year frequency estimate and the estimate developed using data are described here and are used as discussed in Section 4.7.11.

The second accident concerns an impact so hard that the control rods are withdrawn against the restraining and locking mechanism. Therefore, the likelihood of this accident is a subset of the likelihood of the hard impact accidents discussed above (i.e., impact with heavy vehicles, impacts with unyielding objects, and rollovers).

4.5.3.1.3 Frequency of Highway Accidents that Could Result in a Drop to a Lower Elevation

An accident that is included in one of the bounding representative accidents and cannot be derived from truck accident data alone involves a drop to a lower elevation. Correspondingly, a specific hazard of the route topography are locations where there is a drop to a lower elevation surface just off the roadway. If a truck has an accident in these locations (e.g., on a bridge or overpass, or near a steep embankment) and leaves the road, then significantly more damage could occur to the TNPP package if the vehicle drops to a lower elevation. The GIS hazard data and analysis presented in Section 4.5.1.5 and truck accident frequency data and analysis presented in Section 4.5.2 were used to estimate a frequency for this accident.

As described in Section 4.5.1.5, GIS analysis identified 318, 100 ft segments of the route that were considered sufficiently steep that a truck would drop (or roll or slide) to a lower elevation if it left the road as a result of an accident. The total length of the assumed route where this hazard exists translates to 31,800 ft of the route or 5.9 miles. This distance is about 0.46% of the 1,289-mile assumed route.

The overall accident frequency for the route is estimated by multiplying the total (fatal and non-fatal) accident rate for the five states along the route by total miles of the route (i.e., 3.9E-07 per mile per year x 1,289 miles) which yields a frequency of 5.03E-04 per year. Given that only 0.46% of the route can lead to a drop to lower elevation accident and that the accidents are randomly distributed, the estimated frequency of this accident is 2.3E-06 per year assuming one shipment in a year.

4.5.3.2 Development of the Likelihoods for Non-Impact Accidents

This section discusses development of accident frequencies for non-impact vehicle accident scenarios that can occur during transport, specifically fire-only scenarios, non-impact loss of package containment, and increase in radiological dose exposure time events.

Some of the accidents identified in Table 4-4 are fire-only accident scenarios that do not involve mechanical impacts associated with a highway accident. The development of the frequencies for these accidents is provided in Section 4.5.3.2.1. Additionally, there are several lower energy accident scenarios in which the package containment fails but does not involve mechanical impacts associated with highway accidents. One set of these types of transportation accidents concerns a breach or loss of the reactor containment boundary when the system is not pressurized, while another set concerns a breach or loss of the reactor containment boundary when the system is pressurized. A third set concerns a breach of non-reactor containment boundary components, such as in the contaminated Shield Tank. The development of the frequencies for these accidents is provided in Section 4.5.3.2.2.

There is also a set of event scenarios in which technical or logistical difficulties during transport cause a lengthened transport time and an increased exposure of workers to radiation. Discussion of these frequencies is provided in Section 4.5.3.2.3.

4.5.3.2.1 Non-Impact Fire-Only Accident Frequency Development

Non-impact, fire-only TNPP transportation accidents can be of two types: (1) those that originate from inside the transport container, and (2) those that originate from outside the transport container, namely a diesel fuel fire. Truck fire data exist to support estimation of the frequency of diesel fuel fires, but does not exist to base estimation of the frequency of fires that would originate inside the transport container. Fires that occur inside the transport container are due to hazardous conditions inside the container rather than hazardous conditions associated with the truck such as the diesel fuel and hot engine temperatures.

4.5.3.2.1.1 Fires that Originate External to the TNPP Transportation Package

Based on the data presented in Section 4.5, the accident rate is 3.90E-07 per mile for very large truck (greater than 26,000 lb GVW) accidents on all highways for the five states through which the shipment route passes (based on data from MCMIS). Of these accidents, 27.5% did not involve a collision with another vehicle or object and 2.7% of these non-collision accidents resulted in a fire or explosion. The conditional probability of a fire is therefore assumed to be 0.74% (0.275 × 0.027). The likelihood of the TNPP transportation package being involved in a non-collision accident resulting in a fire for a one-way trip of 1,289 miles is therefore estimated to be 3.4E-06 per shipment (3.9E-07 per mile × 1,289 miles × 0.0074).

This result is about an order of magnitude less than the value of 2.8E-05 per shipment obtained using the data from NUREG-2125. In this report, the average accident rate for large trucks is 3.19E-06 per mile based on the average accident rates from 1991 through 2007 for the entire United States – 12.6% of these accidents did not involve collision with another vehicle or object and 5% of the non-collision accidents resulted in a fire or explosion.

4.5.3.2.1.2 Fires that Originate Internal to the TNPP Transportation Package

Based on available information, very little combustible or flammable material will be contained within the Reactor Module. The primary combustibles in this module are cable insulation and lubricants for motors (BWXT 2022[36]). Cable insulation is associated with reactor electrical/instrumentation and control components that are located at the fore end of the module. However, these electrical circuits will not be energized during transportation of the module. A passive venting system is planned to be utilized to cool the Reactor Module during transportation (BWXT 2022[37]). Other cabling is associated with heat-detection devices that will connect to the fire detection system in the Control Module when the TNPP is reassembled. These devices are used to measure the surrounding air and actuate if the surrounding air exceeds a pre-determined air temperature, alerting operators to the potential of a fire in the Reactor Module. Additionally, there is cabling associated with radiation monitors, and temperature and pressure transducers to monitor the internal environmental conditions inside of the module enclosure. This instrumentation is localized at the aft and fore ends of the Reactor Module, in the lowest radiation area possible (BWXT 2022[38]). It is unclear which, if any, of these systems will be energized during transportation. Never-the-less, energized electrical/instrumentation and control systems that could exist in the Reactor Module container during transport may include systems to support such functions as lighting, parameter monitoring (e.g., radiation and heat monitoring), and ventilation and cooling. Specifically, as currently planned, remote monitoring of the Reactor Module systems will be implemented to provide real-time health diagnostics (BWXT 2022[39]). Therefore, it is assumed for the TNPP PRA that there will be energized electrical components in the Reactor Module during transport. However, all cabling in the Reactor Module will be inserted and protected in electrical rated conduit per the National Electrical Code (NFPA 70E).

With regards to lubricants, the Reactor Module contains a handful of small electrical motors. An example are the motors used to operate the reactor control rod drives. While none of these motors are operable, or active, during transport operations, they never-the-less do contain lubricants that are flammable and therefore contribute to the combustible loading within the module.

As stated above, the frequency of fires that occur inside the transport container cannot be based on truck accident data. However, the frequency of these fires could be estimated by using surrogate fire ignition frequencies for a comparable situation. One source of such information is the NRC guidance in NUREG/CR-6850, *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (NRC and EPRI 2005) for performing fire PRA at nuclear power plants and the accompanying fire ignition fire frequencies for different ignition sources provided in NUREG-2169, *Nuclear Power Plant Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database* (NRC and EPRI 2015). The situation in the Reactor Module transport container where the potential fuel load is located is comparable to fire areas in a nuclear power plant in the sense that nuclear safety is vital and so quality control for equipment and procedures should be high. NUREG-2169 presents the fire ignition frequencies for 37 different fire sources (called fire ignition bins), from which the applicable ignition frequencies can be identified.

[36] BWXT Final Design Report, Appendix IV, ATL-TECR-109977 – "MNPP Facility Fire Hazards Analysis".
[37] BWXT Final Design Report, Appendix I, ATL-PLAN-110124 – "Transportation Plan".
[38] BWXT Final Design Report, Section 2.3.1.6.3.
[39] BWXT Final Design Report, Appendix I, ATL-PLAN-110124 – "Transportation Plan".

The fire frequency bins presented in NUREG-2169, Table 4-4 that may be applicable to the Reactor Module transport container during transport are self-ignited cable fires, electric motor fires, junction box fires, electrical cabinet fires, and Main Control Board (instrument and control) fires. Regarding the latter, the TNPP module containing the equivalent of the Main Control Board is the Control Module and so is not applicable in this assessment (additionally, the Control Module instrumentation and control systems are not functional or energized during TNPP transportation and is also not contaminated). The fire frequency associated with the other fire frequency bins identified in Table 4-4 of NUREG-2169 do not apply because those fire sources do not exist in a Reactor Module container during reactor operation or during module transport. For example, because the Reactor Module is not designed to be occupied by plant personnel during normal operation, combustible control programs are assumed to preclude the presence of transients in the Reactor Module during both operation and shipment and so transient fires are not postulated. The fire frequencies for the cited fire frequency bins need to be adjusted to be applicable to a transport container because the number of fire ignition sources in the transport container is substantially less than in a nuclear power plant.

Control, power, and instrumentation cabling used throughout the TNPP are rated for compliance with Institute of Electrical and Electronics Engineers (IEEE) 383, "IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities," (IEEE 2012) and IEEE 1202, "IEEE Standard for Flame Testing of Cables for Use in Cable Tray in Industrial and Commercial Occupancies" (IEEE 2014) (BWXT 2022[40]). Based on IEEE 383, cables that meet the IEEE 1202 flame test meet the flame test requirements of IEEE 383. Per NUREG/CR-6850, Volume 2, Appendix R and associated NRC guidance for modeling self-ignited cable fires in Frequently Asked Question (FAQ) 13-0005, *Cable Fires Special Cases: Self Ignited and Caused by Welding and Cutting* (Hamzehee 2013), self-ignited cable fires are not to be postulated in rooms or fire areas containing qualified cables only (i.e., all cables are qualified per IEEE 383). Hence, based on the design specification for the Reactor Module, self-ignited cable fires are screened as an ignition source during transportation of the Reactor Module. Also, per NUREG/CR-6850, Volume 2, Section 8.5.1.2, cables in conduit are considered potential damage targets, but not ignition targets. Cables in conduit will not contribute to fire growth and spread.

Generally, a junction box is defined as a fully enclosed metal box containing terminals for joining or splicing cables. FAQ 13-0006 (*Modeling Junction Box Scenarios in a Fire PRA* [NRC 2013b]) states:

> Junction box fires generally begin as a relatively small fire or arc within the electrical enclosure. In most cases, these fires do not generate enough heat to be self-sustaining and will self-extinguish prior to spreading outside of the junction box. This is mostly due to the enclosed configuration of the box. In effect, this approach assumes that the zone of influence for these fires is equal to the junction box only. Consequently, the proposed approach provides a method for screening and analysis of such fires without the need for detailed fire growth, damage and suppression modeling.

FAQ 13-0006 further explains that junction box frequencies should include all junction boxes regardless of cable insulation because these fires are not influenced by the cable insulation or jacket type.

Regarding junction box fires, the mean plant-wide fire frequency for this bin from NUREG-2169 is 3.61E-03 per reactor-year. NUREG/CR-6850 and FAQ 13-0006 provide two methods for apportioning this plant-wide frequency to individual fire zones within the plant: (1) the ratio of the number of

[40] BWXT Final Design Report, Appendix IV, ATL-TECR-109977 – "MNPP Facility Fire Hazards Analysis".

junction boxes in the fire zone to the total number of junction boxes in the plant, and (2) the ratio of the cable loading in the fire zone to the total cable loading in the plant. Similarly, the same metrics are used in this report to scale the nuclear power plant junction box fire ignition frequency to develop an applicable frequency for the Reactor Module during transportation. However, since the number of energized junction boxes and cable loading is not available for the Reactor Module, it is assumed, for the purposes of this report, the Reactor Module contains one or two energized junction boxes during transportation and has an associated cable loading, or cable insulation mass, of one to two hundred pounds. A typical nuclear power plant from which the NUREG-2169 fire frequencies were developed contains a few hundred junction boxes (typically at least 150) and a few hundred thousand pounds of cable insulation (typically at least 200,000 lb).[41] Based on this, a conservative scaling factor of 0.001 is assumed. In addition, compared to a typical reactor year of well over 300 days, a Reactor Module shipment is assumed to require just three to four days. Based on this, a shipment duration fraction is assumed to be 0.015. The estimated fire ignition frequency for junction boxes is about 5E-08 per shipment for the Reactor Module. As explained above, junction box fires do not generate enough heat to be self-sustaining and so do not need to be propagated outside of the junction box, thereby limiting damage to the loss of the functions provided by the energized cables in the junction boxes.

As explained above, a passive venting system is planned to be utilized to cool the Reactor Module during transportation. In addition, electric motors used during reactor operations (e.g., control rod drive motors) will not be energized during transportation of the Reactor Module, and so are not potential sources of fire ignition during shipments. However, the design features of the Reactor Module during transportation are not yet fully developed and so this analysis assumes the Reactor Module may have an active ventilation system (e.g., electrically-driven fan). Regarding ventilation system fires, the mean plant-wide fire frequency for this bin from NUREG-2169 is 1.64E-02 per reactor-year, of which 95% are electric systems (per NUREG/CR-6850). A typical nuclear power plant from which the NUREG-2169 fire frequencies were developed contains a few dozen ventilation systems (typically at least 50 subsystems).[23] Based on this, a conservative scaling factor of 0.02 is assumed. Considering the transport time as described above, the estimated fire ignition frequency for a ventilation system (i.e., electrically-driven fan) is about 5E-06 per shipment for the Reactor Module. If all or most electrical cables and wiring are in conduit and there are no other combustibles, then propagation of an electrical motor fire may not need to be considered, thereby limiting damage to the loss of the ventilation function.

As discussed above, there may be an active parameter monitoring system and/or an active ventilation system during transport of the Reactor Module. It is expected that these would be powered and remotely monitored from outside of the module. Regarding instrument and control board fires, the mean plant-wide fire ignition frequency for the electrical cabinets bin from NUREG-2169 is 3.0E-02 per reactor-year. A typical nuclear power plant from which the NUREG-2169 fire frequencies were developed contains several hundred electrical cabinets (typically at least 500).[23] Based on this, a conservative scaling factor of 0.002 is assumed. Considering the transport time as described above, the estimated fire ignition frequency for electric cabinets is about 9E-07 per year. If all or most electrical cables and wiring are in conduit and there are no other combustibles, then propagation of an electrical cabinet fire may not need to be considered, thereby limiting damage to the loss of the applicable functions.

[41] Engineering judgment based on the extensive experience of PNNL staff reviewing nuclear power plant risk-informed applications, including associated fire PRAs, for the NRC.

Accordingly, the total conservative estimate of fire ignition frequency is about 6.1E-06 per year. If all or most electrical cables and wiring are in conduit and there are no other combustibles, then propagation from any of the fire sources described above might be screened and just the consequences from the loss of the applicable functions would need to be considered. Also, if there is an active fire protection system in the Reactor Module during transport, then the risk contribution from these fire sources might be screened given the extremely low likelihood that a fire occurs and propagates to the extent that the Reactor Module is damaged.

4.5.3.2.2 Non-Highway Package Containment Failure Accident Frequency Development

As described in Section 4.5.3, there are several package containment failure accident scenarios that did not involve highway accidents or fires. These accident scenarios are organized into three sets and are presented below along with the possible causes for these accidents (initiating events). Scenarios 1 through 5 involve failure of the non-pressurized reactor containment boundary, Scenarios 6 and 7 involve failure of the pressurized reactor containment boundary, and Scenarios 8 through 11 address scenarios that involve breach of contaminated components of the Reactor Module other than the reactor containment boundary.

Non-pressurized reactor containment boundary failure caused by:

1. Random failure of system components,
2. Vibration and shock from over-the-road travel,
3. Human error in packaging the system,
4. Human error during TNPP disassembly leading to undetected latent failures in reactor containment boundary, and
5. Extreme cold that fails containment.

Pressurized reactor containment boundary failure caused by residual heat build-up in combination with:

6. Mechanical impact on vents or other heat transfer pathway elements that decreases heat removal from the reactor containment boundary and pressurizes containment in combination with containment failure, and
7. High ambient air temperature that in combination with the residual decay heat pressurizes the reactor containment boundary in combination with containment failure.

Package elements failure other than the reactor containment boundary caused by:

8. Pressurization due to radiolysis of hydrogenous material (e.g., Shield Tank not fully drained) and possible hydrogen accumulation and ignition,
9. Pressurization caused by loss of ventilation or high ambient air temperatures,
10. Containment failure caused by random failures and/or vibration, and

11. Containment failure due to a hailstorm that causes general severe vibration.

4.5.3.2.2.1 Accident Frequency for Breach of a Non-Pressurized Reactor Containment Boundary

The following discusses development of the initiating event frequencies for the five accident scenarios identified in the previous section pertaining to loss of containment in a non-pressurized reactor containment boundary. Because the IHX is located in a separate module (i.e., the IHX Module) from the Reactor Module, the Primary Cooling system is dismantled for transport of the TNPP. Hence, containment isolation features or devices (i.e., Grayloc® bolted blind flange [BWXT 2022[42]]) will be installed on the Primary Cooling system inlet/outlet piping after its dismantlement. A HMIS will eventually be included in the TNPP design that may contribute to the ability to mitigate or even prevent a significant release from this set of accidents if operators respond in time. However, this system has not yet been designed and its applicability to these scenarios is unclear, so it is not credited in the estimation of the accident scenario frequencies discussed here.

Regarding random containment failure (e.g., failure of a seal, connection, or joint), the failure probability can be best estimated once the details of the containment features are fully known. However, the containment feature is likely to be a piping fitting or connector that is leak tight and practical to install and uninstall. The DOD Reliability Analysis Center provides failure rates for mechanical piping fittings and disconnects that might be used. The *Nonelectronic Parts Reliability Data* handbook (Denson et al. 1991) presents a failure rate of 1.1E-06 per hour for the “generalized ground operations and test conditions.” The handbook shows that failure rates for quick disconnects and fittings associated with mobile equipment can be an order of magnitude higher than ground-based equipment. However, this potential underestimation of failure frequency might be considered to be offset by: (1) the disconnects used as containment devices for reactor containment boundary that will have low pressure, and (2) the cited failure rate is for general equipment, whereas the fittings used to seal the system are expected to be high quality nuclear grade seals, like hose that might be used on an actual spent nuclear fuel transportation package. The fitting failure rate from the DOD handbook for a general failure mode is judged to a basis for an estimate until more design details are known. If the transport is conservatively assumed to take 100 hours, then the accident frequency for this failure is estimated to be about 1E-04 per shipment.

Regarding the impact of vibration and shock from over-the-road travel on the failure of the reactor containment boundary, it is difficult to find explicitly applicable failure information. However, as mentioned above, the *Nonelectronic Parts Reliability Data* handbook (Denson et al. 1991) indicates that the failure of fittings and disconnects associated with mobile equipment is about one order of magnitude higher than for ground-based environment. Accordingly, one way to estimate the impact of vibration and shock from over-the-road travel is to assume that the random failure of fittings or disconnects are increased by one order of magnitude. This approach is judged to provide a reasonable conservative basis for an estimate until more details are known. Using this approach, it may be beneficial to define a bounding accident that includes random failure and vibration and shock together. If the transport is conservatively assumed to take 100 hours, then the failure rate is estimated to be as about 1E-03 per shipment.

Regarding human error in packaging the reactor containment boundary, a conservative estimate can be generated based on simplified HRA. HRA modeling guidance used by NRC provides an approach for

[42] BWXT Final Design Report, pages 2-13 and 7-36.

estimating the failure probability of operator actions at nuclear power plants in NUREG/CR-6883 (Gertman et al. 2005). The SPAR-H approach uses a base Human Error Probability (HEP) of 1E-02 for execution of an action and 1E-03 for diagnosis for the need for an action. The impact of PSFs such as the presence of a stressor or lack of procedure or training can further increase the failure probability from the base HEP if they can reflect conditions more challenging than nominal conditions. If it is assumed that failure of a single execution step could fail, the containment feature that is used to seal the reactor containment boundary after the modules have been disassembled, then failure probability might be estimated to be 1E-02 with no consideration PSFs. However, it is likely the packaging will be checked or inspected before transport for which another failure would need to occur for the original failure to remain undetected. SPAR-H lists the base HEP for failure to diagnose a problem at 1E-03. Accordingly, the total failure probability leading to a packaging error that fails containment during transport of the Reactor Module could be about 1E-05 assuming the two errors are separate independent mistakes without consideration of any PSFs. However, it is unlikely that certain PSFs such as stressors associated with environmental conditions or complexity of the task would not apply. Based on the possible applicable PSFs, the impact of PSFs could increase the base HEPs by an order of magnitude even if procedures and training are nominal. This approach is judged to provide a basis for an estimate until more details are known. Accordingly, the estimated failure probability associated with human error in packaging the TNPP that could lead to undetected containment failure of the reactor containment boundary during transport is about 1E-04 per shipment.

Regarding human error during TNPP disassembly leading to undetected latent failures in the reactor containment boundary, an estimate can be generated based on simplified HRA modeling in the same way it can be done as described above for estimating the probability of packaging error. If it assumed that failure of a single execution step during disassembly could lead to undetected latent failures in the reactor containment boundary, and there is a separate independent failure to diagnose the problem during a check or inspection, then a failure probability can be determined by multiplying these HEPs together and adjusting the sum using the same assumption about the impact of PSFs as described above. This approach is judged to provide a basis for an estimate until more details are known. Accordingly, the estimated failure probability associated with human error during TNPP disassembly leading to undetected latent failures in the reactor containment boundary that could lead to undetected containment failure during transport of the Reactor Module is about 1E-04 per shipment.

Regarding extreme cold that fails containment, it is difficult to find explicitly applicable failure information. However, the random failure rate of fittings and disconnects from the *Nonelectronic Parts Reliability Data* handbook (Denson et al. 1991) and the distribution of mechanisms may provide some basis for an estimate. The DOD Reliability Analysis Center provides the distribution of failure modes and mechanisms for failures including fitting failures in *Failure Mode/Mechanism Distributions 1997* (Crowell et al. 1997). The reference does not indicate that extreme cold is a failure mode with a measurable contribution, but does show that 2.3% of fitting failures are from being out of tolerance (out of specifications). This is separate failure mode from deterioration, wear-out, breaking, or improper adjustment. The TNPP would not likely be transported if the weather conditions were so cold that tolerances associated with containment features could be exceeded. However, an assumption might be made that extremely cold weather exacerbates failure of the fittings by being out of tolerance, though the temperature may be within the design specification. Given that the failure mode is just 2% of the total failure likelihood, the failure rate of a failure from this failure mode using the failure rate cited above (Crowell et al. 1997) would be 2E-08 per hour. However, if it assumed that the extreme cold weather increases the failure rate by an order of magnitude and the transport is conservatively assumed to take 100 hours, then the failure rate is estimated to be as about 2E-05 per shipment. This approach is judged to provide a basis for an estimate until more details are known.

4.5.3.2.2.2 Accident Frequency for Breach of a Pressurized Reactor Containment Boundary

The following discusses developing the initiating events frequency for the two accident scenarios identified in Section 4.5.3.2.2 pertaining to breach of a pressurized reactor containment boundary. For these accidents it is assumed that residual heat decay pressurizes the reactor containment boundary and that other factors increase the pressure beyond expected levels in combination with containment failure. The containment isolation features described previously are installed on both the reactor and intermediate heat exchanger portions of the Primary Cooling system after it is dismantled in preparation for the shipment of the various modules. It is acknowledged that a HMIS will eventually be included in the TNPP design, that may contribute the ability to mitigate or even prevent a significant release from this set of accidents, if operators respond in time. However, this system has not yet been designed and its applicability to these scenarios is unclear, so it is not credited in the estimation of the accident scenario frequencies discussed in this section.

Regarding impact on vents or other heat transfer pathway elements that decreases heat removal to the extent the reactor containment boundary pressurizes during transport of the Reactor Module, it is difficult to find explicitly applicable failure information. This failure might occur during transport because load restraints loosen or fail during transport allowing the load to shift in a way that damages vents or other elements of the heat transfer pathway (note, the location, quantity, and size of vents for passive cooling have not been determined (BWXT 2022[43]). This in turn results in heat buildup and potential failure of containment through the fittings applied to reactor containment boundary. Degradation of the heat transfer pathway could also occur during preparation of the Reactor Module for transport and remain undetected because the failure is not discovered during inspection before transport. The *Nonelectronic Parts Reliability Data* handbook (Denson et al. 1991) indicates that the failure of restrainers is about 15E-06 per hour. Assuming that failure of the load restraints results in shifting of the load and damage to heat vents or other elements of the heat transfer pathway that in turn leads to additional pressure in the reactor containment boundary provides the basis for estimating a failure rate. If the transport is assumed to take 100 hours, then the failure rate is estimated to be about 1.5E-03 per shipment. As identified above, damage to the passive heat transfer pathway could also occur during the normal process for preparing the Reactor Module for transport. Given that this process consists of dismantling and moving heavy objects (e.g., Primary Cooling system piping) in a relatively small area it might be conservatively assumed that such an error is not uncommon and occurs at a probability of 1E-01. However, it is likely the heat transfer pathway will be checked or inspected before transport. As described above, SPAR-H lists the base HEP for failure to diagnose a problem at 1E-03. If no PSFs are assumed, then the likelihood that damage occurs and goes undetected might be about 1E-04. Given that the estimated frequency for a loss of the load restraints in the Reactor Module during transport is the higher of the two failure modes, the more conservative estimated failure rate of 1.5E-03 per shipment is used for the likelihood that the passive cooling function fails. However, for a release to occur, this failure must be in combination with a failure of the reactor containment boundary. Those failure rates are discussed in Section 4.5.3.2.2.1 for random containment feature failure (1E-03 per transport), vibration and shock (1E-03 per transport), human error in packaging (1E-04 per transport), human error in disassembly (1E-04 per transport), and extreme cold that causes failure (2E-05 per transport). The sum of these failure rates is about 1.3E-03 per shipment. So, the combined

[43] BWXT Final Design Report, Appendix I.2, ATL-PLAN-110124, "Transportation Plan," page 29.

likelihood that failure in the passive heat transfer system occurs undetected and containment of the reactor containment boundary fails is about 2E-06 per shipment. (If heat up and/or pressurization of the reactor containment boundary is monitored during transport and compensatory measures are available to mitigate the pressurization then this accident scenario might be screened.)

Regarding impact of high ambient air temperatures that along with decay heat increases the pressure in the reactor containment boundary in combination with failure of containment, it difficult to know how often or how much ambient heating could increase pressure in the reactor containment boundary or if decay heat by itself is sufficient to cause a pressurized release if containment fails. However, to be conservative it can be assumed (until more design information is provided) that the likelihood of this event is equal to the event frequency of the total reactor containment boundary failures added together as described above (with exception of extreme cold) which is 1.3E-03 per shipment. (Again, if heat up is monitored and compensatory measures are available, then this accident scenario might be screened.)

4.5.3.2.2.3 Accident Frequency for Breach of a Non-Reactor Containment Boundary Component

The following discusses developing the initiating events frequency for the accident scenarios identified in Section 4.5.3.2.2 pertaining to containment breach of other parts of the Reactor Module besides the reactor containment boundary (e.g., the Shield Tank). As shown in Table 4-4 for Accident Class 9 (loss of general package containment – not reactor containment boundary), these accidents only involve release of contamination that is located in Reactor Module components that are not part of the reactor containment boundary.

Given that these accidents only result in release of contamination from package elements that have been handled during disassembly and loading of the TNPP package, the management of the risk from these scenarios can be considered covered by normal radiation safety practices. However, these scenarios should be provided as input to development of those controls. Concerning pressurization caused by loss of ventilation, that event frequency is estimated in Section 4.5.3.2.2.2 (1.5E-03 per shipment). Concerning containment failure caused by random or vibration caused failures, those event frequencies are estimated in Section 4.5.3.2.2.1 (1E-03 and 1E-04 per shipment respectively). Concerning high ambient air temperatures or hailstorms, those conditions should be assumed to occur during a shipment because of their relatively high likelihood. Accordingly, the only accident scenario of this type with initiating event frequency that cannot be derived from earlier discussion is pressurization due to radiolysis of hydrogenous material (e.g., Shield Tank not fully drained) and possible hydrogen accumulation in a contained space and ignition. It is not clear without more specific information to know whether this is even a credible accident, but it would require hydrogenous material (typically water) to generate hydrogen. The hydrogen would have to accumulate a flammable concentration in air to be ignitable and an ignition source would need to be present. This likely requires a combination of human errors, and therefore the accident frequency might be conservatively estimated to be 1E-04 per shipment using the same rational used to develop the packaging and disassembly human errors discussed in Section 4.5.3.2.2.1.

As stated above, these accidents result in minor contamination releases and their risk should be managed by normal radiation safety practices. However, these accident scenarios and the accident scenario frequencies discussed above should be provided as input to the radiation safety program to inform applicable controls. Accordingly, no accident frequencies are developed for these scenarios.

4.5.3.2.3 Incidents that Cause Increased Exposure Time

There is also a set of events that in which technical or logistical difficulties cause a lengthened transport time and an increased exposure of workers to radiation: (1) mechanical breakdown of the transport truck or trailer, (2) technical problems with the Reactor Module that requires resolution due to unanticipated failures or errors, and (3) adverse weather that stalls or delays transport. Given that these accidents only result in increased routine exposure (though unanticipated), the management of the risk from these scenarios can be considered covered by normal radiation safety practices. However, these scenarios should be provided as input to development of those controls. Accordingly, no accident frequencies are developed for these scenarios.

4.5.4 Assumptions Made as Part of Accident Likelihood Estimation

As described above, accident likelihood estimates are based on road hazard information determined using GIS, very large truck (>26,000 lb GVW) interstate and all state highway data for the five states that the assumed route traverses, and nation-wide large truck (>10,000 lb GVW) interstate data. These datasets were used to their greatest advantage, but certain assumptions and approximations were needed to provide accident frequency estimates for various reasons including limitations in these data sources. This section provides a listing of specific main assumptions used in the development of accident likelihoods. Specific assumptions about other aspects of the PRA such as the hazard analysis and factors important to estimating the radiological consequence from a transportation accident are provided in Sections 4.4.2.2 and 4.6.4.

1. The assumed route is from INL to WSMR and uses interstate highways in parts of Idaho, Utah, Wyoming, Colorado, and New Mexico as described in Section 4.5.1.

2. It is assumed that there is one transport in a year to be able to provide the accident frequency estimate on a per-year basis.

3. It is assumed that the known accident rates on all interstates in these five states are the same as accident rates on interstates of the assumed route.

4. The proportion of very large truck fatal accidents on interstate highways and all state highways is assumed to be the same for very large truck all accidents on these interstates and all state highways, as described in Section 4.5.2.3.

5. The known types and proportions of large truck interstate accidents in the nationwide dataset (i.e., in the MCMIS database) are assumed to be the same as the types and proportions of very large truck accidents on the assumed route, as described in Section 4.5.3.1.1.

6. For accidents where fire was the MHE, if a FHE was not specified, then the FHE was assumed to be fire. If another kind of accident was designated as the FHE, then these were assumed to a mixed accident (e.g., a collision and fire). This is based on the nation-wide dataset (i.e., in the MCMIS database) as described in Section 4.5.3.1.1.

7. For the most severe types of impact accidents, hard impacts (i.e., BRA 5H) are assumed to be heavy vehicle collisions, impacts with non-yielding objects, rollovers/overturns, and drops to

lower elevation. Medium impacts (i.e., BRA 5M) are assumed to be all other crashes, including light vehicle collisions, impacts with yielding objects, and jackknifes. This is based on the nation-wide dataset (i.e., in the MCMIS database) as described in Section 4.5.3.1.1.

8. For "less than hard impacts" (BRA 4), those that result in a fatality or injury only are assumed to be medium impacts (BRA 4M) and those that result in property damage only are assumed to be light impacts (BRA 4L). This is based on the nation-wide dataset (i.e., in the MCMIS database) as described in Section 4.5.3.1.1.

9. It was conservatively assumed that the percentage of tankers carrying flammable liquids is about the same for the assumed route as the percentage of tanker truck miles to total heavy-heavy trucks (>26,000 lb) miles nationwide as described in Section 4.5.3.1.1.

10. For the GIS estimation of the submersion accident, it is assumed for locations along the route identified to have bodies of water deep enough to submerge the reactor within 50 m of the highway in combination with an embankment of 1:4 or greater that if a truck in an accident left the road it could slid or roll into the body of water as described Section 4.5.1.4 and Section 4.5.3.1.2.

11. For GIS estimation of the frequency of accident resulting in a drop to a lower elevation, it was assumed that if a truck in an accident left the road with an embankment of 1:3 within 20 m of the road it could result in a drop-to-a-lower elevation accident if confirmed by street views of those locations as described in Section 4.5.1.5 and Section 4.5.3.1.3.

4.5.5 Accident Frequency Results for the Bounding Representative Accidents

This section discusses development of the frequencies for the bounding representative accidents. These frequencies are the sum of frequencies of accidents grouped into a bounding representative accident case. Detailed descriptions of the calculation of the frequencies of the contributing accidents are provided in Section 4.5.3.2. In all cases it is assumed that there is one transport in a year to be able to provide the accident frequency estimate on a per-year basis. A summary of the bounding representative accidents is presented in Table 4-20.

4.5.5.1 Fire Only that Originates Inside Transport Container – BRA 1

BRA 1 is a fire that originates inside the transport container. It is a general fire that originates from such sources as an electrical cable fault, propagates to the package, and ignites combustible material associated with the package. It includes an oil or grease fire that is ignited from a hot surface or electrical fault. The frequency estimated for this accident is based on NRC guidance for estimating fire ignition frequency as described in Section 4.5.3.2.1.2. The estimated frequency is for this accident is 9E-07 per year (assuming one transport in a year).

Table 4-20. Accident Frequency Estimates for Bounding Representative Scenarios

ID	Descriptions	Accident Frequency per Year(1)
BRA 1	Fire-only event that originates inside the transport container.	9.0E-07
BRA 2	Diesel fuel fire-only event that originates outside the transport container and propagates into the transport container and ignites combustible material in the transport container which damages the package.	2.0E-06
BRA 3	Hard impact highway accident that leads to release of radioactive material and loss of shielding. Includes impact with heavy vehicles and unyielding objects (e.g., concrete abutments or rock embankments), significant drops to lower elevation, or rollovers.	7.1E-05
BRA 4M	Less than a hard impact highway accident that results in release of some radiological material and loss of shielding. Medium impact that involves a severe collision with a light vehicle (e.g., one that results in fatality and/or injury).	9.7E-05
BRA 4L	Less than a hard impact highway accident that results in no release of radiological material or loss of shielding. Light impact such as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment) or impact that is not severe with a light vehicle (e.g., results in property damage only).	3.3E-04
BRA 5H	Hard impact highway accidents (i.e., equivalent to the impacts defined by BRA 3) that result in fire with exception of collision with a tanker carrying flammable material.	2.6E-08
BRA 5M	Medium impact highway accidents (i.e., severe collision with a light vehicle that leads to a fatality or injury) that results in fire.	5.9E-07
BRA 6	Collision with a tanker carrying flammable material that leads to fire.	7.1E-08
BRA 7	Loss of non-pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.	1.3E-03
BRA 8	Loss of pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.	1.3E-03
BRA 9	Addition of moderator and a change in core geometry caused by a drop into body of water that results in criticality.	Ranges between 5.1E-09 and 2.1E-06
BRA 10	Control rod withdrawal caused by impact from a road accident that results in criticality.	(2)

(1) For accident frequency calculations, one transport in a year is assumed.
(2) Evaluation pending design data/information.

4.5.5.2 Fire Only that Originates Outside Transport Container – BRA 2

BRA 2 is a diesel fuel fire that originates outside the transport container and propagates into the transport container, igniting combustible material in the transport container and damaging the package. The frequency of a BRA 2 event is estimated using the frequency of fire/explosion accidents (as the MHE) and adjusting it where the initiating event (FHE) is also fire/explosion. This is considered a fire-only event as explained in Section 4.5.3.1.1 and is 74.6% of fire/explosion accidents. Using these bases, the estimated frequency for the assumed route for BRA 2 is 2E-06 per year assuming one transport in a year.

4.5.5.3 Hard Impact Road Accident – BRA 3

BRA 3 is an impact with heavy vehicles and solid unyielding objects (e.g., concrete abutment, rock embankments), fall to lower elevation (e.g., drop from a bridge), and rollovers. The frequency of a BRA 3 event is estimated using the frequency of accidents considered to result in a "hard impact"; these include collision with a heavy vehicle, an impact with a non-yielding object (examples given above), a rollover/overturn accident involving the truck as described in Section 4.5.3.1.2, and an event not specifically identified in crash databases. The drop to a lower elevation was determined using route specific GIS hazard information as described in Section 4.5.3.1.3. Using these bases, the estimated frequency for the assumed route for BRA 3 is 7.1E-05 per year assuming one transport in a year.

4.5.5.4 Less than Hard Impact Road Accident – BRA 4

BRA 4 includes impact with light vehicles or objects that do not create much force when impacted (e.g., impacts with signs, utility poles, guard rails, and live animals), jackknifes that do not involve impact, and impacts with non-rock ground surfaces such as soil or clays. BRA 4 is split into two sub-categories, BRA 4M (for medium impact) and BRA 4L (light impact).

The BRA 4M event is assumed to involve severe collision with a light vehicle that causes some degree of damage to the TNPP package and shielding, resulting in release of radioactive material and direct radiation exposure. This accident is less severe and causes less damage than BRA 3 but more than BRA 4L. The frequency of BRA 4M events is estimated using the frequency of light vehicle collisions and adjusted to consider only those that result in fatality or injury only; this is 26% of light vehicle collisions as discussed in Section 4.5.3.1.1. Using these bases, the estimated frequency for the assumed route for BRA 4M is 9.7E-05 per year assuming one transport in a year.

The BRA 4L event does not result in significant damage to the package; there is no release of radiological material or loss of shielding. The frequency of BRA 4L events is estimated using the frequency of light vehicle collisions and adjusted to consider only those resulting in property damage; this is 74% of light vehicle collisions as discussed in Section 4.5.3.1.1. Using these bases, the estimated frequency for the assumed route for BRA 4L is 3.3E-04 per year assuming one transport in a year.

4.5.5.5 Road Impact and Fire Accident Except with Tanker Carrying Flammable Material – BRA 5

BRA 5 includes all road impact accidents that result in fire except collision with a tanker carrying flammable material. BRA 5 is split into two sub-categories, BRA 5H (for hard impact) and BRA 5M (medium impact). BRA 5 events involve fire, so one of the bases for estimating the fire accident frequency is the same as that used for BRA 2; however, unlike BRA 2, the BRA 5 includes an impact event such as a collision. Fire events designated in the truck accident data as a MHE but not as the FHE are considered to be crashes that result in fire and account for 25.4% of fire/explosion accidents as explained in Section 4.5.3.1.1. Because collisions with tankers are addressed in BRA 6, the frequency of BRA 6 is excluded from BRA 5H and BRA 5M as discussed in Section 4.5.3.3.6. Also, the frequency of the fire/explosion-only accident is excluded because it is addressed in BRA 2. BRA 5H and BRA 5M differ by the type of impact that results in fire/explosion.

The basis for the estimation of accident frequency for BRA 5H is the same as described for BRA 3 in Section 4.5.3.3.3. The accident frequency of BRA 5H was determined using the percentage of hard impact accidents to all impact accidents, which is effectively, BRA/(BRA 3 + BRA 4). Using these bases, the estimated frequency for the assumed route for BRA 5H is 2.6E-08 per year assuming one transport in a year.

BRA 5M medium impacts are the same as described for BRA 4 in Section 4.5.3.3.4. BRA 5M involves the percentage of medium/light impact accidents divided by all impact accidents – effectively BRA 4 ÷ (BRA 3 + BRA 4) as discussed in Section 4.5.3.1.1. Using these bases, the estimated frequency for the assumed route for BRA 5M is 5.9E-07 per year assuming one transport in a year.

4.5.5.6 Collision with a Tanker Carrying Flammable Material and Ensuing Fire – BRA 6

BRA 6 is a collision with a tanker carrying flammable material that results in fire. Here again the basis is the frequency of fire/explosion accidents adjusted by 25.4% to account for initiating events (FHE) that are crash/collision, not FHE of fire/explosion, the same as for BRA 5. However, BRA 6 must also account for the likelihood of striking a heavy truck tanker along the route. The percentage of heavy trucks that are cargo tanker trucks is estimated to be 10.3% as discussed in Section 4.5.3.1.1. Given, there is no distinction in the data between flammable liquids and other types of liquids transported in the tankers, this estimate is conservative. Using these bases, the estimated accident frequency for BRA 6 for the assumed route is 7.1E-08 per year assuming one transport in a year.

4.5.5.7 Loss of Non-Pressurized Reactor Containment Boundary – BRA 7

BRA 7 is a loss of package containment event resulting in a non-pressurized release from the reactor containment boundary not associated with a road impact accident. The development of the accident frequency contributors to BRA 7 are described in detail in Sections 4.5.3.2.1 and 4.5.3.2.2. The contributors to containment failure consist of random loss of a containment device or feature (1E-03 per transport), vibration and shock (1E-03 per transport), human error in packaging (1E-04 per transport), human error in disassembly (1E-04 per transport), and extreme cold (2E-05 per transport). The estimated accident frequency for BRA 7 is the sum of these failure rates which is 1.3E-03 per year assuming one transport in a year.

4.5.5.8 Loss of Pressurized Reactor Containment Boundary – BRA 8

BRA 8 is a loss of package containment event for a pressurized release from the reactor containment boundary but not associated with a road impact accident. The development of accident frequency contributors to BRA 8 are described in Section 4.5.3.2.2. BRA 8 requires a combination of events. It requires either loss of passive heat transfer from the package caused by degradation of the heat transfer pathway or extremely high ambient air temperature along with decay heat in combination with a reactor containment boundary failure. The highest event frequency contributor to BRA 8 is extremely ambient air temperature in combination with a reactor containment boundary failure. The estimated accident frequency for BRA 8 is 1.3E-03 per year assuming one transport in a year.

4.5.5.9 Criticality Event Involving Drop into a Body of Water – BRA 9

BRA 9 is addition of a moderator and a change in core geometry caused by a drop into body of water that results in criticality. The accident frequency estimated for the assumed route ranges from 2.1E-06 per year assuming one transport in a year which is estimated using route specific GIS data and 5.1E-09 per year using submersion accident data from the national MCMIS accident database. The two different approaches and along with pros of cons of using each are discussed in Section 4.5.3.1.2.

4.5.5.10 Criticality Event Caused by Control Rod Withdrawal – BRA 10

This section describes estimation of the frequency of BRA 10, which is control rod withdrawal caused by impact from a road accident that results in criticality. This frequency has not yet been developed because of insufficient design information and will be included in a revision to this report.

4.6 Development of Consequences for TNPP Transportation Accident Scenarios

This section discusses the approach for developing TNPP transportation accident consequences and the radiological dose consequences for each bounding representative accident. Consequence analysis is based on determining the source term for the release, the mobility of that source term (i.e., particle size and behavior), and the corresponding risk/dose to a human receptor. Section 4.6.1 discusses the methodology for determining the source term released from a transportation accident. Section 4.6.2 discusses the source term determined for the bounding representative accidents. Section 4.6.3 describes the approach for determining the radiological dose consequence from TNPP transportation accidents. Section 4.6.4 presents a list of assumptions used in determining the transportation accident source terms and the radiological dose consequences. Section 4.6.5 presents the radiological dose consequences for each bounding representative accident.

4.6.1 Source Term Methodology for Transportation Accident Scenarios

The radiological consequences of an accident can be the result of direct exposure to radiological material either due to loss of shielding or neutrons from an inadvertent criticality. Direct exposures principally impact receptors in the near vicinity of the accident. The dose consequences are highly dependent on materials used in the design.

Radiological consequences can also be the result of material released into the environment which can impact a human receptor through different dose pathways. This released material is referred to the “source term.” For these releases, the principal radiological dose pathway is usually airborne and the dose from the inhalation typically dominates the overall dose. Radiological material that is released produces a direct exposure dose in addition to inhalation dose.

For airborne releases, the source terms will be estimated using the following five factor formula (DOE 2013):

Source Term = MAR x DR x ARF x RF x LPF

where:

MAR = Material at risk
DR = Damage ratio

ARF = Airborne release fraction
RF = Respirable fraction
LPF = Leak path factor

The five-component equation, while traditionally developed for non-reactor nuclear facilities, can be applied to a TNPP transportation accident analysis. The following sections discuss the development of the individual elements making up the source term calculation as applied to the TNPP transportation accident analysis.

4.6.1.1 Material at Risk

The principal MAR relevant to the transportation risk assessment is described in Section 4.2.5. The development of the MAR is based upon the release of fission products and gases from the individual TRISO particles. As discussed in Section 4.2.4, for this evaluation, MAR has been developed for three locations within the TNPP package:

1. Gaseous and nongaseous fission products retained within the TRISO,

2. Fission products that have diffused from the TRISO fuel and are held up in the compact and other core structures,

3. Fission products and gases that have diffused from the TRISO fuel and have condensed or plated-out in the reactor containment boundary.

Based on the approach in Section 4.2.5, release estimates based on the 95% release fractions were developed for each of the 10 Fission Product Classifications (Table 4-1) as shown in Table 4-21. These release fractions were then applied to the core inventory contained in Appendix 8.1 to develop location specific MAR estimates contained in Appendix 8.1, Tables 8.1-2 through 8.1-4.

4.6.1.2 Damage Ratio

The damage ratio represents the fraction of the MAR that is impacted by accident generated stresses. For this evaluation, the damage ratios are estimates based on TNPP package transportation accident stresses transmitted to the MAR and is developed individually for the three primary locations of MAR. A summary of the damage ratios used for each bounding representative accident is provided in Table 4-22. The damage ratio is primarily a function of the energy involved in the accident and physical phenomena that can cause release.

Table 4-21. Fission Product Classification – Normal Operations Release Fractions

Classification	Representative Nuclides	95% Release Fraction MAR in Core Structure	95% Release Fraction MAR in Pressure Boundary
Noble	Xe-133	0.00E+00	**3.17E-05**
	Kr-85	0.00E+00	**3.22E-05**
	Kr-88	0.00E+00	3.15E-05
I, BR, Se, Te	I-131	0.00E+00	3.24E-05
	I-133	0.00E+00	3.24E-05
	Te-132	0.00E+00	3.14E-05
Cs, Rb	Cs-137	5.05E-04	5.50E-04
	Cs-134	5.00E-04	5.50E-04
Sr, Ba, Eu	Sr-90	9.92E-03	6.81E-05
Ag, Pd	Ag-110m	0.00E+00	2.50E-02
	Ag-111	0.00E+00	2.55E-02
Sb	Sb-125	8.62E-04	4.39E-04
Mo, Ru, Rh, Tc	Ru-103	1.10E-04	8.66E-07
La, Ce	Ce-144	1.10E-04	8.61E-07
	La-140	1.09E-04	8.59E-07
Pu, Actinides	Pu-239	1.04E-04	6.54E-08
H-3	H-3	0.00E+00	5.45E-02

Table 4-22. Damage Ratios for Bounding Represented Accidents

Represented Accident	FP(1)/ Gases in TRISO	FP in CORE	FP in Pressure Boundary
BRA 1	0	0	0
BRA 2	0	0.01	1
BRA 3	0.001	0.1	1
BRA 4	0	0.05	0.3
BRA 5	0	0.05	0.3
BRA 6	0.001	0.1	1
BRA 7	0	0	0.2
BRA 8	0	0	0.2

(1) Fission products.

4.6.1.3 Airborne Release Fraction and Respirable Fraction

The airborne release fraction represents the estimate of the total amount of a radioactive material that can be suspended in air and made available for airborne transport under an accident specific set of induced physical stresses. The damage ratio is primarily a function of the energy involved in the accident, physical phenomena that can cause release, and the form of the MAR. The respirable fraction represents the fraction of airborne radionuclides as particles that can be transported through air and inhaled into the human respiratory system and is commonly assumed to include particles 10-µm Aerodynamic Equivalent Diameter (AED) and less. For this evaluation, the airborne release fraction and respirable fraction estimates are based on both the material forms and accident stresses and is developed individually for the three primary locations and forms of MAR (e.g., fission products and gases within the TRISO, within the core and in the coolant loop).

The combined airborne release fractions and respirable fractions assumed for this evaluation are provided in Table 4-23.

Table 4-23. Combined Airborne Release Fractions and Respirable Fractions for Represented Accidents

Represented Accident	Combined Airborne Release Fractions and Respirable Fractions (ARF*RF)		
	FP/Gases in TRISO	FP in CORE	FP in Pressure Boundary
BRA 1	—	—	—
BRA 2	—	6.00E-05	6.00E-05
BRA 3 (FP)(1)	3.00E-04	3.00E-04	3.00E-04
BRA 3 (FG)(2)	1.00E+00	NA	NA
BRA 4	—	3.00E-04	3.00E-04
BRA 5	—	2.50E-06	2.50E-06
BRA 6	6.00E-05	6.00E-05	6.00E-05
BRA 7	—	—	1.00E-04
BRA 8	—	—	8.00E-04

(1) Fission products.
(2) Fission gases.

4.6.1.4 Leak Path Factor

The Leak Path Factor represents the attenuation (including deposition, holdup) of the airborne materials as they are transported from source to the surrounding environment where is it subjected to atmospheric dispersion.

The leak path factors assumed for this evaluation are provided in Table 4-24.

Table 4-24. Leak Path Factors (LPF) for Represented Accidents

Represented Accident	FP(1)/Gases in TRISO	FP in CORE	FP in Pressure Boundary
BRA 1	—	—	—
BRA 2	—	0.01	0.01
BRA 3 (FP)	0.05	0.1	0.5
BRA 3 (Gas)	1	NA	NA
BRA 4	—	0.01	0.05
BRA 5	—	0.01	0.05
BRA 6	0.05	0.1	0.5
BRA 7	—	—	0.001
BRA 8	—	—	0.005

(1) Fission products.

4.6.2 Source Term Determined for the Bounding Representative Accidents

Source terms were developed for each of the bounding representative accidents not screened on the basis of frequency. The source terms are contained in Table 8.1-5

4.6.2.1 Fire Only that Originates Inside Transport Container – BRA 1 Source Term

This section describes the source term development for a fire internal to the Reactor Module. As discussed in the vendor documents (BWXT 2022[44]), the low quantities of combustibles within the module, a fire, should it occur, would be limited in size or potential for growth by the controlled environment. As control, power, and instrumentation cabling is rated for low-flame spread, protected, and routed with adequate shielding or separation to preclude ignition of adjacent components, involvement of more than a single cable is not postulated.

The MAR within the TNPP package includes fission products in the TRISO/compacts, fission products released during normal operations and captured in the core materials, and fission products and gases that have deposited within the reactor containment boundary. All MAR is protected from the direct effects of a fire by the shielding vessel or the reactor pressure vessel and coolant boundary. Due to the limited size of the fire, failure of the reactor containment boundary and release of materials is not postulated for this event. Accordingly, there is no radiological dose consequence calculated from BRA 1.

4.6.2.2 Fire Only that Originates Outside Transport Container – BRA 2 Source Term

This section describes the source term development for a large fire external to the Reactor Module. It assumes a large diesel fuel fire that originates outside the module propagates into the module and ignites combustible material resulting in significant failures of the reactor containment boundary seals and subsequent release of material.

The size of the fire is conservatively assumed to be greater than the fuel in a single truck and the MAR affected includes material in the outer core region and the reactor containment boundary. For the MAR within the core, a damage ratio of 0.01 (1%) is assumed to be impacted by the fire and released. For the MAR within the reactor containment boundary a damage ratio of 1 (100%) is assumed to be impacted by the fire and released. The airborne release fraction of 6.00E-03 and a respirable fraction of 0.01 based on NUREG/CR-6410, Section 3.3.2.10 (*Nuclear Fuel Cycle Facility Accident Analysis Handbook* [NRC 1998]) for air entrainment from fires associated with contaminated non-reactive material was applied. A leak path factor of 0.01 has been assigned based on assumed complete failure of the seals (e.g., Grayloc® connectors) of the reactor containment boundary creating a leak path to the compromised CONEX box and then subsequent release to the environment.

This results in an overall release fraction of 6.00E-09 for the MAR in the core and 6.00E-07 for the MAR in the reactor containment boundary.

4.6.2.3 Hard Impact Road Accident – BRA 3 Source Term

This section describes the source term development for a severe impact road accident that leads to release of radioactive material. It assumes the impacts are sufficient to cause breaches to the TNPP

[44] BWXT Final Design Report, Appendix IV, ATL-TECR-109977 – "MNPP Facility Fire Hazards Analysis"

package components including the CONEX box walls and reactor containment boundary as well as damage to the core and fuel compacts. Affected MAR includes all locations (fuel, core, and reactor containment boundary).

For the MAR within the fuel a damage ratio of 0.001 (0.1%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied to the fission products within the TRISO based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. A leak path factor of 0.05 is assigned based on the pathway, through the damaged core and pressure vessel boundary to the CONEX box and subsequent release to the environment. All fission gases within the TRISO are assumed to be released, with an airborne release fraction, respirable fraction and leak path factor of 1.

For the MAR within the core a damage ratio of 0.1 (10%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. A leak path factor of 0.1 is assigned based on the pathway, through the pressure vessel boundary to the CONEX box and subsequent release to the environment.

For the MAR within the reactor containment boundary a damage ratio of 1 (100%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. A leak path factor of 0.5 is assigned to this material.

This results in an overall release fraction of 1.50E-08 applied to the fission products in the fuel and 1.00E-03 applied to the fission gases in the fuel, a release fraction of 3.0E-06 applied to the MAR in the core and a release fraction of 1.50E-04 applied to the MAR in in the reactor containment boundary.

4.6.2.4 Less than Hard Impact Road Accident – BRA 4 Source Term

This section describes the source term development for a less severe impact road accident that leads to release of radioactive material. It assumes the impacts are sufficient to cause breaches to the CONEX box walls and reactor containment boundary as well as damage to a portion of the core. BRA 4 is further broken down into 4M (medium) and 4L (light). BRA 4M accidents are less than a hard impact highway accident that results in release of some radiological material and loss shielding. These medium impact accidents are defined as a severe collision with a light vehicle (e.g., one that results in fatality and or injury). BRA 4L accidents are light impact highway accident that result in no release of radiological material or loss of shielding. These light impact accidents are defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment) or impact that is not severe with a light vehicle (e.g., results in property damage only). A more precise definition of yielding objects is discussed in the frequency estimation in Section 4.3.3.1 as presented in Table 4-16.

For BRA 4M, the MAR within the core is assumed to have a damage ratio of 0.05 (5%). For this MAR, an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. A leak path factor of 0.01 is assigned based on the pathway through the reactor containment boundary to the CONEX box and subsequent release to the environment.

For the MAR within the reactor containment boundary a damage ratio of 0.3 (30%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. A leak path factor of 0.05 is assigned to this material.

This results in overall release fractions of 1.50E-07 applied to the MAR in the core and 4.50E-06 applied to the MAR in in the reactor containment boundary.

4.6.2.5 Road Impact and Fire Accident Except with Tanker Carrying Flammable Material – BRA 5 Source Term

This section describes the source term development for impact road accidents combined with a fire that leads to release of radioactive material. It assumes the impacts are sufficient to cause breaches to the CONEX box walls and reactor containment boundary as well as damage to a portion of the core which is then subjected to a fire event. BRA 5H addresses a severe (BRA 3) impact combined with the effect of a fire associated with a limited fuel quantity (i.e., a single truck fuel quantity). BRA 5M addresses a medium (BRA 4M) impact combined with the effect of a fire associated with a limited fuel quantity.

4.6.2.5.1 Hard Impact Accident and Ensuing Fire – BRA 5H Source Term

For the MAR within the fuel a damage ratio of 0.001 (0.1%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied to the fission products within the TRISO based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. For the fire portion an airborne release fraction of 2.5E-04 and a respirable fraction of 0.01 based on DOE-HDBK-3010-94, Section 4.4.1.1 for air entrainment from fires associated with non-reactive material was applied. The airborne release and respirable fractions for fire are based on the assumption that the diesel fuel quantity (the source of the fire) is limited to what may be carried in transport vehicle. A leak path factor of 0.05 is assigned based on the pathway through the damaged core and reactor containment boundary to the CONEX box and subsequent release to the environment. All fission gases within the TRISO are assumed to be released, with an airborne release fraction, respirable fraction and leak path factor of 1.

For the MAR within the core a damage ratio of 0.05 (5%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. For the fire portion an airborne release fraction of 2.5E-04 and a respirable fraction of 0.01 based on DOE-HDBK-3010-94, Section 4.4.1.1 for air entrainment from fires associated with non-reactive material was applied. A leak path factor of 0.01 is assigned based on the pathway, through the reactor containment boundary to the CONEX box and subsequent release to the environment.

For the MAR within the reactor containment boundary a damage ratio of 0.3 (30%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. For the fire portion an airborne release fraction of 2.5E-04 and a respirable fraction of 0.01 based on DOE-HDBK-3010-94, Section 4.4.1.1 for air entrainment from fires associated with non-reactive material was applied. A leak path factor of 0.5 is assigned to this material.

This results in overall release fractions of 1.51E-08 applied to the fission products in the fuel and 1.00E-03 applied to the fission gases in the fuel, a release fraction of 3.01E-06 applied to the MAR in the core and a release fraction of 1.51E-04 applied to the MAR in the reactor containment boundary.

4.6.2.5.2 Medium Impact Accident and Ensuing Fire – BRA 5M Source Term

For the MAR within the core a damage ratio of 0.05 (5%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. For the fire portion an airborne release fraction of 2.5E-04 and a respirable fraction of 0.01 based on DOE-HDBK-3010-94, Section 4.4.1.1 for air entrainment from fires associated with non-reactive material was applied. The airborne release and respirable fractions for fire are based on the assumption that the diesel fuel quantity (the source of the fire) is limited to what may be carried in transport vehicle. A leak path factor of 0.01 is assigned based on the pathway, through the reactor containment boundary to the CONEX box and subsequent release to the environment.

For the MAR within the reactor containment boundary a damage ratio of 0.3 (30%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. For the fire portion an airborne release fraction of 2.5E-04 and a respirable fraction of 0.01 based on DOE-HDBK-3010-94, Section 4.4.1.1 for air entrainment from fires associated with non-reactive material was applied. A leak path factor of 0.05 is assigned to this material.

This results in overall release fractions of 1.51E-07 applied to the MAR in the core and 4.540E-06 applied to the MAR in the reactor containment boundary.

4.6.2.6 Collison with a Tanker Carrying Flammable Liquid and Ensuing Fire– BRA 6 Source Term

This section describes the source term development for a severe impact road accident combined with a large fire that leads to release of radioactive material. It assumes the impacts are sufficient to cause breaches to the CONEX box walls and reactor containment boundary as well as damage to portions of the fuel and the core which is then subjected to a large fire event.

For the MAR within the fuel a damage ratio of 0.001 (0.1%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied to the fission products within the TRISO based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. For the fire affects an airborne release fraction of 6.00E-03 and a respirable fraction of 0.01 based on NUREG/CR-6410, Section 3.3.2.10 for air entrainment from fires associated with non-reactive material was applied. A leak path factor of 0.05 is assigned based on the pathway, through the damaged core and reactor containment boundary to the CONEX box and subsequent release to the environment. All fission gases within the TRISO are assumed to be released, with an airborne release fraction, respirable fraction and leak path factor of 1.

For the MAR within the core a damage ratio of 0.1 (10%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. For the fire affects an airborne release fraction of 6.00E-03 and a respirable fraction of 0.01 based on NUREG/CR-6410, Section 3.3.2.10 for air entrainment from fires associated with contaminated non-reactive material was applied. A leak path factor of 0.1 is assigned based on the pathway, through the reactor containment boundary to the CONEX box and subsequent release to the environment.

For the MAR within the reactor containment boundary a damage ratio of 1 (100%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.3 is applied based on NUREG/CR-6410, Section 3.3.4.12a for impacts on powder by debris. For the fire affects an airborne release fraction of 6.00E-03 and a respirable fraction of 0.01 based on NUREG/CR-6410, Section 3.3.2.10 for air entrainment from fires associated with contaminated non-reactive material was applied. A leak path factor of 0.5 is assigned to this material.

This results in overall release fractions of 1.80E-08 applied to the fission products in the fuel and 1.00E-03 applied to the fission gases in the fuel, a release fraction of 3.60E-06 applied to the MAR in the core and a release fraction of 1.80E-04 applied to the MAR in the reactor containment boundary.

4.6.2.7 Loss of Non-Pressurized Reactor Containment Boundary – BRA 7 Source Term

This section describes the source term development for a non-pressurized loss of the reactor containment boundary not caused by a road accident. The MAR affected for this event is assumed to be that contained in the reactor containment boundary.

For the MAR within the reactor containment boundary a damage ratio of 0.2 (20%) is assumed. For this MAR an airborne release fraction of 1.00E-03 and respirable fraction of 0.1 is applied based on DOE-HDBK-3010-94, Section 4.4.3.3.1 for vibration impacts. A leak path factor of 0.001 is assigned to this material, based on failure to properly seal the reactor containment boundary (gasket failure) and release to the CONEX box and then subsequent release to the environment.

This results in overall release fraction of 2.0E-08 applied to the MAR in the reactor containment boundary.

4.6.2.8 Loss of Pressurized Reactor Containment Boundary – BRA 8 Source Term

This section describes the source term development for the loss of a pressurized reactor containment boundary not caused by a road accident. The MAR affected for this event is assumed to be that contained in the reactor containment boundary.

For the MAR within the reactor containment boundary a damage ratio of 0.2 (20%) is assumed. For this MAR an airborne release fraction of 2.00E-03 and respirable fraction of 0.4 is applied based on NUREG/CR-6410, Section 3.3.1.11 for low pressure release of powders from a container. A leak path factor of 0.005 is assigned to this material, based on failure of a seal in the reactor containment boundary (gasket failure) and release to the CONEX box and then subsequent release to the environment.

This results in overall release fraction of 8E-07 applied to the MAR in the reactor containment boundary.

4.6.3 Approach for Developing Transportation Accident Consequences

This section discusses the approach for developing TNPP transportation accident radiological dose consequences. It relies on the release information in Section 4.2.2 to define the radionuclides that leave the transportation package and meet the dosimetry screening criteria which is the source term for the dosimetry calculations. The dose calculations are based on material that has left the transportation package; any external dose from unreleased material within the package will need to be considered in addition to the doses presented in Section 4.6.3 from the released material. The dose calculations are based on the methodology utilized by the IAEA Q system described in IAEA SSG-26 (IAEA 2014) as introduced in Section 3.2.1 and is the basis of the A_1 and A_2 values utilized in 10 CFR Part 71. The specific methodology for calculating radiological dose consequences for human receptors is presented in Appendix I of SSG-26.

The IAEA Q system is a way to define quantity limits for material in a Type A package as well as applications in transport regulations and establishing leakage limits in Type B(U), Type B(M), or Type C package activity leakage limits, LSA and excepted package contents limits, and contents limits for low dispersible radioactive material (LDRM) and special form and non-special form radioactive materials (IAEA 2014). The IAEA Q system methodology was chosen for the dose calculations for this activity based on its wide acceptance and adoption both within the United States transportation regulations as well as the international community. The Q system includes exposure pathways for someone in the vicinity of a Type A package involved in a severe transportation accident. The pathways used to determine a series of Q values are external photon dose, external beta dose, inhalation dose, skin, and ingestion dose due to contamination transfer and submersion dose. For this effort ingestion dose and submersion dose will not be included; ingestion will not be included (consistent with IAEA SSG-26 findings that explicit consideration of the ingestion pathway is unnecessary) and submersion dose will not be included because the assumption is being made that the exposure will take place outside which will limit the time that a receptor might stand in a gaseous cloud of radionuclides. Q value analyses do not consider the content limits for special form alpha and neutron emitters or tritium. A_2 values are defined by the lowest of the Q values (for the exposure pathways) or the A_1 value if it is lower than the Q values. The Q values are derived based on the following radiological criteria in IAEA SSG-26 (IAEA 2014):

- The effective dose or committed effective dose to a person exposed in the vicinity of a transport package following an accident should not exceed a reference dose of 50 mSv.

- The equivalent dose or committed equivalent dose received by individual organs, including the skin, of a person involved in the accident should not exceed 0.5 Sv, or in the special case of the lens of the eye, 0.15 Sv.

- A person is unlikely to remain at 1 m from the damaged package for more than 30 minutes. (Appendix I, page 273, IAEA 2014)

For the purposes of this dose assessment, the dose coefficients developed as a part of IAEA SSG-26 were utilized wherever possible to keep the methodology consistent with the IAEA Q system and dose methodology for development of the A_1 and A_2 values.

4.6.3.1 External Dose Due to Photons

The external dose due to photons is determined by evaluating the external radiation dose due to gamma or X rays to the whole body of a person standing 1 m from the edge of the unshielded radioactive material. There are two possible forms of unshielded radioactive material: (1) material that is released from the TNPP package to the environment, and (2) material that is not released from containment but because there is a loss or degradation in the shielding is a source of direct radiation dose to the worker; however, radiation dose from unreleased radioactive material is not considered. The reactor vessel is judged to remain largely intact even after a transportation accident involving severe impact. The most likely significant degradation is assumed to be dents, bends, and other distortions that possibly create fissures where radiation streaming is then possible. The external exposure scenario for external dose from released material would likely bound dose from fissure streaming. More detailed reactor design information and safety analysis is required to calculate the potential external dose from the reactor core due to significant loss of reactor shielding from a transportation accident.

The external dose coefficients used in this report to calculate dose from released radioactive material are from IAEA SSG-26 (IAEA 2014). For radiological dose from released material, this calculation does not account for dispersion so it will likely be overly conservative because it assumes that the receptor is 1 m away from any released material. The source terms developed in Section 4.6 (using the five-factor equation presented in Section 4.6.1) were multiplied by 100 so that the amount of material used to calculate external dose was more reflective of the total material released rather than the amount of material that is respirable (accounting for the respirable fraction, RF).

External photon dose to a member of the public is estimated based on the worker dose. A member of the public is assumed to be 25 m from the package compared to 1 m for the worker; all other assumptions are the same. The 25 m is the DOT isolation and protective action distance for high level radiological material emergency response (DOT 2020). The source geometry for released material approximates that of a point source at 25 m. The public external dose is reduced from the worker dose by a factor of 1/distance^2, or a reduction factor of 625 (0.16%) for the 25 m distance.

4.6.3.2 External Dose Due to Beta Radiation

The external dose due to beta radiation is evaluated based on the potential for beta dose to the receptor's skin. The IAEA SSG-26 methodology is for beta emitters that are unshielded but includes a concept of residual shielding for beta emitters which has been retained in this dosimetry analysis. The previous beta emitter shielding in the Q system was associated with the materials such as the beta window protector, package debris, etc. and was assumed to be a very conservative shielding factor of 3 for beta emitters of maximum energy (greater than or equal to 2 MeV) (IAEA 2014). The IAEA SSG-26 methodology and associated dose coefficients used in this analysis extended this shielding methodology to include a range of shielding factors depending on the beta energy based on an absorber of approximately 150 mg/cm^2 thickness. In the case of annihilation radiation, this has not been included in the evaluation of beta dose to skin because it will be a very small contribution to the skin dose, but the resulting 0.51 MeV gamma rays are included in the photon energy per disintegration in the derivation of the photon dose coefficients for the radionuclides. In the case of conversion electrons, they are treated as monoenergetic beta particles.

The dose rate coefficients used in this report are from IAEA SSG-26 (IAEA 2014). The use of the dose coefficients for external dose due to beta radiation are for a person standing 1 m away from the released contamination. This calculation does not account for dispersion so it will likely be overly conservative because it assumes that the receptor is 1 m away from any released contamination and the dose not account for the dispersion of released material. The source term developed in Section 4.6 was multiplied by 100 so that the amount of material used to calculate external dose was more reflective of the total material released rather than the amount of material that is respirable.

The exposure distance for a member of the public is increased to 25 m for external dose calculations. At this distance there would be negligible beta dose contribution and no dose is calculated.

4.6.3.3 Inhalation Dose

The inhalation dose is calculated using the effective dose coefficient for inhalation (Sv/Bq) listed in the Appendix I of IAEA SSG-26 (IAEA 2014). The human uptake value of 1E-03 was selected based on its use in IAEA SSG-26 methodology for someone standing within 10 m of the release in an outdoor environment. The uptake value of 1E-03 was derived based on work related to conservative dispersion and human uptake assumptions for a downwind distance of 100 m. Extrapolation of these models to shorter distances is unreliable, but IAEA SSG-26 estimates that uptake values at 10 m would increase by a factor of about 30 compared to those at 100 m which would put uptake factors in the range of 1E-04 to 1E-03. For the purposes of this dose evaluation, uptake factors for the source term calculated in Section 5.3.3 of this report will be assumed to be 1E-03 for a person standing approximately 10 m from the release point. This uptake value represents the amount of material taken up into a human receptor following a release and is separate from the estimate of what material was released as calculated in Sections 5.3.3 of this report. Inhalation doses for this effort were calculated using the inhalation dose coefficients found in Appendix I of IAEA SSG-26 (IAEA 2014).

Inhalation dose to a member of the public is estimated based on the worker dose. The same exposure assumptions are used for the public except the distance from release is assumed to be 30 m instead of 10 m. IAEA SSG-26 states the dose increases by a factor of 30 from 100 m to 10 m. A power function was fit to this change in dose over distance, with a correlation coefficient (R^2) of 1. Using this assumption and the power function, the dose at 30 m was determined to be 5.9 higher than the dose at 100 m. When comparing to the worker dose at 10m, which is 30 times high, the ratio at 30 m is 5.9/30, or 19.7% of the dose at 10 m. The public inhalation dose is therefore about 20% of the worker inhalation dose.

4.6.3.4 Skin Contamination Dose

The skin contamination dose from beta emitters is estimated for a person that has been contaminated with non-special form radioactive materials from the release. For this dose assessment, the dispersed radionuclides (source term) will be evaluated by the criteria set in Section 5.3; this is a deviation from the methodology of IAEA SSG-26 which has a set assumption for amount of material released from the package. The IAEA SSG-26 assumptions are related to ungloved work with debris leading to 10% of radioactive material released getting on the hands and remaining there for 5 hours. The skin contamination dose is based on the source term calculated in Section 5.3.3 which is a respirable release fraction; while the actual amount of material release is higher than the respirable fraction it is also unlikely that a worker would be handling debris around this accident and so it is assumed that the IAEA SSG-26 methodology would still be conservative. For the purposes of this evaluation the skin dose is calculated using the equivalent skin dose rate per unit activity per unit area of the skin ($Sv\ s^{-1}\ TBq^{-1}\ m^2$) found in Appendix I or IAEA SSG-26 (IAEA 2014).

There would be no skin contamination dose to a member of the public. The public is assumed to remain 25-30 m away from the package with no potential for contamination to be transferred to the skin from handling the radiological material.

4.6.3.5 Exclusion of Ingestion and Submersion Dose

Possible exposure from ingestion and submersion are not included in this analysis. Excluding ingestion – as a part of skin contamination – is consistent with IAEA SSG-26 findings that explicit consideration of the ingestion pathway is unnecessary. Internal dose via the inhalation pathway will normally be limiting for internal contamination for both beta and alpha emitters under the Q system. Submersion dose is not included because the assumption is made that the exposure will take place outside with high potential for effective dilution and conditions that limit the time that a receptor might stand in a gaseous cloud of radionuclides. Submersion dose is considered in IAEA SSG-26 only for gaseous radionuclides that do not become incorporated into the body. These include certain isotopes of argon, krypton, xenon, and radon. Only 3 radionuclides identified in Section 4.6.1 would be excluded: Kr-85, Xe-131m, and Xe-133.

4.6.3.6 Radionuclides Not Included in IAEA SSG-26

The screening methodology identified in Section 4.6.1 identified the radionuclides in the release source term that are included in the dosimetry assessment. Most of the radionuclides included in this screened list have associated dose coefficients for the identified exposure pathways in the IAEA SSG-26 and those dose coefficients are utilized in this dose evaluation (IAEA 2014). There are radionuclides included in the screened list of radionuclides that do not have dose coefficients in IAEA SSG-26; these radionuclides are listed in Table 4-25. Some are decay products of other radionuclides included in IAEA SSG-26 and are assumed to be included with the parent dose factor(s). Among the others without dose factors, tritium (H-3), is a very low contributor to inhalation dose compared to other radionuclides and has essentially no external dose as a soft beta-emitter (0.005 MeV average).

Table 4-25. Radionuclides Included in the Dosimetry Source Term
Which Do Not have Dose Coefficients in IAEA SSG-26

Radionuclide	Dosimetry Basis	Status
Ba-136m	Decay product of Cs-136; $T_{1/2}$ = 0.3 s	Included
H-3	Low dose contributor for reactor/transportation accidents	Minor, excluded
Y-89m	Decay product of Sr-89	Included
Pm-146	Negligible source term and dose contributor; $T_{1/2}$ = 5.53 y	Excluded
Sb-127	Negligible source term and dose contributor; $T_{1/2}$ = 3.85 d; 0.3160 MeV β-, 0.6934 MeV γ; decays to Te-127m	Excluded
Tb-161	Negligible source term and dose contributor; $T_{1/2}$ = 6.906 d; 0.2025 MeV β-, 0.0365 MeV γ; decays to Dy-161	Excluded

Other radionuclides without dose factors are less common and screening was performed to determine if they could be important to calculations. The radionuclide Ba-136m is not included among the 1,252 radionuclides in International Commission on Radiological Protection (ICRP) Publication 107 (*Nuclear Decay Data for Dosimetric Calculations* [ICRP 2008]), and neither is it included among radionuclides with inhalation dose coefficients in ICRP Publication 119 (*Compendium of Dose Coefficients based on ICRP Publication 60* [ICRP 2012]). Emissions of Ba-136m are likely included in dose coefficients for Cs-136. On this basis, Ba-136m was removed from further consideration as a dose contributor.

Antimony-127 (Sb-127), Pm-146, and Tb-161, and Np-238 are all near the bottom of the source term in Section 4.6.1 with activities at least 5 orders of magnitude smaller than important dose contributors like Cs-137 and Sr-90. All are beta-emitters, and Sb-127 and Tb-161 have half-lives of only a few days. Pm-146 has a longer half-life but has a source term orders of magnitude smaller than Sb-127 and Tb-161. Sb-127, Pm-146, and Tb-161 were not considered further in the dose estimate process.

4.6.4 Accident Consequence Results for the Bounding Representative Accidents

This section presents a list of the assumptions used in the calculations to determine the transportation accident source terms and the radiological dose consequences for the bounding representative accidents. Specific assumptions about other aspects of the PRA such as the hazard analysis and factors important to estimating the accident likelihood are identified in the sections of the report that address those analyses in detail (i.e., Sections 4.4.2.2 and 4.5.4). The following is for the baseline case:

1. The reactor core has decayed 90 days after three years of operation.

2. The portion of the primary reactor cooling system transported contains all the condensed or plated-out radioactive material (e.g., that released fission product and condensed gases in this system has not been removed before transport).

3. A radioactive material cleanup system and/or the resulting radioactive waste material is not transported with the microreactor.

4. The dose consequences of direct radiation exposure from released material is addressed and contributes to consequence results presented in Section 4.6.5. Excluded is the dose contribution of direct radiation exposure from material that has not been released (including activated material) because the radiation shielding is degraded or lost in a reactor accident. There is not enough design information to calculate external dose contribution from this exposure pathway for this initial report.

5. The baseline case release fractions from normal operations for material residing in the core, reactor structure, and coolant system and the source term factors described in Section 4.6.1 represent best judgment but conservative estimates.

6. Using a standard transportation accident consequence analysis approach based on guidance from IAEA SSG-26 provides reasonable and comparable results without refining the approach for specific considerations. Specific assumptions related to the exposure pathways for both a worker or a member of the public are identified in Section 4.6.3. Each dose assessment is based on a series of exposure assumptions for that pathway of exposure (i.e., time near the released material, distance from the released material). Generally, these assumptions were selected to mimic the assumptions from IAEA SSG-26. Any deviation from these exposure assumptions could lead to changes in the resulting dose.

7. The dose contribution from ingestion submersion was assumed to be negligible, and therefore were not included. Per IAEA SSG-26 findings, explicit consideration of the ingestion pathway is unnecessary and for submersion it is assumed exposure will take place outside and which significantly limits the time that a receptor might stand in a gaseous cloud of radionuclides.

4.6.5 Accident Consequence Results for the Bounding Representative Accidents

This section is a summary of the radiological dose consequences for each of the bounding representative accidents broken down into the different dose pathways for each accident and MAR contribution. The MAR contributions are from: (1) the TRISO fuel itself, (2) the radiological material that diffused into the core structure such as the core compacts during operation, and (3) and radiological material that condensed or plated-out material in the reactor containment boundary during operation. Table 4-26 summarizes this information and provides the Total Effective Dose Equivalent (TEDE) in the last column for each accident. Per the IAEA SSG-26 methodology, external radiation dose from beta emitters that are released and unshielded to a worker who is close by (i.e., 1 m) is accounted for in the TEDE and is presented in the fourth column. However, even though skin contamination equivalent from ungloved work with debris is a radiological dose pathway prescribed by the IAEA SSG-26, Appendix I guidance, it is reasonably assumed that workers involved in handling radioactive material after an accident would wear the appropriate protective clothing including gloves. Therefore, skin contamination equivalent skin dose is presented in the fifth column is for information only and is not converted to effective dose and assumed to be a contributor to the TEDE for the accident.

Accordingly, the proposed risk evaluation guidelines do not specifically consider the radiological dose from skin contamination as result of handling a damaged package.

A discussion of the risk of transportation accidents (i.e., consequences and likelihood) and comparison to the proposed risk evaluation guidelines is presented in Section 4.7.

The results presented in Table 4-26 show that the inhalation pathway dominates the effective dose for the bounding representative accidents in the accident for which it was a contributor. External beta dose is the next highest contributor but is relatively small contributor. Based on the spreadsheets used to generate these results, key radionuclides are Ce-144, Sr-90, and Ru-106 for fission products and Pu-238, Pu-241, and Cm-242 for transuranic radionuclides, all contributing to inhalation dose. Fission products dominate the dose to skin from the beta particle emissions of these radionuclides. The dose contribution of radionuclides differs somewhat between external beta dose and skin contamination dose, but Ce-144, Y-91, and Sr-89 are key radionuclides for both pathways.

Table 4-26 shows that the radiological dose consequences from the TRISO fuel itself dominate the results in the accident for which it was a contributor opposed to radiological material diffused into the reactor core internals or plated out in the primary system. The next biggest contributor to radiological dose consequences is from radiological material that diffused into the reactor core internals during operation and was released in a transportation accident.

Table 4-26. Dose from Bounding Representative Accidents by MAR Contributions and Dose Pathways (2 sheets total)

Bounding Representative Accident	Photon External Effective Dose (rem)		Inhalation Effective Dose (rem)		Beta External Equivalent Skin Dose (rem)(1)	Skin Contamination Equivalent Skin Dose (rem)(2)	Total Effective Dose Equivalent (rem)	
MAR Contributors	Worker	Public	Worker	Public	Worker	Worker	Worker	Public
BRA 1 – Fire Only that Originates Inside Transport Container								
Total	0	0	0	0	0	0	0	0
BRA 2 – Fire Only that Originates Outside Transport Container								
TRISO fuel	0	0	0	0	0	0	0	0
Reactor core	3.2E-05	5.1E-08	1.1E-03	2.1E-04	3.1E-03	3.4E-05	1.1E-03	2.1E-04
Coolant boundary	2.7E-04	4.3E-07	1.0E-03	2.1E-04	3.2E-03	6.5E-03	1.4E-03	2.1E-04
Worker Total	3.0E-04		2.1E-03		6.4E-03	9.9E-03	**2.5E-03**	
Public Total		4.9E-07		4.2E-04				**4.2E-04**
BRA-3 – Hard Impact Road Accident								
TRISO fuel	13.1	2.1E-02	175	34.4	920	2550	197	34.4
Reactor core	1.6E-02	2.6E-05	5.4E-01	1.1E-01	1.5	1.7	5.7E-01	1.1E-01
Coolant boundary	6.8E-02	1.1E-04	2.6E-01	5.1E-02	8.1E-01	1.6	3.4E-01	5.1E-01
Worker Total	13.2		176		923	2550	**198**	
Public Total		2.1E-02		34.6				**34.6**
BRA 4 – Less than Hard Impact Road Accident								
BRA 4M – Medium Impact Road Accident								
TRISO fuel	0	0	0	0	0	0	0	0
Reactor core	8.0E-04	1.3E-06	2.7E-02	5.3E-03	7.8E-02	8.5E-02	2.8E-02	5.3E-03
Coolant boundary	2.0E-03	3.3E-06	7.8E-03	1.5E-03	2.4E-02	4.9E-02	1.0E-02	1.5E-03
Worker Total	2.8E-03		3.5E-02		1.0E-01	1.3E-01	3.9E-02	
Public Total		4.5E-06		6.8E-03				6.8E-03
BRA 4L – Light Impact Road Accident								
Total	0	0	0	0	0	0	0	0
BRA 5 – Road Impact and Fire Accident Except with Tanker Carrying Flammable Material								
BRA 5H – Hard Impact Accident and Ensuing Fire								
TRISO fuel	13.1	2.1E-02	175	34.4	920	2550	197	34.4
Reactor core	1.6E-02	2.6E-05	5.4E-01	1.1E-01	1.5	1.7	5.7E-01	1.1E-01
Coolant boundary	6.8E-02	1.1E-04	2.6E-01	5.1E-02	8.1E-01	1.6	3.4E-01	5.1E-02
Worker Total	13.2		176		923	2550	198	
Public Total		2.1E-02		34.6				34.6
BRA 5M – Medium Impact Accident and Ensuing Fire								
TRISO fuel	0	0	0	0	0	0	0	0
Reactor core	8.1E-04	1.3E-06	2.7E-02	5.3E-03	7.9E-02	8.6E-02	2.9E-02	5.3E-03
Coolant boundary	2.1E-03	3.3E-06	7.9E-03	1.6E-03	2.4E-02	4.9E-02	1.0E-02	1.6E-03
Worker Total	2.9E-03		3.5E-02		1.0E-01	1.4E-01	3.9E-02	
Public Total		4.6E-06		6.9E-03				6.9E-03
BRA 6 – Collision with a Tanker Carrying Flammable Material and Ensuing Fire								
TRISO fuel	13.2	2.1E-02	176	34.6	921	2550	198	34.6
Reactor core	1.9E-02	3.1E-05	6.4E-01	1.3E-01	1.9	2.0	6.8E-01	1.3E-01
Coolant boundary	8.2E-02	1.3E-04	3.1E-01	6.1E-02	9.7E-01	1.0	4.0E-01	6.1E-02
Worker Total	13.3		177		924	2550(3)	199	

Table 4-26. Dose from Bounding Representative Accidents by MAR Contributions and Dose Pathways (2 sheets total)

Bounding Representative Accident	Photon External Effective Dose (rem)		Inhalation Effective Dose (rem)		Beta External Equivalent Skin Dose (rem)(1)	Skin Contamination Equivalent Skin Dose (rem)(2)	Total Effective Dose Equivalent (rem)	
MAR Contributors	Worker	Public	Worker	Public	Worker	Worker	Worker	Public
Public Total		2.1E-02		34.8				34.8
BRA 7 – Loss of Non-Pressurized Reactor Containment Boundary								
TRISO fuel	0	0	0	0	0	0		
Reactor core	0	0	0	0	0	0	0	0
Coolant boundary	9.1E-06	1.5E-08	3.5E-05	6.8E-06	1.1E-04	2.2E-04	4.5E-05	6.8E-06
Worker Total	9.1E-06		3.5E-05		1.1E-04	2.2E-04	4.5E-05	
Public Total		1.5E-08		6.8E-06				6.8E-06
BRA 8 – Loss of Pressurized Reactor Containment Boundary								
TRISO fuel	0	0	0	0	0	0	0	0
Reactor core	0	0	0	0	0	0	0	0
Coolant boundary	3.6E-04	5.8E-07	1.4E-03	2.7E-04	4.3E-03	8.7E-03	1.8E-03	2.7E-04
Worker Total	3.6E-04		1.4E-03		4.3E-03	8.7E-03	1.8E-03	
Public Total		5.8E-07		2.7E-04				2.7E-04
BRA 9 – Criticality Event Involving Drop into a Body of Water								
No radiological dose calculated because of the extremely low likelihood of the scenario.								
BRA 10 – Criticality Event Caused by Control Rod Withdrawal								
Evaluation of this event is pending design data/information.								

(1) Per the IAEA SSG-26 methodology, external radiation dose from beta emitters that are released and unshielded to a worker who is close by (i.e., 1 meter) is accounted for in the TEDE and is presented in the fourth column. Before adding the dose contribution for the worker to get a total effective (whole body) dose, a tissue weighting factor is applied.

(2) Though skin contamination equivalent from ungloved work with debris is a radiological dose pathway prescribed by the IAEA SSG-26 Appendix I guidance, it is reasonably assumed that workers involved in handling radioactive material after an accident would wear the appropriate protective clothing including gloves. Therefore, skin contamination equivalent skin dose is presented for information only and is not converted to effective dose and assumed to be a contributor to the TEDE for the accident.

(3) Small inaccuracy due difference between a rounded-off numerical values and the actual values.

Table 4-26 also shows that the radiological dose consequences from loss of containment accidents, not caused by highway accidents, are very low whether the containment was assumed to be pressurized or not (see BRA 7 and BRA 8). The results show that the radiological dose consequences from fire events, not caused by highway accidents, are very low whether the fire originates from within or outside of the transportation module (see BRA 1 and BRA 2). The results also show that fire as a radiological release mechanism is not as important as mechanical impact by comparing the dose consequence of BRA 3, which is a hard impact without fire, to the dose consequences of BRA 5H which is a hard impact that results in fire. The dose consequences for these two bounding representative accidents are nearly the same. If the dose consequences of same two bounding representative accidents are compared to the radiological dose consequences from BRA 6, which is a collision with a tanker carrying flammable material, it can be seen that the dose consequences are only slight greater for the tanker collision. A collision with a tanker carrying flammable material results in the largest fire that can be postulated. Accordingly, it appears that fire as a radiological release mechanism is not nearly as dominant a factor as mechanical impact.

However, the results presented above should be considered incomplete given that the exposure due to one important dose pathway has not yet been calculated, though it is not clear its contribution would change the risk insights significantly. As explained in Section 4.6.3.1, the radiological dose from unreleased material in which there has been a loss or degradation in the radiation shielding is not addressed in this report because there is not enough design information to calculate the dose contribution from this dose pathway. The primary shielding is the reactor vessel itself and the empty shield tank around the reactor vessel, though there is tungsten shielding beside the reactor vessel. It is judged that the reactor's reactor vessel will remain largely intact, even after a transportation accident involving severe impact. The most likely significant degradation is assumed to be dents, bends, and other distortions that possibly create fissures through which radiation streaming may then be possible. The radiological dose contribution from this dose pathway will be addressed in a future update of the report.

4.7 PRA Baseline Results and Comparison to the Risk Evaluation Guidelines

This section provides summaries of radiological risk for each of the bounding accidents and compares that the risk to the risk acceptance guidelines presented in Section 3, Table 3-7 of this report. Tables are provided for each bounding representative accident that presents: (1) the estimated frequency of the accident based on one transport in a year using the assumed route, (2) the estimated radiological dose consequences of the accident in rem, and (3) the risk limit from the proposed risk evaluation guidelines in terms of likelihood and consequence. The radiological dose results are broken down into the contribution from the three different types of MAR: (1) the TRISO fuel itself, (2) the radiological material that diffused into the core structure such as the core compacts during operation, and (3) and radiological material that condensed or plated-out material in the primary reactor cooling system during operation. The proposed evaluation guidelines from Table 3-7 of this report are presented for each bounding representative accident in terms of the applicable accident likelihood and consequence limits for comparison to calculated accident likelihood and consequence. The acceptability of the risk results is indicated in the table based on the comparison.

4.7.1 Fire Only that Originates Inside Transport Container – BRA 1 Risk Results

BRA 1 is a fire that originates inside the transport container. It is a general fire that originates from such sources as an electrical cable fault, propagates to the package, and ignites combustible material associated with the package. It includes an oil or grease fire that is ignited from a hot surface or electrical fault. All MAR (i.e., the TRISO fuel itself, radiological material diffused into the core during operation, radiological material that has condensed or plated out in the reactor containment boundary) is protected from the direct effects of a fire by the shielding vessel or the reactor pressure vessel and coolant boundary. Due to the limited size of the fire, failure of the reactor containment boundary and release of materials is not postulated for this event. Therefore, the radiological dose consequence for this bounding representative accident was determined to 0 rem without performing consequence analysis as presented in Table 4-27. The estimated frequency of this event is 9.0E-07 per year assuming one trip a year based on likelihood determination presented in Section 4.5. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to acceptable even without comparing the risk to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-27. Risk Results Comparison for BRA 1 – Fire Only
that Originates Inside Transport Container

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	0	0	From Table 4-20	≥25 and <750 rem TEDE for a member of the public
Core Structure	0	0		
Cooling System	0	0		≥100 and <750 rem TEDE for a worker
Total Dose	**0**	**0**		
Accident Frequency (assuming one trip per year)			9.0E-07	≤1E-06 and >5E-07
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.2 Fire Only that Originates Outside Transport Container – BRA 2 Risk Results

BRA 2 is a diesel fuel fire that originates outside the transport container and propagates into the transport container and ignites combustible material in the transport container which damages the package. The quantity of diesel fuel assumed should be limited to the maximum possible fuel in transporter fuel tanks (e.g., 300 gallons). The estimated frequency of this accident as presented in Table 4-28 is extremely unlikely (2E-06 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. The radiological dose consequence for this bounding representative accident was determined to be very low to the worker (1.4E-03 rem) and the public (4.2E-04) based on dose consequence analysis presented in Section 4.6. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-28. Risk Results Comparison for BRA 2 – Fire Only
that Originates Outside Transport Container

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	0	0	From Table 4-20	≥5 and <25 rem TEDE for a member of the public
Core Structure	1.1E-03	2.1E-04		
Cooling System	1.4E-03	2.1E-04		≥25 and <100 rem TEDE for a worker
Total Dose	**2.5E-03**	**4.2E-04**		
Accident Frequency (assuming one trip per year)			2.0E-06	≤1E-04 and >1E-06
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.3 Hard Impact Road Accident – BRA 3 Risk Results

BRA 3 is a hard impact accident and includes impact with heavy vehicles and solid unyielding objects (e.g., concrete abutment or a rock embankment), and drops to a lower elevation (e.g., drop from a bridge), and rollovers which can result in hard impact of the asphalt or concrete roadway. The estimated frequency of this accident as presented in Table 4-29 is very unlikely (7.1E-05 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. The radiological dose consequence for this bounding representative accident is determined to be 198 rem to the worker and 34.6 rem the public based on dose consequence analysis presented in Section 4.6. The greatest contribution to the dose, by far, was the contribution from release of the TRISO fuel itself which is primarily gases. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to unacceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report without compensatory measures.

Table 4-29. Risk Results Comparison for BRA 3 – Hard Impact Road Accident

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	197	34.4	From Table 4-20	≥5 and <25 rem TEDE for a member of the public
Core Structure	5.7E-01	1.1E-01		
Cooling System	3.4E-01	5.1E-01		≥25 and <100 rem TEDE for a worker
Total Dose	**198**	**34.6**		
Accident Frequency (assuming one trip per year)			7.1E-05	≤1E-04 and >1E-06
COMPARISON TO RISK EVAULATION GUIDELINE				Unacceptable

4.7.4 Medium Impact Road Accident – BRA 4M Risk Results

BRA 4M is a less than a hard impact (i.e., a medium impact) highway accident that results in release of some radiological material and loss shielding. These medium impact accidents are defined are a severe collision with a light vehicle. The estimated frequency of this accident as presented in Table 4-30 is very unlikely (9.7E-05 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. The radiological dose consequence for this bounding representative accident is determined to be very low (3.9E-02 rem to the worker and 6.8E-03 rem the public) based on dose consequence analysis presented in Section 4.6. The radiological dose is based on release radiological material diffused into core structure such as the compacts and radiological material condensed or plated out in the reactor containment boundary. No release from the TRISO fuel is postulated. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-30. Risk Results Comparison for BRA 4M – Medium Impact Road Accident

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	0	0	From Table 4-20	≥5 and <25 rem TEDE for a member of the public ≥25 and <100 rem TEDE for a worker
Core Structure	2.8E-02	5.3E-03		
Cooling System	1.0E-02	1.5E-03		
Total Dose	**3.9E-02**	**6.8E-03**		
Accident Frequency (assuming one trip per year)			9.7E-05	≤1E-04 and >1E-06
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.5 Light Impact Road Accident – BRA 4L Risk Results

BRA 4L is less than a hard impact (i.e., light impact) highway accident that results in no release of radiological material or loss of shielding. These light impact accidents are defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment) or impact that is not severe with a light vehicle (e.g., results in property damage only). The estimated frequency of this accident as presented in Table 4-31 is very unlikely (3.3E-04 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. These accidents are defined to results in no release of radiological material or loss of shielding. Therefore, the radiological dose consequence for this bounding representative accident was determined to 0 rem without performing consequence analysis. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to acceptable even without comparing the risk to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-31. Risk Results Comparison for BRA 4L – Light Impact Road Accident

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	0	0	From Table 4-20	≥1 and <5 rem TEDE for a member of the public ≥5 and < 25 rem TEDE for a worker
Core Structure	0	0		
Cooling System	0	0		
Total Dose	**0**	**0**		
Accident Frequency (assuming one trip per year)			3.3E-04	≤1E-03 and >1E-04
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.6 Hard Impact Accident and Ensuing Fire – BRA 5H Risk Results

BRA 5H is a hard impact highway accident and subsequent fire that results in release of radiological material and loss of shielding. Hard impact accidents are defined to be heavy vehicle collisions, impacts with non-yielding objects, rollovers/overturns, and drops to lower elevation (i.e., like BRA 3 but BRA 5H includes fire). The estimated frequency of this accident as presented in Table 4-32 is extremely unlikely

(2.6E-08 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. The radiological dose consequence for this bounding representative accident is determined to be 198 rem to the worker and 34.6 rem the public based on dose consequence analysis presented in Section 4.6. The greatest contribution to the dose, by far, was the contribution from release of the TRISO fuel itself which is primarily gases. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-32. Risk Results Comparison for BRA 5H – Hard Impact Accident and Ensuing Fire

Accident Risk	**Worker Dose (rem TEDE)**	**Public Dose (rem TEDE)**	**Accident Frequency (per year)**	**Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report**
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	197	34.4	From Table 4-20	≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker
Core Structure	5.7E-01	1.1E-01		
Cooling System	3.4E-01	5.1E-02		
Total Dose	**198**	**34.6**		
Accident Frequency (assuming one trip per year)			2.6E-08	≤5E-07
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.7 Medium Impact Accident and Ensuing Fire – BRA 5M Risk Results

BRA 5M is a medium impact highway accident and ensuing fire that results in release of some radiological material and loss shielding. These medium impact accidents are defined are a severe collision with a light vehicle. The estimated frequency of this accident as presented in Table 4-33 is extremely unlikely (5.9E-07 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. The radiological dose consequence for this bounding representative accident is determined to be very low (3.9E-02 rem to the worker and 6.9E-03 rem the public) based on dose consequence analysis presented in Section 4.6. The radiological dose is based on release radiological material diffused into core structure such as the compacts and radiological material condensed or plated out in the reactor containment boundary. No release from the TRISO fuel is postulated. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-33. Risk Results Comparison for BRA 5 – Medium Impact Accident and Ensuing Fire

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	0	0	From Table 4-20	≥25 and <750 rem TEDE for a member of the public ≥100 and <750 rem TEDE for a worker
Core Structure	2.9E-02	5.3E-03		
Cooling System	1.0E-02	1.6E-03		
Total Dose	**3.9E-02**	**6.9E-03**		
Accident Frequency (assuming one trip per year)			5.9E-07	≤1E-06 and >5E-07
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.8 Collision with a Tanker Carrying Flammable Material and Ensuing Fire – BRA 6 Risk Results

BRA 6 is a collision with a tanker carrying flammable material that leads to fire. The estimated frequency of this accident as presented in Table 4-34 is extremely unlikely (7.1E-08 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. The radiological dose consequence for this bounding representative accident is determined to be 199 rem to the worker and 34.8 rem the public based on dose consequence analysis presented in Section 4.6. The greatest contribution to the dose, by far, was the contribution from release of the TRISO fuel itself which is primarily gases. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-34. Risk Results Comparison for BRA 6 – Collision with a Tanker Carrying Flammable Material and Ensuing Fire

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	198	34.6	From Table 4-20	≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker
Core Structure	6.8E-01	1.3E-01		
Cooling System	4.0E-01	6.1E-02		
Total Dose	**199**	**34.8**		
Accident Frequency (assuming one trip per year)			7.1E-08	≤5E-07
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.9 Loss of Non-Pressurized Reactor Containment Boundary – BRA 7 Risk Results

BRA 7 is a non-pressurized loss of the reactor containment boundary not caused by a road accident but rather by human error and failures of containment features. The estimated frequency of this accident as presented in Table 4-35 is low (1.3E-03 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. The radiological dose consequence for this bounding representative accident was determined to be very low to the worker (4.5E-05 rem) and the public (6.8E-06 rem) based on dose consequence analysis presented in Section 4.6. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-35. Risk Results Comparison for BRA 7 – Loss of Non-Pressurized Reactor Containment Boundary

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	0	0	From Table 4-20	≥100 mrem and <1 rem TEDE for a member of the public ≥500 mrem <5 rem TEDE for a worker
Core Structure	0	0		
Cooling System	4.5E-05	6.8E-06		
Total Dose	**4.5E-05**	**6.8E-06**		
Accident Frequency (assuming one trip per year)			1.3E-03	≤1E-02 and >1E-03
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.10 Loss of Pressurized Reactor Containment Boundary – BRA 8 Risk Results

BRA 8 is loss of pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features. The estimated frequency of this accident as presented in Table 4-36 is low (1.3E-03 per year assuming one trip a year) based on accident data and the likelihood determination presented in Section 4.5. The radiological dose consequence for this bounding representative accident was determined to be very low to the worker (1.8E-03 rem) and the public (2.7E-04 rem) based on dose consequence analysis presented in Section 4.6. Accordingly, the risk of this TNPP transportation bounding representative accident is determined to acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report.

Table 4-36. Risk Results Comparison for BRA 8 – Loss of Pressurized Reactor Containment Boundary

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	0	0	From Table 4-20	≥100 mrem and <1 rem TEDE for a member of the public ≥500 mrem <5 rem TEDE for a worker
Core Structure	0	0		
Cooling System	1.8E-03	2.7E-04		
Total Dose	**1.8E-03**	**2.7E-04**		
Accident Frequency (assuming one trip per year)			1.3E-03	≤1E-02 and >1E-03
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

4.7.11 Criticality Event Involving Drop into a Body of Water – BRA 9 Risk Results

BRA 9 is a highway accident that leads to a drop of the TNPP transportation package into a body of water that results in criticality caused addition of moderator and change in core geometry.
The accident frequency estimated for the assumed route ranges from 2.1E-06 per year estimated using route specific GIS data and 5.1E-09 per year using submersion accident from the national MCMIS accident database. The two different approaches and along with pros or cons of using each are discussed in Section 4.5.3.1.2. Using the GIS approach, the route is examined using GIS to identify locations along the assumed route where a body of water sufficient to submerge the reactor vessel and steep slope from the road to the body existed so that if a truck had an accident at that location and left the road, then it is assumed to end up in the body of water. The total length of those locations is multiplied by accident frequency for the assumed route to generate an estimate for this accident. The approach is described in detail in Section 4.5.1.4. The advantage of this approach is that it is route-specific but is likely conservative given assumption that were made which are included in the list provided in Section 4.5.4.

Using the other approach to estimate the frequency of this accident the proportion of "immersion/partial immersion" events to total number of large truck interstate accidents nationwide is provided is multiplied by the route specific accident failure rate as described in Section 4.2.4. The actual rate is judged to be between 2.1E-06 per year and 5.1E-09 per year and likely less than 5E-07 per year as presented in Table 4-37.

The radiological dose consequence for this bounding representative accident was not calculated because: (1) the accident frequency was judged to be beyond extremely unlikely, (2) the radiological dose consequence analysis would be a difficult to perform based on current information, and (3) even with better information the result would likely be uncertain based on the uncertainty modeling issues associated with inputs of the analysis. Accordingly, for this report, the dose consequences of a flood criticality accident were judged to be unacceptable without performing a consequence analysis. Using these bases, the risk of this TNPP transportation bounding representative accident is determined to acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 3-7 of this report based on the estimated low frequency of the accident (i.e., < 5E-07 per year).

Table 4-37. Risk Results Comparison for BRA 9 – Criticality Event Involving Drop into a Body of Water

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable proposed Risk Evaluation Guidelines from Table 3-7 of this report
Accident Consequence (Radiological dose by MAR contribution from Table 4-26)				
TRISO Fuel	—	—	From Table 4-20	≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker
Core Structure	—	—		
Cooling System	—	—		
Total Dose	(see note)	(see note)		
Accident Frequency (assuming one trip per year)			<5E-07	≤5E-07
COMPARISON TO RISK EVAULATION GUIDELINE				Acceptable

Note: For this report, the dose consequences of flooded criticality accident were judged to be unacceptable without performing a consequence analysis for reasons discussed in Section 4.7.11.

4.7.12 Criticality Event Caused by Control Rod Withdrawal – BRA 10 Risk Results

BRA 10, which is control rod withdrawal caused by impact from a road accident that results in criticality. This has not yet been developed because of insufficient design information and will be included in a revision to this report.

4.7.13 Summary Risk Results for Bounding Representative Accidents

Table 4-38 provides a summary of the risk results for the bounding representative accidents and indicates whether the risk was determined to be acceptable compared to the proposed risk evaluation guidelines presented in Table 3-7 of this report. As previously explained the accident frequency is presented on a per-year basis assuming one transport occurs in a year.

Table 4-38. Risk Summary of Bounding Representative Accidents (2 sheets total)

ID	Descriptions	Accident Frequency per year	Radiological Dose Consequences		Meet Proposed Evaluation Guidelines
			Worker (rem TEDE)	Public (rem TEDE)	
BRA 1	Fire-only event that originates inside the transport container.	9.0E-07	0	0	Acceptable
BRA 2	Diesel fuel fire-only event that originates outside the transport container and propagates into the transport container and ignites combustible material in the transport container which damages the package.	2.0E-06	2.5E-03	4.2E-04	Acceptable
BRA 3	Hard impact highway accident that leads to release of radioactive material and loss of shielding. Includes impact with heavy vehicles and unyielding objects (e.g., concrete abutments or rock embankments), significant drops to lower elevation, or rollovers.	7.1E-05	198	34.6	Unacceptable
BRA 4M	Less than a hard impact highway accident that results in release of some radiological material and loss of shielding. Medium impact that involves a severe collision with a light vehicle (e.g., one that results in fatality and/or injury).	9.7E-05	3.9E-02	6.8E-03	Acceptable
BRA 4L	Less than a hard impact highway accident that results in no release of radiological material or loss of shielding. Light impact such as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment) or impact that is not severe with a light vehicle (e.g., results in property damage only).	3.3E-04	0	0	Acceptable

Table 4-38. Risk Summary of Bounding Representative Accidents (2 sheets total)

ID	Descriptions	Accident Frequency per year	Radiological Dose Consequences		Meet Proposed Evaluation Guidelines
			Worker (rem TEDE)	Public (rem TEDE)	
BRA 5H	Hard impact highway accidents (i.e., equivalent to the impacts defined by BRA 3) that result in fire with exception of collision with a tanker carrying flammable material.	2.6E-08	198	34.6	Acceptable
BRA 5M	Medium impact highway accidents (i.e., severe collision with a light vehicle that leads to a fatality or injury) that results in fire.	5.9E-07	3.9E-02	6.9E-03	Acceptable
BRA 6	Collision with a tanker carrying flammable material that leads to fire.	7.1E-08	199	34.8	Acceptable
BRA 7	Loss of non-pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.	1.3E-03	4.5E-05	6.8E-06	Acceptable
BRA 8	Loss of pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.	1.3E-03	1.8E-03	2.7E-04	Acceptable
BRA 9	Addition of moderator and a change in core geometry caused by a drop into body of water that results in criticality.	<5E-07	(3)	(3)	Acceptable
BRA 10	Control rod withdrawal caused by impact from a road accident that results in criticality.	(4)	(4)	(4)	(4)

Notes:
(1) It is assumed that one transport occurs in a year.
(2) Risk is considered unacceptable without application of compensatory actions.
(3) Assumed to unacceptable without performing a radiological dose consequences analysis based on the rational provided in Section 4.7.11.
(4) Evaluation pending design data/information.

4.8 Definition of Sensitivity Studies and Presentation of Results

The section defines the TNPP transportation PRA sensitivity studies that were performed, presents the sensitivity study results, and evaluates the results against the risk evaluation guidelines. The sensitivity studies address the impact of key assumptions and sources of uncertainty by varying parameters and then determining the effects of those changes on the risk results. Section 4.8.1 defines the sensitivity studies performed, Section 4.8.2 presents and evaluates the results against the risk evaluation guidelines. Section 4.8.3 summarizes the insights gained from the sensitivity studies.

4.8.1 Definition of Sensitivity Studies

Selection and definition of the sensitivity cases to be performed will be based on: (1) work that is currently being performed by PNNL but is not yet published that identifies important sources of modeling uncertainty for a microreactor transportation PRA, (2) specific lists of assumptions and bases used in the Project Pele PRA presented in Sections 4.4.2.2, 4.5.4 and 4.6.4 of the report, and (3) the ability of the sensitivity case to provide estimates of the quantitative impacts of selected compensatory actions on TNPP transportation risk. The sensitivity studies might be limited to just BRA 3 because it is the only bounding representative accident to exceed the proposed risk evaluation guidelines presented in Table 3-7 of this report.

Key sources of modeling uncertainty judged to have the potential to significantly impact the radiological dose consequence results from TNPP transportation accidents based on work performed by PNNL, but is not yet fully published include:

- The estimation of source term factors for the various kinds of accidents (i.e., the damage ratio, the airborne release fraction, the respirable fraction, and leak path factor)
- Release fractions during operations of radiological material that is diffused into the reactor core and core structures such as the fuel compacts.
- Duration of radionuclide decay of the core before transport

Sensitivity studies that address a subset, combinations, or all of these factors cited above are proposed in an update of this report. Based on unpublished work of more generic designs, it is judged that the uncertainty associated with estimation of source term factors has the greatest potential to impact the radiological dose consequence results followed by the release fractions during normal operation and duration of radionuclide decay of the core before transport.

Another source of information for defining sensitivity cases are the specific lists of assumptions and bases used in the PRA presented in Sections 4.4.2.2, 4.5.4, and 4.6.4 of the report. Section 4.5.4 lists the assumption that were used to estimate the likelihood of TNPP transportation accidents. Assumptions could be tested that lower or raise the frequency of a bounding representative accident. Section 4.6.4 lists the assumptions that were used to estimate the radiological consequences of TNPP transportation accidents. Assumptions could be tested that lower or raise the consequence of a bounding representative accident. Section 4.4.2.2 lists the assumptions that were used perform the hazard analysis. Assumptions could be tested that lower or raise the frequency or consequences of a bounding representative accident by changing the definition of the accidents. Changes to accident definitions could affect the MAR assumed to be involved in the accident, change the energy sources or release mechanisms assumed to be involved in the accident, or change other factors.

A third set of sensitivity cases will be defined as a way to estimate the impacts of selected compensatory measures. A summary of possible compensatory measures is presented in Section 5.2 and will be reviewed for candidate sensitivity cases. For example, one of the suggested compensatory measures is to ship during periods of low traffic such as at night. If the ratio of low-traffic to average-traffic, which the TNPP PRA accident frequencies are based on, could be estimated then, the accident frequencies might be reduced. This could be particularly important for BRA 3 which could meet the proposed risk evaluation guideline presented in Table 3-7 of this report, if the frequency of the accident was reduced.

4.8.2 Presentation of Sensitivity Study Results

The sensitivity study results will be included in a revision to this report.

4.8.3 Insights from Sensitivity Studies

The insights from the sensitivity studies will be included in a revision to this report.

4.9 Risk Insights for Baseline PRA and Sensitivity Studies

As shown in Table 4-26, all bounding representative accidents meet the proposed risk evaluation guidelines presented in Table 3-7 of this report with one exception. In most cases, the radiological dose consequences are significantly lower than the dose consequence limit for the likelihood range for which the accident falls and lower than 0.1 rem. For two bounding representative accidents (i.e., BRA 5H and BRA 6), the radiological dose consequences are much higher than 0.1 rem, but the estimated accident frequency is well beyond unlikely. Accordingly, to the proposed risk evaluation guidance, the risk from these accident scenarios would generally be acceptable regardless of its radiological dose consequence because of their low frequency (i.e., the frequency of the accident is <5E-07 per year). The criticality event caused by a drop into a body of water was determined to be <5E-07 per year. It is assumed that the frequency of a criticality event caused by control rod withdrawal from the impact of a road accident will be rendered <5E-07 per year based on the design of the TNPP and package. However, demonstration of this assumption has not yet occurred.

The exception to characterization above is BRA 3 which is hard highway accident that results in severe impact and did not meet the proposed risk evaluation guidelines presented in Table 3-7 of this report. The worker dose exceeded the radiological dose limit, given the frequency of the accident, by about a factor of 2, and the public dose exceeded the radiological dose limit, given the frequency of the accident by about 40%. It is notable that the public radiological dose consequences for this worst-case conservatively evaluated accident is just 40% above the risk evaluation guidelines. However, this exceedance means that compensatory measures as described in Section 5.2 of this report are needed to prevent this accident.

However, an important caveat to summarization of the risk results discussed above, is the fact that the radiological dose consequences do not yet include the contribution of an important dose pathway, and therefore, the results should be considered incomplete. As explained in Section 4.6.3.1, the radiological dose from unreleased material in which there has been a loss or degradation in the radiation shielding is not addressed in this report because there is not enough design information to calculate the dose contribution from this dose pathway. The primary shielding is considered to be the reactor vessel itself but includes the empty shield tank around the reactor vessel, the tungsten shielding (e.g., at the ends of the reactor vessel), and other external shielding. It is judged that reactors reactor vessel will remain largely intact even after a transportation accident involving severe impact. The most likely significant degradation is assumed to dents, bends, and other distortions that possibly create fissures making radiation streaming possible. The radiological dose contribution from this dose pathway will be addressed in a future update of the report.

That said, it is not clear that the contribution from the excluded dose pathways will change the risk insights from this study significantly. Meaningful damage to the radiation shielding is judged only to occur as a result of severe impact which could occur in BRA 3, BRA 5H, and BRA 6. As discussed above, based on the proposed risk evaluation guidance, the risk from BRA 5H and BRA 6 would generally be acceptable regardless of its radiological dose consequence because of their low frequency (i.e., below 5E-07 per year). Moreover, BRA 3 already exceeds the proposed risk evaluation guidelines, so an incremental increase in radiological dose due to the contribution from direct radiation exposure from loss of shielding may not change the risk insights. As mentioned above, compensatory measures are already required to reduce the risk from BRA 3.

As mentioned above, BRA 3, which is hard highway accident that results in severe impact, is the only accident that did not meet the proposed risk evaluation guidelines presented in Table 3-7 of this report. The radiological dose limit for this accident (as shown in Table 4-26) at an accident frequency of 7.1E-05 per year (assuming one shipment per year) is ≥5 rem and <25 rem TEDE for a member of the public ≥25 rem and <100 rem TEDE for a worker using the proposed risk evaluation guidelines. If through application of normal and compensatory actions, it can be shown the accident frequency is lower than 1E-06 per year, than the dose consequences are well below the radiological dose limit proposed in Table 3-7 of <750 rem. Though the accident frequencies for transportation of the TNPP were assumed to be random and based on large truck accident data, the actual accident frequencies are likely lower if the impact of controls that will be used for such a significant transport and compensatory measures could be quantified.

There is a possibility that the impact from a highway accident could be hard enough fail the restraining and locking mechanism that keeps the control rods from withdrawing (The current design for demonstration unit does not include transportation poison rods.) This would create a control rod withdrawal criticality event (I.e., BRA 10) which has not yet been analyzed because its design is not sufficiently advanced to estimate the likelihood of such an event.

Note that risk insights from the sensitivity studies will be determined and included in a revision to this report.

4.10 References

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Appendix I – Essential Plans for Deployment.
Appendix II – Engineering Drawings.
Appendix III – Engineering Documentation and Analyses.
Appendix IV – Safety in Design Bases and Reports.

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5. DEFENSE-IN-DEPTH AND SAFETY MARGIN CONCERNS

The U.S. Nuclear Regulatory Commission (NRC) regulations for nuclear power plants require that important risk-informed decisions based on comparison of bounding risk estimates to risk acceptance guidelines also be supported by a philosophy of defense-in-depth and safety margin. The same should be expected for transportation of Transportable Nuclear Power Plant (TNPP) packages. This section defines the defense-in-depth philosophy and includes discussion of safety features and controls that are credited in the TNPP transportation Probabilistic Risk Assessment (PRA). Of special note is the identification of potential compensatory measures used to offset the residual risk associated with TNPP package transport and uncertainty associated with risk calculations. This section also describes the philosophy of incorporating safety margin into design and operation, and how both these philosophies work together with risk assessment and can even be demonstrated using a quantitative risk assessment approach. This is normally done by demonstrating that sufficient conservatism is preserved in the design parameters such that reliability and effectiveness are reasonably ensured against the most demanding challenge. Specifically, for the TNPP transportation PRA, this applies to ensuring that there is a sufficient safety margin to account for modeling and data uncertainties. Accordingly, Section 5.1 discusses the defense-in-depth philosophy as it supports risk-informed decisionmaking in concert with results and insights from the TNPP transportation PRA, and Section 5.2 discusses associated identification of suggested potential compensatory measures. Section 5.3 discusses the safety margin philosophy as it supports risk-informed decisionmaking in concert with results and insights from the TNPP transportation PRA and defense-in-depth.

5.1 Defense-in-Depth Philosophy

Defense-in-depth is a design and operational philosophy that calls for multiple layers of protection to prevent and mitigate accidents as described by the NRC in the RIDM report (*Risk-Informed Decisionmaking for Nuclear Material and Waste Applications* [NRC 2008]) cited in Section 3.1 of this report. It includes the use of controls, multiple physical barriers to prevent release of radiation, redundant and diverse key safety functions, and emergency response measures. The primary element of the defense-in-depth philosophy for this application of the exemption process (10 CFR 71.12, "Specific exemptions") is the fact that the TNPP transportation risk is quantified and shown to be low, but in addition compensatory actions will be administered that reduce the risk to the worker and the public and associated uncertainty through preventative and mitigative features.

NRC Regulatory Guide (RG) 1.174, Revision 3 (*An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis* [NRC 2018]), states about the defense-in-depth philosophy that the key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. The principles in RG 1.174 are generally relevant to different kinds of risk-informed applications. The following principles of the defense-in-depth philosophy extracted from RG 1.174 are dispositioned below for TNPP transportation.

1. Preserve a reasonable balance among the layers of defense.

   For TNPP transportation, compensatory actions (discussed in Section 5.2) will still be applied though the risk is shown to be low. Also, the design will be robust and though it may not meet all the requirements in 10 CFR 71.55 ("General requirements for fissile material packages") after hypothetical accident conditions (HAC), it is expected to meet many or most of the requirements.

2. Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.

   For TNPP transportation, the design goal for the TNPP is to prevent release of radiological material, loss of shielding, and criticality without overreliance programmatic controls and compensatory measures. The TNPP transportation PRA is performed to show that the risk of TNPP transportation is relatively low even when programmatic activities such as compensatory measures are not credited.

3. Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.

   Redundancy, independence, and diversity are concepts that are more relevant to an operating reactor with redundant active systems. However, an active parameter monitoring (though this design is not complete) could be used to prevent or mitigate the risk of certain kinds of TNPP accidents. Uncertainty issues are explicitly addressed using sensitivity studies of the TNPP PRA results to show the impact that different input parameters have on risk results.

4. Preserve adequate defense against potential common cause factors (CCFs).

   Defense-in-depth for TNPP transportation does not rely much on system redundancy because there are very few active and no multiple systems needed for transportation given the TNPP is in shutdown. Therefore, the notion of protecting against CCFs is not as relevant for TNPP transportation as it is for operating nuclear power plants.

5. Maintain multiple fission product barriers.

   The tri-structural isotropic (TRISO) fuel itself is a fission product barrier in addition to the containment afforded by the reactor vessel and containment isolation mechanisms. The reactor is in a shutdown state, so there is no concern for very high temperatures that would challenge the TRISO fuel in a TNPP transportation accident. The contribution to worker and public radiological dose in a transportation accident from other material at risk (MAR [non-TRISO fuel radiological material]) is significantly less.

6. Preserve sufficient defense against human errors.

   The possibility of human error is explicitly addressed in the TNPP transportation PRA as is the risk associated with TNPP accidents that are initiated by human error (e.g., packaging errors).

Additionally, the insights from the TNPP transportation PRA can be used to implement administrative controls on operator actions to prevent error.

As discussed in Section 3, NRC proposes guidance in the RIDM report (NRC 2008). The RIDM report states that for medium-risk and high-risk activities, defense-in-depth measures should consider the concepts listed below. These concepts (which in some cases overlap and are likely drawn from the RG 1.174 concepts assessed above) are dispositioned below even though the risk from TNPP transport is shown by the TNPP PRA to be a generally shown not to be a high-risk activity.

1. Ensure key safety functions do not depend on a single element of design or operation.

   For TNPP transport, this is a possible weakness of the TNPP design if damage from a severe impact (e.g., collision with a heavy truck) leads to significant release of radiological material. Another weakness is that the current design of the demonstration unit does not include transportation poison rods as an additional mechanism to prevent a criticality event from a control insertion event as a result of severe impact. However, the PRA shows that the likelihood of TNPP accidents that produce the highest consequences are beyond extremely unlikely.

2. Use redundancy, diversity, and independence to improve reliability and/or avoid common-mode failure, when necessary, to ensure safety is maintained.

   Again, for TNPP transport, redundancy, independence, and diversity are concepts that are more relevant to an operating reactor with redundant active systems. Defense-in-depth for TNPP transportation does not rely much on system redundancy because there are no multiple and very few active systems that are relied on during transportation given the TNPP is in shutdown, therefore, the notion of protecting against CCFs is not as relevant for TNPP transportation as it is for operating nuclear power plants. However, an active parameter monitoring system (though this design is not complete) could be used to prevent or mitigate the risk of certain kinds of TNPP accidents.

3. Provide safety margins to address uncertainties in modeling or equipment performance.

   Discussion of safety margins to address uncertainties in modeling or equipment performance is provided in Section 5.3.

4. Conduct regulated activities at locations that facilitate protection of public and worker safety.

   Transportation will need to occur over public highways, but the assembly, packing, and disassembly of the TNPP will occur at locations where protection of public and worker safety is highly regulated.

5. Provide time for recovery operations.

   TNPP transport should include a recovery plan for possible transportation accidents and the transportation workers and personnel should be trained on the transportation plan. Quick recovery actions that minimize the risk of release to the public should be included in the transportation plan (e.g., setup of a safety perimeter to keep the public away from the point of release).

6. Ensure the design and operation have both accident prevention and mitigation measures.

   Accident prevention includes preventing release of radioactive material, preventing loss of shielding, and preventing a criticality in a TNPP accident which are addressed in the design. The transport of the TNPP will be supported by an escort forward and aft for the entire route who should be trained in emergency and recovery operations. This should include response to fire and impacts from natural phenomena.

7. Ensure the design includes at least two independent barriers to the uncontrolled release of radioactive material.

   The TRISO fuel itself is a fission product barrier in addition to the containment afforded by the reactor vessel and containment isolation mechanisms. The reactor is in a shutdown state, so there is no concern for very high temperatures that might challenge the TRISO fuel material discussed in the reactor or plated in the reactor coolant boundary in a TNPP transportation accident. The contribution to worker and public radiological dose in a transportation accident from other MAR (non-TRISO fuel radiological material) is significantly less. Moreover, the TNPP PRA shows that the risk from this activity is relatively low. The PRA shows that the likelihood of TNPP accidents that produce the highest consequences beyond extremely unlikely.

5.2 Identification of Potential Compensatory Measures

As described in Section 2.0, the preferred regulatory pathway was determined to be through the exemption process (10 CFR 71.12). Among the requirements to use the exemption process, the following is needed for package approval:

> Identification of compensatory measures such as administrative controls that protect the bases for the exemption by preventing or significantly reducing the likelihood of accident conditions that are outside of the analyzed configurations/conditions; and

This section discusses potential compensatory measures that are either explicitly credited in the TNPP transportation PRA as an underlying assumption in the baseline PRA or could be credited in conjunction with the results of the baseline PRA or as a defense-in-depth measure. Some compensatory measures are explicitly assessed by sensitivity studies presented in Section 4.8 and Section 4.9 to better understand their quantitative impact on the radiological dose consequences from a TNPP transportation accident. Note, however, that sensitivity study results are not presented in this initial report and will be included in a revision to this report.

A list of possible generic compensatory measures identified by the vendor is provided below; this list will be modified based on the results of the transportation PRA.[45]

- Escort the reactor forward and aft for the entire route. Army to provide escorts.
- Choose a route that avoids bodies of water.

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[45] From BWXT Final Design Report App I.2 Section 1010, page 85.

  - This will need to be balanced by the need to use the best quality of road (i.e., interstate highways).

- For bridges over bodies of water:

  - Conduct additional inspections as necessary of the bridges prior to shipping to verify condition.

  - Close bridge to other traffic while the reactor is on the bridge.

  - Reduce speed while crossing the bridge (e.g., 5 mph).

  - Schedule shipment to avoid high winds while on the bridge.

  - For bridges over navigable waterways, close waterway to traffic while reactor is on the bridge.

- Choose a route and schedule the shipment to avoid the potential for flash flooding.

- Ship at night to avoid other traffic.

- Avoid shipping during known times of high traffic volume.

- Conduct training for emergency responders along the route.

- Real time health monitoring. The planned Health Monitoring Instrumentation System (HMIS) will provide real time parameter monitoring of an TNPP package during transport and it is anticipated it will be designed to detect conditions signaling that a TNPP transportation accident (e.g., a leak from containment) has or could occur. Detection would be based on monitoring such parameters as high levels of airborne radioactivity or direct radiation, loss of pressure in the reactor containment boundary, increase in heat in the reactor containment boundary and rod control position. Real time monitoring systems is required for certain radioactive material shipments. For example, real time monitoring of railcars carrying spent nuclear fuel is required by the Association of American Railroads (AAR) Standard S-2043 (*Performance Specification for Trains Used To Carry High-Level Radioactive Material*).

The generic and specific compensatory measures that are explicitly credited in the TNPP transportation PRA as an underlying assumption will be further developed and included in a revision to this report. However, one example is the assumption that the TNPP shipment will be of sufficient weight that it will be subject to heavy haul permitting in each state through which it passes and may be subject to superload permitting in some states. Specific permitting requirements vary by state and in some cases may require specific measures that could be considered compensatory measures. However, to the extent these specific requirements exist, they are reflected in the highway accident rates presented in Section 4.5 and used in the TNPP transportation PRA.

Generic and specific compensatory measures that are explicitly credited in the TNPP transportation PRA sensitivity studies will also be further developed and included in a revision to this report. The results of these studies will provide suggestions for compensatory measures that should be implemented as defense-in-depth measures to reduce transportation risk and/or modeling uncertainty.

Additionally, generic or specific compensatory measures that are credited in the TNPP transportation PRA baseline or sensitivity studies will be further developed and included in a revision to this report. These compensatory measures will be recommended and should be implemented as defense-in-depth measures to reduce transportation risk and/or modeling uncertainty.

5.3 Safety Margin Philosophy

The RIDM report (NRC 2008) defines safety margin as a measure of the conservatism that is employed in a design or process to assure a high degree of confidence that it will perform a needed function. It can be defined as the probability or level of confidence that a design or process will perform an intended function. Sufficient safety margins should be maintained under any proposed regulatory change that relies on a risk-informed decision framework. This is typically done by demonstrating that sufficient conservatism is preserved in the design parameters, such that reliability and effectiveness are reasonably ensured against the most demanding challenge. An alternative approach often used is to demonstrate adherence to the acceptable codes and standards.

RG 1.174, states that sufficient safety margins are maintained when:

- Codes and standards or alternatives approved for use by the NRC are met.
- Safety analysis acceptance criteria in the final safety analysis report (FSAR) are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties.

Again, the guidance in RG 1.174 is applicable for risk-informed applications in general. As indicted above, one way to evaluate safety margin in a microreactor transportation package transportation risk assessment is to ensure that the codes and standards used in the analyses supporting the risk assessment have a high degree of technical quality. TNPP transportation PRA is not yet a well-developed methodology, and in fact, this application advances the state-of-practice, as approval of transportation packages of radiological material has primarily been performed by meeting the deterministic requirements in 10 CFR Part 71 ("Packaging and Transportation of Radioactive Material"). Accordingly, there is no standard for performing a TNPP transportation PRA. The need for a standard and supporting guidance will be addressed in Section 6 and included in a revision to this report. That said the techniques used in the TNPP transportation PRA are not overly challenging and make use of industry tools. For example, the five-factor formula approach used in source term development is commonly used across the DOE complex for determining the possible dose consequences of accidents in non-reactor nuclear facilities (DOE 2013). Also, the approach for determining the radiological dose from a given source term resulting from a TNPP transportation accidents comes from the Q System described in International Atomic Energy Agency (IAEA) Specific Safety Guide No. SSG-26, Appendix I (*Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material* [IAEA 2014]), which is the basis of the A_1 and A_2 values utilized in 10 CFR Part 71. This guidance is described in detail in Sections 4.4.2 and 4.6.3 of this report. The hazard analysis that was used to identify and define the bounding representative accident for TNPP transportation is based on techniques commonly used across the nuclear, chemical and petrochemical, and aerospace industries. The RIDM report (NRC 2008) discusses hazard analysis approaches as does DOE-STD-3009-2014 (*DOE Standard – Preparation of Nonreactor Nuclear Facility Documented Safety Analysis* [DOE 2014]).

From RG 1.174 as indicated above, the second way to evaluate safety margin in a TNPP transportation PRA to ensure that there is a sufficient safety margin in the analyses that support the PRA to account for modeling and data uncertainties. A Safety Analysis Report (SAR) and full testing of the TNPP package has not yet been performed. The deterministic design analyses that are performed to support the SAR that are used in the TNPP transportation PRA needs to employ a safety margin philosophy. Deterministic concerns include the finite element analyses used to determine the fragility of the package to severe impacts that result during a transportation accident and the thermal analyses used to determine the susceptibility of the package to fire that may also occur as part of the transportation accident. When this information becomes available, it can be used to improve the PRA and address the safety margin. Additionally, the safety margin is addressed by performing sensitivity studies as described in Section 4.8 that speak to the impact of deterministic inputs for which there is uncertainty.

Accordingly, commonly used tools and approaches were used in the TNPP transportation PRA, but as discussed above there is a need for a standard and guidance for performing a TNPP transportation PRA which is discussed in Section 6 of this report. However, given that this a first-of-kind endeavor with limitations and uncertainty in the inputs, the TNPP transportation PRA was developed using best judgment that erred on the side of conservativism in: (1) identification of TNPP transportation accidents, (2) estimation of accident likelihood and application of the accident data, (3) estimation of the accident consequences. The safety margin of the deterministic input to the TNPP transportation PRA that come from the design and SAR can be assessed when that information is available and by performing sensitivity studies to test the impact of the uncertainty in PRA inputs.

In summary, a self-evaluation found that defense-in-depth and safety margin philosophies were, in general, applied consistent with NRC guidance and the information available to perform this evaluation to development of the TNPP transportation PRA and to its application to regulatory approval of the TNPP transportation package.

5.4 References

10 CFR Part 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*. Accessed March 28, 2022, at https://www.nrc.gov/reading-rm/doc-collections/cfr/part071/index.html.

AAR Standard S-2043, *Performance Specification for Trains Used to Carry High-Level Radioactive Material*, American Association of Railroads, Washington, D.C.

DOE, 2014, *DOE Standard – Preparation of Nonreactor Nuclear Facility Documented Safety Analysis*, DOE-STD-3009-2014, U.S. Department of Energy, Washington, D.C., November 2014. Available at https://www.standards.doe.gov/standards-documents/3000/3009-astd-2014/@@images/file.

DOE, 2013, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, DOE-HDBK-3010-94, Reaffirmed 2013, U.S. Department of Energy, Washington, D.C. Available at https://www.standards.doe.gov/standards-documents/3000/3010-bhdbk-1994-v1.

IAEA, 2014, *Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2012 Edition)*, Specific Safety Guide No. SSG-26, International Atomic Energy Agency, Vienna, Austria. Available at https://www-pub.iaea.org/MTCD/publications/PDF/Pub1586web-99435183.pdf.

NRC, 2008, *Risk-Informed Decisionmaking for Nuclear Material and Waste Applications*, Revision 1, U.S. Nuclear Regulatory Commission, Washington, D.C., February 2008. Available at https://www.nrc.gov/docs/ML0807/ML080720238.pdf.

NRC, 2018, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, Revision 3, U.S. Nuclear Regulatory Commission, Washington, D.C., January 2018. Available at https://www.nrc.gov/docs/ML1731/ML17317A256.pdf.

6. TECHNICAL ADEQUACY OF TRANSPORTATION RISK ASSESSMENT

The technical adequacy of the transportation risk assessment will be addressed through the process of finalizing the proposed risk-informed methodology for the Project Pele Transportable Nuclear Power Plant.

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7. CONCLUSIONS

This report provides a demonstration implementation of Pacific Northwest National Laboratory's (PNNL's) proposed risk-informed regulatory plan (PNNL-33524, *Plan for Development and Application of Risk Assessment Approach for Transportation Package Approval of an MNPP for Domestic Highway Shipment* [Maheras et al. 2021]) for a hypothetical shipment of the Project Pele microreactor. This demonstration implementation can be used as a guide or template for the development of a hypothetical risk-informed exemption request to the U.S. Nuclear Regulatory Commission (NRC) by the Project Pele microreactor vendor for a one-time ground surface shipment. This report focuses on the Probabilistic Risk Assessment (PRA) that would be used to support an exemption process (10 CFR 71.12, "Specific exemptions") that was determined by PNNL to be the most feasible regulatory option for transportation of a Transportable Nuclear Power Plant (TNPP).

Accordingly, the demonstration implementation includes: (1) development of a workable PRA methodology, (2) development of needed risk evaluation guidelines for assessing the risk of TNPP PRA transport from NRC guidance, federal requirements, and examples of risk evaluation guidance from other applications, and (3) technical information, data, and example analyses that provide a potential template for a vendor to follow when making a request to the NRC. It also addresses the treatment of key assumptions and sources of modeling uncertainty and the concept of defense-in-depth and safety margin.

As shown in Table 4-26, the results of the report show that all bounding representative accidents meet the proposed risk evaluation guidelines presented in Table 3-7 of this report with one exception. In most cases, the radiological dose consequences are significantly lower than the dose consequence limit for the likelihood range for which the accident falls and lower than 0.1 rem. For two bounding representative accidents (i.e., BRA 5H and BRA 6), the radiological dose consequences are much higher than 0.1 rem, but the estimated accident frequency is well beyond unlikely. Accordingly, to the proposed risk evaluation guidance, the risk from these accident scenarios would generally be acceptable regardless of its radiological dose consequence because of their low frequency (i.e., the frequency of the accident is <5E-07 per year). The criticality event caused by a drop into a body of water was determined to be <5E-07 per year. It is assumed that the frequency of a criticality event caused by control rod withdrawal from the impact of a road accident will be rendered <5E-07 per year based on the design of the TNPP and package. However, demonstration of this assumption has not yet occurred.

The exception to the characterization above is BRA 3 which is a hard highway accident that results in severe impact and does not meet the proposed risk evaluation guidelines presented in Table 3-7 of this report. The worker dose exceeded the radiological dose limit, given the frequency of the accident, by about a factor of 2, and the public dose exceeded the radiological dose limit, given the frequency of the accident by about 40%. It is notable that the public radiological dose consequences for this worst-case conservatively evaluated accident is just 40% above the risk evaluation guidelines. However, this exceedance means that compensatory measures as described in Section 5.2 of this report are needed to prevent this accident.

An important caveat to summarization of the risk results discussed above is the fact that the radiological dose consequences do not yet include the contribution of an important dose pathway, and therefore, the results should be considered incomplete. As explained in Section 4.6.3.1, the radiological dose from unreleased material in which there has been a loss or degradation in the radiation shielding is not addressed in this report because there is not enough design information to calculate the dose contribution from this dose pathway even though it is judged to be minimal. However, it is not clear that the contribution from the excluded dose pathways will change the risk insights from this report significantly. Meaningful damage to the radiation shielding is judged only to occur as a result of severe impact which could occur in BRA 3, BRA 5H, and BRA 6. As discussed above, based on the proposed risk evaluation guidance, the risk from BRA 5H and BRA 6 would generally be acceptable regardless of its radiological dose consequence because of their low frequency (i.e., below 5E-07 per year). Moreover, BRA 3 already exceeds the proposed risk evaluation guidelines, so an incremental increase in radiological dose due to the contribution from direct radiation exposure from loss of shielding may not change the risk insights.

As mentioned above, BRA 3, severe impact accident (as shown in Table 4-29), is the only accident that did not meet the proposed risk evaluation guidelines presented in Table 3-7 of this report. The radiological dose limit for this accident (as shown in Table 3-7) at an accident frequency of 7.1E-05 per year (assuming one shipment per year) is ≥5 rem and <25 rem total effective dose equivalent (TEDE) for a member of the public ≥25 rem and <100 rem TEDE for a worker using the proposed risk evaluation guidelines. If through application of normal and compensatory actions it can be shown the accident frequency is lower than 1E-06 per year, then the dose consequences are well below the radiological dose limit proposed in Table 3-7 of <750 rem. Though the accident frequencies for transportation of the TNPP were assumed to be random and based on large truck accident data, the actual accident frequencies are likely lower if the impact of controls that will be used for such a significant transport and compensatory measures could be quantified.

An observation made during PRA development that should be noted even though it did not impact the PRA results concerns latent failures. Failures could occur as a result of TNPP transportation that do not lead to release of radiological material, loss of shielding, or criticality during transport but could have a future effect when the TNPP is operated. For example, if loss of passive heat transfer of decay heat during transportation of the TNPP package led to undetected degradation of the reactor (e.g., damage to materials that exceed their maximum allowable use threshold), then it could have a latent safety related consequence for the plant operation.

Key advantages of using the approach described in this report include: (1) increasing the likelihood of successfully obtaining regulatory transportation package approval, (2) informing the design on the relative risk significance of TNPP containment and shielding, and (3) informing the need for normal and compensatory measures.

Candidate sensitivity cases that will be performed for this report are discussed in Section 4.8 and will address sources of modeling uncertainty judged to have the potential to significantly impact the radiological dose consequence results from TNPP transportation accidents. Selection and definition of the sensitivity cases to be performed will be based on: (1) work that is currently being performed by PNNL (but not yet published) that identifies an important source of modeling uncertainty for a microreactor transportation PRA, (2) specific lists of assumptions and bases used in the PRA presented in Sections 4.4.2.2, 4.5.4 and 4.6.4 of the report, and (3) the ability of the sensitivity case to provide estimates of the quantitative impacts of selected compensatory actions on TNPP transportation risk.

Proposed compensatory measures needed or suggested to reduce the risk associated with TNPP transportation will be developed to support the 10 CFR 71.12 exemption process. However, these measures will be developed after further design information is available to update the PRA and applicable sensitivity studies have been performed.

A self-evaluation of the application defense-in-depth and safety margin philosophies to development of the TNPP transportation PRA and its use for regulatory approval of the TNPP transportation package as part of this study. In general, the evaluation found defense-in-depth and safety margin philosophies were applied consistent with the information available to perform this evaluation and the guidance in RIDM report (NRC 2008) and RG 1.174, Revision 3 (*An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis* [NRC 2018]).

In the next version of this TNPP PRA, greater available design detail will be incorporated to provide more accuracy and reduce uncertainty, and sensitivity studies will be performed as described Section 4.8. to provide decision making with a greater level of confidence about the estimated risk from transportation of an TNPP with its irradiated fuel.

7.1 References

10 CFR Part 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*. Accessed March 28, 2022, at https://www.nrc.gov/reading-rm/doc-collections/cfr/part071/index.html.

Maheras, S. J., S. M. Short, G. A. Coles, H. E. Adkins, J. R. Phillips, P. P. Lowry, C. A. Condon, and K. Banerjee, 2021, *Plan for Development and Application of Risk Assessment Approach for Transportation Package Approval of an MNPP for Domestic Highway Shipment*, PNNL-33524, Pacific Northwest National Laboratory, Richland, Washington, December 28, 2021.

NRC, 2008, *Risk-Informed Decisionmaking for Nuclear Material and Waste Applications*, Revision 1, U.S. Nuclear Regulatory Commission, Washington, D.C., February 2008. Available at https://www.nrc.gov/docs/ML0807/ML080720238.pdf.

NRC, 2018, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, Revision 3, U.S. Nuclear Regulatory Commission, Washington, D.C., January 2018. Available at https://www.nrc.gov/docs/ML1731/ML17317A256.pdf.

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8. APPENDICES

8.1 TNPP Inventory and Development of MAR

This appendix provides the results of the screening of the representative Transportable Nuclear Power Plant (TNPP) inventory[46] and development of the material at risk (MAR) based on this inventory.

As described in Section 4.2.2 of this report, Table 8.1-1 includes all radionuclides greater than 0.01% of its 10 CFR Part 71 A_2 value or greater than 1 millicurie for those nuclides without an assigned A_2 value for cooling periods of 7 days, 30 days, and 90 days after three years of operation.

Table 8.1-1. Prototype TNPP Radionuclide Inventory
(4 sheets total)

Isotope	PELE1 Total, 7 days (Ci)	PELE1 Total, 30 days (Ci)	PELE1 Total, 90 days (Ci)	Inclusion Basis
Ag-109m	1.92E+00	Exclude30	Exclude30	Ci
Ag-110	5.80E-01	5.45E-01	4.61E-01	Ci
Ag-110m	4.27E+01	4.00E+01	3.39E+01	A_2
Ag-111	1.28E+03	1.51E+02	5.67E-01	A_2
Ag-112	6.43E+00	Exclude30	Exclude30	Ci
Am-241	4.90E+00	5.39E+00	6.66E+00	A_2
Am-242	1.63E+00	2.14E-01	2.14E-01	Ci
Am-242m	2.15E-01	2.15E-01	2.15E-01	A_2
Am-243	1.27E-01	1.27E-01	1.27E-01	A_2
As-77	3.18E+01	1.67E-03	Exclude90	A_2
Ba-136m	4.12E+02	1.23E+02	5.20E+00	Ci
Ba-137m	2.35E+04	2.35E+04	2.34E+04	Ci
Ba-140	2.50E+05	7.17E+04	2.75E+03	A_2
Br-82	1.32E+01	Exclude30	Exclude30	A_2
Cd-113m	1.73E-02	1.73E-02	1.71E-02	A_2
Cd-115	1.02E+02	7.92E-02	Exclude90	A_2
Cd-115m	4.63E+01	3.24E+01	1.27E+01	A_2
Ce-139	4.15E-01	3.69E-01	2.73E-01	A_2
Ce-141	2.98E+05	1.83E+05	5.08E+04	A_2
Ce-143	1.03E+04	9.64E-02	Exclude90	A_2
Ce-144	2.96E+05	2.80E+05	2.42E+05	A_2
Cm-242	8.94E+02	8.11E+02	6.28E+02	A_2
Cm-243	1.39E-01	1.39E-01	1.39E-01	A_2
Cm-244	6.41E+00	6.40E+00	6.36E+00	A_2
Cs-132	4.44E-01	3.79E-02	Exclude90	A_2
Cs-134	1.55E+04	1.52E+04	1.44E+04	A_2
Cs-135	1.67E-01	1.67E-01	1.67E-01	A_2
Cs-136	3.72E+03	1.11E+03	4.69E+01	A_2
Cs-137	2.48E+04	2.48E+04	2.47E+04	A_2

[46] BWXT spreadsheet "B1.34-NuclideConcentrations(Ci)-Fuel.xlsx" provided on August 11, 2022.

Table 8.1-1. Prototype TNPP Radionuclide Inventory
(4 sheets total)

Isotope	PELE1 Total, 7 days (Ci)	PELE1 Total, 30 days (Ci)	PELE1 Total, 90 days (Ci)	Inclusion Basis
Eu-152	5.84E-01	5.82E-01	5.77E-01	A_2
Eu-154	5.59E+02	5.56E+02	5.49E+02	A_2
Eu-155	3.76E+02	3.72E+02	3.63E+02	A_2
Eu-156	8.24E+03	2.89E+03	1.87E+02	A_2
Eu-157	3.94E-01	Exclude30	Exclude30	Ci
Ga-72	2.42E-01	Exclude30	Exclude30	A_2
Gd-153	6.14E-01	5.74E-01	4.83E-01	A_2
Gd-159	3.50E-01	Exclude30	Exclude30	A_2
Ge-77	1.55E-02	Exclude30	Exclude30	A_2
H-3	9.70E+01	9.66E+01	9.58E+01	A_2
I-130	1.09E-01	Exclude30	Exclude30	Ci
I-131	9.83E+04	1.35E+04	7.59E+01	A_2
I-132	5.88E+04	4.06E+02	Exclude90	Ci
I-133	1.52E+03	Exclude30	Exclude30	A_2
I-135	7.53E-03	Exclude30	Exclude30	A_2
In-115m	1.11E+02	8.99E-02	1.35E-03	Ci
Kr-85	2.92E+03	2.91E+03	2.88E+03	A_2
La-140	2.86E+05	8.26E+04	3.17E+03	Ci
Mo-99	6.24E+04	1.89E+02	Exclude90	A_2
Nb-95	3.78E+05	3.53E+05	2.39E+05	A_2
Nb-95m	3.97E+03	3.14E+03	1.64E+03	Ci
Nb-96	3.16E+00	Exclude30	Exclude30	Ci
Nb-97	3.39E+02	Exclude30	Exclude30	Ci
Nb-97m	3.21E+02	Exclude30	Exclude30	Ci
Nd-147	8.52E+04	2.00E+04	4.52E+02	A_2
Np-237	3.71E-02	3.75E-02	3.75E-02	A_2
Np-238	1.90E+03	1.02E+00	9.86E-04	Ci
Np-239	7.67E+04	8.85E+01	1.27E-01	Ci
Pa-233	1.25E-01	8.61E-02	4.79E-02	A_2
Pd-109	1.91E+00	Exclude30	Exclude30	A_2
Pd-112	5.47E+00	Exclude30	Exclude30	Ci
Pm-147	5.79E+04	5.76E+04	5.54E+04	A_2
Pm-148	1.02E+04	6.19E+02	3.88E+01	Ci
Pm-148m	3.22E+03	2.19E+03	7.99E+02	A_2
Pm-149	9.66E+03	7.15E+00	4.87E-08	A_2
Pm-151	4.49E+02	Exclude30	Exclude30	A_2
Pr-142	1.47E+01	Exclude30	Exclude30	A_2
Pr-143	2.61E+05	8.11E+04	3.78E+03	A_2
Pr-144	2.96E+05	2.80E+05	2.42E+05	Ci
Pr-144m	2.83E+03	2.67E+03	2.31E+03	Ci
Pu-236	1.44E-02	1.42E-02	1.37E-02	A_2
Pu-238	1.24E+02	1.24E+02	1.25E+02	A_2
Pu-239	1.47E+01	1.47E+01	1.47E+01	A_2
Pu-240	1.58E+01	1.58E+01	1.58E+01	A_2
Pu-241	4.85E+03	4.84E+03	4.80E+03	A_2

Table 8.1-1. Prototype TNPP Radionuclide Inventory
(4 sheets total)

Isotope	PELE1 Total, 7 days (Ci)	PELE1 Total, 30 days (Ci)	PELE1 Total, 90 days (Ci)	Inclusion Basis
Pu-242	3.48E-02	3.48E-02	3.48E-02	A_2
Rb-86	2.01E+02	8.55E+01	9.18E+00	A_2
Rh-102	1.39E-01	1.29E-01	1.05E-01	A_2
Rh-102m	2.35E-02	2.32E-02	2.25E-02	A_2
Rh-103m	1.74E+05	1.16E+05	4.01E+04	Ci
Rh-105	3.26E+03	6.52E-02	Exclude90	A_2
Rh-106	3.11E+04	2.98E+04	2.67E+04	Ci
Ru-103	1.76E+05	1.17E+05	4.06E+04	A_2
Ru-106	3.11E+04	2.98E+04	2.67E+04	A_2
Sb-122	1.09E+01	3.14E-02	Exclude90	A_2
Sb-124	4.23E+01	3.25E+01	1.63E+01	A_2
Sb-125	1.19E+03	1.17E+03	1.13E+03	A_2
Sb-126	4.06E+01	1.12E+01	3.89E-01	A_2
Sb-127	3.05E+03	4.85E+01	9.87E-04	Ci
Se-79	1.86E-02	1.86E-02	1.86E-02	A_2
Sm-151	9.65E+01	9.65E+01	9.64E+01	A_2
Sm-153	3.52E+03	9.39E-01	Exclude90	A_2
Sn-117m	1.45E+00	4.49E-01	2.11E-02	A_2
Sn-119m	2.73E+01	2.59E+01	2.24E+01	A_2
Sn-121	1.38E+01	2.22E+00	2.21E+00	Ci
Sn-121m	2.86E+00	2.86E+00	2.85E+00	A_2
Sn-123	8.94E+01	7.90E+01	5.73E+01	A_2
Sn-125	5.83E+02	1.11E+02	1.49E+00	A_2
Sn-126	3.33E-02	3.33E-02	3.33E-02	A_2
Sr-89	2.45E+05	1.79E+05	7.85E+04	A_2
Sr-90	2.35E+04	2.35E+04	2.34E+04	A_2
Sr-91	1.85E+00	Exclude30	Exclude30	A_2
Tb-160	1.61E+01	1.29E+01	7.27E+00	A_2
Tb-161	1.54E+01	1.53E+00	3.72E-03	Ci
Tc-99	3.47E+00	3.47E+00	3.47E+00	A_2
Tc-99m	6.03E+04	1.83E+02	Exclude90	Ci
Te-123m	1.74E-01	1.52E-01	1.08E-01	A_2
Te-125m	2.59E+02	2.62E+02	2.63E+02	A_2
Te-127	3.64E+03	7.54E+02	4.85E+02	Ci
Te-127m	8.20E+02	7.24E+02	4.95E+02	A_2
Te-129	3.25E+03	2.02E+03	5.86E+02	Ci
Te-129m	5.14E+03	3.20E+03	9.28E+02	A_2
Te-131	2.11E+02	2.12E-03	Exclude90	Ci
Te-131m	8.05E+02	8.10E-03	Exclude90	A_2
Te-132	5.70E+04	3.94E+02	9.07E-04	A_2
Th-234	2.86E-02	2.86E-02	2.86E-02	A_2
U-236	1.06E-01	1.06E-01	1.06E-01	A_2
U-237	4.86E+04	4.58E+03	9.78E+00	Ci
Xe-131m	1.82E+03	7.50E+02	2.98E+01	A_2
Xe-133	1.84E+05	8.90E+03	3.20E+00	A_2

Table 8.1-1. Prototype TNPP Radionuclide Inventory
(4 sheets total)

Isotope	PELE1 Total, 7 days (Ci)	PELE1 Total, 30 days (Ci)	PELE1 Total, 90 days (Ci)	Inclusion Basis
Xe-133m	1.98E+03	1.38E+00	Exclude90	Ci
Xe-135	3.27E+00	Exclude30	Exclude30	A_2
Y-89m	2.36E+01	1.72E+01	7.57E+00	Ci
Y-90	2.36E+04	2.35E+04	2.34E+04	Ci
Y-91	3.08E+05	2.35E+05	1.15E+05	A_2
Y-91m	1.19E+00	Exclude30	Exclude30	Ci
Y-93	3.97E+00	Exclude30	Exclude30	A_2
Zn-72	1.69E-01	Exclude30	Exclude30	Ci
Zr-95	3.52E+05	2.75E+05	1.43E+05	A_2
Zr-97	3.37E+02	Exclude30	Exclude30	A_2

As described in Section 4.2.4 of this report, Tables 8.1-2 through 8.1-4 report the MAR for cooling periods of 7 days, 30 days, and 90 days, respectively, after three years of operation.

Table 8.1-2. 7 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Ag-109m	Ag, Pd	1.87E+00	0.00E+00	4.83E-02
Ag-110	Ag, Pd	5.66E-01	0.00E+00	1.45E-02
Ag-110m	Ag, Pd	4.16E+01	0.00E+00	1.07E+00
Ag-111	Ag, Pd	1.25E+03	0.00E+00	3.23E+01
Ag-112	Ag, Pd	6.27E+00	0.00E+00	1.62E-01
Am-241	Pu, Actinides	4.90E+00	5.02E-04	3.22E-07
Am-242	Pu, Actinides	1.63E+00	1.67E-04	1.07E-07
Am-242m	Pu, Actinides	2.15E-01	2.20E-05	1.41E-08
Am-243	Pu, Actinides	1.27E-01	1.30E-05	8.34E-09
As-77	Sb	3.18E+01	2.75E-02	1.41E-02
Ba-136m	Sr, Ba, Eu	4.07E+02	4.09E+00	2.78E-02
Ba-137m	Sr, Ba, Eu	2.33E+04	2.33E+02	1.59E+00
Ba-140	Sr, Ba, Eu	2.48E+05	2.49E+03	1.69E+01
Br-82	I, Br, Te, Se	1.32E+01	0.00E+00	4.27E-04
Cd-113m	Sb	1.73E-02	1.50E-05	7.70E-06
Cd-115	Sb	1.01E+02	8.77E-02	4.51E-02
Cd-115m	Sb	4.62E+01	4.00E-02	2.06E-02
Ce-139	La, Ce	4.15E-01	4.55E-05	3.69E-07
Ce-141	La, Ce	2.98E+05	3.27E+01	2.65E-01
Ce-143	La, Ce	1.03E+04	1.13E+00	9.17E-03
Ce-144	La, Ce	2.96E+05	3.25E+01	2.58E-01
Cm-242	Pu, Actinides	8.94E+02	9.15E-02	5.87E-05
Cm-243	Pu, Actinides	1.39E-01	1.43E-05	9.15E-09
Cm-244	Pu, Actinides	6.41E+00	6.56E-04	4.21E-07
Cs-132	Cs, Rb	4.43E-01	2.23E-04	2.45E-04
Cs-134	Cs, Rb	1.55E+04	7.81E+00	8.56E+00
Cs-135	Cs, Rb	1.67E-01	8.40E-05	9.22E-05
Cs-136	Cs, Rb	3.71E+03	1.87E+00	2.05E+00
Cs-137	Cs, Rb	2.48E+04	1.24E+01	1.36E+01
Eu-152	Sr, Ba, Eu	5.78E-01	5.80E-03	3.95E-05
Eu-154	Sr, Ba, Eu	5.53E+02	5.55E+00	3.78E-02
Eu-155	Sr, Ba, Eu	3.72E+02	3.73E+00	2.54E-02
Eu-156	Sr, Ba, Eu	8.16E+03	8.18E+01	5.57E-01
Eu-157	Sr, Ba, Eu	3.90E-01	3.91E-03	2.66E-05
Ga-72	Sr, Ba, Eu	2.39E-01	2.40E-03	1.64E-05
Gd-153	Sr, Ba, Eu	6.08E-01	6.09E-03	4.15E-05
Gd-159	Sr, Ba, Eu	3.47E-01	3.48E-03	2.37E-05
Ge-77	Ag, Pd	1.51E-02	0.00E+00	3.91E-04
H-3	H-3, (1*)	9.17E+01	0.00E+00	5.25E+00
I-130	I, Br, Te, Se	1.09E-01	0.00E+00	3.54E-06
I-131	I, Br, Te, Se	9.83E+04	0.00E+00	3.19E+00

Table 8.1-2. 7 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
I-132	I, Br, Te, Se	5.88E+04	0.00E+00	1.90E+00
I-133	I, Br, Te, Se	1.52E+03	0.00E+00	4.75E-02
I-135	I, Br, Te, Se	7.53E-03	0.00E+00	2.44E-07
In-115m	Ag, Pd	1.08E+02	0.00E+00	2.80E+00
Kr-85	Noble Gases	2.92E+03	0.00E+00	9.19E-02
La-140	La, Ce	2.86E+05	3.13E+01	2.54E-01
Mo-99	Mo, Ru, Rh, Tc	6.24E+04	6.85E+00	5.45E-02
Nb-95	Mo, Ru, Rh, Tc	3.78E+05	4.15E+01	3.31E-01
Nb-95m	Mo, Ru, Rh, Tc	3.97E+03	4.36E-01	3.47E-03
Nb-96	Mo, Ru, Rh, Tc	3.16E+00	3.47E-04	2.76E-06
Nb-97	Mo, Ru, Rh, Tc	3.39E+02	3.72E-02	2.96E-04
Nb-97m	Mo, Ru, Rh, Tc	3.21E+02	3.52E-02	2.80E-04
Nd-147	La, Ce	8.52E+04	9.35E+00	7.58E-02
Np-237	Pu, Actinides	3.71E-02	3.79E-06	2.44E-09
Np-238	Pu, Actinides	1.90E+03	1.94E-01	1.25E-04
Np-239	Pu, Actinides	7.67E+04	7.85E+00	5.04E-03
Pa-233	Pu, Actinides	1.25E-01	1.28E-05	8.23E-09
Pd-109	Ag, Pd	1.87E+00	0.00E+00	4.83E-02
Pd-112	Ag, Pd	5.33E+00	0.00E+00	1.38E-01
Pm-147	La, Ce	5.78E+04	6.35E+00	5.14E-02
Pm-148	La, Ce	1.02E+04	1.11E+00	9.03E-03
Pm-148m	La, Ce	3.22E+03	3.53E-01	2.86E-03
Pm-149	La, Ce	9.66E+03	1.06E+00	8.59E-03
Pm-151	La, Ce	4.49E+02	4.92E-02	3.99E-04
Pr-142	La, Ce	1.47E+01	1.62E-03	1.31E-05
Pr-143	La, Ce	2.61E+05	2.87E+01	2.32E-01
Pr-144	La, Ce	2.96E+05	3.25E+01	2.63E-01
Pr-144m	La, Ce	2.83E+03	3.10E-01	2.51E-03
Pu-236	Pu, Actinides	1.44E-02	1.48E-06	9.48E-10
Pu-238	Pu, Actinides	1.24E+02	1.27E-02	8.12E-06
Pu-239	Pu, Actinides	1.47E+01	1.51E-03	9.67E-07
Pu-240	Pu, Actinides	1.58E+01	1.62E-03	1.04E-06
Pu-241	Pu, Actinides	4.85E+03	4.96E-01	3.19E-04
Pu-242	Pu, Actinides	3.48E-02	3.56E-06	2.28E-09
Rb-86	Cs, Rb	2.01E+02	1.01E-01	1.11E-01
Rh-102	Mo, Ru, Rh, Tc	1.39E-01	1.52E-05	1.21E-07
Rh-102m	Mo, Ru, Rh, Tc	2.34E-02	2.57E-06	2.05E-08
Rh-103m	Mo, Ru, Rh, Tc	1.74E+05	1.91E+01	1.52E-01
Rh-105	Mo, Ru, Rh, Tc	3.26E+03	3.58E-01	2.85E-03
Rh-106	Mo, Ru, Rh, Tc	3.11E+04	3.42E+00	2.72E-02
Ru-103	Mo, Ru, Rh, Tc	1.76E+05	1.93E+01	1.54E-01
Ru-106	Mo, Ru, Rh, Tc	3.11E+04	3.42E+00	2.72E-02
Sb-122	Sb	1.09E+01	9.45E-03	4.86E-03
Sb-124	Sb	4.23E+01	3.65E-02	1.88E-02
Sb-125	Sb	1.18E+03	1.02E+00	5.27E-01

Table 8.1-2. 7 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Sb-126	Sb	4.05E+01	3.50E-02	1.80E-02
Sb-127	Sb	3.04E+03	2.63E+00	1.35E+00
Se-79	I, Br, Te, Se	1.86E-02	0.00E+00	6.02E-07
Sm-151	La, Ce	9.65E+01	1.06E-02	8.58E-05
Sm-153	La, Ce	3.52E+03	3.86E-01	3.13E-03
Sn-117m	Ag, Pd	1.41E+00	0.00E+00	3.66E-02
Sn-119m	Ag, Pd	2.66E+01	0.00E+00	6.88E-01
Sn-121	Ag, Pd	1.34E+01	0.00E+00	3.48E-01
Sn-121m	Ag, Pd	2.79E+00	0.00E+00	7.21E-02
Sn-123	Ag, Pd	8.72E+01	0.00E+00	2.25E+00
Sn-125	Ag, Pd	5.68E+02	0.00E+00	1.47E+01
Sn-126	Ag, Pd	3.25E-02	0.00E+00	8.40E-04
Sr-89	Sr, Ba, Eu	2.43E+05	2.44E+03	1.66E+01
Sr-90	Sr, Ba, Eu	2.33E+04	2.33E+02	1.59E+00
Sr-91	Sr, Ba, Eu	1.84E+00	1.84E-02	1.25E-04
Tb-160	Sr, Ba, Eu	1.59E+01	1.60E-01	1.09E-03
Tb-161	Sr, Ba, Eu	1.53E+01	1.53E-01	1.04E-03
Tc-99	Mo, Ru, Rh, Tc	3.47E+00	3.81E-04	3.03E-06
Tc-99m	Mo, Ru, Rh, Tc	6.03E+04	6.62E+00	5.27E-02
Te-123m	I, Br, Te, Se	1.74E-01	0.00E+00	5.64E-06
Te-125m	I, Br, Te, Se	2.59E+02	0.00E+00	8.39E-03
Te-127	I, Br, Te, Se	3.64E+03	0.00E+00	1.18E-01
Te-127m	I, Br, Te, Se	8.20E+02	0.00E+00	2.66E-02
Te-129	I, Br, Te, Se	3.25E+03	0.00E+00	1.05E-01
Te-129m	I, Br, Te, Se	5.14E+03	0.00E+00	1.67E-01
Te-131	I, Br, Te, Se	2.11E+02	0.00E+00	6.84E-03
Te-131m	I, Br, Te, Se	8.05E+02	0.00E+00	2.61E-02
Te-132	I, Br, Te, Se	5.70E+04	0.00E+00	1.85E+00
Th-234	Pu, Actinides	2.86E-02	2.92E-06	1.88E-09
U-236	Pu, Actinides	1.06E-01	1.08E-05	6.95E-09
U-237	Pu, Actinides	4.86E+04	4.97E+00	3.19E-03
Xe-131m	Noble Gases	1.82E+03	0.00E+00	5.84E-02
Xe-133	Noble Gases	1.84E+05	0.00E+00	5.86E+00
Xe-133m	Noble Gases	1.98E+03	0.00E+00	6.27E-02
Xe-135	Noble Gases	3.27E+00	0.00E+00	1.05E-04
Y-89m	La, Ce	2.36E+01	2.59E-03	2.10E-05
Y-90	La, Ce	2.36E+04	2.59E+00	2.10E-02
Y-91	La, Ce	3.08E+05	3.38E+01	2.74E-01
Y-91m	La, Ce	1.19E+00	1.31E-04	1.06E-06
Y-93	La, Ce	3.97E+00	4.36E-04	3.53E-06
Zn-72	Sb	1.69E-01	1.46E-04	7.52E-05
Zr-95	La, Ce	3.52E+05	3.87E+01	3.13E-01
Zr-97	La, Ce	3.37E+02	3.70E-02	3.00E-04

Table 8.1-3. 30 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Ag-110	Ag, Pd	5.31E-01	0.00E+00	1.36E-02
Ag-110m	Ag, Pd	3.90E+01	0.00E+00	1.00E+00
Ag-111	Ag, Pd	1.47E+02	0.00E+00	3.80E+00
Am-241	Pu, Actinides	5.39E+00	5.52E-04	3.54E-07
Am-242	Pu, Actinides	2.14E-01	2.19E-05	1.40E-08
Am-242m	Pu, Actinides	2.15E-01	2.20E-05	1.41E-08
Am-243	Pu, Actinides	1.27E-01	1.30E-05	8.34E-09
As-77	Sb	1.67E-03	1.44E-06	7.43E-07
Ba-136m	Sr, Ba, Eu	1.21E+02	1.22E+00	8.29E-03
Ba-137m	Sr, Ba, Eu	2.32E+04	2.33E+02	1.59E+00
Ba-140	Sr, Ba, Eu	7.10E+04	7.12E+02	4.85E+00
Cd-113m	Sb	1.72E-02	1.49E-05	7.67E-06
Cd-115	Sb	7.91E-02	6.83E-05	3.52E-05
Cd-115m	Sb	3.23E+01	2.79E-02	1.44E-02
Ce-139	La, Ce	3.69E-01	4.05E-05	3.28E-07
Ce-141	La, Ce	1.83E+05	2.00E+01	1.62E-01
Ce-143	La, Ce	9.64E-02	1.06E-05	8.57E-08
Ce-144	La, Ce	2.80E+05	3.07E+01	2.44E-01
Cm-242	Pu, Actinides	8.11E+02	8.30E-02	5.32E-05
Cm-243	Pu, Actinides	1.39E-01	1.42E-05	9.14E-09
Cm-244	Pu, Actinides	6.40E+00	6.55E-04	4.20E-07
Cs-132	Cs, Rb	3.79E-02	1.91E-05	2.09E-05
Cs-134	Cs, Rb	1.52E+04	7.64E+00	8.38E+00
Cs-135	Cs, Rb	1.67E-01	8.40E-05	9.22E-05
Cs-136	Cs, Rb	1.11E+03	5.57E-01	6.11E-01
Cs-137	Cs, Rb	2.47E+04	1.24E+01	1.36E+01
Eu-152	Sr, Ba, Eu	5.76E-01	5.78E-03	3.93E-05
Eu-154	Sr, Ba, Eu	5.50E+02	5.52E+00	3.76E-02
Eu-155	Sr, Ba, Eu	3.68E+02	3.70E+00	2.52E-02
Eu-156	Sr, Ba, Eu	2.86E+03	2.87E+01	1.95E-01
Gd-153	Sr, Ba, Eu	5.69E-01	5.70E-03	3.88E-05
H-3	H-3, (1*)	9.14E+01	0.00E+00	5.23E+00
I-131	I, Br, Te, Se	1.35E+04	0.00E+00	4.38E-01
I-132	I, Br, Te, Se	4.06E+02	0.00E+00	1.31E-02
In-115m	Ag, Pd	8.76E-02	0.00E+00	2.27E-03
Kr-85	Noble Gases	2.91E+03	0.00E+00	9.16E-02
La-140	La, Ce	8.26E+04	9.06E+00	7.35E-02
Mo-99	Mo, Ru, Rh, Tc	1.89E+02	2.07E-02	1.65E-04
Nb-95	Mo, Ru, Rh, Tc	3.53E+05	3.87E+01	3.09E-01
Nb-95m	Mo, Ru, Rh, Tc	3.14E+03	3.45E-01	2.75E-03
Nd-147	La, Ce	2.00E+04	2.19E+00	1.77E-02
Np-237	Pu, Actinides	3.75E-02	3.83E-06	2.46E-09
Np-238	Pu, Actinides	1.02E+00	1.04E-04	6.70E-08
Np-239	Pu, Actinides	8.85E+01	9.05E-03	5.81E-06
Pa-233	Pu, Actinides	8.61E-02	8.81E-06	5.65E-09

Table 8.1-3. 30 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Pm-147	La, Ce	5.76E+04	6.32E+00	5.13E-02
Pm-148	La, Ce	6.19E+02	6.79E-02	5.50E-04
Pm-148m	La, Ce	2.19E+03	2.40E-01	1.95E-03
Pm-149	La, Ce	7.15E+00	7.85E-04	6.36E-06
Pr-143	La, Ce	8.11E+04	8.89E+00	7.21E-02
Pr-144	La, Ce	2.80E+05	3.07E+01	2.49E-01
Pr-144m	La, Ce	2.67E+03	2.93E-01	2.38E-03
Pu-236	Pu, Actinides	1.42E-02	1.46E-06	9.34E-10
Pu-238	Pu, Actinides	1.24E+02	1.27E-02	8.15E-06
Pu-239	Pu, Actinides	1.47E+01	1.51E-03	9.68E-07
Pu-240	Pu, Actinides	1.58E+01	1.62E-03	1.04E-06
Pu-241	Pu, Actinides	4.84E+03	4.95E-01	3.18E-04
Pu-242	Pu, Actinides	3.48E-02	3.56E-06	2.28E-09
Rb-86	Cs, Rb	8.54E+01	4.31E-02	4.72E-02
Rh-102	Mo, Ru, Rh, Tc	1.29E-01	1.41E-05	1.12E-07
Rh-102m	Mo, Ru, Rh, Tc	2.32E-02	2.54E-06	2.03E-08
Rh-103m	Mo, Ru, Rh, Tc	1.16E+05	1.27E+01	1.01E-01
Rh-105	Mo, Ru, Rh, Tc	6.52E-02	7.16E-06	5.70E-08
Rh-106	Mo, Ru, Rh, Tc	2.98E+04	3.27E+00	2.61E-02
Ru-103	Mo, Ru, Rh, Tc	1.17E+05	1.29E+01	1.02E-01
Ru-106	Mo, Ru, Rh, Tc	2.98E+04	3.27E+00	2.61E-02
Sb-122	Sb	3.14E-02	2.71E-05	1.40E-05
Sb-124	Sb	3.24E+01	2.80E-02	1.44E-02
Sb-125	Sb	1.17E+03	1.01E+00	5.21E-01
Sb-126	Sb	1.11E+01	9.63E-03	4.96E-03
Sb-127	Sb	4.84E+01	4.18E-02	2.15E-02
Se-79	I, Br, Te, Se	1.86E-02	0.00E+00	6.02E-07
Sm-151	La, Ce	9.65E+01	1.06E-02	8.58E-05
Sm-153	La, Ce	9.39E-01	1.03E-04	8.35E-07
Sn-117m	Ag, Pd	4.38E-01	0.00E+00	1.13E-02
Sn-119m	Ag, Pd	2.52E+01	0.00E+00	6.52E-01
Sn-121	Ag, Pd	2.16E+00	0.00E+00	5.59E-02
Sn-121m	Ag, Pd	2.79E+00	0.00E+00	7.21E-02
Sn-123	Ag, Pd	7.71E+01	0.00E+00	1.99E+00
Sn-125	Ag, Pd	1.09E+02	0.00E+00	2.81E+00
Sn-126	Ag, Pd	3.25E-02	0.00E+00	8.40E-04
Sr-89	Sr, Ba, Eu	1.77E+05	1.78E+03	1.21E+01
Sr-90	Sr, Ba, Eu	2.32E+04	2.33E+02	1.59E+00
Tb-160	Sr, Ba, Eu	1.28E+01	1.28E-01	8.74E-04
Tb-161	Sr, Ba, Eu	1.52E+00	1.52E-02	1.04E-04
Tc-99	Mo, Ru, Rh, Tc	3.47E+00	3.81E-04	3.04E-06
Tc-99m	Mo, Ru, Rh, Tc	1.83E+02	2.01E-02	1.60E-04
Te-123m	I, Br, Te, Se	1.52E-01	0.00E+00	4.94E-06
Te-125m	I, Br, Te, Se	2.62E+02	0.00E+00	8.50E-03
Te-127	I, Br, Te, Se	7.54E+02	0.00E+00	2.44E-02

Table 8.1-3. 30 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Te-127m	I, Br, Te, Se	7.24E+02	0.00E+00	2.35E-02
Te-129	I, Br, Te, Se	2.02E+03	0.00E+00	6.54E-02
Te-129m	I, Br, Te, Se	3.20E+03	0.00E+00	1.04E-01
Te-131	I, Br, Te, Se	2.12E-03	0.00E+00	6.88E-08
Te-131m	I, Br, Te, Se	8.10E-03	0.00E+00	2.62E-07
Te-132	I, Br, Te, Se	3.94E+02	0.00E+00	1.27E-02
Th-234	Pu, Actinides	2.86E-02	2.92E-06	1.88E-09
U-236	Pu, Actinides	1.06E-01	1.08E-05	6.95E-09
U-237	Pu, Actinides	4.58E+03	4.68E-01	3.01E-04
Xe-131m	Noble Gases	7.50E+02	0.00E+00	2.41E-02
Xe-133	Noble Gases	8.90E+03	0.00E+00	2.83E-01
Xe-133m	Noble Gases	1.38E+00	0.00E+00	4.39E-05
Y-89m	La, Ce	1.72E+01	1.89E-03	1.53E-05
Y-90	La, Ce	2.35E+04	2.57E+00	2.09E-02
Y-91	La, Ce	2.35E+05	2.57E+01	2.09E-01
Zr-95	La, Ce	2.75E+05	3.01E+01	2.44E-01

Table 8.1-4. 90 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Ag-110	Ag, Pd	4.49E-01	0.00E+00	1.15E-02
Ag-110m	Ag, Pd	3.30E+01	0.00E+00	8.47E-01
Ag-111	Ag, Pd	5.53E-01	0.00E+00	1.43E-02
Am-241	Pu, Actinides	6.66E+00	6.81E-04	4.37E-07
Am-242	Pu, Actinides	2.14E-01	2.19E-05	1.40E-08
Am-242m	Pu, Actinides	2.15E-01	2.20E-05	1.41E-08
Am-243	Pu, Actinides	1.27E-01	1.30E-05	8.34E-09
Ba-136m	Sr, Ba, Eu	5.15E+00	5.16E-02	3.52E-04
Ba-137m	Sr, Ba, Eu	2.31E+04	2.32E+02	1.58E+00
Ba-140	Sr, Ba, Eu	2.72E+03	2.73E+01	1.86E-01
Cd-113m	Sb	1.71E-02	1.48E-05	7.61E-06
Cd-115m	Sb	1.27E+01	1.10E-02	5.65E-03
Ce-139	La, Ce	2.73E-01	3.00E-05	2.43E-07
Ce-141	La, Ce	5.08E+04	5.57E+00	4.52E-02
Ce-144	La, Ce	2.42E+05	2.65E+01	2.11E-01
Cm-242	Pu, Actinides	6.28E+02	6.43E-02	4.13E-05
Cm-243	Pu, Actinides	1.39E-01	1.42E-05	9.11E-09
Cm-244	Pu, Actinides	6.36E+00	6.50E-04	4.17E-07
Cs-134	Cs, Rb	1.44E+04	7.23E+00	7.93E+00
Cs-135	Cs, Rb	1.67E-01	8.40E-05	9.22E-05
Cs-136	Cs, Rb	4.69E+01	2.36E-02	2.59E-02
Cs-137	Cs, Rb	2.46E+04	1.23E+01	1.35E+01
Eu-152	Sr, Ba, Eu	5.71E-01	5.73E-03	3.90E-05
Eu-154	Sr, Ba, Eu	5.43E+02	5.45E+00	3.71E-02
Eu-155	Sr, Ba, Eu	3.60E+02	3.61E+00	2.46E-02
Eu-156	Sr, Ba, Eu	1.85E+02	1.85E+00	1.26E-02
Gd-153	Sr, Ba, Eu	4.78E-01	4.80E-03	3.27E-05
H-3	H-3, (1*)	9.06E+01	0.00E+00	5.18E+00
I-131	I, Br, Te, Se	7.59E+01	0.00E+00	2.46E-03
In-115m	Ag, Pd	1.32E-03	0.00E+00	3.41E-05
Kr-85	Noble Gases	2.88E+03	0.00E+00	9.06E-02
La-140	La, Ce	3.17E+03	3.48E-01	2.82E-03
Nb-95	Mo, Ru, Rh, Tc	2.39E+05	2.63E+01	2.09E-01
Nb-95m	Mo, Ru, Rh, Tc	1.64E+03	1.80E-01	1.44E-03
Nd-147	La, Ce	4.52E+02	4.96E-02	4.02E-04
Np-237	Pu, Actinides	3.75E-02	3.84E-06	2.46E-09
Np-238	Pu, Actinides	9.85E-04	1.01E-07	6.47E-11
Np-239	Pu, Actinides	1.27E-01	1.30E-05	8.34E-09
Pa-233	Pu, Actinides	4.79E-02	4.90E-06	3.15E-09
Pm-147	La, Ce	5.54E+04	6.08E+00	4.93E-02
Pm-148	La, Ce	3.88E+01	4.26E-03	3.45E-05
Pm-148m	La, Ce	7.99E+02	8.76E-02	7.10E-04
Pr-143	La, Ce	3.78E+03	4.15E-01	3.36E-03
Pr-144	La, Ce	2.42E+05	2.65E+01	2.15E-01
Pr-144m	La, Ce	2.31E+03	2.53E-01	2.05E-03

Table 8.1-4. 90 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Pu-236	Pu, Actinides	1.37E-02	1.40E-06	8.98E-10
Pu-238	Pu, Actinides	1.25E+02	1.28E-02	8.21E-06
Pu-239	Pu, Actinides	1.47E+01	1.51E-03	9.68E-07
Pu-240	Pu, Actinides	1.58E+01	1.62E-03	1.04E-06
Pu-241	Pu, Actinides	4.80E+03	4.91E-01	3.15E-04
Pu-242	Pu, Actinides	3.48E-02	3.56E-06	2.28E-09
Rb-86	Cs, Rb	9.17E+00	4.62E-03	5.07E-03
Rh-102	Mo, Ru, Rh, Tc	1.05E-01	1.15E-05	9.20E-08
Rh-102m	Mo, Ru, Rh, Tc	2.25E-02	2.47E-06	1.97E-08
Rh-103m	Mo, Ru, Rh, Tc	4.01E+04	4.41E+00	3.51E-02
Rh-106	Mo, Ru, Rh, Tc	2.67E+04	2.93E+00	2.33E-02
Ru-103	Mo, Ru, Rh, Tc	4.06E+04	4.46E+00	3.55E-02
Ru-106	Mo, Ru, Rh, Tc	2.67E+04	2.93E+00	2.33E-02
Sb-124	Sb	1.63E+01	1.40E-02	7.23E-03
Sb-125	Sb	1.12E+03	9.71E-01	5.00E-01
Sb-126	Sb	3.89E-01	3.36E-04	1.73E-04
Sb-127	Sb	9.85E-04	8.52E-07	4.38E-07
Se-79	I, Br, Te, Se	1.86E-02	0.00E+00	6.02E-07
Sm-151	La, Ce	9.64E+01	1.06E-02	8.57E-05
Sn-117m	Ag, Pd	2.06E-02	0.00E+00	5.32E-04
Sn-119m	Ag, Pd	2.19E+01	0.00E+00	5.65E-01
Sn-121	Ag, Pd	2.16E+00	0.00E+00	5.58E-02
Sn-121m	Ag, Pd	2.78E+00	0.00E+00	7.19E-02
Sn-123	Ag, Pd	5.58E+01	0.00E+00	1.44E+00
Sn-125	Ag, Pd	1.45E+00	0.00E+00	3.76E-02
Sn-126	Ag, Pd	3.25E-02	0.00E+00	8.40E-04
Sr-89	Sr, Ba, Eu	7.78E+04	7.80E+02	5.31E+00
Sr-90	Sr, Ba, Eu	2.31E+04	2.32E+02	1.58E+00
Tb-160	Sr, Ba, Eu	7.20E+00	7.22E-02	4.92E-04
Tb-161	Sr, Ba, Eu	3.68E-03	3.69E-05	2.51E-07
Tc-99	Mo, Ru, Rh, Tc	3.47E+00	3.81E-04	3.04E-06
Te-123m	I, Br, Te, Se	1.08E-01	0.00E+00	3.48E-06
Te-125m	I, Br, Te, Se	2.63E+02	0.00E+00	8.53E-03
Te-127	I, Br, Te, Se	4.85E+02	0.00E+00	1.57E-02
Te-127m	I, Br, Te, Se	4.95E+02	0.00E+00	1.60E-02
Te-129	I, Br, Te, Se	5.86E+02	0.00E+00	1.90E-02
Te-129m	I, Br, Te, Se	9.28E+02	0.00E+00	3.01E-02
Te-132	I, Br, Te, Se	9.07E-04	0.00E+00	2.94E-08
Th-234	Pu, Actinides	2.86E-02	2.92E-06	1.88E-09
U-236	Pu, Actinides	1.06E-01	1.08E-05	6.95E-09
U-237	Pu, Actinides	9.78E+00	1.00E-03	6.42E-07
Xe-131m	Noble Gases	2.98E+01	0.00E+00	9.58E-04
Xe-133	Noble Gases	3.20E+00	0.00E+00	1.01E-04
Y-89m	La, Ce	7.57E+00	8.31E-04	6.73E-06
Y-90	La, Ce	2.34E+04	2.56E+00	2.08E-02

Table 8.1-4. 90 Day MAR
(3 sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Y-91	La, Ce	1.15E+05	1.26E+01	1.02E-01
Zr-95	La, Ce	1.43E+05	1.57E+01	1.28E-01

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8.2 TNPP Transportation Hazardous Condition Evaluation

This appendix provides the results of the TNPP transportation hazardous condition evaluation which are filled out worksheets. As described in Section 4.4.2 of the report a series of expert panel sessions were held over the course of few weeks in late February and early March 2022 to identify and assess hazardous conditions associated with TNPP transport. The session participants were experts in Probabilistic Risk Analysis (PRA) (i.e., nuclear power plant PRA and transportation of nuclear material risk assessment), hazard analysis, nuclear safety analysis, and nuclear material packaging safety who made themselves familiar with TNPP vendor designs. The session experts filled out a hazardous condition worksheet to generate comprehensive listing of postulated hazardous conditions that could defeat the safety function of the TNPP transportation package.

The worksheets were filled out by first considering the hazards identified by the vendor Phase I design reports for stationary operation of the TNPP that may also pertain to transport of the TNPP. In addition, hazards exclusively associated with transportation were added based on the description of transport of the TNPP package provided in the vendor Phase I reports a detailed knowledge of transportation risk based on previous transportation risk assessments. The process considered hazards such as such the kinetic energy associated with moving vehicles, and thermal energy associated fires such as diesel fuel fire. The process also considered hazardous conditions that could occur for a stationary reactor but created different hazardous condition for a TNPP in transport. This included loss of confinement of the TNPP package, hazards associated with natural phenomenon like severe weather, and human errors in preparing for transport that could lead to failure or degradation of the TNPP package. These worksheets were produced for following hazard categories and are presented in:

- Table 8.2-1 – Fire Hazard Events.
- Table 8.2-2 – Explosion Events.
- Table 8.2-3 – Kinetic Energy Events.
- Table 8.2-4 – Potential Energy Events.
- Table 8.2-5 – Loss of Containment Events.
- Table 8.2-6 – Direct Radiological Exposure Hazard Events.
- Table 8.2-7 – Criticality Events.
- Table 8.2-8 – Man-Made External Events.
- Table 8.2-9 – Natural Phenomena Hazards.

The hazard analysis does not include consideration of hazardous conditions that occur uniquely during dismantlement of the TNPP, loading it onto the transport trailers, unloading it from the transport trailers, or reassembling the TNPP modules, except to the extent to which latent errors or failures occur that do not manifest themselves until transport of the TNPP package. While these activities might have important contribution to overall risk, they are not considered to be within the scope of the TNPP PRA which provides a risk-informed basis for just on-the-road transportation.

The first column on the left side of the worksheet for a given hazard category (e.g., Fire Hazard Events) is labelled Event Class which is a subdivision of the hazard category. For example, the Events Classes for the Fire Hazard Events category are General Fire, Diesel Fuel Fire, Oil and Grease Fire, and Graphite Fire. The second column is labelled the Initiating Event Category which describes how the hazardous condition came into being (i.e., how it was initiated). For example, the first Initiating Event Category in the Fire Hazard Events worksheet which is under General Fire is “Ignition of flammable material in a transport container (e.g., associated with the module, the overpack, or system components).” The third column is labelled the Hazardous Event Summary and is description of the hazardous condition. For this hazard analysis, the Hazardous Event Summary always concerns: (1) a release of radiological material to the environment, (2) direct radiation exposure (or an increase worker radiation exposure), or (3) a criticality which potentially involves both direct radiation and release of radiological. In terms of the PRA, the Hazardous Event Summary is essentially a description of the accident scenarios. The fourth column is an estimate of the Initiator Frequency identified in the second column. The Initiator Frequency designations are common ranges used in hazard analysis as shown in the following:

- Anticipated (Frequency ≥ 1E-02).
- Unlikely (1E-02 > Frequency ≥ 1E-04).
- Extremely Unlikely (1E-04 > Frequency ≥ 1E-06).
- Beyond Extremely Unlikely (1E-06 > Frequency).

The fifth column is a qualitative description of the physical consequences of the hazardous condition as it concerns the radiological inventory of the TNPP package. The sixth column is a qualitative characterization of risk as High, Moderate, or Low to the workers involved in the transport and to the public. Included in this column is identification material at risk (MAR) potentially released or part of the radiological inventory of the TNPP package that becomes unshielded and could cause direct exposure to a worker or the public. As described in Section 4.2 of this report, the following are contributors to the MAR that are selected as applicable for each hazardous condition (i.e., accident scenario):

1. Nongaseous fission products contained within the TRISO fuel or heavy metal contamination within the compacts that are subsequently damaged in an accident.

2. Fission gases contained within TRISO fuel or heavy metal contamination within the compacts that are subsequently damaged in an accident.

3. Fission products that have diffused from the TRISO fuel and are held up the core structures.

4. Fission products and gases that have diffused from the TRISO fuel and have plated-out in the in the reactor containment boundary (i.e., reactor pressure vessel or primary cooling system).

5. Contamination outside the reactor.

The seventh column of the worksheets identifies structures, systems, and components (SSCs) that could prevent hazardous condition (i.e., accident scenario) and the last column of the worksheets identifies SSCs that could mitigate the risk from the hazardous condition (i.e., accident scenario).

Table 8.2-1. Fire Hazard Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
General Fire	Ignition of flammable materials in a transport container (e.g., associated with the module, the overpack, or system components)	Release of radiological material from the TNPP package to the environment caused by damage due to general fire in the transport container (e.g., associated with module, the overpack, or system components)	Anticipated F ≥ 1E-02	Potential damage to containment boundary and provides a mechanism for release from the core structure and reactor containment boundary (Not hot enough to facilitate release from the TRISO fuel)	High to the worker and public MAR potentially released: 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	Passive: Non-flame propagating rated cabling Low-flame spread coatings Channelized circuit separation design.	Active: Portable or installed fire detection and/or suppression systems Active: Emergency response from caravan team including setting up a safety perimeter
Diesel Fuel Fire	Ignition of diesel fuel from transport vehicle (e.g., about 300 gallons)	Release of radiological material from TNPP package to the environment caused by damage due to ignition of spill or leaked diesel fuel from transport vehicle that propagates to package.	Anticipated F ≥ 1E-02	Potential damage to containment boundary and provides a mechanism for release from core structure and reactor containment boundary (However, it not considered hot enough to cause release from the TRISO fuel.)	High to the worker and public MAR potentially released: 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	Passive: Diesel fuel leak prevention Active: Diesel fuel leak detection	Active: Portable or installed fire detection and/or suppression systems Active: Emergency response from caravan team including setting up a safety perimeter

Table 8.2-1. Fire Hazard Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Oil and Grease Fire	Ignition of grease/oil in a transport container (e.g., associated with module, the overpack, or system components)	Release of radiological material from TNPP package caused by ignition of grease/oil in a transport container (e.g., associated with module, the overpack, or system components)	Anticipated F ≥ 1E-02	Potential damage the containment boundary and could provide a mechanism for release from the MAR. (The quantities of such flammable material are expected to be very low. The consequences of this release would be bounded by a general fire scenario.)	Moderate to the worker MAR potentially released: 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Use of no (or low quantity of) flammable lubricants in transport container	<u>Active:</u> Portable or installed fire detection and/or suppression systems <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter
Oil and Grease Fire	Ignition of grease/oil associated with transport truck or trailer.	Release of radiological material from TNPP package to the environment caused by grease/oil associated with transport truck or trailer.	Anticipated F ≥ 1E-02	Unlikely to damage the containment boundary or provide a mechanism for release from the MAR.	Low to the worker and public (MAR not identified for low-risk hazardous conditions)		<u>Active:</u> Portable or installed fire detection and/or suppression systems <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-1. Fire Hazard Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Graphite Fire	Diesel pool or general fire hot enough to cause burning or ignition of the reactor core material.	Release of radiological material from TNPP package to the environment caused by damage due to diesel pool or general fire followed by subsequent graphite fire in reactor core.	Anticipated F ≥ 1E-02	Though the initiating event is considered anticipated, the possibility of a diesel or general fire that propagates to a graphite fire (which would <u>produce the greatest possible release from the MAR</u> is considered beyond extremely unlikely (<10E-6) (i.e., involve enough other nearby flammable materials to cause burning or ignition of the reactor core material).	Low to the worker and public (MAR not identified for low-risk hazardous conditions)		<u>Active:</u> Portable or installed fire detection and/or suppression systems <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-2. Explosion Events

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Collison with explosive material	Collision with a vehicle in motion with a large amount of explosive material (e.g., a gasoline tanker, tanker carrying explosive chemicals) and subsequent explosion	See Table 9.2-3 (Kinetic Energy Events). Explicitly considered as encompassed by collision of the transport vehicle with TNPP package with a vehicle with a large amount of combustible or explosive material (e.g., a gasoline tanker, tanker carrying explosive chemicals) and subsequent fire and possible explosion	—	—	—	—	—

Table 8.2-3. Kinetic Energy Events
(6 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Vibration and shock	Vibration and shock of the TNPP package during transport (e.g., caused by over the road travel, braking, wind, engine vibration)	Release of radiological material from TNPP package to the environment caused by failure of reactor containment boundary due to vibration and/or shock during transport (e.g., caused by over the road travel, braking, wind, engine vibration) that loosens, degrades or fails component material, seals and connections.	Anticipated F ≥ 1E-02	Potential failure of the containment boundary of the package.	Moderate to the worker MAR potentially released: 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	Passive: Addressed in functional design criteria of the TNPP and package Active: Shock and vibration monitoring	Passive: Active: Continuous radiation monitoring
Rotational Energy	Impact of object or debris from failed equipment with rotational energy on TNPP package during transport	Release of radiological material from TNPP package to the environment caused by damage due to impact from objects or debris from failed equipment with rotational energy (e.g., a failed vehicle wheel, HVAC compressor bearing, or portable generator).	Unlikely 1E-02 > F ≥ 1E-04	Failure of rotational equipment is unlikely to occur and the possibility it leads damage of the TNPP package enough to cause release radiological material is considered extremely unlikely (1E-04 > F ≥ 1E-06).	Low to worker and public (MAR not identified for low-risk hazardous conditions)	Passive: No or limited rotational equipment in transport container Active:	Passive: Active:

Table 8.2-3. Kinetic Energy Events
(6 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Vehicles in motion	Collision with a relatively light vehicle in motion (e.g., car or light truck) during transport	Release of radiological material from TNPP package to the environment caused by damage due to collision of the transport vehicle with a light vehicle in motion (e.g., car, or light truck)	Unlikely 1E-02 > F ≥ 1E-04	Potential damage to containment boundary, core structure, and reactor containment boundary.	High to worker and public MAR potentially released: 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	Passive: Design of TNPP package to remain intact from collision impact. Administration Controls: Travel restrictions such speed controls due to road conditions	Passive: Active: Emergency response from caravan team including setting up a safety perimeter
	Collision with a heavy vehicle in motion (e.g., semi with load, or train) during transport	Release of radiological material from TNPP package to the environment caused by damage due to collision of the transport vehicle with a heavy vehicle in motion (e.g., truck, bus, car, or train)	Unlikely 1E-02 > F ≥ 1E-04	Potential damage to containment boundary, the fuel, the compact the core structure and the Primary Cooling system.	High to worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	Passive: Design of TNPP package to remain intact from collision impact. Administration Controls: Travel restrictions such speed controls due to road conditions	Passive: Active: Emergency response from caravan team including setting up a safety perimeter

Table 8.2-3. Kinetic Energy Events
(6 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Collision with a fixed object	Collision with a fixed object (e.g., wall, road or bridge structures, embankment) during transport	Release of radiological material from TNPP package to the environment caused by damage due to collision of the transport vehicle with TNPP package with a fixed object (e.g., wall, road or bridge structures, embankment, and overpass) Note: Vendor could use 9'-6" high cube container (1 foot higher than normal)	Unlikely 1E-02 > F ≥ 1E-04	Potential damage to containment boundary, the fuel, the compact the core structure and the reactor containment boundary.	High to worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of TNPP package to remain intact from collision impact. <u>Administration Controls:</u> Travel restrictions such speed controls due to road conditions	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-3. Kinetic Energy Events
(6 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Non-collision	Non-collision accident (e.g., rollover, jackknife) during transport	Release of radiological material from TNPP package to the environment caused by damage due to non-collision accident (e.g., rollover, jackknife) involving of transport vehicle with TNPP package.	Anticipated F ≥ 1E-02	Potential damage to containment boundary, the fuel, the compact the core structure and the reactor containment boundary.	High to the worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the Primary Cooling system 5. Contamination outside the reactor	<u>Passive:</u> Design of TNPP package to remain intact from rollover impact. <u>Administration Controls:</u> Travel restrictions such due to road conditions or weather.	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-3. Kinetic Energy Events
(6 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Vehicle accident and fire	Collision with a vehicle in motion (e.g., truck, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (e.g., rollover) and subsequent diesel fuel fire during transport	Release of radiological material from TNPP package to the environment caused by damage due to collision of transport vehicle with TNPP package with a vehicle in motion (e.g., truck, bus, car, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or a non-collision accident (e.g., rollover) and subsequent diesel fuel fire.	Unlikely 1E-02 > F ≥ 1E-04	Potential damage to containment boundary and provides a mechanism for release (fire) from the TRISO fuel, compact and core structure, and reactor containment boundary.	High to the worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of TNPP package to remain intact from collision or rollover impact. <u>Administration Controls:</u> Travel restrictions such due to road conditions or weather.	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter <u>Active:</u> Portable or installed fire detection and suppression systems in transport containers

Table 8.2-3. Kinetic Energy Events
(6 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
	Collision with a vehicle in motion with a large amount of combustible or explosive material (e.g., a gasoline tanker, transport of flammable chemicals) and subsequent fire and possible explosion	Release of radiological material from TNPP package to the environment caused by damage due to collision of the transport vehicle with TNPP package with a vehicle with a large amount of combustible or explosive material (e.g., a gasoline tanker, transport of flammable chemicals) and subsequent fire and possible explosion	Extremely Unlikely 1E-04 > F ≥ 1E-06	This kind of collision has the highest potential to damage the containment boundary and provides a mechanism for release (fire) from the TRISO fuel, compact and core structure, and reactor containment boundary.	High to the worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	Passive: Design of TNPP package to remain intact from collision or rollover impact. Administration Controls: Travel restrictions such due to road conditions or weather.	Passive: Active: Emergency response from caravan team including setting up a safety perimeter Active: Portable or installed fire detection and suppression systems in transport containers

Table 8.2-4. Potential Energy Events
(2 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Height or/and Mass	Drop of the transport vehicle off a bridge, embankment, or elevated surface (e.g., overpass) during transport	Release of radiological material from TNPP package to the environment caused by drop of the transport vehicle with TNPP package off a bridge, embankment, or elevated surface (e.g., overpass).	Unlikely 1E-02 > F ≥ 1E-04	Potential damage to containment boundary, the fuel, the compact the core structure and the reactor containment boundary.	High to worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of TNPP package to remain intact from collision impact. <u>Administration Controls:</u> Travel restrictions such speed controls due to road conditions	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-4. Potential Energy Events
(2 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Height or/and Mass and subsequent fire	Drop of the transport vehicle off a bridge, embankment, or elevated surface (e.g., overpass) and subsequent diesel fuel fire during transport	Release of radiological material from TNPP package to the environment caused by drop of the transport vehicle with TNPP package off a bridge, embankment, or elevated surface (e.g., overpass) and subsequent diesel fuel fire.	Extremely Unlikely 1E-04 > F ≥ 1E-06	Potential damage to containment boundary, the fuel, the compact the core structure and the reactor containment boundary.	High to worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of TNPP package to remain intact from collision impact. <u>Administration Controls:</u> Travel restrictions such speed controls due to road conditions	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-5. Loss of Containment Events
(4 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Loss of containment	Release of radiological material from reactor containment boundary caused by random containment failure.	Release of radiological material to the environment from reactor containment boundary caused by random containment failure (e.g., seal, connection or joint failure).	Anticipated F ≥ 1E-02	Potential failure of the reactor containment boundary and the package.	Moderate to the worker MAR potentially released: 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Should be addressed in functional design criteria of the TNPP and package <u>Active:</u> Monitoring of Primary Cooling temperature and pressure	<u>Passive:</u> <u>Active:</u> Continuous air radiation monitoring <u>Administration Controls:</u> Emergency response

Table 8.2-5. Loss of Containment Events
(4 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Loss of containment	Pressurized air escape from reactor containment boundary caused by residual heat buildup and excessively high ambient air temperatures with containment failure	Release of radiological material to the environment from pressurized reactor containment boundary from residual heat buildup and excessively high ambient air temperatures in combination with failure of reactor containment boundary caused by random failure, human error, or vibration. Note: It is assumed that the reactor containment boundary is somewhat pressurized from residual heat during transport. Note: The X Energy Final Design Report PGN-01-100-RPT-2000405 page VII-45 states that decay heat will be 7.2 kWt at 60 days and the regulatory limit of 185 °F (85 °C) is met at that point in time. Note: The BWXT Transportation Plan (See Appendix I.1 – ATL-PLAN-110124- of Final Design Report dated March 11, 2022) states (page 29/86) that passive cooling will be required during transport to “ensure that critical electronics and systems can properly function.” The “decay heat that needs to be removed is 19.44 BTUs post seven-day shutdown.”	Anticipated F ≥ 1E-02	Potential failure of the reactor containment boundary and the package.	Moderate to the worker MAR potentially released: 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Should be addressed in functional design criteria of the TNPP and package <u>Active:</u> Monitoring of Primary Cooling temperature and pressure	<u>Passive:</u> <u>Active:</u> Continuous air radiation monitoring <u>Administration Controls:</u> Emergency response

Table 8.2-5. Loss of Containment Events
(4 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
	Pressurized air escape from reactor containment boundary caused by residual heat buildup in combination with containment failure	Release of radiological material to the environment from pressurized reactor containment boundary caused residual heat buildup from loss of heat transfer due to minor impacts involving the TNPP package (e.g., damage of vents or impacts on heat transfer pathway) that could occur from movement of the package or other objects in the transport container in combination with failure of reactor containment boundary caused by random failure, human error, vibration or extreme cold. Note: It is assumed that the reactor containment boundary is somewhat pressurized from residual heat during transport.	Anticipated F ≥ 1E-02	Potential failure of the reactor containment boundary and the package.	Moderate to the worker MAR potentially released: 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Should be addressed in functional design criteria of the TNPP and package <u>Active:</u> Monitoring of Primary Cooling temperature and pressure	<u>Passive:</u> <u>Active:</u> Continuous air radiation monitoring <u>Administration Controls:</u> Emergency response

Table 8.2-5. Loss of Containment Events
(4 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
	Air or gas escape caused by pressurization in the TNPP package due to radiolysis and hydrogen generation during transport	Release of radiological material (e.g., activation products or contamination) in escaped air or gas from the TNPP package to the environment in caused by pressurization due to radiolysis of hydrogenous material (e.g., moisture, bound water, plastics, Shield Tank not fully drained) including possible hydrogen accumulation and ignition. (It is assumed that this phenomenon will not occur in system pressure boundary due to lack of hydrogenous material.)	Anticipated F ≥ 1E-02	Potential failure of the containment boundary and the package.	Moderate to the worker MAR potentially released: 5. Contamination outside the reactor 6. Contamination on and outside the TNPP package	<u>Passive:</u> Should be addressed in functional design criteria of the TNPP and package Use of filer vent such as NucFil on for volumes such as the Shield Tank with activated material <u>Active:</u> Monitoring of Primary Cooling temperature and pressure	<u>Passive:</u> <u>Active:</u> Continuous air radiation monitoring <u>Administration Controls:</u> Emergency response
	Air escape caused by pressurization in the TNPP package due to loss of ventilation or high ambient air temperature during transport	Release of radiological material (e.g., contamination) in escaped air from the TNPP package to the environment caused by pressurization in the TNPP package due to loss of ventilation or high ambient air temperature during transport	Anticipated F ≥ 1E-02	Potential failure of the containment boundary and the package.	Moderate to the worker MAR potentially released: 5. Contamination outside the reactor 6. Contamination on and outside the TNPP package	<u>Passive:</u> Should be addressed in functional design criteria of the TNPP and package <u>Active:</u> Monitoring of Primary Cooling temperature and pressure	<u>Passive:</u> <u>Active:</u> Continuous air radiation monitoring <u>Administration Controls:</u> Emergency response
	Air escape from the TNPP package caused by failure of containment due to random or vibration caused failure (e.g., of a seal) or human error during transport.	Release of radiological material (e.g., contamination) in escaped air from the TNPP package to the environment caused by failure of containment due to random or vibration caused failure (e.g., of a seal) or human error during transport.	Anticipated F ≥ 1E-02	Potential failure of the containment boundary and the package.	Moderate to the worker MAR potentially released: 5. Contamination outside the reactor	<u>Passive:</u> Should be addressed in functional design criteria of the TNPP and package <u>Active:</u> Monitoring of Primary Cooling temperature and pressure	<u>Passive:</u> <u>Active:</u> Continuous air radiation monitoring <u>Administration Controls:</u> Emergency response

Table 8.2-6. Direct Radiological Exposure Hazard Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Loss of shielding	Loss of shielding from drop of the transport vehicle off a bridge, embankment, or elevated surface (e.g., overpass) during transport.	Direct radiation exposure caused by loss of shielding (e.g., bolt-on shielding and cable mesh) due to drop of the transport vehicle with TNPP package off a bridge, embankment, or elevated surface (e.g., overpass) during transport.	Unlikely 1E-02 > F ≥ 1E-04	Potential damage to shielding provided as part of the TNPP package and potential damage to the reactor vessel elements and cooling system	High to the worker Moderate to the public (depending on establishment of stand-off distance) Possible direct radiation exposure to: TRISO fuel, fission products held up in compact and other core structures and the reactor containment boundary: activated reactor system components such as the control rods and motors, Reactor Pressure Vessel, copper wires, and tungsten shielding.	<u>Passive:</u> Design of TNPP package to remain intact from collision impact. <u>Administration Controls:</u> Travel restrictions such speed controls due to road conditions	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-6. Direct Radiological Exposure Hazard Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
	Loss of shielding from collision with a vehicle in motion (e.g., truck or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (rollover) during transport.	Direct radiation exposure caused by loss of shielding (e.g., bolt-on shielding and cable mesh) from damage due to collision of transport vehicle with TNPP package with a vehicle in motion (e.g., truck, bus, car, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (e.g., rollover) during transport.	Unlikely 1E-02 > F ≥ 1E-04	Potential damage to shielding provided as part of the TNPP package and potential damage to the reactor vessel elements and cooling system	High to the worker Moderate to the public (depending on establishment of stand-off distance) Possible direct radiation exposure to: TRISO fuel, fission products held up in compact and other core structures and the reactor containment boundary: activated reactor system components such as the control rods and motors, Reactor Pressure Vessel, copper wires, and tungsten shielding.	<u>Passive:</u> Design of TNPP package to remain intact from collision impact. <u>Administration Controls:</u> Travel restrictions such speed controls due to road conditions	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter
Increase in exposure time	Breakdown of transport truck or trailer (e.g., engine, transmission or axile failure)	Increase in worker exposure time to radiation from TNPP package due to breakdown of transport truck or trailer (e.g., engine, transmission or axile failure) that delays transport.	Anticipated F ≥ 1E-02	Workers receive additional radiological dose.	Moderate to the worker and Low to the public Greater exposure to existing routine direct radiation.	<u>Passive:</u> <u>Administration Controls:</u> Radiation worker controls	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-6. Direct Radiological Exposure Hazard Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Increase in exposure time	Breakdown or technical issues associated with the TNPP, the TNPP package, or the overpack and shielding that requires resolution	Increase in worker exposure time to radiation from TNPP package caused by breakdown or technical issues associated with the TNPP, the TNPP package, or the overpack and shielding that requires resolution due to unanticipated random failures or operator errors that delays transport. Note: An off-normal indication from package parameters monitoring could cause a delay such from X Energy's Transportation Monitoring System.	Anticipated F ≥ 1E-02	Workers receive additional radiological dose.	Moderate to the worker and Low to the public Greater exposure to existing routine direct radiation.	Passive: Administration Controls: Radiation worker controls. Confirmatory checks before transport	Passive: Active: Emergency response from caravan team including setting up a safety perimeter
Increase in exposure time	Averse weather causes delay in transport	Increase in worker exposure time to radiation from TNPP package caused by adverse weather that delays transport	Anticipated F ≥ 1E-02	Workers receive additional radiological dose.	Moderate to the worker and Low to the public Greater exposure to existing unreleased MAR (TRISO fuel, fission products held up in compact and other core structures and the reactor containment boundary)	Passive: Administration Controls: Radiation worker controls. Confirmatory checks before transport	Passive: Active: Emergency response from caravan team including setting up a safety perimeter

Table 8.2-7. Criticality Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Addition of moderator	Addition of moderator from drop or rollover of the transport vehicle into a body of water during transport.	Direct radiation exposure and possible release of radiological material to the environment caused by a criticality event due to the immersion of the transport vehicle with TNPP into a body of water (e.g., off a bridge over a body of water or over an embankment into body of water including standing water from rain or flooding) and possible changes core geometry. Note: X Energy and BWXT are planning a core that will remain sub-critical even if immersed in water. (See X Energy Final Design Report PGN-01-100, Section VII.E.2, page VII-40) (See BWXT Final Design Report APP I.2 page 74-78 of 86.)	Extremely Unlikely $1E-04 > F \geq 1E-06$	Criticality event	High to the worker and public Direct radiation exposure from TRISO fuel going critical. MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of core to remain sub-critical after submersion in water. <u>Administration Controls:</u> Travel restrictions such as restriction due to road conditions or weather as rain and heavy flooding.	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-7. Criticality Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Addition of moderator and change in fuel geometry	Addition of moderator and change in fuel geometry from drop or rollover of the transport vehicle into a body of water during transport.	Direct radiation exposure and possible release of radiological material to the environment caused by a criticality event due to the immersion of the transport vehicle with TNPP into a body of water (e.g., off a bridge over a body of water or over an embankment into body of water including standing water from rain or flooding) and possible change in fuel geometry. Note: Reconfiguration of the geometry o the core could defeat design of the core to remain sub-critical after submersion in water.	Extremely Unlikely 1E-04 > F ≥ 1E-06	Criticality event along with loss of shielding Note: The conditional probability that this event leads to Criticality is considered to be very low because the impact of the event would have to break-up the prismatic block inside the reactor vessel and the core would have to reconfigure in way that makes criticality possible. Note: Though a drop or roll-over event of the transport vehicle into a body of water is considered Extremely Unlikely, the likelihood of the entire sequence might be considered Beyond Extremely Unlikely.	High to the worker and public Direct radiation exposure from TRISO fuel going critical. MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of core to remain sub-critical after submersion in water. TNPP design to maintain fuel geometry in case of a drop or rollover accident into a body of water. <u>Administration Controls:</u> Travel restrictions such as restriction due to road conditions or weather as rain and heavy flooding.	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-7. Criticality Events
(3 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Control rod withdrawal	Fast control rod bank withdrawal at shutdown conditions due to collision with a vehicle in motion (e.g., car, truck, bus or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (rollover) during transport.	Direct radiation exposure and possible release of radiological material to the environment caused by a criticality event due to due to fast control rod bank withdrawal at shutdown conditions during transport due to collision with a vehicle in motion (e.g., car, truck, bus or train) or fixed object (e.g., wall, road or bridge structures, embankment) or non-collision accident (rollover) during transport which causes loss of or degraded shielding.	Extremely Unlikely 1E-04 > F ≥ 1E-06	Criticality event along with loss of shielding Note: It is assumed that the impact that causes control rod withdrawal could also fail the mechanism that keeps the transportation poison rod inserted (if installed)	High to the worker and public Direct radiation exposure (e.g., from neutrons) from loss of shielding and TRISO fuel going critical. MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of TNPP against fast control rod bank withdrawal at cold conditions during transport. <u>Administration Controls:</u> Travel restrictions such speed restrictions	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-8. Man-Made External Hazard Events
(2 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
High Speed Impact	Aircraft and debris impact during transport.	Release of radiological material from TNPP package to the environment caused by damage due to impact from aircraft or aircraft debris impact during transport.	Beyond Extremely Unlikely F < 1E-06 (Note: The small size of the TNPP and short exposure duration of a few days makes the likelihood of this impact Beyond Extremely Unlikely.)	Severe damage to containment boundary, the TRISO fuel, the compact and core structure.	Low to worker and public (MAR not identified for Low-risk hazardous conditions)	Passive: Design of TNPP package to remain intact from collision impact. Active:	Passive: Active: Emergency response from caravan team including setting up a safety perimeter
	Missile impact during transport.	Release of radiological material from TNPP package to the environment caused by damage due to impact from missile (e.g., from military facility) during transport.	Beyond Extremely Unlikely F < 1E-06 (Note: The fact that most of the route is not near a military facility and the short exposure duration of a few days makes the likelihood of this impact Beyond Extremely Unlikely.)	Severe damage to containment boundary, the TRISO fuel, the compact and core structure.	Low to worker and public (MAR not identified for Low-risk hazardous conditions)	Passive: Design of TNPP package to remain intact from collision impact. Active:	Passive: Active: Emergency response from caravan team including setting up a safety perimeter
	Train impact	See Table 9.2-3 (Kinetic Energy Events). Explicitly considered as encompassed by collision with a moving “heavy” moving vehicle.	—	—	—	—	—
	Truck impact	See Table 9.2-3 (Kinetic Energy Events). Explicitly considered as part collision with a moving “heavy” moving vehicle.	—	—	—	—	—

Table 8.2-8. Man-Made External Hazard Events
(2 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
	Procedural failures or operator errors in preparing the TNPP package for transport (e.g., sealing reactor containment boundary, IHX Module, and any separated Primary Cooling piping)	Release of radiological material from TNPP package caused by procedural failures or operator errors in preparing the TNPP package for transport (e.g., sealing the reactor containment boundary, IHX Module, and any separated Primary Cooling piping)	Anticipated F ≥ 1E-02	Potential failure of the containment boundary of the package.	Moderate to the worker MAR potentially released: 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor 6. Contamination on and outside the TNPP package	Passive: Active: Air radiation monitoring Shock and vibration monitoring. Primary Cooling temperature and pressure monitoring Administrative Controls Confirmatory checks for before transport	Passive: Active: Air radiation monitoring Administration Controls: Emergency response
	Procedural failures and operational error during plant disassembly leads to undetected latent failures in containment elements (e.g., sealing reactor containment boundary, IHX Module, and any separated Primary Cooling piping)	Release of radiological material from TNPP package caused by procedural failures and operational error during plant disassembly leads to undetected latent failures in containment elements (e.g., sealing reactor containment boundary, IHX Module, and any separated Primary Cooling piping)	Anticipated F ≥ 1E-02	Potential failure of the containment boundary of the package.	Moderate to the worker MAR potentially released: 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	Passive: Active: Air radiation monitoring Shock and vibration monitoring. Primary Cooling temperature and pressure monitoring Administrative Controls Confirmatory checks for before transport	Passive: Active: Air radiation monitoring Administration Controls: Emergency response
	Sabotage of I&C equipment or safety class equipment	Consideration of sabotage is out of scope for the hazard analysis	Out of scope	Out of scope	Out of scope	Out of scope	Out of scope

Table 8.2-9. Natural Phenomena Hazards
(8 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Seismic Activity	Micro earthquakes during transport. Note: Considered to be on the scale of road vibration)	Release of radiological material from TNPP package to the environment from package caused by structural stresses and leaks due to micro earthquakes during transport that loosens, degrades or fails component material, seals and connections.	Extremely Unlikely 1E-04 > F ≥ 1E-06 Note: Unlikely (1E-02 > F ≥ 1E-04) multiplied by 1E-02 of year for duration of transport	Failure of the containment boundary of the package is judged to be unlikely.	Low to the worker and public (MAR not identified for low-risk hazardous conditions) Note: The risk of this event is bounded by risk associated with the Shock and Vibration event which is Anticipated.	Passive: Design of TNPP package to remain intact from seismic event. Active:	Passive: Active: Continuous radiation monitoring Active: Emergency response from caravan team including setting up a safety perimeter
Seismic Activity	Minor earthquakes during transport.	Release of radiological material from TNPP package to the environment from package caused by structural stresses and leaks due to micro or minor earthquakes during transport that loosens, degrades or fails component material, seals and connections.	Extremely Unlikely 1E-04 > F ≥ 1E-06 Note: Unlikely (1E-02 > F ≥ 1E-04) multiplied by 1E-02 of year for duration of transport	Potential failure of the containment boundary of the package.	Low to the worker and public (MAR not identified for low-risk hazardous conditions) Note: The risk of this event is bounded by risk associated with the Shock and Vibration event which is Anticipated.	Passive: Design of TNPP package to remain intact from seismic event. Active:	Passive: Active: Continuous radiation monitoring Active: Emergency response from caravan team including setting up a safety perimeter
	Major earthquakes during transport.	Release of radiological material from TNPP package to the environment from package caused by structural stresses and leaks due to a major earthquake that results in the package coming loose in the transport container or from collision or rollover of the transport vehicle from the ground motion or failure of roadway.	Beyond Extremely Unlikely F < 1E-06 Note: Extremely Unlikely (1E-04 > F ≥ 1E-06) multiplied by 1E-02 of year for duration of transport	Potential failure of the containment and package boundary from the package coming loose in the transport container or from collision or rollover of the transport vehicle.	Low to worker and public (MAR not identified for low-risk hazardous conditions) Note: The risk of this event is bounded by risk associated with Kinetic Energy Events in Table 9.2-3 which occur at a higher likelihood.	Passive: Design of TNPP package to remain intact from seismic event. Active:	Passive: Active: Continuous radiation monitoring Active: Emergency response from caravan team including setting up a safety perimeter

Table 8.2-9. Natural Phenomena Hazards
(8 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Precipitation	Extreme winter snow load on the transport vehicle	Release of radiological material from TNPP to the environment caused by extreme winter snow load leading to structural collapse of transport container.	Unlikely 1E-02 > F ≥ 1E-04	Failure of the transport containers that fails the containment boundary of the package is judged incredible. Note: A normal shipping container will hold 350 pounds per square foot. The transport container will be more robust than a typical shipping container.	Low to the worker and public (MAR not identified for low-risk hazardous conditions)	Passive: Design against snow and ice load Active: Administrative procedures to not drive in severe winter weather and to prevent snow and ice accumulation on transfer container, and	Passive: Active: Radiation monitoring Active: Emergency response from caravan team including setting up a safety perimeter.
	Significant ice formation on the transport vehicle	Release of radiological material from TNPP to the environment caused by significant winter ice formation load leading to structural collapse of transport container.	Unlikely 1E-02 > F ≥ 1E-04	Failure of the transport containers that fails the containment boundary of the package is judged incredible. Note: A normal shipping container will hold 350 pounds per square foot. The transport container will be more robust than a typical shipping container.	Low to the worker and public (MAR not identified for low-risk hazardous conditions)	Passive: Design against snow and ice load Active: Administrative procedures to not drive in severe winter weather and to prevent snow and ice accumulation on transfer container, and	Passive: Active: Radiation monitoring Active: Emergency response from caravan team including setting up a safety perimeter.
	Extreme rain, snow, or ice conditions during transport.	See Table 9.2-3 as contributor to Kinetic Energy Events including collision with a moving vehicle, collision with a fixed object, drop to a lower elevation (e.g., off of a bridge), and non-collision accident (e.g., rollover)	—	Note: These environmental events create special conditions that can impact radioactive material dispersion and transport beside causing an accident	—	—	—

Table 8.2-9. Natural Phenomena Hazards
(8 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
	Severe hailstorm	Release of radiological material from TNPP to the environment caused by failure of package from severe hailstorm that causes significant vibration of the transport vehicle, container and TNPP package	Anticipated F ≥ 1E-02	Potential failure of the containment boundary of the package.	Moderate to worker MAR potentially released: 5. Contamination outside the reactor 6. Contamination on and outside the TNPP package Note: The risk of this event is bounded by risk associated with the Shock and Vibration event which is Anticipated and involves MAR.	<u>Passive:</u> Design against snow and ice load <u>Active:</u> Administrative procedures to not drive in severe winter weather and to prevent snow and ice accumulation on transfer container, and	<u>Passive:</u> <u>Active:</u> Radiation monitoring <u>Active:</u> Emergency response Note: A severe hailstorm will limit emergency response in setting up a safety perimeter
Tornadoes	Tornado event during transport	Release of radiological material from TNPP package to the environment caused by damage to the TNPP and package from tornado event during transport leading to severe impacts (e.g., impacts with moving and fixed objects, rollovers, and drops) and delta pressure impacts.	Extremely Unlikely 1E-04 > F ≥ 1E-06 (Unlikely 1E-02 > F ≥ 1E-04 multiplied by 1E-02)	Likely failure of the containment boundary of the package and potential failure of the reactor containment boundary. Note: Tornadoes create special conditions that can impact radioactive material dispersion and transport Note: Assumptions about the distance to member of the public may be challenged in this event because of the inability or possible delays in setting up a safety perimeter	High to worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of TNPP package to remain intact from tornado event <u>Administrative:</u> Prohibition to transport during potential tornado weather based on national warning system Standard response actions	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-9. Natural Phenomena Hazards
(8 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
High Wind	High wind during transport	Release and dispersion of radiological material from TNPP package caused by damage due high wind that causes collision of transport vehicle with TNPP package with a vehicle in motion (e.g., truck, bus, car, or train) or fixed object (e.g., wall, road or bridge structures, embankment) or a non-collision accident (e.g., rollover).	Unlikely 1E-02 > F ≥ 1E-04	Likely failure of the containment boundary of the package and potential failure of the reactor containment boundary. Note: High wind creates special conditions that can impact radioactive material dispersion and transport	High to worker and public MAR potentially released: 1. Nongaseous fission products from TRISO fuel or heavy metal contamination 2. Fission gases from TRISO or heavy metal contamination 3. Fission products diffused from the TRISO fuel held up in the core structures 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the reactor containment boundary 5. Contamination outside the reactor	<u>Passive:</u> Design of TNPP package to remain intact from collision or rollover impact. <u>Administration Controls:</u> Travel restrictions such due to road conditions or weather.	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter
Lightning Strike	Lightning strike initiating fire during transport.	Release of radiological material from TNPP package damaged to the environment by a lightning strike of the transport vehicle during transport given.	Beyond Extremely Unlikely F < 1E-06	Potential failure of the containment boundary of the package and possible failure of the reactor containment boundary.	Low to worker and public	<u>Passive:</u> <u>Administrative:</u> Prohibition to transport during lightning storms or severe fire danger.	<u>Passive:</u> <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Table 8.2-9. Natural Phenomena Hazards
(8 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Range or Forest Fire	Range or forest fire during transport.	Release of radiological material from TNPP package to the environment caused by range or forest fire during transport.	Beyond Extremely Unlikely F < 1E-06 Note: This likelihood estimate is based on a fire that impacts the transport vehicle. Given typical warning times and the possibility for the transport vehicle to evade or reroute, the likelihood of this event is judged to be beyond extremely unlikely	Potential damage the containment boundary and could provide a mechanism for release from the MAR.	Low to worker and public (MAR not identified for Low-risk hazardous conditions)	Passive: Design of TNPP package to remain intact in extremely high temperatures. Administrative: Prohibition to transport during severe fire danger.	Passive: Active: Emergency response from caravan team including setting up a safety perimeter
External Flooding	External flooding from Local Intense Precipitation, river flooding or dam failure during transport.	Release of radiological material from TNPP package to the environment caused by Local Intense Precipitation, river flooding or dam failure during transport.	Beyond Extremely Unlikely F < 1E-06 Note: This likelihood estimate is based on a flood that impacts the transport vehicle. Given typical warning times and the possibility for the transport vehicle to evade or reroute, the likelihood of this event is judged to beyond extremely unlikely	Potential failure of the containment boundary of the package and possible failure of the reactor containment boundary.	Low to worker and public (MAR not identified for Low-risk hazardous conditions)	Passive: . Administrative: Prohibition to transport during flooding danger.	Passive: Active: Emergency response from caravan team including setting up a safety perimeter

Table 8.2-9. Natural Phenomena Hazards
(8 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Volcanic activity	Volcanic lava flow during transport.	Release of radiological material from TNPP package damaged by volcanic lava flow during transport	Beyond Extremely Unlikely F < 1E-06 Note: No volcanoes along the prototype TNPP route	Potential failure of the containment boundary of the package and the reactor containment boundary.	Low to worker and public (MAR not identified for Low-risk hazardous conditions)	Passive: Administrative: Prohibition to transport during volcanic activity.	Passive: Active: Emergency response from caravan team including setting up a safety perimeter
Tunnel collapse	Tunnel collapse onto transport container	Release of radiological material from TNPP package damaged by tunnel collapse onto transport container	Beyond Extremely Unlikely F < 1E-06 Note: No tunnels on expected route.	Potential failure of the containment boundary of the package and the reactor containment boundary.	Low to worker and public (MAR not identified for Low-risk hazardous conditions)	Passive: Administrative: Prohibition to transport when there is a possibility of environmental conditions that could cause a tunnel collapse, if a tunnel needed.	Passive: Active: Emergency response from caravan team including setting up a safety perimeter

Table 8.2-9. Natural Phenomena Hazards
(8 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Landslide or avalanche	Landslide or avalanche onto transport container	Release of radiological material from TNPP package damaged by landslide or avalanche onto transport container	Beyond Extremely Unlikely F < 1E-06 Note: The expected route does not cross a high mountain pass where a landslide or avalanche is most likely. An avalanche or landslide that could damage the TNPP package is highly unlikely along the route. The conditional probably such an event occurs as the TNPP transverses the hazardous area contributes to total scenario frequency of Beyond Extremely Unlikely.	Potential failure of the containment boundary of the package and the reactor containment boundary.	Low to worker and public (MAR not identified for Low-risk hazardous conditions)	Passive: Administrative: Prohibition to transport when there is a possibility of heavy flooding, landslides, or avalanches	Passive: Active: Emergency response from caravan team including setting up a safety perimeter
High Environmental Temperature	High environmental temperature during transport.	No release of radiological material from TNPP package to the environment damaged by loss of cooling system efficiency or failure of control due to impact on I&C due to operation during transport.	—	—	—	—	—

Table 8.2-9. Natural Phenomena Hazards
(8 sheets total)

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
	High environmental temperature during transport.	See Table 9.2-5: This initiator was included as (1) part loss of reactor containment boundary due to high ambient air temperatures in combination with build-up of residual heat, (2) potential loss of reactor containment boundary due to high ambient air temperatures only, and (3) part of loss of TNPP package containment due to high ambient air temperatures.	—	—	—	—	—
Extreme Cold Environmental Temperature	Extreme cold environmental temperature during transport.	Release of radiological material from TNPP package to the environment NMPP packaging seal and Primary System containment due to extreme cold environmental temperature (e.g., beyond design limits of a containment features during transport.) Note: The temperature specification for materials used in the stationary reactor is -50 °F, the specification for the TNPP package is not known.	Anticipated F ≥ 1E-02	Potential failure of the containment boundary of the package and the reactor containment boundary.	Moderate to the worker MAR potentially released: 4. Fission products and gases that have diffused from the TRISO fuel and plated-out in the Primary Cooling system 5. Contamination outside the reactor	Passive: Design of TNPP package to maintain containment in extremely cold temperatures. Administrative: Prohibition to transport during extremely cold weather.	Passive: Active: Emergency response from caravan team including setting up a safety perimeter