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PNNL-36380 Revision 1

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

August 2024

G.A. Coles
T.A. Ikenberry
S.M. Short
M.S. Taylor
H.E. Adkins, Jr.
P.P. Lowry
C.A. Condon
S.J. Maheras
J.R. Phillips

U.S. DEPARTMENT OF ENERGY

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Pacific Northwest National Laboratory
Richland, Washington 99354

Summary

The U.S. Department of Defense (DoD) Strategic Capabilities Office (SCO) has initiated a project, referred to as Project Pele, to construct and demonstrate a prototype transportable microreactor or Transportable 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 site. Additional future demonstration may be performed at a DoD site 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 used by the public (e.g., the 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.

The SCO has tasked Pacific Northwest National Laboratory (PNNL) to address the regulatory challenges associated with confirming the safe transport of TNPPs containing irradiated nuclear fuel. A previous report—*Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages* (PNNL-31867)—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 the entire suite of NRC requirements set forth in Part 71 of Title 10 of the *Code of Federal Regulations* (CFR) for a Type B package. It was therefore concluded that shipment of this initial TNPP transportation package, as well as possibly others, under existing regulations would likely require NRC approval using the 10 CFR 71.12 ("Specific exemptions") process that relies 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 approving the transportation of radioactive materials. Rather, current regulations specify that packages used to transport radioactive materials must meet deterministic performance standards. The performance standards define normal conditions of transport (NCT) and hypothetical accident conditions a package must be capable of withstanding without exceeding specified acceptance criteria. For a Type B package or a 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 spent 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 transportation package and associated shipment process and controls provide safety equivalent to that of a Type B package.

After publication of the framework report, at the request of the SCO, PNNL developed a plan for the development and application of a risk assessment approach to support a risk-informed pathway for acquiring NRC and DOT approval of an over the road shipment of a single microreactor transportation package at a transport frequency to not exceed one shipment per year. This plan—the *Plan for Development and Application of Risk Assessment Approach for Transportation Package Approval of an MNPP for Domestic Highway Shipment* (PNNL-33524)—identifies the proposed content of a risk-informed exemption request to the NRC for the transport of a TNPP transportation package.

As a follow-on effort related to PNNL-33524, this current report demonstrates how the plan for a hypothetical one-time shipment of the Project Pele microreactor with irradiated fuel might be implemented. This demonstration of how to implement the plan 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 ground surface shipment of a single unit by truck at a maximum frequency of once per year. Though this report only addresses the application of risk information for the TNPP transportation 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 or environmental impact statement, required by 10 CFR Part 51 ("Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions") for exemptions.

Demonstration of the plan implementation 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 requesting an exemption from the NRC for transportation of a TNPP. It also addresses important supporting PRA-related analyses 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 this approach are that it (1) increases the likelihood of successfully obtaining regulatory approval of transportation packages, (2) informs the design relative to the risk significance of TNPP containment and shielding functions, and (3) informs the need for compensatory transportation measures as well as the identification of appropriate measures. Note that although the 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., compact and tri-structural isotropic fuel robustness, mechanical properties, and associated 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 criterion for making a risk-informed decision has been satisfied. In addition, if the risk results are found to be unacceptable, then insights derived from the evaluation can potentially be used to identify design features, operational improvements, or compensatory measures 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 the DOE when 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 transportation package and (2) a member of the public located close to or involved in the accident, defined to be the maximally exposed member of the public. 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 align with NRC nuclear safety goals and corresponding proposed quantitative health objectives and guidelines proposed by the NRC for activities like transportation but have not yet been endorsed. The figure-of-merit in the proposed guidelines is total effective dose equivalent. As a final consideration, given that the TNPP transportation package is assumed to meet the 10 CFR 71.71 deterministic requirements for NCT, development of risk evaluation guidelines was performed in a way that avoids defining pairs of likelihood-dose threshold limits as unacceptable when the limit is comparable to the risk to workers from NCT.

To facilitate application of the risk evaluation guidelines, guide the TNPP transportation PRA, and promote a common understanding, the use of the term "accident" is explicitly defined for this report. It is defined consistent with the safety functions that must be preserved by the TNPP transportation package during transport to be (1) a release of radiological material to the environment, (2) direct radiation exposure from unreleased radiological material (e.g., due to degraded transport or integrated internal shielding), or (3) a criticality that potentially involves both direct radiation and release of radiological material.

The second step in the development of the risk-informed approach was to develop a TNPP transportation package PRA. The TNPP transportation package PRA is based on information available from vendor (i.e., BWX Technologies, Inc.) design material generated for Phase I of Project Pele, which is the design stage. Phase II is the fabrication stage but includes detailed analyses for impact, shock/vibration, criticality, and fire as well as performance testing for analysis verification. The approach, data, and information presented in this report demonstrate the PRA development process and provide usable information that serves to illustrate how a TNPP transportation PRA 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 falls within the proposed risk evaluation guidelines). Further detailed design and safety analysis information from Phase II of Project Pele will better inform the TNPP transportation PRA. However, to the extent that the information was available during Phase I it is reflected in the TNPP transportation PRA 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. Collect the most current information available about 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 of the implementation of the risk-informed approach, only the module containing the reactor system (including spent nuclear fuel) and a portion of the primary cooling system, referred to

as the Reactor Module, was evaluated. This module is estimated to contain over 99 percent of the radiological inventory of a TNPP. Information collected about this module included system design and configuration information, the estimated radionuclide inventory at various time periods following shutdown of the microreactor, and information about the process of preparing the module for shipment. For this demonstration of plan implementation, it is assumed the Reactor Module is shipped 90 days after shutdown of the microreactor that has operated for 3 years. The Reactor Module and its transportation configuration are referred to as the TNPP transportation package.

2. Identify 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. Identify and develop transportation accident scenarios.

   A standard hazard analysis process was used, which included the following steps:

   - Identify possible hazardous conditions that could occur during transportation
   - Postulate accident conditions and assign degree of likelihood and consequence categories to each accident
   - Screen the accident conditions determined to have an extremely low likelihood of occurrence or low consequences (e.g., graphite fire, aircraft impact)
   - Identify and assess a comprehensive set of accident scenarios that are representative of the unscreened accident conditions
   - Group the accident scenarios by accident phenomena
   - Identify and develop accident scenarios for each group for which detailed likelihood of occurrence and consequence analysis is performed.

   A total of 32 events were identified, including crash and non-crash scenarios and some events determined not to be accidents. These scenarios were grouped into 13 accident scenarios, referred to as bounding representative accidents (or BRAs), for detailed analysis.

4. Develop estimates of the likelihood of occurrence of each bounding representative accident.

   For postulated transportation accident scenarios involving a crash, data sources used to develop an estimate of the likelihood of their occurrence included state-level accident data for large trucks and geographic information system information for the assumed transportation route; route-specific Google Street View information was used to supplement the estimation of the likelihood of certain accident scenarios. National accident data for large trucks were also used to supplement the estimation of the likelihood of certain accident scenarios when state-level data were not available. The assumed origin and destination of the shipment route was made only for the purposes of analysis and to establish a credible process and pathway for development of the transportation PRA. These assumptions can be revised as necessary to reflect future program decisions, objectives, and refinements. For postulated transportation accident scenarios that do not involve a crash (e.g., fire-only events and breach of the reactor boundary due to human errors or failure of a containment boundary isolation device), component failure data and simplified Human Reliability Analysis were used to estimate potential failures that could result in a release of radioactive material during shipment of the TNPP transportation package.

5. Develop estimates of the consequence of each bounding representative accident.

   The consequence analysis was performed for both the worker and maximally exposed member of the public. Radiation dose pathways selected for inclusion in the analysis are those used by the IAEA in Specific Safety Guide (SSG)-26 Revision 1 for determining allowable quantity limits (A_1 and A_2 values) 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 report for each pathway, with some refinement to estimate the consequences to a maximally exposed member of the public. The IAEA guidance for dose calculations locates an individual 1 m from the release point or source term who is interpreted in this report to be the worker. The exception to this is for the inhalation dose pathway in which the IAEA assumes the receptor is located 10 m from the release point into an outdoor environment. The IAEA guidance does not distinguish between a worker and the public in its dose calculations. Therefore, for this report, a maximally exposed member of the public was assumed to be located 25 m from the release point, which is based on DOT Emergency Response Guidebook isolation and protective action distance guidance for emergency response to transportation accidents involving high-level radioactive material. However, for airborne releases, rather than using 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 worker and the public 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 based on expert judgment. Where expert judgment was used, values were selected with an objective for them to be bounding.

The results of the PRA for each of the bounding representative accidents are shown in Figure ES.1 for the worker and in Figure ES.2 for the public. These figures also depict the proposed risk evaluation guidelines for comparison. As shown, the risks of bounding representative accidents except for BRA 3 are greater than those defined in the risk evaluation guidelines. For BRA 3, which is a severe collision event with a heavy vehicle or an unyielding object, the risk evaluation guidelines for both the worker and public are exceeded. The risk of BRA 4M, which is a medium-impact accident defined as a severe collision with a light vehicle, is just below the risk defined in the risk evaluation guidelines due to external dose caused by shielding being degraded by the impact energy of the collision. Uncertainty in the allocation of events considered to be light-impact accidents (BRA 4L) versus medium-impact accidents (BRA 4M) could affect the conclusions about risk for BRA 4M. However, controls, future design refinements of the shielding or enhanced consequence calculations could potentially be used to make or show that the risk to the worker from BRA 3 is acceptable and the acceptable risk estimated for BRA 4M to be more certain. Sensitivity studies were performed to assess the risk reduction that may be achieved by implementing certain controls, additional analysis, and compensatory measures.

Three additional bounding representative accidents not shown are criticality events. One of these criticalities (BRA 9A) is an event in which the reactor is submerged in water as result of the accident, thereby creating a flooded criticality. The second of these criticality events (BRA 9B), is inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash that results in fire and TNPP damage. In these two cases, the consequences were not determined because the estimated likelihood of these events occurring was calculated to be less than 5E-07 per year, which, as shown in Figure ES.1, is acceptable risk regardless of the consequence. Therefore, BRA 9A and BRA 9B were not shown as points in the graphic but would fall within the acceptable region. The third criticality event (i.e., BRA 10) is due to a reactivity insertion event (e.g., control rod withdrawal) caused by the impact energy of the accident. This report assumes the design goal will be met, precluding a reactivity insertion event in a TNPP transportation accident, so the risk is addressed.

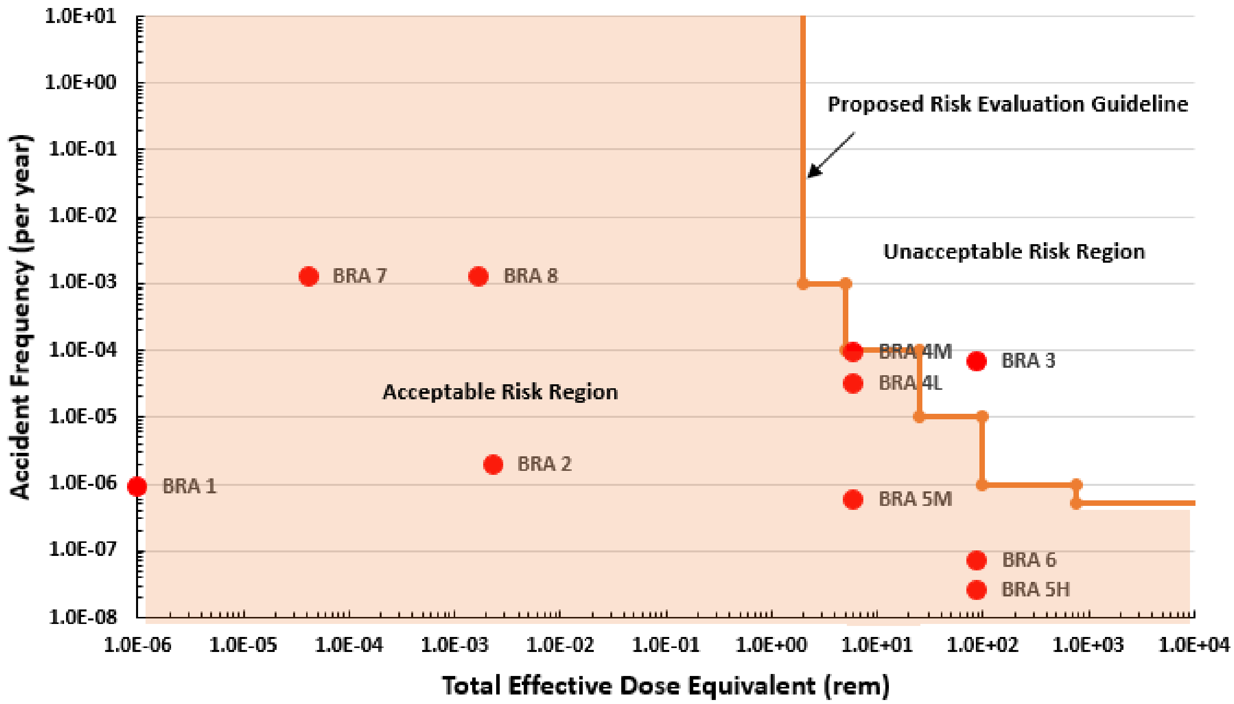

Figure ES.1. PRA Results for the Worker Compared to the Proposed Risk Evaluation Guideline

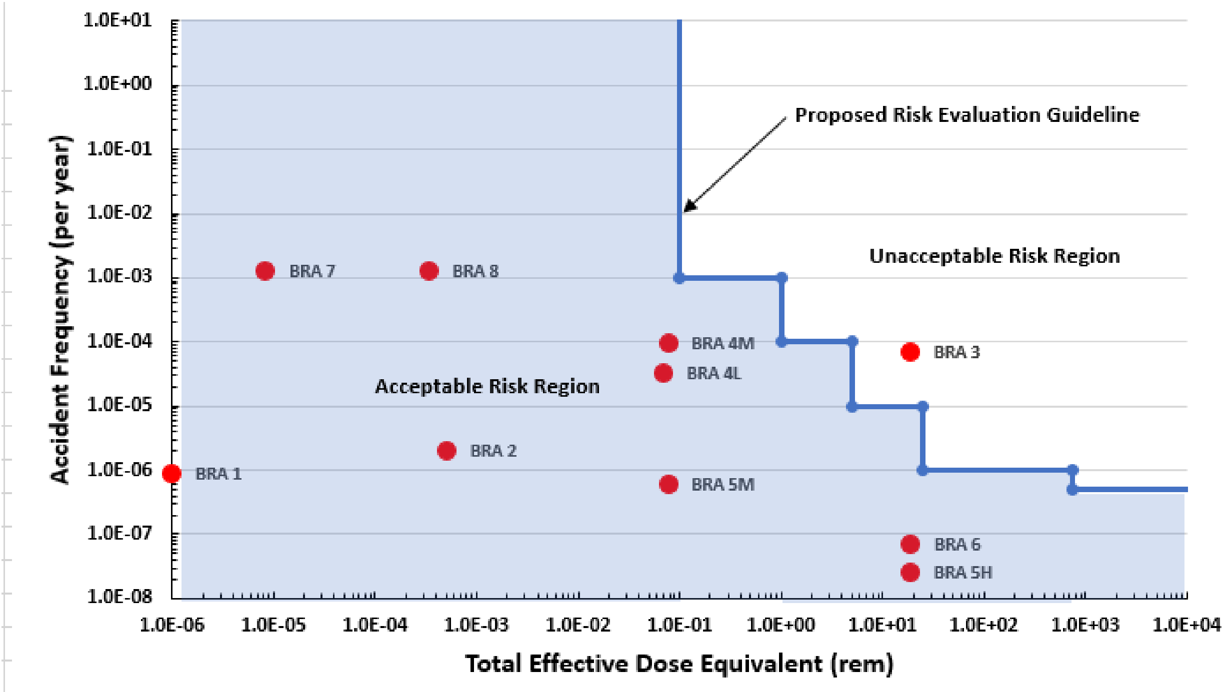

Figure ES.2. PRA Results for the Public Compared to the Proposed Risk Evaluation Guideline

This report includes the results of sensitivity studies that address the impact of modeling key assumptions and sources of uncertainty and the technical adequacy of the TNPP transportation PRA approach. In addition to better characterizing the uncertainty in the TNPP PRA risk results, the sensitivity studies, as mentioned above, support identification of controls and compensatory measures to reduce the risk associated with TNPP transportation to support the 10 CFR 71.12 exemption process. The results of one of the sensitivity studies suggest that for the one bounding representative accident (i.e., BRA 3) for which the risk to public is unacceptable, increasing the isolation distance for the public from 25 m to 100 m (and assuming that 100 m distance in the consequence analysis) yields acceptable risk results for the public, though the risk to the worker for BRA 3 would still be unacceptable. Another sensitivity study shows that decreasing the distance of the public from the accident to be the same distance as that of the worker from the accident does not change the conclusions about whether the risk of any bounding representative accident meets the risk evaluation guidelines with two exceptions (i.e., for BRA 4L and BRA 4M) where the risk to the public would slightly exceed the dose limit for the applicable accident frequencies.

An important sensitivity study result is that the risks of all bounding representative accidents evaluated in this report are below the risk defined in the proposed risk evaluation guidelines if the TNPP reactor core is allowed to decay up to a year (or even somewhat less than a year) after it has been in operation for 3 years before being shipped.

Another important sensitivity study result is that increasing the exposure duration for the worker and public from 30 to 60 minutes to a damaged TNPP has no impact on the conclusion about TNPP transportation risk. Bounding representative accidents that were acceptable in the baseline case based on frequency are still acceptable, accidents that are unacceptable in the baseline are still unacceptable, and the risk of the remaining accidents would remain acceptable.

A final important sensitivity result is that risk conclusions about TNPP transportation are insensitive to the impact of uncertainty on estimating the source term factors given a specific definition of sensitivity discussed below. Best judgment was needed to apply source term factors developed for nonreactor nuclear facilities and specifically for fuel cycle facilities. For this sensitivity study, the radiation dose consequences are considered “not sensitive” to the source term estimates if an increase of more than a factor of 1,000 is needed to exceed the risk evaluation guidelines, and they are considered “not very sensitive” if a factor of more than 100 is needed to exceed the risk evaluation guidelines. The risk of several other bounding representative accidents is either already unacceptable in the baseline case or acceptable based on accident frequency according to the baseline results. Therefore, the impact of this uncertainty is not important to the TNPP PRA risk conclusions.

It is noteworthy that the evaluation described in Section 9.1 related to identifying and defining candidate sensitivity studies was performed by explicitly examining all the PRA modeling assumptions that were made at the various stages of the TNPP PRA for those that could affect the PRA results. This assessment provides confidence that even though a full parametric data uncertainty analysis was not performed the impact of PRA modeling uncertainty was addressed.

The uncertainty analyses performed for the TNPP transportation PRA consisted of (1) the qualitative evaluation of sources of modeling uncertainty from PRA modeling assumptions and inputs, and identification and quantitative evaluation of those considered “key” sources of uncertainty; and (2) quantitative evaluation of the impact of the variability in the very large truck crash data about the conclusions regarding the risk of TNPP transportation accidents. In many cases, the results of the evaluation of key sources of uncertainty, performed using sensitivity studies, demonstrated that the conclusions about risk from the bounding representative accidents were not affected by the sensitivity study results. In other cases, the results can be used to inform compensatory measures for regulatory approval. The results of the uncertainty analysis of the impact of the variability in the very large truck crash data demonstrate that with one exception there is no change in the conclusions about risk (i.e., the risks of bounding representative accidents that were acceptable in the baseline study are still acceptable and the risks of bounding representative accidents that were unacceptable in the baseline study are still equally unacceptable). The exception to this is that accident frequency for the BRA 4M uncertainty analysis case results in an increase that puts it into the next frequency interval of the risk defined in the risk evaluation guidelines for which there is a decrease in the risk dose limit. Consequently, BRA 4M results in risk to a worker that is slightly above that described in the risk evaluation guidelines.

Assessment of defense-in-depth and safety margin philosophies conclude that they can generally be applied consistent with NRC guidance and expectations in support of TNPP transportation PRA and to its application to regulatory approval of the TNPP transportation package. However, given that this is a first attempt to apply these philosophies to this kind of application, further development of the concept of defense-in-depth related to this kind of application is expected, because traditional approaches that worked well in support of risk-informing applications for the current fleet of nuclear power plants may not apply in the same way.

With regards to assuring the technical adequacy of a transportation PRA, it is suggested that a PRA standard on TNPP transportation PRA (or PRA for transportation of any package that does not meet the deterministic requirement of 10 CFR Part 71) that is endorsed by the NRC would greatly aid the NRC approval process for applications that use PRA. If a process were patterned after the one followed for risk-informed application for nuclear power plants, then adherence to

the PRA standard as confirmed by an independent peer review would help ensure the technical adequacy of the transportation risk assessment, promote uniformity, and could significantly expedite the regulatory approval process. It would free up NRC staff to focus on the difficult technical issues, such as underlying design and modeling assumptions, rather than the form and structure of the PRA and corresponding methods.

This report provides a review history of the report by the NRC staff and the NRC Advisory Committee on Reactor Safeguards. This history involves review of the initial draft of the report by the NRC, a request for additional information by the NRC, a response by PNNL to the request for additional information, and an update of the report by PNNL associated with the information request.

In conclusion, the work summarized in this report demonstrates that a risk-formed approach using PRA to support a 10 CFR 71.12 exemption request for TNPP transport with irradiated fuel appears to be feasible and the PRA methods needed to support the request seem to be achievable. The demonstration PRA results appear to indicate that the risk from transportation of a TNPP with irradiated fuel can be shown to be acceptably low. However, this result is contingent on the development of tri-structural isotropic fuel performance testing data under impact conditions to support or validate release fraction assumptions and on structural analyses. This includes structural analysis of the custom-developed International Organization for Standardization container, which resembles a standard container express box forming the exterior of the Reactor Module and the shielding (integrated internal or that applied for transport) integrity under impact conditions to support or validate package performance assumptions.

For the few cases in which the risk acceptance guidelines are not quite met, the risk can be reduced with controls or design improvements and by using compensatory measures. The sensitivity studies and uncertainty analysis performed in support of this demonstration indicate that the sensitivity of the conclusions about risk from accidents to the uncertainty in PRA assumptions and inputs is small (in this case) and can also be reduced by implementing controls or design improvements and by using compensatory measures. Finally, the assessment of defense-in-depth and safety margin philosophies in support of the risk-informed exemption demonstrates that the application of these philosophies is feasible. Given these observations and the fact that this is a first-of-its-kind endeavor, it is recommended, however, that a PRA standard for TNPP transportation (or PRA for transportation of any package that does not meet the deterministic requirement of 10 CFR Part 71) would greatly aid the NRC approval process.

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Acronyms and Abbreviations

°C	degree(s) Celsius
A&I	Analysis and Information Online
ACRS	Advisory Committee on Reactor Safeguards
AF	Attenuation Factors
AGR	advanced gas reactor
AOO	Anticipated Operational Occurrence
ARF	airborne release fraction
BDBE	Beyond Design Basis Event
BRA	bounding representative accident
BWXT	BWX Technologies, Inc.
CC	Capability Category
CCF	common cause factor
CDF	core damage frequency
CFR	*Code of Federal Regulations*
CMV	commercial motor vehicle
CONEX	container express
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
EA	environmental assessment
EAB	Exclusion Area Boundary
EG	Evaluation Guideline
EIS	environmental impact statement
FAQ	frequently asked question
FARS	Fatality Analysis Reporting System
FHE	first harmful event
FHWA	Federal Highway Administration
FMCSA	Federal Motor Carrier Safety Administration
GIS	geographic information system
GVW	gross vehicle weight
HAC	hypothetical accident conditions
HALEU	high-assay low-enriched uranium

HEP	Human Error Probability
HLR	High-level Requirements
HMC	heavy metal contamination
HMIS	Health Monitoring Instrumentation System
HRA	Human Reliability Analysis
HRCQ	highway route-controlled quantity
HTGR	high-temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
IHX	intermediate heat exchange (Module)
INL	Idaho National Laboratory
ISA	Integrated Safety Analysis
ISCORS	Interagency Steering Committee on Radiation Standards
ISO	International Organization for Standardization
LBE	Licensing Basis Events
LCF	latent cancer fatality
LERF	large early release fraction
LPF	leak path factor
LSA	low specific activity
LWR	light water reactor
MAR	material at risk
MCMIS	Motor Carrier Management Information System
MHE	most harmful event
MNPP	Mobile Nuclear Power Plant
MOI	maximally exposed individual
NBD	National Bridge Database
NCT	normal conditions of transport
NEI	Nuclear Energy Institute
NGNP	Next Generation Nuclear Plant
NHD-HR	National Hydrography Dataset High Resolution
NHTSA	National Highway Traffic Safety Administration
NRC	U.S. Nuclear Regulatory Commission
OSM	Open Street Maps
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
RP	release parameter
RPV	Reactor Pressure Vessel
SAR	safety analysis report
SCO	Strategic Capabilities Office
SiC	silicon carbide
SPAR-H	Standardized Plant Analysis Risk-Hazardous Materials Regulation
SR	Supporting Requirement
SSC	structures, systems, and components
SSG	Specific Safety Guide (IAEA)
STATSGO	State Soil Geographic (database)
TED	total effective dose
TEDE	total effective dose equivalent
TNPP	Transportable Nuclear Power Plant
TRISO	tri-structural isotropic (particle)
UCO	uranium oxycarbide
USGS	United States Geological Survey
WebTRAGIS	Web-Based Transportation Routing Analysis Geographic Information System
WSMR	White Sands Missile Range

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1.0 Introduction

The U.S. Department of Defense (DoD) Strategic Capabilities Office (SCO) has tasked Pacific Northwest National Laboratory (PNNL) to address the regulatory challenges associated with safe transport of Transportable Nuclear Power Plants (TNPPs) containing irradiated nuclear fuel. Regulatory approval of such a transportation package based on risk information would be the first of its kind and could help pave the way for use of TNPPs. This report is being written as project work on SCO's Pele TNPP is transitioning through its design process. Phase I of Project Pele encompasses the preliminary and final design stages. Phase II is the fabrication phase but includes detailed analyses for impact, shock/vibration, criticality, and fire as well as performance testing for analysis verification. The risk modeling that supports this demonstration of a risk-informed approach is primarily based on information provided from Phase I of Project Pele. Note that the TNPP transportation probabilistic risk assessment (PRA) model discussed in this report is envisioned to be an in-process model that will be updated as the Project Pele microreactor design matures through the phases and refined information becomes available for inclusion in this risk-informed regulatory approach.

1.1 Background

The SCO has initiated Project Pele to construct and demonstrate a prototype transportable microreactor. The final environmental impact statement (EIS) for this project was released by the SCO in February 2022 (DoD 2022), and the associated Record of Decision was issued in April 2022.[1] In the Record of Decision, the SCO decided to implement the Proposed Action described in the final EIS to fabricate a prototype transportable microreactor and reactor fuel at existing offsite commercial facilities and to demonstrate the microreactor at the U.S. Department of Energy's (DOE's) Idaho National Laboratory (INL) site. Additional future demonstrations may be performed at a DoD site, and in that case, transportation of the irradiated TNPP would occur on public roads and highways. The joint effort between the SCO and DOE, established by interagency agreement, would make use of DOE expertise, material, laboratories, and authority to demonstrate a viable risk-informed regulatory approach to transportation of this 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 the SCO with accurate, current information about the NRC's regulations and licensing processes in connection with fabrication and demonstration of a TNPP. However, consistent with the noncommercial nature of the project, the prototype TNPP is proceeding under authorization by the U.S. Secretary of Energy and does not require an NRC license for operation. The DoD has future plans, which are dependent on the experience gained from this demonstration project, to use TNPPs on military installations and potentially in field operations. If implemented. these plans will necessitate making shipments of TNPPs, potentially containing irradiated nuclear fuel, using transportation infrastructure that is also used by the public (e.g., the interstate highway system). These shipments would therefore likely be regulated by the NRC and U.S. Department of Transportation (DOT).

In a predecessor report, *Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages* (PNNL-31867; Coles et al. 2021), PNNL developed a risk-informed regulatory framework (hereafter referred to as the framework report) for the

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

licensing of the TNPPs as transportation packages 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 TNPP transportation packages is to explicitly meet the deterministic requirements of 10 *Code of Federal Regulations* (CFR) Part 71 ("Packaging and Transportation of Radioactive Material") and be issued a Certificate of Compliance 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 Part 71 (e.g., the test requirements for hypothetical accident conditions [HAC] in 10 CFR 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 PRA, can be used to show safety comparable to that provided by a Type B or fissile material package for over the road transport. The framework report includes guidance on applicable regulations and discusses historical precedence in using risk information for licensing of a prototype TNPP shipment of a single unit that uses risk information to show safety comparable to that provided by a Type B or fissile material package.

The framework report (PNNL-31867) concluded 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 that require an exemption could theoretically be qualitative or semi-quantitative, significant challenges are associated with using qualitative evaluation to demonstrate that transport of a microreactor can be performed at an acceptable level of risk. Concerns include the fact that transport of a microreactor will occur with irradiated fuel in the reactor (a first-of-a-kind endeavor), design and modeling uncertainties, and the potential risk to the public if a transportation accident occurs. Accordingly, a PRA approach was determined to be needed to support the request for regulatory exemption because it provides a rigorous, systematic, quantitative evaluation of risk.

After publication of the framework report, at the request of the 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 an over the road shipment of a single microreactor transportation package. This plan, *Plan for Development and Application of Risk Assessment Approach for Transportation Package Approval of an MNPP*[2] *for Domestic Highway Shipment* (PNNL-33524; Maheras et al. 2021), identifies the proposed content of a request for a risk-informed exemption to the NRC for the transport of a TNPP transportation package, specifically, the Project Pele microreactor containing its irradiated fuel and configured for transport.

1.2 Report Purpose and Objectives

The purpose of this report is to demonstrate the implementation of the plan (PNNL-33524) for a hypothetical shipment of the Project Pele microreactor. This document is intended to be used as a guide or template for the development of a hypothetical risk-informed exemption request to

[2] Mobile Nuclear Power Plant (MNPP) and Transportable Nuclear Power Plant (TNPP) are used interchangeably initially to capture adoption of terminology used by the Pele Program and the later intention of expressing that the subject nuclear power plant that is the focus of the report is not operable under transport conditions.

the NRC by the Project Pele microreactor vendor for a ground surface shipment of a single unit by truck. This report only addresses the application of risk information in the TNPP transportation package safety analysis, but 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, incident-free transport, is not addressed in this report.

Demonstration of the 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 requesting an exemption from the NRC. It also addresses important PRA-supporting analyses 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 that it (1) increases the likelihood of successfully obtaining regulatory transportation package approval, (2) informs the design about the relative risk significance of TNPP containment boundary and shielding, and (3) informs the need for transportation compensatory measures as well as identification of appropriate measures. Although the 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 becomes available.

1.3 Report Content and Organization

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

Section 2.0 describes the division of the TNPP into modules for transport and defines the transportation package modeled in the risk assessment.

Section 3.0 discusses the regulatory approach for licensing the prototype TNPP transportation package (or TNPP Package) and why the 10 CFR Part 71 exemption process was identified to be the most feasible licensing transportation of the prototype TNPP. Section 3 also provides an overview of a risk assessment approach performed to support the selected risk-informed regulatory approach.

Section 4.0 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.

Section 5.0 describes the front-end of the proposed TNPP transportation PRA methodology—the evaluation of hazardous conditions that may exist during transport from which accident scenarios are formulated. It consists of three elements: (1) characterization of the primary hazard (i.e., the TNPP Package radiological material inventory), (2) identification of TNPP Package safety functions designed to prevent or mitigate accident scenarios associated with the radiological material inventory, and (3) processes for identifying and characterizing hazardous conditions from which TNPP transportation package accident scenarios are defined based on the transport and safety design of the TNPP Package.

Section 6.0 discusses determination of the likelihood of occurrence of the TNPP transportation accident scenarios, as derived from different data and information sources. It discusses sources of applicable transportation accident rate data for large trucks. It discusses the collection and analysis of route-specific data on potential transportation hazards such as proximity to bodies of water. This section explains how accident data and hazard data were used to estimate accident frequencies for an assumed route. This section also discusses the development of accident frequencies for accidents that are not highway–related (e.g., leaks).

Section 7.0 discusses the determination of radiological dose consequences from TNPP transportation accidents due to exposure to the release of radiological material, or direct radiation exposure to accidents due to degraded transport or integrated internal shielding.[3] Consequence analysis is based on characterizing the radiological inventory, the phenomena involved in the transportation accident, the source term for the release, the mobility of that source term (i.e., particle size and behavior), and determining the corresponding radiation dose to a human receptor.

Section 8.0 presents the TNPP transportation PRA baseline results, which are a combination of the radiation dose consequences and the accident frequency for each specified accident (i.e., in this case each bounding accident scenario). It also evaluates the results by comparing them to proposed risk evaluation guidelines.

Section 9.0 defines a set of sensitivity studies that were identified by examining key inputs and assumptions made in the TNPP transportation PRA that could potentially affect the conclusions about risk. For each sensitivity study, radiation dose consequences and accident frequencies were updated for applicable bounding representative accidents and compared to the proposed risk evaluation guidelines.

Section 10.0 provides discussion of the role of uncertainty analysis in PRA and in risk-informed decisionmaking and presents the results of the uncertainty analysis that was performed. The NRC has certain expectations about the role of parametric uncertainty analysis for PRAs supporting risk-informed applications.

Section 11.0 describes how the defense-in-depth and safety margin philosophies are incorporated into a risk-informed approach for TNPP Package transportation. This includes discussion of potential compensatory measures that are explicitly credited in the TNPP transportation PRA or implicitly credited as a defense-in-depth measures. It also includes description of how the philosophy of safety margin was considered in the TNPP transportation PRA.

Section 12.0 discusses the need to assure technical adequacy of the transportation risk assessment, including the need for independent peer review of the process and results, and proposes development of applicable national standards. The regulatory authorities need to have confidence that the information developed from a risk assessment is sound and reliable.

[3] Integrated internal shielding is shielding that is incorporated into the design of the TNPP. Transport shielding is additional external shielding added to lower radiation dose rates and facilitate shipping of the TNPP.

Section 13.0 provides the overall insights and conclusions derived from demonstrating the implementation of a risk-informed regulatory approach for TNPP transportation based on the TNPP transportation PRA results, sensitivity studies, uncertainty analysis, defense-in-depth, and safety margin assessment.

Section 14.0 provides a review history of the report by NRC staff and the NRC Advisory Committee on Reactor Safeguards (ACRS). This history involved a review of the initial draft of the report by NRC, a request for additional information by NRC, a response by PNNL to the request for additional information, and an update of the report by PNNL associated with the information request.

Section 15.0 lists the references that are cited in the report.

Four appendices provide supplemental information.

- Appendix A – TNPP Inventory and Development
- Appendix B – Evaluation of TNPP Package Transportation Hazardous Conditions
- Appendix C – Review by NRC Staff
- Appendix D – Review by the Advisory Committee on Reactor Safeguards.

2.0 Breakdown of TNPP and Definition of Transportation Package Modeled in the Risk Assessment

The section discusses the breakdown of the TNPP into modules and defines the transportation package modeled in the risk assessment.

2.1 Breakdown of TNPP into Modules

According to the final EIS (DoD 2022), the Project Pele prototype transportable nuclear microreactor would generate 1 to 5 megawatts of electric power for a minimum of three years of full-power operation. It is a tri-structural isotropic (TRISO) fueled high-temperature gas reactor that uses high-assay low-enriched uranium (HALEU) oxycarbide fuel. 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 such as the reactivity control system, portions of the Reactor Gas System (i.e., primary cooling system) loop, and portions of the 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 the modules would be contained in and integral to separate International Organization for Standardization (ISO)-compliant container express (CONEX) box-like 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 2.1.

Figure 2.1. Standard 20 ft General Purpose ISO Container

A standard 20 ft ISO container loaded on an Army M872A4 semi-trailer with an Army M915A5 tractor truck in shown in Figure 2.2.

Figure 2.2. An Army M915A5 Tractor with a M872A4 Semi-Trailer Carrying a 20 ft ISO Container[4]

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

1. Power Conversion Module
   - Disconnect the Power Conversion Module from its secondary pipes, coolant lines, and wiring
   - Install blank-off covers (i.e., 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.
2. 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 a rollback truck or trailer
   - Remove secondary pipes to the packaging area (Laydown Yard) for loading onto a separate container
   - Initiate Reactor Module wireless parameter monitoring, then disconnect and collect Control Module cables, and load the module on a rollback truck or trailer

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

3. 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)
   - Apply transport shielding
   - Assemble the Reactor Module trailer package

This defines the TNPP Package[5] (TNPP Reactor Module prepared for over the road transport).

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

- Reactor Module (consisting of the TNPP reactor, integrated internal shielding, reactor pressure vessel [RPV], connected piping and other associated components comprising the reactor containment boundary inside a custom-developed ISO container resembling a common CONEX box in appearance) – 42 tons
- Maximum weight of any other individual containers – 26.5 tons
- Maximum total weight of all containers – 70 tons
- Maximum weight for Reactor Module and addition of transport shielding (TNPP Package) – 50 tons
- Approximate weight of semi-trailer – 12 tons.

The shipping package for the prototype TNPP Reactor Module (TNPP Package) will be designed in accordance with NRC requirements in 10 CFR Part 71. It is assumed that the suite of TNPP containers will meet the NRC and DOT regulatory dose rate limits for normal conditions of transport (NCT) during shipment. However, it is also assumed that the TNPP Package will not meet all environmental and test conditions in 10 CFR 71.41(a) and subsequent 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 subjection to the postulated hypothetical accident conditions specified in 10 CFR 71.73 (HAC).

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—the design phase—of Project Pele. Further detailed design and safety analysis information that becomes available during later phases—such as disassembly (and possible reassembly) of the TNPP, and packaging and loading of the various TNPP modules for transport—will better inform the TNPP transportation PRA.

[5] For the purpose of this report, the TNPP Package is defined as the Project Pele TNPP Reactor Module consisting of the TNPP reactor containing its irradiated fuel content, integrated internal shielding, RPV, containment boundary SSCs and associated piping and circulation components housed inside a custom-developed ISO container resembling a common CONEX box in appearance fitted with additional external transport shielding, all prepared for over the road transport.

2.2 Definition of Transportation Package Modeled in the Risk Assessment

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 radionuclide inventory generated during reactor operations (over 99 percent) is expected to be 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 3.0; therefore, only the Reactor Module configured as the TNPP Package 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; instead, they 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[6]). However, if any of these packages contains 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 packaging review and approval.

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

3.0 The Risk-Informed Regulatory Approach

This section discusses the regulatory pathway that was identified to be the most feasible for the transportation package approval of the prototype TNPP Reactor Module and why a risk-informed approach is needed to support this option. It summarizes applicable background and federal regulations, as needed, to support later discussion of the proposed risk evaluation guidelines and TNPP transportation risk assessment approach. It also provides an overview of the risk assessment performed to support demonstration of the implementation of this risk-informed regulatory approach. The demonstration serves to clarify the approach and provides useful data and assessment information that can be used in a PRA application.

3.1 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 PNNL to be the most feasible approach for TNPP transportation package licensing 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 a TNPP transportation package of future TNPP Package definition to meet codified regulatory requirements). Identification of possible regulatory options, evaluation of those options for both the demonstration and production stages of Project Pele, and selection of the most feasible option for each stage is discussed by Coles et al. (2021; *Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages* [PNNL-31867]). 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 a TNPP can be performed at an acceptable level of risk. Challenges include the fact that transport of a TNPP will occur with irradiated fuel in the microreactor (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 a quantitative PRA approach because it provides a rigorous, systematic, quantitative evaluation of risk.

According to 10 CFR Part 71, four regulatory options are available for the approval of a TNPP transportation package in the United States. These options are briefly described below:

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 NRC package approval for shipments involving Type B quantities of radioactive materials are discussed by Coles et al. (2021; PNNL-31867). 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 71.41I), and (3) have a 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 being subjected to postulated 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 for a TNPP Package in the near term. A risk-informed approach is proposed to address the fact that a TNPP transportation package will likely not be able to comply with elements of the deterministic NRC requirements or address the uncertainty of 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 transport is to request exemptions from the specific requirements that are not practical to meet. Based on insights from approval of past transportation package 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 EA
- Acquisition of concurrent exemptions from applicable NRC and 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
- 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 will 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 feature that makes it impractical to transport a TNPP
2. Identify administrative controls that protect the bases and assumptions of the risk-informed assessment
3. Demonstrate that the risk to the public is acceptably low.

In addition to transportation package approval from the NRC, a special permit from the DOT will also be required prior to shipping a TNPP.

3.2 Overview of the Risk Assessment Approach

This section provides an overview of the approach used to determine the level of risk associated with transportation of the prototype TNPP Package (i.e., which is essentially the Reactor Module prepared for over the road transport as described in Section 1.0 and 2.0) to determine whether that risk is acceptable for licensing. The development of a TNPP transportation risk assessment is a novel endeavor; until now there has not been a need to develop a risk-informed licensing bases for a transportable microreactor, so a technical basis has not been thoroughly investigated. The licensing of a TNPP Package for transport 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 are now very mature.

As explained above, the TNPP Package will be designed in accordance with NRC requirements in 10 CFR Part 71 and is expected to meet NRC regulatory dose rate limits during shipment for NCT but not for HAC. Because of design challenges, it is assumed the TNPP Package will not meet (“Additional requirements for Type B packages”) or 10 CFR 71.55 (“General requirements for fissile material packages”) after being subjected to all environmental and test conditions in 10 CFR 71.41(a) associated with HAC (10 CFR 71.73). Thus, the focus of the PRA is to quantify risk associated with accidents that can defeat the safety function of the TNPP transportation package during accidents, that are less likely and more consequential than conditions that might be assumed to be normally encountered during transport. However, it should be noted that not all events that clearly must be considered accidents necessarily result in dose consequences that are significant. For example, as determined by the PRA, fire-only events during transport of TNPP lead to no or minimal radiation dose consequences. An explicit definition of the term “accident” is provided in Section 4.3 along with the proposed risk evaluation guidelines.

When it is known which 10 CFR 71.73 deterministic tests can be met and which cannot, it is possible that certain accidents could be excluded from consideration in the TNPP transportation PRA. However, in practice, it would likely be difficult to align the crash conditions with conditions created by the 10 CFR 71.73 tests. The fact that the hypothetical accident condition tests are performed sequentially, as specified in 10 CFR 71.73, to determine their cumulative effect on the package make this comparison even more difficult.

Accordingly, the reasons for performing a TNPP Package risk assessment are significant and include the following:

1. Demonstrate 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. Identify the design features and administrative controls that should be instituted to show there is an acceptable level of risk and identify compensatory measures that should be used during the transport.
3. Identify the possible trade-off between the design and risk (e.g., the trade-off between reducing the weight and size associated with containment boundary features and the risk associated with potentially breaching the containment boundary in a vehicle accident).

The 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. The paper and the proposed guidance in the risk-informed decisionmaking (RIDM) report (NRC 2008) indicate 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. In *Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages,* Coles et al. (2021; PNNL-31678) describe 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 was 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 when considering proposed compensatory measures. Accordingly, the quantitative risk assessment approach presented in this report is a PRA performed to be consistent with the units of measurement used in risk evaluation guidelines presented in Section 4.0.

The term “risk” is defined by Kaplan and Garrick (1981), 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?
3. What are the consequences?

PRA modeling of accident scenarios typically involves two types of logic analyses: (1) fault analysis and (2) event-tree analysis. 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 deploy or Emergency Diesel Generators to start) using fault trees is not required or beneficial for the 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 estimates of the likelihood of those scenarios occurring, and development of estimates of the consequences of those scenarios. The non-use of transportation fault trees and event trees is discussed in more detail in Section 5.3.1.

That said, preparatory steps are needed to support the process of developing accident scenarios and estimates of their likelihood of occurrence, and their associated consequences. The primary hazard of concern in all TNPP transportation accidents revolves around the radionuclide inventory of a TNPP Package, which needs to be characterized in detail to support accident scenario development and even more importantly to perform the consequence analysis. Another key preparatory step is the identification of the safety functions that must be maintained during transport of a TNPP. This information is key to supporting the postulation of undesired events and outcomes if events occur that could defeat or degrade one of these safety functions.

Concerning modeling assumptions in general, the TNPP transportation PRA presented in this report is primarily based on information available from the reactor vendor at the end of Phase I.

To develop the TNPP transportation PRA, assumptions had to be made, which are identified in applicable discussions of the primary elements of PRA. For example, selection of the INL site as the origination site and White Sands Missile Range (WSMR) in New Mexico as the destination site for the TNPP transport is an assumption that is identified in Section 6.0. Though assumptions are used at this phase of the project, the PRA presented in this report can be updated and revised based on selection of alternate destinations sites or on refined Phase II Project Pele prototype TNPP design information. The results of sensitivity studies presented in this report can be used to show the impact of important assumptions on the risk estimates that may need to be addressed during Phase II.

The following are the primary elements of the TNPP transportation PRA development:

- Characterization of the TNPP Package radiological material inventory (discussed in Section 5.0)
- Identification of the TNPP Package safety functions (discussed in Section 5.0)
- Identification and development of the TNPP transportation accident scenarios (discussed in Section 5.0)
- Determination of the likelihood of the occurrence of TNPP transportation accident scenarios (discussed in Section 6.0)
- Determination of the consequences of TNPP transportation accident scenarios (discussed in Section 7.0).

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 4.0 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. The primary results of the TNPP transportation PRA and the sections in which they are discussed, are as follows:

- Presentation of the TNPP transportation baseline PRA results and comparison to risk evaluation guidelines (Section 8.0)
- Definition of TNPP transportation PRA sensitivity studies and presentation of results (Section 9.0)
- Presentation of the uncertainty analysis and results (Section 10.0)

4.0 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 transportation of a TNPP Package can be found to be acceptable by comparing them to the risk evaluation guidelines, then a key criterion 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 changes, operational improvements, or compensatory measures that reduce the risk to an acceptable level. This section discusses approaches to development of potential risk evaluation guidelines and presents proposed risk evaluation guidelines for a TNPP transportation package risk that are consistent with the NRC's safety goal philosophy, guidance, and historical practice. Section 4.1 discusses NRC-suggested risk evaluation guidelines. Section 4.2 discusses the development of risk evaluation guidelines surrogates for safety goals. Section 4.3 presents proposed surrogate risk evaluation guidelines established using safety goal quantitative health guidelines.

4.1 NRC-Suggested Risk Evaluation Guidelines

In general, impacts to the public from transport of nuclear material can occur in two different ways—routine radiation exposure during normal transport 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, the 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 PRA. The approach involves the use of quantitative health guidelines (QHGs) that are based on the same safety goals as those of the risk evaluation guidance for nuclear power plants. However, the risk evaluation guidance presented in the RIDM report for the transportation of nuclear material has not yet been endorsed by the NRC because challenges remain related 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. Nonetheless, as a starting point to developing risk evaluation guidance for the transportation of a TNPP transportation package, the proposed QHG approach is summarized here.

The proposed quantitative health objectives (QHOs) are based on the 1986 NRC Safety Goal Policy statement published in the *Federal Register* (51 FR 30028-30033, August 21, 1986) for nuclear power plants. The 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." According to the RIDM report, this goal could be applied 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 quantitative definition of the QHOs from the 1986 NRC Safety Goal Policy is as follows:

- "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the 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 1 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 percent of the background fatality risk faced by workers across all industries or equivalently 1 percent 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 *Proposed Risk-Informed Regulatory Framework for Approval of Microreactor Transportation Packages* by Coles et al. (2021; PNNL-31867).

Using terminology from the PRAs developed for nuclear power plants, a full-scope Level III PRA[7] is needed to determine the risk of transporting the TNPP Package in terms consistent with the QHGs (i.e., expected health impacts). This involves the following elements:

- 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
- Determining the physical consequences for possible accident sequences in terms of the extent to which the package is breached or the internal integrated or external shielding is lost
- Calculating the likelihood of those accident sequences
- Calculating radiological consequences for possible accident sequences in terms of the quantity of radionuclides that is released to the environment
- 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” that are determined by multiplying each possible outcome by its likelihood of occurrence and then summing those values. Therefore, the total risk is determined by multiplying the likelihood of occurrence and that 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).

4.2 Development of Risk Evaluation Guideline Surrogates for Safety Goals

Three observations about the approach outlined in Section 4.1 suggest there are advantages to adjusting the proposed RIDM approach using surrogate metrics:

- 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 needed to address important modeling uncertainties.
- If the RIDM QHGs are expressed as pairs of acceptable likelihood and consequence values for which the consequence is expressed as radiation dose without combining the values, then comparisons can made of these radiation dose threshold limits to the radiation 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.

[7] A Level I PRA determines the core damage frequency and large early release fraction and other release categories, a Level II PRA determines the quantity and activity of the radioactive material released from the plant, and a Level III PRA determines the health consequences to the public.

- If the accident sequence results of a PRA are determined as pairs of likelihood and consequence values, then the PRA results provide a greater level of information that can be useful for decisionmaking 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. Note that even nuclear power plant PRAs, for which the PRA technology is mature and well-accepted by the NRC, are not typically taken to Level III to determine public health impacts. Rather, PRAs used to support risk-informed licensing decisions for stationary 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 more feasible and practical to use than QHGs. Accordingly, CDF and LERF are used as surrogate measures to the QHGs. The 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 the 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, the NRC has demonstrated that these are acceptable metrics for the latent and early QHOs using calculations presented in an 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 Sections 4.2.1 through 4.2.4 of this report. 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 proposed dose threshold limits are paired with applicable likelihood limits based on this examination and selected in a way that demonstrates they are equivalent or more conservative than the QHGs proposed in the RIDM report. As a final consideration, given that the TNPP transportation package is assumed to meet the 10 CFR 71.71 deterministic requirements for NCT, development of risk evaluation guidelines was performed in a way that avoids defining pairs of likelihood-dose threshold limits as unacceptable when the limit is comparable to the risk to workers from NCT.

Section 4.2.1 discusses the risk evaluation guideline concepts used by the DOE for safety analysis of nuclear facilities; Section 4.2.2 discusses performance criteria for ISA of nuclear fuel cycle facilities; Section 4.2.3 discusses the referenced dose limit used by the International Atomic Energy Agency (IAEA) in the so-called Q system for radiological material package requirements; Section 4.2.4 discusses the NRC risk evaluation guidelines used to identify Licensing Basis Events (LBEs) in licensing advanced non-LWRs; and Section 4.2.5 discusses the selection of pairs of likelihood-dose threshold limits and the comparison of those limits to the QHGs proposed in the RIDM report.

4.2.1 Risk Evaluation Guidelines Used by DOE for Nuclear Safety

The DOE uses the concept of risk evaluation threshold values to support the nuclear safety basis for non-reactor nuclear facilities, but they are not based on QHOs. Rather than requiring the calculation of the risk of health effects on 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 across the DOE complex.

DOE-STD-3009-2014 provides guidance on using “risk ranking” to support the 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 offsite 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 radiation dose from natural background averages 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 greater than 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[8]) to an individual located on the boundary of the exclusion zone 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.

[8] 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 standard provides guidance on defining consequence and likelihood categories that can be used for risk 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 MOI (maximally exposed offsite individual) and for a CW (co-located worker). 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 a 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 an accident involving TNPP transport package given that 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 transportation package during its shipment.

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 have a likelihood of <1E-06 per year
- Extremely Unlikely accidents have a likelihood of between 1E-04 and 1E-06 per year
- Unlikely accidents have a likelihood of between 1E-02 and 1E-04 per year
- Anticipated accidents have a 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 4.1. Given that the TNPP transportation package is designed to be robust, the most important part of this risk evaluation scheme is the lower likelihood, higher consequence criteria, but higher-likelihood, lower-consequence criteria are also included.

Table 4.1. Hypothetical Radiation Dose Evaluation Guidelines Based on DOE-STD-3009-2014

Annual Accident Frequency (per year)[(a)]	Radiation Dose Consequence to the MOI[(b)]	Radiation Dose Consequence to the CW[(b)]	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 rem 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

CW = collocated worker; MOI = maximally exposed offsite individual; rem = roentgen equivalent man; TED = total effective dose.
(a) The radiation dose consequences are presented as a TED, which is based on the integrated committed dose to all receptor organs, thereby accounting for external exposures as well as a 50-year committed effective dose.
(b) If the accident frequency is <1E-06 per year, the risk of the accident scenario is generally acceptable even if the radiation dose consequence is >25 rem. 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).

Table 4.1 reflects this by not including proposed risk evaluation criteria for Anticipated accidents. It is assumed that these accidents are mitigated by the TNPP transportation package design that meets DOE annual exposure limits of 0.1 rem to the public 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:

- The risk associated with a radiation 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.
- The risk associated with a radiation 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 radiation 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.
- The risk associated with a radiation 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 the MOI and the CW are shown graphically in Figure 4.1 and Figure 4.2, respectively.

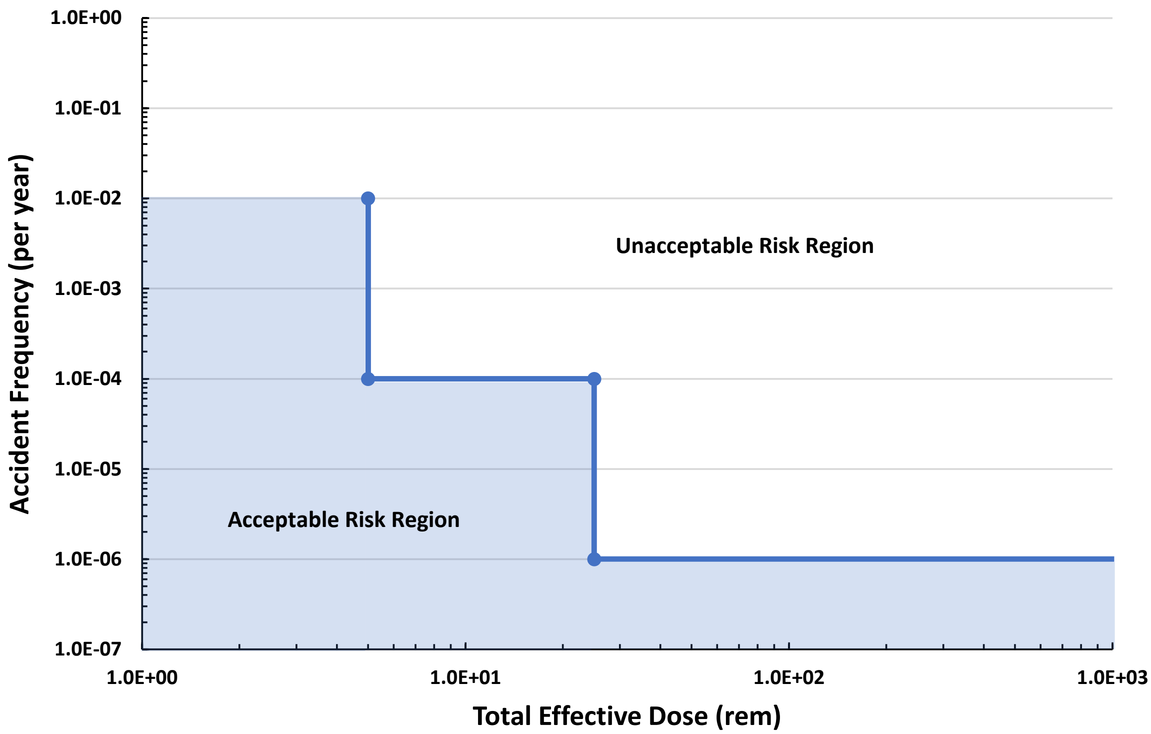


Figure 4.1. Frequency-Consequence Chart for MOI Based on DOE-STD-3009-2014

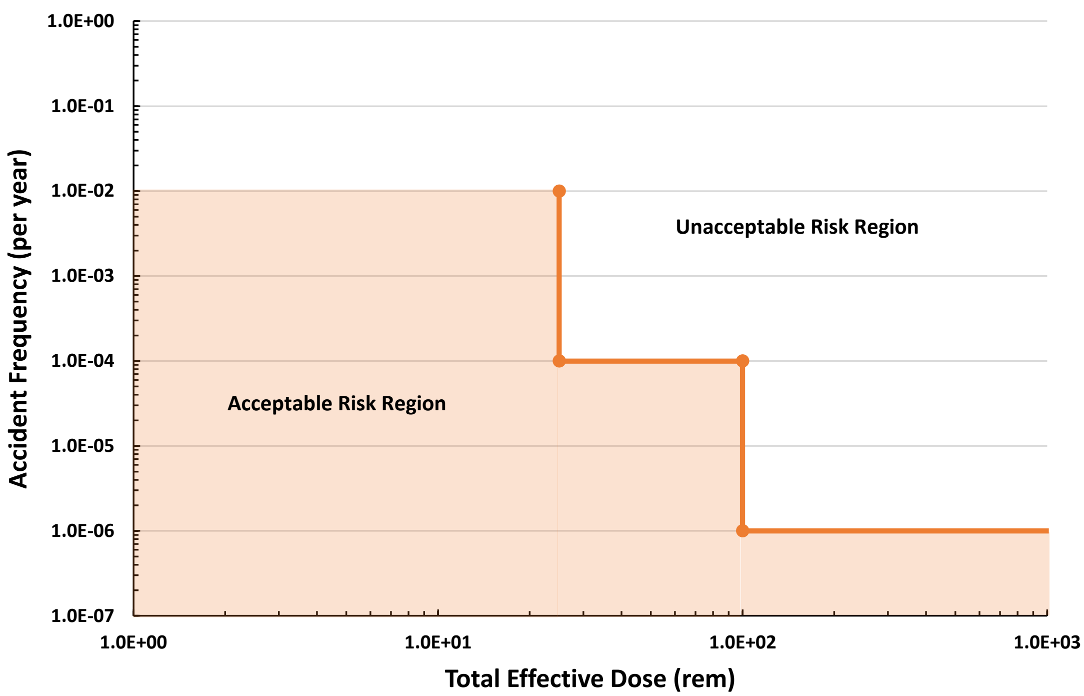


Figure 4.2. Frequency-Consequence Chart for CW Based on DOE-SD-3009-2014

4.2.2 NRC Performance Criteria for the Integrated Safety Analysis of Nuclear Fuel Cycle Facilities

NRC requirements for applications for licenses to possess and use more than a critical mass of special nuclear material, 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 is defined in 10 CFR Part 70, 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 the NRC rather than the transport of radiological material, but ISA is examined in this report because it is an NRC-accepted approach.

In essence, an ISA is a systematic examination of a facility's processes, equipment, structures, and personnel activities to make sure 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 a PRA, an ISA includes a comprehensive identification of potential accident sequences or events that would result in unacceptable consequences. However, unlike a PRA, an 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 processing 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 an ISA for NRC fuel cycle applications is provided in NUREG-1513 (*Integrated Safety Analysis Guidance Document* [NRC 2001]). The NRC has used this method 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 nonreactor 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 that 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 that 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 for showing 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.[9]
- 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 4.2.

Table 4.2. Hypothetical Radiation Dose Evaluation Guidelines Based on 10 CFR Part 70 and NUREG-1520

Annual Accident Frequency (per event, per year)	Radiation Dose Consequence to the Offsite Public[(a)]	Radiation Dose Consequence to the Worker[(a)]	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

TEDE = total effective dose equivalent.
(a) The radiation dose consequences are presented as a TEDE, which is based on the 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 special nuclear material suggests the following:

[9] Per guidance in NUREG-1520, accidents with a probability greater than 1E-04 events per year are subject to additional constraints that would be expected to reduce the dose, especially for those accidents with a high likelihood of occurring (i.e., events greater than 1E-02 evets per year).

- The risk associated with a radiation 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 it is unacceptable if the likelihood of the accident is 1E-05 per year or greater.
- The risk associated with a radiation 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 it is unacceptable if the radiation 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.
- The risk associated with a radiation dose of less than 5 rem to the public and less than 25 rem to workers is acceptable although the likelihood of the accident that produces this consequence is greater than 1E-04 per year.

These regions of acceptable and unacceptable risk for offsite public and the worker are shown graphically in Figure 4.3 and Figure 4.4, respectively.

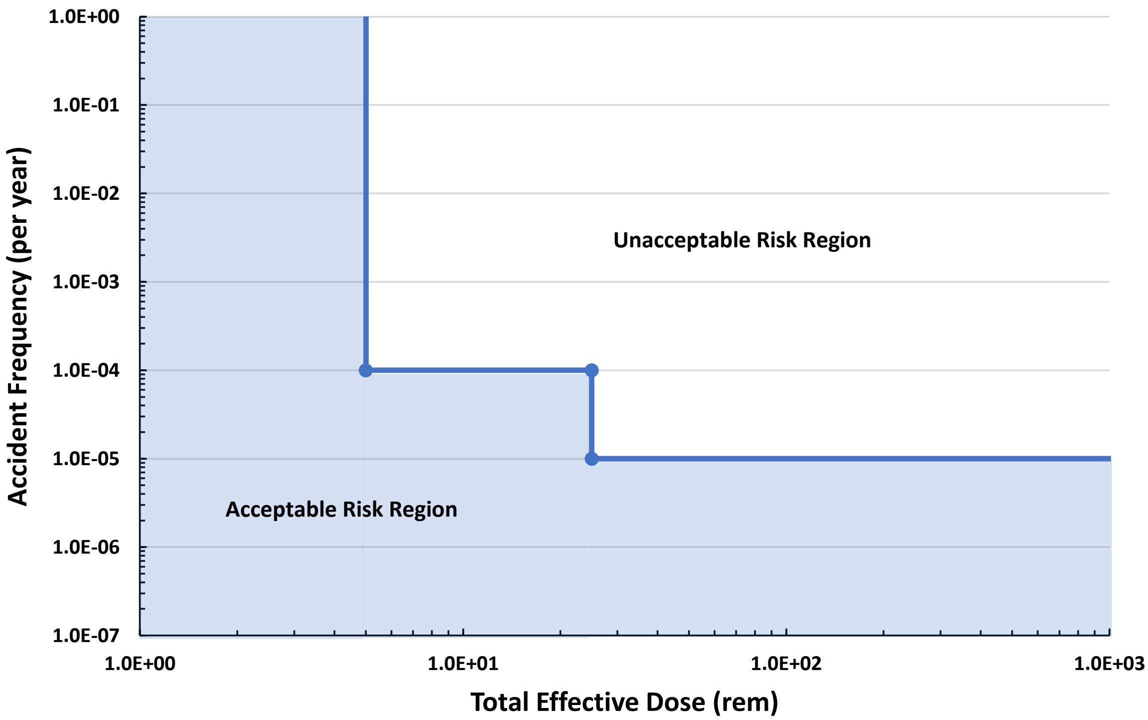


Figure 4.3. Frequency-Consequence Chart for Offsite Public Based on 10 CFR Part 70 and NUREG-1520

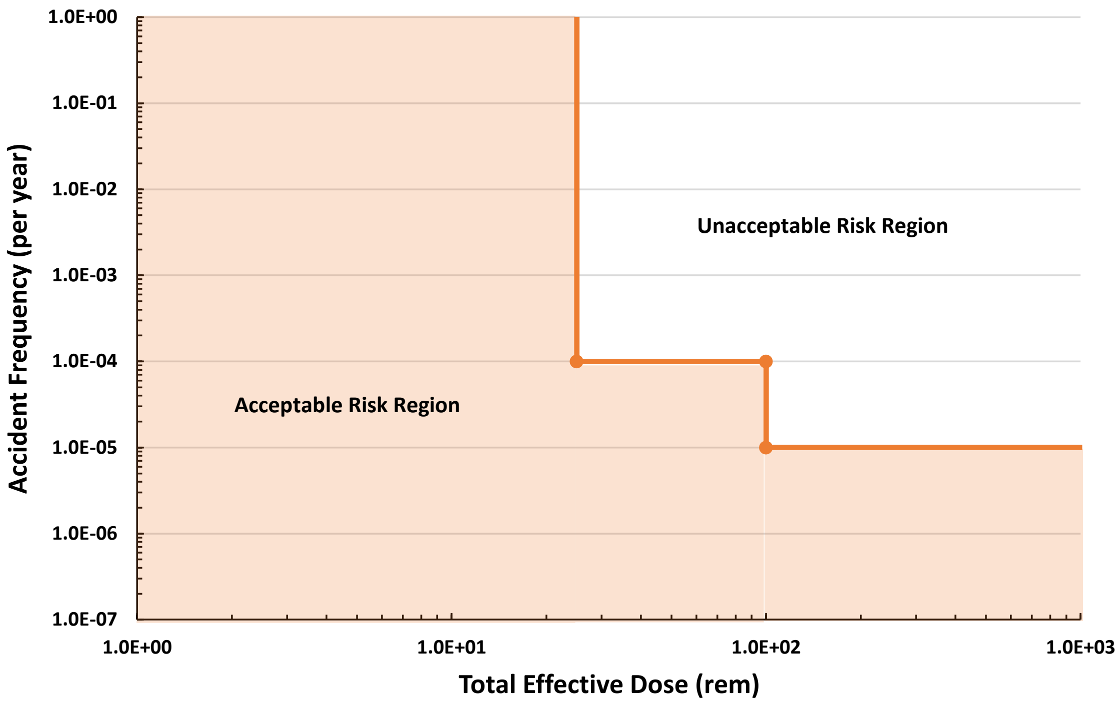


Figure 4.4. Frequency-Consequence Chart for Worker Based on 10 CFR Part 70 and NUREG-1520

4.2.3 Risk References in the IAEA Q System

The Q system was developed by United Kingdom researchers (MacDonald and Goldfinch 1983) for the 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 with units of curie (Ci) or terabecquerel (BTq) for radionuclides that are allowed in a Type A package (IAEA 2022, 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 make sure that the radiological consequences of severe damage to a Type A package are acceptable and that 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 "… intended to demonstrate that some material property (e.g., impact energy) has been shown by previous experience or by full scale prototype tests to give satisfactory performance" in accidents likely to occur in transportation. The use of likelihood and consequence pairs in the Q system as risk acceptance criteria is not obvious, but this guidance was, nonetheless, 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 radiation dose, IAEA Specific Safety Guide (SSG) No. SSG-26 (*Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material* [IAEA 2022]) 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 affect 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[10] ("Packaging and Transportation of Radioactive Material") presents the allowed activity (i.e., terabecquerel or curies) and specific activity (i.e., BTq or Ci) per gram for each radionuclide, which correlates to the quantity limit of each radionuclide allowed before more robust packaging is required. As such, if the A_1 or A_2 value for a radionuclide is exceeded, then more robust packaging is required. 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_1 or 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 the 5-rem limit 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 that a radiation dose of 5 rem is acceptable, if the likelihood of the accident that produces this consequence is highly unlikely or less; and it is unacceptable if the likelihood of the accident is more than highly unlikely.

4.2.4 NRC-Endorsed Risk-Informed Methodology to Support the Licensing of Advanced Reactor Designs

The nuclear industry has produced guidance for a risk-informed performance-based and technology-inclusive process for informing the licensing of advanced non-LWR designs. The process involves a risk-informed approach for selection of LBEs, safety classification of 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-dose 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: therefore, the concepts of CDF and LERF are not necessarily comparable.

[10] A_1 values are for special form (non-dispersible) radioactive material (i.e., sealed sources) and consider only the external photon dose and external beta dose pathways. Special form radioactive material must satisfy the requirements contained in 10 CFR 71.4. A_2 values are for non-special form (dispersible) radioactive material. Non-special form radioactive material is referred to as normal form radioactive material in U.S. regulations.

The 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]). Nonetheless, the frequency-consequence evaluation plot shown in Figure 3-1 of NEI 18-04 presents a scheme in which accidents whose likelihood and consequence fall above the blue line are considered to represent unacceptable risk (presented here as Figure 4.5). Accordingly, this risk must be addressed in the safety basis to make sure 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 TEDE (rem) at the Exclusion Area Boundary (EAB).

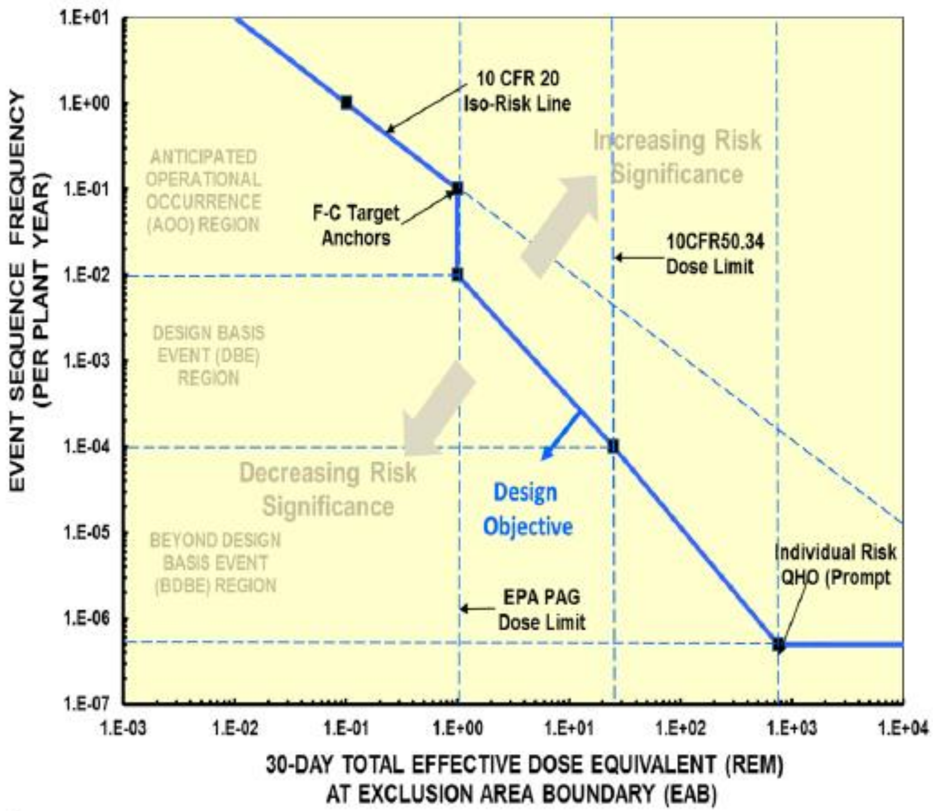


Figure 4.5. Frequency-Consequence Targets from NEI 18-04, Revision 1 (NEI 2019)

NEI 18-04 emphasizes that the frequency-consequence target line shown in Figure 4.5 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.[11] 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.

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

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 during 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.[12] 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 25 rem TEDE).

For low-frequency AOOs (i.e., events with frequencies between 1E-01 and 1E-02 per year), Figure 4.5 shows that the radiation dose consequence should not exceed 1.0 rem, which corresponds to the U.S. Environmental Protection Agency Protective Action Guide 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), Figure 4.5 shows that the radiation dose consequences are based on the iso-risk profile (i.e., having the same risk) 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 4.5 shows that the allowed radiation dose consequences 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 the 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).

[11] An example provided in RG 1.233 is the anchor point at an event sequence frequency of 5E-07 per plant year and the 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 relative to the QHOs for the integrated risks over all the LBEs.

[12] Event sequences with upper 95th percentile frequencies less than 5E-07 per year are retained in the PRA results and used to confirm there are no cliff-edge effects. They also may be considered in the risk-informed, performance-based defense-in-depth evaluation.

For BDBEs (i.e., event frequencies between 1E-04 and 5E-07 per year), Figure 4.5 shows that the allowed radiation dose consequences range from 25 rem to 750 rem. The guidance states that these criteria make sure 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 4.3 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 4.5 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 4.5 were not applied to workers because that level of differentiation was not made in the NEI 18-04 guidance. After controls are applied LBEs are not expected to release any radioactive material.

Table 4.3. Hypothetical Radiation Dose Evaluation Guidelines Based on NEI 18-04

Annual Accident Frequency (per year)[(a)]	Radiation Dose Consequence to the Offsite Public[(b)]	Radiation Dose Consequence to the Worker[(c)]	Risk Acceptability
≤5E-07[(3)]	>750 rem TEDE[(d)]	NA	Acceptable
>5E-07	>750 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	0.1 rem TEDE	NA	Acceptable

NA = not available; NEI = Nuclear Energy Institute; TEDE = total effective dose equivalent.
(a) Determination of the accident frequency should account for multiple shipments per year, if applicable.
(b) The radiation dose consequences are presented as a TEDE, which is based on the integrated committed dose to all organs, thereby accounting for direct exposure as well the 50-year committed effective dose equivalent.
(c) Worker dose limits are not addressed in NEI 18-04. Marked here as "NA."
(d) If the accident frequency is ≤5E-07 per year, the risk of the accident scenario is generally acceptable even if the radiation dose consequence is >750 rem. 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 accounted for 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:

- The risk associated with a radiation dose of greater than 750 rem to the public is acceptable if the likelihood of the accident that produces this consequence is 5E-07 per year or less, and it is unacceptable if the likelihood of the accident is more than 5E-07 per year.
- The risk associated with a radiation 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 radiation dose of greater than 25 rem is unacceptable if the likelihood of the accident is more than 1E-04 per year.
- The risk associated with a radiation 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 radiation dose of greater than 1 rem is unacceptable if the likelihood of the accident is more than 1E-02 per year.

- The risk associated with a radiation dose of 0.1 (100 mrem) to the public is acceptable though 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 4.6.

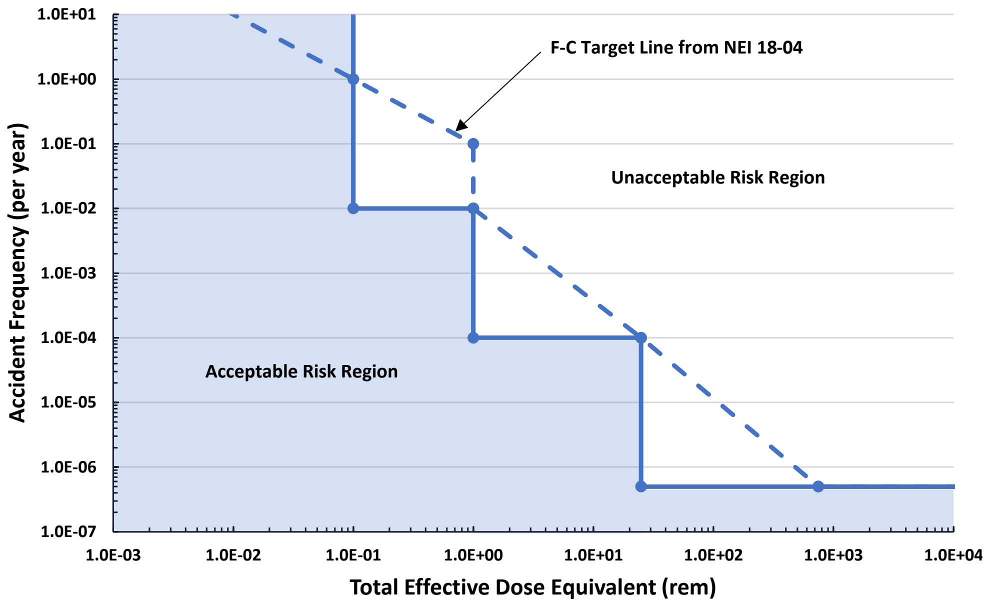


Figure 4.6. Frequency-Consequence Chart for the Offsite Public Based on NEI 18-04

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

This section discusses the selection of pairs of likelihood-consequence (i.e., radiation dose) threshold limits that are bounded by the QHGs proposed in the RIDM report (NRC 2008) and addresses how the risk associated with NCT is considered. Section 4.2.5.1 summarizes risk limits from the sources discussed above. Section 4.2.5.2 describes the conversion of accident frequency and dose to health effects to facilitate showing how selected likelihood-dose pairs that define the proposed risk evaluation guidelines compare to the RIDM QHGs. Section 4.2.5.3 discusses how the risk associated with NCT is considered in the proposed risk evaluation guidelines. Section 4.2.5.4 explains how the likelihood-dose pair limits were selected to meet the goals of the risk evaluation guidelines.

4.2.5.1 Summary of the Risk Limits that Were Explored

Table 4.4 summarizes risk limits derived from the four sources discussed previously, namely DOE-STD-3009-2014 “risk ranking” criteria process for nuclear facilities, the NRC performance criteria for nuclear fuel facilities, the Q system, and the risk-informed licensing of advanced non-LWRs. These pairs of likelihood and consequence criteria are used to help develop risk evaluation guidelines as surrogate measures to the NRC Safety Goal Policy QHOs. These likelihood-dose pairs based on the four sources are consistent and complementary. Table cells containing an entry indicate the dose limits listed in the same row of the first column were derived from the source identified in the column header of those cells. The “Not Applicable” label is meant to indicate that the dose limit was not derived from the source identified in the column header of the cell.

Table 4.4. 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 (NEI-18-04)
750 rem	Not Applicable	Not Applicable	Not Applicable	If the accident frequency is less than 5E-07 per year, then the risk of the accident scenario is acceptable even if the radiation dose is greater than 750 rem.
25 rem	A radiation dose of up to 25 rem to the public and up to 100 rem to workers is acceptable if the likelihood of the accident is less than 1E-04 per year. (If the likelihood is less than 1E-06 per year, then the dose can be greater than 25 rem to the public or 100 rem to the worker.)	A radiation dose of up to 25 rem to the public and up to 100 rem to workers is acceptable if the likelihood of the accident is less than 1E-04 per year. (If the likelihood is less than 1E-05 per year, then the dose can be greater than 25 rem to the public or 100 rem to the worker.)	Not Applicable	A radiation dose of up to 25 rem is acceptable if the likelihood of the accident is less than 1E-04 per year. (If the likelihood is less than 5E-07 per year, then the dose can be greater than 25 rem.)
5 rem	A radiation dose of up to 5 rem to the public and up to 25 rem to workers is acceptable if the likelihood of the accident is less than 1E-02 per year. (If the likelihood is less than 1E-04 per year, then the dose can be greater than 5 rem.)	A radiation dose of up to 5 rem to the public and up to 25 rem to workers is acceptable if the likelihood of the accident is greater than 1E-04 per year.	A radiation 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 radiation dose of up to 1 rem to the public is acceptable if the likelihood of the accident is less than 1E-02 per year. (If the likelihood is less than 1E-04 per year, then the dose can be greater than 1 rem.)
100 mrem	Not Applicable	Not Applicable	Not Applicable	A radiation dose of up to 100 mrem to the public is acceptable if the likelihood of the accident is greater than 1E-02 per year.

ISA = integrated safety analysis; LWR = light water reactor; mrem = millirem; rem = roentgen equivalent man.

4.2.5.2 Conversion of Accident Frequency and Dose-to-Health Effects

Development of surrogate risk evaluation guidance from schemes like those presented above requires selecting likelihood-dose threshold limit pairs that can be encompassed by the RIDM QHGs. If these limits cannot be shown to be encompassed by the QHGs or that there is reasonable comparison, then they might not serve as justifiable surrogates for QHGs. To this end, a simplified approach for converting radiation dose-to-health effects is described below and then used to confirm whether the selected radiation 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[13]), *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 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 on a static population that has 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 RIDM QHGs discussed in Section 4.1 and presented in Table 4.5 are 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 4.5 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 proposed pairs of likelihood-dose threshold limits discussed below as candidate surrogates to the QHGs do not address levels of health concern. However, levels of health concern are 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).

[13] The Interagency Steering Committee on Radiation Standards comprises 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 4.5. 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	QHG-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.
LCF = latent cancer fatality; mrem = millirem; QHG = quantitative health guideline.			

4.2.5.3 How NCT Is Considered in the Proposed Risk Evaluation Guidelines

The regulations in 10 CFR 71.71 (“Normal Conditions of Transport”) require that evaluation of each package design under NCT include a determination of the effect of specified conditions and tests (e.g., a temperature of 38°C and -40°C, water spray, a free drop test, compression test, and a penetration test). The regulations in 10 CFR 71.51 require that “there would be no loss or dispersal of radioactive contents—as demonstrated to a sensitivity of 1E-06 A_2 per hour, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging” as a result of the specified conditions and tests.

Appendix I.64 through I.70 of IAEA SSG-26 explain that the acceptable NCT leak rate of 1E-06 A_2 per hour is derived from limiting the effective dose to 2 rem for a worker spending time in a transport vehicle with the package with a specified air space volume, air exchange rate, and breathing rate for 1 year. This condition differs from a release caused by an accident, which is not expected to be continuous and will be over in far less than a year (e.g., in minutes). Moreover, this limitation does not consider possible impacts on the public just a worker that remains in a room with the package. Though the conditions involving extended exposure are different from a typical accident condition, limiting the effective radiation dose to 2 rem/year for NCT suggests this dose limit may also be appropriate for a worker involved with accidents with frequencies on the most likely side of the likelihood-dose matrix.

Regulations in 10 CFR 20.1301 (“Dose limits for individual members of the public”) stipulates a 0.1 rem annual exposure limit for the public from a licensed operation. This 0.1 rem/year limit is lower than the effective dose allowed for NCT of 2 rem cited above but can be applied to the public in the risk evaluation guidelines to conservatively bound the dose limit allowed by NCT. Given that the 2 rem effective dose allowed for NCT was specifically calculated for a worker and is lower than the 5 rem annual exposure limit stipulated by 10 CFR 1201 (“Occupational dose limits for adults”), the 2 rem limit could be used in the risk evaluation guidelines for the worker to bound the dose limit allowed by NCT.

Accordingly, given that it is assumed that the package design meets the requirements for NCT, development of risk evaluation guidelines was performed in a way that avoids defining pairs of likelihood-dose threshold limits as unacceptable when the limit is comparable to the risk to workers from NCT.

4.2.5.4 Selection of the Likelihood-Dose Pair Limits for the Surrogate Risk Evaluation Guidelines

This section discusses selection of the likelihood-dose pair limits as the surrogate risk evaluation guidelines and demonstration that they encompass the RIDM QHGs. The workable pairing of likelihood and consequence limits are described based on (1) the array of likelihood-consequence pairs presented above in Table 4.4, (2) the objective of the surrogate risk evaluation guidance to be encompassed by the QHGs, and (3) considerations of how NCT should be addressed.

- The risk associated with a radiation 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 it is unacceptable if the likelihood of the accident is more than 5E-07 per year. Although not cited for workers in the applications examined, this limit could also be considered applicable to workers.
- The risk associated with a radiation 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 it is unacceptable if the likelihood of the accident is more than 1E-06 per year.
- The risk associated with a radiation 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-05 per year or less, and it is unacceptable if the likelihood of the accident is more than 1E-05 per year.
- The risk associated with a radiation 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-04 per year or less; and it is unacceptable if the likelihood of the accident is more than 1E-04 per year.
- The risk associated with a radiation dose of 0.1 rem or greater to the public and 2 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 it is unacceptable if the likelihood of the accident is more than 1E-03 per year.
- The risk associated with a radiation dose less than 0.1 rem to the public and less than 2 rem or to workers is acceptable although the likelihood of the accident that produces this consequence is greater than 1E-03 per year.

Table 4.6 below presents an evaluation of the selected likelihood-dose limits based on the evaluation described above to see how they compare to the RIDM QHGs. It presents the conversion of each surrogate likelihood-dose 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 radiation dose to the public and consists of a conversion factor for mortality (i.e., 6E-04 LCFs per rem) and another conversion factor for morbidity (i.e., 8E-04 injuries per rem). The other set is for radiation dose to the worker, and also consists of a conversion factor for mortality (i.e., 5E-04 LCFs 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 RIDM 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 consequence threshold, and therefore, more conservative); QHG-3 for serious injury (cancer illness) to a member of the public; QHG-4 for acute fatality to workers (QHG-5 for LCF in workers might also be applicable but QHG-4 is used because of its lower consequence threshold, and therefore, more conservative); and QHG-6 for serious injury (cancer illness) to workers.

Note that the likelihood-dose pairs shown in Table 4.1, Table 4.2, and Table 4.3 are shown as intervals of radiation dose and accident frequency and the corresponding hypothetical evaluation plots presents a "stair-step" pattern for the criteria. From the selected likelihood-consequence pairs discussed in the bullet points above, the highest dose and frequency limit for a given "step" in the criteria were used to calculate the conversion to health effects in Table 4.6. This assures that the calculations are conservative.

Table 4.6. Comparison of Selected Dose-Consequence Limit Surrogates to the Limiting QHGs

Dose Limit (rem)	Frequency (per year)	Risk of Latent Fatality (fatalities/year)	QHG for Acute Fatality	Risk of Injury (injuries/year)	QHG for Serious Injury
Public		Conversion 6E-04 fatality/rem[(a)]	QHG-1 5E-07 fatality/year	Conversion 8E-04 injury/rem	QHG-3 1E-06 injury/year
Maximum[(b)]	5E-07	Maximum	OK	Maximum	OK
750	1E-06	5.0E-07	OK	6.0E-07	OK
25	1E-05	1.5E-07	OK	2.0E-07	OK
5	1E-04	3.0E-07	OK	4.0E-07	OK
1	1E-03	6.0E-07	OK[(c)]	8.0E-07	OK
0.1	8.3E-03[(d)]	5.0E-07	OK	1.0E-06	OK
Worker		Conversion 5E-04 fatality/rem[(a)]	QHG-4 1E-06 fatality/year	Conversion 7E-04 injury/rem	QHG-6 5E-06 injury/year
Maximum[(b)]	5E-07	Maximum		Maximum	OK
750	1E-06	5.0E-07	OK	3.0E-07	OK
100	1E-05	5.0E-07	OK	7.0E-08	OK
25	1E-04	1.25E-06	OK[(c)]	1.8E-08	OK
5	1E-03	2.5E-06	OK[(c)]	3.5E-06	OK
2	1E-03[(d)]	1.0E-06	OK	3.5E-06	OK

NCT = normal conditions of transport; QHG = quantitative health guideline; rem = roentgen equivalent man; RIDM = risk-informed decisionmaking.

(a) There is exception in this column for use of this conversion factor. For 750 rem, the conversion factor should be for acute fatality as opposed to latent cancer fatality because of the radiation dose is so high. Therefore, it is assumed that the conversion factor to health effects is a 0.5 probability based on guidance provided in Figure 5 of the Office of Environment, Health, Safety and Security Information Brief on "The DOE Ionizing Radiation Dose Ranges Chart" (DOE 2017) for when medical intervention is possible.

(b) The allowed radiation dose is the maximum that can result from a transportation accident involving the TNPP Package.

(c) Within the margin of error or reasonably close the RIDM QHG acute fatality limit which is a conservative limit for comparison to latent cancer fatalities.

(d) Doses lower than 0.1 rem are proposed to be acceptable at any frequency: (1) the dose limits in 10 CFR 20.1301 for individual members of the public from a licensed operation, and (2) because this prevents an effective dose limit that is allowed for NCT. For perspective, the highest accident frequency that still meets the RIDM QHG was calculated and is presented in the Frequency column.

The selected likelihood-dose limits for the public and workers are shown in the first two columns of Table 4.6. 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 columns indicate whether the selected likelihood-dose limits are bounded by the RIDM acute fatality QHGs. The first rows for the public and the worker indicate that if the frequency of an accident is 5E-07 per year (or less), then the allowed radiation dose is the maximum that can result from a transportation accident involving the TNPP Package. The calculated health effects for three of the low consequence but high-likelihood limits slightly exceed the QHGs, but they are reasonably close to meeting the RIDM QHGs given the level of resolution of calculations and latent fatalities from radiation doses of 25 rem or less are conservative compared to RIDM acute fatality QHG limits.

4.3 Proposed Surrogate Risk Evaluation Guidelines Established to Meet 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 4.1 discusses establishing the risk evaluation guidance on RIDM QHGs as proposed by the NRC in a report on risk-informed decisionmaking for activities that include transportation of nuclear material (i.e., the RIDM report [NRC 2008]). Section 4.2 describes the concept of using surrogate risk measures determined to be reasonably bounded by the RIDM QHGs for acute fatalities that are more practical to use and allow helpful comparisons to other radiological risk guidance. Sections 4.2.1 through 4.2.4 discuss risk evaluation guidance concepts that have been established for other nuclear applications using radiation dose and likelihood limits. Section 4.2.5 assesses how those radiation 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.

For the risk evaluation guidelines for TNPP transportation accidents to be appropriately applied, the term “accident” must be clearly defined and used in the PRA in a way that is compatible with criteria used in the risk evaluation guidelines. Given that the safety functions that must be preserved by the TNPP Package during transport as discussed in Section 5.2 are containment, shielding, and prevention of criticality, the accidents of interest addressed in the PRA are:

1. A release of radiological material to the environment
2. Direct radiation exposure from unreleased radiological material (e.g., due to degraded integrated internal or external shielding
3. A criticality that potentially involves both direct radiation and release of radiological material.

Table 4.7 presents the proposed risk evaluation guidelines in terms of likelihood and radiation dose consequences, which are reflective of the development results described in Section 4.2.5 and are compared against the applicable proposed QHGs in Table 4.6. The regions of acceptable and unacceptable risk from are shown graphically in Figure 4.7 and Figure 4.8 for the maximally exposed member of the public and the worker, respectively.

Table 4.7. Proposed Radiological Risk Evaluation Guidelines

Annual Accident Frequency (per year)[(a)]	Radiation Dose Consequence to the Maximally Exposed Member of the Public[(b)]	Radiation Dose Consequence to the Worker[(b)]	Risk Acceptability
≤5E-07[(b)]	≥750 rem TEDE[(c)]	≥750 rem TEDE[(c)]	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-05 and >1E-06	≥5 and <25 rem TEDE	≥25 and <100 rem TEDE	Acceptable
>1E-05	>5 rem TEDE	≥25 rem TEDE	Unacceptable
≤1E-04 and >1E-05	≥1 and <5 rem TEDE	≥5 and <25 rem TEDE	Acceptable
>1E-04	>1 rem TEDE	>5 rem TEDE	Unacceptable
≤1E-03 and >1E-04	≥0.1 and <1 rem TEDE	≥2 and <5 rem TEDE	Acceptable
>1E-03	>0.1 rem TEDE	>2 rem TEDE	Unacceptable
>1E-03	≤0.1 rem TEDE	≤2 rem TEDE	Acceptable

rem = roentgen equivalent man; TEDE = total effective dose equivalent.

(a) Determination of the accident frequency should account for multiple shipments per year, if applicable.

(b) The radiation dose consequences are presented as TEDE, which is based on the integrated committed dose to all organs, thereby accounting for direct exposure and the 50-year committed effective dose equivalent.

(c) If the accident frequency is <5E-07 per year, then the risk of the accident scenario is generally acceptable regardless of its radiation dose consequence. However, accidents with frequencies less than 5E-07 per year could be evaluated (e.g., using sensitivity studies) to confirm there are no cliff edge effects.

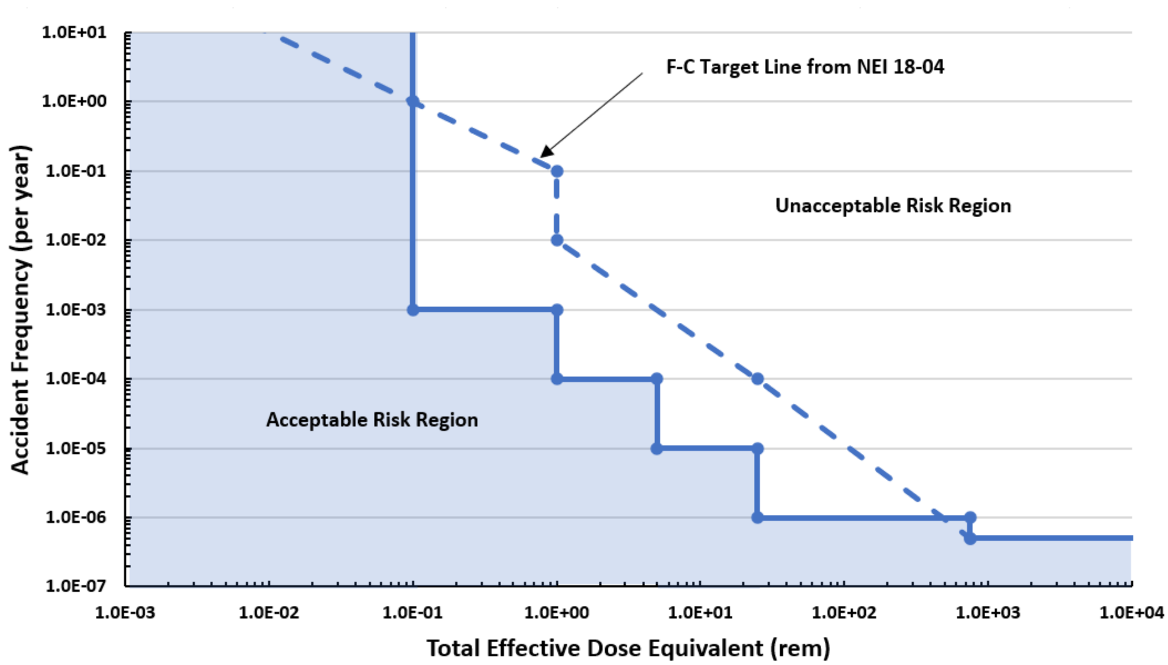


Figure 4.7. Proposed Risk Evaluation Guidelines Chart for the Maximally Exposed Member of the Public for Transport of a TNPP Package

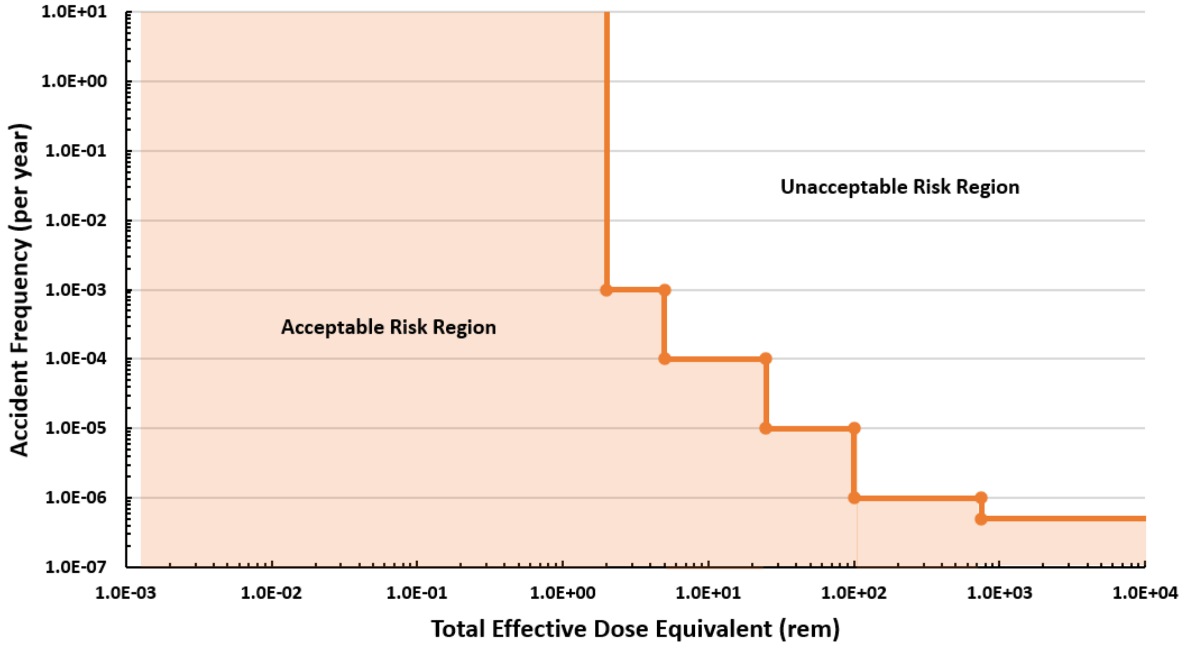


Figure 4.8. Proposed Worker Risk Evaluation Guidelines Chart for Transport of a TNPP Package

Given the challenges, such as lack of accident data for TNPP transport and modeling uncertainties associated with consequence analysis for these accidents, it is more practical to perform the PRA using modeling of bounding representative accidents rather than the modeling of a comprehensive set of accidents as done for nuclear power plants. Hence, the quantified risk results of bounding representative accidents are sufficiently conservative to 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 be more conservative, the dose consequences for each bounding accident or accident sequence within each frequency interval could be added together. The total dose for each frequency interval would then be compared against the risk evaluation guidelines. However, this seems overly conservative. It is noted that members of the public that reside or work near an nuclear power plant can largely be the same population over some meaningful fraction of the operating life of the facility. In contrast, the public who could be exposed to radiation dose consequences from an accident involving a TNPP transportation package are spread out along the route and are individually exposed to additional risk for a relatively short period of time.

To use the risk evaluation guidelines, transportation of a demonstration-phase TNPP Package is assumed to consist of one shipment in a year. If there are multiple shipments of the same TNPP in the same year, then application of the risk evaluations guidelines is still applicable because the risk criteria are provided on a per-year basis.

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 there were to be multiple transports of TNPPs over the same or overlapping routes. The application of the proposed risk evaluation guidelines will need to 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 Section 4.0 in terms of likelihood and radiation dose consequences achieve three objectives:

- They provide practical quantitative guidance to use when evaluating the risk acceptance of transport of the demonstration-phase package under the 10 CFR Part 71 exemption process supported by quantitative risk evaluation.
- They provide 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.
- They provide 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.

5.0 Characterization of Hazardous Conditions to Identify Accident Scenarios

This section describes the front-end of the proposed TNPP transportation PRA methodology—characterization of hazardous conditions to identify accident scenarios using primarily Project Pele Phase 1 design information. Section 5.1 provides characterization of the TNPP Package radiological material inventory. The radiological material inventory is the primary hazard, and an accurate accounting of the inventory is critical because radionuclides exist in different quantities and have different dose impacts on humans depending on the conditions of a TNPP transportation accident. Section 5.2 identifies and discusses the TNPP Package safety functions designed to prevent or mitigate accident scenarios associated with the radiological material inventory. Section 5.3 describes a process for identifying hazardous conditions associated with transport of the radiological material inventory. The purpose of this process is to find and develop TNPP transportation package accident scenarios based on existing hazards and the safety design of the TNPP transportation package definition.

5.1 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 (MAR) during a transportation event that could become the source term in an accident that releases radiological material and (2) define the potential sources of direct radiation exposure during a transportation accident.

Section 5.1.1 discusses the basis for the estimated radiological inventory possible during transport. Section 5.1.2 discuses release mechanisms from the TRISO fuel and fuel compact during reactor operation. This includes discussion of the results of efforts to estimate diffusion from TRISO fuel into the reactor based on published reports, and the impact of manufacturing specifications on these radiological inventory estimates. Section 5.1.3 provides the specific estimated radiological inventory that is used in the TNPP transportation PRA. Section 5.1.4 presents sources of radiation exposure in a transportation accident from different parts of the reactor given that there is diffusion from the TRISO during normal operation.

5.1.1 Bases for the 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 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 period of 3 years, an initial core loading of 0.18 metric tons of uranium, and a uranium enrichment of 19.5 weight-percent (BWXT 2022[14]). 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.

[14] BWXT Final Design Report, Table 2.3.1.1.3-1.

5.1.2 Mechanisms of Release from TRISO Fuel and Fuel Compacts

The Project Pele reactor core is contained within the Reactor Module and is housed within the pressure boundary created by the reactor vessel and primary cooling system. The core comprises fuel assemblies in coolant channels, moderator blocks, and control rods. The fuel assemblies consist of cylindrical TRISO fuel compacts stacked inside graphite fuel sleeves that have 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)[15] TRISO particles contained within a graphite matrix. The TRISO fuel is designed to make sure 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 and passive cooling scenarios do not allow the fuel to exceed its expected maximum allowable temperatures, which makes sure the TRISO fuel will maintain its structural integrity necessary for retention of fission products (BWXT 2022[16]). However, the current BWXT fuel compact design and fabrication is sufficiently different from the AGR fuel compact design that all its characteristics may not be bounded by the AGR testing. For instance, the larger size of the BWXT compact may affect the 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 achieve an acceptable matrix density. One significant concern is that compacting at higher pressure could damage more particles. Sensitivity studies of 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 5.1. An example of AGR fuel compacts, which have a smaller diameter than the Project Pele TNPP fuel compacts, is shown in Figure 5.2.

The HALEU oxycarbide TRISO fuel particles and fuel compacts release fractions of certain fission products[17] during normal operations and AOOs 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 from a transportation accident during shipment of the TNPP Package. This section provides an overview of the available information about these potential release mechanisms.

[15] The Project Pele microreactor UCO is a blend of HALEU uranium dioxide and uranium dicarbide (UC_2).
[16] BWXT Final Design Report, Section 2.3.1.1.3.
[17] 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.

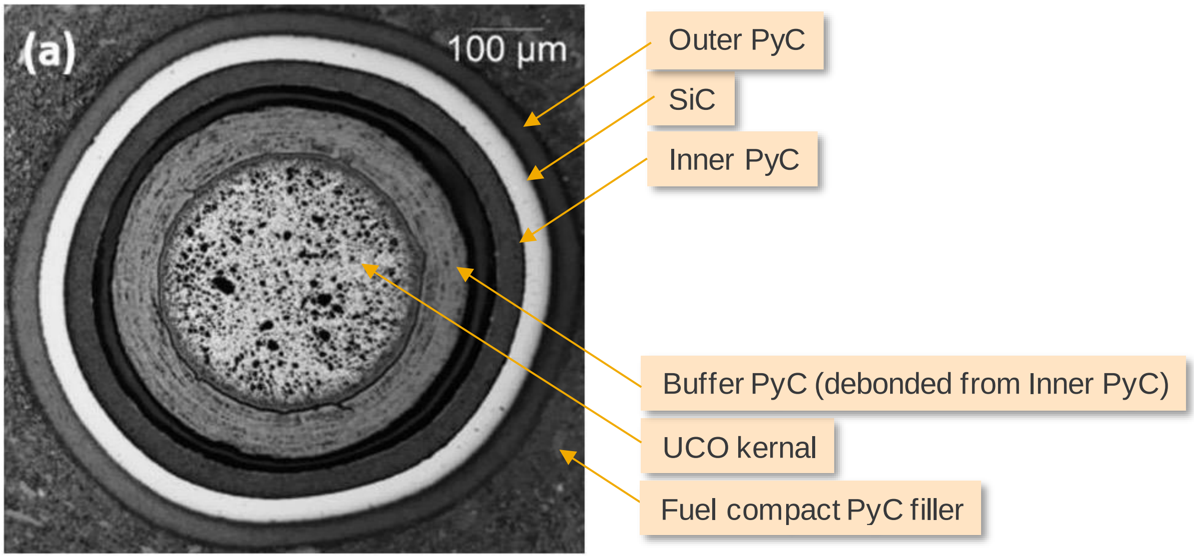


Figure 5.1. Cross Section of an Irradiated UCO TRISO Fuel Particle (EPRI 2020)

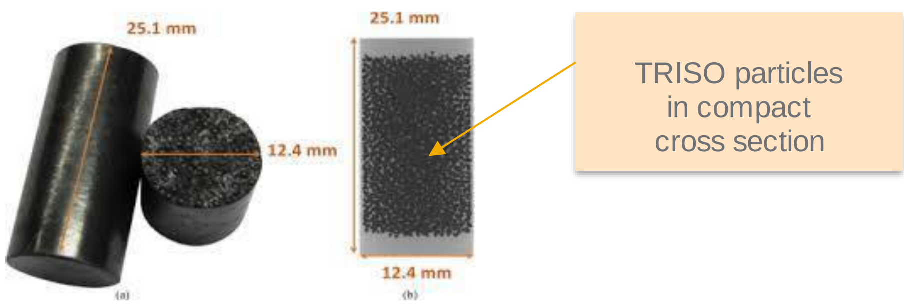


Figure 5.2. AGR TRISO Fuel Compacts (Harp et al. 2014)

5.1.2.1 Release from Normal Operations and Anticipated Operational Occurrences

The most prominent releases are isotopes of silver and a few other metal species that are well known to diffuse through and escape from TRISO particles within the temperature range that the fuel compacts experience during normal operation, which is up to 1,400°C for the TNPP (BWXT 2022[18]). 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 those 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 being saturated by competing species). The most common model for the

[18] BWXT Final Design Report, Table 4.5.1.2-3.

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.

During DBEs and BDBEs, significant heat soak circumstances may occur in which fuel compact temperatures are expected to rise from roughly 1,200°C up to roughly 1,400°C to 1,600°C. At these elevated temperatures, fission product releases increase because diffusion rates increase. However, for the demonstration TNPP transportation PRA, it was assumed that the TNPP being transported has not experienced a DBE or BDBE during operation, which would have affected diffusion rates during operation.

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—a fraction of failures that 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 program.[19] 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 a high technology readiness level and manufacturing readiness level. 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.

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

5.1.2.2 Release Mechanisms Caused by Transportation Accidents

In typical fixed site nuclear power plant deployments in the future, a TRISO-fueled nuclear power plant would meet design requirements for external hazards comparable to other fixed site nuclear power plants. However, transporting a TNPP introduces a broader range of external hazards to the reactor, including the following:

1. Possible exposure of the reactor to different or more severe hazards than those seen at the operating location
2. Tradeoffs in reactor design that are required to meet transportability considerations (e.g., maximum weight of a shippable unit
3. Possible introduction of human and mechanical faults not normally present in a fixed facility because of the requirement to be able to assemble and disassemble the TNPP to deploy and transport the unit.

Hazards associated with disassembly and reassembly 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 configured for transport over the road within the conterminous United States, with NRC approval of the TNPP Package and DOT regulation of its shipment during transit.

A principal difference posed by the user requirement of transportability is the 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 Package. Transportation accidents could involve kinetic hazards (such as impacts with vehicles or other objects), fire hazards, random failures, or human errors (e.g., human error in preparing the TNPP Package for transport), different exposure scenarios to natural hazards (such as earthquakes and tornadoes while in transport), and potential submersion in water.

Information about the performance of the TNPP transportation package or its contents during its transport is currently limited. Preliminary analysis of the TNPP transportation package 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 finite element analysis (BWXT 2022[20]). 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 releasable material. Because transportation testing has yet been performed on the TRISO fuel and compacts, engineering judgments are 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 the maintenance of subcriticality of the fuel. The current prototype TNPP design does not provide assurance of subcriticality in flooded conditions. Further design development

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

and testing are planned to be performed to provide assurance of subcriticality in flooded conditions for production units of the TNPP (BWXT 2022[21]). 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 as a result of 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 the time at temperature and the 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 because layer interactions typically initially unload the SiC layer as burnup occurs. At some point in a mid-burnup range, the SiC barrier begins to load rather than unload 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 percent fissions per initial metal atom with no obvious broad failures of SiC layers. At the expected burnups in the current TNPP demonstration design, which will be 10 percent fissions per initial metal atom or below, gas pressure should not cause the SiC layers to fail. 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 the release of gaseous radioactive material. More likely concerns are vibration spectra and consequences that are critical to understand both the fatigue tolerance of the fuel compacts and the near-fuel materials and tribology that could generate fine particulates that represent a source of released radioactive material.

Similarly, for fire hazards, knowing the irradiated thermal material properties is important to understanding the maximum temperatures that may occur in the fuel and core materials that hold up diffused fission product species. Time at high temperature is one of the common sources of releases 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. Because the TRISO fuel typically performs very well up to 1,400°C, the remaining decay heat is a significant factor in a consequence analysis. The codified regulatory pool fire test (see 10 CFR 71.73(c)(4)) is set at 800°C for a fully engulfing fire—conservative conditions for a liquid fossil fuel pool fire during transport. 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 and

[21] BWXT Final Design Report, Section 7.5.1.

proceed depending strongly on the 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 1,000°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 coolant 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 releases caused by air oxidation up to 1,400°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 coolant boundary.

Moreover, storage of significant Wigner energy in these graphitic structures in this reactor should not pose a significant risk in transportation because 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]).

5.1.2.3 Information Needs of Transportation Risk Analysis

With respect to transportation risk analysis, the likely inability to meet the entire suite of codified regulations for a Type B package, for the Project Pele TNPP Package (Reactor Module containing irradiated fuel and configured for transport), indicates that mechanical data for fresh and irradiated TRISO fuel and compacts, and other closely associated core components, are 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.

Making these risk judgments 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 that 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 material properties of TRISO fuel, compacts, and close core materials, particularly the 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 on 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 might 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 because 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 5.1. 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 that of 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 is 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 in understanding 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, so some limited annealing may affect the impact of radiation damage, especially for the SiC pressure boundary because that material begins to decompose at about 500°C beyond the fuel operating temperature of up to a maximum of about 1,400°C. The PyC materials and UCO kernel decompose at 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 because 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 of 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 more than 10 times the hardness of the typical high-density PyC material in the adjacent layers. The high-density PyC layer material has a hardness of ~3 GPa, whereas the SiC material has a hardness of ~40 GPa. 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 because 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 whether 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 because 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 of 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 to be 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. 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 outside of the TNPP in nearby materials (e.g., temporary shielding that includes a significant amount of water and various minerals that help hold up tritium).

The Project Pele core also substitutes beryllium for graphite as a moderator to a considerable extent. Graphite is known to absorb a considerable amount of tritium. The effect of irradiation on the material properties of beryllium oxide (i.e., mechanical strength and thermal conductivity specifically through irradiation-induced swelling and cracking) is not as mature as for graphite. So, the efficacy of the beryllium oxide 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 because 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).

Because no fractional flow rate particulate and gas cleanup system is currently envisioned in the Project Pele TNPP’s reactor gas system design (other than through gas withdrawal plus fresh gas from the coolant makeup generator), understanding the generation of fines and larger particulate, plate-out, and their carried radionuclide inventories is potentially a more significant part of the transportation risk assessment modeling because 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 gas-related oxidation chemistry in the primary pressure boundary should be evaluated (for example, oxide 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 filter and grab samples may also lead to learning that advises follow-on TNPP variant designs.

5.1.3 Estimated Radiological Inventory

The estimated radionuclide inventory (BWXT 2022[22]) was developed for multiple cooling time periods ranging between time zero (at reactor shutdown) and 2 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 more than 1,000 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 configured for transportation (TNPP Package).

The screening of radionuclides was conducted in two phases, the first based on total curies (radionuclides with more than 1 millicurie) 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 percent 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 specific to each radionuclide and are normalized based on radiological hazard during transport. The A_2 values consider 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 2013-04, "Content Specification and Shielding Evaluations for Type B Transportation Packages" [NRC 2013]). The derivation of the A_2 values is based on the Q system approach, which is described in Appendix I to the IAEA) Specific Safety Guide No. SSG-26 (*Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2018 Edition)* [IAEA 2022]). The A_2 values represent quantities of radionuclides that are tied to specific dosimetric consequences:

> The dosimetric basis of the A_1/A_2 system originally relied upon somewhat pragmatic assumptions. In calculating A_1 values, the whole body exposure was limited to 30 mSv at a distance of 3 m over a period of 3 h. Also, 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 0.1% of the package contents in such a manner that a bystander subsequently received an intake of 0.1% of this released material, page 272, (see IAEA 2022 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 through multiple potential exposure pathways. This screening methodology allows

[22] BWXT spreadsheet "B1.34-NuclideConcentrations(Ci)-Fuel.xlsx" provided on August 11, 2022.

for a targeted dosimetry approach weighted by consequence. In this 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 0.001 (0.1%) of the appropriate A_2 value from 10 CFR 71, Appendix A Table A-3. This approach is consistent with 10 CFR 71, Appendix A ("Determination of A_1 and A_2"), 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. 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.

The screening analysis resulted in the identification of 96 nuclides for a 90-day cooling period that are included in the consequence analysis. Appendix A 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 a negligible impact on dose results because almost all of the radionuclide inventory (i.e., more than 99.99999 percent) at 90 days is included in the consequence analysis.

5.1.4 Sources of Radiation Exposure During a Transportation Accident

The radionuclide inventory identified and described in Section 5.1.3 is a source of radiation dose to the worker and the public in the event of a transportation accident. This radiation dose is from the following 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
3. External exposure to material that is not released due to an accident, but that may or may not be fully 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 TNPP Package.

Dose sources 1 and 2 of are from 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 MAR. The development of the MAR is described in Section 5.1.4.1.

The third source of radiation 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 5.1.4.2 discusses the development of this inventory and summarizes the shielding design for the TNPP Package.

5.1.4.1 Development of Estimates of Material at Risk for Different Accidents

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 SSCs and other materials present in the Reactor Module configured as a TNPP Package. Depending on the magnitude and scope of the physical stress imposed on the TNPP Package contents because of the accident, the MAR may range from the total inventory in the module, as defined in Section 5.1.3, to a subset of the inventory in the module. The primary contributors to the MAR are as follows:

- Radionuclides contained within the TRISO fuel particles
- Radionuclides that were released from the TRISO fuel particles during normal reactor operations due to defects in the SiC coating
- Radionuclides that were released from the TRISO fuel particles during normal reactor operations due to in-service failure of the particles
- Radionuclides that were released from the TRISO fuel particles due to diffusion through the SiC coating
- 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 the following:

- Radionuclides produced from neutron activation of the TNPP Package SSCs, such as the RPV, Shield Tank, graphite moderator, etc.
- Radionuclides produced from neutron activation of any residual gas coolant
- Radionuclides produced from neutron activation of any residual water in the Shield Tank and other components
- Contamination located on inside surfaces of the TNPP Package, including the exterior surfaces of the RPV and primary cooling system
- Contamination located on the exterior surfaces of the TNPP Package.

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 gas coolant and water are removed from the Reactor Module prior to preparation for shipment, leaving minor residual quantities.
3. Contamination levels will be maintained low (in trace amounts) by standard operating procedures to maintain worker dose as low as reasonably achievable 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 used 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 A 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 5.1.

Table 5.1. Radionuclide Classification

INL/EXT-11-24034 Radionuclide Groupings	TNPP Radionuclides Represented (see Appendix A)
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)[(a)]	H-3
(a) 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 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:

- Heavy metal contamination in the outer graphite layer of the fuel particles and potentially in the compact graphite
- Particles that have SiC coating defects
- Incremental in-service fuel failures that occur under normal operations
- Diffusive release through the fuel particle SiC and other coatings.

The values for the fuel-related release parameters, based on assumed 50 percent and 95 percent confidence levels, are shown in Table 5.2. The in-service failure values in Table 5.2 are for a prismatic HTGR that has a reactor outlet temperature of 900°C. This is conservative for the prototype TNPP, which has a reactor outlet temperature of 760°C or 1,033 K (BWXT 2022[23]). These values are about a factor of five higher than allowable 95 percent confidence levels specified in the fuel performance requirements for historical HTGR designs under normal operations (EPRI 2020), and thus conservative for the risk analysis.

Table 5.2. TRISO Fuel Fabrication and Failure Parameters – Normal Operations[(a)]

Fission Product Class	Fabrication				Operations	
	Fraction Heavy Metal Contamination		Fraction SiC Coating Defects		In-Service Failures	
Confidence Limit	50%	95%[(b)]	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)[(c)]	1E-05	1E-04	1E-05	3E-05	1.4E-05	7E-05

(a) Except as noted, values are from INL/EXT-11-24034 (INL 2012).
(b) Heavy Metal Contamination 95% values are from INL/MIS-21-62587, *Microreactor TRISO Fuel Specification* (INL 2021).
(c) Fabrication and operations factors for hydrogen (H-3) are assumed to be the same as for noble gases.

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 the allowable 95 percent 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 percent 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 about TRISO fuel performance (EPRI 2020).

The fuel failure fractions presented in Table 5.2 were assumed to be normally distributed.

The second key element with respect to radionuclide release is the attenuation of the release. attenuation factors (AFs) 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

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

Pebble Bed Modular Reactor (INL 2012). 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 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 percent and 95 percent confidence levels, are shown in Table 5.3. Again, the AFs in Table 5.3 are conservative from the perspective that they are for a prismatic HTGR that has a reactor outlet temperature of 900°C. The AFs are assumed to have a lognormal distribution.

Table 5.3. Normal Operations Attenuation Factors[(a)]

Fission Product Class	Heavy Metal Contamination		Kernel		Diffusive Release Through 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)[(b)]	5	1.5	25	8.33	5E+07	5E+06	1	1

(a) Except as noted, Attenuation Factors are from INL/EXT-11-24034 (INL 2012).
(b) Fabrication and operations factors for hydrogen (H-3) are assumed to be the same as for noble gases.

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 that have 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 through 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 5.3 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 gases (Kr, Xe), halogens (I, Br), and tellurium than metal 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.

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 is assumed to be contained within the reactor core structure (i.e., fuel compacts) or in the reactor cooling system (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. The radiation dose consequence due to potential severe transportation accidents are presented in Sections 8.0.

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 following sequence of equations:

1. 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 A, Table A.1)

   $RP_{i,HMC}$ = release parameter (RP) or fraction of the radionuclide i inventory that is heavy metal contamination (HMC) (Table 5.2)

   $RP_{i,FD}$ = RP or fraction of the radionuclide i inventory that is in defective TRISO fuel particles (FD) (Table 5.2)

   $RP_{i,ISF}$ = RP or fraction of the radionuclide i inventory that is in TRISO fuel particles that failed in-service (ISF) (Table 5.2)

   $AF_{i,HMC}$ = HMC attenuation factor for radionuclide i (Table 5.3)

   $AF_{i,K}$ = fuel kernel (K) attenuation factor for radionuclide i (Table 5.3)

   $AF_{i,DIF}$ = diffusive release through TRISO coating (DIF) attenuation factor for radionuclide i (Table 5.3)

2. 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 5.3)

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

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

4. 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 5.1. 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 5.4. 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 5.4. 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

5.1.4.2 Unreleased Inventory that Could Be the Source of Direct Radiation Exposure

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 radiation dose pathway for 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 transportation shielding is damaged, or (2) an undesired situation develops resulting in longer worker times of exposure to radiation during the transport than planned. The sources of potential direct exposure overlap with those discussed in the previous section and consist of the following:

- Radioactive material contained within irradiated intact TRISO fuel particles
- Radioactive material released from the TRISO fuel particles and from heavy metal contamination during normal reactor operations that is held up in the reactor core structures and primary cooling system

- 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
  - RPV and Shield Tank
  - Saddle mount for the cooling water system or Shield Tank
  - TNPP Package structural components, including a custom-developed ISO container (CONEX box-like exterior structure of the Reactor Module[24]) and transportation shielding
  - Copper wiring for instrumentation and control
  - Lead, steel, and tungsten shielding that is integral to the reactor system.

5.2 Identification of TNPP Package Safety Functions

This section discusses the identification of TNPP Package transportation 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 during 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 SAR. The vendor transportation plan also discusses the need for passive heat removal while the reactor is in shutdown mode during transportation activities. 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 during Phase II to demonstrate transportation safety during NCT and HAC. Dynamic finite element analysis 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. The following discusses testing that has been performed for the two types of conditions cited.

- Normal Conditions of Transport – A shock load evaluation and a vibration load evaluation were performed using 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 CONEX box-like structure.
- 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 CONEX box-like structure, 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 imparted to the reactor internals. Damage to the shielding layers

[24] The term CONEX box is used to describe the exterior of the Reactor Module (i.e., custom-developed ISO container, which is like a CONEX box).

is expected in all angles of drop, and fracture of these 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.

During the hazardous condition evaluation, accident conditions were postulated from a broad range of possible hazards, which includes impacts from highway accidents as well as loss of safety function caused by other hazards extreme weather events (see results documented in Appendix B). 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 of the whole process. As described later in Section 5.3, identification of hazards, which is the starting point of the hazardous condition evaluation, was informed by vendor’s design information plus expert knowledge of additional hazards specific to transportation. Screening of hazardous conditions qualitatively judged to be of low risk is also part of the hazard analysis process.

Satisfactory performance of the TNPP transportation package under exposure to the entire suite of HAC test conditions prescribed in 10 CFR Part 71.73 is likely not feasible; therefore, the TNPP transportation PRA and associated risk information will be used to support the 10 CFR 71.12 exemption 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 relative 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 the overall containment boundary. The fuel is encased in ceramic material that is hardened against high temperature as described in Section 5.1. This barrier must be breached before radioactive material within the TRISO fuel is released during 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 SSC forming the containment boundary 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 5.1 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 coolant 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 a 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 SSC forming the containment boundary is the transportation package or module itself because it includes the CONEX box-like exterior structure 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 during a transportation accident.

Regarding the radiation shielding safety function, shielding external to the reactor vessel called transport shielding is expected to be used during transportation. This feature includes built-in and supplemental bolt-on lead encapsulated by carbon steel. This shielding is used in place of the water shielding during transportation because the amount of water shielding 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 a 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 is used to determine the risk associated with conditions for which the TNPP Package shielding (integrated internal and transport 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 a 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 process of developing accident scenarios by postulating these events in the hazard analysis. Given that 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 keep 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 is 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 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[25],” but is not needed to support 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 parameters such as airborne and direct radiation, reactor containment boundary pressure and temperature, control rod position, and shock and vibration. Decay time is expected to have a significant impact on the decay heat that is possible during

[25] BWXT Final Design Report, Appendix I – Essential Plans for Deployment, Appendix 1.1 Transportation Plan, page 28 of 86.

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 an 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 their impact on the containment safety function.

The conclusions based on examination of safety functions required during transportation are that they are the same as those 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 affect 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 and the reactor is reassembled after transport and operated.

5.3 Identification and Development of TNPP Transportation Package Accident Scenarios

This section describes the identification and development of TNPP transportation package 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 3.2, 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.

Section 5.3.1 describes the overall approach to the development of TNPP transportation accidents. Section 5.3.2 describes use of hazard analysis to identify hazardous conditions and make qualitative estimates of their risk and the assumption used in the analysis. Section 5.3.3 then uses the identified hazardous conditions to develop the detailed TNPP transportation accident scenarios. Section 5.3.4 describes using the detailed TNPP transportation accident to create the bounding representative accidents that are evaluated in the TNPP transportation PRA.

5.3.1 Approach to Development of Accidents Scenarios

In general, development of accident sequences for a PRA consists of three major elements:

1. Identification of the accident sequence initiating events
2. Development of 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
3. Delineation of 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 possible 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 packages such 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 highway structures, or the ground after a fall to a lower elevation), and non-collision accidents (e.g., rollover or jackknife). A fire or explosion could happen randomly while the transport vehicles are stationary or are in motion, or they could happen as a result of a highway vehicle accident. Accordingly, this kind of event tree is typically constructed using transportation accident data and perhaps geographic information system (GIS) data.

Event trees like those shown in NUREG-2125 are mostly useful for identifying the different kinds of highway vehicle accidents that can occur during transport, as 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 NRC transportation risk assessment reports describing 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 adequate likelihood and consequence analyses can be defined and performed. Nonetheless, 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 3.2, 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 deploy or Emergency Diesel Generators to start), the use of fault trees is not required or that 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 after an initiating event that results 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 the 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 systematic 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 transportation package PRA is a first-of-its-kind endeavor, and the associated hazards are apt to be different from those encountered during 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. The hazards identification and assessment are judged to meet those criteria described in the RIDM report and PNNL-31867 (Coles et al. 2021). Therefore, identification and assessment of possible TNPP transportation hazardous conditions are presented in Section 5.3.2. They are used to delineate detailed accident scenarios.

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 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. Another 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 PRA (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 bound 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 5.3.4.

5.3.2 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 who are familiar with TNPP vendor designs to generate a comprehensive list of hazardous conditions of concern. It also describes the information captured by the experts on the hazardous condition worksheets, which were used to 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 4.2.2 of this report), can also be used to support a PRA as mentioned in Section 5.3.1. The use of hazard evaluation methods is discussed in Appendix D of the RIDM report, which describes such methods as a Hazards and Operability Study, 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 nonquantitative risk characterization based on assessment of postulated hazardous conditions. This general approach was applied to identify hazardous conditions for TNPP transportation. The DOE standard defines the term “hazard analysis” as follows:

> 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 a 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 list 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 5.2 of this report and pertain to maintaining criticality safety, maintaining radiation shielding, ensuring containment of radiological material, and maintaining passive heat removal during transport. The Hazardous Condition Evaluation Worksheet used to capture the hazardous analysis results is discussed in Section 5.3.2.1. The assumptions made about the microreactor design and transport to support the hazardous condition evaluations are discussed in Section 5.3.2.2. Applicable hazards are used as 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.

5.3.2.1 Completion of 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 Package.

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 list of the primary assumptions used in the hazard analysis is presented in Section 5.3.2.2.

This process considered hazards such as the kinetic energy associated with moving vehicles and thermal energy associated fires such as diesel fuel fires. 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 the following hazard categories, as presented in Appendix B.

- Fire Hazard Events (Table B.1)
- Explosion Events (Table B.2)
- Kinetic Energy Events (Table B.3)
- Potential Energy Events (Table B.4)
- Loss of Containment Events (Table B.5)
- Direct Radiological Exposure Hazard Events (Table B.6)
- Criticality Events (Table B.7)
- Man-Made External Events (Table B.8)
- Natural Phenomena Hazards (Table B.9).

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 over-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 the CONEX box-like structure (e.g., associated with the module, the possible overpack, or system components external to the TNPP reactor containment boundary).”

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 5.2 (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 from unreleased radiological material (e.g., due to degraded shielding), or (3) a criticality that 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 an Initiator Likelihood that the hazardous condition occurs as defined in the Hazardous Event Summary. The Initiator Frequency (events per year) intervals are common ranges used in hazard analysis, as shown below:

- 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 affects the 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 5.1 of this report, the following contributors to the MAR are identified as applicable for each hazardous condition (i.e., accident scenario):

- Nongaseous fission products contained within the TRISO fuel or heavy metal contamination within the compacts that subsequently were 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 TRISO fuel that have diffused and are 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 (i.e., RPV or primary cooling system)
- 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 the following:

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
3. A qualitative description of the consequences of each hazardous condition
4. A qualitative description of the risk associated with each identified hazardous condition
5. Identification of preventive and mitigative features and systems to reduce the risk associated with the hazardous condition.

For the sake of aligning them 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 the 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 B 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).

5.3.2.2 Assumptions Related to Hazardous Condition Evaluation Assumptions

As stated in Section 5.3.2.1, 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 list 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 consequences of a transportation accident are identified in the sections of the report that address those analyses in detail (i.e., Sections 6.4 and 7.4, respectively).

1. The dominant radiation dose risk is associated with the Reactor Module configured as a TNPP Package because it contains the reactor, the fuel, portions of the primary cooling system and nearly all of the radiological material inventory. It is assumed that other portions of the primary cooling system that contain radioactive material such as the IHX and piping connecting the reactor to the IHX are transported in a separate module or containers as Low Specific Activity (LSA) or surface contaminated objects. It is assumed that radiological contamination or activated material that might exist in the other modules contributes little to radiation dose risk. Accordingly, the hazards analysis focuses on the TNPP Package exclusively.
2. The Reactor Module includes spent fuel after a specified period of decay as described in the consequence analysis presented in Section 7.0.
3. There is no gas cleanup 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. Submersion of the reactor vessel into a body of water could hypothetically lead to criticality based on the available design information.

5. No credit can be taken for a HMIS given that one has not yet been defined, although such a system could reduce the risk from certain kinds of accidents.
6. Loss of passive heat transfer from the reactor in the TNPP Package 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. There is only enough combustible material inside the Reactor Module in the form of cable and wire jacket and insulation to lead to a small fire.
8. No (or minimal) other flammable material, other than cable and wire jacket and insulation and minimal quantities of grease and oil, exist in the Reactor Module configured as the TNPP Package. No significant quantity of plastic wrapping or flammable packing material is used in this module.
9. There will be energized electrical components in the TNPP Package during transport associated with parameter monitoring, lighting, and ventilation.
10. The quantity of diesel fuel in the transport vehicle is about 300 gallons.
11. The only external fire of sufficient magnitude to propagate into the TNPP Package from the outside is a diesel fuel fire. Other external truck fires such engine fires and wheel or tire fires are not of sufficient magnitude to propagate into the TNPP Package.
12. For hard impacts followed by fire including a collision with a tanker carrying flammable liquid, the proportion of collisions that involves an explosion (e.g., deflagration or detonation) is very small compared to those that involve just fire. Therefore, the risk of this accident is considered to be bounded by a collision followed by fire which was an extremely low but much higher likelihood. As such, the dose consequences from a hard impact followed an explosion were not separately evaluated.
13. For the hazard analysis, there is no prohibition about transporting during inclement weather (e.g., extreme wind, rain, or temperature related scenarios were included). However, it was also assumed that a shipment would not deliberately be made in weather conditions so severe that the design/integrity of package would be exceeded.
14. 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. Moreover, the mechanical impact associated with very large truck crashes was assumed to dominate the accident phenomena, and as a result, weather phenomena were not factored into determination of source term factors (e.g., high wind was not assumed to increase the impact or dilute the concentration of released material).
15. There would be no specific control of passing or oncoming vehicles (i.e., collision with other vehicles was assumed possible) in development of the likelihood estimates.
16. Hazardous conditions qualitatively evaluated to be of low risk were not significant enough to be carried forward for detailed accident analysis. Low-risk scenarios were screened out because the likelihood was determined to be well below “Beyond Extremely Unlikely” (i.e., well below 1E-06 per year or below 5E-07 per year) or the consequences were determined not to significantly affect any of the TNPP Package radiological inventory contributors listed in Section 5.1.4.1.
17. The TNPP Package being transported has not experienced a DBE or BDBE during operation that would have affected diffusion rates during operation.

5.3.3 Identification and Development of Accident Scenarios for Detailed Analysis

The hazardous conditions discussed in the previous section that are identified and assessed in Appendix B of this report were used as the basis for defining the accident scenarios for the TNPP transportation PRA. Section 5.3.3.1 describes how the hazardous conditions presented in the Appendix were organized and delineated as TNPP transportation accident scenarios. Sections 5.3.3.2 through 5.3.3.33 provide detailed discussions of each identified accident, including discussion of parameters and factors that are important to accident sequence likelihood and consequence analysis.

5.3.3.1 Delineation of Accident Scenarios from the Identified Hazardous Conditions

As described in Section 5.3.2.1, the hazardous conditions listed in Appendix B were specifically formulated to contain the information needed to define accident scenarios. Accordingly, the hazardous conditions identified in Appendix B are essentially, with some adjustments, 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. An example of this are weather-related events (e.g., ice and snow events) that cause highway accidents. These events are considered to be encompassed by the highway accidents, because the highway accidents include 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). When it was determined that certain variations of an accident might produce different radiological consequences and likelihoods that are important to the risk conclusions, then those accidents were subdivided to account for those variations.

In a PRA, it is typical to organize accident scenarios is by initiating event categories; however, other factors also can play an important role such as accident phenomena and resulting radiation dose pathways to a worker or the public. Table 5.5 at the end of this section presents a condensed summary of potential TNPP accident scenarios after conditions estimated to be of low risk were screened out because their likelihood of occurrence was determined to be “Beyond Extremely Unlikely” or the consequences were determined to be insignificant because the MAR contributors discussed in Section 5.1 are negligibly affected. 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 in Section 3.2. As described above, the hazardous conditions listed in Appendix A were specifically formulated to contain the information needed to define accident scenarios. Accordingly, the hazardous conditions identified in Appendix A are condensed and presented in Table 5.5 as the accident scenarios. However, detailed descriptions of the accident scenarios are provided in Sections 5.3.3.2 through 5.3.3.33.

Many of the potential accidents presented in Table 5.5 are highway accidents typically considered in a transportation risk assessment involving a certified package. These include high-energy events that involve (1) collisions with other types of vehicles or with fixed objects; (2) drops from an elevated surface like a bridge, embankment, or overpass; or (3) rollovers. These same highway accidents could also involve fires resulting from the accident. This second set involving fires 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 5.1.4.1.

Another type of high-energy accidents is 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 when refueling). These fires are considered separately from fires that occur as part of a collision because:

1. There is only one damage and release mechanism.
2. A general fire or diesel pool fire involving the quantities of fuel carried on a transport vehicle (typically 8,000 to 12,000 gallons) 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 1,400°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 might occur at a higher frequency than a highway-related accident, even though they might result in lower levels of radiological consequence to workers or the public. Given that 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 a containment function, and these are locations in the package that might be vulnerable to failure. One set of containment issues concerns loss of the reactor containment boundary when it is not pressurized, while another set concerns loss of reactor containment boundary when it is pressurized. A third set of containment issues concerns the loss of containment from system elements that are not part of the reactor containment boundary.

The first set of low-energy accidents concerns a nonpressurized release from the reactor containment boundary for one of the following reasons:

- Random containment failure (e.g., failure of a seal, connection, or joint
- Vibration and shock from over-the-road travel
- Human error in packaging the system
- Human error during TNPP disassembly leading to undetected latent failures in containment
- Extreme cold that fails containment.

The second set of low-energy accidents concerns a pressurized reactor containment boundary due to reactor decay heat causing pressure that is released for the following reasons:

- High ambient air temperature that in combination with the residual decay heat pressurizes the reactor containment boundary
- 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 containment during transportation appears to be credible. For transportation, the gas that cools the reactor has been discharged (BWXT 2022[26]) 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[27]). During shipment, the Shield Tank would function as an impact limiter for the reactor vessel (BWXT 2022[28]), but this is not explicitly credited in the PRA. 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 decreased thereafter (BWXT 2022[29]). The decay heat generation about 12 days after reactor shutdown is estimated to be 20.2 kW (BWXT 2022[30]). PNNL estimates the decay heat generation will be about half this at 90 days after reactor shutdown. Given that 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 for discharging radioactive material from the reactor containment boundary.

The third set of low-energy events concern loss of containment from the CONEX- box-like structure caused by:

- Pressurization due to radiolysis of hydrogenous material (e.g., Shield Tank not fully drained) and possible hydrogen accumulation and ignition
- Pressurization caused by loss of ventilation or high ambient air temperatures
- Containment failure caused by random or vibration-caused failures
- Containment failure due to a hailstorm that causes general severe vibration.

For this third set, the radiological material available for release is primarily only loose contamination.

Three unique accident groups produce consequences to the worker and public through different radiation 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) as 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 of events concerns additional exposure of the worker to normal direct radiation caused by an increase in exposure time due to:

- Mechanical breakdown of the transport truck or trailer
- Technical problems with the package that requires worker attention
- Adverse weather that delays transport.

[26] BWXT Final Design Report, page 7-17.
[27] BWXT Final Design Report, page 3-9.
[28] BWXT Final Design Report, page 7-3.
[29] BWXT Final Design Report, Appendix III.45, Section 7.3.3.
[30] BWXT Final Design Report, Appendix III.45, page 17.

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:

- Addition of a moderator and potential change in core geometry
- 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.

Table 5.5 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 5.3.2.1, the accident likelihood categories qualitatively 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).

The accident consequence groups that were assigned by the hazard analysis team are presented in the last column of Table 5.5 and are defined using the MAR contributors described in Section 5.1.4.1. The consequence groups indicate which MAR contributors could possibly be released or become unshielded during 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 below:

- 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 coolant system comprising part of the TNPP 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 coolant system comprising part of the TNPP 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 coolant system comprising part of the TNPP reactor containment boundary
- Contamination outside the reactor.

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

Consequence Group E (noncriticality direct radiation exposure) pertains to potential direct exposure, as discussed in Section 5.1.4.2, from

- Existing TRISO fuel
- Fission products held up in the compact and other core structures
- Radiological material condensed or plated-out in the reactor coolant system comprising part of the TNPP reactor containment boundary
- Activated reactor system components such as the control rods and motors, the RPV, 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, respectively, in Sections 6.0 and Section 7.0 of this report.

Sections 5.3.3.2 through 5.3.3.33 contain detailed descriptions of each TNPP transportation accident scenario presented in Table 5.5. Each description includes discussion of factors important to a detailed accident analysis of the accident such as general statements about how the likelihood and consequence of these accidents can be determined.

Table 5.5. Identification of TNPP Accident Scenarios after Screening of Low-Risk Conditions (five sheets total)

Event Class	Candidate Accident Scenario Release of Radiological Material and/or Direct Radiation from Damaged TNPP Package (Reactor Containment Boundary and/or CONEX Box-Like Structure) Caused by:	Qualitative Likelihood Per Year Range	Qualitative Consequence[a] Group
1. Collision with vehicle[b]			
(a) Light	Collision with a light vehicle	Unlikely	B
(b) Heavy	Collision with a heavy vehicle	Unlikely	A
2. Collision (non-vehicle)[b]			
(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 a bridge, embankment, overpass)	Unlikely	A
3. Noncollision road accident[c]			
(a) Rollover	Rollover 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[b]			
(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 containing flammable material and fire	Collision of the transport vehicle with TNPP Package with a vehicle containing 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			
(a) Mechanical impacts and delta pressure	Impacts with moving and fixed objects, rollovers, drops, and delta-pressure impacts	Unlikely	A

Event Class	Candidate Accident Scenario Release of Radiological Material and/or Direct Radiation from Damaged TNPP Package (Reactor Containment Boundary and/or CONEX Box-Like Structure) Caused by:	Qualitative Likelihood Per Year Range	Qualitative Consequence[(a)] Group
6. Fire-only event			
(a) General	General fire in the CONEX box--like structure (e.g., associated with the module, possible overpack, or system components external to the TNPP reactor containment boundary)	Anticipated	B
(b) Diesel fuel	Diesel fuel fire associated with transport vehicle	Anticipated	B
(c) Oil and grease	Oil or grease fire in a CONEX box-like structure (e.g., associated with the module, possible overpack, or system components external to the TNPP reactor containment boundary)	Anticipated	B
7. Loss of Nonpressurized reactor containment boundary[(a)]			
(a) Random failure	Loss of nonpressurized reactor containment boundary caused random containment failure (e.g., seal, connection, or joint failure)	Unlikely	C
(b) Vibration and shock	Loss of nonpressurized 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 nonpressurized 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 nonpressurized reactor containment boundary containment 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 nonpressurized reactor containment boundary caused by extreme cold environmental temperature (e.g., beyond design limits of a containment feature during transport)	Anticipated	C

	Candidate Accident Scenario	**Qualitative Likelihood**	**Qualitative Consequence**[a]
Event Class	**Release of Radiological Material and/or Direct Radiation from Damaged TNPP Package (Reactor Containment Boundary and/or CONEX Box-Like Structure) Caused by:**	**Per Year Range**	**Group**
8. Loss of pressurized reactor containment boundary[a]			
(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 CONEX box-like structure in combination with failure of reactor coolant boundary caused by random failure, human error, vibration or extreme cold.	Anticipated	C
(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 coolant boundary caused by random failure, human error, or vibration.	Anticipated	C
9. Loss of general package containment[a] (not reactor coolant boundary) [d]			
(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, CONEX box-like structure and TNPP Package reactor containment boundary	Anticipated	D

Event Class	Candidate Accident Scenario Release of Radiological Material and/or Direct Radiation from Damaged TNPP Package (Reactor Containment Boundary and/or CONEX Box-Like Structure) Caused by:	Qualitative Likelihood Per Year Range	Qualitative Consequence[a] Group
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[e]			
(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
(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 possible change in core geometry caused by drop into water	Direct exposure and release of radiological material from 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

Event Class	Candidate Accident Scenario Release of Radiological Material and/or Direct Radiation from Damaged TNPP Package (Reactor Containment Boundary and/or CONEX Box-Like Structure) Caused by:	Qualitative Likelihood Per Year Range	Qualitative Consequence[(a)] Group
(b) Addition of moderator and possible change in core geometry caused by addition of water	Direct exposure and release of radiological material from inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash that results in fire and TNPP damage	Extremely Unlikely	F
(c) 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

CONEX = container express; RPV = Reactor Pressure Vessel; TNPP = Transportable Nuclear Power Plant.

(a) In many cases, the presence of parameter monitoring system that measures such parameters as radiation, pressure, and temperature is a mitigation feature that could affect both the likelihood and consequence estimates of these sequences.

(b) High wind, rain, snow, or ice can cause an accident and create special conditions that can affect radioactive material dispersion and transport.

(c) Rollovers and jackknives 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.

(d) Events that cause spread of contamination that affect the worker during transport due to events, environmental conditions, or phenomena that can occur during transport were identified as important hazardous conditions and are shown here for completeness but are not carried forward as accidents that contribute to bounding representative accidents for the reasons explained in Section 5.3.4.6.

(e) Events that cause additional radiation exposure to the worker during transport due to delays caused by environmental conditions or technical problems were identified as important hazardous conditions and are shown here for completeness but are not carried forward as accidents to contribute to bounding representative accidents for the reasons explained in Section 5.3.4.7.

5.3.3.2 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 CONEX box-like structure, and the possibility of damage to the TNPP reactor coolant boundary 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 nongaseous 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 can be set to lower 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) can be considered for its impact on any of the radiation dose pathway consequence analyses.

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

5.3.3.3 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 CONEX box-like structure and the possibility of damage to the TNPP reactor coolant boundary 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 nongaseous release from the TRISO fuel is assumed to occur. Given damage to the CONEX box-like structure, there is a possibility of damage to the transport shielding and increased direct radiation from the TNPP Package.

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 light vehicle. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) can be considered for its impact on any of the radiation dose pathway consequence analyses.

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

5.3.3.4 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 CONEX box-like structure that is higher than standard height (e.g., high cube CONEX), which seems possible given the vendor design information. Like the other highway accidents, there is a strong possibility of mechanical damage to the custom-developed ISO container CONEX box-box structure and the possibility of damage to the TNPP reactor coolant boundary itself. Accordingly, some fraction of all available MAR is assumed to be released including some fraction of the TRISO fuel. Given damage to the CONEX box-like structure, there is a possibility of damage to the transport shielding and increased direct radiation.

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

The likelihood of this accident occurring 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 radiation dose consequences because of collision with one type of object, then consideration can be given to analyzing the collision with the object as a separate scenario with a lower accident frequency.

5.3.3.5 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 CONEX box-like structure and the possibility of damage to the TNPP reactor coolant boundary package itself. Accordingly, some fraction of all available MAR is assumed to be released including some fraction of the TRISO fuel. Given damage to the CONEX box-like structure, there is a possibility of damage to the transport shielding and increased direct radiation. Unlike other highway accidents, there is the possibility that the TNPP Package could be dropped into a body of water sufficient to flood the core. This version of the accident is identified in Table 5.5 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 radiation 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 can be given to analyzing the accident as a separate scenario if the scenario results in high radiation dose consequences. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) can be considered for its impact on any of the radiation dose pathway consequence analyses.

The likelihood of this accident occurring 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 radiation dose consequences and is limited to certain features, then consideration can be given to analyzing those accidents as separate scenarios that have lower accident frequencies.

5.3.3.6 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 CONEX box-like structure and the possibility of damage to the TNPP reactor coolant boundary itself. Accordingly, some fraction of all available MAR is assumed to be released including some fraction of the TRISO fuel. Given damage to the CONEX box-like structure, there is a possibility of damage to the transport shielding and increased direct radiation.

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-case accident is rare and the consequences are high, then consideration can be given to analyzing the accident as a separate scenario if the scenario results in high radiation dose consequences. As with all highway accidents, the possibility that it was caused by adverse weather (e.g., rain or snow) can be considered for its impact on any of the radiation dose pathway consequence analyses.

The likelihood of this accident occurring should be based on non-collision rollover accidents stated above. If the scenario results in high radiation dose consequences and is limited to certain kinds of rollover accidents, then consideration can be given to analyzing those accidents as separate scenarios with lower accident frequencies.

5.3.3.7 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 trailer and TNPP Package. This might lead to some impact inside the CONEX box-like structure for objects (e.g., tools) that become unrestrained, but such impact is not expected to damage the TNPP reactor. Hence, there is some possibility of mechanical damage to the container contents and the possibility of damage to the TNPP reactor or containment boundary. Accordingly, some small fraction of all available MAR might be released, but not the TRISO fuel itself.

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) can be considered for its impact on any of the radiation dose pathway consequence analyses.

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

5.3.3.8 Accident 4(a) – Collision with a Vehicle, Fixed Object, or Rollover and a 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 CONEX box-like structure and the possibility of damage to the TNPP reactor containment boundary itself. Given damage to the CONEX box-like structure, there is a possibility of damage to the transport shielding and increased direct radiation. 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 TNPP Package (e.g., fail reactor containment boundary 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 radiation dose pathway consequence analyses.

The likelihood of this accident occurring 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.

5.3.3.9 Accident 4(b) – Collision with a Tanker and a 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 containing 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 CONEX box-like structure and the possibility of damage to the TNPP reactor containment boundary itself. Given damage to the CONEX box-like structure, there is a possibility of damage to the transport shielding and increased direct radiation. 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 that might 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 reactor containment boundary 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 that might cause greater mechanical impact, but perhaps less thermal damage. This variation would require a different damage ratio, airborne release fraction, and respirable fraction than those for fire events and could be addressed separately and be determined to be bounded by the consequences of a large fire. However, it is assumed for analysis performed for this accident that the explosion risk is bounded by the fire risk, as justified in Section 5.3.2.2. 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 occurring 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.

5.3.3.10 Accident 4(c) – Drop to a Lower Elevation Surface and a 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 CONEX box-like structure and the possibility of damage to the TNPP reactor containment boundary itself. Given damage to the CONEX box-like structure, there is a possibility of damage to the transport shielding and increased direct radiation. 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 from 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 reactor containment boundary 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 could be the sum of the ratio and fractions used for mechanical impact and fire. 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 could affect the duration assumed in the consequence analysis.

The likelihood of this accident occurring 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.

5.3.3.11 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 Package (i.e., CONEX box-like structure, TNPP reactor, core and TRISO containment boundaries) 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 influences. Like most highway accidents, there is a strong possibility of mechanical damage to the CONEX box-like structure and the possibility of damage to the TNPP reactor containment boundary. 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 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 could dilute or decrease radiation dose to the worker and public. However, dilution 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 occurring 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. This cause for an accident might be included with other causes that result in similar consequences or treated separately. Also, the wind events could be separated from the tornado events, particularly if the tornado events result in significantly higher radiological consequences.

5.3.3.12 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 a general fire in the CONEX box-like structure (e.g., associated with the module, possible overpack, or system components external to the TNPP reactor containment boundary). 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, the fire originates in the CONEX box-like structure that impacts the TNPP reactor containment boundary. Accordingly, some fraction of the available MAR is assumed to be released but not 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 1,400°C per INL/EXT-16-40784 (Petti et al. 2017). These are much higher temperatures than testing predicts for a 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 that the fire originates in or directly around the TNPP RPV, that damage could be greater than that caused by a fire that originates from outside the CONEX box-like structure (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 become airborne as discussed in Section 5.3.3.8. However, a general fire is not expected to be big enough to create these effects.

Given the unusual load, the likelihood of this accident occurring should not be based on truck fires, but rather on 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.

5.3.3.13 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 TNPP 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, the fire originates outside the CONEX box-like structure but propagates to the inside to impact the TNPP reactor containment boundary. Some fraction of the available MAR is assumed to be released but not the TRISO fuel itself. As stated above, thermal testing of TRISO fuel suggests that the fuel remains intact with very low radiological material release at temperatures up to 1,400°C (Petti et al. 2017). These are much higher temperatures than testing predicts for large-scale diesel pool fires according to testing by sources such as the 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 consistent 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 CONEX box-like structure, the damage inside the CONEX box may be less than that for fires that originate inside the CONEX box. Also, as explained in Section 5.3.3.8, 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 occurring 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. It might be difficult to separate the fires that occur as a result of impact from fires that occur without impact. If that is the case, then the frequency of this accident might be conservatively developed by overestimating the accident frequency using the frequency of all very large truck fires.

5.3.3.14 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 the CONEX box-like structure (e.g., associated with the module, possible overpack, or system components external to the TNPP reactor containment boundary). This accident could be considered a subset of Accident 6(a), which is a general fire that originates inside the CONEX box-like structure including the TNPP reactor components, as described in Section 5.3.3.12. However, if it is determined that the Reactor Module configured as the TNPP Package itself 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, the fire originates in the CONEX box-like structure in or around the TNPP reactor containment boundary. 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 1,400°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 that the fire originates in or directly around the TNPP Package, that damage could be greater than a fire that originates from outside or close proximity to the TNPP Package (e.g., a diesel fuel fire). Moreover, if it is determined that the TNPP Package has more than minimal quantities of oil (or grease) then the fire might produce even greater damage.

Given the unique load, the likelihood of this accident occurring should not be based on truck fires, but rather on 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. However, in this case, it may be beneficial to combine this scenario with other non-impact–related fire accidents if the likelihood and radiological consequences of the accidents are about the same.

5.3.3.15 Accident 7(a) – Loss of the Reactor Containment Boundary (Nonpressurized) Caused by a Random Failure

This accident concerns release of radiological material to the environment from the reactor coolant boundary caused by a containment failure (e.g., seal, connection, or joint failure). Given that the reactor and heat exchanger will likely be separated into two modules, a containment feature is needed at the points where systems are separated and could be vulnerable to failure. Some fraction of the available MAR is assumed to be released, but not 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 coolant boundary.

The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect the fact that containment has been breached and a nonpressurized condition exists, so there is limited motive force to discharge radiological material from the reactor coolant boundary of the TNPP 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 occurring should be based on the likelihood of random containment failures (e.g., seal, connection, or joint failure) occurring. 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 nonpressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

5.3.3.16 Accident 7(b) – Loss of the Reactor Containment Boundary (Nonpressurized) 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 coolant boundary containment 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, but not 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 coolant boundary. Vibration and shock from road travel could also contribute to loosening radioactive material plated-out in the reactor coolant boundary and surface material diffused onto the compact and other core structures. In this scenario, the reactor containment boundary is assumed to not 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 the fact that a nonpressurized containment has been breached and that vibration and shock may have loosened radioactive material inside the reactor containment boundary such as the core structure.

The likelihood of this accident occurring should be based on the likelihood of containment failures (e.g., component material, seal, connections, or joint failure) occurring that are caused by 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 nonpressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

5.3.3.17 Accident 7(c) – Loss of the Reactor Containment Boundary (Nonpressurized) Caused by Human Error when 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 Reactor Module for transport (e.g., sealing the reactor containment boundary). Systems associated with the reactor coolant boundary could be separated requiring sealing of the interfaces. Some fraction of the available

MAR is assumed to be released, but not 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 coolant boundary. In this scenario, the reactor containment boundary is assumed to not 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 the fact that a nonpressurized 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 coolant boundary.

The likelihood of this accident occurring should be based on human error associated with preparing the TNPP Reactor Module 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 nonpressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

5.3.3.18 Accident 7(d) – Loss of the Reactor Containment Boundary (Nonpressurized) Caused by Human Error During 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, but not 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 coolant 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 the fact that a nonpressurized containment has been breached, so there is limited motive force to discharge radiological material from the reactor coolant boundary of the TNPP Package. It should reflect the fact that road vibration and shock may have loosened radioactive material from the surfaces inside the reactor coolant boundary.

The likelihood of this accident occurring 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 nonpressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

5.3.3.19 Accident 7(e) – Loss of the Reactor Containment Boundary (Nonpressurized) 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 coolant 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, 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, but not 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 coolant boundary. In this scenario, the reactor containment boundary is assumed to not 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 the fact that a nonpressurized containment has been breached, so there is limited motive force to discharge radiological material from within the TNPP Package. It should reflect the fact that road vibration and shock may have loosened radioactive material from the surfaces inside the reactor coolant boundary.

The likelihood of this accident occurring should be based on the likelihood of containment failures (e.g., component material, seals, joints, and connections) occurring as the result of 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 nonpressurized reactor containment boundary accidents, particularly if the likelihood and radiological consequences of the accidents are about the same.

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

This accident concerns release of radiological material to the environment from a pressurized reactor coolant boundary containment caused by residual heat buildup due to loss of heat transfer from mechanical impacts involving the TNPP Package (e.g., damage of vents or influences 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 5.3.3.1.) Some fraction of the available MAR is assumed to be released, but not 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 coolant boundary. The pressurized condition provides a mechanism for discharging 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 the fact that containment has been breached and a pressurized condition exists that provides a mechanism for discharging radioactive material consistent with the conditions for this scenario.

The likelihood of this accident occurring should be based on the likelihood of damage of vents or influences on the heat transfer pathway that could occur as a result of the package or other objects moving within the custom-developed ISO container in combination with reactor containment boundary failures like those discussed in Sections 5.3.3.15 through 5.3.3.18. 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.

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

This accident concerns release of radiological material to the environment from the pressurized reactor coolant boundary containment caused by 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 5.3.3.1.) Some fraction of the available MAR is assumed to be released, but not 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 coolant boundary. The pressurized condition provides a mechanism discharging 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 the fact that the containment has been breached and a pressurized condition exists that provides a mechanism for discharging radioactive material consistent with the conditions for this scenario.

Given that some level of pressurization will exist from decay heat, the likelihood of this accident occurring should be based on the likelihood of very high ambient air temperatures in combination with reactor containment boundary failures like those discussed in Sections 5.3.3.15 through 5.3.3.18. 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 8(a) and 8(c), particularly if the likelihood and radiological consequences of the accidents are about the same.

5.3.3.22 Accident 9(a) – Loss of General Package Containment Caused by Radiolysis

This event 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 reactor containment boundary itself but inside the CONEX box-like structure of TNPP Package. The pressurized condition provides a mechanism for discharging 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 TNPP 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 event occurring 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, accidents that involve loss of general package containment might be combined.

5.3.3.23 Accident 9(b) – Loss of General Package Containment Caused by High Temperature

This event concerns release of radiological material (e.g., contamination) in escaped air from the TNPP containment within the package to the environment caused by pressurization due to loss of ventilation or high ambient air temperature during transport. This primarily concerns contamination outside the TNPP reactor containment itself but inside the CONEX box-like structure of the TNPP Package. The pressurized condition provides a mechanism for discharging 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 CONEX box-like structure but outside the reactor coolant boundary 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 event occurring 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, accidents that involve loss of general package containment might be combined.

5.3.3.24 Accident 9(c) – Loss of General Package Containment Caused by Random Failures

This event concerns release of radiological material (e.g., contamination) in escaped air from the TNPP Package to the environment caused by failure of reactor 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 reactor coolant boundary itself but inside the

TNPP CONEX box-like structure. 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 within the CONEX box-like structure of the TNPP transport 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 event occurring is based on the estimated probability of the random failure 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.

5.3.3.25 Accident 9(d) – Loss of General Package Containment Caused by a Hailstorm

This event concerns release of radiological material from the TNPP Package to the environment caused by failure of package containment caused by a severe hailstorm that causes significant vibration of the transport vehicle, and TNPP Package. This primarily concerns contamination outside the TNPP containment boundary itself but inside the CONEX box-like structure of 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 CONEX box-like structure 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 event occurring 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.

5.3.3.26 Accident 10(a) – Loss of Shielding Caused by the Drop of the Vehicle to a Lower Surface

This accident concerns exposure of the worker to direct radiation from loss of transport 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 radiation 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 TNPP Package drop event. Regarding Accident 4(c), which is a TNPP Package drop event and subsequent 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), 2(b), and 4(c) might be considered together so that all applicable dose pathways are addressed.

The likelihood of this accident occurring should be based on the occurrence of 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 radiation dose consequences and is limited to certain features, then consideration should be given to analyzing those accidents separately with a lower accident frequency.

5.3.3.27 Accident 10(b) – Loss of Shielding Caused by Impact During a Vehicle Collision

This accident concerns exposure of the worker to direct radiation as a result of the loss of transport shielding (e.g., bolt-in shielding and cable mesh) 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) 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 radiation 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 subsequent 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 occurring should be based on the occurrence of 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 radiation dose consequences and is limited to certain features, then consideration should be given to analyzing those accidents separately with a lower accident frequency.

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

This event concerns increased exposure of the worker to normal levels of radiation at the CONEX box-like structure 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 CONEX box-like structure 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 controls associated with a radiation protection program can be considered.

The likelihood of this event occurring could be based on the statistics or truck breakdowns on the highway. It may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set.

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

This event concerns increased exposure of the worker to normal levels of radiation at the boundary of CONEX box--like structure caused by breakdown or technical issues associated with the TNPP Package, the CONEX box, or transport 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 CONEX box-like structure 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 controls associated with a radiation protection program can be considered.

The likelihood of this event occurring 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 failures 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). It may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set.

Events that cause spread of contamination that affect the worker during transport due to events, environmental conditions, or phenomena that can occur during transport were identified as important hazardous conditions but are not incorporated into a bounding representative accident for the reasons explained in Section 5.3.4.6.

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

This event concerns increased exposure of the worker to normal levels of radiation in close proximity to the TNPP transportation package 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 TNPP Package 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 controls associated with a radiation protection program can be considered.

The likelihood of this event occurring could be based on the estimated frequency of severe weather along the route significant enough to delay transport. It may be advantageous to combine certain accidents using the highest likelihood and consequence outcome of the set.

Events that cause additional radiation exposure to the worker during transport due to delays caused by environmental conditions or technical problems were identified as important hazardous conditions but are not incorporated into a bounding representative accident for the reasons explained in Section 5.3.4.7.

5.3.3.31 Accident 12(a) – Criticality Accident Caused by the Addition of Moderator Caused by Drop into Body of Water

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 Package 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 reactor 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 by 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 significant damage to the TNPP Package including the reactor cooling boundary and the reactor core itself. For the direct radiation portion of the consequence (non-release), the loss of shielding should be consistent with the drop accident that can create significant damage to the shielding. 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 radiation dose pathway consequence analyses.

The likelihood of this accident occurring should be based on the occurrence of 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 is about 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.

5.3.3.32 Accident 12(b) – Criticality Accident Caused by the Addition of Moderator Caused by Fire Suppression

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 inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash and possible change in core geometry that results in fire and TNPP Package damage. 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 by 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 significant 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 significant damage to the shielding. 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 radiation dose pathway consequence analyses.

The likelihood of this accident occurring should be based on the occurrence of accidents that involve a significant impact and ensuing fire plus a set of probabilities to account for the other conditions needed to create a flooded criticality. Conditional probabilities need to account for the fact that (1) the TNPP Package and internal reactor coolant boundary would have to be damaged in a way that makes it possible for water (or other material) to enter the core and (2) fire suppression water (or other material) would have to be directed at the TNPP Package in manner that results in filling and inundating the core even though the reactor vessel and associated containment may not necessarily be directly in or near the fire.

5.3.3.33 Accident 12(c) – 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 under 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 significant 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 transport radiation shielding of the TNPP Package.

The likelihood of this accident occurring should be based on the occurrence of 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 is 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.

5.3.4 Development of Bounding Representative Accident Scenarios

As described at the end of Section 5.3.1, there is sufficient rationale and a practical advantage to only performing detailed accident analysis of 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 the determination of TNPP transportation risk and are used in this report to determine the level of impact that certain PRA assumptions have on the risk results.

The 32 accident scenarios discussed in detail in Section 5.3.3 and summarized in Table 5.5 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 of occurrence, and potential radiation 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 makes 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 occurring 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, 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 nonpressurized reactor containment boundary
6. Release of radioactive material from other than the nonreactor containment boundary element of the package
7. Unplanned increases in exposure time to radiation
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 5.3.4.9 includes Table 5.6, which lists and defines the resulting bounding representative accidents. Accident frequency development for bounding representative accidents is discussed in Section 6.5 and radiation dose consequences for these accidents are presented in Section 7.5.

5.3.4.1 Fire-Only Accidents

This section describes development of bounding representative accidents for the fire-only accidents. There are six separate fire accidents with three pertaining 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 inside the CONEX box-like structure (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 though oil and grease are expected to be present in limited quantities in the transport CONEX box-like structure. Though these fires could potentially affect the TNPP reactor coolant boundary within the package directly because they are inside the CONEX box-like structure, they are apt to be small, given the lack of flammable material in the container. There are no electrically active systems inside the reactor, so fire initiated inside the reactor is not expected during transport. The other fire (i.e., Accident 6(b)) is a diesel fuel fire that originates outside of the CONEX box-like structure of the TNPP Package. This fire is likely to be much bigger but must propagate into the CONEX box-like structure to cause damage to the TNPP reactor containment, RPV, and possible degradation of internal integrated shielding within the packaging. If the internal and external fire-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 CONEX box-like structure). Additionally, the likelihood of spurious fire in the CONEX box-like structure 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 CONEX box, which is more likely.

Therefore, the three fire-only accident scenarios are divided into two cases: (1) BRA) 1 and (2) BRA 2.

BRA 1 is a fire that originates inside the CONEX box-like structure. It is a general fire caused by sources such as an electrical cable fault that ignites combustible material associated with the package. Combustible material includes an oil or grease that is ignited by 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 containment boundary SSCs. Due to the limited space that the fire can occupy, failure of the reactor containment boundary, marked degradation of integrated internal shielding, and release of materials is not postulated for this event.

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

BRA 2 is a diesel fuel fire that originates outside the CONEX box-like structure, propagates into the CONEX box, and ignites combustible material, which damages the reactor coolant boundary of the RPV. The assumed quantity of diesel fuel is limited to the maximum possible fuel in transporter fuel tanks (e.g., about 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.

5.3.4.2 Road Impact Accidents

This section describes development of bounding representative accidents for the impact-only accidents (no fire) that occur on the highway. 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 5(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 a 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 inside the container for objects that become unrestrained (e.g., tools), but such impacts are not expected to damage the internal safety components of the TNPP Package. The high-wind event (i.e., Accident 5(a)) can lift or remove the TNPP Package from the conveyance 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 the most significant forces on the TNPP Package. These forces can create damage to the TNPP Package components (i.e., CONEX box-like structure, containment, RPV, reactor core, integrated internal shielding, etc.) and its transport shielding, which results in release of radiological material and increased direct radiation. Conversely, impact with light vehicles or objects, and impacts that 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), falls to a lower elevation (e.g., drop from a bridge), and rollovers, which can result in hard impact on the asphalt or concrete roadway. It is assumed that this bounding accident results in damage to the TNPP Package and associated 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 to 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 occurring 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 a 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 of 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 transport 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 occurring 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 but there is some degradation in the transport shielding which is enough to get radiation streaming through an open gap in the CONEX box-like structure of the TNPP Package. These light-impact accidents are defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or an impact with a light vehicle that is not severe (e.g., results in property damage only). Accordingly, the likelihood of BRA 4L is based these kinds of accidents. A precise definition of yielding versus unyielding objects is discussed in the frequency estimation in Section 6.3.1.1 and as presented in Table 6.10.

5.3.4.3 Road Impact and Subsequent Fire Accidents

This section describes development of bounding representative accidents for the highway 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 a 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, significant mechanical damage is assumed to occur caused by impact, as discussed for accidents addressed in Section 5.3.3, along with fire that 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.
- 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.

The impact and fire can also cause damage to the transport shielding of the TNPP Package. Therefore, determination of the damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this bounding representative accident considers these conditions and 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 that has the potential to create strong mechanical impact forces in a collision. An additional consideration is the fact that the tanker could contain explosive material, which might cause greater mechanical impact, and perhaps less thermal 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 a collision with a tanker carrying flammable material. BRA 6 is then defined as collision with a tanker carrying flammable material and subsequent fire. For both BRAs 5 and 6, it is assumed that the proportion of collisions that involves an explosion (e.g., deflagration or detonation) is very small compared to collisions that involve just fire whose likelihood is already extremely small. Therefore BRA 5 and BRA 6 were defined as leading only to fire and assumed to be bounding.

The likelihood of BRA 5 occurring is based on the sum of the accident frequencies for the scenarios assigned to this group and the likelihood of BRA 6 occurring 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.

5.3.4.4 Loss of the Nonpressurized Reactor Containment Boundary

This section describes development of the bounding representative accidents for loss of package containment events specifically for a nonpressurized release from the reactor coolant boundary containment that is 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:

- Random containment failure (e.g., failure of a seal, connection, or joint
- Vibration and shock from over-the-road travel
- Human error in packaging the reactor containment boundary
- Human error during TNPP disassembly leading to undetected latent failures in containment
- Extreme cold that causes the containment to fail.

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 cause one of the accident scenarios in this group, are also a factor in the other scenarios of this group even though they do not cause the breach. Table 5.5 indicates that vibration and shock could loosen surface material held up in the compact or other core structures, adding to the radiological material that might be released (i.e., the shock and vibration accident scenario is assigned to Consequence Category B).

Accordingly, some fraction of the available MAR is assumed to be released, but not 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 could also be released. The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect the fact that a nonpressurized 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 occurring is based on the sum of the accident frequencies for the accident scenarios assigned to this group.

5.3.4.5 Loss of the 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 that specifically involve pressurized release from the reactor coolant boundary containment. There are two accidents of this type (i.e., Accidents 8(a) and 8(b)), which consist of a 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 noncontinuous force, exists to drive material out of the containment. As soon as the pressure inside and outside the contained elements equalizes (which could happen quickly), the motive force dissipates. The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident need to reflect the fact that the containment is pressurized, and that normal road travel vibration and shock may have loosened radioactive material in the reactor coolant boundary. Accordingly, even though Table 5.5 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 coolant boundary such as the core structure. The two accident scenarios discussed define BRA 8.

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

5.3.4.6 Loss of the Nonpressurized Nonreactor Containment Boundary

This section describes events involving loss of package containment events for nonpressurized release accident scenarios from outside the reactor containment boundary. There are four events 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 5.5, these accidents have been assigned to Consequence Category D and only involve release of contamination from the package that could reside outside the reactor containment. 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 are developed.

5.3.4.7 Unplanned Exposure to Radiation

This section describes unplanned increases in the time of exposure to radiation. This set of events consist of Events 11(a), 11(b), and 11(c), in which technical or logistic 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 CONEX box-like structure and transport shielding; (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 events only result in increased routine (though unanticipated) exposure, the management of the risk from these scenarios can be considered covered by normal radiation safety practices. Although, these scenarios should be provided as input to development of those controls, they do not define a bounding representative accident for which detailed likelihood and consequences are developed.

5.3.4.8 Criticality Accidents

This section describes the development of bounding representative accidents for criticality events that happen during transport. There are three types of accidents of this type, Accidents 12(a), 12(b), and 12(c). Accident 12(a) consists of the addition of a moderator and possible change in core geometry from immersion of the TNPP into a body of water. Accident 12(b) consists of the addition of a moderator and possible change in core geometry from inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash that results in fire and TNPP damage. Accident 12(c) consists of a control rod withdrawal (or another event that inserts reactivity) caused by energetic collision.

BRA 9A is a flooded criticality event defined as the addition of a moderator and possible change in core geometry caused by a drop into a body of water (i.e., Accident 12(a)). For the demonstration TNPP design, the change in core geometry is not required to cause criticality if the core is inundated, but an alteration in core geometry is possible from a drop event and could contribute to criticality. 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 to immerse the TNPP as discussed later in Section 6.0. The frequency of this accident can be determined by estimating the likelihood a road accident over those limited sections of highway this event can occur or by looking at the data for accidents that result in a submerged vehicle.

BRA 9B is a flooded criticality event defined as the addition of a moderator and possible change in core geometry caused by inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash that results in fire and TNPP damage (i.e., Accident 12(b)). Again, for the demonstration TNPP design, the change in core geometry is not required to cause criticality if the core is inundated, but an alteration in core geometry is possible from the collision that could contribute to criticality. This accident requires a highly unlikely set of circumstances because the crash would need to result in fire, the TNPP would have to be damaged in a way that water (or other material) could enter the core, and fire suppression water (or other material) would have to be directed at the TNPP Package in a way that it enters and inundates the core, which is inside the CONEX box-like structure of the TNPP Package and not necessarily near the fire.

BRA 10 is defined as a control rod withdrawal event or another event that inserts reactivity due to damage to the reactivity control system caused by a collision that damages the TNPP Package and associated components (i.e., Accident 12(c)). The likelihood of this accident occurring is based on the accident frequency and the conditional probability that control rod withdrawal occurs from the accident.

5.3.4.9 Summary of the Bounding Representative Accidents

Table 5.6 provides a summary definition of the bounding representative accidents discussed in Sections 5.3.4.1 through 5.3.4.8. Accident frequency development for bounding representative accidents is discussed in Section 6.0 and summarized for each of the bounding representative accidents in Table 6.16. Radiation dose consequence for these accidents is presented in Section 7.0 and summarized for the bounding representative accidents in Table 7.6.

Table 5.6. Definitions of the Bounding Representative Accidents

ID	Descriptions
BRA 1	Fire-only event that originates inside of the CONEX box-like structure of the TNPP Package.
BRA 2	Diesel fuel fire-only event that originates outside the Reactor Module and propagates into the CONEX box-like structure and ignites combustible material which damages the reactor containment boundary and impacts containment capacity.
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 a 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 of shielding. Medium impact that involves a severe collision with a light vehicle (e.g., one that results in fatality or injury). Less than a hard-impact highway accident that results in no release of radiological material but some degradation of transport shielding. Light impact such as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or impact with a light vehicle that is not severe (e.g., results in property damage only).
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 leads to fatality or injury) that results in fire.
BRA 6	Collision with a tanker carrying flammable material that leads to fire.
BRA 7	Loss of the nonpressurized reactor containment boundary not caused by a road accident but by human error and failures of containment features.
BRA 8	Loss of the pressurized reactor containment boundary not caused by a road accident but by human error and failures of containment features.
BRA 9A BRA 9B	Addition of moderator and possible change in core geometry caused by a drop into body of water that results in criticality. Addition of moderator and possible change in core geometry caused by a crash that results in RPV damage, fire, and inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality.
BRA 10	Control rod withdrawal (or another reactivity insertion event) caused by impact from a road accident that results in criticality[(a)].

BRA = bounding representative accident; CONEX = container express; RPV = Reactor Pressure Vessel; TNPP = Transportable Nuclear Power Plant.
(a) This study assumes the design goal of precluding a reactivity insertion event in a TNPP transportation package accident will be met.

6.0 Development of Estimates of the Likelihood of Occurrence of the TNPP Transportation Accident Scenarios

This section describes the development of estimates of the likelihood of TNPP transportation accidents occurring and provides the bases for those estimates. The assumed route from INL to WSMR passes through parts of Idaho, Utah, Wyoming, Colorado, and New Mexico, as shown in Figure 6.1. The route traverses a diverse geography and population from very rural Wyoming to metropolitan Denver. For the metropolitan Denver area, a bypass route to the east (on Colorado highway E-470) was included in the analysis. Section 6.1 discusses collection, analysis, and characterization of route-specific hazards. Section 6.2 discusses collection and analysis of large truck accident data. Section 6.3 discusses development of estimates of the likelihood of the accidents occurring during TNPP transportation. Section 6.4 provides a list of the primary assumptions that were made as part the accident likelihood development process. Section 6.5 provides the estimation of the frequencies for each bounding representative accident.

6.1 Characterization of Route-Specific Spatially Derived Hazards

Many environments with different risks will be encountered as the TNPP 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 the following:

- Soil types. The relative hardness was quantified for the assumed routes and is discussed in Section 6.1.1.
- Bridges. The presence of underpasses and overpasses was enumerated and is discussed in Section 6.1.2.
- Rivers and waterbodies. Bodies of water sufficient to submerge the reactor vessel near the assumed route were investigated and are discussed in Section 6.1.3.
- 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 were investigated and are discussed in Section 6.1.4.
- Population density. Population density data, though not used in the TNPP transportation PRA, may be of concern for the state of Colorado as it pertains to DOT regulations (i.e., the route should avoid populated areas), as discussed in Section 6.1.5. Population density is used in an EIS.

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

Figure 6.1. Assumed Route from INL to WSMR, Bypassing Metropolitan Denver Metro using Colorado E-470

6.1.1 Soil Types

The following definitions are used for the identification of Map Unit subcomponents that behave like "hard rock," "soft rock," "rocky soil," or "other soils, slay, silt." Previous work used the 1:250,000 scale 1996 State Soil Geographic (STATSGO) (USDA-SCS 1993) data, but those data have been superseded by the more well-resolved Soil Survey Geographic data (USDA-NRCS 2005). The Soil Survey Geographic data use the 1:24,000 scale and a different data model than the earlier STATSGO data.

A new soils data model was used to classify the data into the four categories following the work of Mills et al. (2006), which was a source document for the transportation risk assessment studies performed by the NRC, like the study presented in NUREG-2125 (NRC 2014) and described in Section 6.3.1. Multiple tables were needed, including "chorizons" (soil horizons), which include information about the existence and fraction of rocky soil) and "corestrictions," which includes 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 whether the bedrock was removable with a backhoe or only by blasting. Because those data are not included in the current data model, the cementation information from the corestrictions table was used. That table classifies hardness (from hardest to least) as follows: "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 based on the highest hardness within a unit:

- 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 noncemented.
- 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 percent, the average diameter of these rocks was ≥3 in., and the sum of the thicknesses of these layers was ≥2 ft.
- If the Map Unit subcomponent was not hard rock, soft rock, or rocky soil, then it was defined to be "other soils, clay, or silt" or water.

GIS methods of analysis used ArcMap™ software by Esri (an American multinational geographic information system software company headquartered in Redlands, California) 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. Table 6.1 and Table 6.2 summarize the wayside surface types for each state for the two potential routes.

Table 6.1. 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[(a)]	0.111	0.068	0.067	0.753
(a) Distance-weighted average values.				

Table 6.2. 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[(a)]	0.111	0.069	0.066	0.754
(a) Distance-weighted average values.				

The assumed highway transport route for the Project Pele prototype TNPP Package was from INL to WSMR in New Mexico. The highway transport route for assessment of soil types in this section and bridges assessed in Section 6.1.2 was estimated using the Web-Based Transportation Routing Analysis Geographic Information System (WebTRAGIS) computer code (Peterson 2018). Figure 6.2 through Figure 6.6 show 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 6.5 shows both the route through the city of Denver and the E-470 bypass around the center of Denver, Colorado.

The destination of WSMR was assumed for the purposes of analysis in the transportation PRA as well as for demonstration of the process and can be altered later, 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.

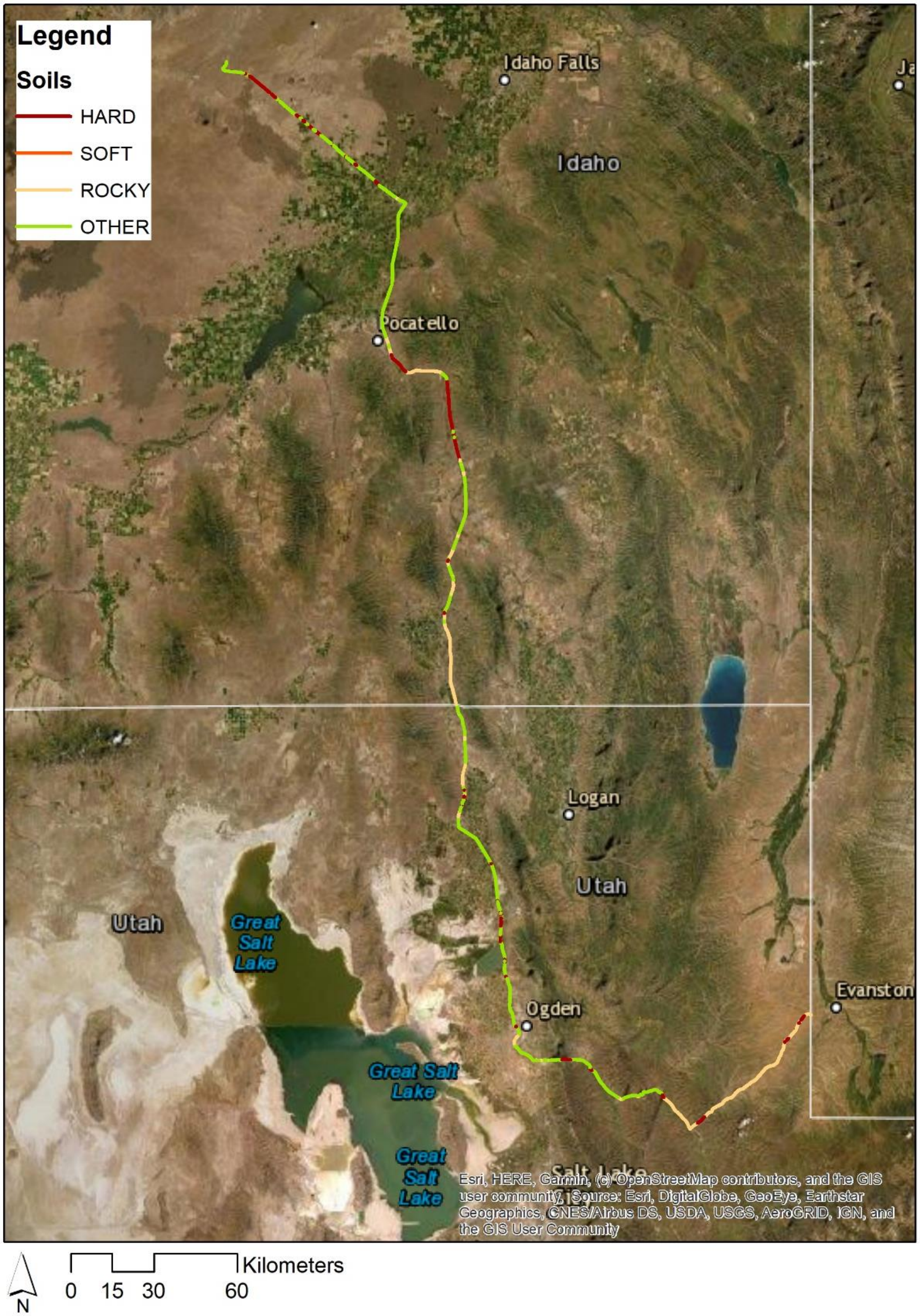


Figure 6.2. Potential Route from INL to WSMR Showing the Wayside Soil Types: Idaho and Utah

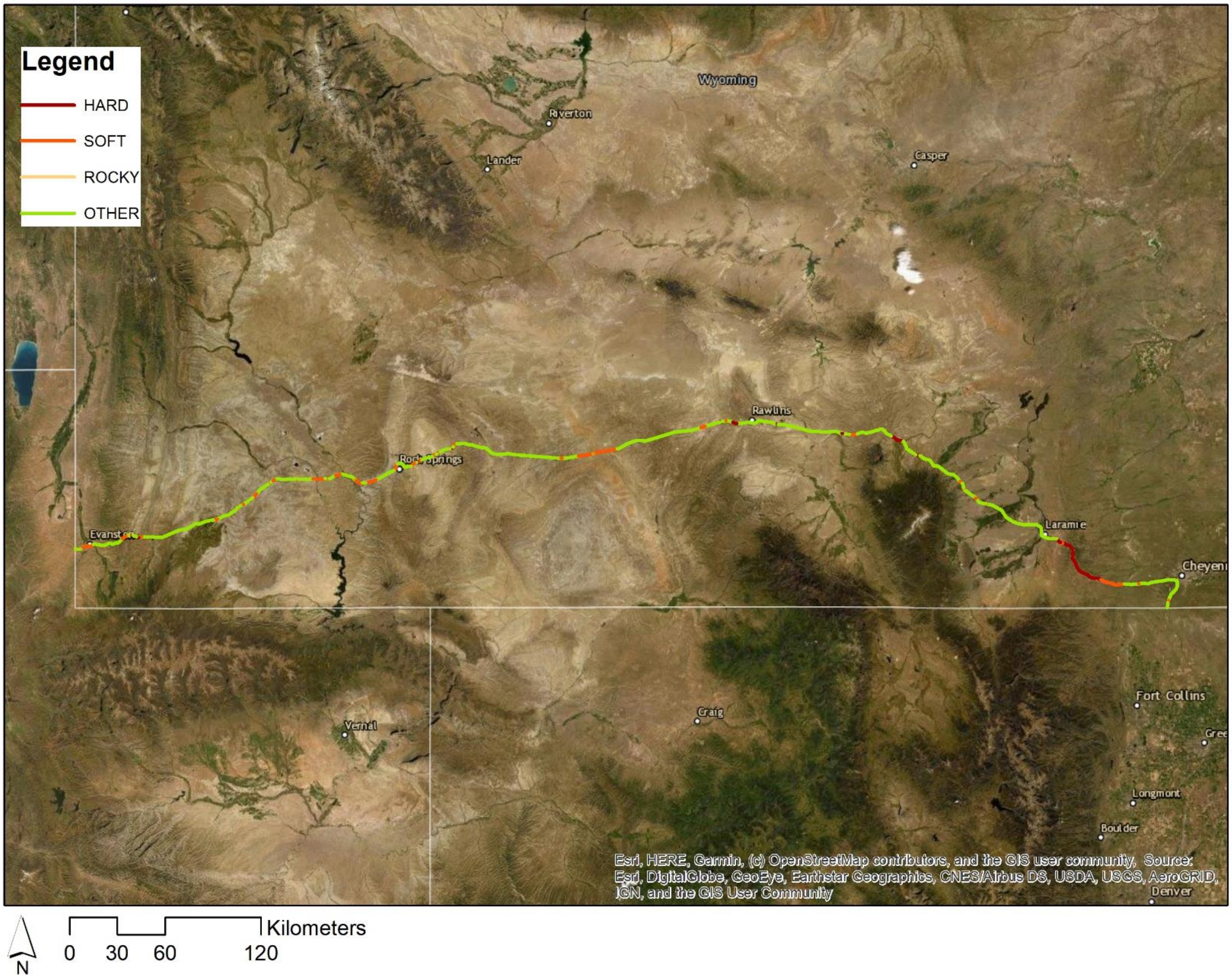


Figure 6.3. Potential Route from INL to WSMR Showing the Wayside Soil Types: Wyoming

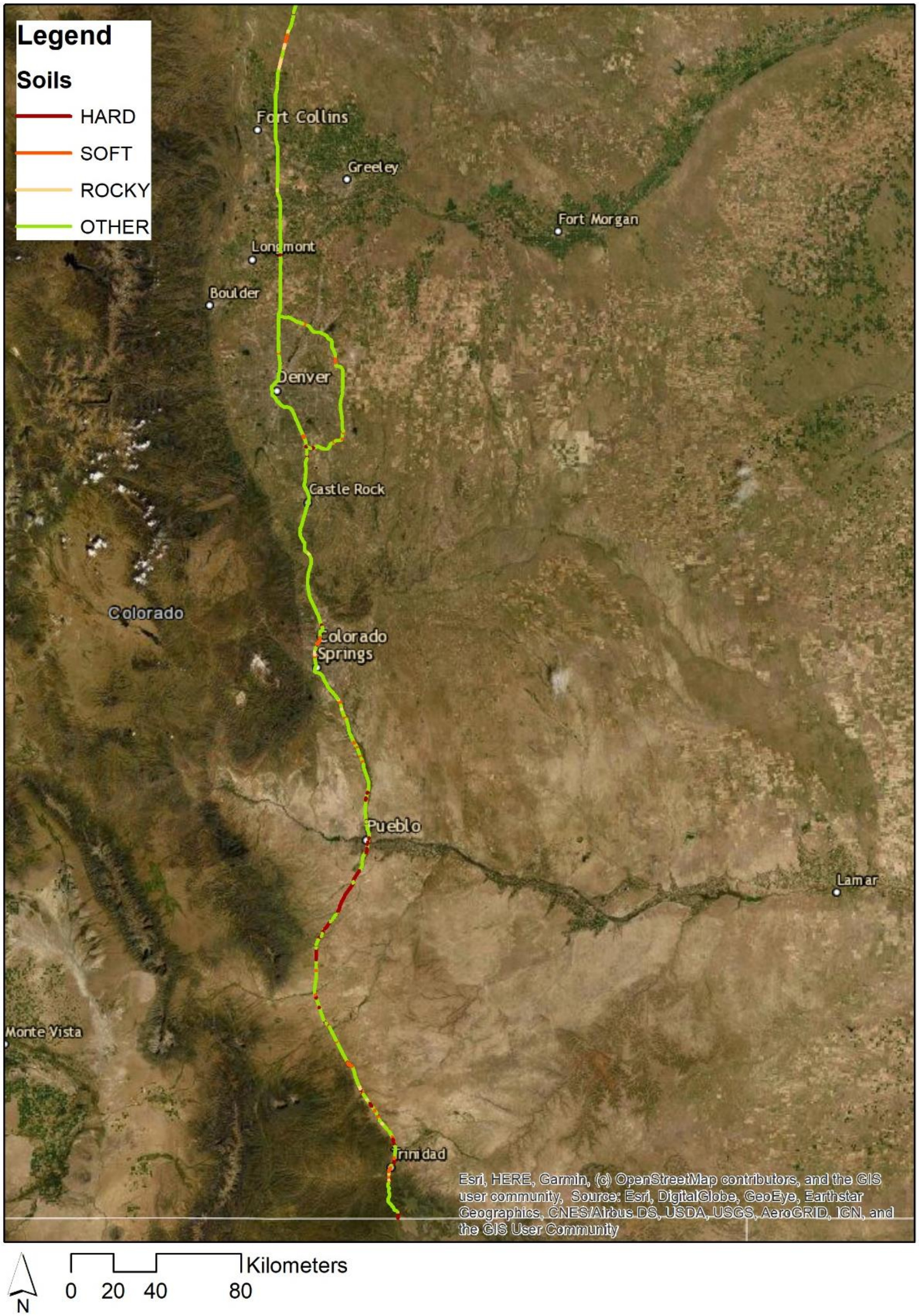


Figure 6.4. Potential Route from INL to WSMR Showing the Wayside Soil Types: Colorado

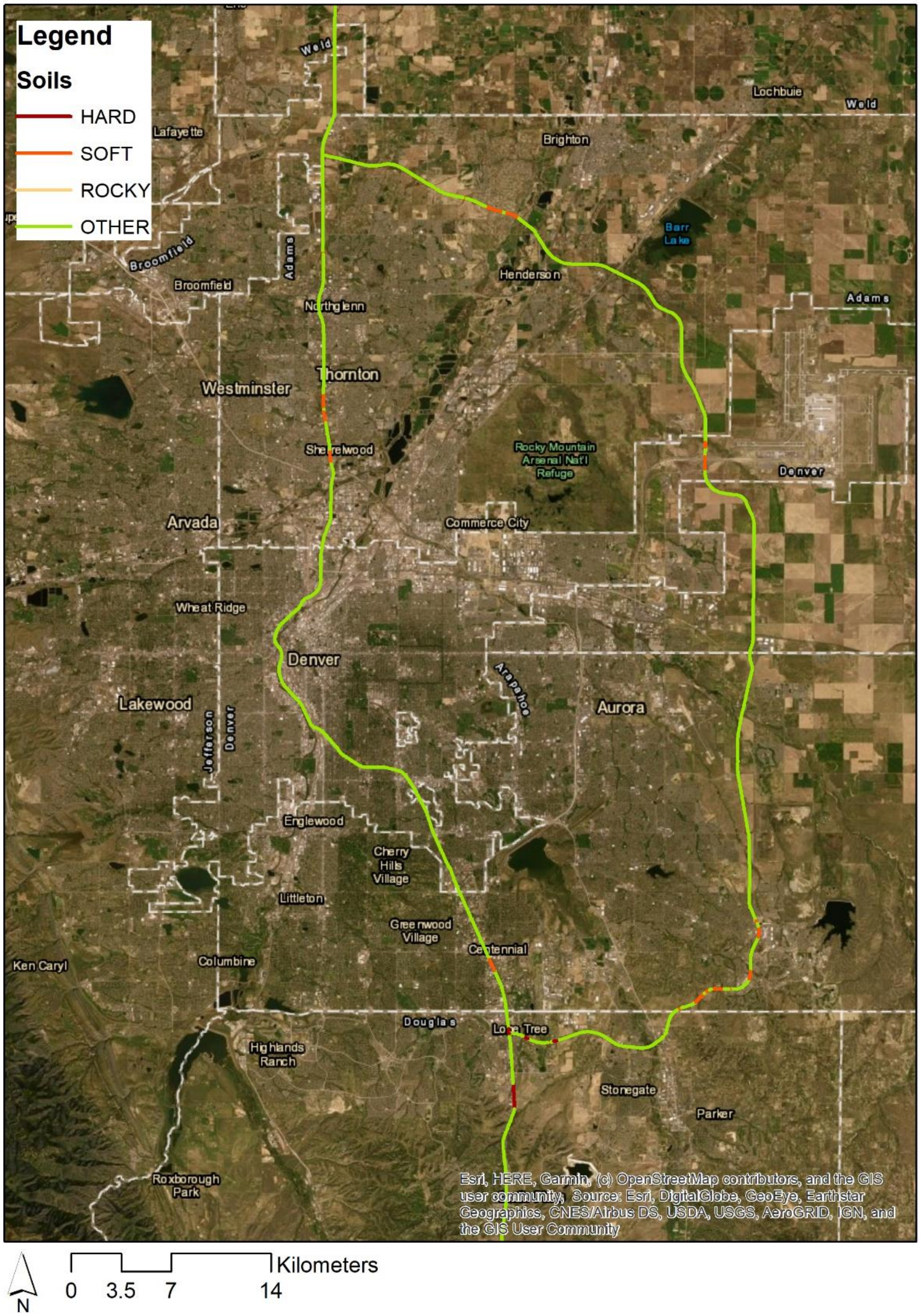


Figure 6.5. Potential Routes from INL to WSMR Showing the Wayside Soil Types: Denver[31]

[31] The eastern route uses Colorado E-470 to bypass Metropolitan Denver and the intersection of I-25 and I-70.

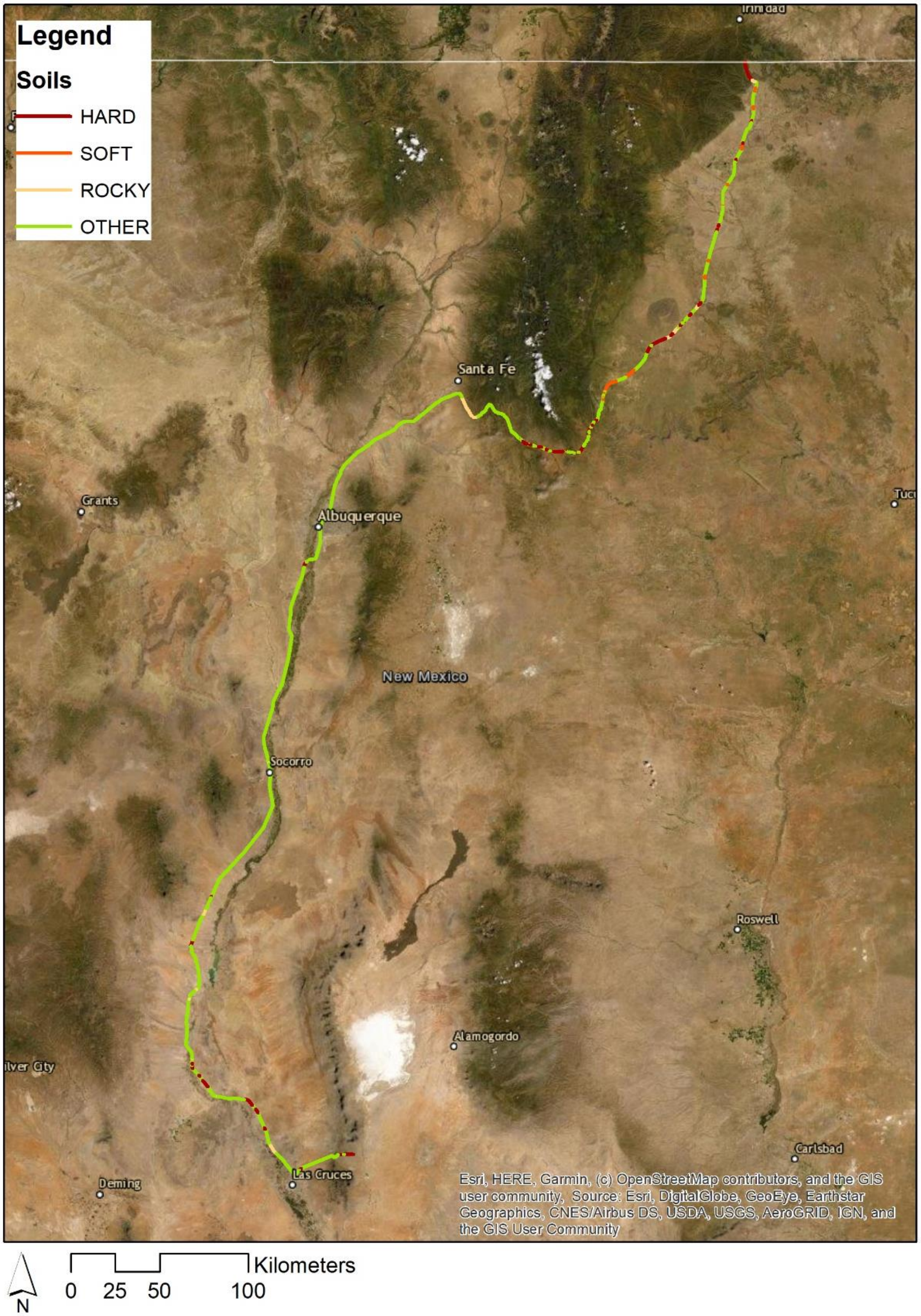


Figure 6.6. Potential Route from INL to WSMR Showing the Soil Types: New Mexico

6.1.2 Bridges

The National Bridge Database (NBD)[32] was used to assess the bridges along the transportation route. There are two types of bridges: overpasses and underpasses. Overpasses can be over roads, rivers, railroads, or other features, while underpasses are typically under roads or railroads. Overpasses present “fall” hazards, while 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 locations and variations in coding from state-to state. Table 6.3 presents state-by-state information about the number of underpasses and overpasses, and the minimum drop from the overpasses (maximum drop is not in that database). This information was not used explicitly in the PRA for determination of accident likelihood, but it provides some perspective of this hazard. The likelihood of the occurrence of accidents that involve drops into a body of water or to a lower elevation are explained in Sections 6.1.3 and Section 6.1.4, respectively.

Table 6.3. National Bridge Inventory of Overpasses and Underpasses Along the 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

6.1.3 Rivers and Bodies of Water

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, several mountainous areas must be traversed. In these areas, the interstate crosses a few major rivers and streams, and there are several locations where rivers and streams run parallel to the interstates. Note in Figure 6.7 that the river is adjacent to the interstate highway.

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.

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

Figure 6.7. I-84 along the Weber River (from Google Maps)

The evaluation described in Sections 6.1.1 and 6.1.2 used an assumed route at a level of resolution that did not consider the difference between northbound and southbound lanes, including the 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 routes traveled 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 were extracted from Open Street Maps (OSM) (2022) using QGIS 3.2,[33] a Java-based OSM data query tool (Stadtherr et al. 2022). These data proved to be more spatially accurate than the data used to generate the hazard results described in Section 6.1.2 about 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 were 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.

Stream data were 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, three-dimensional 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 gauges scattered throughout the network. This dataset is considered to have the available “best flow and velocity estimates” (USGS 2022a, page 62).

[33] formerly Quantum GIS

The presence of water in streams at various flow rates was investigated using images 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 6.8. Most streams with flow rates <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 of 3 or >3, 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.[34] 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.

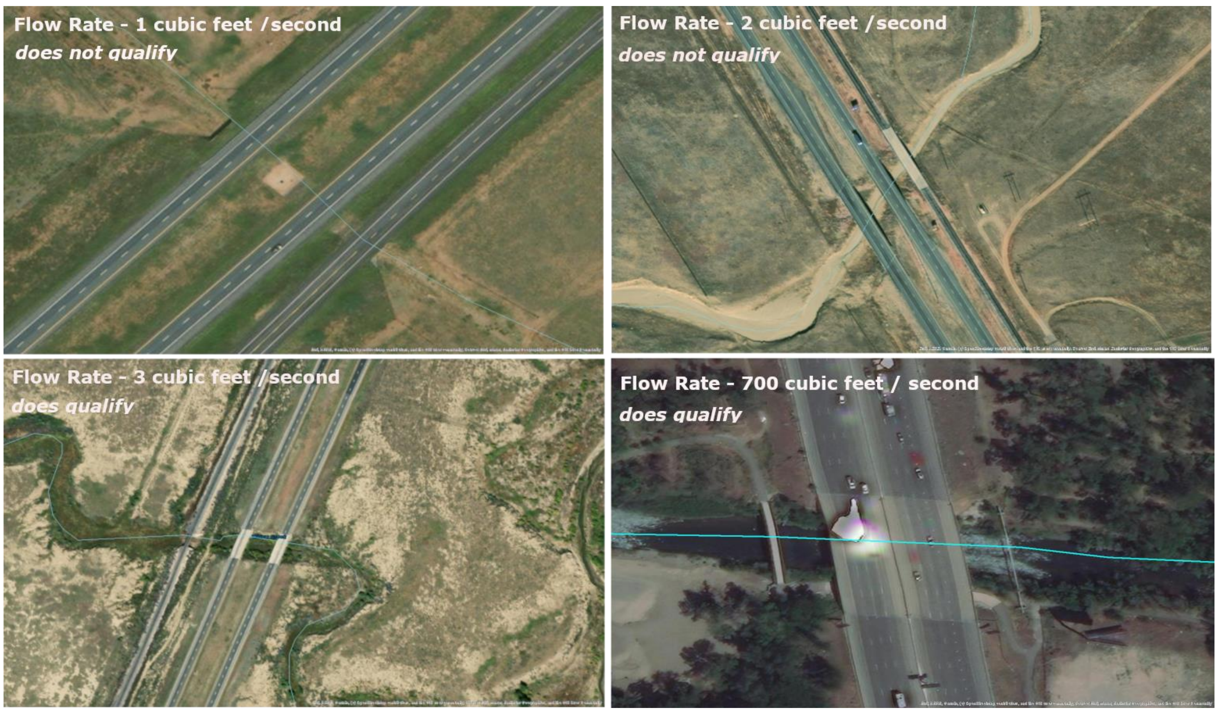


Figure 6.8. Stream Images at Various Flow Rates Used to Determine the Minimum Threshold for Qualifying Streams in the Analysis

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, if it left the road. Slope data were 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 an 8.3 × 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 were used to derive a slope dataset for 8.3 m × 8.3 m sections away from the road by querying whether the slope was 25 percent or greater between sections. Slope data were then

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

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 make sure that the sloped area was, in fact, between the roadway and the downhill stream as illustrated in Figure 6.9. Route segments closer than 8.3 m to the adjacent stream were added back to the qualifying segments to make up for the lack of slope analysis resolution and slope data immediately next to the road.

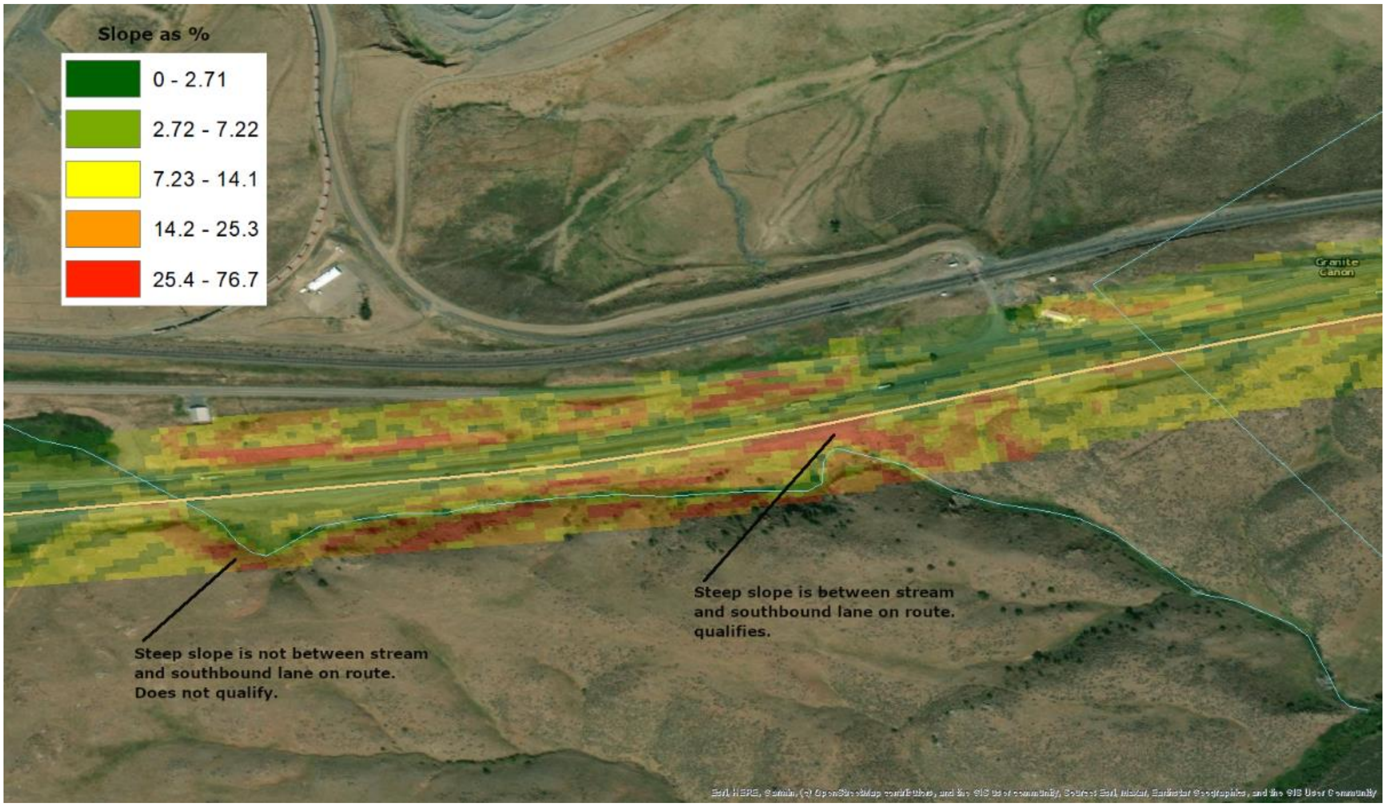


Figure 6.9. Slopes Adjacent to Roadside Checked Manually to Make Sure They Are Downhill to the Stream

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 (i.e., adjacency refers to portions of the route where the assumed route runs along a river or stream). 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. 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 Figure 6.10 and Figure 6.11. The image identification numbers in the image name correspond to the route segment identification number.

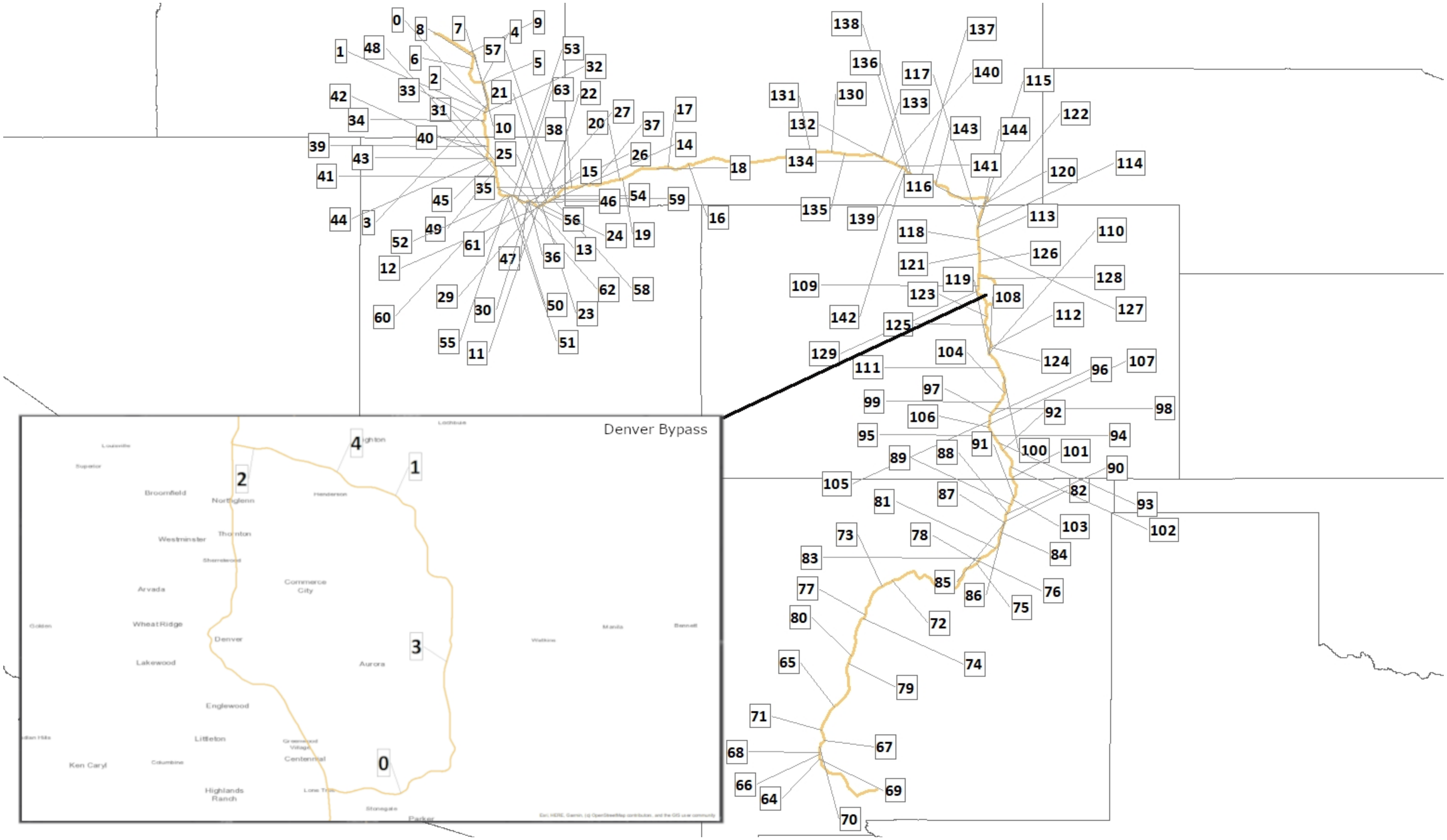


Figure 6.10. Road Segments Crossing Streams that Have a Flow Rate Greater than 3 ft^3/sec[35]

[35] The identification number (tag) corresponds to the Google Street View Image.

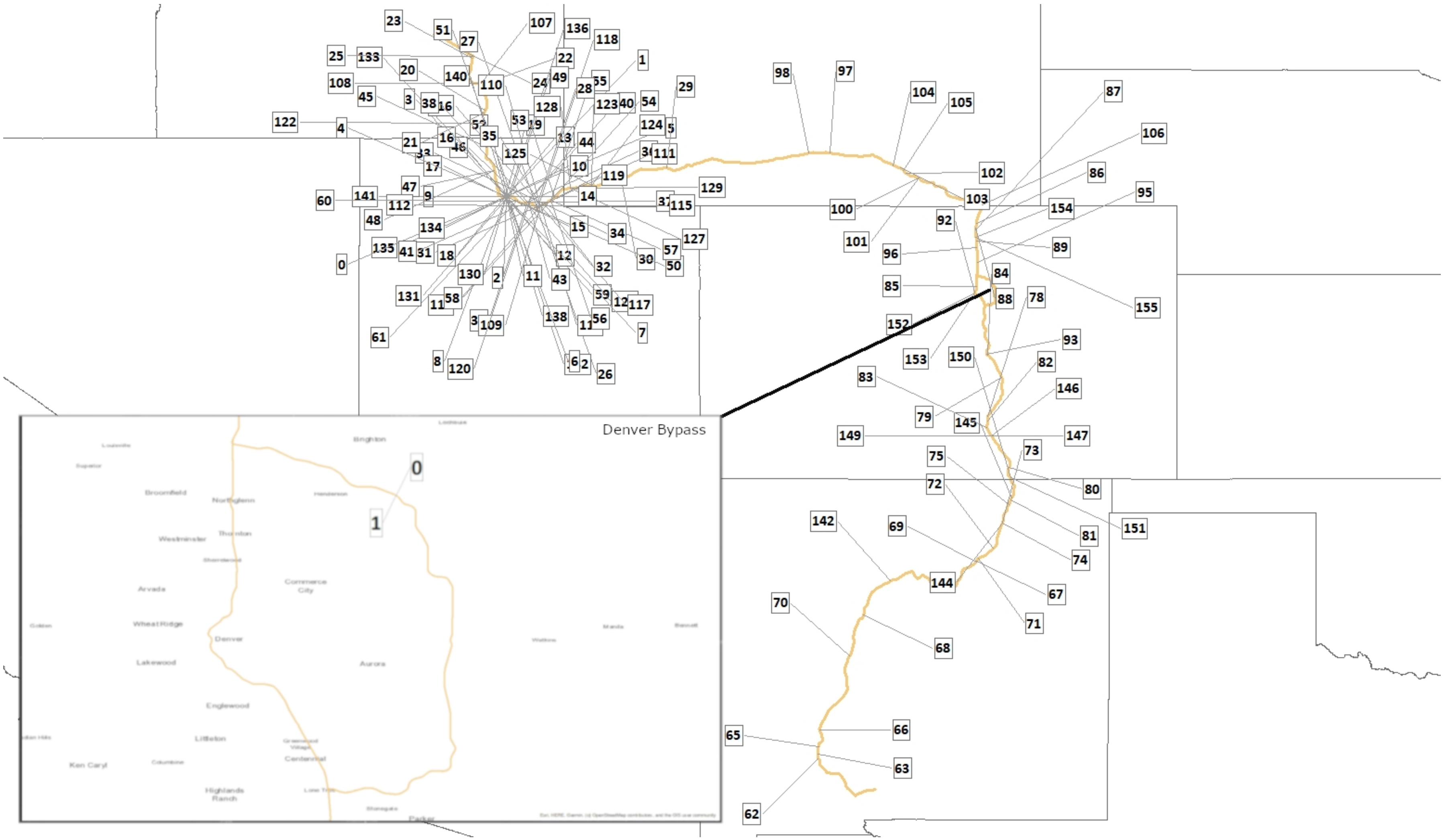


Figure 6.11. Within 50 m of a Stream that has a Flow Rate Greater than 3 ft^3/sec[29]

This assessment of river and stream crossings and adjacency resulted in the final route segment counts and length totals listed in Table 6.4.

Table 6.4. River and Stream Crossings 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 the best available location data of the stream/waterbody network derived from the best available three-dimensional elevation, LiDAR sourced (Light Detection and Ranging technology), 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 about the exact stream location without physical confirmation. Second, 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. Finally, 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 to be flat in that area.

However, a physical road survey was performed to visually verify that the assumptions used in the assessment described above to support the determination of an accident frequency associated with a flooded criticality event caused by a drop into a body water appear to be reasonable. The conclusions of the verification work described in a report published by Taylor et al. (2023) and confirm that the results of the evaluation using GIS are reasonable or conservative for use in estimating the likelihood of applicable accidents as described in Section 6.3.1.2.

6.1.4 Drops to a Lower Elevation

Another specific hazard of the route is the 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 6.1.3, the evaluations described in Sections 6.1.1 and 6.1.2 of this 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, as for the investigation of bodies of water, the route data were extracted from Open Street Maps (2022) using QGIS 3.2 Java-based OSM data query tool (Stadtherr et al. 2022). As described in the Section 6.1.3 investigation of bodies of water sufficient to submerge the reactor vessel along 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 percentages. If the slope of the adjacent land was greater a than 33 percent 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 were derived from a DEM available for the entire United States from the USGS. The DEM had 8.3 × 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 were used to derive a slope dataset for 8.3 m × 8.3 m sections away from the road by querying whether the slope was 33 percent or greater between sections. The slope data were screened to include only areas greater than a 33 percent grade. Then, the route segments within 25 m of sloped areas greater than 33 percent grade were examined to determine the direction of the slope to make sure that it was down and away from the route and not down and toward the route, as illustrated in Figure 6.12.

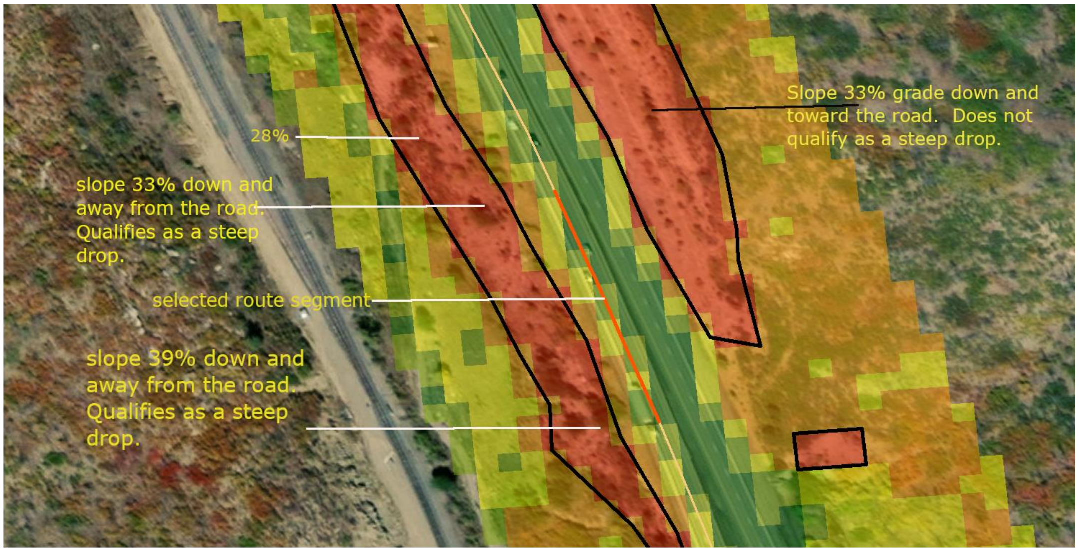


Figure 6.12. Route with a Qualifying and Non-Qualifying Slope of a 33 Percent or More Grade to the Sides of the Route Based on Its Direction of Slope

Finally, street view images were generated to visually confirm that the drop-off locations exist, 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 application programming interface. The road segment identification number are shown in Figure 6.13 and Figure 6.14. These are the same locations of the Google Street View images shown in Figure 6.15 through Figure 6.18.

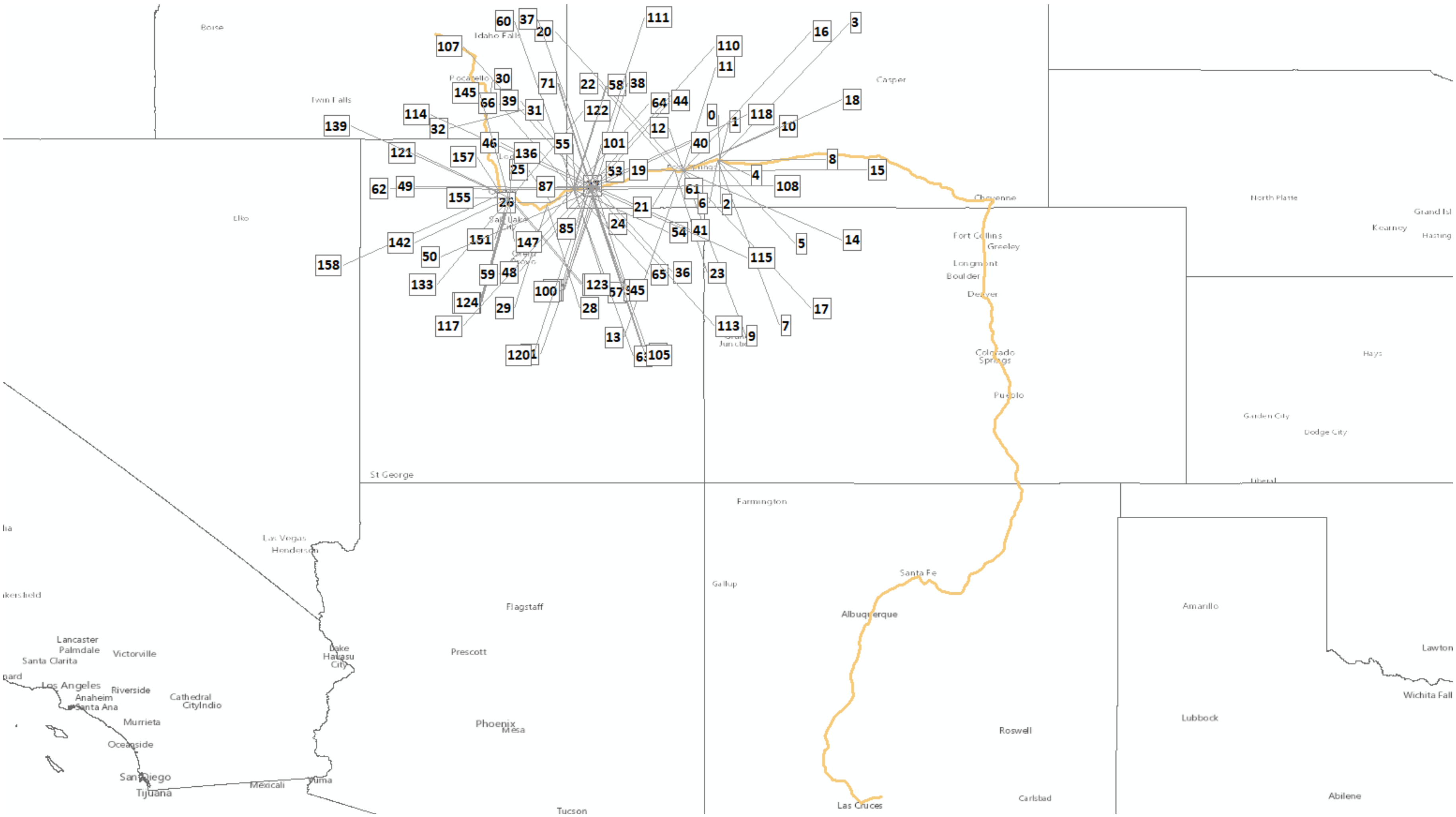


Figure 6.13. Road Segments 1–159 Where a Drop-off Was Identified Within 25 m of the Route with an Immediate Slope of at a Least 33 percent Grade – North

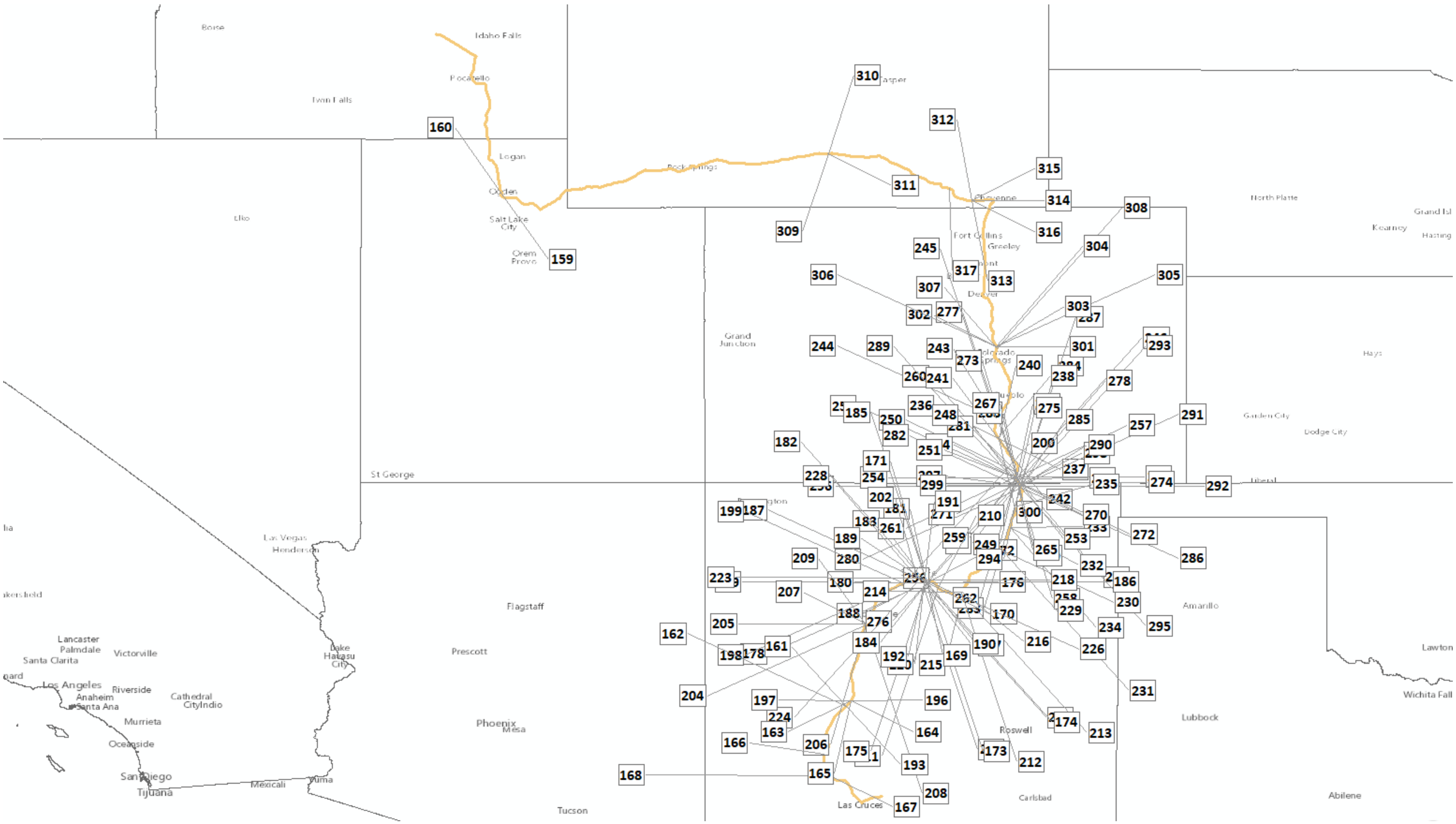


Figure 6.14. Road Segments 160–318 Where a Drop-off was Identified Within 25 m of the Route with an Immediate Slope of at Least 33 percent Grade – South

Figure 6.15. Bridge Drop-Off from Route Determined by GIS Analysis

Figure 6.16. Large Ditch Where the Slope Appears to Slope Down and Away as Well as Down and Toward the Route

Figure 6.17. Relatively Gentle Slopes Along the Route with a Deep Hole Emerging Next to the Road

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 6.15. In other cases, the general slope appears to be up and away from the road but contains a steep drop (e.g., into a ditch) near the road before the slope goes up, as shown in Figure 6.16. 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 6.17. Finally, 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 6.18A; where the road is built up to a higher elevation than a rail line valley below, as shown in Figure 6.18B and Figure 6.18C; and where water from a drainage canal eroded the earth creating a steep drop along the side of the road, as shown in Figure 6.18D.

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 of the total 1,289-mile route.

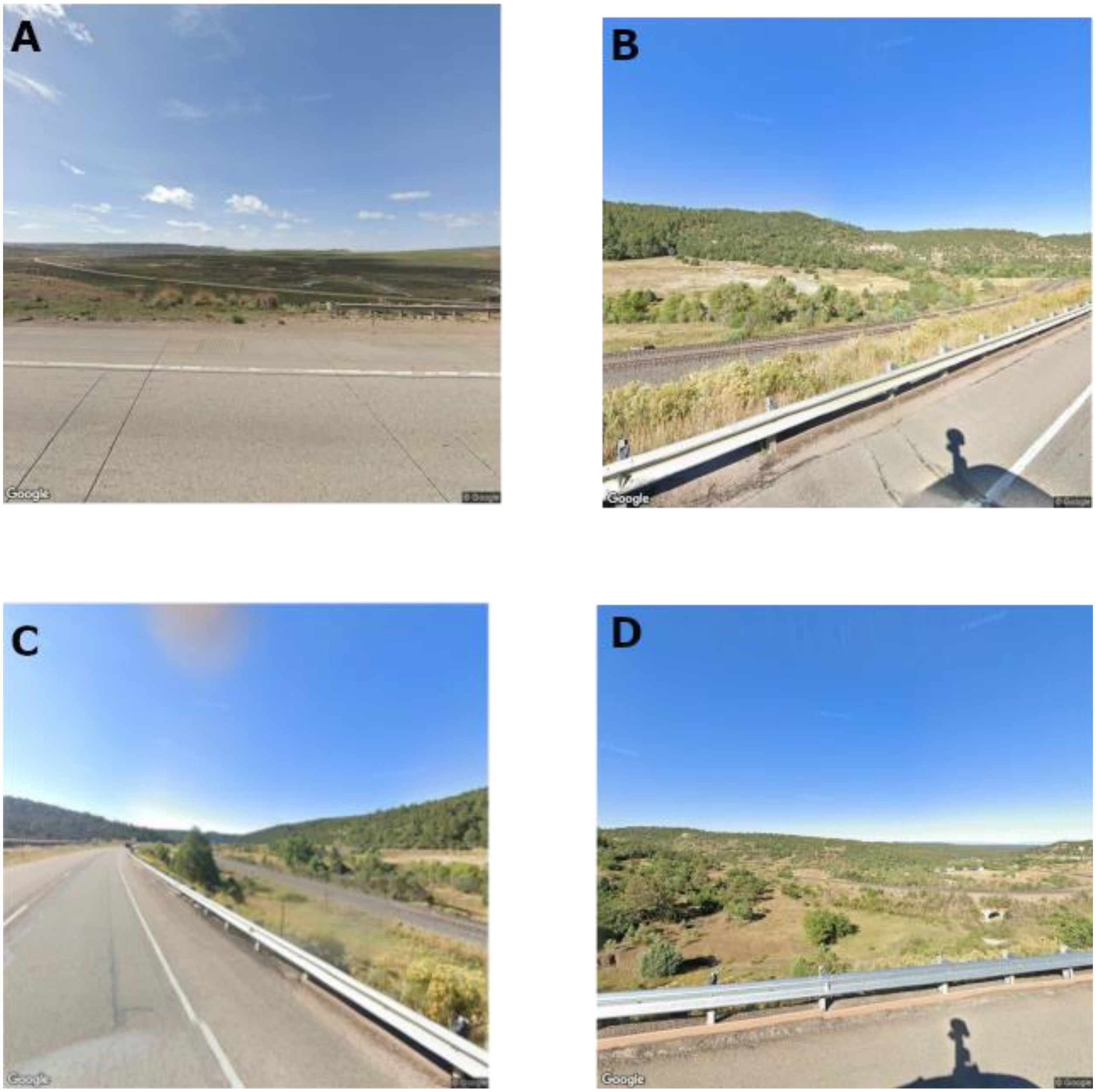


Figure 6.18. Views of Various Steep-Drop Locations Along the Route

The primary limitations of this GIS analysis concern the 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 m × 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 the second 8.3 m section from the road. Second, any barrier, such as a wall or ground feature (e.g., rock outcropping) less than 8.3 m wide, that protects the truck from getting to the sloped ground 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 percent grade criterion defines “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 transportation package. More data would be needed to accurately estimate criteria used to define a “steep slope.”

However, a road survey was performed to verify that the assumptions used in the assessment described above to support the determination of an accident frequency associated with a drop-to-lower-elevation event appear to be reasonable. The conclusion of the verification work is described in a report published by Taylor et al. (2023) and confirms that the results of the evaluation using GIS are reasonable or conservative for use in estimating the likelihood of applicable accidents as described in Section 6.3.1.3.

6.1.5 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 making sure that the motor vehicle is operated on routes that minimize radiological risk. Determination of radiological risk is required to consider available information about accident rates, transit time, population density and activities, and the time of day and the day of week during which transportation occurs. 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.

Figure 6.19 through Figure 6.21 present the spatial distribution of the population density along the route. Figure 6.19 presents the whole route and Figure 6.20 and Figure 6.21 present the populated parts of the route.

This route transits the Metropolitan Denver 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 6.4), the total population within 800 m of the route would be reduced to 1,650,000, a reduction of about 1 percent. 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 was assumed for the PRA that the route used the E-470 beltway around Denver to reduce proximity to the population. It is recommended that this issue be discussed with the state government of Colorado.

As stated in Section 6.1.1, population density data are not used in the TNPP transportation PRA but may be of concern for the state of Colorado because population density pertains to DOT regulations cited above (i.e., the route should avoid populated areas).

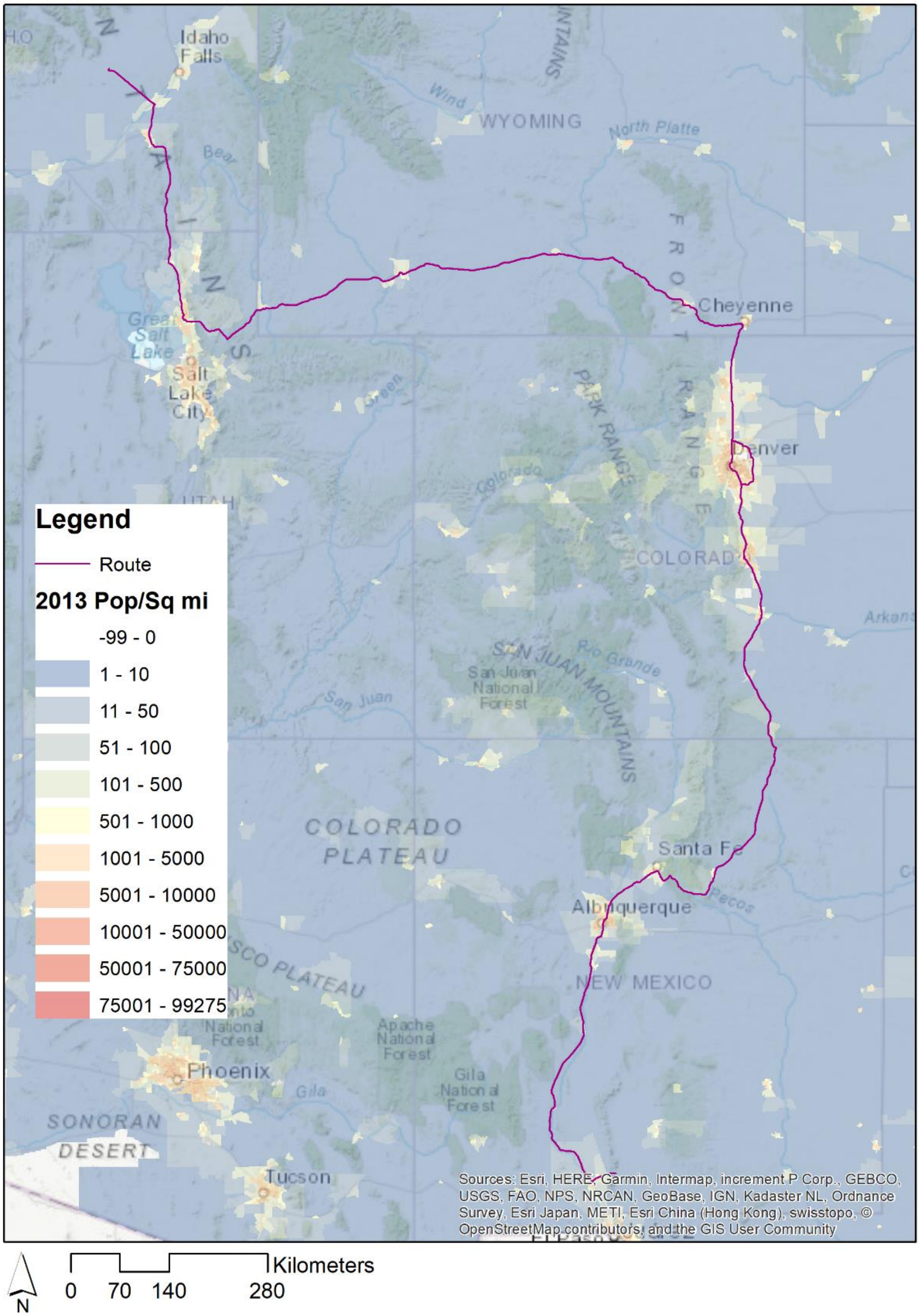


Figure 6.19. Population Density for the Entire Route

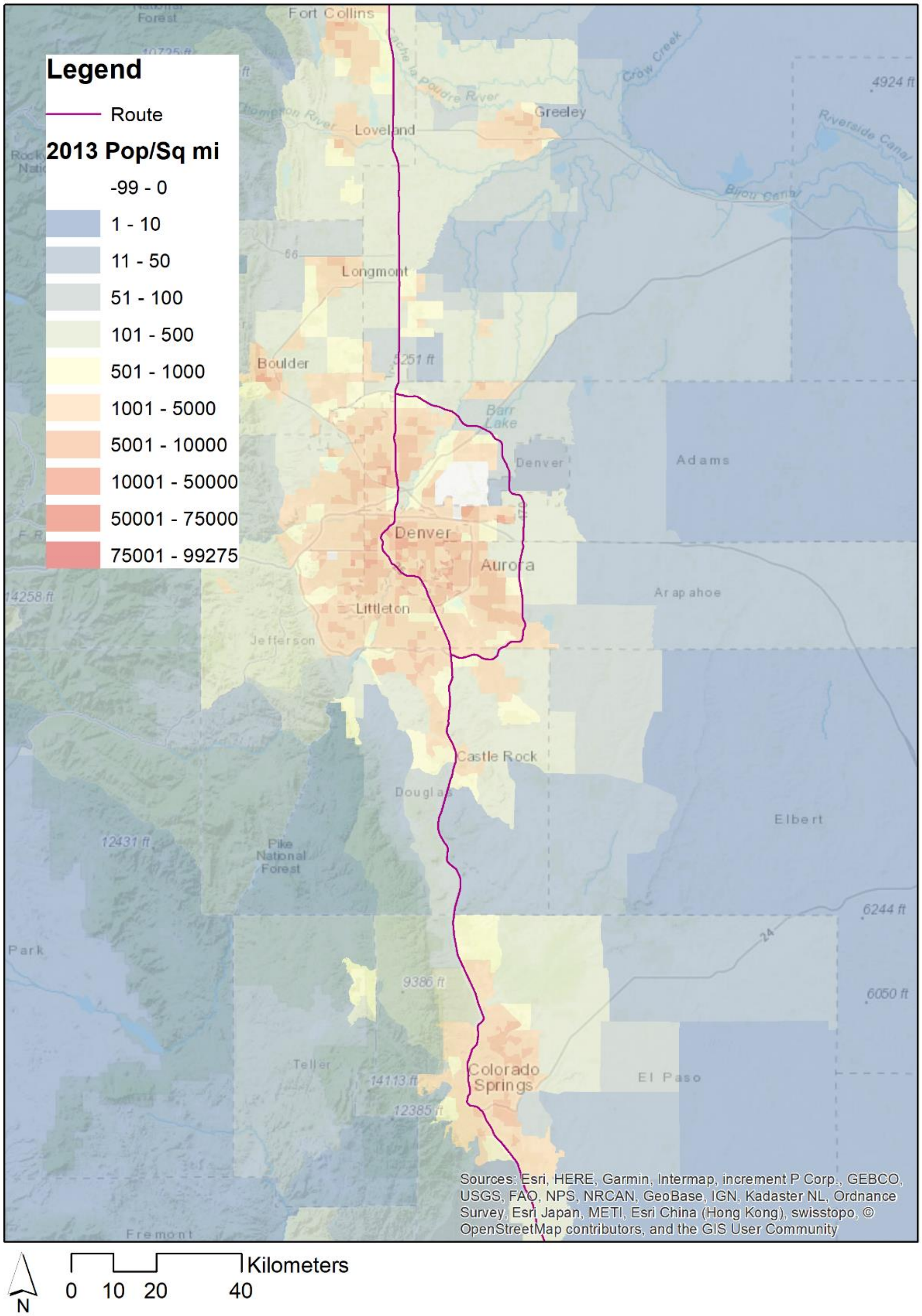


Figure 6.20. Population Density Along the Route for the Colorado Front Range Including a Denver Bypass Route (Colorado E-470) to the East of the Metro Area

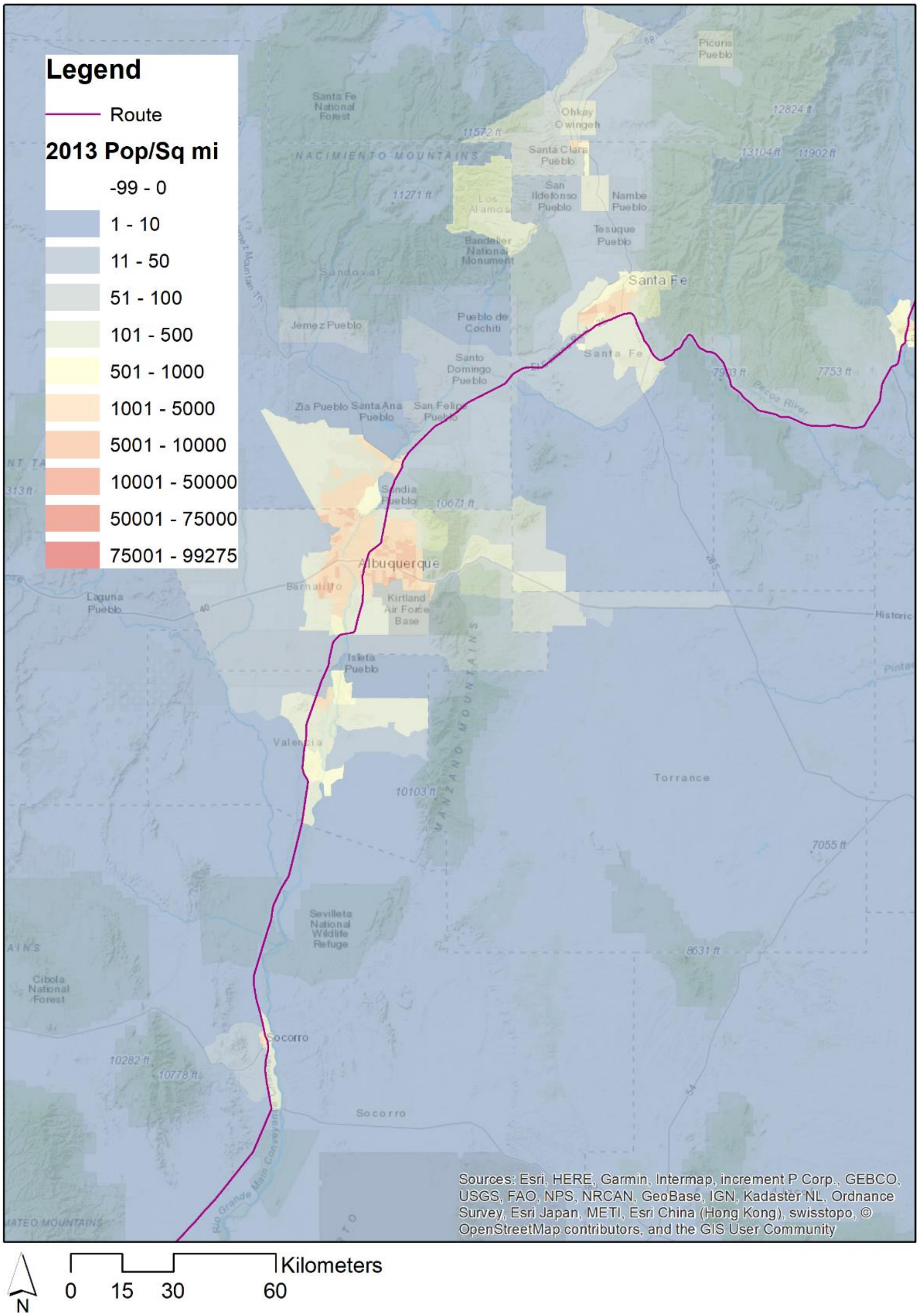


Figure 6.21. Population Density Along the Route in New Mexico, Including the Greater Albuquerque and Santa Fe Regions

6.2 Collection of Transportation Accident Rate Data for Very Large Trucks

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

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 pounds 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 pounds GVW) may be omitted.

Accident rates were determined for very large truck travel from INL in Idaho to WSMR in New Mexico (using the bypass around Denver). 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 3 years 2017–2019 for the five states and nationwide where state-specific data were not available. Key information sources for the number of accidents and vehicles involved annually were the Motor Carrier Management Information System (MCMIS) for large truck "all-accident" (crash[36]) 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 and 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 of the mileage data used to determine accident rates per mile for 2017–2019.

Section 6.2.1 discusses very large truck mileage for each of the five states the route traverses by interstate and all state highway miles. Section 6.2.2 discusses very large accident event data and presents the accident rate for each of the five states along the route as well as the total accident rate for all five states. Section 6.2.3 discusses the development of the accident rates for specific types of very large truck accidents modeled in the TNPP transportation PRA. It also presents the proportions of different accident types as well as the estimated accident rates for those types.

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

6.2.1 Very Large Truck Mileage for the Five States Along the Route

Detailed information about annual miles traveled by all vehicles, including categories of combination trucks and single-unit trucks, are available from the FHWA for each individual year.[37] 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."

Table VM-2 breaks down functional system travel into two major categories of "rural" and "urban" travel. In each of these are seven subcategories, with the one of interest being interstate systems. The other subcategories are other freeways and expressways, another 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 subcategories 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 is included to address all large trucks (>10,000 pounds GVW). As noted earlier, some single-unit trucks may be very large trucks, but their mileage is 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 percent were traveled on interstate highways, which is characteristic of the assumed TNPP Package transport route. Annual and total interstate mileages in each of the five states are listed in Table 6.5 and all state highway mileage in each state is listed in Table 6.6.

Table 6.5. Very Large Truck Interstate Mileage, 2017 to 2019[(a)]

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%

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

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

Table 6.6. Very Large Truck All-State Highways Mileage, 2017 to 2019[(a)]

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%

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

6.2.2 Accident Rates for Very Large Trucks Based on All Highways of States Along the Route

The basis for the accident rates of very large trucks is the state-specific number of "all crash events" that 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)*.[38] The A&I crash statistics module provides users with an ability to view crash data reports from either MCMIS or FARS:

- MCMIS includes crashes involving large trucks and buses (commercial motor vehicles [CMVs]) that are reported by states to the FMCSA through the SAFETYNET computer reporting system. It includes data elements collected about trucks and buses that meet the National Governors Association recommended crash threshold. The FMCSA operates and maintains the MCMIS.
- 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.[39] 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 CMV involved in a crash. The MCMIS contains information about the safety fitness of commercial motor carriers (trucks and bus) and hazardous material shippers subject to the Federal Motor Carrier Safety Regulations and the Hazardous Materials Regulations.

Crash statistics data can be filtered for large trucks using the A&I Crash Query Tool. The crash statistics were accessed during the early months of 2022. There are also published crash reports for CMVs that provide additional information,[40] such as the *FMCSA Pocket Guide to Large Truck and Bus Statistics*.[41] The data query used readily available summary reports from

[38] Available at https://ai.fmcsa.dot.gov/default.aspx

[39] MCMIS data are considered preliminary for 22 months to allow for changes. Therefore, 2019 was the most recent year of final data and the 3 years 2017–2019, which were also without the effect of the COVID-19 pandemic in 2020, were used as the basis for accident rates.

[40] Available at https://ai.fmcsa.dot.gov/CrashStatistics/CrashProfile.aspx

[41] Available at https://www.fmcsa.dot.gov/safety/data-and-statistics/commercial-motor-vehicle-facts

the A&I Crash Query Tool. A snip of a summary report is shown in Figure 6.22 at the end of this section, 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 within this figure provides state-specific data for “Gross Vehicle Weight – Over 26,000 lbs.”

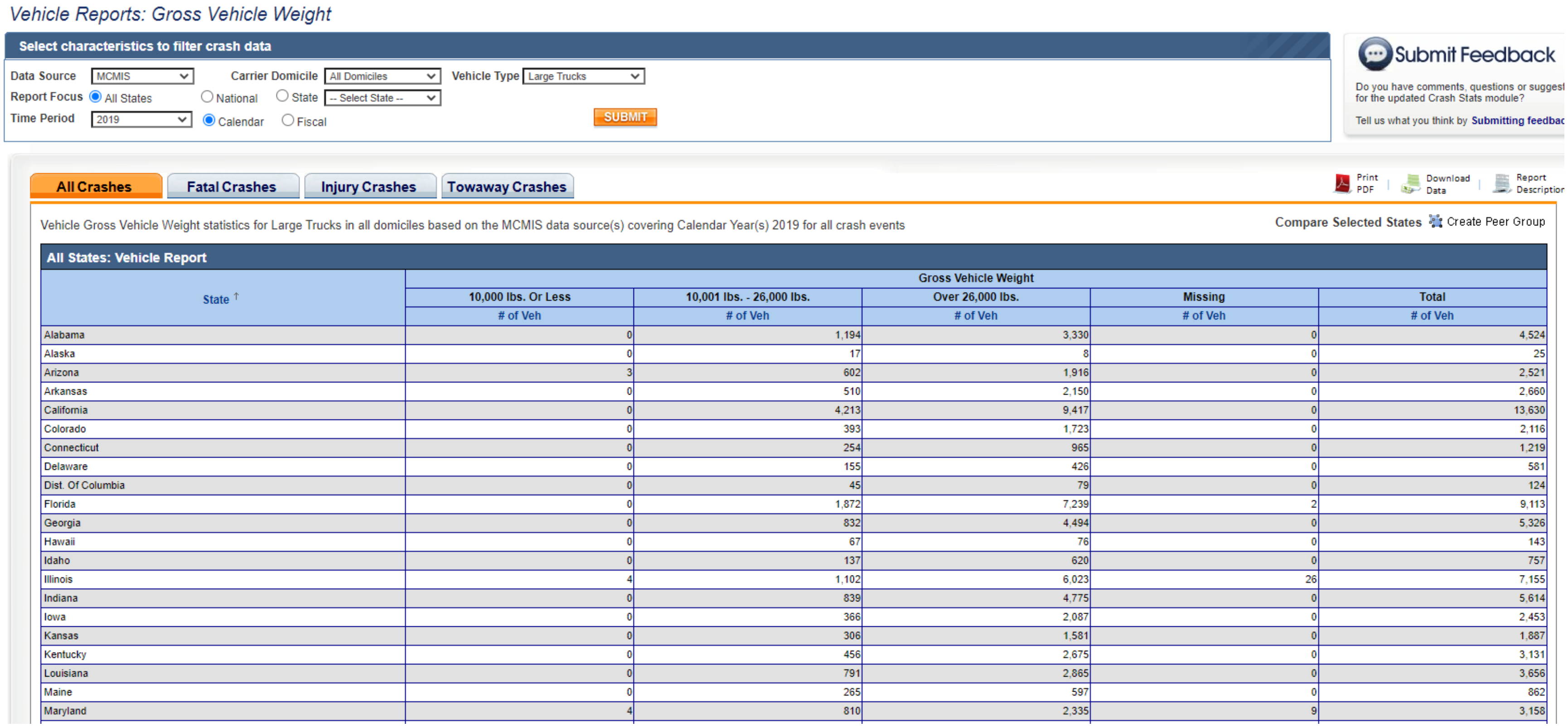
Table 6.7 shows the number of all crashes involving very large trucks on all state highways in the five western states of interest for 2017–2019 (i.e., a total of 16,207 crash events both fatal and non-fatal). The accident rate on all state highways is determined using all the state highway mileages presented in Table 6.6, which is a total of approximately 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 6.7. Rates of Very Large Trucks Crashes and Accidents for All Five State Highways[(a)]

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 five states	5,224	5,379	5,604	16,207	5.59E-07

(a) Source: MCMIS, All States: Vehicle, GVW over 26,000 lb. Includes all crash events so the rate uses all states highway mileages.

These accident rates use crash events for >26,000 pounds 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.



Vehicle Reports: Gross Vehicle Weight

Select characteristics to filter crash data

Data Source: MCMIS | Carrier Domicile: All Domiciles | Vehicle Type: Large Trucks
Report Focus: All States / National / State -- Select State --
Time Period: 2019 | Calendar / Fiscal | SUBMIT

Submit Feedback
Do you have comments, questions or suggest for the updated Crash Stats module?
Tell us what you think by Submitting feedbac

All Crashes | Fatal Crashes | Injury Crashes | Towaway Crashes

Print PDF | Download Data | Report Descriptior

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

Compare Selected States | Create Peer Group

All States: Vehicle Report

State	Gross Vehicle Weight				
	10,000 lbs. Or Less	10,001 lbs. - 26,000 lbs.	Over 26,000 lbs.	Missing	Total
	# of Veh	# of Veh	# of Veh	# of Veh	# 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 6.22. FMCSA A&I Data Query Tool Showing State Accidents by Goss Vehicle Weight for Large Trucks, 2019

6.2.3 Very Large Truck Accident Rates for the Interstate Highways in the States Along the Route

The accident rates in Section 6.2.2 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, but 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 acquire this information.

Fatal crash data are available from the NHTSA using the Fatality and Injury Reporting System Tool. This query tool allows a user to construct customized queries from the FARS and also from the General Estimates System/Crash Report Sampling System.[42]

Data were queried for fatal accidents on interstate and non-interstate highways involving very large trucks (>26,000 pounds GVW) for 2017–2019. Very large trucks involved in fatal accidents totaled 240, and they were involved in 626 fatal accidents that occurred on all highways in the five states over the 3 years. Summarized results of the query (for the most harmful event [MHE]) are shown in Table 6.8.

Table 6.8. Very Large Truck Fatal Accidents by Type of Highway, 2017–2019

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 five 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 6.7 is multiplied by the percentage of fatal accidents occurring on interstate highways in Table 6.8; overall, this assumes the number of all types of very large truck accidents on interstate highways is 38.3 percent 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 6.5, 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 6.9. 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 miles on interstate highway. In general, the all-highway all-accident rates in Table 6.7 are reduced by about a factor of two when estimating the interstate highway all-accident rates in Table 6.9.

[42] Fatality and Injury Reporting System is available at https://cdan.dot.gov/query.

Table 6.9. Determination of Interstate All-Accident Rates for Very Large Truck Using Fatal Accident Comparison, 2017–2019[(a),(b)]

State	Total Events	% Fatal Accidents on Interstates	Interstate Miles × 10^6	% Interstate Miles of all Highways Miles[(c)] (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**

(a) Total events from are from Table 6.7; the percentage of fatal accidents on interstates is from Table 6.8; and the interstate miles are from Table 6.5.
(b) 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 of all accidents on the interstate only.
(c) The information in this column is not used in the calculation of accident frequencies but provides perspective.

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.

6.3 Development of Estimates of the Likelihood of the Occurrence of Types of TNPP Transportation Package Accidents

This section discusses development of the estimates of the likelihood of occurrence of types of accidents that can occur during TNPP transportation package and estimation of the frequencies for each bounding representative accident. Section 6.3.1 discusses development of the estimates of the likelihood of the occurrence of TNPP transportation package accidents that involve highway accidents including non-collisions (e.g., jack-knife accidents) as well as collisions that involve external impact. Section 6.3.2 discusses development of the estimates of the likelihood of the occurrence of TNPP transportation package accidents that result in loss of containment events that do not involve external impacts on the transportation package caused by a highway accident.

6.3.1 Development of Estimates of the Likelihood of the Occurrence of Highway Accidents

This section discusses determination of the estimated frequencies of highway accidents to which a TNPP transportation package could be subjected. Their frequencies are 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 impacts on the TNPP Package and non-collisions (e.g., jack-knife accidents) that may induce loading interior to the TNPP Package subjecting the reactor and containment to impact if an object becomes unrestrained within the CONEX box-like structure.

The basis for developing the estimates of the likelihood of route-specific accidents occurring is the very large truck interstate accident data for the five states described in Section 6.2. However, these state-specific data only have sufficient resolution to determine the frequencies of those classified as fatal accidents. Developing likelihood estimates for all accidents (i.e., fatal, injury-only, and property damage-only accidents) requires the use of 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 the frequency of different accident types. Development of these accident frequencies is discussed in Section 6.3.1.1.

For certain kinds of accidents, GIS data for route-specific road hazards not addressed 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, as discussed in Section 6.3.1.2, and a drop to a lower elevation, as discussed in Section 6.3.1.3.

6.3.1.1 Development of Frequencies of Accident Occurrences from Accident Data for Different Types of Highway Events

Nationwide all-accident data about interstate highway systems involving large trucks—single-unit and combination trucks greater than 10,000 pounds GVW—are available. Most of these trucks are very heavy trucks, greater than 26,000 pounds 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 6.10 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), unyielding (6), yielding (27), and split (3). Split accident categories reflect types for which additional resolution is needed (as described below). Three accident types are not included in Table 6.10 because they were determined not to be relevant to TNPP Package transportation based on their description.

Event data for the three split categories are shown in Table 6.11. The first entry in Table 6.11 pertains to accidents involving the “motor vehicle in transport,” which is 84 percent of the total number of 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 about 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 percent 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 percent).

Table 6.10. Nationwide Interstate Accident Events Involving Large Truck Including Those Resulting in Fatality, Injury Only, and Property Damage Only Events, from 2017–2019[(a)]

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	Unyielding	5,703	0.83%
Bridge Overhead Structure	Unyielding	937	0.14%
Other Fixed Object	Unyielding	686	0.10%
Bridge Rail (Includes Parapet)	Unyielding	271	0.04%
Bridge Pier or Support	Unyielding	172	0.03%
Unknown Fixed Object	Unyielding	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 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	—

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

The next two entries in Table 6.11 pertain to the number of crashes with embankments or the ground and for the purposes of the PRA are split into unyielding or yielding collisions. The percentage of unyielding 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 Table 6.1 and Table 6.2. The fraction of hard rock wayside surface value is 11.1 percent of the proposed transport route mileage. The remainder of the wayside surfaces are considered yielding.

Table 6.11. Accident Category Splits for Large Truck Crash Events on Interstate Highways

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

(1) Combination trucks and buses are considered heavy vehicles (>26,000 pounds GVW). Does not include large single-unit trucks, for which information is not available.
(2) Based on information provided in Table 6.1 and Table 6.2.

Using the data described above, the large truck accident events are organized into the eight categories shown in Table 6.12 and the percentage of accidents that occur in each accident category is determined (e.g., collisions with light vehicles is 74.3 percent of all crashes).

Table 6.12. Nationwide Estimated Likelihood of the Occurrence of Large Truck Accidents on the Interstate by Key Accident Categories[(a)]

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
Unyielding 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

(a) Accident categories based on information in Table 6.10 and Table 6.11.

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 6.12 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 pounds GVW, and not for very heavy trucks greater than 26,000 pounds GVW.

The frequencies of accident occurrences for categories of very large trucks 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 6.2.3 and presented in Table 6.9. To obtain each accident category identified in Table 6.12, the total accident frequency is multiplied by the percent of 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 6.13 for the entire 1,289-mile transport route.

Table 6.13. Estimated Likelihood of Very Large Truck Accidents Occurring on Five-State Interstate Route[(a)]

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
Unyielding 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

(a) Accident rates based on the very large truck total accident frequency from Table 6.9 and percentage information from Table 6.12.

Additional queries of the MCMIS database were performed to support development of frequencies for the bounding representative accidents identified in Table 5.6 and discussed in Section 5.3.4.9. Given that bounding accidents are representative of a subset of the TNPP transportation accidents identified by the hazard analysis summarized in Table 5.5 and discussed in Sections 5.3.3.2 through 5.3.3.33, the frequencies of contributors were determined and summed together to calculate the frequency of each bounding representative accident. To determine the frequency of certain bounding representative accidents, frequencies for applicable accident categories in Table 6.13 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 6.12, a total of 2,340 fire/explosion accidents occurred involving large trucks during this period. Of these accidents, the FHE was specified as some other type of accident for 25.4 percent 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 percent of the fire/explosion events. These percentages are used to develop estimates of 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 break out accidents involving tanker trucks from other large trucks. However, information about miles driven by tanker trucks for each state is available 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 pounds) in 2002. Tanker trucks were estimated to compose 10.3 percent 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 estimates of 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 5.6. The MCMIS dataset does not support the breakout of data into these categories, so estimates were based on further evaluating the accidents categorized as “Motor Vehicle in Transport,” as shown in Table 6.11. 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 percent of the 581,859 accident events considered. Light impacts were assumed for property damage-only accidents, accounting for 76.0 percent of the transport accidents. These percentages were used to develop estimates of 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, unyielding impacts, rollovers/overturns, and drops. Medium impacts are other crashes, including light vehicle collisions, yielding impacts, and jackknives.

6.3.1.2 Frequency of Highway Accidents that Could Result in Criticality Events

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
- Crash that results in RPV damage, fire, and inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash that results in fire and TNPP Package damage
- Control rod withdrawal (or other reactivity insertion event) caused by impact during a 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 liquid Shield Tank that surrounds it and could be ruptured during the accident. This accident is a subset of accidents that result in a drop to a lower elevation; therefore, the likelihood of the occurrence of this criticality accident is a subset of the likelihood that the transport vehicle drops to a lower elevation. Two approaches were taken in developing an estimate of the frequency for this accident; they each 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 6.1.3 and truck accident frequency data and analysis presented in Section 6.3.1.1 were used to estimate a frequency for this accident. In Table 6.4 in Section 6.1.3, the results of the GIS analysis show that there are 301 segments, 100 ft in length, along 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. The required conditions for such locations were assumed to be a body of water near the roadway (i.e., 50 m or less) sufficiently deep (i.e., 5 ft) where there is enough slope (i.e., a 1-to-3 slope) between the roadway and body of water to cause the truck and trailer to slide or roll. These segments represent 28,612 ft of the route that bypasses Denver for a total of about 5.4 miles (whether or not the bypass is used). This distance is about 0.42 percent 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 the 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 percent of the route can lead to a 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 accident frequency estimate is conservative because in addition to the conditions specified above (i.e., nearness to a sufficiently deep body of water and sufficient slope to cause the truck and trailer to slide or roll) other conditions are needed that would reduce the estimated frequency of this accident. However, it is difficult to estimate the probabilities of these other conditions, which include the fact that:

1. The required conditions do not necessarily exist simultaneously along the 100-ft segment.
2. Many accidents would not leave the road enough to be caused to slide or roll down the adjacent slope.
3. The truck, trailer, and TNPP Package may come to rest short of the body of water depending on the circumstances of the cash, the ground surface between the roadway and body of water, and the presence of rocks, scrubs, and trees that may impede their slide or roll.
4. The water may not be sufficiently deep during the time of year the accident occurs or at the point in the stream, river, or other body of water were the TNPP Package ultimately comes to rest.

Given the uncertainty in the estimated frequency using GIS information, it is judged that the estimate could be too conservative by an order of magnitude or more. (If the individual conditional probabilities are assumed to be 50 percent the combined probability of the four conditions would be 6.3 percent.) Even though the estimated accident frequency using this approach is clearly conservative in this case, the approach illustrates the potential value of using GIS to identify road hazards and estimate accident frequencies.

In the other approach, data for the proportion of “immersion/partial immersion” events to total number of large truck interstate accidents nationwide was developed as presented in Table 6.12. This ratio was then multiplied by the route-specific five-state interstate accident rate times the number of miles in the route in Table 6.13 to get an accident frequency of 5.12E-09 per year, assuming one shipment in a year. This estimate is based on immersion events identified in the dataset as the MHE. From 2016–2019, subcategories of FHE included immersion or partial immersion, motor vehicle in-transport, and collision with a guardrail face. A total of 12 immersion or partial immersion MHEs were reported during this period for large trucks on interstate highways; all of them resulted in fatalities and there were no injury-only or property-damage-only events that involved immersion. It is judged that immersion or partial immersion events for large trucks are likely to be clearly identified and reported because of their uniqueness. Section 6.5.9 describes how the likelihood of the bounding representative accident associated with this event is determined using this data. Even allowing for possible underreporting using the data approach because there may be uncounted non-MHEs, it is judged that the frequency of this flooded criticality event is less than 5E-07 per year. Given, this estimate along with the conservatism explained above for estimating the frequency of this accident using the GIS approach, the final estimate is judged to be less than 5E-07 per year.

The second flooded criticality accident concerns addition of a moderator and possible change in core geometry caused by inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash that results in fire and TNPP Package damage. For the demonstration TNPP design, the change in core geometry is not required to cause criticality if the core is inundated, but an alteration in core geometry is possible from the collision that could contribute to criticality. This accident requires a highly unlikely set of circumstances because the TNPP Package would have to be damaged in a way that water (or other material) could enter and inundate the reactor core, fire must ensue, fire suppression water (or other material) would have to be directed at the TNPP Package in a way that it enters and inundates the core, which is inside the Reactor Module of the TNPP Package but could sustain damage, and there would have to be a need to direct fire suppression water toward the TNPP Package, though the most probable source of fire would be the tractor engine, the diesel fuel, or the tire/wheels. The frequency of this accident is determined by estimating the likelihood of a collision that results in fire and damage to the TNPP and the conditional probability that the core is inundated with fire suppression water or other hydrogenous material. Section 6.5.10 describes how the likelihood of the bounding representative accident associated with this event is determined using the likelihood of bounding representative accidents that lead to fire and can damage the TNPP Package plus the conditional probability that the core is inundated.

The third criticality accident concerns an impact so hard that the control rods are withdrawn against the restraining and locking mechanism. Therefore, the likelihood of this accident occurring would be a subset of the likelihood of the hard-impact accidents discussed above (i.e., impact with heavy vehicles, impacts with unyielding objects, and rollovers). However, this frequency has not been developed because it is understood that the design will eventually preclude the possibility of this TNPP transportation accident occurring.

6.3.1.3 Frequency of Highway Accidents that Could Result in a Drop-to-a-Lower Elevation Event

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 a lot more damage could occur to the TNPP Package if the vehicle drops to a significantly lower elevation. The GIS hazard data and analysis presented in Section 6.1.4 and truck accident frequency data and analysis presented in Section 6.2.3 were used to estimate a frequency for this accident.

As described in Section 6.1.4, GIS analysis identified 318 segments each 100 ft in length along 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 because 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 percent 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 percent of the route can lead to a drop to a 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.

6.3.2 Development of Estimates of the Likelihood of Occurrence of Non-Crash Accidents

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

Some of the accidents identified in the Table 5.5 accident summary 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 6.3.2.1. Additionally, there are a set of lower energy accident scenarios in which the package containment fails but does not involve mechanical impact from a highway accident. One set of these accident types concerns a breach or loss of the reactor containment boundary when the system is not pressurized, while another type concerns a breach or loss of the reactor containment boundary when the system is pressurized. A third type concerns a breach of other than reactor containment boundary components, such as in the contaminated Shield Tank. The development of the frequencies for these accidents is provided in Section 6.3.2.2. There is also a set of event scenarios defined to be technical or logistic difficulties during transport that cause a lengthened transport time and an increased exposure of workers to radiation. Discussion of these frequencies is provided in Section 6.3.2.3.

6.3.2.1 Development of Estimates of the Frequency of Non-Impact Fire-Only Accidents

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

The following discussion pertains to fire that originates outside of the TNPP transportation package. Based on the data presented in Section 6.2.3, the accident rate is 3.90E-07 per mile for very large truck (greater than 26,000 pounds 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 percent did not involve a collision with another vehicle or object and 2.7 percent of these non-collision accidents resulted in a fire or explosion. The conditional probability of a fire is therefore assumed to be 0.74 percent (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 (NRC 2014). 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 percent of these accidents did not involve collision with another vehicle or object and 5 percent of the non-collision accidents resulted in a fire or explosion.

The following discussion pertains to fire that originates within the CONEX box-like structure of the TNPP transportation package. Based on available information, very little combustible or flammable material will be contained within the entire Reactor Module (configured as the TNPP transportation package). The primary combustibles in this module are cable insulation and lubricants for motors (BWXT 2022[43]). Cable insulation is associated with reactor electrical/instrumentation and control components that are located at the fore end of this module. However, these electrical circuits will not be energized during transportation. A passive venting system is planned to be used to cool the reactor within the TNPP Package during transportation (BWXT 2022[44]). Other cabling is associated with heat-detection devices that 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 occurring within the Reactor Module of the TNPP Package. Additionally, there is cabling associated with radiation monitors, and temperature and pressure transducers to monitor the internal environmental conditions inside 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[45]). It is unclear which, if any, of these systems will be energized during transportation. Nevertheless, energized electrical/instrumentation and control systems that could exist in the TNPP Package during transport may include systems to support functions such 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[46]). Therefore, it is assumed for the TNPP PRA that there will be energized electrical components in the TNPP Package during transport. However, all cabling in the TNPP Package will be inserted and protected in electrical rated conduit per the National Electrical Code (NFPA 70).

Regarding lubricants, the Reactor Module of the TNPP Package 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 is operable, or active, during transport operations, they nevertheless do contain lubricants that are flammable and therefore contribute to the combustible loading within the module.

[43] BWXT Final Design Report, Appendix IV, ATL-TECR-109977 – “MNPP Facility Fire Hazards Analysis”.
[44] BWXT Final Design Report, Appendix I, ATL-PLAN-110124 – “Transportation Plan”.
[45] BWXT Final Design Report, Section 2.3.1.6.3.
[46] BWXT Final Design Report, Appendix I, ATL-PLAN-110124 – “Transportation Plan”.

As stated above, the frequency of fires that occur inside the CONEX box-like structure 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. Two sources of such information are 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 TNPP Package CONEX box-like structure 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, 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.

The fire frequency bins presented in Table 4-4 of NUREG-2169 that may be applicable to the TNPP Package CONEX box-like structure 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 it is not applicable in this assessment (additionally, the Control Module instrumentation and control systems are not functional or energized during TNPP Package transportation initiatives and are 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 during reactor operation or during 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, so transient fires are not postulated. The fire frequencies for the cited fire frequency bins need to be adjusted to be applicable to a CONEX box-like structure because the number of fire ignition sources in the structure are substantially less than those 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 2015) (BWXT 2022[47]). 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 2013a), 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 configured as the TNPP Package. 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 do 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* [Hamzehee 2013b]) states:

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

> 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 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. The same metrics are used in this report to scale the nuclear power plant junction box fire ignition frequency to determine 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, that the Reactor Module contains one or two energized junction boxes during transportation and has an associated cable loading, or cable insulation mass, of 100 to 200 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 pounds).[48] 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 (TNPP Package) shipment is assumed to require just 3 to 4 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 TNPP Package. As explained above, junction box fires do not generate enough heat to be self-sustaining, so they 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 used to cool the reactor within the TNPP Package during transportation initiatives. In addition, electric motors used during reactor operations (e.g., control rod drive motors) will not be energized during transportation of the TNPP Package, so they are not potential sources of fire ignition during shipments. However, the design features of the Reactor Module during transportation are not yet fully developed, so this analysis assumes the TNPP Package 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 percent are electric systems (per NUREG/CR-6850). A typical nuclear power plant based upon which the NUREG-2169 estimates of 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

[48] This estimate was performed based on engineering judgment based on the extensive experience of staff at PNNL reviewing nuclear power plant risk-informed applications, including associated fire PRAs, for the NRC.

the TNPP Package. 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 TNPP Package. These systems are expected to be powered and remotely monitored from outside 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 based upon which the NUREG-2169 estimates of 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.

In summary and as shown in Table 6.14, the estimate of the ignition frequency for a fire that originates outside of the TNPP Package is about 3.4E-06 per year assuming one shipment per year. A conservative estimate of the ignition frequency for a fire that originates inside the TNPP Package is about 7.0E-06 per year assuming one shipment per year based on summing contributions from possible junction boxes (5E-08 per year), electric fans (5E-06 per year), and electrical panels 9E-07 per year). Regarding fires that originate inside the TNPP Package, 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 TNPP Package during transport, then the risk contribution from these fire sources might be screened given the extremely low likelihood that a fire would occur and propagate to the extent that the Reactor Module (TNPP transportation package) is damaged.

Table 6.14. Frequencies of Fire-Only Events that Originate Inside and Outside the TNPP Package

Accident Type[(a)]	Basis	Frequency Per Year (assuming one shipment per year) [(b)]
Fire that originates outside the TNPP Package	Based on highway accident data for fire that do not involve a cash	3.4E-06
Fire that originates inside the TNPP Package	Based on fire ignition data for nuclear applications	7.0E-06

TNPP = Transportable Nuclear Power Plant.
(a) No credit was taken for early detection and mitigative response actions or the fact that all wiring was in conduit.
(b) This estimate is conservative and does not credit cable protection or limitations on combustibles.

6.3.2.2 Development of Estimates of the Frequency of Non-Crash TNPP Package Containment Failure Accidents

As described above, a set of postulated TNPP Package containment failure accident scenarios did not involve highway accidents or fires. These accident scenarios are organized into the three sets presented below along with their possible causes (initiating events). Scenarios 1 through 5 involve failure of the nonpressurized reactor coolant boundary containment, 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 TNPP Package (Reactor Module prepared for over the road transport) other than the reactor containment boundary.

Nonpressurized reactor containment boundary failure can be caused by:

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

Pressurized reactor containment boundary failure can be caused by residual heat buildup in combination with the following:

- 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 a containment failure
- High ambient air temperature that in combination with the residual decay heat pressurizes the reactor containment boundary in combination with a containment failure.

Package element failure other than the reactor containment boundary can be caused by the following:

- Pressurization due to radiolysis of hydrogenous material (e.g., Shield Tank not fully drained) and possible hydrogen accumulation and ignition
- Pressurization caused by loss of ventilation or high ambient air temperatures
- Containment failure caused by random failures and/or vibration
- Containment failure due to a hailstorm that causes general severe vibration.

Concerning the loss of containment in a nonpressurized reactor coolant boundary containment, the following addresses development of estimates of the initiating event frequencies for the five accident scenarios identified above. The primary cooling system will be dismantled for transport of the TNPP and certain piping segments and the IHX will be packaged separately from the Reactor Module. Hence, containment isolation features or devices (e.g., Grayloc® bolted blind flanges [BWXT 2022[49]]) will be installed on the primary cooling system inlet/outlet piping after its dismantlement. An HMIS will eventually be included in the TNPP design and 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.

[49] BWXT Final Design Report, pages 2-13 and 7-36.

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 pipe 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 coolant boundary containment that will have low pressure
2. The cited failure rate being for general equipment, whereas the fittings used to seal the system are expected to be high-quality nuclear-grade seals, like an item or device that might be used on a spent nuclear fuel transportation package.

The fitting failure rate from the DoD handbook for a general failure mode is judged to be 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 reasonably 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 about 1E-03 per shipment.

Regarding human error in packaging the reactor containment boundary and preparing the Reactor Module as a TNPP transportation package, a conservative estimate can be generated based on simplified HRA. HRA modeling guidance used by the NRC provides an approach for estimating the failure probability of operator actions at nuclear power plants in NUREG/CR-6883 (Gertman et al. 2005). The Standardized Plant Analysis Risk-HAR (SPAR-H) approach uses a base Human Error Probability (HEP) of 1E-02 for execution of an action and 1E-03 for diagnosis of 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. An estimated failure probability of 1E-02 is used assuming a single execution step could fail the containment feature that is used to seal the reactor coolant boundary containment after the modules have been disassembled. This estimate does not include any specific consideration of PSFs. However, it is likely the packaging will be checked or inspected before transport, so 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 causes containment failure during transport of the TNPP Package 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 coolant 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 is 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 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 TNPP Package is about 1E-04 per shipment.

Regarding extreme cold that causes containment failure, 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 it does show that 2.3 percent of fitting failures are caused by being out of tolerance (out of specifications). This is a separate failure mode from deterioration, wear-out, breaking, or improper adjustment. The TNPP would not likely be transported if 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 percent of the total failure likelihood, the rate of 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 about 2E-05 per shipment. This approach is judged to provide a basis for an estimate until more details are known.

Regarding the breach of a pressurized reactor containment boundary, the following describes the development of the estimate of the initiating event frequencies for the two accident scenarios identified above. For these accidents it is assumed that residual decay heat 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 an HMIS, which will eventually be included in the TNPP design, 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 the impact on vents or other heat transfer pathway elements that decrease heat removal to the extent the reactor containment boundary pressurizes during transport of the TNPP Package, it is difficult to find explicitly applicable failure information. This failure might occur during transport because load restraints loosen 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 202250). This in turn results in heat buildup and potential failure of containment through the fittings applied to the reactor coolant boundary containment. Degradation of the heat transfer pathway could also occur during preparation of the Reactor Module for transport and could 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 rate of 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 of preparing the Reactor Module for transport as the TNPP Package. 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, the heat transfer pathway likely 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 of 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 above for random containment feature failure (1E-04 per shipment), vibration and shock (1E-03 per shipment), human error in packaging (1E-04 per shipment), human error in disassembly (1E-04 per t shipment), and extreme cold that causes failure (2E-05 per shipment). The sum of these failure rates is about 1.3E-03 per shipment. So, the combined 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 the impact of high ambient air temperatures that along with decay heat increases the pressure in the reactor coolant boundary containment in combination with failure of containment, it is 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 the containment fails. However, to be conservative it can be assumed (until more design information is provided) that the likelihood of this event occurring 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

[50] BWXT Final Design Report, Appendix I.2, ATL-PLAN-110124, “Transportation Plan,” page 29.

up is monitored and compensatory measures are available, then this accident scenario might be screened.)

Regarding the containment breach of other parts of the TNPP Package besides the reactor containment boundary (e.g., the Shield Tank), the following presents the development of the estimate of the initiating events frequency for the accident scenarios identified above. As shown in Table 4-4 in NUREG-2169 for Event Class 9 (loss of general package containment – not reactor containment boundary), these accidents only involve release of contamination in the TNPP Package components that are not part of the reactor containment boundary. Any pathways into or out of the Shield Tank are expected to be closed during transport.

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.

In summary the accident frequencies for non-crash package containment failure accident scenarios are presented in Table 6.15.

Table 6.15. Non-crash Containment Failure Accident Frequencies

Non-crash Loss of Containment Accident Scenarios	Accident Frequency
Breach of Nonpressurized Containment[(2)] caused by one of the following:	
Random failure of transport containment devices	1E-04
Vibration and shock from over-the-road travel	1E-03
Human error in assembling the TNPP Package	1E-04
Human error in disassembling the TNPP Package	1E-04
Extreme cold that causes containment failure	2E-05
TOTAL	**1.3E-03**
Breach of Pressurized Containment caused by one of the following combinations:	
Same failures above that breach containment in combination with Condition 1 or 2 below.	1.3E-03
Condition 1 – 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.	1.5E-03
Condition 2 – High ambient air temperature that in combination with the residual decay heat pressurizes the reactor containment boundary. (Total random containment breach frequency) minus (Exclusion of extreme cold contribution) times (Conservative probability of high ambient air temperature) = ((1.3E-03) – (2E-05)) × 1.00	1.3E-03 × 1.00
TOTAL for Condition 1	2E-06
TOTAL for Condition 2 (This case used to bound the accident frequency)	**1.3E-3**
TNPP = Transportable Nuclear Power Plant.	

6.3.2.3 Incidents that Cause Increased Exposure Time

There is also a set of events in which technical or logistic 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 TNPP Package that require 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 (though unanticipated) exposure, 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.

6.4 Assumptions Made When Estimating the Likelihood of Accidents to Occur

As described above, accident likelihood estimates are based on road hazard information determined using GIS, very large truck (>26,000 pounds GVW) interstate and all state highway data for the five states that the assumed route traverses, and nationwide large truck (>10,000 pounds 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 the limitations of these data sources. 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 5.3.2.2 and 7.4, respectively. Specific main assumptions used in the development of the estimates of accident likelihoods were as follows:

1. The route is from INL to WSMR and uses interstate highways in parts of Idaho, Utah, Wyoming, Colorado, and New Mexico as described in Section 6.0.
2. There is one transport in a year to be able to provide the accident frequency estimate on a per-year basis.
3. 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 versus all state highways is the same for very large truck accidents of all types (i.e., including those that are not fatal) on these interstates versus all state highways, as described in Section 6.2.3.
5. The known types and proportions of large truck interstate accidents in the nationwide dataset (i.e., in the MCMIS database) are the same as the types and proportions of very large truck accidents on the assumed route, as described in Section 6.3.1.1.
6. For accidents in the nationwide dataset (i.e., MCMIS) described in Section 6.3.1.1, where fire was the MHE, if a FHE was not specified, then the FHE is fire. If another kind of accident was designated as the FHE, then it was assumed to be a mixed accident (e.g., a crash and fire) as described in Section 6.3.1.1.
7. For accidents in the nationwide dataset (i.e., MCMIS) described in Section 6.3.1.1, the most severe types of impact accidents, hard impacts (i.e., BRA 5H) are considered to be heavy vehicle collisions, impacts with unyielding objects, rollovers/overturns, and drops to lower elevation. Medium impacts (i.e., BRA 5M) are considered to be all other crashes, including light vehicle collisions, impacts with yielding objects, and jackknives.

8. For “less than hard impacts” accidents (i.e., BRA 4 as described in Section 6.3.1.1), those that result in a fatality or injury-only in the nationwide dataset (i.e., MCMIS) are considered medium impacts (BRA 4M) and those that result in property damage-only are considered light impacts (BRA 4L).
9. 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 pounds) miles nationwide, as described in Section 6.3.1.1.
10. For the GIS estimation of the submersion accident, locations along the route identified 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 and that if a truck in an accident left the road it could slide or roll into the body of water, as described in Sections 6.1.3 and 6.3.1.2.
11. For GIS estimation of the frequency of accident resulting in a drop to a lower elevation, 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 6.1.4 and Section 6.3.1.3.
12. For non-crash fire-only events, no credit was quantified for early detection or fire response mitigation as described in Section 6.3.2.1.
13. For BRA 4L, it was conservatively assumed that there was 10 percent chance that a worker could be located in the radiation stream through a gap or fissure in the radiation shielding for the duration assumed in the consequence analysis (i.e., 30 minutes).

6.5 Accident Frequency Results for the Bounding Representative Accidents

This section discusses development of estimates 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 6.3. 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 discussion of the development of estimates of the frequency of each bounding representative accident is presented here based on the detailed discussions of accident frequency development for specific accident types. A summary table of the frequencies for all bounding representative accidents is presented in Table 6.16.

Table 6.16. Accident Frequency Estimates for the Bounding Representative Scenarios

ID	Descriptions	Accident Frequency per Year[(a)]
BRA 1	Fire-only event that originates inside the CONEX box-like structure of the TNPP Package.	7.0E-06
BRA 2	Diesel fuel fire-only event that originates outside the Reactor Module, propagates into the CONEX box-like structure and ignites combustible material which damages the reactor containment boundary and impacts containment capacity.	3.4E-06
BRA 3	Hard-impact highway accident that leads to release of radioactive material and loss of shielding. Includes impact with heavy vehicles	7.1E-05

ID	Descriptions	Accident Frequency per Year[(a)]
	and unyielding objects (e.g., concrete abutments or rock embankments), significant drops to lower elevation, or rollovers.	
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 but some degradation of transport shielding. Light impact such as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or impact with a light vehicle that is not severe (e.g., results in property damage only).	3.3E-05
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 the nonpressurized 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 the pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.	1.3E-03
BRA 9A	Addition of moderator and possible change in core geometry caused by a drop into body of water that results in criticality.	<5E-07[(b)]
BRA 9B	Addition of moderator and possible change in core geometry caused by a crash that results in RPV damage, fire, and inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash.	<5E-07
BRA 10	Control rod withdrawal (or another reactivity insertion event) caused by impact from a road accident that results in criticality.	[(c)]

BRA = bounding representative accident; CONEX = container express; ID = identification; RPV = Reactor Pressure Vessel.
(a) For accident frequency calculations, one transport in a year is assumed.
(b) Assessed to be between 5.1E-9 and 2.1E-06 but judged to be less than 5E-07 per year per discussion in Section 6.3.1.2
(c) This study assumes the design goal of precluding a reactivity insertion event in a TNPP transportation package accident will be met.

6.5.1 Fire Only that Originates Inside the CONEX Box-Like Structure – BRA 1 Frequency

BRA 1 is a fire that originates inside the CONEX box-like structure of the Reactor Module. It is a general fire that originates from sources such as an electrical cable fault, potentially propagates to the reactor coolant boundary containment, and ignites associated combustible material. It does not involve impact from a crash. It includes consideration of 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 6.3.2.1. The estimated frequency is for this accident is 9E-07 per year (assuming one transport in a year).

6.5.2 Fire Only that Originates Outside the CONEX Box-Like Structure – BRA 2 Frequency

BRA 2 is a diesel fuel fire that originates outside the CONEX box-like structure, propagates into the CONEX box, and ignites combustible material, which possibly damages the reactor coolant boundary. It does not involve impact from a crash. The frequency of a BRA 2 event is estimated using the MHE as the frequency of fire/explosion accidents and adjusting it when the FHE is also fire/explosion. This is considered a fire-only event as explained in Section 6.3.1.1 and Section 6.3.2.1 and is 74.6 percent 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.

6.5.3 Hard-Impact Road Accident – BRA 3 Frequency

BRA 3 is an impact with heavy vehicles and solid unyielding objects (e.g., concrete abutment, rock embankments), fall to a 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,” including collision with a heavy vehicle, an impact with an unyielding object (examples given above), a rollover/overturn accident involving the truck as described in Section 6.3.1.1, and an event not specifically identified in crash databases. The frequency of a drop to a lower elevation event was determined using route-specific GIS hazard information as described in Section 6.3.1.3. 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.

6.5.4 Less than Hard-Impact Road Accident – BRA 4M and BRA 4L Frequency

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), jackknives that do not involve impact, and impacts with non-rock ground surfaces such as soil or clays. BRA 4 is split into two subcategories, BRA 4M (for medium impact) and BRA 4L (light-impact).

The BRA 4M events are 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 BRA 4L event does not result in significant damage to the TNPP Package; there is no release of radiological material but some degradation of transport shielding.

6.5.4.1 BRA 4M Frequency

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; this is 26 percent of light vehicle collisions, as discussed in Section 6.3.1.1. Using these bases, the estimated frequency of BRA 4M events for the assumed route is 9.7E-05 per year assuming one transport in a year.

6.5.4.2 BRA 4L Frequency

The frequency of BRA 4L events is determined starting with estimating the frequency of light vehicle collisions adjusted to consider only those resulting in property damage; this is 74 percent of light vehicle collisions, as discussed in Section 6.3.1.1. Using these bases, the estimated frequency of BRA 4L events for the assumed route would be 3.3E-04 per year assuming one transport in a year. BRA 4L does not result in release of radiological material but damages the transport shielding to create a gap or fissure in the CONEX box-like structure and transport shielding of the TNPP Package (e.g., damage to a corner of the TNPP Package) allowing radiation streaming to occur. Accordingly, a worker would not receive a radiation dose unless they were located in the radiation stream. It was conservatively assumed that there was a 10 percent chance that a worker could be in the radiation stream for the duration assumed in the consequence analysis (i.e., 30 minutes). Therefore, the frequency of BRA 4L was estimated to 3.3E-05 per year (i.e., 3.3E-04 per year times 0.1).

6.5.5 Road Impact and Fire Accident Except with a Tanker Carrying Flammable Material – BRA 5H and BRA 5M Frequency

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 subcategories, 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, 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 crashes that result in fire and account for 25.4 percent of fire/explosion accidents, as explained in Section 6.3.1.1. 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 5.3.4.3. 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.

6.5.5.1 BRA 5H Frequency

The basis for the estimation of accident frequency for BRA 5H is like the estimation described for BRA 3 above with the exception that it is impact followed by a fire. The accident frequency of BRA 5H was determined using the percentage of hard-impact accidents to all impact accidents, which is effectively, BRA 3/(BRA 3 + BRA 4M + BRA 4L) multiplied by the frequency of relevant fire/explosion events and subtracting the accident frequency for tanker collisions leading to fire and possible explosion (BRA 6). Using these bases, the estimated frequency of BRA 5H events for the assumed route is 2.6E-08 per year assuming one transport in a year.

6.5.5.2 BRA 5M Frequency

BRA 5M medium impacts are the same as those described for BRA 4M above which only those that result in fatality or injury. BRA 5M involves the percentage of medium-/light-impact accidents divided by all impact accidents—effectively (BRA 4M + BRA 4L)/(BRA 3 + BRA 4M + BRA 4L) multiplied by the frequency of relevant fire/explosion events, as discussed in Section 6.3.1.1. Using these bases, the estimated frequency of BRA 5M events for the assumed route is 5.9E-07 per year assuming one transport in a year.

6.5.6 Collision with a Tanker Carrying Flammable Material and an Ensuing Fire – BRA 6 Frequency

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 percent 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 percent, as discussed in Section 6.3.1.1. Given that 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 of BRA 6 events for the assumed route is 7.1E-08 per year assuming one transport in a year.

6.5.7 Loss of Nonpressurized Reactor Containment Boundary – BRA 7 Frequency

BRA 7 is a loss of reactor boundary containment event resulting in a nonpressurized release from the reactor coolant boundary containment not associated with a road impact accident. The development of the accident frequency contributors to BRA 7 are described in detail in Sections 6.3.2.2. 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 of BRA 7 is the sum of these failure rates, which is 1.3E-03 per year assuming one transport in a year.

6.5.8 Loss of the Pressurized Reactor Containment Boundary – BRA 8 Frequency

BRA 8 is a loss of reactor boundary containment event for a pressurized release from the reactor coolant boundary containment that is not associated with a road impact accident. The development of accident frequency contributors to BRA 8 are described in Section 6.3.2.2. 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 extreme ambient air temperature in combination with a reactor containment boundary failure. The estimated accident frequency of BRA 8 is 1.3E-03 per year assuming one transport in a year.

6.5.9 Criticality Event Involving a Drop into a Body of Water – BRA 9A Frequency

BRA 9A is the 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, along with the pros of cons of using each, are discussed in Section 6.3.1.2. Given that the GIS estimate is very conservative, the frequency was set to be less than 5E-07 per year.

6.5.10 Criticality Event Involving Inundation of the Core with Fire Suppression Water – BRA 9B Frequency

BRA 9B is the addition of a moderator and possible change in core geometry caused inundation of the core with fire suppression water or other hydrogenous material that enters the reactor core of the TNPP Package in sufficient quantities to cause criticality after a crash that results in fire and TNPP Package damage. This accident requires a highly unlikely set of circumstances because the TNPP Package would have to be damaged in such a way that water (or other material) could enter and inundate the core. Fire must ensue from the crash, fire suppression water (or other material) would have to be directed at the TNPP Package in a way that it enters and inundates the core. For water to enter the reactor pressure boundary the CONEX box-like structure of the TNPP Package would have to be damaged, and the reactor containment would have to be breached. There would also need to be a reason to direct fire suppression water toward the TNPP Package even though the most probable source of fire would be the transport tractor engine, the diesel fuel, or the tire/wheels.

There are five fire scenarios (i.e., BRA 1, BRA 2, BRA 5-M, BRA 5H, and BRA 6). BRA 1 and BRA 2 are fire-only events that do not involve a crash, and therefore, the containment is intact, and no water intrusion occurs. BRA 5H (hard impact and fire) and BRA 6 (crash with tanker carrying combustible liquids and fire) have accident frequencies well below the risk evaluation guideline frequency of 5E-07 per year. The risk of accident below a frequency of 5E-07 per year is acceptable regardless of consequence using the proposed risk evaluation guidelines.

BRA 5M (medium impact and fire) has an accident frequency of 5.9E-07 per year just over the risk evaluation guideline frequency of 5E-07 per year. However, as mentioned above, there are other conditions besides a crash, subsequent fire, and fire suppression response needed to produce a flooded criticality event that would reduce the estimated frequency of this accident. However, it is difficult to estimate the probabilities of these other conditions. The crash would need to cause an opening in both the CONEX box-like structure of TNPP Package and likely the RPV of the TNPP Package in such a manner that water could run into the reactor core. Fire suppression water would be directed at the fire which would likely be associated with the engine of the transport tractor, wheels or tires, or a fuel spill near or under the diesel fuel tanks rather than at the TNPP Package, which is carried by the trailer behind the truck. However, a fuel pool could form below the TNPP Package and there could be an opening in both the CONEX box-like structure of TNPP Package and possibly RPV of the TNPP Package caused by impact that allows fire suppression water to enter and inundate the RPV and core. It is judged that the probabilities of these conditions (though not quantitatively estimated) are enough to reduce the frequency of a flood criticality from fire suppression water for BRA 5M to below the risk evaluation guideline frequency of 5E-07 per year even without quantitative estimation.

6.5.11 Criticality Event Caused by Control Rod Withdrawal – BRA 10 Frequency

This section describes estimation of the frequency of BRA 10, which is a control rod withdrawal caused by impact from a road accident that results in criticality. This frequency has not been developed because it is assumed for this study the design goal of precluding this reactivity insertion event in a TNPP transportation accident will be met.

7.0 Development of the Estimated Consequences of the TNPP Transportation Package Accident Scenarios

This section discusses the approach for developing TNPP transportation accident consequences, specifically the radiation 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 7.1 discusses the methodology for determining the source term released from a TNPP transportation accident. Section 7.2 discusses the source term determined for each bounding representative scenario. Section 7.3 describes the approach for determining the radiation dose consequence from TNPP transportation accidents. Section 7.4 presents the list of assumptions used when determining the transportation accident source terms and the radiation dose consequences. Section 7.5 presents the radiation dose consequences for each bounding representative accident.

7.1 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 affect 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 that can impact a human receptor through different dose pathways. This released material is referred to as the “source term.” For these releases, the principal radiation 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 an inhalation dose.

For airborne releases, the source terms are estimated using the following five-factor formula (DOE 2013):

$$\text{Source Term} = MAR \times DR \times ARF \times RF \times LPF$$

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 nonreactor nuclear facilities, can be applied to a TNPP transportation package 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.

7.1.1 Material at Risk

Characterization of the TNPP Package inventory is addressed in Section 5.1 and the MAR TNPP transportation accident is discussed in Section 5.1.4.1. The development of the MAR is based upon the release of fission products and gases from the individual TRISO particles during normal operation that diffuse elsewhere into the reactor. Accordingly, MAR has been developed for three locations within the TNPP Package (as described in Section 5.1.4.1):

- Gaseous and nongaseous fission products retained within the TRISO
- Fission products that have diffused from the TRISO fuel and are held up in the compact and other core structures
- Fission products and gases that have diffused from the TRISO fuel and have condensed or plated-out in the reactor coolant boundary.

Using the approach as discussed in Section 5.1.4.1, release estimates were based on the 95 percent release fractions (shown in Table 7.1) developed for each of the 10 fission product classifications discussed in Section 5.1.4.1. These release fractions were then applied to the prototype core radionuclide inventory presented in Table A.1 in Appendix A to develop location-specific MAR (i.e., locations in the reactors where MAR exist). Estimates for different decay times are presented in Table A.1 through Table A.4 in Appendix A.

Table 7.1. 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
MAR = material at risk.			

7.1.2 Damage Ratio

The damage ratio represents the fraction of the MAR that is affected by accident-generated stresses. For this evaluation, the damage ratios are estimates based on TNPP Package transportation accident stresses transmitted to the MAR, and they are 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 7.2. The damage ratio is primarily a function of the energy involved in the accident and physical phenomena that can cause release. (For reference, a summary of the bounding representative accidents is p resented in Table 5.6 and described in some detail in Section 5.3.4.)

Table 7.2. Damage Ratios for the Bounding Represented Accidents

Represented Accident	FP/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
BRA = bounding representative accident; FP = fission products; TRISO = tri-structural isotropic (particle).			

7.1.3 Airborne Release Fraction and Respirable Fraction

The airborne release fraction represents the estimate of the total amount of radioactive material that can be suspended in air and made available for airborne transport under an accident-specific set of induced physical stresses. The airborne release fraction is primarily a function of the energy involved in the accident, physical phenomena that can cause releases, 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 in Aerodynamic Equivalent Diameter and less. For this evaluation, the airborne release fraction and respirable fraction estimates are based on both the material forms and accident stresses, and they are 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 7.3.

Table 7.3. Combined Airborne Release Fractions and Respirable Fractions for Represented Accidents

	Combined Airborne Release Fractions and Respirable Fractions		
Represented Accident	FP/Gases in TRISO	FP in CORE	FP in Pressure Boundary
BRA 1	—	—	—
BRA 2	—	6.00E-05	6.00E-05
BRA 3 (FP)	3.00E-04	3.00E-04	3.00E-04
BRA 3 (FG)	1.00E+00	NA	NA
BRA 4	—	3.00E-04	3.00E-04
BRA 5	—	2.50E-06	2.50E-06

Represented Accident	Combined Airborne Release Fractions and Respirable Fractions		
	FP/Gases in TRISO	FP in CORE	FP in Pressure Boundary
BRA 6	6.00E-05	6.00E-05	6.00E-05
BRA 7	—	—	1.00E-04
BRA 8	—	—	8.00E-04
BRA = bounding representative accident; FG = fission gases; FP = fission products; NA = not available; .RF = respirable fraction; TRISO = tri-structural isotropic (particle).			

7.1.4 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 it is subjected to atmospheric dispersion. The leak path factors assumed for this evaluation are provided in Table 7.4.

Table 7.4. Leak Path Factors for Represented Accidents

Represented Accident	FP/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
BRA = bounding representative accident; FP = fission products; TRISO = tri-structural isotropic (particle).			

7.2 Description of the Source Term for Each Bounding Representative Accident

The bases for the source terms developed for each of the bounding representative accidents that are not screened are described in the following sections. The source term values themselves are presented in Table A.1 through Table A.4 in Appendix A. The bases are described by discussing the source term factors (i.e., damage ratios, airborne release factors, respirable fractions, and leak path factors) associated with each bounding representative accident. BRA 9A, BRA 9B, and BRA 10 are criticality accidents but are screened from quantification of dose consequences. BRA 9A and BRA 9B are screened because they meet the proposed risk evaluation guidelines based on their exceptionally low accident frequencies (i.e., 5E-07 per year). BRA 10 is screened from quantification of dose consequences because the design goal of precluding a reactivity insertion event in a transportation accident involving the TNPP Package will be met.

7.2.1 Fire Only that Originates Inside the CONEX Box-Like Structure – BRA 1 Source Term

This section describes the source term development for a fire between the inside of the CONEX box-like structure and the reactor coolant boundary of the reactor within the TNPP Package. As discussed in the vendor documents (BWXT 2022[51]), given the low quantities of combustibles within the module, a fire, if it occurs, would be limited in size or potential for growth by the controlled environment. Because control, power, and instrumentation cabling are 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 by SSCs in the reactor containment boundary. There are no electrically active systems inside the reactor, so fire initiated inside the reactor is not expected during transport. 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 radiation dose consequence calculated for BRA 1.

7.2.2 Fire Only that Originates Outside the CONEX Box-Like Structure – BRA 2 Source Term

This section describes the source term development for a large fire outside the CONEX box-like structure of the TNPP Package. It assumes a large diesel fuel fire that originates outside of 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 that resulting from 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 percent) is assumed to be affected by the fire and released. For the MAR within the reactor containment boundary, a damage ratio of 1 (100 percent) is assumed to be affected 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 nonreactive material is applied. A leak path factor of 0.01 has been assigned based on the assumed complete failure of the seals (e.g., Grayloc® connectors) of the reactor coolant boundary containment creating a leak path to the compromised CONEX box-like structure 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.

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

7.2.3 Hard-Impact Road Accident – BRA 3 Source Term

This section describes the source term development for a severe impact highway accident that leads to release of radioactive material. It assumes the impacts are sufficient to cause failure or breaches in the TNPP Package components, including the CONEX box-like structure walls, transport shielding, and reactor coolant boundary, as well as damage to the core and fuel compacts. Affected MAR includes all locations (fuel, core, and reactor coolant system piping).

For the MAR within the fuel, a damage ratio of 0.001 (0.1 percent) 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-like structure 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 percent) 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-like structure and subsequent release to the environment.

For the MAR within the reactor containment boundary, a damage ratio of 1 (100 percent) 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 the reactor containment boundary.

7.2.4 Less than Hard-Impact Road Accident – BRA 4M and BRA 4L Source Term

This section describes the source term development for a less severe highway accident impact that leads to release of radioactive material. It assumes the impacts are sufficient to cause breaches in the CONEX box-like structure walls and reactor coolant 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 shielding loss. 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 accidents that result in insignificant damage to the TNPP Package. These light-impact accidents are defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or impact with a light vehicle that is not severe (e.g., results in property damage only). A more precise definition of yielding versus unyielding objects is discussed in the frequency estimation in Section 6.3.1.1 and presented in Table 6.10. Accordingly, BRA 4L is assumed to cause some damage to the CONEX box-like structure walls and transport shielding but not to the reactor coolant boundary.

7.2.4.1 BRA 4M Source Term

For BRA 4M, the MAR within the core is assumed to have a damage ratio of 0.05 (5 percent). 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-like structure and subsequent release to the environment.

For the MAR within the reactor containment boundary, a damage ratio of 0.3 (30 percent) 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 the reactor containment boundary.

7.2.4.2 BRA 4L Source Term

BRA 4L events are light-impact highway accidents that result in no release of radiological material but some degradation of transport shielding, and therefore, no source term for this accident was estimated.

7.2.5 Road Impact and Fire Accident Except with a Tanker Carrying Flammable Material – BRA 5H and BRA 5M 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 in the CONEX box-like structure 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.

7.2.5.1 BRA 5H Source Term

For BRA 5H, the MAR within the fuel is assumed to have a damage ratio of 0.001 (0.1 percent). 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 Section 4.4.1.1 of DOE-HDBK-3010-94 (*Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* [DOE 2013]) for air entrainment from fires associated with nonreactive material is applied. For the airborne release and respirable fractions, it was assumed that the diesel fuel quantity (the source of the fire) is limited to what may be carried in a 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-like structure 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 percent) 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 nonreactive material is applied. A leak path factor of 0.01 is assigned based on the pathway through the reactor containment boundary to the CONEX box-like structure and subsequent release to the environment.

For the MAR within the reactor containment boundary, a damage ratio of 0.3 (30 percent) 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 nonreactive material is 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.

7.2.5.2 BRA 5M Source Term

For BRA 5M, the MAR within the core is assumed to have a damage ratio of 0.05 (5 percent). 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 nonreactive material is applied. For the airborne release and respirable fractions for fire, it was assumed that the diesel fuel quantity (the source of the fire) is limited to what may be carried in the transport vehicle. A leak path factor of 0.01 is assigned based on the pathway through the reactor containment boundary to the CONEX box-like structure of the TNPP Package and subsequent release to the environment.

For the MAR within the reactor containment boundary, a damage ratio of 0.3 (30 percent) 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 nonreactive material is 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.

7.2.6 Collison with a Tanker Carrying Flammable Liquid and an 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 in the CONEX box-like structure walls and reactor coolant boundary as well as damage to portions of the reactor 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 percent) 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 effects, 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 nonreactive material is 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-like structure of the TNPP Package 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 percent) 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 effects, 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 nonreactive material is applied. A leak path factor of 0.1 is assigned based on the pathway through the reactor containment boundary to the CONEX box-like structure of the TNPP Package and subsequent release to the environment.

For the MAR within the reactor containment boundary, a damage ratio of 1 (100 percent) 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 effects, 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 nonreactive material is 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.

7.2.7 Loss of the Nonpressurized Reactor Containment Boundary – BRA 7 Source Term

This section describes the source term development for a nonpressurized 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 coolant boundary.

For the MAR within the reactor containment boundary, a damage ratio of 0.2 (20 percent) 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-like structure of the TNPP Package and then subsequent release to the environment.

This results in an overall release fraction of 2.0E-08 applied to the MAR in the reactor containment boundary.

7.2.8 Loss of the 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 by 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 percent) 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-like structure of the TNPP Package and then subsequent release to the environment.

This results in an overall release fraction of 8E-07 applied to the MAR in the reactor containment boundary.

7.3 Approach for Developing Estimates of Transportation Accident Consequences

This section discusses the approach for developing TNPP transportation package accident radiation dose consequences. It relies on the inventory information as described in Section 5.1.3 to define the radionuclides that are released from the transportation package and meet the dosimetry screening criteria, which make up the source term for the radiation dose calculations. The dose calculations are based on material that is released from the TNPP Package and any direct radiation dose from unreleased material within the package. The dose calculations are based on the methodology used by the IAEA Q system described in IAEA SSG-26 (IAEA 2022) as introduced in Section 4.2.1 of this report as the basis of the A_1 and A_2 values used in 10 CFR Part 71.[52] The specific methodology for calculating radiation 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 and special form and non-special form radioactive materials (IAEA 2022). The IAEA Q system methodology was chosen for the dose calculations for this activity based on its wide acceptance and adoption both in the United States transportation regulations as well as by the international community.

For the purposes of this dose assessment, the dose coefficients developed as a part of IAEA SSG-26 were used whenever possible to keep the methodology consistent with the IAEA Q system and dose methodology for development of the A_1 and A_2 values. However, the IAEA Q

[52] A_1 values are for special form (non-dispersible) radioactive material (i.e., sealed sources) and consider only the external photon dose and external beta dose pathways. Special form radioactive material must satisfy the requirements contained in 10 CFR 71.4. A_2 values are for non-special form (dispersible) radioactive material. Non-special form radioactive material is referred to as normal form radioactive material in U.S. regulations.

system dose methodology was developed for a single receptor typically addressed in this study as the worker, whereas a maximally exposed member of the public is also considered in this demonstration PRA. Therefore, exposure of the public to radiation had to be separately defined for the different radiation dose pathways for the maximally exposed individual assumed to be located 25 meters from the release. For the radiation dose associated with released material, the approach is refined to tailor the source term to the accident phenomena for the public.

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 are not included; ingestion is not included consistent with IAEA SSG-26 findings that explicit consideration of the ingestion pathway is unnecessary, and submersion dose is not included because it is assumed 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 from IAEA SSG-26 (IAEA 2022):

- 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 2022).

7.3.1 External Dose Due to Photons from Released Material

The external dose due to photons from released radiological material 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 of the shielding, it is a source of direct radiation dose. This calculation concerns external dose due to photons from released material.

The external dose coefficients used in this report to calculate dose from released radioactive material are from IAEA SSG-26 (IAEA 2022). For radiation dose from released material, this calculation does not account for dispersion, so it is likely overly conservative because it assumes that the receptor is 1 m away from any released material. The source terms developed for each bounding representative accident as described in Section 7.2 (using the five-factor equation presented in Section 7.1) are multiplied by 100 so that the amount of material used to calculate external dose is more reflective of the total material released rather than the amount of material that is respirable (accounting for the 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, according to the *Emergency Response Guidebook* (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 percent) for the 25 m distance.

7.3.2 External Dose Due to Photons from Unreleased Material

The external dose due to photons from unreleased radioactive inventory inside the reactor is determined by the Q system 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 TNPP Package. The exposure is caused by degradation of the transport shielding. The reactor vessel is judged to remain largely intact even after a transportation accident involving severe impact, but the transport shielding is assumed to be significantly degraded. Even in less than the highest impact crashes, it may be possible to create gaps where radiation streaming occurs. Accordingly, the calculation of external dose due to direct photon radiation from unreleased reactor inventory was determined for loss of the transport shielding (i.e., loss of the CONEX box-like structure and additional transport shielding provided on the walls for transport of the Reactor Module that define the TNPP Package).

Due to a lack of direct radiation calculation data from the vendor, a MicroShield® (version 9.06) model was developed and used when estimating exposure from the unshielded reactor vessel during accident scenarios that involve impact during a crash. Using the design plans provided by the vendor along with an expected radionuclide inventory, as listed in Appendix A, exposure data were calculated for various distances and decay times.

The MicroShield model used for the evaluation incorporated the top 99 percent of dose contributors from the expected nuclide inventory. The shielding layers included in the model were taken from the design plans provided by the vendor. The density of the reactor vessel core was adjusted to obtain similar exposure rates outside the vessel to those reported by the vendor.

This model was used to determine the exposure rates from the unshielded reactor vessel for distances between 1 m and 100 m from the vessel, with post shutdown times varying between 7 days and 2 years. For scenarios involving the degradation of shielding, exposures for workers and the public were estimated at 1 m for 30 minutes and 25 m for 30 minutes, respectively. For distances and times not provided directly by the model, an interpolated equation was used to estimate values.

It was assumed that all crash events could damage the transport shielding (i.e., additional shielding provided on the walls of the CONEX box-like structure for transport of the Reactor Module of the TNPP Package). Accordingly, the dose contribution from direct photon radiation from unreleased radiological material was added to all crash events (i.e., BRA 3, BRA 4M, BRA 4L, BRA 5H, BRA 5M, and BRA 6). Though BRA 4L is a light-impact event, it is defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or an impact with a light vehicle that is not severe (e.g., results in property damage only), and therefore, damage to the transport shielding was not ruled out. For BRA 4L, it was assumed that enough damage to the transport shielding occurred to create a gap in the wall of the CONEX box-like structure of the Reactor Module (e.g., damage to a corner of the TNPP Package) allowing radiation streaming to occur. Accordingly, there is some likelihood a worker or member of the public could be exposed to this stream for a meaningful period of time (i.e., 30 minutes was assumed) as described in Section 8.1.5. Last, although direct radiation for any material not released is clearly a dose contributor to BRA 9A, BRA 9B, and BRA 10, the dose consequences for these criticality events were not determined.

7.3.3 External Dose Due to Beta Radiation

The external dose due to beta radiation is evaluated based on the potential for beta dose to the skin of receptors from released radiological material. The IAEA SSG-26 methodology is for beta emitters that are unshielded, but it 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 is associated with materials, such as the beta window protector, package debris, etc., and is assumed to be a very conservative shielding factor of 3 for beta emitters of maximum energy (greater than or equal to 2 MeV) (IAEA 2022). 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 2022). 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 is likely overly conservative because it assumes that the receptor is 1 m away from any released contamination and the dose does not account for the dispersion of released material. The source term developed as described in Section 7.2 is multiplied by 100 so that the amount of material used to calculate external dose is 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.

7.3.4 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 2022). 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 is 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 as described in Section 7.2 of this report for each accident depending on the accident phenomena is assumed to be 1E-03 for a person standing within approximately 10 m from the release point. This uptake value represents the amount of material taken up into a human receptor after a release and is separate from the estimate of what material is released as described in Section 7.2. Inhalation doses for this effort are calculated using the inhalation dose coefficients found in Appendix I of IAEA SSG-26 (IAEA 2022).

Inhalation dose to a member of the public is estimated by scaling the uptake factor based on distance from the accident. The same exposure assumptions are used for the public as for the worker, except that the distance from release is assumed to be 25 m. While worker dose is assumed to be at 1 m from the accident site, a conservative 10 m uptake factor is used for inhalation dose. Within the 10 m distance, other factors that reduce the uptake come into effect and may become dominant. IAEA SSG-26 states the dose increases by a factor of 30 from 100 m to 10 m. A power function is 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 25 m is determined to be 7.5 times higher than the dose at 100 m. When comparing this to the worker dose at 10 m, which is 30 times higher, the ratio at 25 m is 25.8 percent of the dose at 10 m. The public inhalation dose is therefore about 20 percent of the worker inhalation dose.

7.3.5 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) are evaluated using the bases presented in Section 7.2 of this report, which are different for each bounding representative accident; 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 percent 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 7.2, 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, 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 of IAEA SSG-26 (IAEA 2022).

It is assumed that workers involved in handling radioactive material after an accident would wear the appropriate protective clothing including gloves. Therefore, even though skin contamination equivalent skin dose is calculated and shown in the radiation dose consequences results table in the next section (because it is used in the IAEA SSG-26 approach), this dose is not counted toward the risk results that are compared to risk evaluation guidelines.

7.3.6 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 is normally the limiting contributor to 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 a 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 three radionuclides identified in Table 5.1 in Section 5.1.4.1 would be excluded: Kr-85, Xe-131m, and Xe-133.

7.3.7 Exposure Pathways Not Addressed by the Q System

The exposure pathways used in the Q system to calculate the A_1 and A_2 values reported in IAEA SSG-26 were selected for evaluation in the TNPP transportation package risk assessment methodology because they were judged by the IAEA in SSG-26 to be the dominant pathways for the public and workers to be exposed to radiation as a result of transportation accidents involving radioactive materials. Furthermore, the development of the Q system specifically examined implications for activity release limits for Type B packages in the context of the transport of irradiated nuclear fuels. While other exposure pathways could be evaluated (e.g., external exposure to neutrons, resuspension, skyshine, drinking water ingestion, etc.), they are not expected to be significant exposure pathways for transportation accidents involving irradiated fuel. This is especially the case for pathways such as resuspension, skyshine, and drinking water ingestion that would be expected to be mitigated by emergency response to a transportation accident.

With specific regard to exposure to neutrons, SSG-26 (2018 Edition) states:

> In the case of neutron emitters, it was originally suggested under the Q system that there were no known situations with (α,n) or (γ,n) sources or the spontaneous neutron emitter Cf-252 for which neutron dose would contribute significantly to the external or internal radiation pathways considered earlier [I.4]. However, neutron dose cannot be neglected in the case of Cf-252 sources.

Neutron dose from Cf-252 sources (spontaneous fission) is now specifically considered in the development of the Q_A and A_1 values, but this is done to explicitly address the neutron dose risk associated with Cf-252 sources (special form material); the quantity of Cf-252 in irradiated fuel is insignificant, so it is a negligible contributor to dose.

Nevertheless, a bounding assessment was performed of the potential dose contribution from spontaneous neutron emitters present in the Project Pele irradiated fuel. This assessment accounted for the spontaneous fission neutrons emitted from the dominant spontaneous fission neutron sources (specifically, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, U-235, U-238, Am-241, Cm-242, Cm-244, Cf-252, and Np-237). The spontaneous fission neutron dose is dominated by Cm-242 and Cm-244 for short cooling times (less than 5 years), which is also consistent with studies of LWR irradiated fuel. The bounding dose contribution from spontaneous fission neutrons was determined to be less than 0.5 percent of the photon dose and would be even less if making less bounding but realistic yet conservative assumptions (e.g., the bounding analysis assumes all neutrons are at the peak dose conversion energy, while a realistic conservative assessment could use the full neutron energy distribution).

Because of the complexity of assessing the neutron emission rate for alpha, neutron (α,n) reactions, PNNL did not perform a separate evaluation of the neutron dose contribution from these reactions. However, significant evidence is provided in the literature that the neutron dose contribution from these reactions is less than 10 percent of the dose contribution from spontaneous fission neutrons for the time periods of interest (e.g., transportation of the TNPP Package within a few years after reactor shutdown). See, for example, https://publications.jrc.ec.europa.eu/repository/bitstream/JRC112361/report_eur_29301en.pdf. These results are for irradiated LWR uranium oxide fuel. The UCO TRISO fuel used in Project Pele, in addition to having significant quantities of oxygen, also has significant quantities of carbon. While the presence of carbon is not expected to significantly alter these results for LWR fuel, this difference is a source of uncertainty that may need to be addressed by an applicant using the risk-informed methodology for transporting irradiated UCO TRISO fuel.

PNNL also did not perform a separate evaluation of neutron startup sources incorporated into the Reactor Module. These sources could be included in future assessments.

7.3.8 Radionuclides Not Included in IAEA SSG-26

The screening methodology discussed in Section 5.1.3 and the MAR identification discussed in Section 5.1.4.1 result in the radionuclides in the release source term that are included in the dosimetry assessment identified in Table 5.1. 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 used in this dose evaluation (IAEA 2022). Some radionuclides included in the screened list of radionuclides do not have dose coefficients in IAEA SSG-26; these radionuclides are listed in Table 7.5. 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)[53] 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 7.5. Radionuclides Included in the Dosimetry Source Term That 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

7.4 Assumptions Made for the Accident Consequence Analyses

This section presents a list of assumptions used in the calculations to determine the transportation accident source terms and the radiation 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 5.3.2.2 and Section 6.4). The following assumptions are for the baseline case:

1. The reactor core has decayed for 90 days after 3 years of operation.
2. The portion of the primary reactor cooling system transported contains all the condensed or plated-out radioactive material (e.g., the released fission products and condensed gases in this system have not been removed before transport).

[53] Tritium was considered separately in SSG-26, see para. I.59, page 309.

3. A radioactive material cleanup system and/or the resulting radioactive waste material is not transported with the microreactor.
4. The direct radiation dose is estimated using a MicroShield model developed by PNNL. The model is based on design information from the vendor about the reactor (i.e., the types and thicknesses of different layers of material) and the thickness of the CONEX box-like structure and transport shielding. The model and results were benchmarked against the results of preliminary transport radiation results from the vendor for a design in which the radiation limits at the outside of the CONEX box-like structure were met.
5. The baseline case release fractions from diffusion occurring during normal operations leading to material residing in the core, reactor structure, and coolant system as well as the source term factors described in Section 7.1 represent best judgment but conservative estimates.
6. For BRA 4L events, which are light-impact highway accidents, it is assumed that there is no release of radiological material. These light-impact accidents are defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or impact with a light vehicle that is not severe (e.g., results in property damage only). However, it was assumed that enough damage to the transport shielding occurred to create a gap in the CONEX box-like structure of the Reactor Module (e.g., damage to a corner) allowing radiation streaming to occur. It was further assumed that there was a 10 percent chance that a worker could be located in the radiation stream for 30 minutes.
7. The radiation dose pathways and the transportation accident consequence analysis of these pathways are generally based on the IAEA SSG-26 (IAEA 2022) methodology and assumptions made for assessing package performance, such as the 30 min assumed duration time for exposure to the damaged TNPP Package.
8. A deviation from the approach identified in assumption 7 above is that the uptake by a human receptor is derived in consideration of the accident phenomena, as described in Section 7.2. This is done for each bounding representative accident using the five factors associated with defining the source term, as described in Section 7.1. This assumption provides a more refined estimate than the guidance in IAEA SSG-26 (IAEA 2022) because it tailors the source term to the accident phenomena. The guidance in IAEA SSG-26 is simply based on the assumption that the radioactive material intake by a “bystander” is 0.1 percent of the released material.
9. As a follow-on to assumption 8 above, use of accident-specific source term factors, required use of best but conservative judgment to apply factors developed specifically for a different use (i.e., specifically for fuel cycle and nonreactor nuclear facilities.
10. Another deviation from the approach identified in assumption 7 above is that the TNPP PRA approach considers two receptors—a worker and a member of the public—whereas SSG-26 does not differentiate between receptors. The distance between the worker and the point of the release is assumed to be 1 m (except for the inhalation dose pathway for which the distance is assumed to be 10 m), which is equivalent to the approach described in SSG-26. A member of the public is assumed to be located 25 m from the point of the release because that is the protective action distance for high-level radiological material emergency response (DOT 2020).
11. The dose contribution from ingestion submersion is assumed to be negligible, and therefore is 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, which significantly limits the time that a receptor might stand in a gaseous cloud of radionuclides.

12. It is assumed that workers involved in handling radioactive material after an accident would wear the appropriate protective clothing including gloves. Therefore, even though skin contamination equivalent skin dose is calculated and shown in the radiation dose consequences results table in the next section (because it is used in the IAEA SSG-26 approach), it is not counted toward the risk results that are compared to risk evaluation guidelines.

7.5 Dose Consequences for Each of the Bounding Representative Accidents

This section summarizes the radiation 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
3. The radiological material that condensed or plated-out in the reactor cooling boundary during operation.

As discussed in Section 7.3, the dose pathways addressed for released radiological material are from external photon radiation, inhalation, external beta radiation to the skin, and skin contamination from handling radiological debris. The external beta radiation to the skin and skin contamination from handling debris are not applicable to a member of the public. Also included is the radiation dose pathway from unreleased radiological material caused by loss of external shielding (the CONEX box-like structure and transport shielding on the walls of the CONEX box) in accidents that involve a crash (i.e., BRA 3, BRA 4M, BRA 5H, BRA 5M, and BRA 6). Table 7.6 summarizes this information and provides the total TEDE for the worker and the public for each bounding representative accident in the last two columns.

The results presented in Table 7.6 show that the inhalation pathway dominates the effective dose for the bounding representative accidents for which it is a contributor. For workers, the external photon dose from unreleased material (from degraded transport shielding) and the external beta dose from released material are the next highest contributors. For the public (compared to the worker), inhalation dose is a more dominant radiation dose pathway than the dose from other dose pathways.

Table 7.6 shows that the radiation dose consequences from the TRISO fuel itself dominate the results for the accident for which it is a contributor compared to radiological material diffused into the reactor core internals or plated-out in the primary system. The next biggest contributor to radiation dose consequences is from radiological material that diffused into the reactor core internals during operation and is released during a transportation accident.

Table 7.6. Dose from Bounding Representative Accidents by MAR Contributions and Dose Pathways (2 sheets total)

Bounding Representative Accident	Dose Pathways for Released Material								Dose Pathway for Unreleased Material		All Dose Pathways	
	Photon External Effective Dose from Released Material (rem)		Inhalation Effective Dose (rem)		Beta External Effective Skin Dose (rem)[a]	Skin Contamination Equivalent Skin Dose (rem)[b]	Total Effective Dose Equivalent from Released Material (rem)		Photon External Effective Dose from Unreleased Material (rem)		Total Effective Dose Equivalent from Released and Unreleased (rem)	
MAR Contributors	Worker	Public	Worker	Public	Worker	Worker	Worker	Public	Worker	Public	Worker	Public
BRA 1 – Fire Only that Originates Inside CONEX Box-Like Structure												
Total	0	0	0	0	0	0	0	0	NA[c]	NA[c]	0	0
BRA 2 – Fire Only that Originates Outside CONEX Box-Like Structure												
TRISO Fuel	0	0	0	0	0	0	0	0	NA[c]	NA[c]	**2.3E-03**	**5.1E-04**
Reactor Core	1.1E-05	1.7E-08	1.0E-03	2.6E-04	2.2E-5	2.3E-03	1.0E-03	2.6E-04				
Coolant Boundary	2.5E-04	3.9E-07	9.7E-04	2.5E-04	2.4E-5	5.5E-03	1.2E-03	2.5E-04				
Totals	2.6E-04	4.1E-07	2.0E-03	5.1E-04	4.6E-5	7.9E-03	2.3E-03	5.1E-04				
BRA-3 – Hard-Impact Road Accident												
TRISO Fuel	4.0	6.4E-03	71.7	18.5	5.2	1530	80.9	18.5	6.0	6.9E-02	**87.7**	**18.8**
Reactor Core	5.5E-03	8.7E-06	5.0E-01	1.3E-01	1.1E-02	1.2	5.2E-01	1.3E-01				
Coolant Boundary	6.1E-02	9.8E-05	2.4E-01	6.3E-02	5.9E-03	1.4	3.1E-01	6.3E-02				
Totals	4.1	6.5E-02	72.4	18.7	5.2	1540[d]	81.7	18.7	6.0	6.9E-02		
BRA 4 – Less than Hard-Impact Road Accident												
BRA 4M – Medium-Impact Road Accident												
TRISO Fuel	0	0	0	0	0	0	0	0	6.0	6.9E-02	**6.0**	**7.7E-02**
Reactor Core	2.7E-04	4.4E-07	2.5E-02	6.5E-03	5.5E-04	5.8E-02	2.6E-02	6.5E-03				
Coolant Boundary	1.8E-03	3.0E-06	7.3E-03	1.9E-03	1.8E-04	4.2E-02	9.3E-03	1.9E-03				
Totals	2.1E-03	3.4E-06	3.2E-02	8.4E-03	7.3E-04	1.0E-01	3.5E-02	8.4E-03	6.0	6.9E-02		
BRA 4L – Light-Impact Road Accident												
TRISO Fuel	0	0	0	0	0	0	0	0	6.0	6.9E-02	**6.0**	**6.9E-02**
Reactor Core	0	0	0	0	0	0	0	0				
Coolant Boundary	0	0	0	0	0	0	0	0				
Total	0	0	0	0	0	0	0	0	6.0	6.9E-02		
BRA 5 – Road Impact and Fire Accident Except with Tanker Carrying Flammable Material												
BRA 5H – Hard-Impact Accident and Ensuing Fire												
TRISO Fuel	4.0	6.4E-03	71.7	18.5	5.2	1530	80.9	18.5	6.0	6.9E-02	**87.8**	**18.8**
Reactor Core	5.5E-03	8.8E-06	5.0E-01	1.3E-01	1.1E-02	1.2	5.2E-01	1.3E-01				
Coolant Boundary	6.2E-02	9.9E-05	2.5E-01	6.3E-02	6.0E-03	1.4	3.1E-01	6.3E-02				
Totals	4.1	6.5E-02	72.5	18.7	5.2	1540[d]	81.7	18.7	6.0	6.9E-02		
BRA 5M – Medium-Impact Accident and Ensuing Fire												
TRISO Fuel	0	0	0	0	0	0	0	0	6.0	6.9E-02	**6.0**	**7.7E-02**
Reactor Core	2.8E-04	4.4E-07	2.5E-02	6.5E-03	5.6E-04	5.9E-02	2.6E-02	6.5E-03				
Coolant Boundary	1.9E-03	3.0E-06	7.3E-03	1.9E-03	1.8E-04	4.2E-02	9.4E-03	1.9E-03				
Totals	2.1E-03	3.4E-06	3.3E-02	8.4E-03	7.4E-04	1.0E-01	3.5E-02	8.4E-03	6.0	6.9E-02		

Bounding Representative Accident	Dose Pathways for Released Material								Dose Pathway for Unreleased Material		All Dose Pathways	
	Photon External Effective Dose from Released Material (rem)		Inhalation Effective Dose (rem)		Beta External Effective Skin Dose (rem)[a]	Skin Contamination Equivalent Skin Dose (rem)[b]	Total Effective Dose Equivalent from Released Material (rem)		Photon External Effective Dose from Unreleased Material (rem)		Total Effective Dose Equivalent from Released and Unreleased (rem)	
MAR Contributors	Worker	Public	Worker	Public	Worker	Worker	Worker	Public	Worker	Public	Worker	Public
BRA 6 – Collision with a Tanker Carrying Flammable Material and Ensuing Fire												
TRISO Fuel	4.0	6.4E-03	72.5	18.7	5.2	1530	81.7	18.7	6.0	6.9E-02	**88.7**	**18.9**
Reactor Core	6.5E-03	1.1E-05	6.0E-01	1.6E-01	1.3E-02	1.4	6.2E-01	1.6E-03				
Coolant Boundary	7.4E-02	1.2E-04	2.9E-01	7.1E-03	7.1E-03	1.7	3.7E-01	7.5E-02				
Totals	4.1	6.6E-03	73.4	19.0	5.2	1540[d]	82.7	18.9	6.0	6.9E-02		
BRA 7 – Loss of Nonpressurized Reactor Containment Boundary												
TRISO Fuel	0	0	0	0	0	0	0	0	NA[c]	NA[c]	**4.1E-05**	**8.4E-06**
Reactor Core	0	0	0	0	0	0	0	0				
Coolant Boundary	8.2E-06	1.3E-08	3.2E-05	8.4E-06	7.9E-07	1.8E-04	4.1E-05	8.4E-06				
Totals	8.2E-06	1.3E-08	3.2E-05	8.4E-06	7.9E-07	1.8E-04	4.1E-05	8.4E-06				
BRA 8 – Loss of Pressurized Reactor Containment Boundary												
TRISO Fuel	0	0	0	0	0	0	0	0	NA[c]	NA[c]	**1.7E-03**	**3.4E-04**
Reactor Core	0	0	0	0	0	0	0	0				
Coolant Boundary	3.3E-04	5.2E-07	1.3E-03	3.4E-04	3.2E-05	7.4E-03	1.7E-03	3.4E-04				
Totals	3.3E-04	5.2E-07	1.3E-03	3.4E-04	3.2E-05	7.4E-03	1.7E-03	3.4E-04				
BRA 9A – Criticality Event Involving Drop into a Body of Water												
No radiation dose calculated because the accident meets the risk evaluation guidelines based on its exceptionally low likelihood.												
BRA 9A – Criticality Event Involving Inundation with Fire Suppression Water												
No radiation dose was calculated because the accident meets the risk evaluation guidelines based on its exceptionally low likelihood.												
BRA 10 – Criticality Event Caused by Control Rod Withdrawal												
The consequence of the accident has not been developed because this study assumes that the design goal of precluding a reactivity insertion event in a TNPP transportation accident will be met.												

BRA = bounding representative accident; CONEX = container express; MAR = material at risk; NA = not available; rem = roentgen equivalent man; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).

(a) 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.

(b) Though skin contamination equivalent from ungloved work with debris is a radiation 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.

(c) Loss of transport shielding does not occur for non-crash accidents, so the contribution from photon radiation from unreleased material does not apply.

(d) Small inaccuracy in the total for this column due to difference between a rounded-off numerical values and the actual values.

Table 7.6 also shows that the radiation dose consequences from loss of containment accidents, not caused by highway accidents, are very low whether the containment boundary is assumed to be pressurized or not (see BRA 7 and BRA 8). The results show that the radiation dose consequences from fire events, not caused by highway accidents, are very low whether the fire originates from inside or outside of the CONEX box-like structure (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 consequences 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. The dose consequences of the same two bounding representative accidents compared to the radiation dose consequences from BRA 6 (a collision with a tanker carrying flammable material) are only slightly greater for the tanker collision. A collision with a tanker carrying flammable material results in the largest fire that can be postulated. Accordingly, it can be concluded that fire as a radiological release mechanism is not nearly as dominant a factor as mechanical impact.

The skin contamination equivalent skin dose presented in the fifth column is for information only; it is not converted to effective dose or assumed to be a contributor to the TEDE for the accident. Accordingly, the proposed risk evaluation guidelines do not specifically consider the radiation dose from skin contamination as result of handling a damaged TNPP Package. Consideration of this dose contribution comes from 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). However, even though skin contamination equivalent from ungloved work with debris is a radiation 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.

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 of which contribute 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.

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 8.0.

8.0 PRA Baseline Risk Results and Evaluation

This section provides summaries of radiological risk for each bounding accident and a comparison to the risk acceptance guidelines presented in Table 4.7 in this report. In the following: Section 8.1 presents the worker and public risk for each bounding representative accident and compares them to the proposed risk acceptance guidelines. Section 8.2 presents an overview of the risk results for the bounding representatives in one table.

8.1 Risk Comparisons to Proposed Risk Evaluation Guidelines

This section presents risk summary tables for each bounding representative accident that provide:

- The estimated frequency of the accident based on one transport in a year using the assumed route
- The estimated radiation dose consequences of the accident in rem (TEDE
- The risk limit from the proposed risk evaluation guidelines in terms of likelihood and consequence.

The radiation 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
3. The radiological material that condensed or plated-out in the primary reactor cooling system during operation.

The proposed evaluation guidelines from Table 4.7 in this report are presented in the last column for each bounding representative accident table along with the applicable accident likelihood and consequence limits for comparison. The last column of the table also presents the acceptability of the risk results based on the comparison. Note, these risk results and comparisons to the risk evaluation guidelines are based on one ship made in a year.

8.1.1 Fire Only that Originates Inside the CONEX Box-Like Structure – BRA 1 Risk Results

BRA 1 is a fire that originates inside the CONEX box-like structure of the TNPP Package. This is a general fire that originates from sources such as an electrical cable fault, propagates into the package, and ignites combustible material internal to the TNPP Package. It includes an oil or grease fire that is ignited by 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 RPV vessel integrated shielding and containment 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. There are no electrically active systems inside the reactor, so fire initiated inside the reactor is not expected during transport. Therefore, the radiation dose consequence for this bounding representative accident was determined to be 0 rem without performing consequence analysis, as discussed in Section 7.2.1 and presented in Table 7.6.

The estimated frequency of this event is 7.0E-06 per year assuming one trip in a year based on likelihood determination discussed in Section 6.5.1 and presented in Table 6.16. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable even without comparing the risk to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report as presented in Table 8.2.

Table 8.1. Risk Results Comparison for BRA 1 – Fire Only that Originates Inside the CONEX Box-Like Structure

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥5 and <25 rem TEDE for a member of the public ≥25 and <100 rem TEDE for a worker when the accident frequency is ≤1E-05 and >1E-06
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	0	0		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Radiation dose	**0**	**0**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**7.0E-06**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

8.1.2 Fire Only that Originates Outside the CONEX Box-Like Structure – BRA 2 Risk Results

BRA 2 is a diesel fuel fire that originates outside the Reactor Module, propagates into the CONEX box-like structure of the TNPP Package, and ignites combustible material which damages the reactor containment boundary. The quantity of diesel fuel assumed should is limited to full transporter fuel tanks (e.g., 300 gallons). The estimated frequency of this accident as presented in Table 6.16 is 3.4E-06 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.2. The radiation dose consequence for this bounding representative accident was determined to be very low to the worker (2.3E-03 rem) and the public (5.1E-04 rem) based on dose-consequence analysis discussed in Section 7.2.2 and presented in Table 7.6. The radiation dose limits at an accident frequency of 3.4E-06 per year meet the risk evaluation guidelines. 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 4.7 of this report as presented in Table 8.2.

Table 8.2. Risk Results Comparison for BRA 2 – Fire Only that Originates Outside the CONEX Box-Like Structure

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥5 and <25 rem TEDE_for a member of the public ≧25 and <100 rem TEDE for a worker when the accident frequency is ≤1E-05 and >1E-06
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	1.0E-03	2.6E-04		
Cooling System	1.2E-03	2.5E-04		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Radiation dose	**2.3E-03**	**5.1E-04**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**3.4E-06**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

8.1.3 Hard-Impact Road Accident – BRA 3 Risk Results

BRA 3 is a hard-impact accident that includes impact with heavy vehicles and solid unyielding objects (e.g., concrete abutment or a rock embankment), drops to a lower elevation (e.g., drop from a bridge), and rollovers which can result in hard impact on the asphalt or concrete roadway. The estimated frequency of this accident as presented in Table 6.16 is estimated to be 7.1E-05 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.3. The radiation dose consequence for this bounding representative accident is determined to be 87.7 rem to the worker and 18.8 rem to the public based on dose-consequence analysis discussed in Section 7.2.3 and presented in Table 7.6. The radiation dose is based on (1) direct radiation from unreleased material caused by degraded shielding (i.e., loss of transport shielding) caused by the crash, and (2) a release of radiological material from the TRISO fuel, radiological material diffused into the core structure such as the compacts and radiological material condensed or plated-out in the reactor cooling boundary. Section 7.3.2 explains that due to lack of direct radiation calculations from the vendor a MicroShield model was developed by the authors to estimate the impact from this radiation dose pathway. The greatest contribution to the dose, by far, was the contribution of release from the TRISO fuel itself. The table shows that the estimated radiation dose from this bounding representative accident exceeds the risk evaluation guidelines for the public of 5 rem and the worker of 25 rem for an accident frequency of 7.1E-05 per year. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be unacceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report without controls and/or compensatory measures as presented in Table 8.3.

Table 8.3. 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	80.9	18.5		
Core Structure	5.2E-01	1.3E-01		
Cooling System	3.1E-01	6.3E-02		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**87.7**	**18.8**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**7.1E-05**	
COMPARISON TO RISK EVALUATION GUIDELINE				Unacceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

8.1.4 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 of shielding. These medium-impact accidents are defined as severe collisions with a light vehicle leading to fatality or injury. The estimated frequency of this accident as presented in Table 6.16 is estimated to be 9.7E-05 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.4.1. The radiation dose consequence for this bounding representative accident is determined to be moderate for the worker (6 rem) and low for the public (7.7E-02 rem) based on dose-consequence analysis described in Section 7.2.4.1 and presented in Table 7.6. The radiation dose is based on:

1. Direct radiation from unreleased material caused by degraded shielding (i.e., loss of transport shielding) caused by the crash
2. A release of radiological material diffused into the core structure such as the compacts and radiological material condensed or plated-out in the reactor coolant boundary.

No release from the TRISO fuel is postulated. Section 7.3.2 explains that due to lack of direct radiation calculations from the vendor a MicroShield model was developed by the authors to estimate the impact from this radiation dose pathway. The total radiation dose for BRA 4M meets the proposed risk evaluation guidelines given its frequency of 9.7E-05 per year. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report as presented in Table 8.4. Uncertainty in the allocation of events considered to be light-impact accidents (BRA 4L) versus medium-impact accident (BRA 4M) event could affect the conclusion about risk for BRA 4M. The uncertainty analysis presented in Section 10.2.2 shows that the uncertainty associated with the estimated accident frequency can impact the risk conclusion about BRA 4M.

Table 8.4. 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 4.7 of this Report
Accident Consequence by MAR contribution (Radiation dose from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	6.5E-03		
Cooling System	9.3E-03	1.9E-03		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**6.0**	**7.7E-02**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**9.7E-05**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

8.1.5 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 but some degradation of transport shielding. These light-impact accidents are defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or an impact with a light vehicle that is not severe (e.g., results in property damage only). The estimated frequency of this accident as presented in Table 6.16 is 3.3E-05 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.4.2. BRA 4L was determined to result in no release of radiological material but does result in some degradation of the transport shielding. Per the radiation dose consequence results discussed in Section 7.2.4.2 and presented in Table 7.6, the total dose of 6 rem to the worker and 6.9E-02 rem to the public meets the proposed risk evaluation guidelines at a frequency of 3.3E-05 per year. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable as presented in Table 8.5 below. Section 7.3.2 explains that due to lack of direct radiation calculations from the vendor, a MicroShield model was developed by the authors to estimate the impact from this radiation dose pathway. It was assumed for BRA 4L that enough damage to the transport shielding occurred to create a gap or fissure in the CONEX box-like structure and transport shielding of the Reactor Module (e.g., damage to a corner of the TNPP Package) allowing radiation streaming to occur. It was further conservatively assumed that there was 10 percent chance that a worker or member of the public could be located in the radiation stream for the duration of 30 minutes. This last assumption has the effect of reducing the accident frequency which is reflected in Table 8.5.

Table 8.5. 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	0	0		
Total Dose	**0**	**0**		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**6.0**	**6.9E-02**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**3.3E-05**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable

MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).

8.1.6 Hard-Impact Accident and an 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 a loss of shielding. Hard-impact accidents are defined to be heavy vehicle collisions, impacts with unyielding 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 6.16 is estimated to be 2.6E-08 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.5.1. The radiation dose consequence for this bounding representative accident is determined to be 87.8 rem to the worker and 18.8 rem to the public based on dose-consequence analysis as discussed in Section 7.2.5.1 and presented in Table 7.6. The greatest contribution to the dose, by far, was the contribution of release from the TRISO fuel itself. The radiation dose limits at an accident frequency of 2.6E-08 per year meet the risk evaluation guidelines. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report as presented in Table 8.6.

Table 8.6. Risk Results Comparison for BRA 5H – Hard-Impact Accident and an Ensuing Fire

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker when the accident frequency is ≤5E-07
MAR contribution from released material				
TRISO Fuel	80.9	18.5		
Core Structure	5.2E-01	1.3E-01		
Cooling System	3.1E-01	6.3E-02		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**87.8**	**18.8**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**2.6E-08**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

8.1.7 Medium-Impact Accident and an 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 a loss of shielding. These medium-impact accidents are defined as a severe collision with a light vehicle that leads to fatality or injury. The estimated frequency of this accident as presented in Table 6.16 is estimated to be 5.9E-07 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.5.2. The radiation dose consequence for this bounding representative accident is determined to be low (6 rem to the worker and 7.7E-02 rem to the public) based on dose-consequence analysis as discussed in Section 7.2.5.2 and presented in Table 7.6. The radiation dose is based on release of radiological material diffused into core structure such as the compacts and radiological material condensed or plated-out in the reactor coolant boundary. No release from the TRISO fuel is postulated. The radiation dose limits at an accident frequency of 5.9E-07 per year meet the risk evaluation guidelines. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report as presented in Table 8.7.

Table 8.7. Risk Results Comparison for BRA 5M – 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥25 and <750 rem TEDE for a member of the public ≥100 and <750 rem TEDE for a worker when the accident frequency is ≤1E-06 and >5E-07
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	6.5E-03		
Cooling System	9.4E-03	1.9E-03		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**6.0**	**7.7E-02**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**5.9E-07**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

8.1.8 Collision with a Tanker Carrying Flammable Material and an 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 6.16 is estimated to be 7.1E-08 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.6. The radiation dose consequence of this bounding representative accident is determined to be 88.7 rem to the worker and 18.9 rem to the public based on dose-consequence analysis, as discussed in Section 7.2.6 and presented in Table 7.6. The greatest contribution to the dose, by far, was the contribution of release from the TRISO fuel itself. The radiation dose limits at an accident frequency of 7.1E-08 per year meet the risk evaluation guidelines. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report as presented in Table 8.8.

Table 8.8. 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker when the accident frequency is ≤5E-07
MAR contribution from released material				
TRISO Fuel	81.7	18.7		
Core Structure	6.2E-01	1.6E-03		
Cooling System	3.7E-1	7.5E-02		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**88.7**	**18.9**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**7.1E-08**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

8.1.9 Loss of the Nonpressurized Reactor Containment Boundary – BRA 7 Risk Results

BRA 7 is a nonpressurized 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 6.16 is estimated to be 1.3E-03 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.7. The radiation dose consequence for this bounding representative accident was determined to be very low to the worker (4.1E-05 rem) and the public (8.4E-06 rem) based on dose-consequence analysis as discussed in Section 7.2.7 and presented in Table 7.6. The radiation dose limits at an accident frequency of 1.3E-03 per year meet the risk evaluation guidelines. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report as presented in Table 8.9.

Table 8.9. Risk Results Comparison for BRA 7 – Loss of the Nonpressurized Reactor Containment Boundary

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≤0.1 rem TEDE for a member of the public ≤2 rem TEDE for a worker when the accident frequency is >1E-03
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	4.1E-05	8.4E-06		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Dose	**4.1E-05**	**8.4E-06**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**1.3E-03**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable

MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).

8.1.10 Loss of the 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 6.16 is estimated to be 1.3E-03 per year assuming one trip in a year based on accident data and the likelihood determination presented in Section 6.5.8. The radiation dose consequence for this bounding representative accident was determined to be very low to the worker (1.6E-03 rem) and to the public (3.4E-04 rem) based on the dose-consequence analysis as discussed in Section 7.2.7 and presented in Table 7.6. The radiation dose limits at an accident frequency of 1.3E-03 per year meet the risk evaluation guidelines. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report as presented in Table 8.10.

Table 8.10. Risk Results Comparison for BRA 8 – Loss of the 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≤0.1 rem TEDE for a member of the public ≤2 rem TEDE for a worker when the accident frequency is >1E-03
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	1.6E-03	3.4E-04		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Dose	**1.6E-03**	**3.4E-04**		
Accident Frequency assuming one trip in a year (from Table 6.16)			**1.3E-03**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

8.1.11 Criticality Event Involving a Drop into a Body of Water – BRA 9A Risk Results

BRA 9A is a highway accident that leads to a drop of the TNPP Package into a body of water that results in a criticality caused by the addition of moderator and a possible change in core geometry. The estimated frequency of this bounding representative accident is <5E-07 per year, as described in Section 6.3.1.2, concluded in Section 6.5.9, and presented in Table 6.16, based on a set of analyses presented in Section 6.1.3 and Table 6.12.

The radiation dose consequence for this bounding representative accident was not calculated because the risk of this accident meets risk evaluation guidelines based on its exceptionally low estimated accident without calculating the dose consequences. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report based on the estimated accident frequency of <5E-07 per year as presented in Table 8.11.

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Table 8.11. Risk Results the Comparison for BRA 9A – Criticality Event Involving a 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 4.7 of this Report
Accident Consequence not determined because of the exceptionally low accident frequency				≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker ≤5E-07
MAR contribution from released material				
TRISO Fuel	—	—		
Core Structure	—	—		
Cooling System	—	—		
Total Dose	(a)	(a)		
Accident Frequency assuming one trip in a year (from Table 6.16)			**<5E-07**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable

MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).
(a) For this report, the dose consequences of flooded criticality accident were judged to be unacceptable without performing a consequence analysis for reasons discussed above.

8.1.12 Criticality Event Involving Inundation with Fire Suppression Water – BRA 9B Risk Results

BRA 9B is addition of moderator and a possible change in core geometry caused by a crash that results in RPV damage, fire, and inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality. The estimated frequency of this bounding representative accident is <5E-07 per year, as described in Section 6.3.1.2, concluded in Section 6.5.10, and presented in Table 6.16.

The radiation dose consequence for this bounding representative accident was not calculated because the risk of this accident meets risk evaluation guidelines based on its exceptionally low estimated accident without calculating the dose consequences. Accordingly, the risk of this TNPP Package transportation bounding representative accident is determined to be acceptable when compared to the applicable proposed risk evaluation guidelines presented in Table 4.7 of this report, based on the estimated accident frequency of <5E-07 per year as presented in Table 8.12.

Table 8.12. Risk Results the Comparison for BRA 9B – Criticality Event Involving Inundation with Fire Suppression Water

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence not determined because the low accident frequency and the complexity of such an analysis				≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker ≤5E-07
MAR contribution from released material				
TRISO Fuel	—	—		
Core Structure	—	—		
Cooling System	—	—		
Total Dose	(a)	(a)		
Accident Frequency assuming one trip in a year (from Table 6.16)			**<5E-07**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable

MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).

(a) For this report, the dose consequences of flooded criticality accident were judged to be unacceptable without performing a consequence analysis for reasons discussed above.

8.1.13 Criticality Event Caused by Control Rod Withdrawal – BRA 10 Risk Results

BRA 10 is a reactivity insertion event (e.g., control rod withdrawal) caused by the impact energy of an accident that results in a TNPP Package criticality event. The frequency and consequence of the accident have not been developed because it is assumed in this study that the design goal of precluding a non-flooding reactivity insertion event in a transportation accident involving a TNPP Package will be met. If the design does not preclude criticality in the event of a severe accident, then the risk of this accident would need to be assessed.

8.2 Summary of Risk Results for the Bounding Representative Accidents

Table 8.13 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 4.7 of this report. As previously explained, the accident frequency is presented on a per-year basis assuming one transport occurs in a year. As shown, the only bounding representative accident that does not meet the proposed risk evaluation guidelines is BRA 3.

Table 8.13. Risk Summary of the Bounding Representative Accidents (2 sheets total)

ID	Descriptions	Accident Frequency per Year[(a)]	Radiation Dose Consequences		Meets Proposed Risk Evaluation Guidelines
			Worker (rem TEDE)	Public (rem TEDE)	
BRA 1	Fire-only event that originates inside the CONEX box-like structure of the TNPP Package.	7.0E-06	0	0	Acceptable
BRA 2	Diesel fuel fire-only event that originates outside the Reactor Module and propagates into the CONEX box-like structure and ignites combustible material, which damages the reactor containment boundary and impacts containment capacity.	3.4E-06	**2.3E-03**	**5.1E-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 a lower elevation, or rollovers.	7.1E-05	**87.7**	**18.8**	Unacceptable[(b)]
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 or injury).	9.7E-05	**6.0**	**7.7E-02**	Acceptable
BRA 4L	Less than a hard-impact highway accident that results in no release of radiological material but some degradation in transport shielding. Light impact such as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or impact with a light vehicle that is not severe (e.g., results in property damage only).	3.3E-05	**6.0**	**6.9E-02**	Acceptable
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	**87.8**	**18.8**	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	**6.0**	**7.7E-02**	Acceptable
BRA 6	Collision with a tanker carrying flammable material that leads to fire.	7.1E-08	**88.7**	**18.9**	Acceptable

ID	Descriptions	Accident Frequency per Year[(a)]	Radiation Dose Consequences		Meets Proposed Risk Evaluation Guidelines
			Worker (rem TEDE)	Public (rem TEDE)	
BRA 7	Loss of the nonpressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.	1.3E-03	**4.1E-05**	**8.4E-06**	Acceptable
BRA 8	Loss of the pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.	1.3E-03	**1.7E-03**	**3.4E-04**	Acceptable
BRA 9A	Addition of moderator and possible change in core geometry caused by a drop into body of water that results in criticality.	<5E-07	(c)	(c)	Acceptable
BRA 9B	Addition of moderator and possible change in core geometry caused by a crash that results in RPV damage, fire, and inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality.	<5E-07	(d)	(d)	Acceptable
BRA 10	Control rod withdrawal (or other reactivity insertion event) caused by impact from a road accident that results in criticality.	(e)	(e)	(e)	(e)

BRA = bounding representative accident; CONEX = container express; ID = identification; MAR = material at risk; rem = roentgen equivalent man; RPV = Reactor Pressure Vessel; TEDE = total effective dose equivalent; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).

(a) It is assumed that one transport occurs in a year.
(b) Risk is considered unacceptable without application of compensatory measures
(c) Assumed to be acceptable without performing a radiation dose consequences analysis based on the estimated accident frequency as discussed in Section 8.1.11, though per the proposed risk evaluation guidance in Table 4.7 there should be confirmation that there are no cliff edge effects.
(d) Assumed to be acceptable without performing a radiation dose consequences analysis based on the estimated accident frequency, as discussed in Section 8.1.12, though per the proposed risk evaluation guidance in Table 4.7 there should be confirmation that there are no cliff edge effects.
(e) The consequence and likelihood of this accident have not been developed because it is understood that the design will eventually preclude the possibility of this TNPP transportation accident.

9.0 Sensitivity Study on PRA Modeling Inputs

This section addresses the impact of the uncertainty of PRA modeling assumptions and inputs using sensitivity studies. Sensitivity studies were performed by varying PRA input parameters and then determining the effects of the changes on the risk results. Candidate sensitivity studies were identified by examining key TNPP Package transportation PRA modeling assumptions and other factors that could potentially affect the risk conclusions.

For sensitivity studies that were performed, radiation dose consequences and accident frequencies were updated for the applicable bounding representative accidents from the baseline case and compared to the proposed risk evaluation guidelines. From these results, insights were generated about the importance of the sources of PRA modeling uncertainty relative to the risk conclusions. Section 9.1 describes first how candidate sensitivity studies were identified and then defines the studies that were performed. Section 9.2 presents and evaluates the results of the sensitivity studies against the risk evaluation guidelines. Section 9.3 presents a summary of lessons learned by performing the sensitivity studies.

9.1 Identification and Definition of Sensitivity Studies

Selection and definition of candidate sensitivity cases to be performed were based on evaluation of (1) specific lists of assumptions and bases used in the TNPP Package transportation demonstration PRA and (2) the compensatory measures listed by the vendor to reduce or mitigate risk. This process was performed in two steps.

In the first step, a screening process was performed by evaluating each listed PRA modeling assumption and each compensatory measure listed by the vendor for its potential impact on the TNPP Package transportation PRA risk results. Qualitative evaluation was performed using the quantitative results from the baseline PRA results to screen assumptions or bases from further consideration or identify it as a candidate sensitivity study. In the second step, an examination and analysis of the assumptions, bases, and measures not screened out in the first step was performed to define feasible sensitivity studies. Section 9.1.1 describes the screening step and Section 9.1.2 describes the delineation of feasible sensitivity studies.

9.1.1 Screening for Candidate Sensitivity Studies

The screening process evaluated each of the PRA modeling assumptions listed in the report and the list of the vendor’s compensatory measures. The specific lists of assumptions and bases used in the PRA are documented in Section 5.3.2.2, which describes the hazardous condition evaluation leading to delineation of accident scenarios; Section 6.4, which describes the accident likelihood determination; and Section 7.4, which describes the radiation dose consequence determination. The compensatory measures listed by the vendor to reduce or mitigate risk are presented in Section 11.2.

Each assumption or measure was qualitatively evaluated to determine whether the impact from the source of uncertainty associated with the assumptions, or the potential risk reduction associated with compensatory measure could affect the PRA risk results enough to change the conclusions about the risk from TNPP Package transportation. The evaluation was facilitated by the quantitative risk results for the baseline case. In many cases the accident frequency or radiation dose consequences for the bounding representative accidents were determined to be so low for the baseline case that the impact of the assumption or compensatory measure could be qualitatively judged to have a minimal impact on risk based on examination. The qualitative evaluation was used to screen each assumption or measure from further consideration or identify it as a candidate for sensitivity study.

The qualitative screening process is documented in Table 9.1 through Table 9.4. In each table:

- The first column provides an identifier associated with the assumption or measure from the lists presented in Sections 5.3.2.2, 6.4, 7.4, or 11.2. The first two letters indicate which list the assumption or measure came from (e.g., HA indicates it came from the hazardous condition evaluation assumption list, LA indicates it came from the likelihood analysis assumption list, CA indicates it came from the radiation dose consequence analyses assumption list, and CM indicates it came from the list compensatory actions).
- The second column presents the assumptions and measures cited in sections of the report being evaluated.
- The third column dispositions the assumption or measure as either screened or further considered as providing the basis for a candidate sensitivity study. The discussion in the third column under disposition of "SCREENED" or "CANDIDATE STUDY" provides justification for the disposition.

As discussed above, qualitative evaluation was performed aid by the quantitative results from the baseline PRA results to screen the assumption, bases, or measure from further consideration or identify it as a candidate for sensitivity study.

Table 9.1. Evaluation of Hazard Analysis and Accident Identification Assumptions for Candidate Sensitivity Studies

Disposition of Hazard Analysis and Accident Identification Assumptions		
ID	**Assumption**	**Disposition**
HA1	The dominant radiation dose risk is associated with the Reactor Module configured as a TNPP Package because it contains the reactor, the fuel, portions of the primary cooling system, and nearly all of the radiological material inventory. It is assumed that other portions of the primary cooling system that contain radioactive material such as the IHX and piping connecting the reactor to the IHX are transported in a separate module or containers as LSA or surface contaminated objects. It is assumed that radiological contamination or activated material that might exist in the other modules contributes little to radiation dose risk. Accordingly, the hazards analysis focuses on the TNPP Package exclusively.	**SCREENED** If other non-Reactor Modules, contain sufficiently high levels of radioactivity to warrant shipment in a Type B package, the Probabilistic Risk Assessment (PRA) approach delineated in this report would apply. However, the additional risk determined for these modules is very likely to be negligible because (1) the entire radiological material inventory is assumed to reside in the TNPP Package, and therefore, any accident scenarios postulated for separate modules would be encompassed by accident scenarios postulated for the TNPP Package; and (2) the baseline results show that the radiation dose consequences from piping is a very small contributor. Therefore, a sensitivity study was not performed that addresses non-Reactor Modules.

Disposition of Hazard Analysis and Accident Identification Assumptions		
ID	**Assumption**	**Disposition**
HA2	The Reactor Module includes spent fuel after a specified period of decay, as described in the consequence analysis presented in Section 7.0.	**SCREENED** A sensitivity study was not performed to address the non-irradiated fuel case because the risk associated with this case is significantly lower than and bounded by the irradiated fuel case.
HA3	There is no gas cleanup 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 material at risk.	**SCREENED** In the baseline case, it is assumed that the entire radiological material inventory resides in the Reactor Module of the TNPP Package. The radiation dose from TNPP Package transportation accident scenarios involving the cleanup system is likely encompassed by accident scenarios postulated for the TNPP Package. Moreover, the cleanup system could be shipped separably in an approved package. Therefore, a sensitivity study was not performed that addresses the separate contribution of the gas cleanup system to radiation dose.
HA4	Submersion of the reactor vessel into a body of water could hypothetically lead to a criticality based on the available design information.	**SCREENED** Assessment of a TNPP Package transportation accident that involves a drop into a body of water of sufficient depth to submerge the reactor vessel was addressed in the baseline study. The likelihood of this scenario is estimated to be <5E-07 per year. Accordingly, the risk of this scenario is found to be an acceptable event though the assumption that submersion always leads to a criticality is conservative. Therefore, a sensitivity study was not performed that assumes a conditional probability of criticality.
HA5	No credit is taken for a HMIS given that one has not yet been defined, though such a system could reduce the risk from certain kinds of accidents.	**SCREENED** The HMIS could be credited as a detection system that alerts the crew to take mitigating actions for indications of a possible fire or radiation leak associated with BRA 1, 2, 7, or 8. However, the radiation dose consequences of these accidents are very low (i.e., in the 1E-03 to 1E-06 range), and therefore, a sensitivity study was not performed crediting a HMIS.

Disposition of Hazard Analysis and Accident Identification Assumptions		
ID	**Assumption**	**Disposition**
HA6	Loss of passive heat transfer from the reactor in the TNPP Package 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.	**SCREENED** The TNPP design documents indicate that the function of passive heat transfer is not safety related and that its purpose is to preserve sensitive electronics (see Section 5.2) and other sensitive components. Therefore, in the baseline study, failure of passive heat transfer in a TNPP Package transportation accident it is not postulated to lead to failure of containment due to increased pressure. Moreover, even if failure of a containment device is assumed in containment failure accident scenarios (i.e., BRA 7 and BRA 8), the radiation dose consequences of these accidents are very low (i.e., in the 1E-03 to 1E-06 range). Accordingly, postulating containment failure due to loss of passive heat transfer leads to a negligible increase in risk. Therefore, a sensitivity study was not performed that assumes containment failure due only to loss of passive heat transfer.
HA7	There is only enough combustible material inside Reactor Module in the form of cable, wire jacket and insulation to lead to a small fire.	**SCREENED** The radiation dose from fire accidents that originate inside or outside the CONEX box-like structure (i.e., BRA 1 and 2) was determined to be very low in the baseline case (i.e., 0 to 2.3E-03 rem). This is also true for impact with a tanker carrying flammable material and subsequent fire (i.e., BRA 6), which involves a much larger fire. The contribution to the radiation dose consequences from the fire phenomenon in BRA 6 is minimal (i.e., <1 rem). Therefore, a sensitivity study was not performed that postulates more combustible material inside the CONEX box-like structure.
HA8	No (or minimal) other flammable material, other than cable, wire jacket and insulation and minimal quantities of grease and oil, exist in the Reactor Module configured as the TNPP Package. No significant quantity of plastic wrapping or flammable packing material is used in this module.	
HA9	There will be energized electrical components in the TNPP Package during transport associated with parameter monitoring, lighting, and ventilation.	**SCREENED** This assumption ensures there are ignition sources considered for the postulated fires scenarios not caused by impact (i.e., BRA 1 and 2). However, the radiation dose consequences associated with the BRA 1 and 2 fires are quite low (i.e., 0 to 2.3E-03 rem). Therefore, the risk associated with energized electrical components in the TNPP Package during transport is acceptably low. A sensitivity study assuming there are no ignition sources inside or outside the TNPP Package to ignite a fire was not performed because the results would have negligible impact on the conclusions about fire risk.

Disposition of Hazard Analysis and Accident Identification Assumptions		
ID	**Assumption**	**Disposition**
HA10	It is assumed that the quantity of diesel fuel in the transport vehicle is about 300 gallons.	**SCREENED** This assumption about quantity of diesel fuel impacts the source term factors determined to be applicable for BRA 5M and BRA 5H, which are impact accidents followed by fire. The risk results for the baseline case show that the contribution to radiation dose for these fires is negligible. Moreover, this is also true for impact with a tanker carrying flammable material and subsequent fire (i.e., BRA 6), which can result in a much larger fire. Accordingly, even if the quantity of fuel is assumed to be less than or greater than the assumed amount (e.g., 150 gallons or 600 gallons), the calculated dose consequences would be the same. Therefore, a sensitivity study was not performed that postulates more (or less) diesel fuel than assumed in the baseline case.
HA11	The only external fire of sufficient magnitude to propagate into the TNPP Package from the outside is a diesel fuel fire. Other external truck fires such as engine fires and wheel or tire fires are not of sufficient magnitude to propagate into the TNPP Package.	**SCREENED** This assumption affects the frequency of BRA 2 (Diesel fuel fire-only event that originates outside the CONEX box-like structure). Even if non-diesel fuel related truck fires were assumed to propagate into the TNPP Package (which increases the accident frequency) the estimated dose consequence would still be very low (2.3E-03 rem to the worker and 5.1E-04 rem to the public). Accordingly, the conclusions about risk would remain unchanged.
HA12	For hard impacts followed by fire including a collision with a tanker carrying flammable liquid it was assumed that the proportion of collisions that involves an explosion (e.g., deflagration or detonation) is very small compared to that of collisions involving just fire. Therefore, the dose consequences from a hard impact followed by an explosion were not separately evaluated but assumed to be bounded by hard impact followed by fire, which over time can cause significant heat up of the TNPP Package.	**SCREENED** The accident frequencies for BRA 5H and BRA 6 are <5E-7 per year, and so even though the consequences of these accidents are high (i.e., a little higher than for BRA 3, which is a hard impact event without fire), the risk of these accidents falls below the risk evaluation guidelines. The accident frequency for BRA 5M is only slightly higher than the lower limit of the beyond extremely unlikely frequency category. If fire events that make up the basis for the accident frequencies of BRA 5M, BRA-5H, and BRA 6 were parsed into fire and explosion events, then the likelihood of explosion events for each of these types of impact accidents would clearly be below the risk evolution guidelines (See note (c) of Table 4.7). Therefore, a sensitivity study was not performed that considers the radiation dose consequences of explosion.

Disposition of Hazard Analysis and Accident Identification Assumptions		
ID	**Assumption**	**Disposition**
HA13	For the HA, there is no prohibition about transporting during inclement weather (e.g., extreme, wind, rain, or temperature related scenarios were included). However, it was also assumed that a shipment would not deliberately be made in weather conditions so severe that the design/integrity of the package would be exceeded.	**CANDIDATE STUDY** Per the next assumption (i.e., HA14), extreme weather events that can contribute to the occurrence of highway accidents that damage the TNPP Package are included in the large truck data. However, a sensitivity study could be performed that removes accidents from the very large truck accident count associated with inclement or extreme weather. Results of this study would show the potential impact on radiation dose consequences of placing controls on shipping during inclement weather. This is the same candidate sensitivity study proposed for HA14 below.
HA14	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 considered in separate scenarios. Moreover, the mechanical impact associated with very large truck crashes was assumed to dominate the accident phenomena, and as a result weather phenomena were not factored into determination of source term factors (e.g., high wind was not assumed to increase the impact or dilute the concentration of released material.)	**CANDIDATE STUDY** Extreme weather is assumed to be one of the causes of an accident and is reflected in the very large truck accident data. However, unacceptable accident frequencies could be reduced by removing the contribution of severe weather-related accidents and implementing a prohibition against shipping during severe weather. This is same candidate sensitivity study proposed for HA13 above.
HA15	There would be no specific control of passing or oncoming vehicles (i.e., collision with other vehicles was assumed possible) in development of the likelihood estimates.	**CANDIDATE STUDY** Controls associated with passing or oncoming vehicles is a way to reduce accident frequencies, and therefore, the risk of TNPP Package transportation accidents that result in problematic radiation dose consequences and frequencies. The assumed route is nearly all interstate, and the very large truck accident rate was determined for the interstate highway that makes up the assumed route. Accordingly, rolling road closures, which is an option used for nonradioactive, overweight, and over-dimension shipments, could be used to constrain passing of the transport by other vehicles. This control would mainly just benefit the risk determined for BRA 3, which has an accident frequency of 7.1E-05 per year and high dose consequences. (For accidents other than BRA 3, the risk is acceptable because either the accident frequency or consequence is exceptionally low compared to the risk evaluation guidelines.) It is not clear whether enough reduction in accident frequency can be obtained to render the risk of the accident acceptable, but a sensitivity study could be performed to investigate the possibilities.

Disposition of Hazard Analysis and Accident Identification Assumptions		
ID	**Assumption**	**Disposition**
HA16	Hazardous conditions qualitatively evaluated to be of low risk were not significant enough to be carried forward for detailed accident analysis. Low-risk scenarios were screened out because the likelihood was determined to be "Beyond Extremely Unlikely" (i.e., <5E-07 per year) or the consequences from the transportation accident were determined not to significantly affect any of the TNPP Package radiological inventory contributors listed in Section 5.1.4.	**SCREENED** A reasonableness check was performed of the cited qualitative screening and no candidate sensitivity studies were identified from this review. The screened hazardous conditions can be identified in Appendix B, as those assigned "Low risk to the worker and public." Accordingly, no sensitivity study was performed to estimate the impact of the scenario screening process.
HA17	The TNPP Package being transported has not experienced aDBE or BDBE during operation that would have affected diffusion rates during operation.	**SCREENED** Besides its impact on diffusion during operation the impact of experiencing a DBE or BDBE would require an evaluation and analysis of the condition of the TNPP Package that is beyond the scope of this study.
BDBE = Beyond Design Basis Event; BRA = bounding representative accident; CONEX = container express; DBE = Design Basis Event; HA = Hazard Analysis; HMIS = Health Monitoring Instrumentation System; IHX = intermediate heat exchange; LSA = Low Specific Activity; rem = roentgen equivalent man; SCO = surface contaminated object; TNPP = Transportable Nuclear Power Plant.		

Table 9.2. Evaluation of Accident Likelihood Analysis Assumptions for Candidate Sensitivity Studies

Dispositions of Likelihood Analysis Assumptions		
ID	**Assumption**	**Disposition**
LA1	The route is from INL to WSMR and uses interstate highways in parts of Idaho, Utah, Wyoming, Colorado, and New Mexico as described in Section 6.0.	**SCREENED** The route from INL to WSMR was chosen based on a preference for interstate highways between INL and WSMR. There are few practical variations of the route using interstate highways. One exception is going through Denver rather than around it, but the difference is minimal, as discussed in Section 6.0 and shown in Figure 6.5. Therefore, a sensitivity study to address this assumption has effectively been performed.
LA2	There is one transport in a year to be able to provide the accident frequency estimate on a per-year basis.	**SCREENED** Multiple shipments are not planned for this phase of the project. Moreover, the likelihood of transportation accidents associated with multiple shipments would be proportional to the likelihood of one shipment. Therefore, a sensitivity study that assumed more than one transport in a year was not performed.

Dispositions of Likelihood Analysis Assumptions		
ID	**Assumption**	**Disposition**
LA3	The known accident rates on all interstates in these five states are the same as accident rates on interstates of the assumed route.	**SCREENED** It does not appear possible to extract highway-specific accident data from the five-state accident data source. Therefore, a sensitivity study that extracts accident data from the five-state accident data source was not performed for specific highways.
LA4	The proportion of very large truck fatal accidents on interstate highways versus all state highways is the same for very large truck accidents of all types (i.e., including those that are not fatal) on these interstates versus all state highways, as described in Section 6.2.3.	**SCREENED** Because of the lack of data, the relative occurrence of fatal accidents on interstate and non-interstate highways was assumed to be representative of the relative occurrence of non-fatal accident on these highways and is used to estimate the all-accident rate for interstate only. The assumption is conservative in the sense that the accident rate on interstate highway is expected to be less than on non-interstate highway. On the other, the assumption is also non-conservative in the sense that only fatal accidents are considered though non-fatal accident may also result from severe impact crashes that can damage the TNPP Package. Since this assumption is combination of conservatism and non-conservatism and the base accident rate used in the PRA directly affects the risk results, a sensitivity study was not performed. However, an uncertainty analyses was performed on the base accident rate and is presented in Section 10.
LA5	The known types and proportions of very large truck interstate accidents in the nationwide dataset (i.e., in the MCMIS database) are assumed to be representative of the types and proportions of very large truck accidents on the assumed route, as described in Section 6.3.1.1.	**SCREENED** There is not enough state-specific data to support calculation of the accident frequency for the different accident types identified to lead to different radiation dose consequences. However, use of the nationwide data provides a way to estimate the five-state accident frequency for different accident types by multiplying the five-state accident frequency by the ratio of accidents of a particular type nationwide to total number of very large truck accidents nationwide. This is considered a refinement and a sensitivity that provides further refinement was not performed due to the lack of more refined data.
LA6	For accidents in the nationwide dataset (i.e., MCMIS) described in Section 6.3.1.1, where fire was the MHE, if a FHE was not specified, then the FHE is fire. If another	**SCREENED** The accidents involving both a crash and fire that do not involve a tanker carrying flammable liquids are BRA 5M, and BRA 5H. The radiation

Dispositions of Likelihood Analysis Assumptions		
ID	**Assumption**	**Disposition**
	kind of accident was designated as the FHE, then it was assumed to be a mixed accident (e.g., a crash and fire), as described in Section 6.3.1.1.	dose from BRA 5M is very low for released material (i.e., 3.5E-02 to the worker and 8.4E-03 rem to the public). Therefore, the accident meets the risk evaluation guidelines even if the estimated frequency of this accident changes. The accident frequency of BRA 5H is quite low (i.e., 2.6E-08 per year), and therefore BRA 5M meets the risk evaluation guidelines. On the other hand, if mixed crash-and-fire events are under-counted, then the accident frequency could be greater than 5E-07 per year. This would make the risk associated with this accident unacceptable. However, even if the ratio of mixed accidents (i.e., fire and crash) to all fire events (i.e., fire-only events and crash-and-fire events) was increased by an order of magnitude the frequency of BRA 5H would be 2.6E-07 per year would still be below the risk evaluation frequency guidelines. Using the percentage of total “Fire/explosion” accident shown in Table 6.12 (i.e., 0.54%), it can be concluded that even if all fire events are assumed to be mixed (i.e., impact and fire events), the risk evaluation guideline of 5E-07 per year is not exceeded for BRA 5H. Therefore, a sensitivity study was not performed in which the proportion of mixed crash accidents (i.e., fire and crash) to all accidents (i.e., fire-only events and crash-and-fire events) was increased.
LA7	For accidents in the nationwide dataset (i.e., MCMIS) described in Section 6.3.1.1, the most severe types of impact accidents—hard impacts (i.e., BRA 5H)—are considered to be heavy vehicle collisions, impacts with unyielding objects, rollovers/overturns, and drops to lower elevation. Medium impacts (i.e., BRA 5M) are considered to be all other crashes, including light vehicle collisions, impacts with yielding objects, and jackknives.	**SCREENED** It is a refinement to use the crash description categories from a nationwide dataset as shown in Table 6.10 to determine a proportion of hard and medium impact to total impacts and apply that proportion to the very large truck accident rate. Further refinement with more refined data is not possible. Moreover, even if this proportion of heavy and light cases was changed to increase the accident frequency of BRA 5M or BRA 5H, the risk insights do not change. The estimated radiation dose consequences for BRA 5M are low (i.e., 6 rem to the worker and 7.7E-02 rem to the public) and any increase in accident frequency cannot cause the risk associated with BRA 5M to exceed the risk evaluation guidelines of 750 rem to the worker or the

Dispositions of Likelihood Analysis Assumptions		
ID	**Assumption**	**Disposition**
		public (which is the allowed dose for an accident frequency of 5.9E-07 per year). Accordingly, a sensitivity study was not performed that adjusts the portion of medium and hard impact accidents.
LA8	For “less than hard impacts” accidents (i.e., BRA 4, as described in Section 6.3.1.1), those that result in a fatality or injury-only in the nationwide dataset (i.e., MCMIS) are considered medium impacts (BRA 4M) and those that result in property damage-only are considered light impacts (BRA 4L).	**SCREENED** Adjustment of the accident frequency between BRA 4M and 4L will have no impact on the conclusion about the risk associated with BRA 4L, because the frequency of BRA 4L cannot be impacted enough to raise it to the next accident frequency interval in the risk evaluation guidelines for which dose limit is lower. Adjustment of the accident frequency between BRA 4M and 4L could have an impact on the conclusion about the risk associated with BRA 4M. because he frequency of BRA 4M could increase enough to raise it to the next accident frequency interval in the risk evaluation guidelines for which the dose limit is lower. However, this is already acknowledged in the presentation of the risk results presented in Section 8.1.4 and conclusions presented in Section 13.0. Accordingly, a sensitivity study was not performed re-validate this finding.
LA9	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 pounds) miles nationwide, as described in Section 6.3.1.1.	**SCREENED** It is very conservative to base the frequency of BRA 6 on all tanker miles as opposed to tankers carrying flammable liquids; even so, the estimated frequency of BRA 6 is just 7.1E-08. Given the conservatism, a sensitivity study should explore a less conservative estimate, but radiological risk associated with BRA 6 already meets the risk evaluation guidelines by virtue of its very low likelihood. Therefore, a sensitivity study was not performed that adjusts accident frequency for impact with a tanker carrying flammable liquid.

Dispositions of Likelihood Analysis Assumptions		
ID	**Assumption**	**Disposition**
LA10	For the GIS estimation of the submersion accident, locations along the route identified 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, and if a truck in an accident left the road it could slide or roll into the body of water, as described in Sections 6.1.3 and 6.3.1.2.	**SCREENED** In addition to this conservative GIS approach to estimating the accident frequency for BRA 9A (Flooded criticality), an accident frequency was calculated using nationwide very large truck data, which resulted in a beyond extremely unlikely accident frequency for BRA 9A. The combination of these insights produced an acceptably low frequency for this accident. Given that two methods were used to investigate the frequency of this accident, no sensitivity study was performed to address the impact of this assumption.
LA11	For GIS estimation of the frequency of an 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 Sections 6.1.3 and 6.3.1.2.	**SCREENED** The calculated frequency using the GIS approach of a drop-to-lower-elevation accident is 2.3E-06 per year and is a contributor to BRA 3, which is a hard impact without fire, and BRA 5M and 5H, which are impacts with fire. The frequency of a drop-to-lower-elevation accident is a minor contributor to the accident frequency BRA 3 (i.e., 7.1E-05 per year). The GIS approach involves conservative assumptions and likely results in a conservative estimate of frequency. The estimate of the frequency of a drop-to-lower-elevation accident would have to be made significantly more conservative to affect the frequency of BRA 3. Likewise, the frequency contribution of a drop-to-lower-elevation accident to BRA 5H (i.e., 2.6E-08 per year) and BRA 5M is 5.9E-07 is insignificant compared to other contributors by two or more orders of magnitude. So, a very large increase in drop frequency would be needed to increase the frequency of BRA 5H to an unacceptable level. The consequence of BRA 5M is quite low (i.e., in the 1E-01 to 1E-2 rem range), so an increase of the frequency above 5E-07 per year does not change the risk insights. Therefore, no sensitivity study was performed to address the impact of this assumption.

Dispositions of Likelihood Analysis Assumptions		
ID	**Assumption**	**Disposition**
LA12	For non-crash fire-only events, no credit was quantified for early detection or fire response mitigation as described in Section 6.3.2.1.	**SCREENED** Non-crash fire-only events are very low risk accidents given that they produce radiation dose consequences in the zero to 2.3E-03 rem at frequencies in the 9.0E-07 to 2.0E-06 per year range. Therefore, a sensitivity was not performed to evaluate the impact of decreasing risk by crediting early detection or fire response mitigation.
LA13	For BRA 4L, it was conservatively assumed that there was a 10 percent chance that a worker could be located in the radiation stream through a gap or fissure in the radiation transport shielding for the duration assumed in the consequence analysis (i.e., 30 minutes).	**SCREENED** The direct radiation dose is estimated using a conservative MicroShield model developed by PNNL, so rather than a sensitivity study, a more accurate model of transport shielding from the vendor is needed.
BDBE = Beyond Design Basis Event; BRA = bounding representative accident; CONEX = container express; FHE = first harmful event; GIS = geographic information system; HMIS = Health Monitoring Instrumentation System; INL = Idaho National Laboratory; LA = Likelihood Analysis; LSA = Low Specific Activity; MCMIS = Motor Carrier Management Information System; MHE = most harmful event; PRA = Probabilistic Risk Assessment; rem = roentgen equivalent man; TNPP = Transportable Nuclear Power Plant; WSMR = White Sands Missile Range.		

Table 9.3. Evaluation of Accident Radiological Consequence Assumptions for Candidate Sensitivity Studies

Disposition of Consequence Analysis Assumptions		
ID	**Assumption**	**Disposition**
CA1	The reactor core has decayed for 90 days after 3 years of operation.	**CANDIDATE STUDY** A sensitivity study is needed to investigate the impact of decay times on the radiation dose consequences of a TNPP Package in a transportation accident because the MAR is significantly affected by decay time.
CA2	The portion of the primary reactor cooling system transported contains all the condensed or plated-out radioactive material (e.g., the released fission products and condensed gases in this system have not been removed before transport).	**SCREENED** The results of the baseline case show that the contribution to radiation dose from primary cooling system release, even for the worst-case accident, is minimal (i.e., <1 rem for the worker and public). Therefore, a sensitivity study that removes the dose consequences associated with the reactor cooling system was not performed.

Disposition of Consequence Analysis Assumptions		
ID	**Assumption**	**Disposition**
CA3	A radioactive material cleanup system and/or the resulting radioactive waste material is not transported with the microreactor.	**SCREENED** In the baseline case, it is assumed that the entire radiological material inventory resides in the TNPP Package. The radiation dose from the TNPP Package transportation accident scenarios involving the cleanup system is likely encompassed by accident scenarios postulated for the TNPP Package. Moreover, the cleanup system could be shipped separably in an approved package. Therefore, a sensitivity study that addresses the separate contribution of the gas cleanup system to radiation dose was not performed.
CA4	The direct radiation dose is estimated using a MicroShield model developed by PNNL. The model is based on design information from the vendor about the reactor (i.e., the types and thicknesses of different layers of material) and the thickness of the CONEX box-like structure and transport shielding. The model and results were benchmarked against the results of preliminary external radiation results from the vendor for a design in which the radiation limits at the outside of the CONEX were met.	**SCREENED** The results of the baseline study show that the contribution of direct radiation to the dose from a TNPP Package transportation accident for the public is low (e.g., about 7% of the radiation dose contribution in the worst-case accident for the worker) and extremely low to the public. The model is conservative in the sense that the transport shielding is assumed to be completely removed. Therefore, this contribution is unlikely to affect the conclusions about risk. Therefore, a sensitivity study was not performed that uses a different direct radiation contribution from the damaged radiation shielding.
CA5	The baseline case release fractions from diffusion occurring during normal operations leading to material residing in the core, reactor structure, and coolant system as well as the source term factors described in Section 7.1 represent best judgment conservative estimates.	**SCREENED** The results of the baseline case show that the radiation dose contribution in an accident from diffused or plated-out material is very small (two orders of magnitude less than the contribution from the release from TRISO fuel from impact.). Therefore, a sensitivity study was not performed that adjusts the release fractions from normal operations for material residing in the core, reactor structure, and coolant system.
CA6	For BRA 4L events, which are light-impact highway accidents, it is assumed there is no release of radiological material. These light-impact accidents are defined as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or impact with a light vehicle that is not severe (e.g., results in property damage only). However, it was assumed that enough damage to the transport shielding occurred to create a gap in the CONEX box-like	**SCREENED** Concerning release of radiological material, it is assumed that the greatest damage that can occur is to produce a leak similar to that which occurs as a result of BRA 7 or BRA 8. It can be concluded that the impact of this assumption on risk is inconsequential because the dose consequences calculated for BRA 7 and BRA 8 are extremely low (i.e., ranges between 1.7E-03 and 8.4E-06 for the worker and public). Therefore, a sensitivity study was not

Disposition of Consequence Analysis Assumptions		
ID	**Assumption**	**Disposition**
	structure of the Reactor Module (e.g., damage to a corner) allowing radiation streaming to occur. It was further assumed there is a 10% chance that a worker could be located in the radiation stream for 30 minutes.	performed that examines the impact of assuming no release for BRA 4L. **CANDIDATE STUDY** Concerning direct radiation from unreleased radiological material, the chance that a worker could be located in the radiation stream for 30 minutes is encompassed below in the sensitivity study proposed for CA7 concerning the impact to dose consequences from different exposure times.
CA7	The radiation dose pathways and the transportation accident consequence analysis of these pathways are generally based on the IAEA SSG-26 (IAEA 2022) methodology and assumptions made for assessing package performance, such as the 30-minute duration time for exposure to the damaged TNPP Package.	**CANDIDATE STUDY** A sensitivity study that determines the dose consequences from different exposure times to a damaged package could provide valuable risk insights and help inform emergency response.
CA8	A deviation from the approach identified in assumption CA7 above is that the uptake by a human receptor is derived in consideration of the accident phenomena, as described in Section 7.2. This is done for each bounding representative accident using the five factors associated with defining the source term, as described in Section 7.1. This assumption provides a more refined estimate than the guidance in IAEA SSG-26 (IAEA 2022) because it tailors the source term to the accident phenomena. The guidance in IAEA SSG-26 is simply based on the assumption that the radioactive material intake by a "bystander" is 0.1 percent of the released material.	**SCREENED** The approach of assuming uptake by a human receptor is derived in consideration of the accident phenomena and is more refined than the approach presented in SSG-26. Therefore, further refinement in the form a sensitivity study was not performed due to the lack of applicable data.
CA9	As a follow-on to assumption CA8 above, use of accident-specific source term factors, required use of best but conservative judgment for factors developed for a different use (i.e., specifically for fuel cycle and nonreactor nuclear facilities).	**CANDIDATE STUDY** The source term factors are estimated using guidance from NUREG/CR-6410 (NRC 1998) and DOE-HDBK-3010-94 (DOE 2013), but the accident conditions and phenomena for a TNPP Package transportation accident can be different from a fuel cycle facility or waste package accident for which the data was derived. The approach of assuming uptake by a human receptor is derived in consideration of the accident phenomena and is more refined than the approach presented in SSG-26. However, best judgment was needed to apply source term factors from NUREG/CR-6410 (NRC 1998) and DOE-HDBK-3010-94.

Disposition of Consequence Analysis Assumptions		
ID	**Assumption**	**Disposition**
		Therefore, a sensitivity study is warranted to determine in practice how much impact this source of uncertainty has on the TNPP Package transportation risk results.
CA10	Another deviation from the approach identified in assumption CA7 above is that the TNPP Package Probabilistic Risk Assessment approach considers two receptors—a worker and a member of the public—whereas SSG-26 does not differentiate between receptors. The distance between the worker and the point of the release is assumed to be 1 m (except for the inhalation dose pathway for which the distance is assumed to be within 10 m of the point of release), which is equivalent to the approach described in IAEA SSG-26. A member of the public is assumed to be located 25 m from the point of the release because that is the protective action distance for high-level radiological material emergency response (DOT 2020).	**CANDIDATE STUDY** A sensitivity study that determines the dose consequences for a distance other than that used in the baseline study could provide valuable risk insights help inform emergency response procedures.
CA11	The dose contribution from ingestion submersion is assumed to be negligible, and therefore is 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, which significantly limits the time that a receptor might stand in a gaseous cloud of radionuclides.	**SCREENED** Submersion dose is not included because the assumption is made that the exposure will take place outside with a 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. A sensitivity study was not performed because it is not anticipated that a release cannot occur indoors for a TNPP Package transportation accident.
CA12	It is assumed that workers involved in handling radioactive material after an accident would wear the appropriate protective clothing including gloves. Therefore, even though skin contamination equivalent skin dose is calculated and shown in the radiation dose consequences results table (because it is used in the IAEA SSG-26 approach), it is not counted toward the risk results that are compared to risk evaluation guidelines.	**SCREENED** It is considered unrealistic to assume that workers escorting the TNPP Package during transport will not be trained in radiation safety and equipped with the appropriate radiation protective clothing and gear. That said, there is a chance that a radiation safety protection mishap could occur, but such events would occur after the accident. A sensitivity study was not performed because, radiation safety protection mishaps are considered separately from the nuclear safety function of the TNPP Package.

Disposition of Consequence Analysis Assumptions		
ID	**Assumption**	**Disposition**
BRA = bounding representative accident; CA = Consequence Analysis; CONEX = container express; IAEA = International Atomic Energy Agency; INL = Idaho National Laboratory; LA = Likelihood Analysis; LSA = Low Specific Activity; MAR = material at risk; PNNL = Pacific Northwest National Laboratory; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic.		

Table 9.4. Evaluation of Compensatory Measures for Candidate Sensitivity Studies

Evaluation of the Impact of Compensatory Measures		
ID	**Assumption**	**Disposition**
CM1	Escort the reactor forward and aft for the entire route. Army to provide escorts.	**SCREENED** There appears to be very little available data about the extent to which use of escorts may decrease the very large truck accident rate. Therefore, a quantitative sensitivity study was not performed to estimate the impact of using the cited escort configuration.
CM2	Choose a route that avoids bodies of water and bridges over bodies of water.	**SCREENED** The route-specific hazard is explicitly addressed in the report in the baseline as it pertains to the possibility of a flooded criticality and was shown to be beyond extremely unlikely. No further investigation of this route-specific hazard was addressed in sensitivity studies.
CM3	Choose a route and schedule the shipment to avoid the potential for flash flooding.	**CANDIDATE STUDY** The large truck accident databases used in the demonstration PRA do not refer to accidents during or caused by flash flooding but do refer to more general weather conditions. The impact of extreme weather was addressed during evaluation of hazardous conditions. A sensitivity study could be performed that removes accidents from the very large truck accident count associated with inclement or extreme weather. Results of this study would show the potential impact on radiation dose consequences of placing controls on shipping during inclement weather. This candidate sensitivity study is similar to the candidate study identified for assumption HA13 and HA14 in Table 9.1.
CM4	Ship at night to avoid other traffic.	**CANDIDATE STUDY** The large truck accident databases used in the demonstration PRA include environmental lighting conditions at the time of the crash. Crash data in combination with truck flow rate data could be used in a sensitivity study to

Evaluation of the Impact of Compensatory Measures		
ID	**Assumption**	**Disposition**
		estimate the quantitative impact of shipping at night.
CM5	Avoid shipping during known times of high traffic volume.	**SCREENED** The large truck accident databases used in the demonstration PRA do not refer to traffic volume data or indicate whether a crash occurred during times of high traffic volume. Therefore, a sensitivity study was not performed to determine the quantitative impact of avoiding shipment during known times of high traffic volume.
CM6	The planned HMIS will provide real-time parameter monitoring of a TNPP Package during transport and it is anticipated it will be designed to detect conditions signaling that a TNPP Package transportation accident (e.g., a leak from containment) has or could occur.	**SCREENED** The HMIS could be credited as a detection system that alerts the crew to take mitigating actions upon indications of a possible fire or leak that could lead to BRA 1, 2, 7, or 8. However, the dose consequences of these accidents are very low (i.e., in the 1E-03 to 1E-05 range for the worker and public). Therefore, any decrease in the likelihood of these accidents has minimal impact on the risk conclusion for these accidents. Accordingly, a sensitivity study was not performed that credits the HMIS in TNPP Package transportation accidents.
CM = Compensatory Measures; PRA = Probabilistic Risk Assessment; HA = Hazard Analysis; HMIS = Health Monitoring Instrumentation System; TNPP = Transportable Nuclear Power Plant.		

9.1.2 Delineating Feasible Sensitivity Studies

In the second step of identifying helpful sensitivity studies, an examination and analysis of the assumptions and measures not screened out in the first step were performed to define feasible studies. As explained in Section 9.1.1, Table 9.1, Table 9.2, Table 9.3, and Table 9.4 identify certain TNPP Package transportation PRA modeling assumptions or compensatory measures as being the basis for a candidate sensitivity study and are further discussed in this section. For many of the assumptions or compensatory actions dispositioned with the label “CANDIDATE STUDY,” sensitivity studies were defined and performed as described below. In some cases, candidate sensitivity studies were found to be infeasible to perform quantitatively primarily due to the lack of data.

9.1.2.1 Sensitivity Study – Decay Time After Operation

In the baseline case, it was assumed that the reactor core had decayed for 90 days after 3 years of operation as discussed above in assumption CA1 listed in Table 9.3. The selection of a 90-day decay time for the baseline case stemmed from a preliminary assessment indicating the radiological inventories from shorter decay times may result in radiation dose consequences that exceed acceptable doses for a TNPP Package transportation accident. The decay time of the core after operation and up to the time that the TNPP Package is transported is a major contributor to how much radiation dose is received in a transportation accident because of the impact that decay time has on the radiological inventory.

Accordingly, a sensitivity study was performed to determine the impact on radiation dose consequences of different decay times after 3 years of operation (i.e., 30 days, 60 days, 1 year, and 2 years). The radionuclide inventories for the baseline case and the sensitivity studies were provided by the vendor.[54] The risk results from this sensitivity study are presented in Section 9.2.1.

9.1.2.2 Sensitivity Study – Distance from a Member of the Public to Point of Release

In the baseline case, it was assumed that a member of the public is located 25 m from the point of a release, as indicated in assumption CA10 in Table 9.3. Pathways that contribute to the radiation dose for a member of the public are external and inhalation dose from released material and external dose from unreleased material. Skin contamination from working with debris from the accident is not applicable for a member of the public and direct radiation dose from loss or degradation of the transport shielding is expected to be very minimal at distances considered for public dose. The distance of 25 m used in the baseline case comes from the protective action distance for high-level radiological material emergency response (DOT 2020). If that distance was increased from 25 m to 100 m, then the estimated dose consequences to the public would decrease. On the other hand, if the distance was decreased, then the corresponding estimated dose consequences to the public would Increase. Accordingly, a sensitivity case was performed assuming a member of the public is at 100 m from the point of the release and another case was performed assuming a member of the public was at the same distance from the point of the release as a worker.

For the case in which a member of the public is assumed to be 100 m from the point of the release, the consequence analysis evaluated relevant standards from national and international agencies. Inhalation dose was evaluated applying the approach used in IAEA SSG-26 (IAEA 2022) and comparing it to guidance from DOE-STD-3009-2014 (DOE 2014), the standard for a nonreactor nuclear facility Documented Safety Analysis. These references provide generic uptake fractions for a ground-level release relevant to a transportation accident.

For many of the bounding representative accidents, inhalation is the dominant radiation dose pathway. Results from the baseline case presented in Table 7.6 in Section 7.5 show that the contribution to the total radiation dose from photon and beta radiation for a member of the public is negligible in comparison to the inhalation dose. The determination of inhalation dose is based on the source term material released, the atmospheric dispersion factor (identified as χ/Q) at the selected receptor location, the receptor breathing rate, and the inhalation dose coefficient. For the bounding representative accident that does not meet the proposed risk evaluation for the public (which is BRA 3) controlling the distance of the public from the accident at 100 m opposed to 25 m might be especially beneficial by reducing the potential radiation dose to the public.

Appendix I.36 of IAEA SSG-26 provides a generic uptake fraction range of 1E-04 to 1E-03 for individuals located 10 m downwind from a radioactive material ground release. It is estimated, as discussed in Section I.36, that the uptake fraction decreases by a factor of 30 for an

[54] Radionuclide inventories came from BWXT spreadsheet "B1.34-NuclideConcentrations(Ci)-Fuel.xlsx" provided August 11, 2022.

individual at 100 m downwind. Using a generic power function for interpolation, it was estimated that when comparing an individual at 25 m to 100 m, the uptake fraction would be reduced by a factor of 7.75. The generic uptake fraction at 100 m downwind from DOE-STD-3009-2014 was compared and found to be less conservative than the IAEA SSG-26 uptake fraction. Accordingly, the IAEA SSG-26 Section I.36 uptake fractions were chosen for this application to be bounding. The risk results from this sensitivity study are discussed in Section 9.2.2 and presented in Table 9.14.

For the other case in which a member of the public is assumed to be at the same distance from the point of the release as a worker, the estimated radiation dose consequences for the worker can be used directly to estimate the dose to the member of the public at the same distance. The distance that was used for the photon and beta external radiation dose pathway was 1 m using the guidance in IAEA SSG-26 and 10 m for the inhalation dose pathway. The inhalation dose pathway is the dominant contributor to radiation dose in an accident. A 10 m distance is used for the inhalation dose pathway because inside 10 m, other factors that would tend to reduce the uptake come into effect and may even become dominant, as stated in SSG-26, Appendix I.36. The impacts on the PRA results are presented and discussed in Section 9.2.2.

9.1.2.3 Sensitivity Study – Exposure Time to a Damaged TNPP Package

In the baseline study, the exposure time for the worker and public to the damaged TNPP Package after an accident was assumed to 30 minutes per the approach used in IAEA SSG-26 as discussed under assumption CA7 in Table 9.3. Increasing the time leads to a higher radiation dose. The increase in dose would apply to exposure to external photon and beta radiation for released and unreleased radiological material but not to inhalation exposure. For the inhalation dose, it is expected that any airborne release of source term material during an accident scenario would result in a short-term release. Any uptake would be received shortly after the incident, so an increase in duration is not expected to increase inhalation dose.

The inhalation doses are based on the amount of released material that could be taken up into the lungs and result in an inhalation dose. The release is assumed to be of short duration in comparison to the length of time the individual is present at the accident site, with all of the intake occurring while the individual is present. These intake values reported in SSG-26 are based on a χ/Q release material distribution and breathing rate for an individual at 100 m, which is then extrapolated down to a 10 m dose for the entire release duration. In both cases the most conservative uptake factors were used. The inhalation doses are rough estimates. The actual inhalation dose values during an accident will be dependent on the location and the meteorological conditions at the time of the accident.

In the sensitivity study the exposure time was increased to 60 minutes from 30 minutes. BRA 5H, BRA 6, BRA 9A, and BRA 9B were found to be acceptable based on their exceptionally low accident frequency, so increasing the exposure time in these scenarios would have no impact on the conclusions about risk for these accidents. For the remaining bounding representative accidents (i.e., BRA 2, BRA 4L, BRA 4M, BRA 5M, BRA 7, and BRA 8), the radiation dose duration is doubled, and the new radiation dose consequences are compared to the proposed risk evaluation guidelines. This impact on the TNPP Package transportation PRA risk for these bounding representative accidents is presented in sensitivity study results provided in Section 9.2.3.

9.1.2.4 Sensitivity Study – Uncertainty in Source Term Fraction Estimates

In the baseline study, the source factors (described in Section 7.1 of this report) were estimated using guidance from NUREG/CR-6410 (NRC 1998) and DOE-HDBK-3010-94 (DOE 2013) as stated in assumption CA9 of Table 9.3. The approach of deriving an uptake by a human receptor based on accident phenomena instead of assuming a given uptake is more refined than the approach presented in IAEA SSG-26. However, best judgment was needed to apply source term factors from NUREG/CR-6410 (NRC 1998), which are for fuel cycle facilities, and DOE-HDBK-3010-94, which are for radiological waste containers. Therefore, a sensitivity study was performed in which the sum of the aggregate source term factors was increased for bounding representative accidents that currently meet the risk evaluation guidelines up to a factor of 1,000 to determine the point the risk evaluation guidelines were exceeded. This was done to determine how sensitive the risk results were to the uncertainty in the source fraction estimates. When the sum of the aggregate source term factors must be increased by more than a factor of 1,000 to exceed the risk evaluation guidelines, then the risk results are considered to be insensitive to the source term estimates. This study does not include the bounding representative accident that exceed the risk evaluation guidelines (i.e., BRA 3) or the four bounding representative accidents that meet the risk evaluation guidelines based on their exceptionably low accident frequencies (i.e., BRA 5H, BRA 6, BRA 9A, and BRA 9B) in the baseline case.

Section 9.2.4 presents the results of this sensitivity study by showing the aggregate source factor increases that would need to be assumed before the bounding representative accident exceeded the risk evaluation guidelines.

9.1.2.5 Sensitivity Study – Restriction of Transport During Extreme Weather

Assumption HA13 and HA14 in Table 9.1 and compensatory action CM3 in Table 9.4 suggest that the likelihood of very large truck accidents could be reduced, if shipments were postponed when there is inclement or extreme weather. The disposition of HA14 explains that extreme weather is assumed to be one of the causes of an accident and is reflected in the very large truck accident data. Therefore, unacceptable accident frequencies could be reduced by removing the contribution of severe weather-related accidents and implementing a prohibition against shipping during severe weather. Vary large truck crash data sources described in Section 6.2 were examined for data on crashes during increment or extreme weather.

Figure 9.1 provides the results of a search of the MCMIS data source in calendar year 2019 for the environmental weather conditions for crashes involving very large trucks.

State	Clear	Blowing Sand, Soil, Dirt, or Snow	Fog, Smog, Smoke	Rain	Severe Crosswinds	Sleet, Hail	Snow	Other
Colorado	80.2%	0.09%	1.33%	3.56%	1.85%	0.00%	13.00%	0.00%
Idaho	84.7%	2.16%	0.54%	2.16%	0.54%	0.81%	9.05%	0.00%
New Mexico	83.6%	0.47%	1.04%	3.96%	0.09%	0.85%	7.17%	2.83%
Utah	73.6%	1.44%	0.83%	6.97%	0.45%	0.38%	16.29%	0.00%
Wyoming	54.6%	11.76%	3.10%	2.07%	4.61%	1.11%	22.73%	0.00%
Subset of fatal accidents only.								
Colorado	79.6%	2.04%	1.02%	7.14%	1.02%	0.00%	9.18%	0.00%
Idaho	83.9%	12.90%	0.00%	0.00%	0.00%	0.00%	3.23%	0.00%
New Mexico	96.2%	0.00%	0.00%	0.00%	0.00%	0.00%	2.56%	1.28%
Utah	78.3%	2.17%	2.17%	6.52%	0.00%	0.00%	10.87%	0.00%
Wyoming	58.7%	2.17%	8.70%	0.00%	0.00%	0.00%	30.43%	0.00%

Figure 9.1. Environmental Weather Conditions of Crashes Involving Very Large Trucks from MCMIS

Figure 9.1 shows that that crashes occur most often during clear weather by a significant margin. This is likely because there is more very large truck traffic during clear weather. Very large truck crashes could occur at a higher rate during inclement and extreme weather than they do for clear weather, but this cannot be determined from the data presented above. Without having corresponding data about the number of very large trucks per unit of time for the weather conditions reported in the dataset, quantitative assessment cannot be performed to determine the percentage of accidents that might be avoided by prohibiting transport during inclement or extreme weather.

Figure 9.2 provides the results of a search of the MCMIS data source in calendar year 2019 for road surface conditions that could be caused by environmental weather conditions that involve crashes of very large trucks. The top of the figure presents the percentages for all very large truck crashes and the bottom half presents the percentage for fatal large truck crashes.

State	Dry	Ice	Sand, Mud, Dirt, Oil, or Gravel	Slush, Snow	Water (Standing, Moving)	Wet	Other
Colorado	73.67%	9.44%	0.09%	8.40%	0.00%	8.21%	0.19%
Idaho	73.32%	7.00%	0.26%	11.62%	0.40%	7.40%	0.00%
New Mexico	84.35%	6.97%	0.00%	3.30%	0.00%	5.28%	0.09%
Utah	52.74%	7.18%	0.00%	26.37%	0.39%	13.32%	0.00%
Wyoming	40.94%	43.16%	0.32%	8.11%	0.00%	7.47%	0.00%
Subset of fatal accidents only.							
Colorado	76.53%	8.16%	0.00%	7.14%	0.00%	7.14%	1.02%
Idaho	83.87%	3.23%	0.00%	6.45%	0.00%	6.45%	0.00%
New Mexico	93.59%	2.56%	0.00%	0.00%	0.00%	3.85%	0.00%
Utah	69.23%	0.00%	0.00%	11.54%	0.00%	19.23%	0.00%
Wyoming	47.83%	39.13%	0.00%	0.00%	0.00%	13.04%	0.00%

Figure 9.2. Road Conditions of Crashes Involving Very Large Trucks from MCMIS

Figure 9.2 shows that crashes occur most often during dry-road conditions by a significant margin. Again, this is likely because there is more very large truck traffic during dry-road conditions. Very large truck crashes could occur at a higher rate during poor road conditions than during dry conditions, but this cannot be determined from the data presented above. Without having corresponding data about the number of very large trucks per unit of time for the road conditions reported in the dataset, quantitative assessment cannot be performed to determine the percentage of accidents that might be avoided by prohibiting transport during inclement or extreme weather that causes poor environmental road conditions. The exception to the observation above, are the percentages of crashes during icy road conditions in Wyoming, as shown in Figure 9.2, which might be due to the fact that Wyoming has more icy roads during the year, However, even in this case, more information is needed to perform a quantitative study. Without further data the possibility that crash rates for very large trucks are less during inclement weather cannot be ruled out for reasons such as there is less traffic and drivers are more attentive.

Accordingly, a quantitative sensitivity study of the impacts of avoiding shipping during inclement or extreme weather or poor road conditions on the PRA result are not addressed quantitatively in Section 9.2 due lack of data.

9.1.2.6 Sensitivity Study – Transport at Night

Shipping at night to avoid other traffic is identified in Table 9.4 as compensatory measure CM4. The large truck accident databases used in the demonstration PRA include environmental lighting conditions at the time of the crash. Accordingly, Table 9.4 identified potential use of these data in a candidate sensitivity study. Figure 9.3 provides the results of a search of the MCMIS data source in calendar year 2019 for the environmental lighting conditions for crashes involving large trucks.

	Daylight	Dark - Not Lighted	Dark - Lighted	Dark - Unknown Roadway Lighting	Dawn	Dusk	Other
Colorado	73.26%	14.22%	7.73%	0.00%	3.08%	1.71%	0.00%
Idaho	71.39%	18.15%	3.58%	0.40%	0.93%	0.26%	5.30%
New Mexico	66.07%	23.25%	4.63%	0.00%	2.84%	2.74%	0.47%
Utah	71.80%	16.28%	8.02%	1.03%	1.67%	1.19%	0.00%
Wyoming	60.84%	31.14%	3.26%	0.00%	2.94%	1.51%	0.32%
	72.10%	13.73%	10.17%	0.40%	2.29%	1.22%	0.09%

Figure 9.3. Environmental Lighting Conditions of Crashes Involving Large Trucks from MCMIS

Like the weather-related environmental conditions presented in Section 9.1.2.5, Figure 9.4 shows that crashes occur most often during daylight hours. This is likely because there is more very large truck traffic during daylight hours making shipping at night possibly more desirable. A paper by Statistics Canada (2005), a Canadian national statistical agency, indicates this is the case, as shown in Figure 9.4.

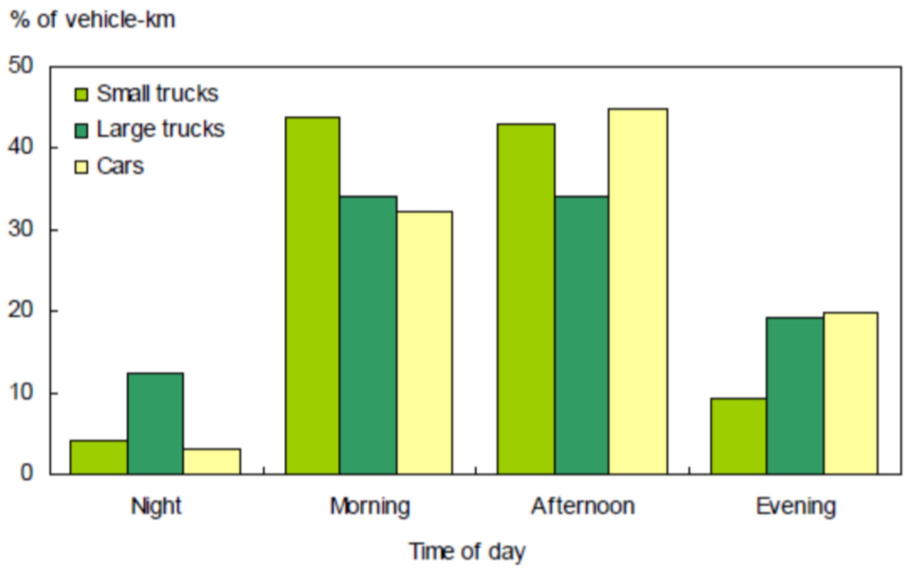

Figure 9.4. Vehicles per Kilometer for Times of the Day (Statistics Canada 2005)

Again, however, data corresponding to the number of very large trucks per unit of time for the environmental lighting conditions reported on the MCMIS dataset are needed to perform a quantitative assessment to determine the percentage of accidents that might be avoided by prohibiting transport during poor environmental lighting conditions. Without further data, it is hard to show that that crash rates for very large trucks are less at night. Lighting problems, wildlife on the road, and other factors that cause accidents may be more prominent during nighttime, dawn, and dusk driving.

Accordingly, a quantitative sensitivity study of the impact of shipping at night is not addressed quantitatively in Section 9.2.

9.1.2.7 Sensitivity Study – Control of Passing or Oncoming Traffic

The fact that the demonstration PRA takes no credit for control of passing or on-coming traffic is identified in hazardous condition evaluation assumption HA15. The evaluation of this assumption in Table 9.1 states that controls associated with limitations on passing or on oncoming vehicles is a way to reduce accident frequencies, and therefore, the risk of TNPP Package transportation accidents. A control of this nature could have been listed by the vendor but is not included in the compensatory measures presented in Section 11.2 of this report. Rolling road closures, which are an option used for nonradioactive overweight and over-dimension shipments, could be used to help control passing traffic.

However, no available data was found for such controls that could be used for estimating the reduction in very large truck accident rates. Additionally, a control like “rolling road closures” would need to be evaluated for reliability and for whether it may create unintended and unevaluated hazardous conditions. Based on engineering judgment, it seems likely that such controls would reduce accident risk but the data for determining the degree of risk reduction was not found.

9.2 Risk Results of Sensitivity Studies and Evaluations

The results of quantitative sensitivity studies of adjustments to selected PRA input assumptions for TNPP Package transportation accidents are provided in this section. This includes the impact of:

- The decay time of the reactor core since operation
- The distance of a public receptor from the accident
- The duration of exposure to the accident
- The uncertainty in source term fractions.

9.2.1 Results of Sensitivity Study – Decay Time after Operation

A sensitivity study was performed to determine the impact of the radiation dose consequences of different decay times after 3 years of operation (i.e., 30 days, 60 days, 1 year, and 2 years). The radionuclide inventories for the baseline case and the sensitivity studies were provided by the vendor.[55]

Table 9.5 through Table 9.13 present results of the sensitivity study for applicable bounding representative accidents (i.e., there are no radiation dose consequences for BRA 1 and consequences are not calculated for BRA 9A, BRA 9B, and BRA 10). The primary insight gained from this sensitivity study is that decaying the core for up to a year (or even somewhat less than a year) will ensure an acceptable level of risk for all bounding representative accidents based on the proposed risk evaluation guidelines.

Table 9.5 shows that for BRA 2 (Fire-only originating external to the Rector Module), the risk is acceptable for this accident for all five decay times (i.e., 30 days, 60 days, 90 days, 1 year and 2 years). In general, the TNPP Package transportation risk results are not very sensitive to a fire hazard compared to the kinetic energy associated crash impact.

Table 9.5. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 2

Risk and Comparison to Risk Evaluation Guidelines				
Delay from Shutdown to Transport	**Worker Dose (rem TEDE)**	**Public Dose (rem TEDE)**	**Accident Frequency (per year)**	**Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report**
Accident Consequence (Radiation dose from Table 7.6)				
30 days	3.0E-03	6.2E-04		≥5 and <25 rem TEDE for a member of the public ≥25 and <100 rem TEDE for a worker when the accident frequency is ≤1E-05 and >1E-06
60 days	2.5E-03	5.5E-04		
90 days	2.3E-03	5.1E-04		
1 year	1.8E-03	4.2E-04		
2 years	1.7E-03	3.9E-04		
Accident Frequency assuming one trip per year (from Table 6.16)			3.4E-06	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable for all delay times
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

[55] Radionuclide inventories came from BWXT spreadsheet "B1.34-NuclideConcentrations(Ci)-Fuel.xlsx" provided August 11, 2022.

The results presented in Table 9.6 show that for BRA 3 (Hard impact), the risk is acceptable for this accident for decay times of 1 year or more and would be acceptable for a delay time of less than 1 year if the results were interpolated.

Table 9.6. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 3

<table>
<tr><th colspan="5">Risk and Comparison to Risk Evaluation Guidelines</th></tr>
<tr><th>Delay from Shutdown to Transport</th><th>Worker Dose (rem TEDE)</th><th>Public Dose (rem TEDE)</th><th>Accident Frequency (per year)</th><th>Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report</th></tr>
<tr><td colspan="5">Accident Consequence (from Table 7.6)</td></tr>
<tr><td>30 days</td><td>1420</td><td>319</td><td rowspan="6">7.1E-05</td><td rowspan="6">≥1 and <5 rem TEDE for a member of the public
≥5 and <25 rem TEDE for a worker
when the accident frequency is
≤1E-04 and >1E-05</td></tr>
<tr><td>60 days</td><td>208</td><td>45.9</td></tr>
<tr><td>90 days</td><td>87.7</td><td>18.8</td></tr>
<tr><td>1 year</td><td>14.5</td><td>3.3</td></tr>
<tr><td>2 years</td><td>7.8</td><td>1.7</td></tr>
<tr><td colspan="3">Accident Frequency assuming one trip per year (from Table 6.16)</td></tr>
<tr><td colspan="4">COMPARISON TO RISK EVALUATION GUIDELINE</td><td>Acceptable for delay times of 1 year or more</td></tr>
<tr><td colspan="5">rem = roentgen equivalent man; TEDE = total effective dose equivalent.</td></tr>
</table>

The results presented in Table 9.7 show that for BRA 4L (Light impact), the risk is acceptable for this accident for decay times of 60 days or more.

Table 9.7. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 4L

<table>
<tr><th colspan="5">Risk and Comparison to Risk Evaluation Guidelines</th></tr>
<tr><th>Delay from shutdown to transport</th><th>Worker Dose (rem TEDE)</th><th>Public Dose (rem TEDE)</th><th>Accident Frequency (per year)</th><th>Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report</th></tr>
<tr><td colspan="5">Accident Consequence (from Table 7.6)</td></tr>
<tr><td>30 days</td><td>22.6</td><td>2.5E-01</td><td rowspan="6">3.3E-05</td><td rowspan="6">≥1 and <5 rem TEDE for a member of the public
≥5 and <25 rem TEDE for a worker
when the accident frequency is
≤1E-04 and >1E-05</td></tr>
<tr><td>60 days</td><td>7.9</td><td>8.7E-02</td></tr>
<tr><td>90 days</td><td>6.0</td><td>6.9E-02</td></tr>
<tr><td>1 year</td><td>9.0E-01</td><td>1.0E-02</td></tr>
<tr><td>2 years</td><td>5.3E-01</td><td>5.8E-03</td></tr>
<tr><td colspan="3">Accident Frequency assuming one trip per year (from Table 6.16)</td></tr>
<tr><td colspan="4">COMPARISON TO RISK EVALUATION GUIDELINE</td><td>Acceptable for delay times of 60 days or more</td></tr>
<tr><td colspan="5">rem = roentgen equivalent man; TEDE = total effective dose equivalent.</td></tr>
</table>

The results in Table 9.8 show that for BRA 4M (Medium impact), the risk is acceptable for this accident for decay times of 30 days or more.

Table 9.8. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 4M

Risk and Comparison to Risk Evaluation Guidelines				
Delay from Shutdown to Transport	**Worker Dose (rem TEDE)**	**Public Dose (rem TEDE)**	**Accident Frequency (per year)**	**Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report**
Accident Consequence (from Table 7.6)				
30 days	22.6	2.6E-01		≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
60 days	7.9	9.6E-02		
90 days	6.0	7.7E-02		
1 year	9.3E-01	1.7E-02		
2 years	5.5E-01	1.2E-02		
Accident Frequency assuming one trip per year (from Table 6.16)			9.7E-05	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable for all delay times
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

The results presented in Table 9.9 show that for BRA 5H (Hard impact and fire), the risk is acceptable for this accident for all decay times based on the frequency of the event.

Table 9.9. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 5H

Risk and Comparison to Risk Evaluation Guidelines				
Delay from Shutdown to Transport	**Worker Dose (rem TEDE)**	**Public Dose (rem TEDE)**	**Accident Frequency (per year)**	**Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report**
Accident Consequence (from Table 7.6)				
30 days	1420	319		≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker when the accident frequency is ≤5E-07
60 days	208	45.9		
90 days	87.8	18.8		
1 year	14.5	3.3		
2 years	7.8	1.7		
Accident Frequency assuming one trip per year (from Table 6.16)			2.6E-08	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable for all delay times
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

The results presented in Table 9.10 show that for BRA 5M (Medium impact and fire), the risk is acceptable for this accident for decay times of 30 days or more.

Table 9.10. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 5M

Risk and Comparison to Risk Evaluation Guidelines				
Delay from Shutdown to Transport	**Worker Dose (rem TEDE)**	**Public Dose (rem TEDE)**	**Accident Frequency (per year)**	**Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report**
Accident Consequence (from Table 7.6)				
30 days	22.6	2.6E-01		≥25 and <750 rem TEDE for a member of the public ≥100 and <750 rem TEDE for a worker when the accident frequency is ≤1E-06 and >5E-07
60 days	7.9	9.6E-02		
90 days	6.0	7.7E-02		
1 year	9.3E-01	1.7E-02		
2 years	5.5E-01	1.2E-02		
Accident Frequency assuming one trip per year (from Table 6.16)			5.9E-07	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable for all delay times
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

Table 9.11 shows that for BRA 6 (Collision with tanker carrying flammable liquids and fire), the risk is acceptable for this accident for all decay times based on the frequency of the event.

Table 9.11. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 6

Risk and Comparison to Risk Evaluation Guidelines				
Delay from Shutdown to Transport	**Worker Dose (rem TEDE)**	**Public Dose (rem TEDE)**	**Accident Frequency (per year)**	**Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report**
Accident Consequence (from Table 7.6)				
30 days	1420	319		≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker when the accident frequency is ≤5E-07
60 days	209	46.1		
90 days	88.7	18.9		
1 year	15.3	3.5		
2 years	8.5	1.9		
Accident Frequency assuming one trip per year (from Table 6.16)			7.1E-08	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable for all delay times
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

The results presented in Table 9.12 show that for BRA 7 (Nonpressurized loss of reactor containment), the risk is acceptable for this accident for all decay times examined. Given the very low radiation dose consequence from this accident, the risk from this accident is not sensitive to the decay time of the core since operation.

Table 9.12. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 7

Risk and Comparison to Risk Evaluation Guidelines				
Delay from Shutdown to Transport	**Worker Dose (rem TEDE)**	**Public Dose (rem TEDE)**	**Accident Frequency (per year)**	**Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report**
Accident Consequence (from Table 7.6)				
30 days	5.5E-05	1.0E-05		≤0.1 rem TEDE for a member of the public ≤2 rem TEDE for a worker when the accident frequency is >1E-03
60 days	4.5E-05	9.0E-06		
90 days	4.1E-05	8.4E-06		
1 year	3.2E-05	6.6E-06		
2 years	2.8E-05	6.0E-06		
Accident Frequency assuming one trip per year (from Table 6.16)			1.3E-03	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable for all delay times
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

Table 9.13 shows that for BRA 8 (Pressurized loss of reactor containment), the risk is shows that for BRA 8 (Pressurized loss of reactor containment), the risk is acceptable for this accident for all decay times of examined. Given the very low radiation dose consequence from this accident, the risk from this accident is not sensitive to decay time of the core since operation.

Table 9.13. Sensitivity Study of Impact of Decay Time After Operation Until Transport – BRA 8

Risk and Comparison to Risk Evaluation Guidelines				
Delay from Shutdown to Transport	**Worker Dose (rem TEDE)**	**Public Dose (rem TEDE)**	**Accident Frequency (per year)**	**Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report**
Accident Consequence (from Table 7.6)				
30 days	2.2E-03	4.1E-04		≤0.1 rem TEDE for a member of the public ≤2 rem TEDE for a worker when the accident frequency is >1E-03
60 days	1.8E-03	3.6E-04		
90 days	1.7E-03	3.4E-04		
1 year	1.3E-03	2.7E-04		
2 years	1.1E-03	2.4E-04		
Accident Frequency assuming one trip per year (from Table 6.16)			1.3E-03	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable for all delay times
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

The primary insight gained from this sensitivity study is that decaying the core for up to a year (or even somewhat less than a year) after it has been in operation for 3 years will make sure there is an acceptable level of risk for all bounding representative accidents based on the proposed risk evaluation guidelines. (This assumes the design goal of precluding a reactivity insertion event (e.g., control rod withdrawal) caused by the impact energy from a TNPP Package transportation accident has been met.)

9.2.2 Results of Sensitivity Study – Distance of a Member of the Public from Damaged Package

Controlling the distance of the public from a TNPP Package transportation accident at 100 m as opposed to 25 m is beneficial because it reduces the potential radiation dose to the public. Similarly, decreasing the distance of a member of the public to less than 25 m would increase the radiation dose to the public for a TNPP Package transportation accident. As shown in Table 8.13, the only bounding representative accident that does not meet the proposed risk evaluation for the public is BRA 3 (Hard impact). Therefore, BRA 3 is the one accident for which benefit could be realized by increasing the distance to public from the accident in the radiation dose consequence analysis.

Using the estimation approach described in Section 9.1.2.2, Table 9.14 shows that the if the public is controlled at 100 m as opposed to 25 m, then the risk from BRA 3 (Hard impact) to the public is found to be acceptable using the proposed risk evaluation guidelines (though the risk to the worker is still unacceptable).

On the other hand, if the distance of the public from the accident is not controlled to 25 m then the radiation dose consequences would be higher and as high as the radiation dose determined for the worker, as discussed in Section 9.1.2.2. Based on inspection of the results presented in Table 8.13 and sensitivity results presented in Table 9.15 through Table 9.22, the risk conclusions of all bounding representative accidents would remain the same if the radiation dose to the public was increased to be the same as the radiation dose to the worker with a couple of exceptions. The exceptions are for BRA 4L and BRA 4M in which the dose to the public would slightly exceed the risk evaluation guideline dose of 5 rem at an accident frequency of 3.3E-05 per year and 9.7E-04 per year, respectively.

Table 9.14. Member of the Public 100 Meters from the Accident for BRA 3 – Hard-Impact

<table>
<tr><th>Accident Risk</th><th>Worker Dose (rem TEDE)</th><th colspan="2">Public Dose (rem TEDE)</th><th>Accident Frequency (per year)</th><th>Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report</th></tr>
<tr><td colspan="4">Accident Consequence (from Table 7.6)</td><td rowspan="9">7.1E-05</td><td rowspan="9">≥1 and <5 rem TEDE_for a member of the public
≥5 and <25 rem TEDE for a worker

when the accident frequency is
≤1E-04 and >1E-05</td></tr>
<tr><td colspan="2">MAR contribution from released material</td><td>25 m</td><td>100 m</td></tr>
<tr><td>TRISO Fuel</td><td>80.9</td><td>18.5</td><td>2.4</td></tr>
<tr><td>Core Structure</td><td>5.2E-01</td><td>1.3E-01</td><td>1.7E-02</td></tr>
<tr><td>Cooling System</td><td>3.1E-01</td><td>6.3E-02</td><td>8.1E-3</td></tr>
<tr><td colspan="4">Contribution from Unreleased Material</td></tr>
<tr><td>Degraded shielding</td><td>6.0</td><td colspan="2">6.9E-02</td></tr>
<tr><td>Total Dose</td><td>87.7</td><td>18.8</td><td>2.4</td></tr>
<tr><td colspan="4">Accident Frequency assuming one trip per year (from Table 6.16)</td></tr>
<tr><td colspan="5">COMPARISON TO RISK EVALUATION GUIDELINE</td><td>Unacceptable</td></tr>
<tr><td colspan="6">rem = roentgen equivalent man; TEDE = total effective dose equivalent.</td></tr>
</table>

Table 9.15. Member of the Public Same Distance from the Accident as the Worker for BRA 2 – Fire Only that Originates Outside the CONEX Box-Like Structure

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥5 and <25 rem TEDE for a member of the public ≥25 and <100 rem TEDE for a worker when the accident frequency is ≤1E-05 and >1E-06
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	1.0E-03	1.0E-03		
Cooling System	1.2E-03	1.2E-03		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Radiation dose	**2.3E-03**	**2.3E-03**		
Accident Frequency assuming one trip per year (from Table 6.16)			**3.4E-06**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

Table 9.16. Member of the Public Same Distance from the Accident as the Worker 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	0	0		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.0		
Total Dose	**6.0**	**6.0**		
Accident Frequency assuming one trip per year (from Table 6.16)			**3.3E-05**	
COMPARISON TO RISK EVALUATION GUIDELINE				Unacceptable
rem = roentgen equivalent man; TEDE = total effective dose equivalent.				

Table 9.17. Member of the Public Same Distance from the Accident as the Worker 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	2.6E-02		
Cooling System	9.3E-03	9.3E-03		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.0		
Total Dose	**6.0**	**6.0**		
Accident Frequency assuming one trip per year (from Table 6.16)			**9.7E-05**	
COMPARISON TO RISK EVALUATION GUIDELINE				Unacceptable

MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).

Table 9.18. Member of the Public Same Distance from the Accident as the Worker for BRA 5M 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥25 and <750 rem TEDE for a member of the public ≥100 and <750 rem TEDE for a worker when the accident frequency is ≤1E-06 and >5E-07
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	2.6E-02		
Cooling System	9.3E-03	9.3E-03		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.0		
Total Dose	**6.0**	**6.0**		
Accident Frequency assuming one trip per year (from Table 6.16)			**5.9E-07**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable

MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).

Table 9.19. Member of the Public Same Distance from the Accident as the Worker for BRA 5H – Hard-Impact Accident and an Ensuing Fire

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker when the accident frequency is ≤5E-07
MAR contribution from released material				
TRISO Fuel	80.9	80.9		
Core Structure	5.2E-01	5.2E-01		
Cooling System	3.1E-01	3.1E-01		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.0		
Total Dose	**87.8**	**87.8**		
Accident Frequency assuming one trip per year (from Table 6.16)			**2.6E-08**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 9.20. Member of the Public Same Distance from the Accident as the Worker 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker when the accident frequency is ≤5E-07
MAR contribution from released material				
TRISO Fuel	81.7	81.7		
Core Structure	6.2E-03	6.2E-03		
Cooling System	3.7E-03	3.7E-03		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.0		
Total Dose	**88.7**	**88.7**		
Accident Frequency assuming one trip per year (from Table 6.16)			**7.1E-08**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 9.21. Member of the Public Same Distance from the Accident as the Worker for BRA 7 – Loss of the Nonpressurized Reactor Containment Boundary

Accident Risk	Worker Dose	Public Dose	Accident	Applicable Proposed Risk Evaluation Guidelines from

	(rem TEDE)	(rem TEDE)	Frequency (per year)	Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≤0.1 rem TEDE for a member of the public ≤2 rem TEDE for a worker when the accident frequency is >1E-03
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	4.1E-05	4.1E-05		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Dose	**4.1E-05**	**4.1E-05**		
Accident Frequency assuming one trip per year (from Table 6.16)			**1.3E-03**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 9.22. Member of the Public Same Distance from the Accident as the Worker for BRA 8 – Loss of the 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≤0.1 rem TEDE for a member of the public ≤2 rem TEDE for a worker when the accident frequency is >1E-03
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	1.6E-03	1.6E-03		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Dose	**1.6E-03**	**1.6E-03**		
Accident Frequency assuming one trip per year (from Table 6.16)			**1.3E-03**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

The conclusions of this sensitivity study are that for the one accident (i.e., BRA 3) in which the risk to the public is unacceptable, increasing the isolation distance for public from 25 m to 100 m and assuming that distance in the consequence analysis yields acceptable risk when using the proposed risk evaluation guidelines (though the risk to the worker from BRA 3 is still unacceptable). However, if the distance to the public from the accident is decreased to be the same as the distance for the worker, the conclusions about whether the risk of a representative bounding accident meets the risk evaluation guidelines remains the same for all bounding representative accidents with two exceptions. The exceptions are for BRA 4L and BRA 4M in which the dose to the public (i.e., 5 rem) would exceed the risk evaluation guideline dose of 5 rem at an accident frequency of 3.3E-05 per year and 9.7E-04 per year, respectively.

9.2.3 Results of Sensitivity Study – Exposure Time to Damaged TNPP Package

Limiting the time of exposure to the radiological impacts of a TNPP Package transportation accident reduces the potential radiation dose to the worker and the public. As described in Section 9.1.2.3, an exposure time of 30 minutes is used in the baseline case based on the approach used in IAEA SSG-26. However, if that time was increased to 60 minutes, the total radiation dose to the worker and the public would increase. As explained in Section 9.1.2.3, this sensitivity study only applies to BRA 2, BRA 5M, BRA 7, and BRA 8 because other bounding representative accidents are either already unacceptable or acceptable based on accident frequency.

Table 9.23 through Table 9.28 show that by increasing the exposure time from 30 minutes to 60 minutes as described in Section 9.1.2.3, the risk for these accidents would remain acceptable using the proposed risk evaluation guidelines.

Table 9.23. 60-minute Exposure Time to Damaged TNPP Package for BRA 2 – Fire Only that Originates Outside the CONEX Box-Like Structure

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥5 and <25 rem TEDE_for a member of the public ≥25 and <100 rem TEDE for a worker when the accident frequency is ≤1E-05 and >1E-06
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	1.0E-03	2.6E-04		
Cooling System	1.2E-03	2.5E-04		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Baseline Radiation dose	2.3E-03	5.1E-04		
Total dose from increased exposure to 60 from 30 minutes	**2.6E-03**	**5.1E-04**		
Accident Frequency assuming one trip per year (from Table 6.16)			**3.4E-06**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 9.24. 60-Minute Exposure Time to Damaged TNPP Package 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥1 and <5 rem TEDE_for a member of the public
MAR contribution from released material				
TRISO Fuel	0	0	**3.3E-05**	

Core Structure	0	0		≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
Cooling System	0	0		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Baseline Radiation dose	6.0	6.9E-02		
Total dose from increased exposure to 60 from 30 minutes	**12**	**1.4E-01**		
Accident Frequency assuming one trip per year (from Table 6.16)				
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 9.25. 60-Minute Exposure Time to Damaged TNPP Package 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	6.5E-03		
Cooling System	9.3E-03	1.9E-03		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Baseline Radiation dose	6.0	6.9E-02		
Total dose from increased exposure to 60 from 30 minutes	**12**	**1.4E-01**		
Accident Frequency assuming one trip per year (from Table 6.16)			**9.7E-05**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 9.26. 60-Minute Exposure Time to Damaged TNPP Package for BRA 5M – 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥25 and <750 rem TEDE for a member of the public ≥100 and <750 rem TEDE for a worker
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	6.5E-03	**5.9E-07**	

Cooling System	9.3E-03	1.9E-03		when the accident frequency is ≤1E-06 and >5E-07
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Baseline Radiation dose	6.0	7.7E-02		
Total dose from increased exposure to 60 from 30 minutes	**12.0**	**1.5E-01**		
Accident Frequency assuming one trip per year (from Table 6.16)				
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 9.27. 60-Minute Exposure Time to Damaged TNPP for BRA 7 – Loss of the Nonpressurized Reactor Containment Boundary

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≤0.1 rem TEDE for a member of the public ≤2 rem TEDE for a worker when the accident frequency is >1E-03
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	4.1E-05	8.4E-06		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Baseline Radiation dose	4.1E-05	8.4E-06		
Total dose from increased exposure to 60 from 30 minutes	**5.0E-05**	**8.4E-06**		
Accident Frequency assuming one trip per year (from Table 6.16)			**1.3E-03**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 9.28. Risk Results Comparison for BRA 8 – Loss of the 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≤0.1 rem TEDE for a member of the public ≤2 rem TEDE for a worker when the accident frequency is >1E-03
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	1.7E-03	3.4E-04		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Baseline Radiation dose	1.7E-03	3.4E-04		
Total dose from increased exposure to 60 from 30 minutes	**2.0E-03**	**3.4E-04**		
Accident Frequency assuming one trip per year (from Table 6.16)			**1.3E-03**	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

In summary, increasing the duration of worker and public exposures to a damaged TNPP Package from 30 to 60 minutes has no impact on the conclusion about TNPP Package transportation risk. Bounding representative accidents that were acceptable in the baseline case based on frequency are still acceptable, accidents that are unacceptable in the baseline are still unacceptable, and the risk of the remaining accidents would remain acceptable.

9.2.4 Results of Sensitivity Study – Uncertainty in Source Term Fraction Estimates

A sensitivity study was performed in which the sum of the aggregate source term factors was increased for bounding representative accidents with radiation dose consequences up to a factor of 1,000 to determine the point at which the risk evaluation guidelines may be exceeded. This was done to determine how sensitive the risk results were to the uncertainty in the source fraction estimates. When the sum of the aggregate source term factors must be increased by more than a factor of 1,000 to exceed the risk evaluation guidelines, then the risk results are considered to be insensitive to the source term estimates.

The bounding representative risk results in Table 8.1 through

Table 8.11, indicate how much increase in radiation dose is needed to exceed the risk evaluation guidelines. This sensitivity study does not include the bounding representative accident that exceeds the risk evaluation guidelines (i.e., BRA 3) or the four bounding representative accidents that meet the risk evaluation guidelines based on their exceptionally low accident frequencies (i.e., BRA 5H, BRA 6, BRA 9A, and BRA 9B) in the baseline case. Table 9.29 below summarizes the results of this study.

Table 9.29. Sensitivity Study of the Impact of the Uncertainty in Estimating Source Term Factors

Bounding Representative Accident	Dose Limitation (TEDE rem)		Factor Increase that Exceeds Dose Limit		Sensitivity
	Worker	Public	Worker	Public	
BRA 1	750	750	>1000	>1000	Not sensitive
BRA 2	100	25	>1000	>1000	Not sensitive
BRA 5M	750	750	125	>1000	Not very sensitive
BRA 7	2	0.1	>1000	>1000	Not sensitive
BRA 8	2	0.1	294	>1000	Not sensitive
BRA = bounding representative accident; rem = roentgen equivalent man; TEDE = total effective dose equivalent.					

In summary, the risk conclusions are not very sensitive to the impact of uncertainty on estimating the source term factors given the following definition of sensitivity. The radiation dose consequences are considered “not sensitive” to the source term estimates if an increase of more than a factor of 1,000 is needed to exceed the risk evaluation guidelines and is considered “not very sensitive” if an increase of more than a factor of 100 is needed to exceed the risk evaluation guidelines. The risk of the other bounding representative accidents not shown in Table 9.29 are either already unacceptable or acceptable based on accident frequency according to the baseline results. In this case, the risk to BRA 4M and BRA 4L would remain the same because all the contribution to dose (or in the case of BRA 4M nearly all the contribution to dose) is direct radiation from unreleased material (i.e., damage to the transport shielding). Therefore, the impact of this source of modeling uncertainty is not important to the TNPP Package PRA risk conclusions.

9.3 Insights Gained from the Sensitivity Studies

The insights gained from the screening, identification, and performance of quantitative sensitivity studies are discussed in this section. They include (1) takeaways from the process of screening PRA modeling assumptions to identify PRA input uncertainties that could affect the risk results, as discussed in Section 9.1.1, (2) delineation of feasible sensitivity studies as discussed in Section 9.1.2, and (3) the results of quantitative sensitivity studies presented in Sections 9.2.1, 9.2.2, 9.2.3, and 9.2.4. These insights are summarized below.

1. Most of the PRA modeling assumptions listed in Sections 5.3.2.2, 6.4, and 7.4 were qualitatively dispositioned as not having a significant impact on the conclusions about risk from the bounding representative accidents. These dispositions shown in Table 9.1 on the hazard analysis accident identification assumptions, Table 9.2 on the accident likelihood determination assumptions, and Table 9.3 on the accident consequence determination assumptions often used the risk results presented in Section 8 to make these conclusions. In some cases, the input itself did not have much impact on the risk results. In other cases, the input could have an impact on the risk results, but it could be qualitatively reasoned not

to affect the risk conclusions (i.e., it would not change the determination of whether the representative accident meets the risk evaluation guidelines.)

2. In addition to PRA modeling assumptions, Table 9.4 presents disposition of compensatory measures listed by the vendor for candidate sensitivity studies. While various compensatory measures may be feasible to implement, the quantitative impact of the compensatory measures is difficult to quantify without having more refined accident data, as discussed in Section 9.1.2. Accordingly, quantitative assessment of the impact of compensatory actions were not as feasible to perform as the sensitivity studies performed certain sources of modeling uncertainty.
3. Allowing the TNPP reactor core to decay up to a year (or even somewhat less than a year) after it has been in operation for 3 years will make sure there is an acceptable level of risk for all bounding representative accidents addressed in this study based on the proposed risk evaluation guidelines, as described in Section 9.2.1. This assumes the design goal of precluding a reactivity insertion event (e.g., control rod withdrawal) caused by the impact energy of the accident from the TNPP Package transportation accident has been met.
4. For the one bounding representative accident (i.e., BRA 3) for which the risk to the public is unacceptable, increasing the isolation distance for the public from 25 m to 100 m and assuming that distance in the consequence analysis yields acceptable risk results for the public using the proposed risk evaluation guidelines (though the risk to the worker for BRA 3 is still unacceptable). Decreasing the distance of the public from the accident to be the same distance as the worker from the accident does not the change the conclusions about whether the risk the representative bounding accident meets the risk evaluation guidelines. The conclusion remains the same for all bounding representative accidents with two exceptions. The exceptions are for BRA 4L and BRA 4M in which the dose to the public (i.e., 6 rem) would exceed the risk evaluation guideline dose of 5 rem at an accident frequency of 3.3E-05 per year and 9.7E-04 per year, respectively.
5. Increasing the duration of worker and public exposure to a damaged TNPP Package from 30 to 60 minutes has minimal impact on the conclusions about TNPP Package transportation risk. Bounding representative accidents that were acceptable in the baseline case based on frequency are still acceptable, accidents that are unacceptable in the baseline are still unacceptable, and the risk of the remaining accidents would remain acceptable.
6. The risk conclusions about TNPP Package transportation are not very sensitive to the impact of uncertainty on estimating the source term factors given the following definition of sensitivity. The radiation dose consequences are considered “not sensitive” to the source term estimates if an increase of more than a factor of 1,000 is needed to exceed the risk evaluation guidelines and is considered “not very sensitive” if an increase of more than a factor of 100 is needed to exceed the risk evaluation guidelines. The risk of several bounding representative accidents is either already unacceptable in the baseline case or acceptable based on accident frequency according to the baseline results. Therefore, the impact of this uncertainty is not important to the TNPP Package PRA risk conclusions. In this case, the risk to BRA 4M and BRA 4L would remain the same because all the contribution to dose is from direct radiation from unreleased material (i.e., damage to the transport shielding) for BRA 4L and nearly all the dose contribution for BRA 4M.

A final insight is that even though compensatory measures examined in Section 9.1.2.5 about the restriction of transport during extreme weather and in Section 9.1.2.6 about shipping at night during low traffic make intuitive sense, the impact of these controls could not be quantitatively assessed using the large truck crash datasets described in Section 6.2. Although environmental weather conditions, road conditions, and lighting conditions are recorded in a nationwide dataset for crash entries, there is no corresponding data about very large truck traffic. Therefore, reduction in accident rates cannot be shown. The fact that most accidents happen during clear weather and during daylight hours is likely an indication that truck traffic is higher during those hours. However, without further data, the possible reduction in crash rates for very large trucks for not driving during inclement weather or driving at night cannot be demonstrated and may not be true.

10.0 Uncertainty Analysis

Since the advent of PRA, uncertainty analysis has always played an important role in the assessment of nuclear power plant risk. The first large-scope (i.e., Level 1, 2, and 3) PRA and perhaps the most well-known PRA study of nuclear power plant risk is NUREG-1150, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants* (NRC 1990). It states:

> an important characteristic of the probabilistic risk analyses… is that they have explicitly included an estimation of the uncertainties in the calculations of core damage frequency and risk that exist because of incomplete understanding of reactor systems and severe accident phenomena.

The two different basic classes of uncertainties are aleatory and epistemic (Apostolakis 1994). Aleatory uncertainty pertains to the random nature of events such as initiating events and component failures and is associated with physical phenomena that cannot be reduced by model or data improvements. The PRA model is an explicit model of random processes and thus is a model of aleatory uncertainty. Epistemic uncertainties arise when making statistical inferences from data and, perhaps more significantly, from incompleteness in the collective state of knowledge about how to represent plant behavior in the PRA model. The RIDM report states that epistemic uncertainty represents the uncertainty associated with the human understanding of a particular process or physical phenomenon. As the knowledge base increases, our understanding of the physical phenomena is also enhanced, thereby reducing the uncertainty associated with the data and the model. Epistemic uncertainty is reducible as the modeling aspects are refined. In practice, the probability of a sequence could be regarded as representing an aleatory uncertainty and the probability of parameters and models as representing epistemic uncertainties.

This section discusses the general role of uncertainty analysis in PRA in Section 10.1 and describes the uncertainty analyses that were performed for demonstration TNPP Package transportation PRA in Section 10.2.

10.1 Role of Uncertainty in PRA

Since 1990 when NUREG-1150 was issued, quantitative uncertainty analysis for nuclear power plant PRA has mostly focused on the parametric uncertainty analyses associated with modeling fault-tree and event-tree inputs, particularly the probability distribution of failure rates. Failure rates have a direct impact on the determination of CDF and LERF. These metrics are important because the NRC has issued guidance in 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 the CDF and LERF levels at which a change in a plant's operating license would not be allowed using a risk-informed approach. Currently, the risk acceptance guidelines in RG 1.174 do not include criteria associated with limitations in uncertainty. However, NRC-accepted guidance on uncertainty can affect the use of the risk acceptance guidelines in RG 1.174 as discussed below.

NUREG-1855, Revision 1, *Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making* (NRC 2009), provides background on the use of uncertainty analysis in PRA that supports risk-informed decisions. It states that in a 1995 policy statement (NRC 1995), the NRC encouraged the use of PRA in all regulatory matters and declared that the "use of PRA technology should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements NRC's deterministic approach," including sensitivity studies and uncertainty analyses and importance measures.[56] It further states, citing the policy statement, that the "treatment of uncertainty is an important issue for regulatory decisions." Uncertainties exist because of knowledge limitations and PRA has exposed some of these limitations and provides a framework for assessing their significance and assisting in developing a strategy to accommodate them in the regulatory process.

The guidance in RG 1.200, Revision 3, *Acceptability of Probabilistic Risk Assessment Results for Risk Informed Activities* (NRC 2020) acknowledges the importance of understanding uncertainties and their impacts and states:

> "An important aspect in understanding the base PRA results is knowing what are the sources of uncertainty and assumptions and understanding their potential impact. Uncertainties can be either parameter or model uncertainties, and assumptions can be related either to PRA scope and level of detail or to model uncertainties. The impact of parameter uncertainties is gained through the actual quantification process. The assumptions related to PRA scope and level of detail are inherent in the structure of the PRA model. The requirements of the applications will determine whether they are acceptable. The impact of model uncertainties and related assumptions can be evaluated qualitatively or quantitatively. The sources of model uncertainty and related assumptions are characterized in terms of how they affect the base PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event)."

The national PRA consensus standard (ASME/ANS 2009), which is endorsed by RG 1.200, Revision 3, also recognizes the importance of identifying and understanding uncertainty analysis. While the PRA consensus standard does not provide explicit guidance on the treatment of uncertainties in risk-informed decisionmaking, it does provide guidance in the form of High-level Requirements (HLRs) and Supporting Requirements (SRs). One of the HLRs for PRA quantification (QU-E) stipulates that "Uncertainties in the PRA results shall be characterized. Sources of model uncertainty and related assumptions shall be identified, and their potential impact on the results understood." SR QU-E2 addresses quantitative uncertainty analysis by stipulating that the CDF estimate must consider parametric modeling uncertainties. As stated above, parametric uncertainty analyses focus primarily on propagating the probability distribution of failure rates that contribute to CDF. However, as mentioned above the risk acceptance guidelines in RG 1.174 do not include criteria associated with limitations on uncertainty.

As described in Section 4.1, for activities like transportation, the NRC has proposed guidance in a report titled *Risk-Informed Decisionmaking for Nuclear Material and Waste Applications* (NRC 2008; referred to in this report as the RIDM report) for accepting the risk associated with such activities based on a risk assessment approach such as a PRA. The RIDM report provides general guidance on uncertainty analysis. It states that "Sources of uncertainties should be carefully considered to ensure that all uncertainties …. are included, properly characterized, and propagated through the risk model." However, it also states that "Modeling uncertainties can also be reduced by making models as realistic as possible, with compensating assumptions and modeling constraints."

[56] Importance analysis in PRA is specific to the generation of cutsets quantified from fault- and event-tree models. Because cutsets are not generated as part of the TNPP Package transportation PRA (for reasons explained in Section 5.3.1), importance analysis as defined for PRA (using factors like Fusel Vesely and Risk Achievement Worth) cannot be used.

It is noteworthy, concerning parametric uncertainty analysis, that RG 1.174 and Section 6.4 of NUREG-1855, Revision 1 state that for a Capability Category II risk evaluation (which is generally required by RG 1.200, Revision 3 for risk-informed applications), the mean values of the risk metrics (total and incremental values) need to be compared to the risk acceptance guidelines. However, PRA results for nuclear power plants are typically calculated and reported as point values. Therefore, risk-informed programs developed based on RG 1.174 must determine mean values from the propagation of parametric probability distributions of the PRA inputs (e.g., failure rates) into the quantified PRA results or show that there is an insignificant difference between using the point values and using the mean values as far the decisionmaking is concerned. The risk results for nuclear power plant risk-informed applications are commonly presented as CDF and LERF because, as explained above, the risk acceptance guidelines in NUREG-1.174 use those metrics. A formal propagation of uncertainty may not be required if it can be demonstrated that there are negligible differences in the comparison of point values to risk evaluation guidelines and comparison of mean values to the risk evaluation guidelines.

In nuclear power plant PRAs, point estimate PRA results are commonly calculated and reported, but they are typically lower than the mean values and do not account for the state-of-knowledge correlation between nominally independent basic event probabilities that come from the same dataset. However, this effect is specific to generation of cutsets (combinations of failure events) from fault- and event-tree modeling, and as mentioned above, cutsets are not generated as part of the TNPP Package transportation PRA for reasons explained in Section 5.3.1.

In any event, TNPP Package transportation PRA is a new endeavor, and there are not enough data to fully perform parametric uncertainty analysis for certain likelihood determinations for bounding representative accidents. Moreover, this would be very burdensome to perform for radiation dose consequence analyses which is not currently required for PRAs developed in risk-informed applications for the current fleet. That said, for TNPP Package transportation accident frequency development many of the bounding accident scenario frequency estimates are founded on very large truck accident rates for which data exist. Therefore, the base accident frequency for very large trucks driven in the five states of the assumed route used in the TNPP Package transportation PRA could be examined for variability. This variability could then be assessed for its potential impact on the conclusions about risk for bounding representative accidents.

In addition to quantitative assessment, Section 5.1 of NUREG-1855 discusses the need to identify and assess key sources of model uncertainty. It states:

> A source of model uncertainty is labeled key when it could impact the PRA results that are being used in a decision and, consequently, may influence the decision being made. Therefore, a key source of model uncertainty is identified in the context of an application. This impact would need to be significant enough that it changes the degree to which the risk acceptance criteria are met and, therefore, could potentially influence the decision. For example, for an application for a licensing basis change using the acceptance criteria in RG 1.174, a source of model uncertainty or related assumption could be considered “key” if it results in uncertainty regarding whether the result lies in Region II or Region I or if it results in uncertainty regarding whether the result becomes close to the region boundary or not.

For TNPP Package transportation PRA, the concept cited above of being “close to a Region I or II boundary” can be interpreted as being “close to exceeding the risk evaluation guidelines,” which are defined by likelihood and radiation dose consequences. The need to identify and assess key model uncertainty is consistent with the requirement in HLR-QU-E of the PRA standard that stipulates “Uncertainties in the PRA results shall be characterized. Sources of model uncertainty and related assumptions shall be identified, and their potential impact on the results understood.” This assessment and identification process is further discussed in the next section.

In summary, the kinds of uncertainty assessments performed for the TNPP Package transportation PRA consists of the (1) qualitative assessments and dispositions of PRA model uncertainties based on review of PRA modeling assumptions and inputs performed for the sensitivity studies, (2) quantitative sensitivity assessment of key sources of uncertainty performed for the sensitivity studies, and (3) evaluation of the impact of the variability in the very large truck accident frequency used in the TNPP Package transportation PRA on the conclusions about risk for bounding representative accidents. These assessments are discussed in Section 10.2.

10.2 Uncertainty Analysis Performed for the Demonstration TNPP Package PRA

This section focuses on epistemic uncertainty (i.e., uncertainties in the formulation of the PRA model). NUREG-1855 provides the general guidance on the types of uncertainty, which include parametric, model, and completeness uncertainties. The guidance addresses:

- How to treat parameter uncertainty and its propagation
- How to assess model uncertainty by assessing the impact on PRA results and insights used to support risk-informed decisions
- How to assess completeness uncertainty using conservative or bounding analysis.

The uncertainty analyses performed for the TNPP Package transportation PRA consist of the identification and evaluation of key sources of modeling uncertainty from PRA modeling assumptions and inputs, as described in Section 10.2.1, and the quantitative evaluation of the impact of the variability in the very large truck crash data on the conclusions about risk of TNPP Package transportation accidents, as described in Section 10.2.2.

10.2.1 Identification and Assessment of Key Sources of Modeling Uncertainty

This section describes the mix of qualitative and then quantitative evaluation of key sources of modeling uncertainty and their impacts on the risk results, which was essentially performed as part of the sensitivity analysis described in Section 9.0.

The qualitative assessment and disposition of sources of PRA model uncertainties identified from review of PRA modeling assumptions and inputs was the first stage of the sensitivity study, as described in Section 9.1.1. As such, it also constitutes the identification and assessment of key sources of TNPP Package PRA modeling uncertainty. This assessment is presented in Table 9.1, Table 9.2, and Table 9.3 for different sets of PRA assumptions and inputs. The dispositions used in those tables to evaluate the source of uncertainty are primarily a combination of the concepts cited above for evaluating model uncertainty and completeness uncertainty. In practice, this often consisted of asking the question: “How would the specific source of uncertainty change the insights generated by the TNPP Package transportation PRA if the assumption or input were set to a bounding or conservative value?” When the answer to this question was that there would be no change in the conclusions about risk from the bounding representative accident or that the impact would be negligible, then the source of uncertainty was screened against further consideration. If it was concluded that an impact was possible, it was identified as the source for a “Candidate Study,” and a sensitivity study was performed to determine the quantitative extent of the impact. The exception was that for certain PRA inputs identified as sources for a “Candidate Study” the possibility of defining an applicable sensitivity study was investigated but found infeasible to perform (e.g., for lack of data). These exceptions were primarily related to studies in support of possible compensatory actions, as addressed in detail in Sections 9.1.2.5 and 9.1.2.6.

The quantitative assessment of the impact of unscreened sources of uncertainty is the second stage of the sensitivity study described in Section 9.2. Using the terminology defined in NUREG-1855 and cited above, the sources of uncertainty for which quantitative sensitivity was performed could be considered candidate “key” sources of uncertainty. (Other quantitative treatment of sources of uncertainty is described in Section 10.2.2 based on parametric uncertainty analysis.) The results of the quantitative sensitivity studies of the impact of key sources of uncertainty on the risk results are summarized at the end of Section 9.3. In many cases, the sensitivity study demonstrated that the conclusions about risk from the bounding representative accidents were not affected by the sensitivity study results. In other cases, the results can be used to inform compensatory measures.

10.2.2 Evaluation of the Impact of the Variability in the Very Large Truck Crash Data

This section describes the evaluation of the impact of the variability in the very large truck accident frequency used in the TNPP Package transportation PRA on the conclusions about risk for bounding representative accidents.

As discussed above, TNPP Package transportation PRA is a new practice and there are not enough data to fully perform parametric uncertainty analysis for certain likelihood determinations for certain bounding representative accidents. However, many of the bounding accident scenario frequency estimates are founded on very large truck accident rates for which data exist. Therefore, the base very large truck accident frequency is examined below for its variability and its potential impact on the conclusions about risk for bounding representative accidents. It should be noted that uncertainty analysis Is not proposed for radiation dose consequences for the reasons described in Section 10.1.

In Section 6.2.1 of this report, Table 6.6 displays the very large truck highway mileage for each of the five states of the proposed route and for the 3 years the data were collected (i.e., 2017, 2018, and 2019). Table 6.7 in the same section displays the number of very large truck events for each of the five states and the 3 years of data. From the data presented in Table 6.6 and Table 6.7, crash frequencies were calculated for each state of the proposed route for the 3 years the data was collected, as displayed below in Table 10.1. Inspection of the accident frequencies presented in Table 10.1 shows that the highest frequency is 7.9E-07 per year for Colorado in 2019. This rate is about 41 percent higher than the average crash rate of 5.59E-07 per year.

Table 10.1. Rates of Very Large Truck Crashes by the Compiled States and Years[a]

State	2017 Events per Mile	2018 Events per Mile	2019 Events per Mile	Accident Rate per Mile
Colorado	6.51E-07	7.39E-07	7.90E-07	7.27E-07
Idaho	6.75E-07	5.53E-07	5.44E-07	5.90E-07
New Mexico	3.59E-07	5.75E-07	7.73E-07	5.21E-07
Utah	3.93E-07	4.17E-07	3.81E-07	3.97E-07
Wyoming	6.84E-07	6.09E-07	7.33E-07	6.75E-07
All 5 states	6.51E-07	7.39E-07	7.90E-07	5.59E-07

(a) Source: Derived from Table 6.6 and Table 6.7, which are from the Motor Carrier Management Information System.

Using a smaller time interval (e.g., a monthly breakdown of the crash data), or some other approach, more data points could be generated than the 15 values shown in Table 10.1 and a probability distribution might be developed. Once a distribution is established, by whatever means, it can be used to explore the impact of this source of uncertainty on the TNPP Package transportation PRA risk results. For example, the 95th percentile could be selected to determine what the high-end impact on the risk results may be. That said, for this study another approach is used that does not involve the development of a probability distribution for the crash data but provides similar insights. In this alternate approach, the frequency of applicable bounding representative accidents (i.e., those founded on crash data) were increased by 41 percent to match the highest value presented in Table 10.1.

The results of increasing the accident frequency by 41 percent for applicable bounding representative accidents and comparing the resulting risk to the risk evaluation guidelines are presented in Table 10.2 through Table 10.7.

Both the original accident frequency and the increased frequency are shown in the second column from the right. These tables demonstrate that there is minimal change in the conclusions about risk (i.e., the risks of accidents that were acceptable in the baseline study are still acceptable and the risks of accidents that were unacceptable in the baseline study are still equally unacceptable). This is because the increased frequency remained in the same risk evaluation guideline accident frequency interval as that defined in the baseline case. The exception to this is the accident frequency for the BRA 4M uncertainty analysis presented in Table 10.3, in which the increased frequency resulted in a decrease in the risk evaluation guideline radiation dose limit, as shown in the right-hand column, BRA 4M results in risk to worker slightly above the risk evaluation guidelines.

In summary, Table 10.2 through Table 10.7 demonstrate that there is minimal change in the conclusions about risk (i.e., the risks of bounding representative accidents that were acceptable in the baseline study are still acceptable and the risks of bounding representative accidents that were unacceptable in the baseline study are still equally unacceptable). The exception being for BRA 4M, for which the uncertainty analysis presented in Table 10.3 shows that the risk to the worker is slightly above the risk evaluation guidelines.

This quasi-uncertainty analysis was performed on the base highway accident rate because not enough data was found to generate a probability distribution for the base highway accident rate.

Table 10.2. Uncertainty Analysis 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	80.9	18.5		
Core Structure	5.2E-01	1.3E-01		
Cooling System	3.1E-01	6.3E-02		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		≥0.1 and <1 rem TEDE for a member of the public ≥2 and <5 rem TEDE for a worker when the accident frequency is ≤1E-03 and >1E-04
Total Dose	**87.7**	**18.8**		
Accident Frequency assuming one trip per year (from Table 6.16)			7.1E-05	
Accident Frequency multiplied by 41% to match highest state and year combination			**1.0E-04**	No change in acceptability based on comparison to risk evaluation guidelines
COMPARISON TO RISK EVALUATION GUIDELINE				Unacceptable

MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).

Table 10.3. Uncertainty Analysis 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 4.7 of this Report
Accident Consequence by MAR contribution (Radiation dose from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	6.5E-03		
Cooling System	9.3E-03	1.9E-03		
Contribution from Unreleased Material				≥0.1 and <1 rem TEDE for a member of the public ≥2 and <5 rem TEDE for a worker when the accident frequency is ≤1E-03 and >1E-04
Degraded shielding	6.0	6.9E-02		
Total Dose	**6.0**	**7.7E-02**		
Accident Frequency assuming one trip per year (from Table 6.16)			9.7E-05	

Accident Frequency multiplied by 41% to match highest state and year combination	**1.4E-04**	Worker risk changed from acceptable to unacceptable from comparison to risk evaluation guidelines
COMPARISON TO RISK EVALUATION GUIDELINE		Unacceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).		

Table 10.4. Uncertainty Analysis 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥0.1 and <1 rem TEDE for a member of the public ≥2 and <5 rem TEDE for a worker when the accident frequency is ≤1E-03 and >1E-04
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	0	0		
Cooling System	0	0		
Total Dose	**0**	**0**		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**6.0**	**6.9E-02**		
Accident Frequency assuming one trip per year (from Table 6.16)			3.3E-05	
Accident Frequency multiplied by 41% to match highest state and year combination			**4.7E-05**	No change in acceptability based on comparison to risk evaluation guidelines
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 10.5. Uncertainty Analysis for BRA 5H – Hard-Impact Accident and an Ensuing Fire

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker when the accident frequency is ≤5E-07
MAR contribution from released material				
TRISO Fuel	80.9	18.5		
Core Structure	5.2E-01	1.3E-01		
Cooling System	3.1E-01	6.3E-02		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**87.8**	**18.8**	2.6E-08	

Accident Frequency assuming one trip per year (from Table 6.16)		
Accident Frequency multiplied by 41% to match highest state and year combination	**3.7E-08**	No change in acceptability based on comparison to risk evaluation guidelines
COMPARISON TO RISK EVALUATION GUIDELINE		Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).		

Table 10.6. Uncertainty Analysis for BRA 5M – 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥25 and <750 rem TEDE for a member of the public ≥100 and <750 rem TEDE for a worker when the accident frequency is ≤1E-06 and >5E-07
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	6.5E-03		
Cooling System	9.3E-03	1.9E-03		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**6.0**	**7.7E-02**		
Accident Frequency assuming one trip per year (from Table 6.16)			5.9E-07	
Accident Frequency multiplied by 41% to match highest state and year combination			**8.3E-07**	No change in acceptability based on comparison to risk evaluation guidelines
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).				

Table 10.7. Uncertainty Analysis 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 4.7 of this Report
Accident Consequence (from Table 7.6)				≥750 rem TEDE for a member of the public ≥750 rem TEDE for a worker when the accident frequency is ≤5E-07
MAR contribution from released material				
TRISO Fuel	81.7	18.7		
Core Structure	1.6E-01	1.5E-01		
Cooling System	7.5E-2	7.5E-02		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	**88.7**	**18.9**	7.1E-08	

Accident Frequency assuming one trip per year (from Table 6.16)		
Accident Frequency multiplied by 41% to match highest state and year combination	**1.0E-07**	No change in acceptability based on comparison to risk evaluation guidelines
COMPARISON TO RISK EVALUATION GUIDELINE		Acceptable
MAR = material at risk; rem = roentgen equivalent man; TEDE = total effective dose equivalent; TRISO = tri-structural isotropic (particle).		

11.0 Defense-in-Depth and Safety Margin Concerns

The 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 is expected for transportation of TNPP Packages. This section defines the defense-in-depth philosophy and includes discussion of safety features and controls that are credited in the TNPP Package transportation PRA. Of special note is the identification of potential compensatory measures used to offset the residual risk associated with TNPP Package transport and the uncertainty associated with risk calculations. This section also describes the philosophy of incorporating a safety margin into design and operation, and how both of 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 Package transportation PRA, this applies to ensuring that there is a sufficient safety margin to account for modeling and data uncertainties. Accordingly, Section 11.1 discusses the defense-in-depth philosophy as it supports risk-informed decisionmaking in concert with results and insights from the TNPP Package transportation PRA, and Section 11.2 discusses identification of suggested potential compensatory measures. Section 11.3 discusses the safety margin philosophy as it supports risk-informed decisionmaking in concert with results and insights from the TNPP Package transportation PRA and defense-in-depth.

11.1 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 elements of the defense-in-depth philosophy for this application of the exemption process (10 CFR 71.12, "Specific exemptions") are: (1) the robustness of the TRISO fuel and containment, (2) support of safety functions during transport do not rely on active systems, (3) the TNPP Package transportation risk is quantified and shown to be low, (4) sensitivity studies show that most sources of uncertainty in PRA modeling assumptions and inputs do not impact the conclusions about risk from accidents, and (5) because compensatory measures will be administered and not credited in the TNPP Package PRA to reduce risk to the worker and the public and uncertainty about risk through preventive and mitigative actions and 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 that, relative to the defense-in-depth philosophy, the key is creating multiple independent and redundant layers of defense to compensate for potential human error 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 Package transportation:

1. Preserve a reasonable balance among the layers of defense.

   A reasonable balance among the layers of defense is maintained because no given layer by itself is relied on primarily for nuclear safety. The layers of defense are the TRISO fuel itself; the containment system, which consists of the reactor coolant boundary and isolation devices for disconnected piping; the low risk associated with TNPP Package transportation accidents; and the implementation of compensatory actions not credited in the PRA. The TRISO fuel is designed to be tolerant of the elevated temperatures and heat that can occur in nuclear power plant accidents. However, it remains to be verified that release of radiological material from a transportation accident that could involve significant mechanical forces on the fuel is minimal or nil. The same kind of verification is also still needed for the reactor coolant boundary for transportation accidents. However, even though the TNPP Package is not expected to meet all 10 CFR 71.55 (b) tests associated with hypothetical accident conditions, it is expected, nonetheless, to possess a very high level of robustness.

2. Preserve adequate capability of design features without an overreliance on programmatic activities such as compensatory measures.

   For TNPP Package transportation, the design goal for the TNPP Package is to prevent release of radiological material, loss of shielding, and criticality without overreliance on programmatic controls and compensatory measures. The TNPP Package transportation PRA is performed to show that the risk of TNPP Package transportation is relatively low even when programmatic activities such as compensatory measures are not credited. Additionally, sensitivity studies show that most sources of uncertainty in PRA modeling assumptions and inputs do not affect the conclusions about risk from transportation accidents.

3. Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.

   The primary safety functions of containment, shielding, and maintaining criticality safety are performed by different design features and components. None of the features and components that perform these functions rely on (i.e., are dependent on) active alternating current power. Shielding and containment are afforded by a combination of the robust design of the TNPP itself independent from the metal CONEX box-like structure of the Reactor Module configured as the TNPP Package. Though the Reactor Module does not provide a strong containment function (e.g., it is not leak tight), it would absorb much of the energy of a crash involving impact, and therefore, protects the reactor coolant boundary (which does provide a containment function) from more significant damage. Much of the radiation shielding is built into and afforded by the reactor vessel itself but is augmented by transport shielding in the walls of the CONEX box-like structure of the Reactor Module. Criticality safety is maintained by a completely different set of design features that help prevent reactivity insertion.

4. Preserve adequate defense against potential common cause factors (CCFs).

   An adequate defense against potential CCFs is maintained given that the safety functions during transport do not rely on redundant trains or circuits, and there are no active systems needed when the TNPP is in shut down. Therefore, the notion of protecting against CCFs is not as relevant for TNPP Package transportation as it is for operating nuclear power plants for which there are numerous redundant trains and circuits to ensure that even if one train or circuit fails sufficient functionality remains.

5. Maintain multiple fission product barriers.

   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 about very high temperatures that would challenge the TRISO fuel during a TNPP Package transportation accident. The contribution to worker and public radiation dose in a transportation accident from other MAR (non-TRISO fuel radiological material) is significantly less than the TRISO fuel contributions (see Table 7.6).

6. Preserve sufficient defense against human errors.

   The possibility of human error is explicitly identified and addressed in the TNPP Package transportation PRA and the risk associated with human error (e.g., packaging errors) is quantified. As such, the insights from the TNPP Package transportation PRA can be used to implement administrative controls on operator actions to prevent error. This includes the results of the sensitivity studies that, in general, show most sources of uncertainty in PRA modeling assumptions and inputs do not affect the conclusions about risk from accidents.

As discussed in Section 4.0, the NRC proposes guidance in the RIDM report (NRC 2008) concerning defense-in-depth. 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 cited above) are dispositioned below even though the risk from TNPP Package transport is shown by the TNPP Package PRA to generally not be a high-risk activity.

1. Make sure key safety functions do not depend on a single element of design or operation.

   There is no single element of the design or operation that is relied on to ensure key safety functions remain intact. Again, the safety functions that need to be protected are containment, shielding, and maintaining criticality safety. Shielding and containment are afforded by a combination of the robust design of the TNPP itself independent of the metal CONEX box-like structure of the TNPP Package and they complement each other in the way described above. Criticality safety is maintained by the design features that help prevent reactivity insertion largely independent from features created for shielding and containment (e.g., rod locking mechanisms). The PRA results demonstrate that the risk of one of these safety systems failing is low. Additionally, the reliability of these safety functions will be supported by administrative transportation controls (e.g., it is expected that there will be an escort vehicle in the front of and behind the truck carrying the TNPP Package) and other compensatory measures that are not credited in the TNPP Package PRA.

2. Use redundancy, diversity, and independence to improve reliability and/or avoid common-mode failure, when necessary, to make sure safety is maintained.

   The defenses-in-depth philosophy of redundancy, independence, and diversity related to TNPP Package transport (identified in the NRC RIDM report [NRC 2008]) and the concept of avoidance of CCF are addressed in the first list from RG 1.200 as Item 3 and Item 4, respectively. As described above, system redundancy, independence, and diversity and avoidance of CCF are preserved and there is little opportunity for CCF. Unlike operating nuclear power plants for which there are numerous redundant trains and circuits to ensure that even if one train or circuit fails sufficient functionality remains, there are no active or passive trains or circuits that are relied upon to fulfill safety functions.

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 11.3.

4. Conduct regulated activities at locations that facilitate protection of the 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 are highly regulated.

5. Provide time for recovery operations.

   It is expected that TNPP Package transportation will include a recovery plan for possible transportation incidents and accidents, and the transportation workers and personnel should be trained to 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). It is expected that the TNPP Package would be transported using escorts in the front and back of the truck carrying the TNPP Package and that personnel in these vehicles would be trained in the emergency response procedures. Results of the TNPP Package PRA and associated sensitivity studies can be used to enhance recovery response. For example, sensitivity studies were explicitly performed that address assumptions made about the distance of the worker and public from the location of the accident and the duration that they were exposed to the release (or direct radiation from unreleased material).

6. Make sure the design and operation have both accident prevention and mitigation measures.

   Accident prevention includes preventing the release of radioactive material, preventing the loss of shielding, and preventing a criticality in a TNPP Package accident, all of which are addressed in the design. The transport of the TNPP Package will be supported by transportation procedures with specific controls to reduce risk and 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 of natural phenomena.

7. Make sure 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. Also, as stated above, the CONEX box-like structure of the TNPP Package provides transport shielding and some barrier to release of fission products, but more importantly it protects the reactor coolant boundary from damage in a transportation accident. The reactor is in a shutdown state, so there is no concern in a TNPP Package accident about very high temperatures that might challenge the TRISO fuel, or the radiological material diffused into the reactor or plated-out in the reactor coolant boundary during a TNPP Package transportation accident. Moreover, the TNPP Package PRA shows that the risk from transportation is relatively low. The PRA shows that the likelihood of the occurrence of TNPP Package accidents that produce the highest consequences is also very low.

11.2 Identification of Potential Compensatory Measures

As described in Section 3.0, 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.

This section discusses potential compensatory measures that are:

- Explicitly credited in the TNPP Package transportation PRA as an underlying assumption in the baseline PRA, and therefore are reflected in the baseline PRA results
- Credited in a sensitivity study that shows the quantitative risk decrease (if implemented as a defense-in-depth measure
- Not credited quantitatively but could be qualitatively credited to decrease risk.

The list of possible generic compensatory measures identified by the vendor is provided below[57] and is amended by additional compensatory measures identified as part of the Hazardous Condition Evaluation. This list of compensatory measures is explicitly dispositioned, either qualitatively or by being addressed in a sensitivity study, as discussed in Section 9.1 of this report.

- Escort the reactor forward and aft for the entire route. Army to provide escorts.
- Choose a route that avoids bodies of water. 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 their condition.
  - Close each 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 the waterway to traffic while the 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.
- Implement real-time health monitoring (see discussion below).
- Install fire detection and suppression on the transport vehicle and in the Reactor Module of the TNPP Package.
- Implement travel restrictions, such as speed controls due to road or weather conditions.

[57] From BWXT Final Design Report App I.2 Section 10.10, page 85.

Though a planned HMIS will provide real-time parameter monitoring of a TNPP Package during transport, the design is not known at this time. It is anticipated that the HMIS will be designed to detect conditions signaling that a TNPP Package transportation accident (e.g., a leak from containment) has occurred or could occur. Detection would be based on monitoring parameters such as high levels of airborne radioactivity or direct radiation, loss of pressure in the reactor containment boundary, increase in heat in the reactor coolant boundary, and rod control position. Real-time monitoring systems are 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 Standard S-2043 (*Performance Specification for Trains Used To Carry High-Level Radioactive Material* [AAR Standard S-2043 2017]).

It is noteworthy that the TNPP Package shipment will be of sufficient weight that it will be subject to heavy haul permitting in each state through which it passes, and it may be subject to super load 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 that these specific requirements exist, they are reflected in the highway accident rates presented in Section 6.0 and used in the TNPP Package transportation PRA.

11.3 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 verified relative to 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 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 indicated above, one way to evaluate the safety margin in a microreactor transportation package transportation risk assessment is to make sure that the codes and standards used in the analyses supporting the risk assessment have a high degree of technical quality. The TNPP Package transportation PRA is not yet a well-developed methodology, and in fact, this application advances the state-of-practice, because 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 Package transportation PRA. That said, the techniques used in the TNPP Package 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 nonreactor nuclear facilities (DOE 2013). Also, the approach for determining the radiation dose from a given source term resulting from a TNPP Package transportation accident comes from the Q System described 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 used in 10 CFR Part 71. This guidance is described in detail in Sections 5.1.3 and 7.3 of this report. The hazard analysis that was used to identify and define the bounding representative accidents for TNPP Package 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]).

As stated, RG 1.174 indicates the second way to evaluate safety margin in a TNPP Package transportation PRA is to make sure that there is sufficient safety margin in the analyses results that support the PRA to account for modeling and data uncertainties. An SAR, supporting safety basis analytical evaluations, and physical testing of the TNPP Package has not yet been developed or performed. The deterministic design analyses that are performed to support the SAR that are used in the TNPP Package transportation PRA needs to employ a safety margin philosophy. Deterministic concerns include the finite element analyses used to determine the fragility of the package relative 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. The safety margin was assessed to the extent possible by performing sensitivity studies, as described in Section 9.0, that address the impact of the uncertainty associated with deterministic inputs to the TNPP Package transportation PRA.

Commonly used tools and approaches were used in the TNPP Package transportation PRA, but, as discussed above, there is a need for a PRA standard and corresponding guidance for performing a TNPP Package transportation PRA (as discussed later in Section 12.0 of this report; however, given that this is a first-of-a-kind endeavor, with limitations and uncertainty in the inputs, the TNPP Package transportation PRA was developed using a "best judgment" approach that erred on the side of conservativism in (1) the identification of TNPP Package transportation accidents, (2) the estimation of accident likelihood and application of the accident data, and (3) the estimation of the accident consequences. The safety margin of the deterministic input to the TNPP Package transportation PRA that comes from the design and SAR can be assessed when that information is available and by performing sensitivity studies to test the influence of the uncertainty in PRA inputs.

In summary, the assessment of defense-in-depth and safety margin philosophies concludes that they can generally be applied consistent with NRC guidance and expectations in support of TNPP Package transportation PRA and to its application to regulatory approval of the TNPP transportation package. However, given that this is a first attempt to apply these philosophies to this kind of application, further development of the concept of defense-in-depth to this kind of application is expected, because traditional approaches that worked well in support of risk-informing applications for the current fleet of nuclear power plants may not apply in the same way.

12.0 Technical Adequacy of Transportation PRA

This section discusses the technical adequacy of the TNPP Package transportation PRA, which includes identification of applicable national standards and the use of an independent peer review process. 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.

The NRC has issued an RG that removes the need for an in-depth review by NRC of the PRA for LWRs in a risk-informed application (RG 1.200, Revision 3 [NRC 2020]). RG 1.200 states the following concerning regulatory decisionmaking for LWRs:

> This regulatory guide (RG) describes one approach acceptable to the U.S. Nuclear Regulatory Commission (NRC) staff for determining whether a base probabilistic risk assessment (PRA), in total or in the portions that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors (LWRs). When used in support of an application, this RG will obviate the need for an in-depth review of the base PRA by NRC reviewers, allowing them to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application.

This approach is facilitated by meeting PRA industry consensus standards with exceptions and clarifications by the NRC as provided in RG 1.200, Revision 3:

> This RG endorses, with staff exceptions and clarifications, national consensus PRA standards provided by standards development organizations, and guidance provided by nuclear industry organizations.

The NEI has issued guidance on how to perform an independent industry peer review of internal events, internal fires, and external event PRAs in the form NEI 17-07, Revision 1, *Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard*, (issued August 2019 [NRC 2019]). This guidance is an update of a collection of separate guidance documents issued earlier including guidance on how to close a finding made by an independent peer review panel. Per the guidance provided in NEI 17-07, an independent peer review team (that meets the criteria for independence and expertise defined in the guidance) can generate findings or suggestions in a form referred to as Facts and Observations about a PRA in cases for which a PRA standard SR does not appear to be fully met or is not met at the appropriate Capability Category (CC). (In general, the appropriate CC level for risk-informed application to NRC is CC-II). All finding level Facts and Observations must be resolved for the application they support for the NRC to find a risk-informed application acceptable. The NRC, in RG 1.200, Revision 3, has endorsed the use of NEI 17-07, Revision 2 for ensuring the technical adequacy of PRAs supporting risk-informed applications.

The guidance in NEI 17-07 also address details of process such as the criteria for deciding when a PRA requires a new independent peer review or a Focused Scope peer review of a portion of the PRA that has been updated, particularly those portions of the PRA where new PRA methods are employed (i.e., new methods as defined in the guidance). It is noteworthy that the process described above has been used to ensure the appropriate level of PRA technical adequacy for more than a decade for a very large number of risk-informed licensing amendment requests to the NRC by the industry.

Of course, this approach is predicated on having a mature and well-developed set of PRA requirements in the form of a PRA standard. RG 1.200, Revision 3 endorses the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard in ASME/ANS RA-Sa-2009, “Addenda to ASME RA-S-2008, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications” (ASME/ANS 2009). This standard is organized by hazard type (e.g., internal events PRA versus internal fire or seismic PRA). Each PRA type is broken down by PRA technical elements (such as initiating events, system analysis, HRA, and data analysis), which are further broken into HLRs. HLRs are broken down further into SRs, which are then levied against the PRA being reviewed.

Detailed development of a PRA for TNPP Package transportation or other packages that do not meet (or do not quite meet) the deterministic requirements of 10 CFR Part 71 is a recent undertaking, so methods required to perform TNPP Package transportation accident PRA will likely evolve. Therefore, this beginning stage is a good time to begin development of TNPP Package transportation PRA standards. In many regards, a TNPP Package transportation PRA is much less complex than one for a nuclear power plant because the engineered systems required to fulfill safety functions during transport are less complex and predominantly passive. However, this activity is novel at this stage and the PRA presents different kinds of challenges than PRAs for nuclear power plants.

It is suggested that a PRA standard on TNPP Package transportation PRA (or PRA for transportation of any package that does not meet the deterministic requirement of 10 CFR Part 71) would greatly aid the NRC approval process. If a process were patterned after the one followed for risk-informed application for nuclear power plants, then adherence to the PRA standard as confirmed by an independent peer review would help ensure the technical adequacy of the transportation risk assessment, promote uniformity, and could significantly expedite the regulatory approval process. It could also free up NRC staff to be able to focus on other technical issues associated with the application besides PRA adequacy.

13.0 Risk Conclusions and Insights

This section presents conclusions about the use of proposed risk evaluation guidelines, the risk of TNPP Package transport with irradiated fuel and insights drawn from the baseline PRA results, the sensitivity study results, and the uncertainty analysis results. It also makes general conclusions about the use of defense-in-depth and safety margin for this application and summarizes a suggested approach to ensuring the technical adequacy of a transportation PRA.

The proposed risk evaluation guidelines were developed to be consistent with existing DOE and NRC guidance and regulations judged applicable and to align with NRC nuclear safety goals and corresponding proposed QHOs and corresponding QHGs that have been proposed by NRC for activities like transportation but have not been endorsed. The benefit of having risk evaluation guidelines is that if TNPP Package transportation PRA results can be found to be acceptable by comparing them to the risk evaluation guidelines, then a key criterion for making a risk-informed decision is satisfied. If not, then controls, design changes, or compensatory measures can be devised from the comparison to reduce the unacceptable risk.

The results presented in Sections 8.1 and 8.2 show that the risks of the bounding representative accidents, except for BRA 3, meet the risk evaluation guidelines. The risk for BRA 3, which is a severe collision event with a heavy vehicle or an unyielding object, exceeds the risk evaluation guidelines for both the worker and public. However, the results of one of the sensitivity studies presented in Section 9.2.2 suggest that for the one bounding representative accident (i.e., BRA 3) for which the risk to the public is unacceptable, increasing the isolation distance for the public from 25 m to 100 m (and assuming that distance in the consequence analysis) yields acceptable risk results for the public, though the risk to the worker for BRA 3 would still be unacceptable.

Another sensitivity study shows that decreasing the distance of the public from the accident to be the same distance as the worker from the accident does not the change the conclusions about whether the risk of the representative bounding accident meets the risk evaluation guidelines with one exception. The exception is that for BRA 4M and BRA 4L the risk to the public would slightly exceed the risk evaluation guidelines dose of 5 rem at their respective accident frequencies. This is significant because a member of the public may be involved in the accident if the vehicle that the person is driving is part of the collision, and therefore, the person is initially within the 25 m or 100 m cordoned off area.

The risk of BRA 4M, which is a medium-impact accident defined as a severe collision with a light vehicle, is just slightly below the risk evaluation guidelines due to external dose caused by degraded shielding from the impact energy of the collision. Uncertainty in the allocation of events considered to be light-impact accidents (BRA 4L) versus medium-impact accidents (BRA 4M) could affect the conclusions about risk for BRA 4M. However, controls, future design refinements of the shielding or enhanced consequence calculations might be used to decrease the estimated risk of BRA 4M.

Concerning criticality events, the TNPP Package transportation results for BRA 9A and BRA 9B (flooded criticality) show that the risk evaluation guidelines are met due to the exceptionally low calculated frequency of such accidents. For a criticality event due to a reactivity insertion event (e.g., control rod withdrawal) caused by the impact energy of the accident (i.e., BRA 10), this study assumes the design goal of precluding a reactivity insertion event in a TNPP Package transportation accident will be met.

An important sensitivity study result is that allowing the TNPP reactor core to decay up to a year (or even somewhat less than a year) after it has been in operation for 3 years will ensure an acceptable level of risk for all bounding representative accidents evaluated in the study based on the proposed risk evaluation guidelines.

Another important sensitivity study result is that increasing the exposure duration for the worker and public from 30 to 60 minutes to a damaged TNPP Package has no impact on the conclusion about TNPP Package transportation risk. Bounding representative accidents that were acceptable in the baseline case based on frequency are still acceptable, accidents that are unacceptable in the baseline are still unacceptable, and the risk of the remaining postulated accidents would remain acceptable.

A final important sensitivity result is that risk conclusions about TNPP Package transportation are not very sensitive to the impact of uncertainty on estimating the source term factors given a specific definition of sensitivity discussed below. This alleviates a concern because source term factors have not been explicitly developed for TNPP Package transportation, so best judgment was needed to apply source term factors from for fuel cycle facilities and radiological waste containers. For this study the radiation dose consequences are considered “not sensitive” to the source term estimates if an increase of more than a factor of 1,000 is needed to exceed the risk evaluation guidelines and are considered “not very sensitive” if more than a factor of 100 is needed to exceed the risk evaluation guidelines. The risk of several other representative accidents considered to be bounding is either already unacceptable in the baseline case or acceptable based on accident frequency according to the baseline results. Therefore, the impact of this uncertainty is not important to the TNPP Package PRA risk conclusions. In this case, the risk to BRA 4M and BRA 4L would remain the same because all the contribution (for BRA 4M nearly all the contribution) to dose is from direct radiation from unreleased material (i.e., damage to the transport shielding) for BRA 4L.

This report found that even though compensatory measures examined in Section 9.1.2.5 for restriction of transport during extreme weather and in Section 9.1.2.6 for shipping at night during low traffic make intuitive sense, the impact of these controls could not be quantitatively assessed using the large truck crash datasets described in Section 6.2. Although environmental weather conditions, road conditions, and lighting conditions are recorded in a nationwide dataset for the crash entries, there are no corresponding data about very large truck traffic. Therefore, reduction in accident rates by avoiding these conditions cannot be shown. The fact that most accidents happen when the weather and roads are clear and during daylight hours is likely an indication that truck traffic is higher during clear weather. However, without further data, the possible reduction in crash rates for very large trucks for not driving during inclement weather or driving at night cannot be demonstrated and may not be true. That said, it should be noted that it is routine for radioactive material shipments (such as shipments of transuranic waste to the Waste Isolation Pilot Plant) to consider weather and road conditions and the time of day and the day of week prior to and during transport.

It is noteworthy that the evaluation described in Section 9.1 on identifying and defining candidate sensitivity studies was performed by explicitly examining all the PRA modeling assumptions that were made for various elements of the TNPP Package PRA for those that could affect the PRA results. This assessment provides confidence that even though a full parametric data uncertainty analysis was not performed the impact of PRA modeling uncertainty was addressed.

The uncertainty analyses performed for the TNPP Package transportation PRA consisted of:

- Qualitative evaluation of sources of modeling uncertainty from PRA modeling assumptions and inputs, and identification and quantitative evaluation of those considered “key” sources of uncertainty
- Quantitative evaluation of the impact of the variability in the very large truck crash data about the conclusions about risk of TNPP Package transportation accidents.

In many cases, the results of the evaluation of key sources of uncertainty, performed using sensitivity studies, demonstrated that the conclusions about risk from the bounding representative accidents were not affected by the sensitivity study results. In other cases, the results can be used to inform compensatory measures for regulatory approval. The results of the uncertainty analysis on the impact of the variability in the very large truck crash data on the conclusions demonstrate that with one exception there is no change in the conclusions about risk (i.e., the risks of bounding representative accidents that were acceptable in the baseline study are still acceptable and the risks of bounding representative accidents that were unacceptable in the baseline study are still equally unacceptable). The exception to this is that accident frequency for the BRA 4M uncertainty analysis case (as presented in Table 10.3) results in an increase that puts it into the next frequency interval of the risk evaluation guidelines for which there is a decrease in the risk dose limit. Consequently, BRA 4M results in risk to a worker slightly above the risk evaluation guidelines.

It is expected that key PRA parameters will be examined to establish bounding estimates unless the risk results (likelihoods and consequences) are demonstratively bounding.

Assessment of defense-in-depth and safety margin philosophies concluded that they can generally be applied consistent with NRC guidance and expectations in support of TNPP Package transportation PRA and to its application to regulatory approval of the TNPP transportation package. However, given that this is a first attempt to apply these philosophies to this kind of application, further development of the concept of defense-in-depth related to this kind of application is expected, because traditional approaches that worked well in support of risk-informing applications for the current fleet of nuclear power plants may not apply in the same way.

With regard to ensuring the technical adequacy of a transportation PRA, it is suggested that a PRA standard on TNPP Package transportation PRA (or PRA for transportation of any package that does not meet the deterministic requirement of 10 CFR Part 71) would greatly aid the NRC approval process. If a process were patterned after the one followed for risk-informed application for nuclear power plants, then adherence to the PRA standard as confirmed by an independent peer review would help ensure the technical adequacy of the transportation risk assessment, promote uniformity, and could significantly expedite the regulatory approval process. It could also free up NRC staff to focus on the difficult technical issues, such underlying design and modeling assumptions, as opposed to the form and structure of the PRA and corresponding methods.

In conclusion, the work summarized in this report demonstrates that a risk-formed approach using PRA to support a 10 CFR 71.12 exemption request for TNPP Package transport with irradiated fuel appears feasible and the PRA methods needed to support the request seem achievable. The demonstration PRA results appear to indicate that the risk from transportation of a TNPP Package with irradiated fuel is acceptably low. For the few cases where the risk acceptance guidelines are not quite met, the risk can be reduced with controls or design improvements and by using compensatory measures. The sensitivity studies and uncertainty analysis performed in support of this demonstration indicate that the sensitivity of the conclusions about risk from accidents to the uncertainty in PRA assumptions and inputs is small (in this case) and can also be reduced with controls or design improvements and by using compensatory measures. Finally, the assessment of defense-in-depth and safety margin philosophies in support of the risk-informed exemption demonstrates that the application of these philosophies is feasible. However, given these observations and the fact that a risk-informed approach using PRA for approval of TNPP Package transport carrying irradiated fuel is a first-of-its-kind endeavor, development of a PRA standard on TNPP Package transportation PRA (or PRA for transportation of any package that does not meet the deterministic requirement of 10 CFR Part 71) is recommended, because it would greatly aid the NRC approval process.

14.0 NRC Review History of This Report

On February 20, 2023, PNNL on behalf of the SCO asked the NRC in an email message (PNNL 2023a) to provide a review of an initial draft of this report (PNNL 2022). On April 19, 2023, PNNL received an email message forwarded by the SCO from the NRC (NRC 2023a) that contained a letter dated April 14, 2023, from the NRC to the SCO (NRC 2023b) transmitting a request for Information on the initial draft and asking that the response include a marked revision of the report updated to incorporate information associated with the requests. Enclosure 1 to the letter contained the information request, and Enclosure 2 contained observations that an applicant might find useful when using the approach in the report in an application for package approval.

On September 18, PNNL transmitted to NRC by email (PNNL. 2023b) responses to the request for information and an updated version of the report (PNNL. 2023c) that incorporated responses associated with the requests. Documentation of those the NRC request for information and the PNNL response is provided in Appendix C.

Additionally, the NRC ACRS Subcommittee reviewed the report. On November 17 and December 6, 2023, PNNL, the NRC and the SCO provided presentations to the ACRS Subcommittee and Full Committee in Washington, D.C., after their review. The presentations focused on the approach presented and demonstrated in the report and the acceptability of the approach to the NRC for licensing. The presentations provided in the November 17, 2023, meeting was to the Fuels, Material, and Structures Subcommittee. The SCO provided opening remarks about the purpose of the report, PNNL followed with a presentation on development and demonstration of the proposed risk-informed approach for regulatory approval of highway shipment of microreactor, and the NRC provided its review of the approach. Based on ACRS interest and questions about the approach, the SCO, PNNL, and the NRC were invited back for a meeting December 6, 2023, with the full committee to provide follow-up information of interest to the ACRS. A summary report dated December 22, 2023 (NRC 2023c) was written by the ACRS that includes discussion of the December 6 meeting and documents comments offered by the individual members of the ACRS. The December 6 version of the presentation contains refinements from the November 17 to clarify or correct technical issues of interest to the ACRS. The two presentations provided by PNNL on November 17 and December 6 to the ACRS Subcommittee and Full Committee are documented in this report in Attachment D.

The interaction with the ACRS helped inform the NRC review of the report and suggest to PNNL refinements that could be made to improve the report. Interaction with the ACRS is discussed in Appendix D.

This version of the report reflects incorporation of feedback from the NRC and self-identification by PNNL of needed improvements based on the review history described above and associated discussions.

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Appendix II – Engineering Drawings.
Appendix III – Engineering Documentation and Analyses.
Appendix IV – Safety in Design Bases and Reports.

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Appendix A – TNPP Inventory and Development

This appendix provides the results of the screening of the representative Transportable Nuclear Power Plant (TNPP) inventory[1] and development of the material at risk (MAR) based on this inventory. As described in Section 5.1.3 of this report using a screening process, Table A.1 includes all radionuclides greater than 0.01 percent 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 30 days, 60 days, 90 days, 1 year, and 2 years after 3 years of operation.

As described in Section 5.1.4 of this report, Table A.2 through Table A.6 present the MAR for cooling periods of 30 days, 60 days, 90 days, 1 year, and 2 years, respectively, after 3 years of operation based on where in the TNPP Package the radiological material exists, as explained in Section 5.1.4 (i.e., the tri-structural isotropic (TRISO) fuel, material diffused into the reactor core, and material in the coolant boundary).

Table A.1. Prototype TNPP Radionuclide Inventory (three sheets total)

Isotope	PELE Total 30 Days (Ci)	PELE Total 60 Days (Ci)	PELE Total 90 Days (Ci)	PELE Total 1 Year (Ci)	PELE Total 2 Years (Ci)
Ag-110m	9.34E+01	8.60E+01	7.91E+01	3.69E+01	1.34E+01
Ag-111	3.53E+02	2.17E+01	1.33E+00	0.00E+00	0.00E+00
Am-241	3.25E+01	3.57E+01	3.89E+01	6.73E+01	1.03E+02
Am-242m	1.11E+00	1.11E+00	1.11E+00	1.10E+00	1.10E+00
Am-243	1.88E-01	1.88E-01	1.88E-01	1.88E-01	1.88E-01
Ba-136m	1.30E+02	2.67E+01	5.50E+00	0.00E+00	0.00E+00
Ba-140	7.05E+04	1.38E+04	2.70E+03	0.00E+00	0.00E+00
Cd-113m	1.55E-01	1.54E-01	1.53E-01	1.48E-01	1.41E-01
Cd-115	1.04E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	4.30E+01	2.70E+01	1.69E+01	2.35E-01	0.00E+00
Ce-139	1.41E-01	1.21E-01	1.04E-01	0.00E+00	0.00E+00
Ce-141	1.81E+05	9.55E+04	5.04E+04	1.43E+02	5.97E-02
Ce-143	9.25E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	2.70E+05	2.51E+05	2.34E+05	1.20E+05	4.92E+04
Cm-242	2.26E+03	1.99E+03	1.75E+03	5.45E+02	1.16E+02
Cm-243	2.26E-01	2.25E-01	2.25E-01	2.21E-01	2.16E-01
Cm-244	5.51E+00	5.50E+00	5.48E+00	5.32E+00	5.12E+00
Cs-134	1.17E+04	1.14E+04	1.11E+04	8.61E+03	6.16E+03
Cs-135	2.59E-01	2.59E-01	2.59E-01	2.59E-01	2.59E-01
Cs-136	1.17E+03	2.41E+02	4.97E+01	0.00E+00	0.00E+00
Cs-137	2.53E+04	2.52E+04	2.52E+04	2.48E+04	2.42E+04
Eu-152	5.55E+00	5.53E+00	5.51E+00	5.30E+00	5.03E+00
Eu-154	4.64E+02	4.61E+02	4.58E+02	4.31E+02	3.98E+02
Eu-155	3.31E+02	3.27E+02	3.23E+02	2.90E+02	2.50E+02

[1] BWXT spreadsheet "B1.34-NuclideConcentrations(Ci)-Fuel.xlsx" provided on August 11, 2022.

Isotope	PELE Total 30 Days (Ci)	PELE Total 60 Days (Ci)	PELE Total 90 Days (Ci)	PELE Total 1 Year (Ci)	PELE Total 2 Years (Ci)
Eu-156	1.89E+03	4.80E+02	1.22E+02	0.00E+00	0.00E+00
Gd-153	2.17E+00	1.99E+00	1.82E+00	8.24E-01	2.88E-01
H-3	1.05E+02	1.04E+02	1.04E+02	9.97E+01	9.42E+01
I-131	1.46E+04	1.09E+03	8.20E+01	0.00E+00	0.00E+00
I-132	4.23E+02	6.42E-01	0.00E+00	0.00E+00	0.00E+00
In-115m	1.18E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	2.76E+03	2.74E+03	2.73E+03	2.60E+03	2.44E+03
La-140	8.12E+04	1.59E+04	3.11E+03	0.00E+00	0.00E+00
Mo-99	1.91E+02	9.90E-02	0.00E+00	0.00E+00	0.00E+00
Nb-95	3.40E+05	2.87E+05	2.30E+05	1.52E+04	2.98E+02
Nd-147	1.97E+04	2.97E+03	4.47E+02	0.00E+00	0.00E+00
Np-237	6.24E-02	6.24E-02	6.24E-02	6.24E-02	6.24E-02
Np-239	4.49E+02	2.53E-01	1.88E-01	1.88E-01	1.88E-01
P-32	3.83E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pa-233	6.95E-02	6.57E-02	6.39E-02	6.24E-02	6.24E-02
Pm-146	4.41E-03	4.37E-03	4.32E-03	3.93E-03	3.47E-03
Pm-147	6.11E+04	6.00E+04	5.87E+04	4.81E+04	3.70E+04
Pm-148m	4.72E+03	2.85E+03	1.72E+03	1.71E+01	3.72E-02
Pm-149	6.29E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-143	8.02E+04	1.73E+04	3.74E+03	0.00E+00	0.00E+00
Pu-236	5.42E-03	5.31E-03	5.21E-03	4.34E-03	3.41E-03
Pu-238	1.73E+02	1.74E+02	1.75E+02	1.80E+02	1.81E+02
Pu-239	1.44E+02	1.44E+02	1.44E+02	1.44E+02	1.44E+02
Pu-240	9.86E+01	9.86E+01	9.86E+01	9.86E+01	9.86E+01
Pu-241	2.42E+04	2.41E+04	2.41E+04	2.32E+04	2.21E+04
Pu-242	7.72E-02	7.72E-02	7.72E-02	7.72E-02	7.72E-02
Rb-86	3.25E+01	1.07E+01	3.49E+00	0.00E+00	0.00E+00
Rh-102	2.56E-02	2.32E-02	2.10E-02	0.00E+00	0.00E+00
Rh-103m	1.41E+05	8.30E+04	4.89E+04	3.80E+02	0.00E+00
Rh-105	1.11E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-103	1.43E+05	8.39E+04	4.94E+04	3.84E+02	6.09E-01
Ru-106	5.86E+04	5.54E+04	5.24E+04	3.14E+04	1.59E+04
Sb-122	1.74E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-124	1.94E+01	1.37E+01	9.71E+00	4.09E-01	0.00E+00
Sb-125	1.48E+03	1.45E+03	1.42E+03	1.17E+03	9.13E+02
Sb-126	1.24E+01	2.31E+00	4.34E-01	0.00E+00	0.00E+00
Sb-127	6.62E+01	2.99E-01	1.35E-03	0.00E+00	0.00E+00
Sm-151	2.45E+02	2.44E+02	2.44E+02	2.43E+02	2.41E+02
Sm-153	6.85E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sn-117m	5.89E-01	1.28E-01	2.77E-02	0.00E+00	0.00E+00
Sn-119m	3.09E+01	2.87E+01	2.68E+01	1.40E+01	5.89E+00

Isotope	PELE Total 30 Days (Ci)	PELE Total 60 Days (Ci)	PELE Total 90 Days (Ci)	PELE Total 1 Year (Ci)	PELE Total 2 Years (Ci)
Sn-121m	3.70E+00	3.69E+00	3.69E+00	3.64E+00	3.59E+00
Sn-123	1.64E+02	1.40E+02	1.19E+02	2.72E+01	3.84E+00
Sn-125	1.69E+02	1.96E+01	2.26E+00	0.00E+00	0.00E+00
Sn-126	4.24E-02	4.24E-02	4.24E-02	4.24E-02	4.24E-02
Sr-89	1.59E+05	1.05E+05	6.98E+04	1.60E+03	1.07E+01
Sr-90	2.20E+04	2.19E+04	2.19E+04	2.15E+04	2.10E+04
Tb-160	1.42E+01	1.06E+01	7.98E+00	5.72E-01	1.73E-02
Tb-161	4.18E+00	2.06E-01	1.01E-02	0.00E+00	0.00E+00
Tc-99	3.56E+00	3.56E+00	3.56E+00	3.56E+00	3.56E+00
Tc-99m	1.85E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-123m	6.53E-02	5.48E-02	4.61E-02	0.00E+00	0.00E+00
Te-125m	3.29E+02	3.31E+02	3.31E+02	2.87E+02	2.24E+02
Te-127	1.69E+03	1.35E+03	1.11E+03	1.93E+02	1.90E+01
Te-127m	1.66E+03	1.37E+03	1.13E+03	1.97E+02	1.94E+01
Te-129	2.50E+03	1.35E+03	7.25E+02	2.49E+00	0.00E+00
Te-129m	3.96E+03	2.13E+03	1.15E+03	3.95E+00	0.00E+00
Te-132	4.10E+02	6.23E-01	0.00E+00	0.00E+00	0.00E+00
Th-231	5.77E-02	5.77E-02	5.77E-02	5.77E-02	5.77E-02
Th-234	4.64E-02	4.64E-02	4.64E-02	4.64E-02	4.64E-02
U-232	1.40E-03	1.40E-03	1.41E-03	1.43E-03	1.46E-03
U-234	2.52E+00	2.52E+00	2.52E+00	2.52E+00	2.52E+00
U-236	1.56E-01	1.56E-01	1.56E-01	1.56E-01	1.56E-01
Xe-129m	7.48E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131m	7.83E+02	1.68E+02	3.14E+01	0.00E+00	0.00E+00
Xe-133	9.45E+03	1.79E+02	3.39E+00	0.00E+00	0.00E+00
Xe-133m	1.50E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-88	1.58E-02	1.30E-02	0.00E+00	0.00E+00	0.00E+00
Y-89m	1.53E+01	1.02E+01	6.73E+00	1.55E-01	1.04E-03
Y-90	2.20E+04	2.19E+04	2.19E+04	2.15E+04	2.10E+04
Y-91	2.12E+05	1.48E+05	1.04E+05	4.00E+03	5.30E+01
Zr-95	2.64E+05	1.91E+05	1.38E+05	7.03E+03	1.35E+02
Totals	**2.24E+06**	**1.54E+06**	**1.16E+06**	**3.34E+05**	**2.02E+05**

Table A.2. 30-Day MAR (three sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Ag-110m	Ag, Pd	9.11E+01	0.00E+00	2.35E+00
Ag-111	Ag, Pd	3.51E+02	0.00E+00	1.91E+00
Am-241	Pu, actinides	3.25E+01	3.37E-03	2.15E-06
Am-242m	Pu, actinides	1.11E+00	1.15E-04	7.32E-08
Am-243	Pu, actinides	1.88E-01	1.94E-05	1.24E-08
Ba-136m	Sr, Ba, Eu	1.28E+02	1.28E+00	8.72E-03
Ba-140	Sr, Ba, Eu	6.98E+04	6.97E+02	4.74E+00
Cd-113m	Sb	1.55E-01	1.33E-04	6.92E-05
Cd-115	Sb	1.04E-01	8.99E-05	4.66E-05
Cd-115m	Sb	4.30E+01	3.71E-02	1.92E-02
Ce-139	La, Ce	1.41E-01	1.55E-05	1.23E-07
Ce-141	La, Ce	1.81E+05	1.99E+01	1.58E-01
Ce-143	La, Ce	9.25E-02	1.02E-05	8.08E-08
Ce-144	La, Ce	2.70E+05	2.95E+01	2.34E-01
Cm-242	Pu, actinides	2.26E+03	2.34E-01	1.50E-04
Cm-243	Pu, actinides	2.26E-01	2.34E-05	1.49E-08
Cm-244	Pu, actinides	5.51E+00	5.71E-04	3.65E-07
Cs-134	Cs, Rb	1.17E+04	5.94E+00	6.45E+00
Cs-135	Cs, Rb	2.58E-01	1.31E-04	1.42E-04
Cs-136	Cs, Rb	1.17E+03	5.94E-01	6.44E-01
Cs-137	Cs, Rb	2.53E+04	1.26E+01	1.40E+01
Eu-152	Sr, Ba, Eu	5.50E+00	5.49E-02	3.73E-04
Eu-154	Sr, Ba, Eu	4.60E+02	4.59E+00	3.12E-02
Eu-155	Sr, Ba, Eu	3.28E+02	3.28E+00	2.23E-02
Eu-156	Sr, Ba, Eu	1.87E+03	1.87E+01	1.27E-01
Gd-153	Sr, Ba, Eu	2.14E+00	2.14E-02	1.46E-04
H-3	H-3, (1*)	1.05E+02	0.00E+00	3.31E-03
I-131	I, Br, Te, Se	1.46E+04	0.00E+00	4.62E-01
I-132	I, Br, Te, Se	4.23E+02	0.00E+00	1.37E-02
In-115m	Ag, Pd	1.15E-01	0.00E+00	2.98E-03
Kr-85	Noble Gases	2.76E+03	0.00E+00	8.96E-02
La-140	La, Ce	8.12E+04	8.94E+00	7.09E-02
Mo-99	Mo, Ru, Rh, Tc	1.91E+02	2.08E-02	1.65E-04
Nb-95	Mo, Ru, Rh, Tc	3.40E+05	3.71E+01	2.94E-01
Nd-147	La, Ce	1.97E+04	2.17E+00	1.72E-02
Np-237	Pu, actinides	6.23E-02	6.46E-06	4.12E-09
Np-239	Pu, actinides	4.49E+02	4.65E-02	2.97E-05
P-32	Sb	3.83E-02	3.31E-05	1.71E-05

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Pa-233	Pu, actinides	6.95E-02	7.20E-06	4.60E-09
Pm-146	La, Ce	4.41E-03	4.85E-07	3.85E-09
Pm-147	La, Ce	6.11E+04	6.73E+00	5.34E-02
Pm-148m	La, Ce	4.72E+03	5.20E-01	4.12E-03
Pm-149	La, Ce	6.29E+00	6.93E-04	5.50E-06
Pr-143	La, Ce	8.02E+04	8.83E+00	7.01E-02
Pu-236	Pu, actinides	5.42E-03	5.61E-07	3.59E-10
Pu-238	Pu, actinides	1.73E+02	1.79E-02	1.14E-05
Pu-239	Pu, actinides	1.44E+02	1.49E-02	9.51E-06
Pu-240	Pu, actinides	9.86E+01	1.02E-02	6.52E-06
Pu-241	Pu, actinides	2.42E+04	2.51E+00	1.60E-03
Pu-242	Pu, actinides	7.72E-02	8.00E-06	5.11E-09
Rb-86	Cs, Rb	3.25E+01	1.65E-02	1.79E-02
Rh-102	Mo, Ru, Rh, Tc	2.56E-02	2.80E-06	2.22E-08
Rh-103m	Mo, Ru, Rh, Tc	1.41E+05	1.54E+01	1.22E-01
Rh-105	Mo, Ru, Rh, Tc	1.11E-01	1.21E-05	9.60E-08
Ru-103	Mo, Ru, Rh, Tc	1.43E+05	1.56E+01	1.23E-01
Ru-106	Mo, Ru, Rh, Tc	5.86E+04	6.40E+00	5.07E-02
Sb-122	Sb	1.74E-02	1.51E-05	7.80E-06
Sb-124	Sb	1.94E+01	1.67E-02	8.66E-03
Sb-125	Sb	1.47E+03	1.27E+00	6.60E-01
Sb-126	Sb	1.24E+01	1.07E-02	5.55E-03
Sb-127	Sb	6.62E+01	5.72E-02	2.96E-02
Sm-151	La, Ce	2.45E+02	2.69E-02	2.14E-04
Sm-153	La, Ce	6.85E-01	7.54E-05	5.99E-07
Sn-117m	Ag, Pd	5.74E-01	0.00E+00	1.48E-02
Sn-119m	Ag, Pd	3.01E+01	0.00E+00	7.77E-01
Sn-121m	Ag, Pd	3.60E+00	0.00E+00	9.31E-02
Sn-123	Ag, Pd	1.60E+02	0.00E+00	4.14E+00
Sn-125	Ag, Pd	1.65E+02	0.00E+00	4.26E+00
Sn-126	Ag, Pd	4.14E-02	0.00E+00	1.07E-03
Sr-89	Sr, Ba, Eu	1.57E+05	1.57E+03	1.07E+01
Sr-90	Sr, Ba, Eu	2.17E+04	2.17E+02	1.48E+00
Tb-160	Sr, Ba, Eu	1.41E+01	1.40E-01	9.54E-04
Tb-161	Sr, Ba, Eu	4.14E+00	4.14E-02	2.81E-04
Tc-99	Mo, Ru, Rh, Tc	3.56E+00	3.89E-04	3.08E-06
Tc-99m	Mo, Ru, Rh, Tc	1.84E+02	2.01E-02	1.60E-04
Te-123m	I, Br, Te, Se	6.53E-02	0.00E+00	2.11E-06
Te-125m	I, Br, Te, Se	3.29E+02	0.00E+00	1.07E-02
Te-127	I, Br, Te, Se	1.69E+03	0.00E+00	5.47E-02
Te-127m	I, Br, Te, Se	1.66E+03	0.00E+00	5.38E-02

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Te-129	I, Br, Te, Se	2.50E+03	0.00E+00	8.09E-02
Te-129m	I, Br, Te, Se	3.96E+03	0.00E+00	1.28E-01
Te-132	I, Br, Te, Se	4.10E+02	0.00E+00	1.33E-02
Th-231	Pu, actinides	5.77E-02	5.98E-06	3.82E-09
Th-234	Pu, actinides	4.64E-02	4.81E-06	3.07E-09
U-232	Pu, actinides	1.40E-03	1.45E-07	9.26E-11
U-234	Pu, actinides	2.52E+00	2.61E-04	1.66E-07
U-236	Pu, actinides	1.56E-01	1.62E-05	1.03E-08
Xe-129m	Noble Gases	7.48E-03	0.00E+00	2.43E-07
Xe-131m	Noble Gases	7.83E+02	0.00E+00	2.54E-02
Xe-133	Noble Gases	9.45E+03	0.00E+00	2.98E-01
Xe-133m	Noble Gases	1.50E+00	0.00E+00	4.86E-05
Y-88	La, Ce	1.58E-02	1.74E-06	1.38E-08
Y-89m	La, Ce	1.53E+01	1.69E-03	1.34E-05
Y-90	La, Ce	2.20E+04	2.42E+00	1.92E-02
Y-91	La, Ce	2.12E+05	2.33E+01	1.85E-01
Zr-95	La, Ce	2.64E+05	2.91E+01	2.31E-01
Totals	—	**2.24E+06**	**2.74E+03**	**5.54E+01**

TRISO = tri-structural isotropic (particle).

Table A.3. 60-Day MAR (three sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Ag-110m	Ag, Pd	8.38E+01	0.00E+00	2.17E+00
Ag-111	Ag, Pd	2.15E+01	0.00E+00	1.17E-01
Am-241	Pu, actinides	3.57E+01	3.70E-03	2.36E-06
Am-242m	Pu, actinides	1.11E+00	1.15E-04	7.31E-08
Am-243	Pu, actinides	1.88E-01	1.94E-05	1.24E-08
Ba-136m	Sr, Ba, Eu	2.64E+01	2.64E-01	1.80E-03
Ba-140	Sr, Ba, Eu	1.37E+04	1.37E+02	9.28E-01
Cd-113m	Sb	1.54E-01	1.33E-04	6.89E-05
Cd-115	Sb	0.00E+00	0.00E+00	0.00E+00
Cd-115m	Sb	2.69E+01	2.33E-02	1.21E-02
Ce-139	La, Ce	1.21E-01	1.33E-05	1.06E-07
Ce-141	La, Ce	9.55E+04	1.05E+01	8.34E-02
Ce-143	La, Ce	0.00E+00	0.00E+00	0.00E+00
Ce-144	La, Ce	2.51E+05	2.74E+01	2.18E-01
Cm-242	Pu, actinides	1.99E+03	2.06E-01	1.32E-04
Cm-243	Pu, actinides	2.25E-01	2.34E-05	1.49E-08
Cm-244	Pu, actinides	5.50E+00	5.69E-04	3.64E-07
Cs-134	Cs, Rb	1.14E+04	5.78E+00	6.27E+00
Cs-135	Cs, Rb	2.58E-01	1.31E-04	1.42E-04
Cs-136	Cs, Rb	2.41E+02	1.22E-01	1.33E-01
Cs-137	Cs, Rb	2.52E+04	1.26E+01	1.40E+01
Eu-152	Sr, Ba, Eu	5.47E+00	5.47E-02	3.72E-04
Eu-154	Sr, Ba, Eu	4.56E+02	4.56E+00	3.10E-02
Eu-155	Sr, Ba, Eu	3.24E+02	3.24E+00	2.20E-02
Eu-156	Sr, Ba, Eu	4.75E+02	4.75E+00	3.23E-02
Gd-153	Sr, Ba, Eu	1.97E+00	1.96E-02	1.34E-04
H-3	H-3, (1*)	1.04E+02	0.00E+00	3.29E-03
I-131	I, Br, Te, Se	1.09E+03	0.00E+00	3.47E-02
I-132	I, Br, Te, Se	6.42E-01	0.00E+00	2.08E-05
In-115m	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Kr-85	Noble Gases	2.74E+03	0.00E+00	8.91E-02
La-140	La, Ce	1.59E+04	1.75E+00	1.39E-02
Mo-99	Mo, Ru, Rh, Tc	9.89E-02	1.08E-05	8.55E-08
Nb-95	Mo, Ru, Rh, Tc	2.87E+05	3.13E+01	2.48E-01
Nd-147	La, Ce	2.97E+03	3.27E-01	2.59E-03
Np-237	Pu, actinides	6.24E-02	6.46E-06	4.13E-09
Np-239	Pu, actinides	2.53E-01	2.62E-05	1.68E-08
P-32	Sb	0.00E+00	0.00E+00	0.00E+00
Pa-233	Pu, actinides	6.57E-02	6.80E-06	4.34E-09

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Pm-146	La, Ce	4.36E-03	4.80E-07	3.81E-09
Pm-147	La, Ce	6.00E+04	6.60E+00	5.24E-02
Pm-148m	La, Ce	2.85E+03	3.14E-01	2.49E-03
Pm-149	La, Ce	0.00E+00	0.00E+00	0.00E+00
Pr-143	La, Ce	1.73E+04	1.91E+00	1.51E-02
Pu-236	Pu, actinides	5.31E-03	5.50E-07	3.52E-10
Pu-238	Pu, actinides	1.74E+02	1.80E-02	1.15E-05
Pu-239	Pu, actinides	1.44E+02	1.49E-02	9.51E-06
Pu-240	Pu, actinides	9.86E+01	1.02E-02	6.52E-06
Pu-241	Pu, actinides	2.41E+04	2.50E+00	1.60E-03
Pu-242	Pu, actinides	7.72E-02	8.00E-06	5.11E-09
Rb-86	Cs, Rb	1.07E+01	5.41E-03	5.87E-03
Rh-102	Mo, Ru, Rh, Tc	2.32E-02	2.53E-06	2.00E-08
Rh-103m	Mo, Ru, Rh, Tc	8.30E+04	9.06E+00	7.18E-02
Rh-105	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Ru-103	Mo, Ru, Rh, Tc	8.39E+04	9.15E+00	7.25E-02
Ru-106	Mo, Ru, Rh, Tc	5.54E+04	6.05E+00	4.79E-02
Sb-122	Sb	0.00E+00	0.00E+00	0.00E+00
Sb-124	Sb	1.37E+01	1.18E-02	6.13E-03
Sb-125	Sb	1.45E+03	1.25E+00	6.47E-01
Sb-126	Sb	2.31E+00	1.99E-03	1.03E-03
Sb-127	Sb	2.98E-01	2.58E-04	1.34E-04
Sm-151	La, Ce	2.44E+02	2.69E-02	2.13E-04
Sm-153	La, Ce	0.00E+00	0.00E+00	0.00E+00
Sn-117m	Ag, Pd	1.24E-01	0.00E+00	3.22E-03
Sn-119m	Ag, Pd	2.80E+01	0.00E+00	7.24E-01
Sn-121m	Ag, Pd	3.60E+00	0.00E+00	9.30E-02
Sn-123	Ag, Pd	1.36E+02	0.00E+00	3.52E+00
Sn-125	Ag, Pd	1.91E+01	0.00E+00	4.93E-01
Sn-126	Ag, Pd	4.14E-02	0.00E+00	1.07E-03
Sr-89	Sr, Ba, Eu	1.04E+05	1.04E+03	7.08E+00
Sr-90	Sr, Ba, Eu	2.17E+04	2.17E+02	1.47E+00
Tb-160	Sr, Ba, Eu	1.05E+01	1.05E-01	7.16E-04
Tb-161	Sr, Ba, Eu	2.04E-01	2.04E-03	1.39E-05
Tc-99	Mo, Ru, Rh, Tc	3.56E+00	3.89E-04	3.08E-06
Tc-99m	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Te-123m	I, Br, Te, Se	5.48E-02	0.00E+00	1.78E-06
Te-125m	I, Br, Te, Se	3.31E+02	0.00E+00	1.07E-02
Te-127	I, Br, Te, Se	1.35E+03	0.00E+00	4.36E-02
Te-127m	I, Br, Te, Se	1.37E+03	0.00E+00	4.45E-02
Te-129	I, Br, Te, Se	1.35E+03	0.00E+00	4.36E-02

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Te-129m	I, Br, Te, Se	2.13E+03	0.00E+00	6.91E-02
Te-132	I, Br, Te, Se	6.23E-01	0.00E+00	2.02E-05
Th-231	Pu, actinides	5.77E-02	5.98E-06	3.82E-09
Th-234	Pu, actinides	4.64E-02	4.80E-06	3.07E-09
U-232	Pu, actinides	1.40E-03	1.45E-07	9.28E-11
U-234	Pu, actinides	2.52E+00	2.61E-04	1.66E-07
U-236	Pu, actinides	1.56E-01	1.62E-05	1.03E-08
Xe-129m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Xe-131m	Noble Gases	1.68E+02	0.00E+00	5.45E-03
Xe-133	Noble Gases	1.79E+02	0.00E+00	5.65E-03
Xe-133m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Y-88	La, Ce	1.30E-02	1.43E-06	1.13E-08
Y-89m	La, Ce	1.02E+01	1.12E-03	8.87E-06
Y-90	La, Ce	2.19E+04	2.41E+00	1.92E-02
Y-91	La, Ce	1.48E+05	1.63E+01	1.30E-01
Zr-95	La, Ce	1.91E+05	2.10E+01	1.67E-01
Totals	—	**1.53E+06**	**1.58E+03**	**3.93E+01**

TRISO = tri-structural isotropic (particle).

Table A.4. 90-Day MAR (three sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Ag-110m	Ag, Pd	7.71E+01	0.00E+00	1.99E+00
Ag-111	Ag, Pd	1.32E+00	0.00E+00	7.19E-03
Am-241	Pu, actinides	3.89E+01	4.03E-03	2.57E-06
Am-242m	Pu, actinides	1.11E+00	1.14E-04	7.31E-08
Am-243	Pu, actinides	1.88E-01	1.94E-05	1.24E-08
Ba-136m	Sr, Ba, Eu	5.44E+00	5.44E-02	3.70E-04
Ba-140	Sr, Ba, Eu	2.68E+03	2.67E+01	1.82E-01
Cd-113m	Sb	1.53E-01	1.32E-04	6.86E-05
Cd-115	Sb	0.00E+00	0.00E+00	0.00E+00
Cd-115m	Sb	1.69E+01	1.46E-02	7.56E-03
Ce-139	La, Ce	1.04E-01	1.15E-05	9.11E-08
Ce-141	La, Ce	5.04E+04	5.55E+00	4.40E-02
Ce-143	La, Ce	0.00E+00	0.00E+00	0.00E+00
Ce-144	La, Ce	2.34E+05	2.55E+01	2.03E-01
Cm-242	Pu, actinides	1.75E+03	1.82E-01	1.16E-04
Cm-243	Pu, actinides	2.25E-01	2.33E-05	1.49E-08
Cm-244	Pu, actinides	5.48E+00	5.68E-04	3.62E-07
Cs-134	Cs, Rb	1.11E+04	5.62E+00	6.10E+00
Cs-135	Cs, Rb	2.58E-01	1.31E-04	1.42E-04
Cs-136	Cs, Rb	4.96E+01	2.52E-02	2.73E-02
Cs-137	Cs, Rb	2.52E+04	1.26E+01	1.40E+01
Eu-152	Sr, Ba, Eu	5.45E+00	5.45E-02	3.70E-04
Eu-154	Sr, Ba, Eu	4.53E+02	4.53E+00	3.08E-02
Eu-155	Sr, Ba, Eu	3.20E+02	3.20E+00	2.17E-02
Eu-156	Sr, Ba, Eu	1.21E+02	1.21E+00	8.21E-03
Gd-153	Sr, Ba, Eu	1.80E+00	1.80E-02	1.22E-04
H-3	H-3, (1*)	1.04E+02	0.00E+00	3.28E-03
I-131	I, Br, Te, Se	8.20E+01	0.00E+00	2.60E-03
I-132	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
In-115m	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Kr-85	Noble Gases	2.73E+03	0.00E+00	8.86E-02
La-140	La, Ce	3.11E+03	3.43E-01	2.72E-03
Mo-99	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Nb-95	Mo, Ru, Rh, Tc	2.30E+05	2.51E+01	1.99E-01
Nd-147	La, Ce	4.47E+02	4.92E-02	3.90E-04
Np-237	Pu, actinides	6.24E-02	6.46E-06	4.13E-09
Np-239	Pu, actinides	1.88E-01	1.94E-05	1.24E-08
P-32	Sb	0.00E+00	0.00E+00	0.00E+00
Pa-233	Pu, actinides	6.39E-02	6.62E-06	4.23E-09

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Pm-146	La, Ce	4.32E-03	4.76E-07	3.77E-09
Pm-147	La, Ce	5.87E+04	6.46E+00	5.13E-02
Pm-148m	La, Ce	1.72E+03	1.90E-01	1.51E-03
Pm-149	La, Ce	0.00E+00	0.00E+00	0.00E+00
Pr-143	La, Ce	3.74E+03	4.12E-01	3.27E-03
Pu-236	Pu, actinides	5.21E-03	5.39E-07	3.45E-10
Pu-238	Pu, actinides	1.75E+02	1.81E-02	1.16E-05
Pu-239	Pu, actinides	1.44E+02	1.49E-02	9.51E-06
Pu-240	Pu, actinides	9.86E+01	1.02E-02	6.52E-06
Pu-241	Pu, actinides	2.41E+04	2.49E+00	1.59E-03
Pu-242	Pu, actinides	7.72E-02	8.00E-06	5.11E-09
Rb-86	Cs, Rb	3.49E+00	1.77E-03	1.92E-03
Rh-102	Mo, Ru, Rh, Tc	2.10E-02	2.29E-06	1.81E-08
Rh-103m	Mo, Ru, Rh, Tc	4.89E+04	5.33E+00	4.22E-02
Rh-105	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Ru-103	Mo, Ru, Rh, Tc	4.94E+04	5.39E+00	4.27E-02
Ru-106	Mo, Ru, Rh, Tc	5.24E+04	5.72E+00	4.53E-02
Sb-122	Sb	0.00E+00	0.00E+00	0.00E+00
Sb-124	Sb	9.70E+00	8.38E-03	4.34E-03
Sb-125	Sb	1.42E+03	1.22E+00	6.34E-01
Sb-126	Sb	4.33E-01	3.74E-04	1.94E-04
Sb-127	Sb	1.35E-03	1.16E-06	6.03E-07
Sm-151	La, Ce	2.44E+02	2.69E-02	2.13E-04
Sm-153	La, Ce	0.00E+00	0.00E+00	0.00E+00
Sn-117m	Ag, Pd	2.70E-02	0.00E+00	6.97E-04
Sn-119m	Ag, Pd	2.61E+01	0.00E+00	6.75E-01
Sn-121m	Ag, Pd	3.59E+00	0.00E+00	9.29E-02
Sn-123	Ag, Pd	1.16E+02	0.00E+00	3.00E+00
Sn-125	Ag, Pd	2.21E+00	0.00E+00	5.70E-02
Sn-126	Ag, Pd	4.14E-02	0.00E+00	1.07E-03
Sr-89	Sr, Ba, Eu	6.91E+04	6.90E+02	4.69E+00
Sr-90	Sr, Ba, Eu	2.17E+04	2.16E+02	1.47E+00
Tb-160	Sr, Ba, Eu	7.91E+00	7.90E-02	5.37E-04
Tb-161	Sr, Ba, Eu	1.00E-02	1.00E-04	6.82E-07
Tc-99	Mo, Ru, Rh, Tc	3.56E+00	3.89E-04	3.08E-06
Tc-99m	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Te-123m	I, Br, Te, Se	4.61E-02	0.00E+00	1.49E-06
Te-125m	I, Br, Te, Se	3.31E+02	0.00E+00	1.07E-02
Te-127	I, Br, Te, Se	1.11E+03	0.00E+00	3.60E-02
Te-127m	I, Br, Te, Se	1.13E+03	0.00E+00	3.68E-02
Te-129	I, Br, Te, Se	7.25E+02	0.00E+00	2.35E-02

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Te-129m	I, Br, Te, Se	1.15E+03	0.00E+00	3.72E-02
Te-132	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
Th-231	Pu, actinides	5.77E-02	5.98E-06	3.82E-09
Th-234	Pu, actinides	4.64E-02	4.80E-06	3.07E-09
U-232	Pu, actinides	1.41E-03	1.46E-07	9.30E-11
U-234	Pu, actinides	2.52E+00	2.61E-04	1.66E-07
U-236	Pu, actinides	1.56E-01	1.62E-05	1.03E-08
Xe-129m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Xe-131m	Noble Gases	3.14E+01	0.00E+00	1.02E-03
Xe-133	Noble Gases	3.39E+00	0.00E+00	1.07E-04
Xe-133m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Y-88	La, Ce	0.00E+00	0.00E+00	0.00E+00
Y-89m	La, Ce	6.73E+00	7.41E-04	5.88E-06
Y-90	La, Ce	2.19E+04	2.41E+00	1.91E-02
Y-91	La, Ce	1.04E+05	1.15E+01	9.09E-02
Zr-95	La, Ce	1.38E+05	1.52E+01	1.21E-01
Totals	—	**1.16E+06**	**1.07E+03**	**3.41E+01**
TRISO = tri-structural isotropic (particle).				

Table A.5. 1-Year MAR (three sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Ag-110m	Ag, Pd	3.59E+01	0.00E+00	9.29E-01
Ag-111	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Am-241	Pu, actinides	6.73E+01	6.97E-03	4.45E-06
Am-242m	Pu, actinides	1.10E+00	1.14E-04	7.28E-08
Am-243	Pu, actinides	1.88E-01	1.94E-05	1.24E-08
Ba-136m	Sr, Ba, Eu	0.00E+00	0.00E+00	0.00E+00
Ba-140	Sr, Ba, Eu	0.00E+00	0.00E+00	0.00E+00
Cd-113m	Sb	1.48E-01	1.28E-04	6.61E-05
Cd-115	Sb	0.00E+00	0.00E+00	0.00E+00
Cd-115m	Sb	2.34E-01	2.03E-04	1.05E-04
Ce-139	La, Ce	0.00E+00	0.00E+00	0.00E+00
Ce-141	La, Ce	1.43E+02	1.58E-02	1.25E-04
Ce-143	La, Ce	0.00E+00	0.00E+00	0.00E+00
Ce-144	La, Ce	1.20E+05	1.31E+01	1.04E-01
Cm-242	Pu, actinides	5.45E+02	5.65E-02	3.61E-05
Cm-243	Pu, actinides	2.21E-01	2.29E-05	1.46E-08
Cm-244	Pu, actinides	5.32E+00	5.51E-04	3.52E-07
Cs-134	Cs, Rb	8.60E+03	4.37E+00	4.74E+00
Cs-135	Cs, Rb	2.58E-01	1.31E-04	1.42E-04
Cs-136	Cs, Rb	0.00E+00	0.00E+00	0.00E+00
Cs-137	Cs, Rb	2.47E+04	1.24E+01	1.37E+01
Eu-152	Sr, Ba, Eu	5.24E+00	5.24E-02	3.56E-04
Eu-154	Sr, Ba, Eu	4.27E+02	4.26E+00	2.90E-02
Eu-155	Sr, Ba, Eu	2.87E+02	2.87E+00	1.95E-02
Eu-156	Sr, Ba, Eu	0.00E+00	0.00E+00	0.00E+00
Gd-153	Sr, Ba, Eu	8.16E-01	8.15E-03	5.54E-05
H-3	H-3, (1*)	9.97E+01	0.00E+00	3.14E-03
I-131	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
I-132	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
In-115m	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Kr-85	Noble Gases	2.60E+03	0.00E+00	8.44E-02
La-140	La, Ce	0.00E+00	0.00E+00	0.00E+00
Mo-99	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Nb-95	Mo, Ru, Rh, Tc	1.52E+04	1.66E+00	1.31E-02
Nd-147	La, Ce	0.00E+00	0.00E+00	0.00E+00
Np-237	Pu, actinides	6.24E-02	6.46E-06	4.13E-09
Np-239	Pu, actinides	1.88E-01	1.94E-05	1.24E-08
P-32	Sb	0.00E+00	0.00E+00	0.00E+00
Pa-233	Pu, actinides	6.24E-02	6.46E-06	4.13E-09
Pm-146	La, Ce	3.93E-03	4.33E-07	3.43E-09

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Pm-147	La, Ce	4.81E+04	5.30E+00	4.20E-02
Pm-148m	La, Ce	1.70E+01	1.88E-03	1.49E-05
Pm-149	La, Ce	0.00E+00	0.00E+00	0.00E+00
Pr-143	La, Ce	0.00E+00	0.00E+00	0.00E+00
Pu-236	Pu, actinides	4.34E-03	4.49E-07	2.87E-10
Pu-238	Pu, actinides	1.80E+02	1.87E-02	1.19E-05
Pu-239	Pu, actinides	1.44E+02	1.49E-02	9.51E-06
Pu-240	Pu, actinides	9.86E+01	1.02E-02	6.52E-06
Pu-241	Pu, actinides	2.32E+04	2.40E+00	1.53E-03
Pu-242	Pu, actinides	7.72E-02	8.00E-06	5.11E-09
Rb-86	Cs, Rb	0.00E+00	0.00E+00	0.00E+00
Rh-102	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Rh-103m	Mo, Ru, Rh, Tc	3.80E+02	4.15E-02	3.28E-04
Rh-105	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Ru-103	Mo, Ru, Rh, Tc	3.84E+02	4.19E-02	3.32E-04
Ru-106	Mo, Ru, Rh, Tc	3.14E+04	3.43E+00	2.71E-02
Sb-122	Sb	0.00E+00	0.00E+00	0.00E+00
Sb-124	Sb	4.09E-01	3.53E-04	1.83E-04
Sb-125	Sb	1.17E+03	1.01E+00	5.25E-01
Sb-126	Sb	0.00E+00	0.00E+00	0.00E+00
Sb-127	Sb	0.00E+00	0.00E+00	0.00E+00
Sm-151	La, Ce	2.43E+02	2.67E-02	2.12E-04
Sm-153	La, Ce	0.00E+00	0.00E+00	0.00E+00
Sn-117m	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Sn-119m	Ag, Pd	1.36E+01	0.00E+00	3.52E-01
Sn-121m	Ag, Pd	3.55E+00	0.00E+00	9.18E-02
Sn-123	Ag, Pd	2.65E+01	0.00E+00	6.86E-01
Sn-125	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Sn-126	Ag, Pd	4.14E-02	0.00E+00	1.07E-03
Sr-89	Sr, Ba, Eu	1.59E+03	1.59E+01	1.08E-01
Sr-90	Sr, Ba, Eu	2.13E+04	2.13E+02	1.44E+00
Tb-160	Sr, Ba, Eu	5.66E-01	5.66E-03	3.84E-05
Tb-161	Sr, Ba, Eu	0.00E+00	0.00E+00	0.00E+00
Tc-99	Mo, Ru, Rh, Tc	3.56E+00	3.89E-04	3.08E-06
Tc-99m	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Te-123m	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
Te-125m	I, Br, Te, Se	2.87E+02	0.00E+00	9.29E-03
Te-127	I, Br, Te, Se	1.93E+02	0.00E+00	6.26E-03
Te-127m	I, Br, Te, Se	1.97E+02	0.00E+00	6.40E-03
Te-129	I, Br, Te, Se	2.49E+00	0.00E+00	8.07E-05
Te-129m	I, Br, Te, Se	3.95E+00	0.00E+00	1.28E-04

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Te-132	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
Th-231	Pu, actinides	5.77E-02	5.98E-06	3.82E-09
Th-234	Pu, actinides	4.64E-02	4.80E-06	3.07E-09
U-232	Pu, actinides	1.43E-03	1.48E-07	9.47E-11
U-234	Pu, actinides	2.52E+00	2.61E-04	1.67E-07
U-236	Pu, actinides	1.56E-01	1.62E-05	1.03E-08
Xe-129m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Xe-131m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Xe-133	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Xe-133m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Y-88	La, Ce	0.00E+00	0.00E+00	0.00E+00
Y-89m	La, Ce	1.55E-01	1.70E-05	1.35E-07
Y-90	La, Ce	2.15E+04	2.37E+00	1.88E-02
Y-91	La, Ce	4.00E+03	4.41E-01	3.50E-03
Zr-95	La, Ce	7.03E+03	7.74E-01	6.14E-03
Totals	—	**3.34E+05**	**2.83E+02**	**2.30E+01**

TRISO = tri-structural isotropic (particle).

Table A.6. 2-Year MAR (three sheets total)

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Ag-110m	Ag, Pd	1.31E+01	0.00E+00	3.37E-01
Ag-111	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Am-241	Pu, actinides	1.03E+02	1.07E-02	6.84E-06
Am-242m	Pu, actinides	1.10E+00	1.13E-04	7.25E-08
Am-243	Pu, actinides	1.87E-01	1.94E-05	1.24E-08
Ba-136m	Sr, Ba, Eu	0.00E+00	0.00E+00	0.00E+00
Ba-140	Sr, Ba, Eu	0.00E+00	0.00E+00	0.00E+00
Cd-113m	Sb	1.41E-01	1.21E-04	6.30E-05
Cd-115	Sb	0.00E+00	0.00E+00	0.00E+00
Cd-115m	Sb	0.00E+00	0.00E+00	0.00E+00
Ce-139	La, Ce	0.00E+00	0.00E+00	0.00E+00
Ce-141	La, Ce	5.97E-02	6.57E-06	5.21E-08
Ce-143	La, Ce	0.00E+00	0.00E+00	0.00E+00
Ce-144	La, Ce	4.92E+04	5.38E+00	4.27E-02
Cm-242	Pu, actinides	1.16E+02	1.20E-02	7.68E-06
Cm-243	Pu, actinides	2.16E-01	2.24E-05	1.43E-08
Cm-244	Pu, actinides	5.12E+00	5.31E-04	3.39E-07
Cs-134	Cs, Rb	6.15E+03	3.12E+00	3.39E+00
Cs-135	Cs, Rb	2.58E-01	1.31E-04	1.42E-04
Cs-136	Cs, Rb	0.00E+00	0.00E+00	0.00E+00
Cs-137	Cs, Rb	2.42E+04	1.21E+01	1.34E+01
Eu-152	Sr, Ba, Eu	4.98E+00	4.98E-02	3.38E-04
Eu-154	Sr, Ba, Eu	3.94E+02	3.93E+00	2.67E-02
Eu-155	Sr, Ba, Eu	2.48E+02	2.48E+00	1.68E-02
Eu-156	Sr, Ba, Eu	0.00E+00	0.00E+00	0.00E+00
Gd-153	Sr, Ba, Eu	2.85E-01	2.85E-03	1.93E-05
H-3	H-3, (1*)	9.42E+01	0.00E+00	2.97E-03
I-131	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
I-132	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
In-115m	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Kr-85	Noble Gases	2.44E+03	0.00E+00	7.92E-02
La-140	La, Ce	0.00E+00	0.00E+00	0.00E+00
Mo-99	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Nb-95	Mo, Ru, Rh, Tc	2.98E+02	3.25E-02	2.57E-04
Nd-147	La, Ce	0.00E+00	0.00E+00	0.00E+00
Np-237	Pu, actinides	6.24E-02	6.47E-06	4.13E-09
Np-239	Pu, actinides	1.87E-01	1.94E-05	1.24E-08
P-32	Sb	0.00E+00	0.00E+00	0.00E+00
Pa-233	Pu, actinides	6.24E-02	6.47E-06	4.13E-09
Pm-146	La, Ce	3.47E-03	3.82E-07	3.03E-09

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Pm-147	La, Ce	3.70E+04	4.07E+00	3.23E-02
Pm-148m	La, Ce	3.72E-02	4.10E-06	3.25E-08
Pm-149	La, Ce	0.00E+00	0.00E+00	0.00E+00
Pr-143	La, Ce	0.00E+00	0.00E+00	0.00E+00
Pu-236	Pu, actinides	3.41E-03	3.53E-07	2.25E-10
Pu-238	Pu, actinides	1.81E+02	1.87E-02	1.20E-05
Pu-239	Pu, actinides	1.44E+02	1.49E-02	9.51E-06
Pu-240	Pu, actinides	9.86E+01	1.02E-02	6.52E-06
Pu-241	Pu, actinides	2.21E+04	2.29E+00	1.46E-03
Pu-242	Pu, actinides	7.72E-02	8.00E-06	5.11E-09
Rb-86	Cs, Rb	0.00E+00	0.00E+00	0.00E+00
Rh-102	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Rh-103m	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Rh-105	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Ru-103	Mo, Ru, Rh, Tc	6.09E-01	6.65E-05	5.27E-07
Ru-106	Mo, Ru, Rh, Tc	1.59E+04	1.73E+00	1.37E-02
Sb-122	Sb	0.00E+00	0.00E+00	0.00E+00
Sb-124	Sb	0.00E+00	0.00E+00	0.00E+00
Sb-125	Sb	9.12E+02	7.87E-01	4.08E-01
Sb-126	Sb	0.00E+00	0.00E+00	0.00E+00
Sb-127	Sb	0.00E+00	0.00E+00	0.00E+00
Sm-151	La, Ce	2.41E+02	2.65E-02	2.10E-04
Sm-153	La, Ce	0.00E+00	0.00E+00	0.00E+00
Sn-117m	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Sn-119m	Ag, Pd	5.74E+00	0.00E+00	1.48E-01
Sn-121m	Ag, Pd	3.50E+00	0.00E+00	9.04E-02
Sn-123	Ag, Pd	3.74E+00	0.00E+00	9.68E-02
Sn-125	Ag, Pd	0.00E+00	0.00E+00	0.00E+00
Sn-126	Ag, Pd	4.14E-02	0.00E+00	1.07E-03
Sr-89	Sr, Ba, Eu	1.06E+01	1.06E-01	7.22E-04
Sr-90	Sr, Ba, Eu	2.08E+04	2.08E+02	1.41E+00
Tb-160	Sr, Ba, Eu	1.71E-02	1.71E-04	1.16E-06
Tb-161	Sr, Ba, Eu	0.00E+00	0.00E+00	0.00E+00
Tc-99	Mo, Ru, Rh, Tc	3.56E+00	3.89E-04	3.08E-06
Tc-99m	Mo, Ru, Rh, Tc	0.00E+00	0.00E+00	0.00E+00
Te-123m	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
Te-125m	I, Br, Te, Se	2.23E+02	0.00E+00	7.24E-03
Te-127	I, Br, Te, Se	1.90E+01	0.00E+00	6.15E-04
Te-127m	I, Br, Te, Se	1.94E+01	0.00E+00	6.28E-04
Te-129	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
Te-129m	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00

Isotope	Grouping	TRISO (Ci)	Core (Ci)	Coolant Boundary (Ci)
Te-132	I, Br, Te, Se	0.00E+00	0.00E+00	0.00E+00
Th-231	Pu, actinides	5.77E-02	5.98E-06	3.82E-09
Th-234	Pu, actinides	4.64E-02	4.80E-06	3.07E-09
U-232	Pu, actinides	1.46E-03	1.51E-07	9.63E-11
U-234	Pu, actinides	2.52E+00	2.61E-04	1.67E-07
U-236	Pu, actinides	1.56E-01	1.62E-05	1.03E-08
Xe-129m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Xe-131m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Xe-133	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Xe-133m	Noble Gases	0.00E+00	0.00E+00	0.00E+00
Y-88	La, Ce	0.00E+00	0.00E+00	0.00E+00
Y-89m	La, Ce	1.04E-03	1.14E-07	9.04E-10
Y-90	La, Ce	2.10E+04	2.31E+00	1.83E-02
Y-91	La, Ce	5.30E+01	5.84E-03	4.63E-05
Zr-95	La, Ce	1.35E+02	1.49E-02	1.18E-04
Totals	—	**2.02E+05**	**2.46E+02**	**1.96E+01**

TRISO = tri-structural isotropic (particle).

Appendix B – Evaluation of TNPP Package Transportation Hazardous Conditions

This appendix provides the results of the Transportable Nuclear Power Plant (TNPP) Package transportation Hazardous Conditions Evaluation, which are filled out in worksheets. As described in Section 5.3.2 of this report, a series of expert panel sessions were held over the course of 2 weeks in early March of 2022 to identify and assess hazardous conditions associated with TNPP Package transport. The session participants were Pacific Northwest National Laboratory staff members, who are 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 the TNPP vendor designs and are the authors of this report. The experts filled out a hazardous condition worksheet to generate a comprehensive list of postulated hazardous conditions that could defeat the safety functions 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 Package. 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 and a detailed knowledge of transportation risk based on previous transportation risk assessments. The process considered hazards such as the kinetic energy associated with moving vehicles, and thermal energy associated with fires such as a diesel fuel fire. The process also considered hazardous conditions that could occur for a stationary reactor but created different hazardous conditions for the TNPP Package in transport. This included loss of confinement of the TNPP Package reactor cooling boundary, hazards associated with natural phenomenon like severe weather, and human errors in preparing the Reactor Module for transport that could lead to failure or degradation of the TNPP Package. These worksheets were produced for the following hazard categories and are presented in the tables listed below:

- Table B.1 – Fire Hazard Events
- Table B.2 – Explosion Events
- Table B.3 – Kinetic Energy Events
- Table B.4 – Potential Energy Events
- Table B.5 – Loss of Containment Events
- Table B.6 – Direct Radiological Exposure Hazard Events
- Table B.7 – Criticality Events
- Table B.8 – Man-Made External Events
- Table B.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, they are not considered to be within the scope of the TNPP Package PRA, which provides a risk-informed basis for just over-the-road transportation.

The first column on the left side of the worksheet 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 material in the CONEX box-like structure (e.g., associated with the Reactor Module, the CONEX box-like structure, or system components external to the TNPP reactor containment boundary)." The third column is labeled the Hazardous Event Summary and is a 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 from unreleased radiological material (e.g., due to degraded shielding), or (3) a criticality that 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 an estimate of the Initiator Frequency identified in the second column. The Initiator Frequency (events per year) intervals listed below are common ranges used in hazard analysis:

- 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 of the 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 5.1 of this report, the following are contributors to the MAR that are identified as applicable for each hazardous condition (i.e., accident scenario):

1. Nongaseous fission products contained in the tri-structural isotropic (TRISO) fuel or heavy metal contamination in the compacts that are subsequently damaged in an accident
2. Fission gases contained in the TRISO fuel or heavy metal contamination in the compacts that are subsequently damaged in an accident
3. Fission products that have diffused from the TRISO fuel and are held up in the core structures
4. Fission products and gases that have diffused from the TRISO fuel and have plated-out in the reactor containment boundary (i.e., RPV 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 B.1. Fire Hazard Events (three 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 CONEX box-like structure e.g., associated with the module, the overpack, or system components external to the TNPP reactor containment boundary)	Release of radiological material from the TNPP Package to the environment caused by damage due to general fire in the CONEX box (e.g., associated with the module, the overpack, or system components external to the TNPP reactor containment boundary).	Anticipated F ≥ 1E-02	Potential damage to one or more of the TNPP containment boundaries 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 containment boundary	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 one or more of the TNPP containment boundaries 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 containment boundary	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

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 CONEX box-like structure (e.g., associated with the module, the overpack, or system components external to the TNPP reactor containment boundary)	Release of radiological material from TNPP Package caused by ignition of grease/oil in a CONEX box-like structure (e.g., associated with the module, the overpack, or system components external to the TNPP reactor containment boundary).	Anticipated F ≥ 1E-02	Potential damage to one or more of the TNPP containment boundaries 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 containment boundary	<u>Passive:</u> Use of no (or low quantity of) flammable lubricants in CONEX box-like structure	<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 Conex box-like 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

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 produce the greatest possible release from the MAR 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)	—	Active: Portable or installed fire detection and/or suppression systems Active: Emergency response from caravan team including setting up a safety perimeter
CONEX = container express; MAR = material at risk; RPV = Reactor Pressure Vessel; SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).							

Table B.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 B.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	—	—	—	—	—
SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant.							

Table B.3. Kinetic Energy Events (six 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 one or more of the containment boundaries 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 containment boundary	Passive: Addressed in functional design criteria of the TNPP Package. Active: Shock and vibration monitoring	Passive: NA 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 CONEX box-like structure Active: NA	Passive: NA Active: NA

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 containment boundary	<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> NA <u>Active:</u> 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 containment boundary	<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> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

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 CONEX box-like 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 containment boundary	Passive: Design of TNPP Package to remain intact from collision impact. Administration Controls: Travel restrictions such speed controls due to road conditions	Passive: NA Active: Emergency response from caravan team including setting u up a safety perimeter

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 reactor containment boundary 5. Contamination outside the reactor containment boundary	Passive: Design of TNPP Package to remain intact from rollover impact. Administration Controls: Travel restrictions such due to road conditions or weather.	Passive: NA Active: Emergency response from caravan team including setting up a safety perimeter

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 containment boundary	<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> NA <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 CONEX box-like structure

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 containment boundary	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: NA Active: Emergency response from caravan team including setting up a safety perimeter Active: Portable or installed fire detection and suppression systems in CONEX box-like structure
CONEX = container express; MAR = material at risk; NA = not available; RPV = Reactor Pressure Vessel; SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).							

Table B.4. Potential Energy Events (two 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 containment boundary	<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> NA <u>Active:</u> Emergency response from caravan team including setting up a safety parameter

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 containment boundary	Passive: Design of TNPP Package to remain intact from collision impact. Administration Controls: Travel restrictions such speed controls due to road conditions	Passive: NA Active: Emergency response from caravan team including setting up a safety parameter

CONEX = container express; MAR = material at risk; NA = not available; RPV = Reactor Pressure Vessel; SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).

Table B.5. Loss of Containment Events (four 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 containment boundary	Passive: Should be addressed in functional design criteria of the TNPP Package Active: Monitoring of Primary Cooling temperature and pressure	Passive: NA Active: Continuous air radiation monitoring Administration Controls: Emergency response

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 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 Btu 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 containment boundary	<u>Passive:</u> Should be addressed in functional design criteria of the TNPP Package <u>Active:</u> Monitoring of Primary Cooling temperature and pressure	<u>Passive:</u> NA <u>Active:</u> Continuous air radiation monitoring <u>Administration Controls:</u> Emergency response

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 CONEX box 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 containment boundary	Passive: Should be addressed in functional design criteria of the TNPP Package Active: Monitoring of Primary Cooling temperature and pressure	Passive: NA Active: Continuous air radiation monitoring Administration Controls: Emergency response

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 containment boundary 6. Contamination on and outside the TNPP Package	Passive: Should be addressed in functional design criteria of the TNPP Package Use of filer vent such as NucFil on for volumes such as the Shield Tank with activated material Active: Monitoring of Primary Cooling temperature and pressure	Passive: NA Active: Continuous air radiation monitoring Administration Controls: 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 containment boundary 6. Contamination on and outside the TNPP Package	Passive: Should be addressed in functional design criteria of the TNPP Package Active: Monitoring of Primary Cooling temperature and pressure	Passive: NA Active: Continuous air radiation monitoring Administration Controls: 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 containment boundary	Passive: Should be addressed in functional design criteria of the TNPP Package Active: Monitoring of Primary Cooling temperature and pressure	Passive: NA Active: Continuous air radiation monitoring Administration Controls: Emergency response
BWXT = BWX Technologies, Inc.; CONEX = container express; MAR = material at risk; NA = not available; RPV = Reactor Pressure Vessel; SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).							

Table B.6. Direct Radiological Exposure Hazard Events (three 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> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

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> NA <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> NA <u>Administration Controls:</u> Radiation worker controls	<u>Passive:</u> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

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.	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: NA Administration Controls: Radiation worker controls. Confirmatory checks before transport	Passive: NA 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: NA Administration Controls: Radiation worker controls. Confirmatory checks before transport	Passive: NA Active: Emergency response from caravan team including setting up a safety perimeter
CONEX = container express; MAR = material at risk; NA = not available; RPV = Reactor Pressure Vessel; SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).							

Table B.7. Criticality Events (three 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 Package 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.	Extremely Unlikely 1E-04 > F ≥ 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 containment boundary	<u>Passive:</u> Design of core to remain subcritical 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> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
	Addition of moderator from fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause to criticality after a crash that results in fire and TNPP damage.	Direct radiation exposure and possible release of radiological material to the environment caused by a criticality event due inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash that results in fire and TNPP Package damage *Entry made as the result of review after the hazards analysis sessions were completed.	Extremely Unlikely 1E-04 > F ≥ 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 containment boundary	Passive: Design of core to remain subcritical after submersion in water. Administration Controls: Restrict delivery of fire suppression water or other hydrogenous material onto a breached TNPP Package and use other fire suppression options.	Passive: NA Active: Emergency response from caravan team including setting up a safety perimeter

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 Package 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 subcritical 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 rollover 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 containment boundary	<u>Passive:</u> Design of core to remain subcritical after submersion in water. TNPP Package 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> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

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 containment boundary	<u>Passive:</u> Design of TNPP Package against fast control rod bank withdrawal at cold conditions during transport. <u>Administration Controls:</u> Travel restrictions such speed restrictions	<u>Passive:</u> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

CONEX = container express; MAR = material at risk; RPV = Reactor Pressure Vessel; SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).

Table B.8. Man-Made External Hazard Events (two 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 Package 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: NA	Passive: NA 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: NA	Passive: NA Active: Emergency response from caravan team including setting up a safety perimeter
	Train impact	See Table B.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.	—	—	—	—	—

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Human caused events	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 containment boundary 6. Contamination on and outside the TNPP Package	Passive: NA Active: Air radiation monitoring Shock and vibration monitoring. Primary Cooling temperature and pressure monitoring Administrative Controls: Confirmatory checks for before transport	Passive: NA 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 containment boundary	Passive: NA Active: Air radiation monitoring Shock and vibration monitoring. Primary Cooling temperature and pressure monitoring Administrative Controls: Confirmatory checks for before transport	Passive: NA 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

CONEX = container express; I&C = instrumentation and controls; IHX = intermediate heat exchange (Module); MAR = material at risk; NA = not available; RPV = Reactor Pressure Vessel; SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).

Table B.9. Natural Phenomena Hazards (seven 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: NA	Passive: NA 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: NA	Passive: NA 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 CONEX box 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 CONEX box 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 B.3 which occur at a higher likelihood.	Passive: Design of TNPP Package to remain intact from seismic event. Active: NA	Passive: NA Active: Continuous radiation monitoring Active: Emergency response from caravan team including setting up a safety perimeter

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 Package to the environment caused by extreme winter snow load leading to structural collapse of CONEX box.	Unlikely 1E-02 > F ≥ 1E-04	Failure of the CONEX box that fails the containment boundary of the package is judged incredible. Note: A normal shipping container will hold 350 pounds per square foot. The CONEX box 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 TNPP Package.	Passive: NA 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 Package to the environment caused by significant winter ice formation load leading to structural collapse of CONEX box.	Unlikely 1E-02 > F ≥ 1E-04	Failure of the CONEX box that fails the containment boundary of the package is judged incredible. Note: A normal shipping container will hold 350 pounds per square foot. The CONEX box 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 TNPP Package.	Passive: NA 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 B.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 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	—	—	—

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 Package to the environment caused by failure of package from severe hailstorm that causes significant vibration of the transport vehicle, containment 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 containment boundary 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.	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 TNPP Package.	Passive: NA Active: Radiation monitoring Active: 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 packaging 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 parameter	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 containment boundary	Passive: Design of TNPP Package to remain intact from tornado event Administrative: Prohibition to transport during potential tornado weather based on national warning system Standard response actions	Passive: NA Active: Emergency response from caravan team including setting up a safety perimeter

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 containment boundary	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: NA Active: 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	Passive: NA Administrative: Prohibition to transport during lightning storms or severe fire danger.	Passive: NA Active: Emergency response from caravan team including setting up a safety perimeter

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: NA 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 be 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: NA Administrative: Prohibition to transport during flooding danger.	Passive: NA Active: Emergency response from caravan team including setting up a safety perimeter
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 Package transportation 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: NA Administrative: Prohibition to transport during volcanic activity.	Passive: NA Active: Emergency response from caravan team including setting up a safety perimeter

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
Tunnel collapse	Tunnel collapse onto CONEX box	Release of radiological material from TNPP Package damaged by tunnel collapse onto CONEX box	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)	<u>Passive:</u> NA <u>Administrative:</u> Prohibition to transport when there is a possibility of environmental conditions that could cause a tunnel collapse, if a tunnel needed.	<u>Passive:</u> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter
Landslide or avalanche	Landslide or avalanche onto CONEX box	Release of radiological material from TNPP Package damaged by landslide or avalanche onto CONEX box	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 Package 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)	<u>Passive:</u> NA <u>Administrative:</u> Prohibition to transport when there is a possibility of heavy flooding, landslides, or avalanches	<u>Passive:</u> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter

Event Class	Initiating Event Category	Hazardous Event Summary	Initiator Likelihood	Physical Consequences	Qualitative Risk Characterization	Preventive SSCs	Mitigative SSCs
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.	—	—	—	—	—
	High environmental temperature during transport.	See Table B.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 TNPP 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 reactor containment boundary 5. Contamination outside the reactor containment boundary	<u>Passive:</u> Design of TNPP Package to maintain containment in extremely cold temperatures. <u>Administrative:</u> Prohibition to transport during extremely cold weather.	<u>Passive:</u> NA <u>Active:</u> Emergency response from caravan team including setting up a safety perimeter
CONEX = container express; I&C = instrumentation and controls; IHX = intermediate heat exchange (Module); MAR = material at risk; NA = not available; RPV = Reactor Pressure Vessel; SSC = structures, systems, and components; TNPP = Transportable Nuclear Power Plant; TRISO = tri-structural isotropic (particle).							

Appendix C – Review by NRC Staff

This appendix provides a summary of the review by U.S. Nuclear Regulatory Commission (NRC) staff of this report. On February 20, 2023 in an email message (PNNL 2023a), Pacific Northwest National Laboratory (PNNL) on behalf of the U.S. Department of Defense's Strategic Capabilities Office (SCO) asked the NRC to provide a review of an initial draft of this report (PNNL 2022). On April 19, 2023, PNNL received an email message forwarded by the SCO from the NRC (NRC 2023a) that contained a letter dated April 14, 2023, from the NRC to the SCO (NRC 2023b) transmitting a request for Information on the initial draft and asking that the response include a marked revision of the report updated to incorporate information associated with the requests. Enclosure 1 to the letter contained the information request, and Enclosure 2 contained observations that an applicant might find useful when using the approach in the report In an application for package approval.

In a September 18, 2023, email (PNNL 2023b), PNNL transmitted to NRC responses to the request for information and an updated version of the report (PNNL 2023c) that incorporated responses associated with the requests.

Additionally, the NRC Advisory Committee on Reactor Safeguards (ACRS) reviewed the report. On November 17 and December 6, 2023, PNNL, NRC staff, and representatives of the SCO provided presentations to the ACRS Subcommittee and Full Committee in Washington, D.C., on the approach presented and demonstrated in the report. The interaction with the ACRS helped inform the NRC staff review of the report and suggest to PNNL refinements that could be made to improve the report. The presentations to the ACRS are discussed in Appendix D.

This version of the report reflects incorporation of feedback from NRC staff and self-identification by PNNL of needed improvements based on the review history described above and associated discussions.

Given that the NRC request for information and the PNNL response to the NRC request for information are both contained in the PNNL response document, that PNNL response is documented in this appendix as documentation of review by NRC staff.

Enclosure 1

PNNL Response to
Request for Additional Information Transmitted by NRC April 19, 2023
Docket No. 71-9396
Project Pele

Response to Request for Additional Information

The following is the PNNL response to NRC request for additional information (RAIs) on a draft report regarding demonstration of a risk-informed approach to support transportation of a transportable microreactor with irradiated fuel using Probabilistic Risk Assessment (PRA). The request is available in the NRC Agencywide Documents Access and Management System (ADAMS) as ML23087A109 (the transmittal letter) and ML23087A110 (the questions). The questions along with the lead-in paragraphs provided below are extracted directly from the NRC file (ML23087A110) and are followed by responses by PNNL in a different font as marked. The questions are organized by section number of the report and question within that section. In some cases, the question was broken down into parts as determined by PNNL to provide an interim response to a request or observation. Some of the responses also address relevant observations made in the Observation attachment to the request (ML23087A111) as a way to provide clarification to the report.

<u>Questions by NRC</u>
By request dated February 20, 2023 (Agencywide Documents Access and Management System ML23066A202), on behalf of the Strategic Capabilities Office (SCO) within the Department of Defense, the Pacific Northwest National Laboratory (PNNL) requested U.S. Nuclear Regulatory Commission (NRC) review of PNNLs document titled "Development and Application of Risk Assessment Approach for Transportation Package Approval of a Transportable Nuclear Power Plant [TNPP] for Domestic Highway Shipment."

This request for additional information identifies information needed by the NRC staff in connection with its review of the request for endorsement. Since, the regulations in 10 CFR Part 71 are prescriptive requirements, the NRC staff evaluated the request for endorsement against some regulatory approaches and methods discussed in Regulatory Guide 1.200 "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities" and integrated safety analyses in NUREG-1520, "Standard Review Plan for Fuel Cycle Facilities License Applications," as they are applicable to evaluating risk of transportation accidents. The requested information is listed by chapter number and title in the report. Each question describes information needed and the staff's justification for asking the question.

1.0 INTRODUCTION

1. Clarify whether the risk assessment approach for transport of a TNPP will be used only for accidents or will it be used for normal conditions of transport. If the approach is used for normal conditions of transport, then different dose criteria may be needed than for accidents.

<u>PNNL Response to Section 1, Question 1, Part 1:</u>
The proposed risk assessment approach for supporting approval of transport of a TNPP was developed to be used only for evaluating accidents. The assumption was made that the design would meet the deterministic requirements like CFR 71.71 for normal conditions of transport (NCT). If a TNPP package cannot meet the regulatory requirements for NCT, that would suggest a different package robustness than assumed in the demonstration PRA, and the package performance would need to be reconsidered against hypothetical accident conditions (HAC) (i.e., the assumptions made in the consequence analyses supporting the transportation PRA would need to be reconsidered). A PRA approach could be developed to evaluate the risk

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associated with normal conditions of transport (NCT) if the requirements for NCT are not met but that condition is not addressed in the current study.

A discussion of NCT and HAC that explains their relevance to the TNPP PRA has been added to new section ("Overview of the Risk Assessment Approach") that was moved to Section 2 (which is now Section 3 in the updated report). Section 3.2 of the updated report refers to the definition of "accident" used in the PRA. Also, a discussion was added to Section 3 (now Section 4 of the updated report) that explains how NCT was considered in developing the proposed risk evaluation guidelines. This discussion is provided in Section 4.2.5.3 of the updated report and described later in response to other NRC questions.

Section 1.3 states that the dose rate regulatory limits will be met during transport, which is in conflict with the statement in section 2.1 which states that "Compliance with all environmental and test conditions in 10 CFR [Title 10 of the *Code of Federal Regulations*] 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." In addition, 10 CFR 71.43 requires no substantial reduction in the effectiveness of the packaging under the conditions specified in normal conditions of transport (10 CFR 71.71); it does not state an acceptable probability.

<u>PNNL Response to Section 1, Question 1, Part 2:</u>
The cited statement from Section 1.3 of the report is meant to refer to regulatory limits for normal conditions of transport. The cited statement from Section 1.3 (now Section 2.1 of the updated report) has been modified to state: "It is assumed that the suite of TNPP containers will meet the NRC and DOT regulatory dose rate limits during shipment for normal conditions of transport (NCT), although the distance at which dose rate limits are met may be part of the exemption request."

However, as indicated, it is assumed that not all 10 CFR 71.73 deterministic tests can be met. In Section 2.1 of the updated report, a sentence was added stating: "However, it is also assumed that the TNPP Package will not meet all environmental and test conditions in 10 CFR 71.41(a) and subsequent 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 subjection to the postulated hypothetical accident conditions specified in 10 CFR 71.73 (HAC)." As a follow-on to this statement and to address an observation made in the NRC Observations attachment (ML23087A111), a statement was added to Section 3.2 of the updated report stating "When it is known which 10 CFR 71.73 deterministic tests can be met and which cannot, it is possible that certain accidents could be excluded from consideration in the TNPP transportation PRA. However, in practice, it would likely be difficult to align the crash conditions with conditions created by the 10 CFR 71.73 tests. The fact that the hypothetical accident condition tests are performed sequentially as specified in 10 CFR 71.73 to determine their cumulative effect on the package make this comparison even more difficult."

We agree with the last statement ("it does not state an acceptable probability") and take it to mean that per 10 CFR 71.43 there is no probabilistic element to meeting normal conditions of transport. We note that the current requirements for packaging under 10 CFR Part 71 are deterministic.

The transportation regulations in 10 CFR Part 71, have different dose rate and containment criteria for normal conditions of transport and hypothetical accident conditions in 10 CFR 71.47 and 10 CFR 71.51(a). (The dose rate criteria are located in 10 CFR 71.41 and containment criteria in 10 CFR 71.51(a)(1) for normal conditions of transport and the dose rate and containment criteria is in 10 CFR 71.51(a)(2) for hypothetical accident conditions.) This recognizes the fact that the impact of radiological material to the public should be lower during normal transport conditions than in an accident. In addition, development of the approach recognizes this in section 3.1 where it states, "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, some of the "accidents" in table 4-5 appear to be similar to the tests and conditions for normal conditions of transport in 10 CFR 71.71, such as items 7e, 8b, 9a, 9c and potentially 11c.

PNNL Response to Section 1, Question 1, Part 3

We acknowledge that the following cited events from Table 4-5 (now Table 5-5 of the updated report) appear to be caused by normal condition of transport:

- Accident 7e (Extreme cold that fails containment),
- 8b (High ambient air temperature and containment failure),
- 9a (Radiolysis and possible hydrogen accumulation),
- 9c (Random vibration or human error),
- 11c (Adverse weather that causes delay).

These events (and others) were identified in the hazard analysis as hazardous conditions that could potentially be considered accidents and are included in Table 4-5 (now Table 5-5 of the updated report) and evaluated for completeness. As stated in Note (d) and (e) at the bottom of new Table 5-5, certain events were evaluated and considered not to be accidents.

Accident 7e and 8b are grouped with other accidents to create bounding representative accidents BRA 7 and BRA 8 as described in Sections 5.3.4.4 and 5.3.4.5 which pertain to breach of the reactor containment boundary (pressurized and unpressured) caused by failure or error and in some cases facilitated by an environmental condition.

However, potential Accidents 9a,9b, 9c and 9d involve only release of contamination from the package but outside the reactor cooling boundary and are therefore considered to be covered by the Radiation Safety program. A note was added to Table 4-5 (now Table 5-5 of the updated report) stating that "Events that cause spread of contamination that affect the worker during transport due to events, environmental conditions, or phenomena that can occur during transport were identified as important hazardous conditions and are shown here for completeness but are not carried forward as accidents that contribute to bounding representative accidents for the reasons explained in Section 5.3.4.6." (Section 5.3.4.6 in the updated report was originally Section 4.4.3.2.6).

Likewise, potential Accidents 11a, 11b, and 11c potentially result in increased unplanned additional exposure to normal radiation from the package and are therefore considered to be covered by the Radiation Safety program. A note was added to Table 4-5 (now Table 5-5) stating that "Events that cause additional radiation exposure to the worker during transport due to delays caused by environmental conditions or technical problems were identified as

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important hazardous conditions and are shown here for completeness but are not carried forward as accidents to contribute to bounding representative accidents for the reasons explained in Section 5.3.4.7." (Section 5.3.4.7 in the updated report was originally Section 4.4.3.2.7).

2.0 DEFINITION OF REGULATORY APPROACH

No questions

3.0 DEFINITION OF SAFETY GOALS AND RISK EVALUATION GUIDELINES

1. Clarify the discussion on document No. DOE-STD-3009-2014, "Preparation of Nonreactor Nuclear Facility Documented Safety Analysis," relating to dose acceptance criteria for an accident.

Section 3.2.1 discusses that the acceptance criteria in DOE-STD-3009-2014 includes "[a] radiological dose of greater than 25 rem to the public and 100 rem to workers are 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." However, on page 17, it also states: "The standard [DOE-STD-3009-2014] states that if the unmitigated offsite 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." These appear to be in conflict with one another for a member of the public.

<u>PNNL Response to Section 3, Question 1, Part 1</u>
The cited statement from Section 3.2.1 (which is in a bullet point on page 18) concerns "a hypothetical risk evaluation scheme" postulated based on "investigation of potential risk evaluation concepts" in DOE-STD-3009-2014 as explained in the preceding paragraphs. (The cited statement is now in Section 4.2.1 of the updated report.)

We acknowledge that the purpose of DOE-STD-3009-2014 is not to define acceptable risk but rather to define when controls are needed. Appendix A.10 of the DOE-STD-3009-2014 standard states that the 25 rem TED Evaluation Guideline is not a safety standard because it does not define acceptable or unacceptable dose.

However, DOE-STD-3009-2014 does endorse the concept of risk ranking in the Hazard Evaluation as discussed in Appendix A.4 and elsewhere of the standard. Based on the frequency and consequence categories defined in the standard and other regulatory guidance we hypothesized criteria shown in Table 3-1 of the report (which are like schemes widely used in the hazardous condition evaluations supporting Documented Safety Analyses for DOE nuclear nonreactor facilities). This is now presented in Table 4-1 of the updated report.

Also, on page 18, last paragraph the DOE-STD-3009-2014 states "This analysis shall demonstrate how SC [safety class] mitigative SSCs [structures, systems, and components] and/or SACs [specific administrative controls] reduce consequences below the EG and how SC (if identified) and SS [safety significant] mitigative SSCs and/or SACs reduce co-located worker consequences below 100 rem," which appears to state that dose to co-located worker should be mitigated to less than 100 rem, which is not what the document shows in its tabular form for the DOE Standard in tables 3-1 and 3-2,

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as both tables show that, for accidents with a frequency less than 10^{-6}, there is no upper limit on the dose.

PNNL Response to Section 3, Question 1, Part 2
Please refer to the discussion above in response to Part 1 of this section. Table of 3-1 (now Table 4-1 of the updated report) presents a "hypothetical risk evaluation scheme" postulated based on "investigation of potential risk evaluation concepts" in DOE-STD-3009-2014 as indicated in the title of the table. In like vein, Table 3-2 (now Table 4-2 of the updated report) is "a hypothetical risk evaluation scheme" based on our examination of guidance in 10 CFR Part 70 and NUREG-1520.

2. Clarify whether the results of an accident that meets the dose rate and containment criteria in 10 CFR 71.51, will also have to meet the quantitative health guidelines (QHGs.)

PNNL Response to Section 3, Question 2, Part 1
We propose that accidents meet only the suggested risk evaluation guidelines like those presented in Table 3-7 (now Table 4-7 of the updated report). However, we show (as discussed below) that if bounding representative accidents meet the proposed surrogate risk evaluation guidelines, then the RIDM acute fatality QHGs would also be met. Concerning the requirements in 10 CFR 71.51, we assume that the criteria for normal condition of transport (NCT) are met but that the criteria for hypothetical accent conditions (HAC) are not met.

Discussion at end of section 3.2, "Development of Risk Evaluation Guidelines Surrogates for Safety Goal QHOs [quantitative health objectives]," notes that some of the lower consequence higher likelihood bins violate the QHGs but seems to argue that the package will be designed to prevent these events anyway. That's more an argument that the package will easily meet these limits rather than an argument that the limits are acceptable.

PNNL Response to Section 3, Question 2, Part 2
The purpose of Table 3-6 at the end of Section 3.2 (now Table 4-6 at the end of Section 4.2 in the updated report) is to present an evaluation of the selected likelihood-dose limits and compare those limits to the RIDM QHGs based on a conversion from dose limits to health effects. Adjustments were made to the proposed risk evaluation guidelines presented in Table 3-7 (now Table 4-7 of the updated report), to ensure the resulting health effects meet the RIDM QHGs. The cited sentence about lower-consequence higher-frequency events being addressed by safety programs was deleted. (Though the statement may be true it caused confusion about the purpose of the table.)

We refined Table 3-6 (Table 4-6 of the updated report) to address Item 2 of Section 3 of the NRC Observations attachment (ML23087A111) that notes that determination of health effects from the proposed dose limits are based only on the lower end of defined frequency intervals for acceptable dose. We adjusted the determination of health effects to be based on the upper end of the defined frequency intervals for an acceptable dose. Accordingly, the conversion to health effects and subsequent comparison to RIDM QHGs is based on using highest allowed frequency of the accident frequency interval and the highest allowed consequence of the consequence

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interval to calculate health effects. Use of this approach and other conservations described in Section 4.2.5.4 of the updated report results in conservative determination of risk evaluation guidelines limits.

After updating Table 3-6 (now Table 4-6 of the updated report), it was necessary to adjust our proposed risk evaluation guidelines presented in Table 3-7 (now Table 4-7 of the updated report) to ensure the proposed risk evaluation guidelines meet the RIDM QHGs.

3. Clarify whether terminology such as accidents, anticipated occurrences, etc., are defined in a manner that is consistent with NRC regulations or provide a definition of the terms.

<u>PNNL Response to Section 3, Question 3, Part 1</u>
Terms such as "accidents" and "Anticipated Operational Occurrences (AOOs)" were to be used in a manner that is consistent with NRC regulations. However, clarifications were made in the report about the way the term "accident" is used in the study. The term "anticipated occurrences" is not used in the draft report. The term AOO is used in Section 3.2.4 (now Section 4.2.4) of the report along with terms like Beyond Design Basis Events (BDBEs) which are defined in Section 3.2.4 (now Section 4.2.4) of the report based on their use in NEI 18-04. The guidance in NEI 18-04 has been endorsed by NRC in RG 1.233 which noted no clarifications of terminology. The accident frequency category of "Anticipated" used in the Hazardous Condition Evaluation is defined in Section 4.4.2.1 (Section 5.3.2.1).

The last paragraph of Section 4.4.1, "Approach to Development of Accident Scenarios" (now Section 5.3.1 of the updated report) states that the accidents of interest in the TNPP transportation PRA are events that lead to release of radiological material into the environment or direct radiation exposure to workers or the public. Such events were identified using hazard analysis as a comprehensive approach to identifying potential accident events evaluated for applicability. A definition of accidents was added to the report in Section 3.3 (which is now Section 4.3 of the updated report) titled "Proposed Surrogate Risk Evaluation Guidelines Established to Meet the Safety Goal QHOs" as follows:

> "For the risk evaluation guidelines for TNPP transportation accidents to be appropriately applied, the term "accident" must be clearly defined and used in the PRA in a way that is compatible with criteria used in the risk evaluation guidelines. Given that the safety functions that must be preserved by the TNPP Package during transport as discussed in Section **Error! Reference source not found.** are containment, shielding, and prevention of criticality, the accidents of interest addressed in the PRA are (1) a release of radiological material to the environment, (2) direct radiation exposure from unreleased radiological material (e.g., due to degraded internal or external shielding), or (3) a criticality that potentially involves both direct radiation and release of radiological material."

Section 3, page 13, "discusses potential risk evaluation guideline approaches and presents proposed risk evaluation guidelines for TNPP transportation package risk that are consistent with the U.S. NRC's safety goal philosophy, guidance, and historical practice."

Although the classification of "not unlikely" as greater than 10^{-4} per year could imply the scale could go all the way to 1 or more per year, the NRC has typically used other terms such as "anticipated occurrences," or "off-normal conditions" to describe events or

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occurrences that can be anticipated to occur rather than referring to these conditions/events as "accidents". In contrast, figures 3-1 and 3-2 in the SCO report provide accident frequency versus consequences based on DOE-STD-3009-2014 that cuts off the curve at a frequency of 0.01 per year, which is consistent with an accident as an event that does not have a frequency of 1.

In summary, discussion regarding the intent of calling items that are expected to occur as accidents versus what NRC might consider normal conditions of transport should be included in order to distinguish them from hypothetical accident conditions, see question 1 above, in section 1, "INTRODUCTION." The term 'normal conditions' is not used in 10 CFR Part 70 to represent such things as expected conditions and events. Importantly the regulatory requirements are different for accidents than for normal conditions (e.g., the occupational dose limit in 10 CFR Part 20.1201 for adults) only apply to normal operating conditions and are the primary guidelines in emergencies – see 56 FR 23365; May 21, 1991; the 10 CFR Part 20 dose limit of 100 mrem for the public is the limit for normal operations. Discussion of the use of the term 'accidents' versus normal conditions (e.g., is the intention to apply guidelines/limits for accidents to normal conditions of transport?) may be helpful.

Figure 3-5 (taken from NEI 18-04) uses the term 'event sequence'. Regardless of the term used discussion regarding the limits/requirements for normal conditions of transport (e.g., events expected to occur) and accidents (e.g., those events with a low likelihood of occurrence) – including how this is represented in figures and tables in the document may be helpful. In discussing TNPP Safety Functions (section 4.3, page 61) there is an identification of 'normal conditions of transport' and 'hypothetical accident conditions,' but figures just reflect all events as accidents. Additionally, the term 'anticipated' is provided on page 73 as a frequency greater than or equal to 0.01 as an accident likelihood category - accidents should not be considered normal operating conditions but this implies they are.

PNNL Response to Section 3, Question 3, Part 2

Discussion of the deterministic requirements in 10 CFR Part 71 concerning normal conditions of transport (NCT) and hypothetical accident conditions (HAC) was added to the Section 2 (now Section 3 titled "The Risk-Informed Regulatory Approach" of the updated report). This discussion provided in Section 3.2 "Overview of the Risk Assessment Approach," makes reference to the definition of accidents used in the PRA. This discussion points out that for the most part, the consequences of TNPP transportation accidents are less likely and more consequential than conditions that might be assumed to be normally encountered during transport, but this is not always the case. For example, as determined by the PRA, fire-only events during transport of TNPP, which clearly must be considered accidents, lead to no or very minimal radiation dose consequence.

We note that the term "NCT" is deterministic as there is no probabilistic element to the NCT requirements but for releases of radioactive material, a maximum radiological leak rate (10E-06 A_2 per hour) after a set of conditions and test are met is specified. (These include high and low temperature, high and low external pressure, vibration, water spray, a free drop (from about 1 meter depending on the weight), a corner drop, compression, and a penetration test.)

A new section was added to Section 3 (now Section 4) that explains how NCT was considered in developing the proposed risk evaluation guidelines. Section 4.2.5.3 of the updated report cites

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IAEA SSG-26, Appendix I.64 through I.70 which explains that the acceptable NCT leak rate of 1E-06 A_2 per hour is derived from limiting the effective dose to 2-rem for a worker spending time in a transport vehicle with the package with a specified air space volume, air exchange rate, and breathing rate for one year. This condition differs from a release caused by an accident which is not expected to be continuous and will be over in far less than a year (e.g., in minutes). None-the-less, limiting the effective radiation dose to 2 rem per year for NCT suggests this dose limit may also be acceptable for a worker for accidents with frequencies from the most likely side of the frequency consequence matrix. Accordingly, given the package design meets the requirements for NCT, development of risk evaluation guidelines was performed in way that avoids defining pairs of likelihood-dose threshold limits as unacceptable when the limit is comparable to the risk to workers from NCT.

4. Revise section 3.2.2, to clarify that not all accidents with a dose less than 5 rem are acceptable.

Section 3.2.2 "NRC Performance Criteria for Integrated Safety Analyses of Nuclear Fuel Cycle Facilities" depicts acceptable and unacceptable accident risk regions for the offsite public (Figure 3.3) and workers (figure 3.4) based on information in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," and NUREG-1520, "Standard Review Plan for Fuel Cycle Facilities License Applications." Although NRC understands how these figures were constructed from portions of NRC's regulations and NUREG-1520, the construct of these figures need clarification to accurately represent key elements of NRC's regulations and NUREG-1520.

In particular, figure 3-3 depicts all accidents below 5 rem are acceptable for a 10 CFR Part 70 license. Although this is consistent with the development of the Integrated Safety Analysis (ISA) under 10 CFR Part 70, there are other requirements in 10 CFR Part 70 and expressed in NUREG-1520 that would make unacceptable a 'blanket' approval of a 5 rem dose to the public (e.g., 10 CFR 70.61(c) requires controls to ensure an event with a dose of 5 rem is unlikely). The regulations provide requirements for unlikely and highly unlikely accidents at 10 CFR 70.61 and NUREG-1520 provides guidance regarding a numerical definition of unlikely and highly unlikely (i.e., unlikely is less than 10^{-4} per event per year and highly unlikely is less than 10^{-5} per event per year; page 3-32) and identify dose limits with respect to high and intermediate consequences (e.g., greater than 25 rem and 5 rem, respectively, for the offsite public; page 3-A-2). As explained in NUREG-1520, this construct was done to "identify accidents for which the consequences and likelihoods yield an unacceptable risk index and to which items relied on for safety must be applied" (page 3-A-3). The report by the SCO appears to interpret this information as a public dose less than 5 rem is acceptable in all situations including accidents with a probability of 1 (there is a later discussion on concern with considering high probability events as accidents). The SCO report ignores the fundamental aspect of the 10 CFR Part 70 and NRC regulations, in general, that the 'acceptability' of the 'not unlikely' accidents is evaluated under the radiation protection program as described in NUREG-1250, "Report on the Accident at the Chernobyl Nuclear Power Station," (section 4):

> "[T]he reviewer should be aware that accident sequences considered 'not unlikely' in the ISA summary are constricted, under the ALARA requirement in 10 CFR Part 20, to minimize exposure to personnel and the public" (NUREG-1520; page 4-13). The not unlikely category includes those accidents with a probability

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greater than 10^{-4} per event per year (NUREG-1520; page 3-A-6). Thus, the dose for the 'not unlikely' accidents are subject to additional constraints that would be expected to reduce the dose especially for those accidents with a high likelihood of occurring (events greater than 10^{-2}). Additionally, 10 CFR 70.62(c)(i-v) requires the licensee/applicant to identify all credible accident sequences including those that are "not unlikely."

Although the identification of the "not unlikely" accident sequences are not required to be submitted to the NRC, the licensee is required to maintain the analysis of these events onsite including the consequences and likelihood. This information is reviewed by the NRC staff during the initial horizontal and vertical slice review and can be reviewed by the NRC inspectors during routine inspection.

PNNL Response to Section 3, Question 4

The investigation of the NRC performance criteria for integrated safety analyses (ISA) of nuclear fuel cycle facilities was performed (along with investigation of other concepts) to arrive at the proposed risk evaluation guidelines presented in Table 3-7 and Figures 3-7 and 3-8 (now Table 47 and Figures 4-7 and 4-8 of the updated report). The hypothetical radiation dose evaluation guidelines postulated in Table 3-2 and Figures 3-3 and 3-4 (now Table 4-2 and Figures 4-3 and 4-4 of the updated report) is a derivation based on reviewing guidance in 10 CFR 70.61 and NUREG-1520.

However, concerning the caveat noted in the NRC observation above, an explanation on events defined to be greater than 1E-04 being subject to additional restraints was added to Section 3.2.2 (Section 4.2.2 in the updated report) for perspective and completeness.

5. Revise section 3.2.3 to clarify the intent of the use of the Q system from the International Atomic Energy Agency's Specific Safety Guide No. SSG-26 (Rev. 1), "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2018 Edition)."

Section 3.2.3 states "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." The Q system isn't based on analyses of accidents to determine when Type A package is insufficient, as the Q system uses dose to an individual, without regard to evaluation of specific accidents. For special form radioactive material, the Q system uses calculation of a whole-body dose limit of 30 mSv (3 Rem) assuming a distance of 3 m over a period of 3 h. For normal form material, the dose limit for A_2 is set based on a release of $10^{-6}A_2$, which is a "median accident". The median accident is defined as one which leads to complete loss of shielding and to a release of 0.1% of the package contents in such a manner that a bystander subsequently received an intake of 0.1% of this released material, hence the $10^{-6}A_2$ release. Based on this calculation the A_2 value is set to limit the dose to a radiation worker to half the annual limit on intake for each specific radionuclide.

PNNL Response to Section 3, Question 5

We understand the guidance in IAEA SSG-26 Edition 2018 to say that the assumption you refer to "a whole-body dose limit of 30 mSv (3 Rem) assuming a distance of 3 m over a period of 3 h" is an old assumption that has been superseded in the current version of the safety guide. Appendix I.7 discusses this original assumption but goes on to state that the current basis includes consideration of a broader range of exposure pathways than the earlier A_1/A_2 system.

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The parameters of those exposure pathways were used or referenced as a starting point as explained in Section 4.6.3 of the draft report (now Section 7.3). SSG-26 states that for external dose due to photons a reference dose of 5 rem (0.05 Sv) is used. For external dose to beta emitters, a reference dose of 50 rem (0.05 Sv) is used. For internal dose due to inhalation a reference dose of 5 rem (0.05 Sv) is used. For submersion dose due to gas, a reference dose of 5 rem (0.05 Sv) and 50 to the skin is used. The consequence analysis performed in support of the TNPP transportation PRA starts from this more current basis.

However, as far as Section 3 (now Section 4) is concerned, it is the reference dose of 5 rem that is the primary contributor to development of the risk evaluation guidelines.

6. Clarify what appear to be errors/typos in the following:

a. Table 3-4 contains many acceptable/unacceptable, more than/less than phrases which appear to be reversed. e.g., "A …dose…is acceptable if the likelihood is more than…"

<u>PNNL Response to Section 3, Question 6, Part a</u>
The "not applicable" blocks are intended to indicate that the dose limit in the first column did not come from the source identified in the column header. Concerning Table 3-4 (now table 4-4 of the updated report), the report has been updated to explain the meaning of the "not applicable" blocks in the body of text where the table is discussed.

We recognize that the use of "more and less than phrases" in the table can be confusing, perhaps because the likelihood intervals have an upper and lower value. Table 3-4 (now table 4-4 of the updated report) has been updated for clarity.

b. In table 3-6:
- Numbers for worker (last row) appear to be incorrect
- Risk columns don't have any unit labels (fatalities/yr)
- QHG for acute fatality would not apply for the lower consequence bins

<u>PNNL Response to Section 3, Question 6, Part b</u>
Concerning the first bullet, we agree that values for the worker in the last row needed to be updated. That said, Table 3-6 (now Table 4-6 in the updated report) has been updated for other reasons described above and correction of these values has been superseded by other changes. (These other changes include those associated with adjusting surrogate risk evaluation guidelines to meet RIDM QHGs as described in the response to Question 2 of Section 3 and addressing how NCT should be considered for the worker as described in the end of the response to Question 3 of Section 3.)

Concerning the second bullet, unit labels were added to the column headers (i.e., "fatalities/year" and "injuries/year") of Table 3-6 (now Table 4-6 of the updated report) also, differentiation of acute from latent and fatalities for 750 rem is explained in Note (a) of updated Table 4-6.

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Concerning the third bullet, Table 3-6 (now Table 4-6 in the updated report) has been updated to meet RIDM QHGs as described in the response to Question 2 of Section 3.

c. Section 3.2.4, "NRC Endorsed Risk-Informed Methodology in Support of Licensing Advanced Reactor Design," should the phrase "None-the-less, the guidance document presents the frequency-consequence evaluation plot shown in Figure 3-3 ..." really be Figure 3-5, since the caption for Figure 3-5 states "Frequency-Consequence Targets from NEI 18-04, Revision 1"? The caption for Figure 3-3 states "Frequency Consequence Chart for Offsite Public Based on 10 CFR Part 70 and NUREG-1520."

PNNL Response to Section 3, Question 6, Part c

This error had been internally noted. The reference was updated to be Figure 3.5 (now Figure 4.5) of the report (which is Figure 3.1 of the NEI 18-04 report).

7. Revise the statement in the first paragraph in section 3.2.3, "Risk Reference Used in Developing the IAEA Q System," regarding Type B(U) and Type B(M) package testing.

The statement in the first paragraph of section 3.2.3: "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" is incorrect. Hypothetical accident conditions were not designed to represent an actual accident the package would experience during transport but, as stated in the proposed rulemaking dated December 21, 1965 (30 FR 15748), was "chosen that satisfactory performance of a package exposed to them may be considered to give reasonable assurance of satisfactory performance in accidents likely to occur in transportation."

PNNL Response to Section 3, Question 7

The cited statement in the first paragraph of Section 3.2.3 (now Section 4.2.3) has been replaced with the statement quoted from the Federal Register as suggested as it provides a more accurate articulation of the purpose of the HAC tests.

8. Clarify what is meant by the "not applicable" blocks in table 3-4.

Table 3-4, "Summary of Relevant Risk Limits from Other Applications," shows a number of dose rate blocks labeled "not applicable." It is not clear what "not applicable" means in this context.

PNNL Response to Section 3, Question 8

Concerning Table 3-4 (now Table 4-4 of the updated report), the report has been updated to explain the meaning of the "not applicable" blocks in the body of text where the table is discussed. The "not applicable" blocks are intended to indicate that the dose limit in the first column did not come from the source identified in the column header.

9. Clarify the following statement in section 3.3, "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."

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The term "remain intact" is a vague description. Does this mean, no release of radioactive material from the package? Also, it appears that these events seem to be normal conditions of transport; however, the acceptance criteria in Part 71 for normal conditions of transport (dose rates in 10 CFR 71.41 and containment criteria in 10 CFR 71.51(a()1)) are lower than the acceptance criteria for hypothetical accident conditions (dose rate and containment criteria in 10 CFR 71.51(a)(2). The dose criteria listed in the document appear to be for accidents, not normal conditions of transport. (See question 1, above, in the Introduction.)

<u>PNNL Response to Section 3, Question 9</u>

The term "remain intact" is admittedly a vague description and has been removed given that its use in reference to Table 3-6 (which is now Table 4-6 in the update) is no longer needed.

As described earlier we provide an explicit definition in Section 3.3 (now Section 4.3 of the updated report) for the term "accident" for the purposes of this report as "accidents of interest addressed in the PRA are: (1) a release of radiological material to the environment, (2) direct radiation exposure from unreleased radiological material (e.g., due to degraded shielding), or (3) a criticality that potentially involves both direct radiation and release of radiological material." Section 4.2.5.3 was added to the updated report to explain how NCT is considered in the development of the risk evaluation guidelines. Additionally, the report has been updated to more clearly explain that certain kinds of events were not carried forward to become part of bounding representative accidents, but rather are seen as operational upsets that are managed using normal programs such a radiation safety programs (see Table 5-5, Note (d) and (e), Section 5.3.4.6 and Section 5.3.4.7).

4.0 TNPP TRANSPORTATION PRA METHODOLOGY, DATA, AND RESULTS

1. Clarify the following provided in the risk assessment approach for a TNPP:

   a. Screening Analysis

   Provide information on when a screening analysis will be performed and what screening criteria is used, including, if applicable, what types of scenarios will be screened out. The basis for screening and how it will be performed and documented as part of the framework should be further described/explained. This should include an initial list of events, the screening process, and a final list of events. Currently, it is not clear which, if any, scenarios have been excluded and is also not clear how several scenarios that have no, or minimal consequences survived the screening process.

   <u>PNNL Response to Section 4, Question 1, Part a, Subpart 1</u>

   Item #14 of Section 4.4.2.2 (now Item 16 of Section 5.3.2.2 of the updated report and second paragraph of Section 4.4.3.1 (now Section 5.3.3.1 of the updated report) explain that accidents with "conditions estimated to be of low risk were screened out because (1) the likelihood was determined to be "Beyond Extremely Unlikely" (which is defined for the hazard analysis in Section 4.4.3.1 (now Section 5.3.3.1 of the updated report) or (2) the consequences were determined not to significantly impact any of the TNPP radiological inventory contributors listed in Section 4.4.2 (now Section 5.3.2 in the updated report). In the hazardous condition evaluation tables presented in Tables 8.2-1 through 8.2-9 (Now Appendix B, Tables B.1 through B.9 of the updated report) the low-

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risk hazardous conditions are identified in the 6th column from the left. These conditions were evaluated as low risk during the hazard analysis process based on the criteria above and then screened out during the accident identification process. The remaining hazardous conditions were converted into accident events which are listed in Table 4-5 (now Table 5-5) and described in Sections 4.4.3.1.1 through 4.4.3.1.31 (which is now Section 5.3.3.2 through 5.3.3.33 of the updated report)

The accident scenarios that were qualitatively screened out based on the criteria cited above are not explicitly listed in the draft report other than in the Appendix 8 hazardous condition evaluation tables (now Appendix B of the updated report).

Section 4.4.2.2 (Page 70, Item 14) states: "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…"; however, table 4-26 presents a risk summary of the bounding representative accidents (BRA) and includes accidents that have no release of radiological material and no loss of shielding (presenting 0 consequences in the table) for BRA 1, BRA 4L; consequences on the order of a microrem for BRA 7; and consequences on the order of a millirem or less for BRA 2 and BRA 8 (5 of the 12 BRAs have very low consequences).

<u>PNNL Response to Section 4, Question 1, Part a, Subpart 2</u>
We acknowledge that Table 4-26 (now Table 7-6 in the updated report) shows bounding representative accidents that were determined after evaluation to result in no or very low dose consequences. Accident scenarios were defined based on identification of hazardous conditions that were qualitatively judged to produce enough damage to the TNPP package to result in non-zero dose consequences as described Section 4.4 (now Section 5.3 of the updated report) on accident identification and development. During, the consequence analysis stage described in Section 4.6 (now Section 7.0 of the updated report), detailed evaluations of the radiological consequences were performed that determined the dose to be very low or zero for certain bounding representative accidents. The bases (i.e., source term development) for these determinations are described in Section 4.6.2 (now Section 7.2 of the updated report). For example, Section 4.6.2.1 (now Section 7.2.1 of the updated report) describes why fires that originate with within the transport module do not produce enough damage to the package to cause a release. These consequence results for these scenarios were, none-the-less, presented with the other results to provide perspective and to show completeness, (though they might have been screened).

b. Clarify the definition of fission products used in the document.

A wide range of radionuclides (e.g., Pu isotopes, which are actinides) are included as a class of fission products, for example, in table 4-1; however, in section 4.2.4.1 (page 56) the text indicates that fission products and actinides are separate groupings. Consider defining what the term "fission products" includes in the document and then use it consistently throughout the document.

<u>PNNL Response to Section 4, Question 1, Part b</u>
The term "radionuclides," as it is used to define Material at Risk and Source Term released, encompasses actinides, fission products and activation products, as appropriate. As such, this is consistent with the way the term "fission products" is used in the first sentence of Section 4.2.4.1 (now Section 5.1.4.1 of the updated report). The report was updated for consistent use of the term "fission products" which involved revising the title of Table 4-1 (now Table 5-1 in the updated report) to be "Radionuclide Classification," and revising the first column of this table to be "Radionuclide Grouping" consistent with the terminology used in NUREG-1465.

c. Clarify the release fractions in section 4.4.3.1.1, "Accident 1(a) – Collision with a Light Vehicle," for collision with a light vehicle.

Section 4.4.3.1.1 (Page 79) states: "The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set <u>lower than values used for collision with a light vehicle.</u>" This sentence states that the values should be set <u>lower</u> than the values used for collision with a light vehicle; however, <u>this section is for "collision with a light vehicle".</u>

<u>PNNL Response to Section 4, Question 1, Part c</u>
We acknowledge this misstatement (which we had also internally identified). The cited sentence in Section 4.4.3.1.1 (now Section 5.3.3.2 of the updated report) revised to state: "The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set lower than the values used for collision with a <u>heavy</u> vehicle."

2. With respect to section 4.4.2.2, "Hazardous Condition Evaluation Assumptions," provide clarifications for the following:

a. In item number 11 (page 70) it states that the hazards analysis assumed no prohibition of transport in extreme weather, and that "this assumption was reconsidered in the accident analysis." It is not clear what was meant by that statement.

<u>PNNL Response to Section 4, Question 2, Part a</u>
The cited statement in Item 11 of the assumption list in Section 4.4.2.2 (now Item 13 in Section 5.3.2.2 of the updated report) has been updated for clarity. We intended to indicate that an assumption was made (later) that a shipment would not deliberately be made in severe weather conditions. Therefore, we deleted the cited statement (i.e., "this assumption was reconsidered in the accident analysis") and replaced it with an updated version of the second part of the item (i.e., "With the exception that it was assumed a shipment would not deliberately be made in severe weather conditions.").

However, in practice it should be noted that the way extreme or inclement weather events are included in the PRA are as contributors to the crash events counted to derive the very large truck accident frequency as explained in Item #12 of the assumption list in Section 4.4.2.2 (now item 14 of the assumption list in Section 5.3.2.2 of the updated report).

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3. With respect to section 4.4.3.1.10, "Tornado or High Wind Event," provide clarifications for the following:

   a. Additional information is necessary to describe how the likelihood of these events are defined. The section states that "the highest frequency along the route could be used to be conservative." Considering the potential that the highest frequency event may result in lower consequences due to lower wind speeds vs. a lesser frequency event that could result in higher consequences due to the higher wind speeds, clarify how these differences in events levels and their associated frequencies are being considered when defining likelihood and consequences in the accident progression analysis (e.g., 40-mph wind event could be more likely to occur (i.e., higher frequency) than a 90-mph event; however, one level of high wind event may fall into anticipated event with low consequences vs. the other be an unlikely with higher consequences; or for tornadoes an F4 tornado may just be considered as extremely unlikely, vs. an F1 tornado may just be unlikely).

   PNNL Response to Section 4, Question 3, Part a

   As explained in Item #12 of the assumption list in Section 4.4.2.2 (now item 14 of the assumption list in Section 5.3.2.2), the way extreme or inclement weather events are included in the PRA are as contributors to the crash events counted to derive the very large truck accident frequency. However, there is not enough information in the large truck datasets used to separate large truck accident events by different wind speeds. Furthermore, deriving an event frequency using location dependent wind speed data for a moving vehicle that causes a crash would be overly complex. Therefore, the risk impact of high frequency lower wind speed versus low frequency higher consequence speed events is not explicitly addressed. However, the frequency of high wind correlated to crashes is accounted for in the data used to calculate crash frequencies.

   b. The frequency of a tornado or a high wind event will vary from one type of event vs. another. Because of this, for similar wind intensity or wind loads each event will be associated with a different frequency. Clarify how these differences will be captured and evaluated for each type of event if there are evaluated under the same accident condition.

   PNNL Response to Section 4, Question 3, Part b

   As indicated in the response above 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, are not treated considered in separate scenarios for wind events. To further clarify, we also updated Item 12 of the assumption list in Section 4.4.2.2 (now Item #14 in assumption list in Section 5.3.2.2 of the updated report) to state: "Moreover, the mechanical impact associated with very large truck crashes was assumed to dominate the accident phenomena, and as a result weather phenomena were not factored into determination of source terms factors (e.g., High wind was not assumed to increase the impact or dilute the concentration of released material.)"

4. Clarify the accident progression described in section 4.4.3.2.8, "Criticality Accidents," of the report regarding criticality under accident BRA 9.

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This section of the report states that the accident consists of "drop into a body of water (e.g., from a bridge) and enough impact to cause a change in core geometry." Clarify whether the change in geometry is a necessary precursor to package criticality, or if flooding with water alone is enough to initiate criticality. This information may affect the frequency determination of accident BRA 9.

PNNL Response to Section 4, Question 4
The wording used to define BRA 9 (now BRA 9A) includes the criterion of "enough impact to cause a change in core geometry." However, the vendor clarified that the "prototype unit will not preclude criticality during a water immersion and inundation event." Therefore, it is assumed in the TNPP transportation PRA that if the TNPP is submerged a criticality will occur. That said, an alteration in core geometry is possible from the drop event and could contribute to criticality. The description of criticality events in Section 4.4.3.2.8 (now Section 5.3.4.8 of the updated report) was updated to state: "For the demonstration TNPP design, the change in core geometry is not required to cause criticality if the core is inundated but an alteration in core geometry is possible as a result of a drop event and could contribute to criticality."

5. Clarify the basis for the frequency of accident BRA 9, "Criticality Event Involving Drop into a Body of Water."

Section 4.7.11 of the report states, regarding the frequency of criticality events involving a drop of the package into a body of water:

> "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 basis for the conclusion that the frequency is less than 5E-07 per year is not clear. The 2.1E-06 estimate is well within the frequency range considered in figures 3-1 and 3-2 for the maximally exposed offsite individual and co-located worker, respectively. The applicant does not give a reason for assuming that the frequency is lower than the 2.1E-06 estimate developed in section 4.5.3.1.2, "Frequency of Highway Accidents that Could Result in a Criticality Event," of the report.

PNNL Response to Section 4, Question 5
We acknowledge that Section 4.7.11 (now Section 8.1.11 of the updated report) and Section 4.5.3.1.2 (now Section 6.3.1.2 of the updated report) did not provide the full basis for determining that the frequency of this flooded criticality event is less than 5E-07 per year. The summary in Section 4.5.3.1.2 (now Section 6.3.1.2 of the updated report) of this flood criticality event involving a drop into a body of water has been updated to provide a more complete explanation of the basis for the accident frequency estimate of this accident.

As summarized in Section 4.5.3.1.2 (now Section 6.3.1.2) of the report two approaches were used to estimate the frequency of this accident, one using GIS information and the other using nationwide very large truck data. A detailed description of the GIS approach is provided in Section 4.5.1.4 (now Section 6.1.3 of the updated report). The summary in Section 6.3.1.2 of the updated report, as described below, draws from the detailed description of this GIS approach and from Section 4.5.3.1.2 (now Section 6.3.1.1 in the updated report) on nationwide data for submersion events.

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The description in Section 4.5.1.4 (now Section 6.1.3 of the updated report) of the report explains that the estimate for the frequency of a flooded criticality event using geographic information system (GIS) data is based on the following conditions:

- an accident occurs within a 100-foot segment of the route near a sufficiently deep body of water (i.e., five feet or greater for at least part of the year) that is 50 meters or so closer from the highway, and
- where there is sufficient slope from the highway to the body of water (i.e., sufficient slope is assumed to be a 1-to-4 slope), so
- that as a result of any accident, the truck and tailor (with the reactor vessel) will always slide or roll into the body of water from the crash.

The probabilities of these conditions are multiplied by the very large truck accident rate for the five states along the assumed route to produce the 2.1E-06 per year estimate. However, in reality there are other conditions needed that would reduce the estimated frequency of this accident if credited. Though difficult to estimate the probabilities of these other conditions include the fact that: 1) the required conditions do not necessarily exist simultaneously along the 100 foot segment, 2) many accidents would not leave the road enough to be caused to slide or roll down the adjacent slope, 3) the truck, trailer and Reactor Module may come to rest short of the body of water depending on the circumstances of the cash, the ground surface between the roadway and body of water, and the presence of rocks, shrubs and trees that may impeded their slide or roll, and 4) and the water may not be sufficiently deep during the time of year the accident occurs or at the point in the stream, river, or other body of water were the Reactor Module ends up.

Given the uncertainty in the estimated frequency using GIS data, it is judged that the estimate could be too conservative by an order of magnitude or more. (If the individual conditional probabilities are assumed to be 50 percent the combined probability of the four conditions would be 6.3%). Even though the estimated accident frequency using this approach is clearly conservative in this case, the approach illustrates the potential value of using GIS to identify road hazards and estimate accident frequencies.

Using the other approach to estimating this accident frequency, as explained in Section 4.5.3.1.2 (now Section 6.3.1.1 of the updated report) on nationwide data for submersion events (the database refers to these as "immersion events") are estimated to occur at a frequency of 5.12E-09 per year (as shown in Table 6-13 of the updated report). This estimate is based on immersion events identified in the dataset as the most harmful events (MHE). The 2016-2019 subcategories of first harmful event (FHE) included immersion or partial immersion, motor vehicle in-transport, and collision with a guardrail face. A total of 12 immersion or partial immersion MHEs were reported during this period for large trucks on interstate highways and all resulted in fatalities and there were no injury-only or property-damage-only events that involved immersion events. It is judged that immersion or partial immersion events for large trucks are likely to be clearly identified and reported because of their uniqueness.

Even allowing for possible underreporting using the data approach because there may be uncounted non-MHEs for BRA 9A, it is judged that the frequency of a flooded criticality event is less than 5E-07 per year. Given, this estimate along with the conservatism explained above for estimating the frequency of this accident using the GIS approach, the final estimated is judged to be less than 5E-07 per year. (The updated report now has two flooded criticality events - BRA 9A and BRA 9B.)

6. The probabilistic risk assessment (PRA) should consider a less than completely flooded package or fire scenarios for criticality under accident scenario BRA 9.

The TNPP core is significantly moderated by graphite as designed and built, such that small amounts of water added to the system could significantly increase system k_{eff}, and result in criticality. The reactor pressure vessel may not need to be fully flooded to achieve criticality. Bodies of water with less depth than required to completely submerge the package may still result in criticality. Criticality analyses of the package with varying levels of water moderation will be necessary to determine the depth for bodies of water to be included in the frequency determination for criticality under accident BRA 9.

PNNL Response to Section 4, Question 6, Part 1
The estimation of the frequency for BRA 9 (i.e., flooded criticality) would not change for bodies of water of less depth based on two separate approaches to estimating this frequency as described in response to Item 5 above.

Additionally, fires in or near TNPP packages are likely to be aggressively suppressed to prevent radionuclide release. In the event the containment is failed, due to impact or other event, water or other hydrogenous fire suppression materials may enter the core in sufficient quantities to cause criticality. Criticality analyses of the package with varying amounts of water or other hydrogenous fire suppression materials in the core may be necessary to determine the frequency of criticality under this accident scenario.

This information may affect the frequency determination of accident BRA 9.

PNNL Response to Section 4, Question 6, Part 2
It is acknowledged that if fire suppression water enters the reactor vessel, it could potentially cause criticality. Therefore, a criticality caused by fire suppression water (or other hydrogenous material) was added as a second variation of a flooded criticality and is identified as bounding representative accident BRA 9B. The original variation of this accident involving a drop into a body of water was re-identified as BRA 9A. This second possibility was added as a hazardous condition in the worksheet presented in Table 8.2-7 (now Appendix B, Table B.7 in the updated report) with a note stating the entry was made as the result of review after the hazards analysis sessions were completed. This scenario was also added to Table 4-5 of Section 4.4.3.1 (now Table 5-5 of Section 5.3.3.1 in the updated report) as potential accident and corresponding discussion was added as Section 5.3.3.32 of the updated report. Description of the likelihood development of this bounding representative accident (i.e., BRA 9B) was added to Sections 4.5.3.1.2 (now Section 6.3.1.2 of the updated report) and a detailed explanation of the development of the accident frequency was added to Section 4.5.5.9 (now Section 6.5.10 of the updated report).

The radiation dose consequences of BRA 9B (like BRA 9A) are not developed but an entry was added to the radiation dose consequence summary in Table 4-26 (now Table 7-6 of the updated

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report). Also, a description of the overall risk results for BRA 9B were added as Section 8.1.12 of the updated report.

The following provides the key elements the detailed description of the development of the accident frequency added to 4.5.5.9 (now Section 6.5.10 of the updated report):

There are five bounding representative accidents that involve fire (i.e., BRA 1, BRA 2, BRA 5-M, BRA 5H, and BRA 6). BRA 1 and BRA 2 are fire-only events that do not involve a crash, and therefore, the containment is intact, and no water intrusion occurs.

BRA 5H (hard impact and fire) and BRA 6 (crash with tanker carrying combustible liquids and fire) have accident frequencies well below the risk evaluation guideline frequency of 5E-07 per year. The risk of accident below 5E-07 per year are acceptable regardless of consequence using the proposed risk evaluation guidelines.

The remaining bounding representative accident - BRA 5M (medium impact and fire) - has an accident frequency of 5.9E-07 per year just over the risk evaluation guideline frequency of 5E-07 per year. However, there are other conditions besides a crash and fire suppression response needed to produce a flooded criticality event that would decrease the estimated frequency of this accident. It is difficult to estimate the probabilities of these other conditions, but they can be characterized in the following way. The crash would need to cause opening in both the Reactor Module and TNPP that would allow water to run into the reactor core. Fire suppression water (or other hydrogenous material) would be directed at the fire which would likely be associated with the engine, wheels or tires, or a fuel spill near under the diesel fuel tanks rather than at the Reactor Module which is carried by the trailer behind the truck. However, a fuel pool could form below the Reactor Module and there could be an opening in the both the Reactor Module and reactor vessel caused by the impact of the crash that allows fire suppression water to enter and inundate the reactor vessel. It is judged that the probabilities of these conditions (though not quantitatively estimated) are enough to reduce the frequency of a flood criticality from fire suppression water due to BRA 5M to below the risk evaluation guideline frequency of 5E-07 per year even without quantitative estimation.

7. Justify using the Q system is appropriate to calculate doses during an accident.

The methodology in SSG-26 was developed to calculate A_1 and A_2 values for individual radionuclides to determine the maximum quantity of material in a package that is not evaluated for hypothetical accident conditions. While the methodology includes external photon dose, external beta dose, inhalation dose, skin, and ingestion dose due to contamination transfer and submersion dose, it does not include neutron sources, except for Cf-252, and does not include interactions that may generate neutrons, such as alpha, neutron (α,n) reactions.

<u>PNNL Response to Section 4, Question 7</u>
The following explanation was inserted into the report after Section 4.6.3.5, "Exclusion of Ingestion and Submersion Dose" (now Section 7.3.6 of the updated report) as Section 7.3.7 "Exposure Pathways Not Addressed by the Q System."

The exposure pathways used in the Q System to calculate the A_1 and A_2 values reported in SSG-26 were selected for evaluation in the TNPP risk assessment methodology because those were

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judged (by the Special IAEA Working Group) to be the dominant pathways for the public and workers to be exposed to radiation as a result of transportation accidents involving radioactive materials. Furthermore, the development of the Q System specifically examined implications on activity release limits for Type B packages in the context of the transport of irradiated nuclear fuels. While other exposure pathways could be evaluated (e.g., external exposure to neutrons, resuspension, skyshine, drinking water ingestion, etc.), these are not expected to be significant exposure pathways for transportation accidents involving radioactive materials. This is especially the case for pathways such as resuspension, skyshine, and drinking water ingestion that would be expected to be mitigated by emergency response to a transportation accident.

With specific regards to exposure to neutrons, SSG-26 (2018 Edition) states: "In the case of neutron emitters, it was originally suggested under the Q system that there were no known situations with (α,n) or (γ,n) sources or the spontaneous neutron emitter Cf-252 for which neutron dose would contribute significantly to the external or internal radiation pathways considered earlier [I.4]. However, neutron dose cannot be neglected in the case of Cf-252 sources." As noted in the RAI, neutron dose from Cf-252 sources (spontaneous fission) is now specifically considered in the development of the Q_A and A_1 values, but this is done to explicitly address the neutron dose risk associated with Cf-252 sources (special form material); the quantity of Cf-252 in irradiated fuel is insignificant and so is a negligible contributor to dose.

Nevertheless, PNNL has performed a bounding assessment of the potential dose contribution from spontaneous neutron emitters present in the Project Pele irradiated fuel. This assessment accounted for the spontaneous fission neutrons emitted from the dominant spontaneous fission neutron sources (specifically, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, U-235, U-238, Am-241, Cm-242, Cm-244, Cf-252, and Np-237). The spontaneous fission neutron dose is dominated by Cm-242 and Cm-244 for short cooling times (less than 5 years), which is also consistent with studies of LWR irradiated fuel. The bounding dose contribution from spontaneous fission neutrons was determined to be less than 0.5% of the photon dose and would be even less if making less bounding but realistic yet conservative assumptions (e.g., the bounding analysis assumes all neutrons are at the peak dose conversion energy while a realistic conservative assessment could use the full neutron energy distribution).

Because of the complexity with assessing the neutron emission rate for alpha, neutron (α,n) reactions, PNNL did not perform a separate evaluation of the neutron dose contribution from these reactions. However, significant evidence is provided in the literature that the neutron dose contribution from these reactions is less than 10% of the dose contribution from spontaneous fission neutrons for the time periods of interest (e.g., transportation of the TNPP package within a few years after reactor shutdown). See, for example, https://publications.jrc.ec.europa.eu/repository/bitstream/JRC112361/report_eur_29301en.pdf These results are for irradiated LWR uranium oxide (UO_2) fuel. The UCO TRISO fuel used in Project Pele, in addition to having significant quantities of oxygen, also has significant quantities of carbon. While the presence of carbon is not expected to significantly alter these results for LWR fuel, this difference is a source of uncertainty that may need to be addressed by an applicant utilizing the risk-informed methodology for transporting irradiated UCO TRISO fuel.

8. Clarify section 4.2.2 with regard to 10 CFR 50.71, "Maintenance of records, making of reports", referencing A_2 values.

Section 4.2.2 states: "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." However, 10 CFR 50.71 does not reference 10 CFR Part 71, Appendix A for isotopes that do not have a specified A_2 value. It is unclear what the statement is conveying.

PNNL Response to Section 4, Question 8
The citation in Section 4.2.2 (now Section 5.1.3 in the updated report) was corrected to say:

> 10 CFR 71, "Packaging and Transportation of Radioactive Material", Appendix A "Determination of A_1 and A_2".

9. Clarify whether the following statement in section 4.2.3.1 is discussing in-reactor operations or during transport:

   "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."

   PNNL Response to Section 4, Question 9
   The cited sentence in Section 4.2.3.1 (now Section 5.1.3.1 of the updated report):

   > "However, transportation of an TNPP that has experienced a DBE or BDBE is beyond the scope of this assessment"

   was revised to state that

   > "However, for the demonstration TNPP transportation PRA, it was assumed that the TNPP being transported has not experienced a DBE and BDBE during operation which would have affected diffusion rates during operation."

   This assumption was also added to the end of the list of hazard analysis assumption list in Section 4.2.2 (now Section 5.3.2.2, Item 17 list of assumptions in the updated report)

10. Clarify table 4-5, item 6, and section 4.4.3.1.12, "Accident 6(b) – Diesel Fuel Fire Only Event," to indicate whether any of the fires include any other combustible components of the truck such as tires. If it does not, justify not including combustible portions of the truck.

    PNNL Response to Section 4, Question 10
    It was assumed that the only external fire of sufficient magnitude to propagate into the Reactor module from the outside is a diesel fuel fire. Other external truck fires such engine fires and wheel or tire fires were assumed not to be of sufficient magnitude to propagate into the TNPP Package. This assumption was also added to the list of hazard analysis assumptions presented in Section 4.2.2 (now Section 5.3.2.2 as Item 11).

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11. Clarify the language in section 4.4.3.1.1, "Accident 2(a) – Collision with a Fixed Object."

In several places there is language like the following "If a worst-case collision with an object is rare and the consequences are high, then..." Consider reviewing the document for language consistency. Is rare considered unlikely, highly unlikely? Is high considered "High"- Consequence group A or "Very High" – Consequence group B. Several places use the term high consequences that cover both groups.

PNNL Response to Section 4, Question 11

In the cited case, a general likelihood term was used because we were not trying to associate its use with a defined likelihood category like those defined in Section 4.4.3.1 (now Section 5.3.2.1 of the updated report) used for hazardous condition evaluation. Also, concerning the proposed risk evaluation guidelines presented in Table 3-7 (now Table 4-7 of Section 4.3 of the updated report), we purposely did not label the likelihood and consequence intervals to avoid confusion with the likelihood and consequence categories defined in Section 4.4.3.1 (now Section 5.3.3.1 of the updated report). In the updated report use of accident likelihood and consequence terms such as in the one cited was reviewed for consistency and adjustments were made in some cases.

12. Clarify whether any of the accidents in section 4.4.3.1, "Identification and Description of the Full Set of Important Accident Scenarios," includes drop onto a lower elevation which could be caused by another accident, such as impact with a light or heavy vehicle or fixed object, jackknife, or rollover.

It appears that the discussion of a drop onto a lower elevation result appears to be a single event or due to a fire, however, it appears that a drop onto a lower surface could be caused by another initiating event.

PNNL Response to Section 4, Question 12

Events that involve a drop to a lower elevation are grouped into the bounding representative accidents (see BRA 3, BRA 5H and BRA 9) as indicated in Table 4-6 of Section 4.4.3.2.9 (now Table 5-6 of Section 5.3.4.9 of the updated report). As such, they contribute to "hard impacts" events and are aggregated with other hard impacts into BRA 3 (crash only) and BRA 5H (crash and subsequent fire). These other hard impacts include collision with a heavy vehicle or an unyielding object with have their own event data as shown in Table 4-16 (now Table 6-10 of Section 6.3.1.1 of the updated report) and own frequencies as shown in Table 4-19 (now Table 6-13 of Section 6.3.1.1 of the updated report). BRA 9A, which is a drop into a body of water that results in criticality, can also be considered a drop to a lower elevation. However, estimation of the accident frequency of BRA 9A has already been described earlier in response to Question 5 of Section 4.

In the nationwide datasets, data fields that could indicate a crash involved "a drop to a lower elevation" were not found. Therefore, a GIS approach used was to estimate the frequency contribution of this possibility (i.e., "drops to a lower elevation") to BRA 3 and BRA 5H as described in detail in Section 4.5.1.5 (now Section 6.1.4 of the updated report) and summarized in Section 4.5.3.1.3 (now Section 6.3.1.3 of the updated report).

As described in Section 4.5.1.5 (now Section 6.1.4 of the updated report), GIS was used to identify topography along the route where there is a drop to a lower surface just off the

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roadway (e.g., on a bridge or overpass, or near a steep embankment). This was assumed to be a drop-off or a slope of at least 1-to-3 within 25 meters of the edge of the roadway. The assumption was also made that if the truck has an accident at these locations and leaves the road, then significantly more damage could occur to the TNPP package if the vehicle drops to a lower elevation. It is conservatively assumed that every crash leaves the roadway at the defined locations and drops to a lower elevation.

As described in Section 4.5.3.1.3 (now Section 6.3.1.3 of the updated report), the probability that segments of the route meet the criteria discussed above was multiplied by the general crash rate for very large trucks for the five states along the assumed route to yield a frequency of 2.3E-06 per year. This frequency was then added with other contributors to estimate the accident frequency of BRA 3 as described in Section 4.5.5.3 (now Section 6.5.3 of the updated report) and BRA 5H as described in the later part of Section 4.5.5.5 (now Section 6.5.5.1 of the updated report).

13. Justify the statement in section 4.6.3.1, "External Dose Due to Photons," that the distance to the closest member of the public is 25 meters from the accident.

The report assumes that the closest member of the public is 25 meters from the accident based on U.S. Department of Transportation isolation and protective action distance for high level radiological material emergency response. There is no justification for why a member of the public cannot be closer than 25 meters during an accident. While the U.S. Department of Transportation Emergency Response Guide states that a cordon of 25 meters surrounding a spill or leak of radioactive material, this would occur after the accident. The report includes two consequence-probability curves to account for public and worker dose for accidents; however, it is not clear why there needs to be two curves if a member of the public can be located closer than a worker.

<u>PNNL Response to Section 4, Question 13</u>

The uncertainty associated with the cited assumption is recognized (i.e., that the public is assumed to 25 meters from the accident), and it is acknowledged that a member of the public could potentially be closer than 25 meters after an accident until a barrier is established of at least 25 meters surrounding accident site. Therefore, the update of the draft report includes a sensitivity study described in the second half of Section 9.2.2 supported by results presented in Table 9-15 through Table 9-20 that addresses this concern. In the sensitivity study a member of the public assumed to be at the same distance from the accident as a worker (which 10 meters for inhalation dose and 1 meter for other dose pathways based the SSG-26 approach). The results of the sensitivity study show that the overall conclusions about risk from two bounding representative accidents are changed from the baseline case given this change in assumptions. This result is described in Section 9.2.2 of the updated report.

For the baseline case, we judged that the public will be 25 meters or further from point of a release and would be quickly evacuated to a safe location if they are closer than 25 meters. The transport is expected to include an escort vehicle in the front and back of the vehicle carrying the Reactor Module. The escort vehicles create buffer space in front of and behind the truck carrying Reactor Module where other vehicles are expected to be prevented from occupying. The escort vehicles are expected also to provide emergency responders in case of an accident. Also, the assumed route will be entirely on interstate highways which significantly limits crashes involving oncoming traffic.

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The question at the end the question concerns why separate risk evaluation risk calculations for the worker and the public given the observation in the question. In defining the two receptors, we are following the lead of the RIDM report (*Risk-Informed Decisionmaking for Nuclear Material and Waste Applications*) and (2) the guidance provided by nuclear non-reactor facility safety approaches such as the approach discussed in DOE-STD-3009. Moreover, by differentiating the risk evaluation guidelines for the worker from the public, we are recognizing that the "worker" has made a choice to be nuclear worker or support nuclear activities and would be expected to be trained in in radiation safety. Also, workers travelling with the transport are at greater risk than a member of public because workers are with the transport for the duration of the transport.

5.0 DEFENSE-IN-DEPTH AND SAFETY MARGIN CONCERNS

1. Revise this section with a focus on what will be developed/presented to describe what is relied on for safety, including the uncertainties with estimating the performance of those items relied on for safety.

While the overall methodology contains the main topics to be addressed in a PRA approach for estimating risk, it appears that the treatment of some of the topic areas, such as defense-in-depth and uncertainty, may not be to the appropriate level of detail or possibly do not address the primary regulatory aspect of the topic. NRC notes that an applicant for package approval should address both areas in much greater detail in its application.

PNNL Response to Section 5, Question 1
It is acknowledged that the defense-in-depth discussion presented in the draft report did not provide the level-of-detail needed for an application and did not sufficiently address many of the unique challenges presented to the demonstration application by applying the defense-in-depth and safety margin philosophies. Therefore, the discussion of the application of each defense-in-depth philosophy in Section 5 (which is now Section 11 in the updated report) has been expanded beyond its original focus.

The updated primary elements of the defense-in-depth philosophy for this application of the exemption process (10 CFR 71.12 are: (1) the robustness of the TRISO fuel and containment, (2) support of safety functions during transport do not rely on active systems, (3) the TNPP transportation risk is quantified and shown to be low, (4) sensitivity studies show that most sources of uncertainty in PRA modeling assumptions and inputs do not impact the conclusions about risk from accidents, and (5) because compensatory measures will be administered and not credited in the TNPP PRA to reduce risk to the worker and the public and uncertainty about risk through preventive and mitigative actions and features.

a. Defense-in-Depth

A number of statements in section 5 appear to imply that defense-in-depth is not really needed due to the low risk. Although a low-risk value may be estimated for a certain activity, NRC's regulatory approach does not dismiss a need for defense-in-depth simply based on risk. NRC considers risk insights gained from

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conducting a PRA to promote an improved understanding of the system in support of the appropriate level of defense-in-depth:

> "Risk insights can make the elements of defense-in-depth more clear by quantifying them to the extent practicable. Although the uncertainties associated with the importance of some elements of defense may be substantial, the fact that these elements and uncertainties have been quantified can aid in determining how much defense makes regulatory sense. Decisions on the adequacy of or the necessity for elements of defense should reflect risk insights gained through identification of the individual performance of each defense system in relation to overall performance." (NRC White Paper on Risk- Informed and Performance-Based Regulation; March 11, 1999; [ML17348B272]).

> "Defense in depth is invoked primarily as a strategy to ensure public safety given the unquantified uncertainty in risk assessments. The nature and extent of compensatory measures should be related, in part, to the degree of uncertainty." (Letter to Chairman Meserve from B. John Garrick [Chairman Advisory Committee on Nuclear Waste] and Dana A. Powers [Chairman Advisory Committee on Reactor Safeguards]); Use of Defense in Depth in Risk-Informing NMSS [Office of Nuclear Materials Safety and Safeguards] Activities; May 25, 2000; [ML003718610]).

Additionally, in Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement (60 FR 42627; August 16, 1995) the NRC has made clear that a defense-in-depth approach is appropriate to all its programs:

> "Deterministic-based regulations have been successful in protecting the public health and safety and PRA techniques are most valuable when they serve to focus the traditional, deterministic-based, regulations and support the defense-in-depth philosophy."

Below are some of the statements that need further consideration regarding the defense-in-depth approach and how the PRA would support an understanding of the defense-in-depth approach appropriate for the TNPP:

Section 5.1, "Defense in Depth Philosophy"

"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."

<u>Comment</u>: It would appear the defense-in-depth approach for design is based solely on a low-risk estimate rather than an articulation of the design basis for the low risk (e.g., the compensatory measures represent operational constraints and not attributes of the design – such as ship at night to avoid other traffic, escort provided forward and aft, etc.). Defense-in-depth precludes a complete reliance

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on one single safety component for safety of the design. The question to be answered is why is the risk low? – what are the safety components that are relied on for safety of the transportation system?

<u>PNNL Response to Section 5, Question 1, Part a, Subpart 1</u>
The cited summary sentence from Section 5.1 (now Section 11.1 of the updated report) was expanded to include other elements of the defense-in-depth philosophy (as primary elements) for this application as shown below:

"The primary elements of the defense-in-depth philosophy for this application of the exemption process (10 CFR 71.12, "Specific exemptions") are: (1) the robustness of the TRISO fuel and containment, (2) support of safety function during transport do not rely on active systems, (3) the TNPP transportation risk is quantified and shown to be low, (4) sensitivity studies show that most uncertainties in modelling assumptions and inputs do not impact the conclusions about risk from accidents, and (5) because compensatory measures will be administered and not credited in the TNPP PRA to reduce risk to the worker and the public and uncertainty about risk through preventive and mitigative actions and features."

Page 192, Item 1, "Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures…"

"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 HAC, it is expected to meet many or most of the requirements."

<u>Comment</u>: It is unclear what is being conveyed in stating that the Project Pele design "may not meet all" of the safety requirements. The question to be answered is: what are the different design aspects that reduce the risk (such as the tri-structural isotropic (TRISO) fuel, the reactor containment, etc.)? Defense-in-depth is about describing the various 'safety' components and explaining the limits of their functionality with respect to reducing risk.

<u>PNNL Response to Section 5, Question 1, Part a, Subpart 2</u>
The question refers to Item 1 and cites text from the report associated Item 1 but cites defense-in-depth philosophy associated with Item 2 ("Preserve adequate capability…."). Therefore, in the response we describe the enhancement of our assessment of the defense-in-depth philosophy for both Item 1 and Item 2 of the first list from RG 1.200.

Concerning Item 1 (Preserve a reasonable balance among the layers of defense), the discussion was amended to discuss the layers of defense (which includes design features as the robustness of the TRSIO fuel to heat and pressure and the robustness of the containment system against release) and explain that no given layer by itself is relied on primarily for nuclear safety. (In the original version, we intended to convey that even though the TNPP package is not expected to meet all 10 CFR 71.55 (b) tests associated with hypothetical accident conditions, it is expected, none-the-less, to possess a very high level of robustness.)

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Concerning Item 2 (Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures), the discussion was amended to state that sensitivity studies show that most sources of uncertainty in PRA modelling assumptions and inputs do not impact the conclusions about risk from transportation accidents.

Page 192, Item 3, "Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences…"

"Redundancy, independence, and diversity are concepts that are more relevant to an operating reactor with redundant active systems."

Comment: While a TNPP is a reactor, it is not an operating reactor. Section 5 does not appear to focus on what makes transporting a microreactor safe (e.g., the TRISO fuel limits release, the reactor core limits release, and the container, express (CONEX) box offering some protection) and how there are protections beyond just the CONEX box.

PNNL Response to Section 5, Question 1, Part a, Subpart 3

The cited text in Item 3 of Section 5.1 (now Section 11.1) has been rewritten to focus on preservation of redundancy, independence, and diversity. The primary safety functions of containment, shielding and maintaining criticality safety are performed by different design features and components. None of the features and components that perform these functions rely on (i.e., are dependent on) active AC power. Shielding and containment are afforded by a combination of the robust design of the TNPP itself independent from the metal CONEX-box-like Reactor Module. Though the Reactor Module does not provide a strong containment function (e.g., it is not leak tight), it would absorb much of the energy of a crash involving impact, and therefore, protects the reactor coolant boundary (which does provide a containment function) from more significant damage. Much of the radiation shielding is bult into and afforded by the reactor vessel itself but is augmented by lead plates in the walls of the Reactor Module. Criticality safety is maintained by a completely different set design features that help prevent reactivity insertion.

Page 193, item 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 a 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."

Comment: The tone of this statement is that there could be reliance on a single component, which is contrary to a defense-in-depth approach. This does not align with other statements in this section that identify such items as the fuel itself and the reactor vessel as significant barriers to release.

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PNNL Response to Section 5, Question 1, Part a, Subpart 4

The cited text in Item 1 (of the second list) of Section 5.1 (now Section 11.1 of the updated report) has been rewritten to focus on how there is not a focus of key safety functions on a single element of design or operation. Again, the safety functions that need to be protected are containment, shielding and maintaining criticality safety. There is no single element of the design or operation that is relied on to ensure key safety functions. Shielding and containment are afforded by a combination of the robust design of the TNPP itself independent from the metal CONEX-box-like Reactor Module and complement each other in the way described above. Criticality safety is maintained by the design features that help prevent of reactivity insertion largely independent from features created for shielding and containment (e.g., rod locking mechanisms). The PRA results demonstrate that the risk one of these safety systems failing is low. Additionally, the reliability of these safety functions will be supported by administrative transportation controls (e.g., is expected that there will an escort vehicle in the front and behind the truck carrying the Reactor Module) and other compensatory measures that are not credited in the TNPP PRA.

Page 192, item 5, "provide time for recovery operations," includes a statement regarding the fuel itself and the reactor vessel as key safety barriers; however, the safety significance of these barriers is completely undermined by items 3 and 4 and emphasis on the lack of redundancy and potential common-cause failures continues on the page 193 with a new set of points (items 1 and 2).

PNNL Response to Section 5, Question 1, Part a, Subpart 5

The question refers to Item 5 from the first list based on RG 1.200 but cites defense-in-depth philosophy associated with Item 5 from the second list from the NRC RIDM report ("Provide time for recovery actions"). Therefore, in the response we describe the update of our assessment of the defense-in-depth philosophy for both Item 5 of the first list from RG 1.200 and Item 5 of the second list from NRC RIDM.

Concerning assessment of Item 5 from the first list from RG 1.200 (Maintain multiple fission product barriers), the question asserts that assessment of safety barriers is "undermined" by the original assessment of Item 3 on redundancy, independence, and diversity and the original assessment of Item 4 on CCF. The assessment of the defense-in-depth philosophies addressed in Item 3 and Item 4 have been significantly updated as described above which now act to support the assessment in Item 5 (which has not been changed.)

Concerning Item 5 of the second list from the NRC RIDM report, the assessment of the philosophy of "Provide time for recovery" has been updated. It is expected that TNPP transportation will 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). It is expected that that the TNPP would be transported using and escorts in the front and back of the truck carrying the TNPP and that personnel in these vehicles would be trained in the emergency response procedures. Results of the TNPP PRA and associated sensitivities studies can be used to enhance recovery

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response. For example, sensitivity studies were explicitly performed that address assumptions made about the distance of the worker and public from the location of the accident and the duration that they were exposed to the release (or direct radiation from unreleased material).

The end of section 5 summarizes by stating defense-in-depth was applied consistent with NRC guidance and available information; however, the summary does not appear consistent with a number of the points made in section 5 that seem to state precisely the opposite. The application should demonstrate that the principle of defense-in-depth is satisfied.

If the package itself is insufficient for this, then the application should identify other attributes of the design that are relied upon or provide a compelling basis for reliance on administrative measures, such as those identified, and compensatory measures in the document that are not explicitly credited in the PRA.

PNNL Response to Section 5, Question 1, Part a, Subpart 6
We acknowledge that the discussion does not address the points made by NRC in this section and that it cannot be concluded that defense-in-depth principles are met based on the original discussion. Accordingly, the section has been updated as described above.

2. Provide a description of a more robust treatment of uncertainty, which could be based on extensive sensitivity analyses.

The proposed framework appears to lack a formal treatment of uncertainties with the exception of proposing sensitivity studies. This is a reasonable approach to characterize the uncertainty and should focus on key parameters that could significantly increase or decrease the estimated risk. If sensitivity studies are the primary method of characterizing the uncertainty, one would expect the number and level of detail of the sensitivities to be robust.

PNNL Response to Section 5, Question 2
An assessment of uncertainty has been performed and is presented in the updated report in Section 10 which includes identification and evaluation of key sources of uncertainty and assessment of parametric uncertainty (to the extent it can be addressed). The examination PRA modelling assumptions to identify and disposition sources of uncertainty that could impact the conclusions about risk is extensive (see Section 10.2.1 of the updated report). It is based on assessment performed for the sensitivity analyses described in Section 9.1.1 in the new report) which informs the sensitivity analyses as well as the uncertainty analysis.

6.0 TECHNICAL ADEQUACY OF TRANSPORTATION RISK ASSESSMENT

No Questions

7.0 CONCLUSIONS

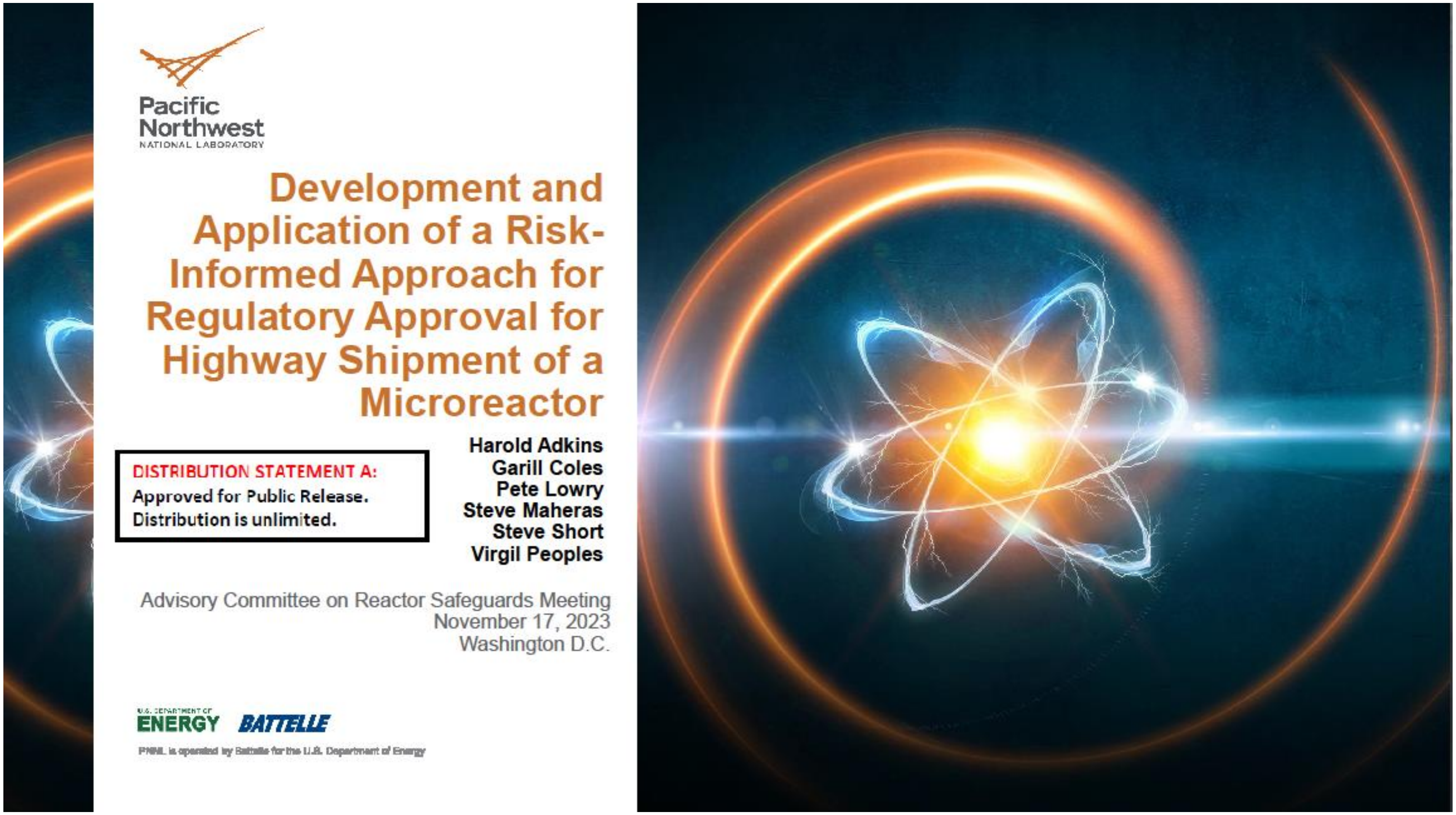
No Questions

Appendix D – Review by the Advisory Committee on Reactor Safeguards

In addition to the review by U.S. Nuclear Regulatory Commission (NRC) described in Section 14.0 and Appendix C, this report was also reviewed by the NRC Advisory Committee on Reactor Safeguards (ACRS). On November 17 and December 6, 2023, Pacific Northwest National Laboratory (PNNL), NRC and the NRC Strategic Capabilities Office (SCO) provided presentations to the ACRS Subcommittee and Full Committee in Washington D.C. after their reviews of report. The presentations focused on the approach presented and demonstrated in the report and the acceptability of the approach to the NRC for licensing. The presentations provided in the November 17, 2023, meetings were made to the Fuels, Material, and Structures Subcommittee. The SCO provided opening remarks about the purpose of the report, PNNL followed with a presentation on development and demonstration of the proposed risk-informed approach for regulatory approval of highway shipment of microreactor, and NRC then provided its review of the approach. Based on ACRS interest and questions about the approach, the SCO, PNNL, and the NRC were invited back for a meeting December 6, 2023, with the full committee to provide focused information of interest to the ACRS. A summary report written by the ACRS and dated December 22, 2023, includes a discussion of the December 6 meeting and documents comments offered by the individual members of the ACRS (which is referenced in Section 14.0). The December 6 version of the presentation contains refinements from the November 17 presentation to clarify or correct technical issues of interest to the ACRS. The November 17, 2023, presentation is provided in this appendix in Section D.1 (pages D.2 through D.37), and the December 6, 2023, presentation is provided in in Section D.2 (pages D.38 through D.54).

This version of the report reflects incorporation of feedback from NRC and self-identification by PNNL of needed improvements based on the review history described above and associated discussions.

D.1 Slides from the November 17, 2023, Presentation



Purpose and Major Elements of Presentation

Purpose: Provide background information on proposed risk-informed regulatory approach for the transportation of a transportable nuclear power plant (TNPP) in support of NRC draft safety evaluation

1. Brief description of the demonstration TNPP
2. Description of the proposed risk-informed regulatory pathway for TNPP transport and why it is needed
3. Development of proposed risk evaluation guidelines
4. Description of quantitative risk assessment process using an integrated assessment process based on probabilistic risk assessment (PRA) methods which includes consideration of defense in depth (DID) and Safety Margin
5. Example results of applying the proposed PRA and risk evaluations guidelines to the demonstration TNPP using PNNL's proposed approach
6. Description of approach to and results of sensitivities studies and uncertainty analyses
7. Insights gained from implemented demonstration of PNNL's proposed approach

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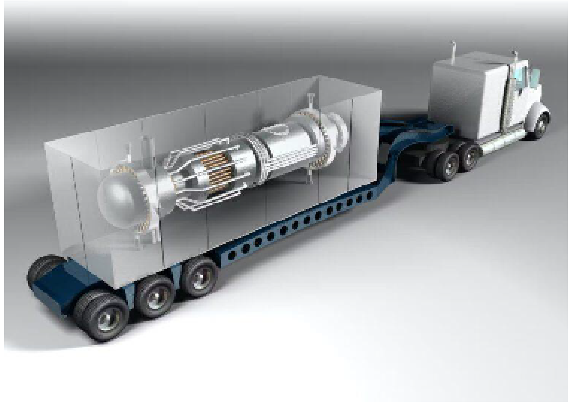
Transportable Nuclear Power Plant (TNPP) Package

- Many advanced reactor vendors are developing TNPPs to make higher density energy readily available for:
  - Department of Defense's (DOD's) domestic infrastructure resilient to electric grid attack
  - Enabling rapid response during Humanitarian Aid and Disaster Relief (HADR) operations
  - Clean, zero-carbon energy in a variety of austere conditions and off-grid locations
- These TNPP conventions would be factory produced, fueled, acceptance tested, and deployed as sealed units prepared for transport and retrieved for refueling and reapplication



Semi-Tractor and Trailer Carrying Reactor Module

Photo courtesy of News & Technology for Global Energy Industry, April 21, 2022
https://www.powermag.com/green-light-for-project-pele-defense-departments-mobile-nuclear-microreactor-demonstration/

3

Project Pele used to Demonstrate Risk-Informed Regulatory Pathway

- 1 to 5 MWe, minimum of 3 years of full power operation
- HTGR using HALEU UCO TRISO fuel
- Multiple modules
    - Reactor Module
    - IHX Module
    - Control Module
    - Power Conversion Module
- Reactor Module contains a vast majority of radioactivity at EOL (remainder in IHX Module)
- Each module contained in and integral with separate ISO-compliant CONEX box-like containers

Artist's rendering of BWXT's Project Pele transportable reactor modules arriving for set up and operation. (Image: BWXT)

Photo courtesy of NuclearNewswire, June 9, 2022
https://www.ans.org/news/article-4035/bwxt-wins-project-pele-contract-to-supply-nations-first-microreactor/

Acronyms: MWe – megawatt electric; HTGR – high temperature gas-cooled reactor, HALEU – high-assay low-enriched uranium; UCO – uranium oxycarbide; TRISO – tri-structural isotropic; IHX – intermediate heat exchanger; EOL – end of life; ISO- International Organization for Standardization; CONEX – container express

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Need for Risk-Informed Regulatory Approach

- US transportation regulatory requirements contained in 10 CFR Part 71 primarily focus on the definition for thick-wall steel vessel for SNF transportation package
- A TNPP with its irradiated fuel contents prepared as a package for transport could be challenged to meet the entire suite of codified regulatory performance requirements in 10 CFR 71
  - It is anticipated that the TNPP will be capable of being deterministically shown to comply with the Normal Conditions of Transport (NCT) as outlined in 10 CFR 71.71
  - However, it may be challenging to demonstrate that the level of robustness of current proposed TNPP technology can fully meet the dose rate and containment success criteria after Hypothetical Accident Conditions (HAC) tests as outlined in 10 CFR 71.73
    - ✓ E.g., Sequential 30 ft free drop, crush, puncture free drop, 30-minute engulfing hydrocarbon fire, and water immersion tests
- Leverage compensatory measures and defense-in-depth approaches and philosophies to <u>reestablish equivalent safety</u>
- Leverage consideration of TRISO, compact, fuel sleeve, core, and reactor structure related inherent retention and protection boundaries

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Basis for Proposed Regulatory Approach

- If Fissile Material or Type B package postulated HAC requirements (10 CFR 71.73) cannot be directly met, then other package approval options are possible:
  - 10 CFR 71.41(c) Alternative Environmental and Test Conditions (10-160B and 8-120B Transportation Casks)
  - 10 CFR 71.41(d) Special Package Authorization (West Valley Melter Package)
  - 10 CFR 71.12 Exemption (Trojan Reactor Vessel)
- Approval of transporting the Trojan Reactor Vessel up the Columbia River and on the Hanford Site was based on compensatory actions as it could not be fully tested.
- Preferred initial pathway identified by PNNL is the Exemption process that allows compensatory actions to protect the basis of exemption if acceptable risk is demonstrated
  - Can apply to more than a single shipment unlike Special Package Authorization
  - Flexibility in deviating from deterministic requirements compared to Alternative Environmental and Test Conditions

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Reasoning Behind Selection of this Regulatory Approval Pathway

- Quantitative risk analysis approaches such as Probabilistic Risk Assessment (PRA) are used in risk-informed regulatory approaches for the NRC:
  - PRAs have been conducted since the 1970s for nuclear reactors starting with WASH-1400 and used since the 2000s for risk informed licensing applications.
  - PRA has also been used to assess:
    - ✓ Dry cask storage systems at a nuclear power plants (see NUREG-1864)
    - ✓ Transportation of spent nuclear fuel (SNF), most notably in NUREG/CR-4829, NUREG/CR-6672, and NUREG-2125
- Proposed to NRC as an aid in developing a near-term approval pathway to drive Advanced Factory-Produced TNPP development and deployment
- Bridges the gap between the current regulatory framework (thick-wall steel vessel based) and the level of robustness of current proposed TNPP technology
- Provides buffer time for strategic regulatory considerations and possible rule making to more so accommodate advanced, transportable, microreactor conventions

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Risk-Informed Regulatory Approval – Using Exemption Process

- Quantitative Risk Assessment - Demonstration of acceptable risk will require a quantitative assessment given (1) the complexities and uncertainties about package performance and (2) potential risk to public. PRA provides a rigorous quantitative approach
  - Unlike the approval pathways used in the past (e.g., Trojan Reactor Vessel), it is unlikely that all accident scenarios can be screened based on likelihood.
- Risk Evaluation Guidelines - Quantitative risk assessments work best when supported by guidelines about acceptable risk as a key basis for regulatory decisionmaking
- However - risk-informed regulatory guidelines using PRA do not exist for transportation packages like they do for nuclear power plants (NPPs)
- That said – The proposed risk evaluation guidelines are based on the risk-informed decision making (RIDM) guidance in NRC 2008 report for nuclear material and waste applications (ML080720238)
  - This guidance includes proposed quantitative health guidelines developed from the 1986 NRC Safety Policy Statement
  - Challenges remain in its implementation and the approach has not been endorsed for use by NRC as that would be a policy decision

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Proposed Risk Acceptance Guidelines 2008 in RIDM Report

NRC-Proposed Qualitative Health Guidelines (QHGs) Based on Interpretation of Safety Policy Statement

Receptor	Acute Fatality	Latent Cancer Fatality (LCF)	Serious Injury (Cancer Illness)
Public	QHG-1 - Public individual risk of acute fatality is negligible if it is less than or equal to 5×10^{-7} fatality per year.	QHG-2 - Public individual risk of a LCF is negligible if it is less than or equal to 2×10^{-6} fatality per year or 4 mrem per year	QHG-3 - Public individual risk of serious injury is negligible if it is less than or equal to 1×10^{-6} injury per year.
Worker	QHG-4 - Worker individual risk of acute fatality is negligible if it is less than or equal to 1×10^{-6} fatality per year.	QHG-5 - Worker individual risk of LCF is negligible if it is less than or equal to 1×10^{-5} 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 5×10^{-6} injury per year.

- **1986 NRC Safety Goal Policy –** The premise is that risk to people from a nuclear power plant should be very small compared to the sum of other accident risk (e.g., 0.1% prompt fatality)
- **Workers** are not specifically addressed in the Safety Goal Policy, so the 2008 RIDM report proposes that worker risk be small compared to other risk but not as small as for the public who are not trained in radiation protection

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Justification for Using Surrogate Measures for QHGs

- As an analog - Levels of NPP PRA include Level I (CDF/LERF), II (release), and III (health effects)
  - However, NPP PRAs (which are mature and well used) are not typically taken to Level III, but rather use the surrogates of CDF and LERF for risk-informed applications, as they are more feasible (see RG 1.200)
- PNNL proposes using surrogates for the QHGs suggested by the 2008 RIDM report by formulating goals in terms of radiological dose and likelihood limits to an individual receptor, which are more feasible to achieve:
  - Reduces calculational burden by eliminating determination of health effects
  - Dose limits can be compared to other federal/international dose limits used in related contexts
  - Determining likelihood and consequence as pairs provides added information for decisionmaking
- PNNL examined the use of dose consequence-likelihood pairs from other applications
  - NEI 18-04 provides risk-informed licensing basis development for advanced non-light-water NPPs
  - DOE-STD-3009 applies risk ranking using dose and likelihood for nonreactor facility nuclear safety analysis
  - NUREG-1513, NUREG-1520, and 10 CFR Part 70 Subpart H provide guidance used in Integrated Safety Analysis (ISA) for determining performance requirements for nuclear fuel cycle facilities
  - The Q system in Appendix I of IAEA Specific Safety Guide (SSG)-26 uses a reference dose to determine an upper quantity limit of radionuclides in Type A package (greater quantities require Type B)

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Development of Proposed Risk Evaluation Guidelines

- Hypothetical risk evaluation guidelines for radiation dose based on guidance for ISA (NUREG-1513)
  - 10 CFR Part 70 defines radiation dose levels for **High** and **Intermediate** consequences for the worker and for an individual member of the public
  - NUREG-1520 provides per year frequency definitions for **Unlikely** and **Highly Unlikely** events

Annual Accident Frequency (per event, per year)	Radiation Dose Consequence to the Offsite Public[(a)]	Radiation Dose Consequence to the Worker[(a)]	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

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

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Development of Proposed Risk Evaluation Guidelines

- Hypothetical risk evaluation guidelines for radiation dose based on guidance for ISA (NUREG-1513)
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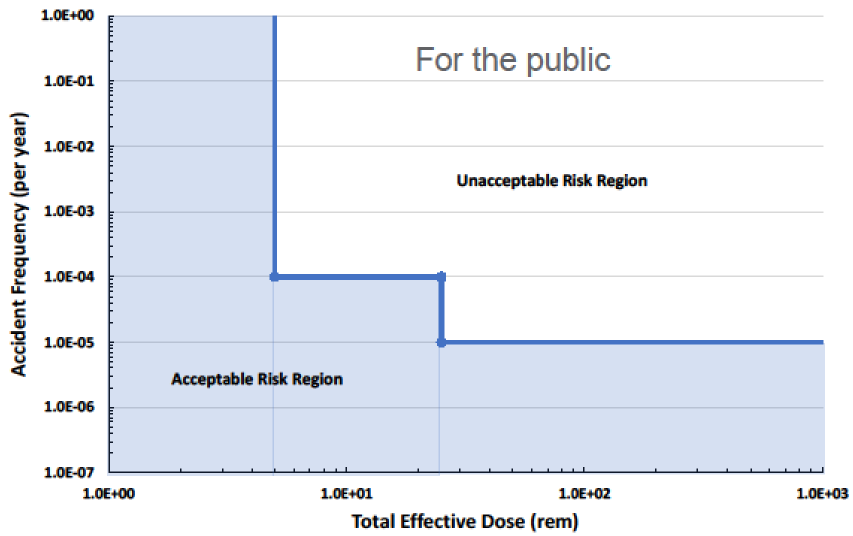
  - NUREG-1520 provides per year frequency definitions for Unlikely and Highly Unlikely events



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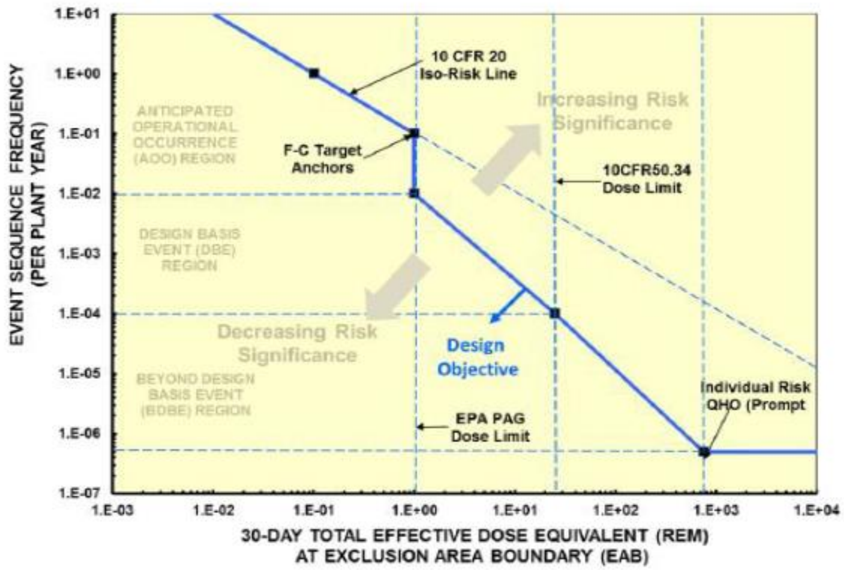
Development of Proposed Risk Evaluation Guidelines

Frequency-Consequence Targets from NEI 18-04, Revision 1



- Illustration of the concept of risk evaluation guidelines based on the combination of radiological dose and likelihood

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Development of Proposed Risk Evaluation Guidelines

1. Synthesized a set of the limits using the likelihood-consequence pairs from or based on the applications investigated for facilities
2. Converted the radiological dose consequence limits to health effects to the worker and a member of the public by multiplying the:
   - Accident frequency
   - Radiation dose consequence from the accident
   - Conversion factors published by DOE used to convert radiation dose to mortality and morbidity[(1)]
3. Readjusted some of the likelihood-consequence pairs to ensure that each limit was less than or equal to the QHGs for acute fatalities proposed in the NRC 2008 RIDM report

Note:
(1) DOE Environmental Policy and Guidance Memorandum, "Radiation Risk from Effective Dose Equivalents (TEDEs)," dated August 2002 based on an Interagency Steering Committee on Radiation Standards (ISCORS) for implementing standards for protection from ionizing radiation

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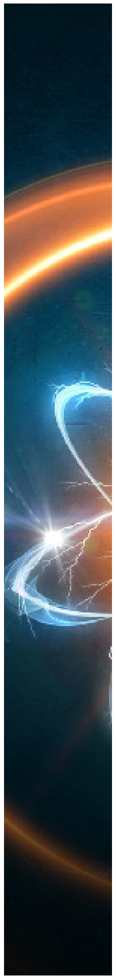
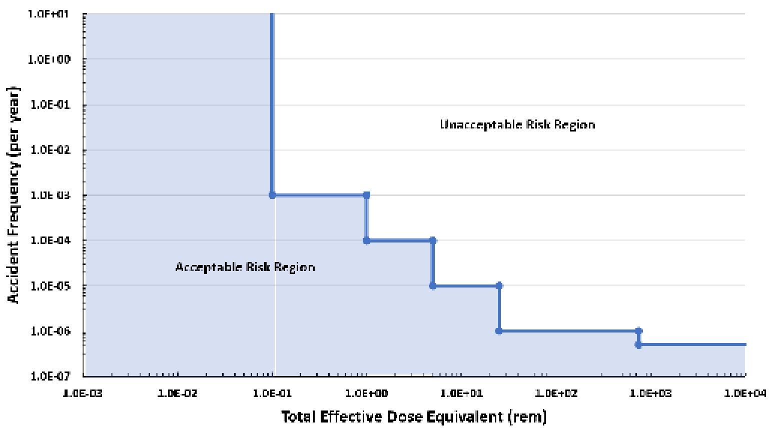


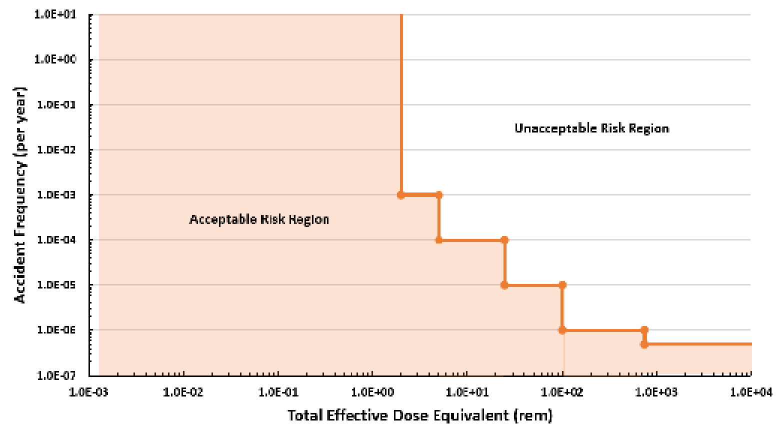
Proposed Risk Evaluation Guidelines

Proposed risk evaluation guidelines compatible with NRC nuclear safety goals, Qualitative Health Objectives, and NRC-proposed QHGs in the NRC 2008 RIDM report

For the Maximum Exposed Member of the Public



For the Worker



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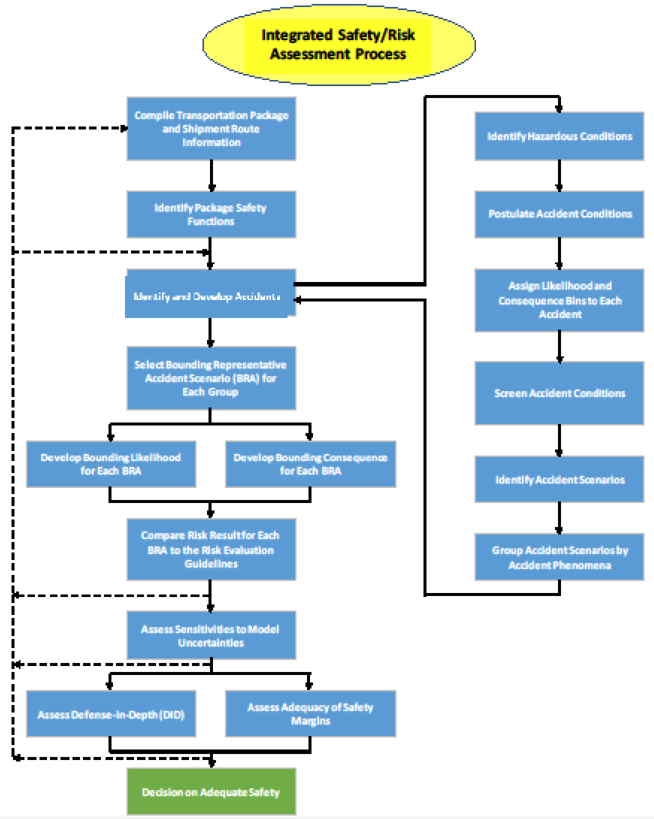
Quantitative Risk Assessment Process

- Uses an integrated risk assessment process based on probabilistic risk assessment (PRA) approaches and methods
- Uses standard methods acceptable to both NRC and DOE for assessing the risk of nuclear facilities
- The process was implemented as a demonstration on a hypothetical shipment of the Project Pele TNPP



Probabilistic Risk Assessment Development Process

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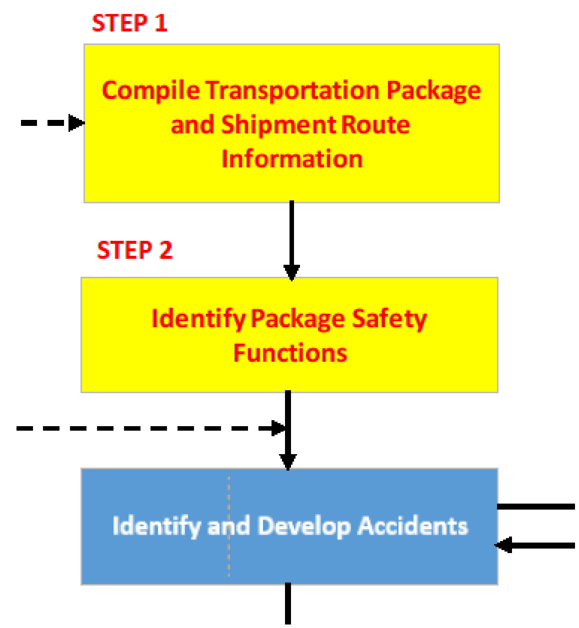
Step 1 – Compile TNPP and Shipment Route Information and Step 2 - Identify Package Safety Functions

Step 1: Information Collection

- TNPP transportation package (Reactor Module only); System design and configuration information, estimated radionuclide inventory at various time periods following reactor shutdown, information on the process for preparing the module for shipment
- Route hazard information, very large truck accident data, and non-vehicle accident data

Step 2: Package Safety Functions

- provide containment of radiological materials
- provide radiation shielding
- maintain a criticality-safe configuration
- maintain passive cooling (considered)



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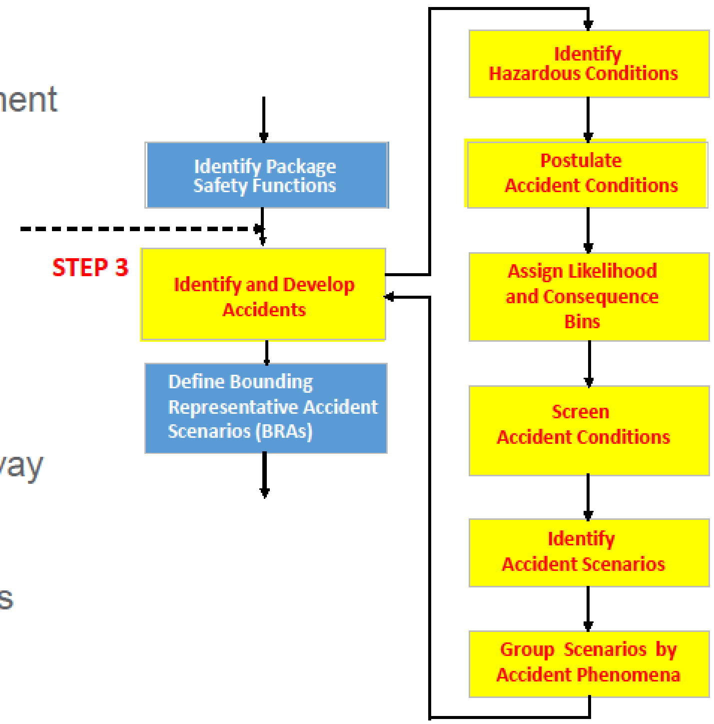
Step 3 – Identify and Develop Accidents

- Performed accident identification and development using Hazard Analysis
- Use of subject matter experts to identify and assess hazardous conditions that could occur during TNPP transport Hazards ID Checklist
- Complete hazardous condition evaluation worksheets that assign likelihood and consequence categories
- Consider both highway accident and non-highway accident initiating events
- Formulate hazardous conditions to contain information needed to define accident scenarios
- Total of 31 accident scenarios representing 8 accident phenomena classes were defined



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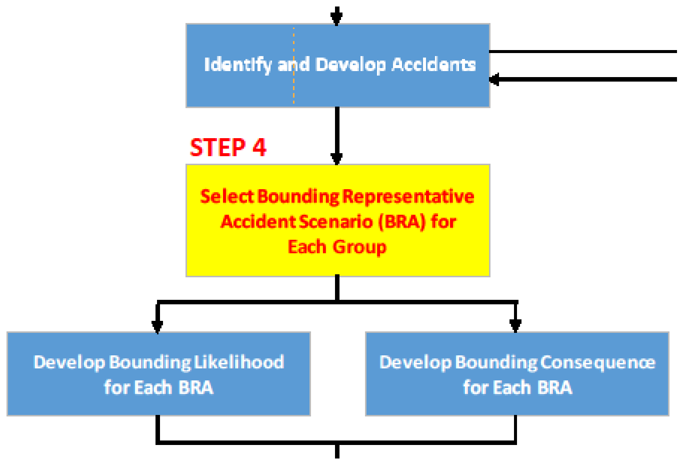
Step 4 – Define Bounding Representative Accident Scenarios (BRAs)



- A BRA is representative of a group of accident scenarios that are phenomenologically similar
- The likelihood for the BRA is determined by the sum of the accidents in the group
- The consequence for the BRA is then determined by the worst consequence of the accidents in the group
- This bounds the risk of all accident scenarios in the group

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Step 4 – List of Resulting Bounding Representative Accidents for this Demonstration Implementation

BRA ID	Description
1	Fire-only event that originates inside the transport module.
2	Diesel fuel fire-only event that originates outside the transport module and propagates into the transport module and ignites combustible material in the transport container, which damages the package.
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 a lower elevation, or rollovers.
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.
4L	Less than a hard-impact highway accident that results in no release of radiological material but some degradation of external shielding. Light impact such as a jackknife, impact with a yielding object (e.g., a road sign or soil embankment), or impact with a light vehicle that is not severe.
5H	Hard impact highway accidents that result in fire with exception of collision with a tanker carrying flammable material.
5M	Medium impact highway accidents (i.e., severe collision with a light vehicle) that results in fire.
6	Collision with a tanker carrying flammable material that leads to fire.
7	Loss of non-pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.
8	Loss of pressurized reactor containment boundary not caused by a road accident but rather by human error and failures of containment features.
9A	Addition of moderator and a possible change in core geometry caused by a drop into body of water that results in criticality.
9B	Addition of moderator and possible change in core geometry caused inundation of the core with fire suppression water or other hydrogenous material that enters the core in sufficient quantities to cause criticality after a crash that results in fire and TNPP damage
10	Control rod withdrawal (or another reactivity insertion event) caused by impact from a road accident that results in criticality.

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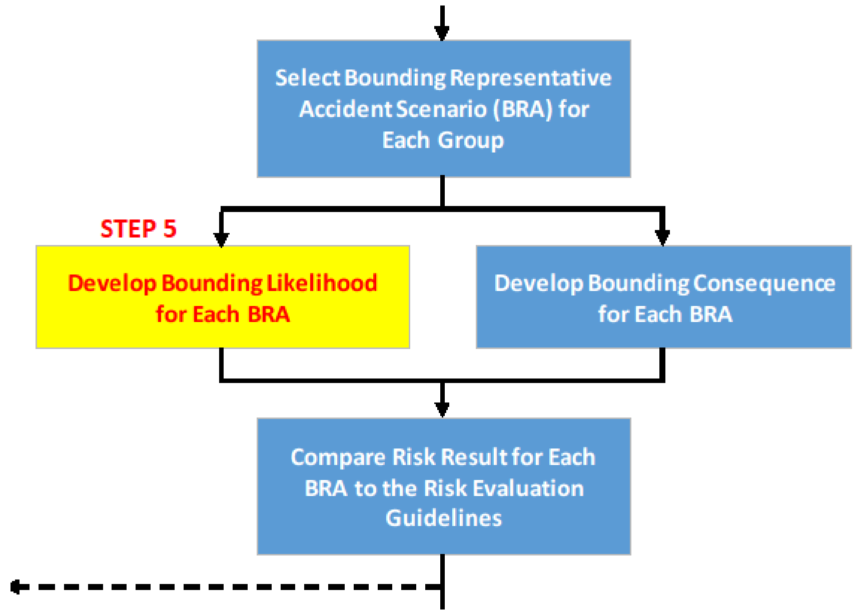
Step 5 – Develop Likelihood for Each BRA

- Very large truck accident data
  - Frequency of impacts, fires, non-impacts, rollovers
  - Use route specific data to the extent possible
- Package-specific failures not in accident rate data
  - Internal-initiated fires, random failures, human error
- Specific route hazard information
  - such as bridges, bodies of water, steep drops to a lower elevation



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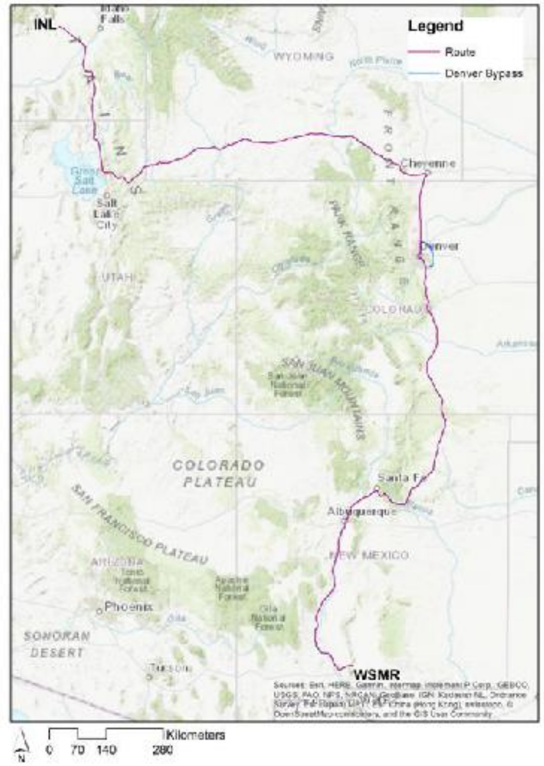
Step 5 – Develop Bounding Likelihood for Each BRA

- The assumed hypothetical route for this demonstration was from Idaho National Lab to White Sands NM (about 1300 miles of Interstate)
- GIS was used to identify portions of the route where hazards existed to compute the percentage of total route where the hazard existed. This includes:
  - Steep drop-offs. If an accident happened here, the truck and package could drop or roll to lower elevation.
  - Sufficient slope to a body of water deep enough to submerge the reactor vessel. If an accident happened here, a criticality could occur.
- Using very large truck crash rate data and hazard data, an accident frequency was computed.



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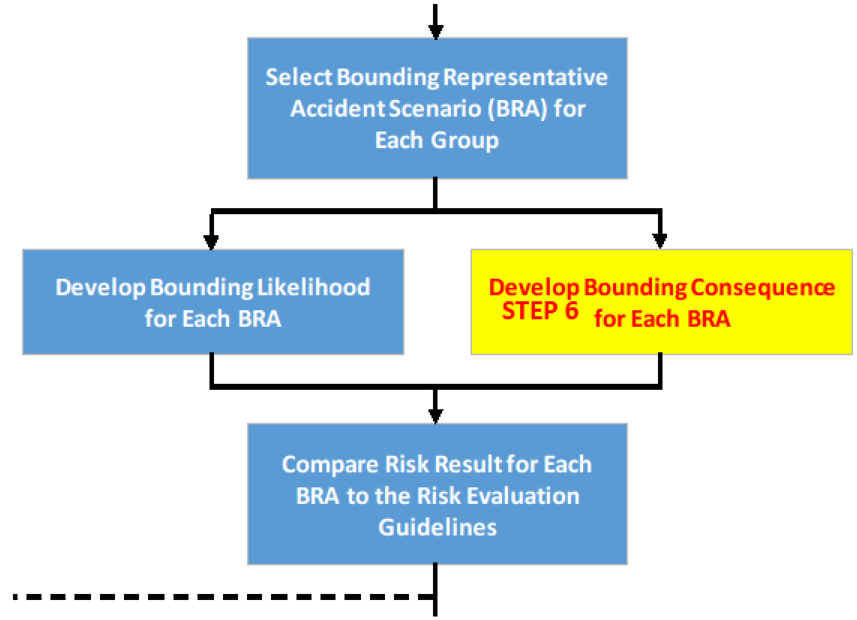
Step 6 – Develop Bounding Consequence for Each BRA

- Estimated effective radiation dose from each dose pathway methodology is based on Appendix I of IAEA SSG-26, with refinements mostly to account for the public receptor.
- The source term was calculated using DOE/NRC methods/data used to determine source term (e.g., MAR × DR × ARF × RF × LPF)
- Source term includes used fuel inventory and inventories diffused into reactor during operation
  - Fuel (concerns about performance under mechanical impact)
  - Core/compact (concerns about fuel qualification)
  - Pressure Boundary (concerns about plating)



Material at Risk (MAR), Damage Ratio (DR), Airborne Release Fraction (ARF), Release Fraction (RF), and Leak Path Factor (LPF)

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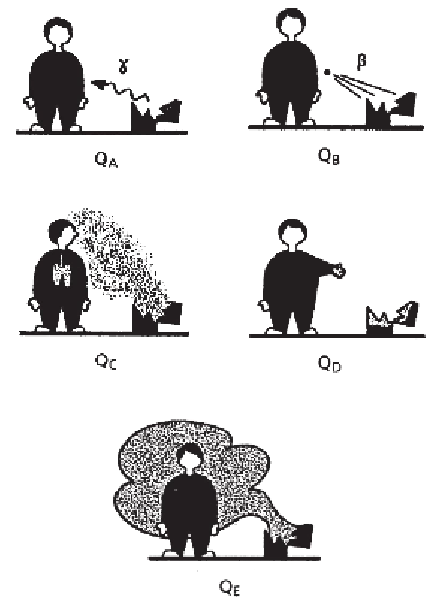
Step 6 – Develop Consequence for Each BRA

- Radiological dose pathways from IAEA SSG-26 (Q System) were used which are the same as in NRC regulations).
  - External Photon Dose (Q_A): External dose due to released material (with added contribution for unreleased material for an individual at given distances from the package with degraded shielding.)
  - External Beta Dose (Q_B): External direct dose from skin contamination due to released material for individual at given distances from the release.
  - Inhalation Dose (Q_C): : Inhalation dose calculated using an airborne source term and human uptake value
  - Skin contamination (Q_D): Calculated from equivalent skin dose but not used because responders are assumed to use protective clothing
  - Neutron Dose: Determined by PNNL to be a minimal contributor in the demonstration for released material. Gamma dominates



*Other pathways excluded by Q system: (e.g., resuspension, skyshine, drinking water ingestion) are not significant contributors for irradiated fuel and would likely be mitigated by the emergency response

**Submersion pathway (see Q_E in the Figure) excluded because the release is outdoors where there will be a high level of dilution

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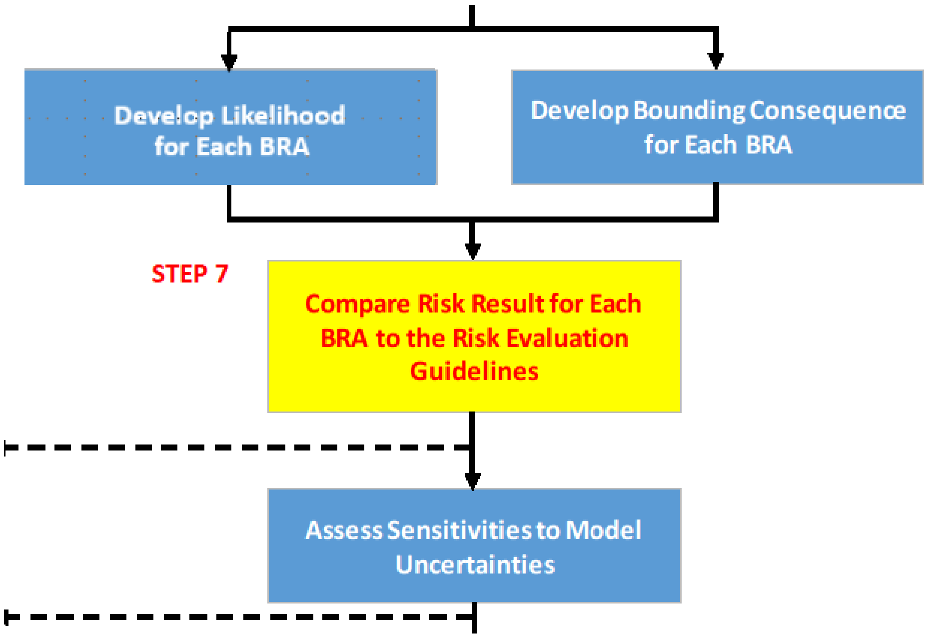
Step 7 – Compare Risk Results to Proposed Risk Evaluation Guidelines

The risk results are reported as the likelihood and consequence for each Bounding Representative Accident



Example Comparisons to Risk Evaluation Guidelines

When the accident frequency is
≤1E-05 and >1E-06 per year

Then the dose limits are:
≥5 and <25 rem TEDE for a member of the public
≥25 and <100 rem TEDE for a worker

or

When the accident frequency is
≤1E-04 and >1E-05 per year

Then the dose limits are:
≥1 and <5 rem TEDE for a member of the public
≥5 and <25 rem TEDE for a worker

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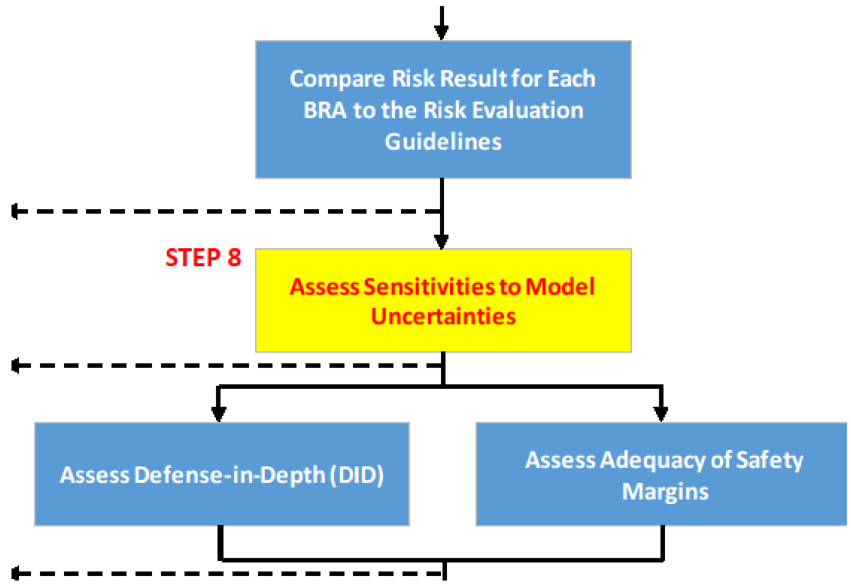
Step 8 – Assess Sensitivities and Model Uncertainties

- Sensitivity studies were performed to address the impact of key assumptions and sources of uncertainty (examples are provided later)
- Sensitivity studies were also considered to address the impact of compensatory actions
- Limited parameter uncertainty analysis typical of PRAs was performed
  - Data for a parametric uncertainty analysis is limited
  - Because each BRA is evaluated with a bounding estimate of the likelihood and consequence



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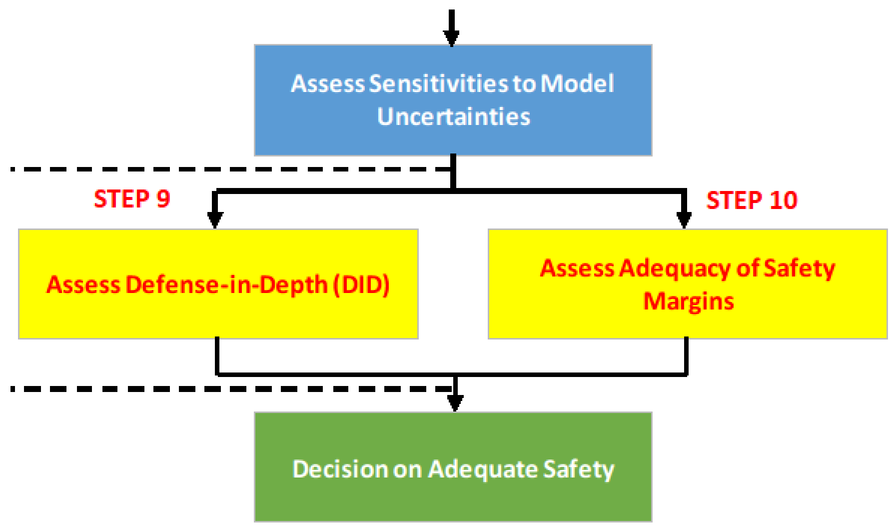
Step 9 – Assess Defense-in-Depth
Step 10 – Assess Adequacy of Safety Margins

- DID is a design and operational philosophy that calls for multiple layers of protection to prevent and mitigate accidents
  - multiple physical barriers to prevent release of radiation
  - passive features
  - PRA shows low risk
  - administrative controls
  - accident recovery plans
- Safety margin is 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
  - Typically to demonstrate adherence to acceptable codes and standards



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Example Risk Results for Bounding Representative Accidents

BRA 2 – Fire Only that Originates from Outside the Transport Container

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥5 and <25 rem TEDE for a member of the public ≥25 and <100 rem TEDE for a worker when the accident frequency is ≤1E-05 and >1E-06
MAR contribution from released material				
TRISO Fuel	0	0		
Core Structure	1.0E-03	2.6E-04		
Cooling System	1.2E-03	2.5E-04		
Contribution from Unreleased Material				
Degraded shielding	0	0		
Total Radiation dose	2.3E-03	5.1E-04		
Accident Frequency assuming one trip per year (from Table 6.16)			2.0E-06	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable

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Example Risk Results for Bounding Representative Accidents

BRA 3 – Hard Impact Road Accident that leads to release of radioactive material and degraded shielding

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from released material				
TRISO Fuel	80.9	18.5		
Core Structure	5.2E-01	1.3E-01		
Cooling System	3.1E-01	6.3E-02		
Contribution from Unreleased Material				
Degraded shielding	6.0	6.9E-02		
Total Dose	87.7	18.8		
Accident Frequency assuming one trip per year (from Table 6.16)			7.1E-05	
COMPARISON TO RISK EVALUATION GUIDELINE				Unacceptable

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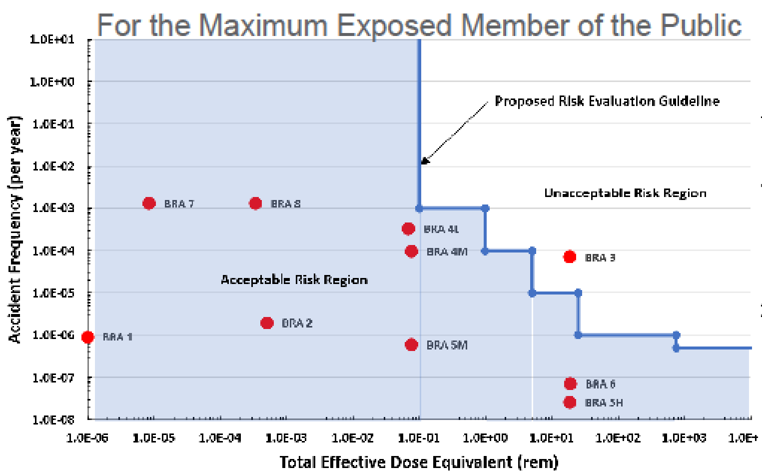
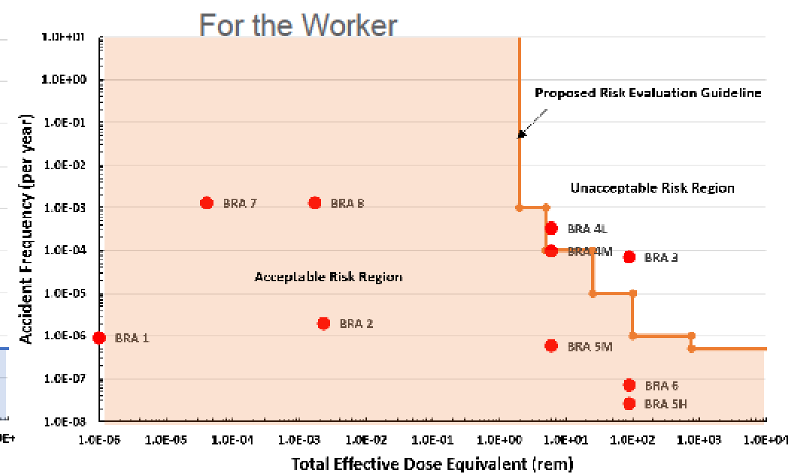
Summary of Demonstration TNPP PRA Risk Results

- Risk for the Bounding Representative Accident Results Shown Graphically





Note: BRA 9A and 9B - two kinds of flooded criticality events - are not shown here because their consequences were not calculated given that their likelihoods were determined to be extremely low.
BRA 10 – reactivity insertion caused by crash impact leading to criticality was not developed because it was anticipated the demonstration design will preclude (or design against) this possibility (e.g., using locking mechanisms)

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Sensitivity Studies

- Selection and definition of the sensitivity cases to be performed were based on:
  - Comprehensive examination of specific lists of assumptions and bases for the hazards, likelihood, and consequence analysis, and
  - Compensatory measures listed for the demonstration design to reduce or mitigate risk
- Quantitative sensitivity studies defined and performed
  1. Decay time after operation
  2. Distance of a member of the public to point of release
  3. Exposure time to a damaged TNPP package
  4. Uncertainty in source term fraction estimates
  5. Restriction of transport during extreme weather (compensatory action)
  6. Transport at night (compensatory action)
- In sensitivity studies - reran the models for applicable BRA to determine new risk results

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Example Sensitivity Study Results

Results of Sensitivity Study on decay time after shutdown on BRA 3 – Hard Impact Road Accident

Sensitivity Study for Impact of Decay after Shutdown				
Delay from Shutdown to Transport	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence (from Table 7.6)				
30 days	1420	319		≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
60 days	208	45.9		
90 days	87.7	18.8		
1 year	14.5	3.3		
2 years	7.8	1.7		
Accident Frequency assuming one trip per year (from Table 6.16)			7.1E-05	
COMPARISON TO RISK EVALUATION GUIDELINE				Acceptable for delay times of 1 year or more

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Uncertainty Analysis and Insights

- In general, there is insufficient data to perform parametric uncertainty analysis (hence the sensitivity studies)
- However, a limited uncertainty analysis was performed on the very large truck accident data

Accident Risk	Worker Dose (rem TEDE)	Public Dose (rem TEDE)	Accident Frequency (per year)	Applicable Proposed Risk Evaluation Guidelines from Table 4.7 of this Report
Accident Consequence by MAR Contribution (Radiation dose from Table 7.6)				≥1 and <5 rem TEDE for a member of the public ≥5 and <25 rem TEDE for a worker when the accident frequency is ≤1E-04 and >1E-05
MAR contribution from Released Material				
TRISO Fuel	0	0		
Core Structure	2.6E-02	6.5E-03		
Cooling System	9.3E-03	1.9E-03		
Contribution from Unreleased Material				≥0.1 and <1 rem TEDE for a member of the public ≥2 and <5 rem TEDE for a worker when the accident frequency is ≤1E-03 and >1E-04
Degraded shielding	6.0	6.9E-02		
Total Dose	6.0	7.7E-02		
Accident Frequency assuming one trip per year (from Table 6.16)			9.7E-05	
Accident Frequency multiplied by 41% to match highest state and year combination			1.4E-04	Worker risk changed from acceptable to unacceptable from comparison to risk evaluation guidelines
COMPARISON TO RISK EVALUATION GUIDELINE				Unacceptable

- The limited uncertainty analysis did not change the conclusions about risk of the BRAs with the exception above for BRA 4 Medium Impact Accident which becomes unacceptable.

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Key Insights from Demonstration PRA Results and Sensitivity Studies

- Allowing the TNPP reactor core to decay up to one year after it has been in operation for 3 years will result in an acceptable level of risk for all BRAs based on the proposed risk evaluation guidelines.
- The conclusions about the risk of BRAs are not sensitive to the uncertainty in estimating the source term factors.
- The conclusions about the risk of BRAs are not sensitive to increasing the accident duration from 30 minutes to one hour.
- The conclusions about the risk of BRAs are not sensitive to decreasing the distance that the public is to the accident to be the same distance as the worker is to the accident, except for light impact accidents (BRA 4L and BRA 4M) in which a direct dose of 6 rem is estimated from degraded shielding.
- While certain compensatory actions are feasible to implement, their impact is difficult to evaluate

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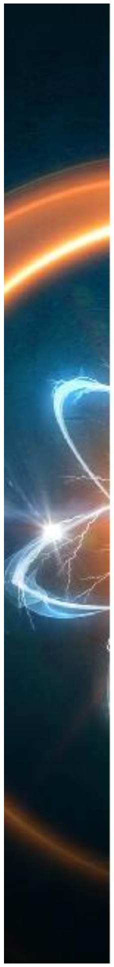
Summary

- Current NRC regulations provide a feasible regulatory pathway for licensing a first-of-kind transportation of a microreactor with irradiated fuel
- Proposed workable risk evaluation guidelines were developed that are compatible with QHGs proposed in the 2008 NRC RIDM report
- The risk-informed PRA-based approach does support an application to the NRC for approval of shipment of a TNPP package (containing irradiated fuel)
- The demonstration application of this approach for a hypothetical single shipment per year of the Project Pele microreactor has shown that the proposed risk evaluation guidelines can be met



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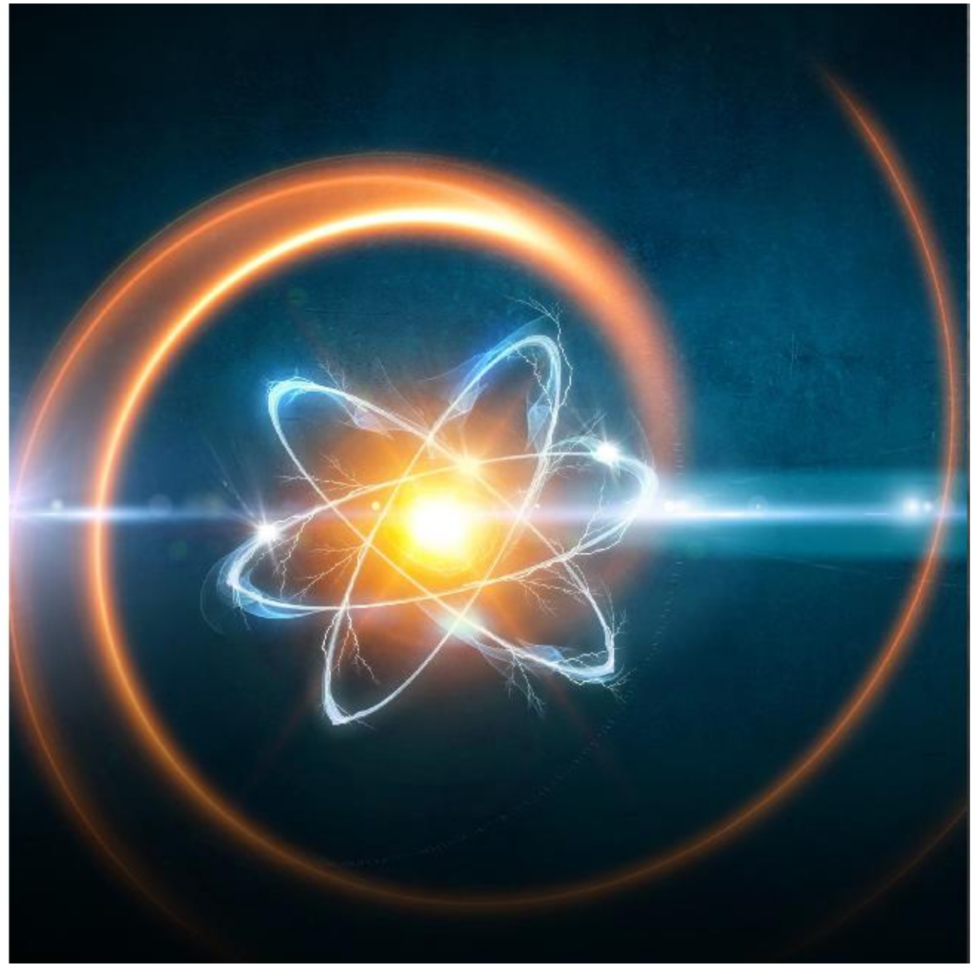
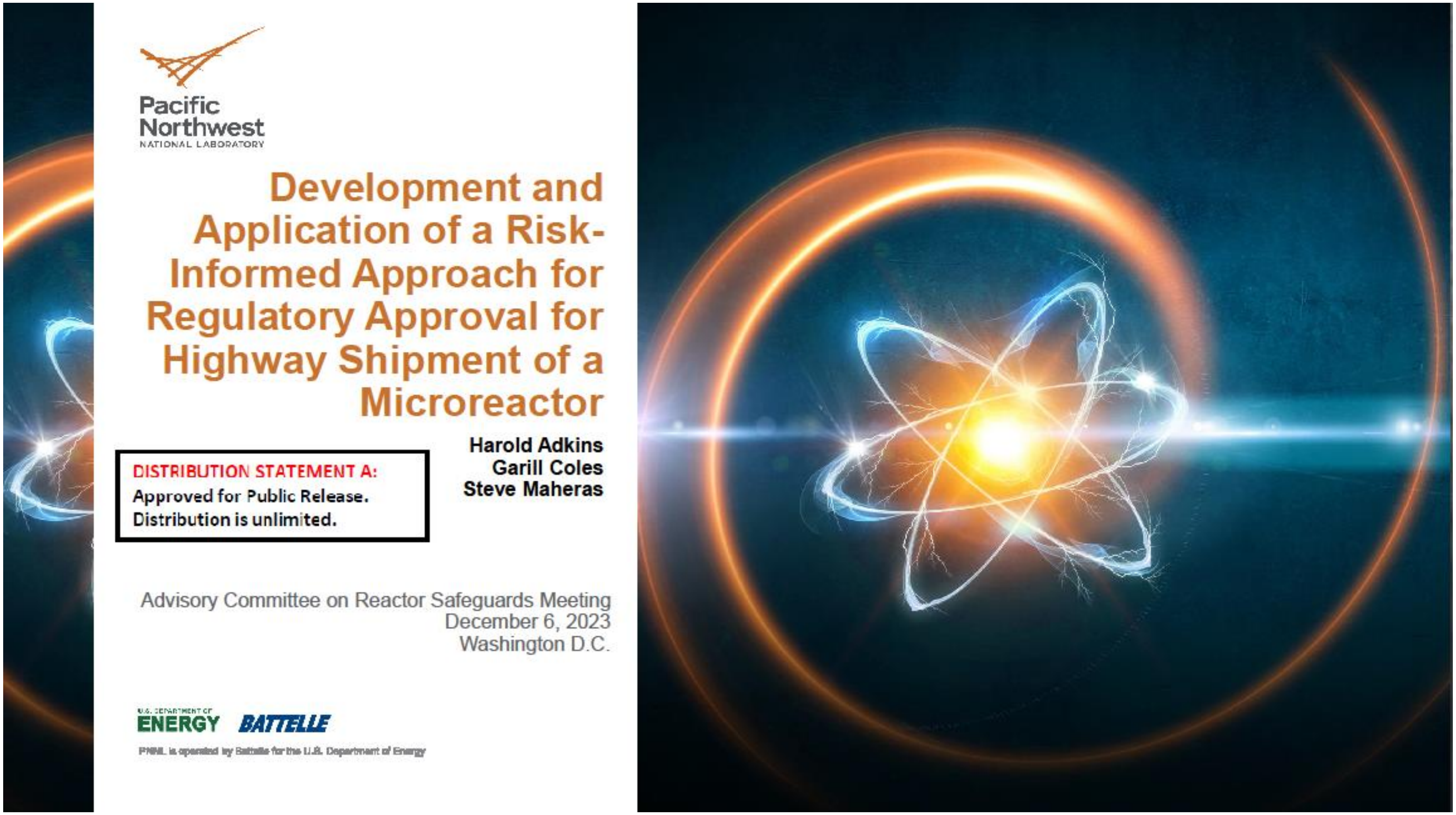
Questions & Discussion



D.2 Slides from the December 6, 2023, Presentation



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Purpose and Major Elements of Presentation

Purpose: Provide background information on proposed risk-informed regulatory approach for the transportation of a transportable nuclear power plant (TNPP) in support of NRC draft safety evaluation

1. Brief description of the demonstration TNPP
2. Description of the proposed risk-informed regulatory pathway for TNPP transport and why it is needed
3. Overview of proposed risk evaluation guidelines
4. Overview of quantitative risk assessment process using an integrated assessment process based on probabilistic risk assessment (PRA) methods which includes use of sensitivities and uncertainty analysis and consideration of defense in depth (DID) and Safety Margin (SM)
5. Example results of applying the proposed PRA and risk evaluations guidelines to the demonstration TNPP using proposed approach
6. Brief clarification in response to questions raised during the November 17 ACRS meeting.

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Project Pele TNPP Package used to Demonstrate Risk-Informed Regulatory Pathway

- Many advanced reactor vendors are developing TNPPs to make higher density energy readily available for:
  - Department of Defense's (DOD's) domestic infrastructure resilient to electric grid attack
  - Enabling rapid response during Humanitarian Aid and Disaster Relief (HADR) operations
  - Clean, zero-carbon energy in a variety of austere conditions and off-grid locations
- These TNPP conventions would be factory produced, fueled, acceptance tested, and deployed as sealed units prepared for transport and retrieved for refueling and reapplication
- Project Pele is a HTGR using HALEU UCO TRISO fuel
  - 1 to 5 MWe, minimum of 3 years of full power operation
  - Comprised of a Reactor, IHX, Control, and Power Conversion Module
  - Reactor Module contains a vast majority of radioactivity at EOL (remainder in IHX Module)
  - Each module contained in and integral with separate ISO-compliant CONEX box-like containers

Semi-Tractor and Trailer Carrying Reactor Module

Photo courtesy of News & Technology for Global Energy Industry, April 21, 2022
https://www.powermag.com/green-light-for-project-pele-defense-departments-mobile-nuclear-microreactor-demonstration/

Acronyms: MWe – megawatt electric; HTGR – high temperature gas-cooled reactor, HALEU – high-assay low-enriched uranium; UCO – uranium oxycarbide; TRISO – tri-structural isotropic; IHX – intermediate heat exchanger, EOL – end of life; ISO- International Organization for Standardization; CONEX – container express

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Need for Risk-Informed Regulatory Approach and Basis for Proposed Regulatory Approach

- A TNPP with its irradiated fuel contents prepared as a package for transport could be challenged to meet the entire suite of regulatory performance requirements in 10 CFR 71 as they are intended for thick-wall steel vessel for SNF transportation package
  - It is anticipated that the TNPP will be capable of being deterministically shown to comply with the Normal Conditions of Transport (NCT) as outlined in 10 CFR 71.71
  - However, it may be challenging to demonstrate that the level of robustness of current proposed TNPP technology can fully meet the dose rate and containment success criteria after Hypothetical Accident Conditions (HAC) tests as outlined in 10 CFR 71.73
    - ✓ E.g., Sequential 30 ft free drop, crush, puncture free drop, 30-minute engulfing hydrocarbon fire, and water immersion tests
- Leverage compensatory measures, defense-in-depth approaches, and philosophies to establish equivalent safety. Also leverage consideration of TRISO, compact, fuel sleeve, core, and reactor structure related inherent retention and protection boundaries
- If Fissile Material or Type B package postulated HAC requirements (10 CFR 71.73) cannot be directly met, then other options such as 10 CFR 71.41(c), 10 CFR 71.41(d), or 10 CFR 71.12 (Exemption) are possible
- Preferred initial pathway identified by PNNL is the Exemption process that allows compensatory actions to protect the basis of exemption if acceptable risk is demonstrated
  - Can apply to more than a single shipment unlike Special Package Authorization (10 CFR 71.41(d))
  - Flexibility in deviating from deterministic requirements compared to Alternative Environmental and Test Conditions

4

Reasoning Behind Selection of this Regulatory Approval Pathway

- Quantitative risk analysis approaches such as Probabilistic Risk Assessment (PRA) are used in risk-informed regulatory approaches for the NRC:
  - PRAs have been conducted since the 1970s for nuclear reactors starting with WASH-1400 and used since the 2000s for risk informed licensing applications.
  - PRA has also been used to assess:
    - ✓ Dry cask storage systems at a nuclear power plants (see NUREG-1864)
    - ✓ Transportation of spent nuclear fuel (SNF), most notably in NUREG/CR-4829, NUREG/CR-6672, and NUREG-2125
    - ✓ Risks of transporting SNF to the Yucca Mountain repository by truck and rail (DOE/EIS-0250)
- Proposed to NRC as an aid in developing a near-term approval pathway to drive Advanced Factory-Produced TNPP development and deployment
- Bridges the gap between the current regulatory framework (thick-wall steel vessel based) and the level of robustness of current proposed TNPP technology
- Provides buffer time for strategic regulatory considerations and possible rule making to accommodate advanced, transportable, microreactor conventions

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Risk-Informed Regulatory Approval – Using Exemption Process

- Quantitative Risk Assessment - Demonstration of acceptable risk will require a quantitative assessment given (1) the complexities and uncertainties about package performance and (2) potential risk to public. PRA provides a rigorous quantitative approach
  - Unlike the approval pathways used in the past (e.g., Trojan Reactor Vessel), it is unlikely that all accident scenarios can be screened based on likelihood.
- Risk Evaluation Guidelines - Quantitative risk assessments work best when supported by guidelines about acceptable risk as a key basis for regulatory decisionmaking
- However – risk-informed regulatory guidelines using PRA do not exist for transportation packages like they do for nuclear power plants (NPPs)
- That said – The proposed risk evaluation guidelines are based on the risk-informed decision making (RIDM) guidance in NRC 2008 report for nuclear material and waste applications (ML080720238)
  - This guidance includes proposed quantitative health guidelines (QHGs) developed from the 1986 NRC Safety Policy Statement for the worker as well as the public
  - Challenges remain in its implementation and the approach has not been endorsed for use by NRC as that would be a policy decision

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Justification for Using Surrogate Measures for QHGs

- However, PNNL proposes using surrogate measures for QHGs proposed in the NRC 2008 RIDM report
  - In the same way that Core Damage Frequency and Large Early Release Fraction are used instead of health effects for risk-informed applications for the current fleet as justified in NRC RG 1.200.
- Specifically, PNNL proposes formulating goals in terms of pairs of radiological dose and likelihood limits to an individual receptor which are more feasible to achieve:
  - Reduces calculational burden by eliminating determination of health effects
  - Dose limits can be compared to other federal/international dose limits used in related contexts
  - Determining likelihood and consequence as pairs provides added information for decisionmaking
- PNNL examined the use of dose consequence and likelihood from other applications
  - NEI 18-04 provides risk-informed licensing basis development for advanced non-light-water NPPs
  - DOE-STD-3009 applies risk ranking using dose and likelihood for nonreactor facility nuclear safety analysis
  - NUREG-1513, NUREG-1520, and 10 CFR Part 70 Subpart H provide guidance used in Integrated Safety Analysis (ISA) for determining performance requirements for nuclear fuel cycle facilities
  - The Q system in Appendix I of International Atomic Energy Agency (IAEA) Specific Safety Guide (SSG)-26 uses a reference dose to determine an upper quantity limit of radionuclides in Type A package (greater quantities require Type B)
    - ✓ Exposure time – 30 min

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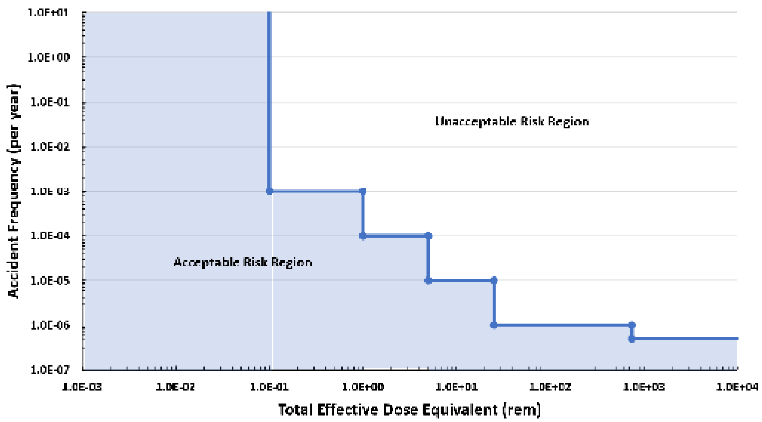
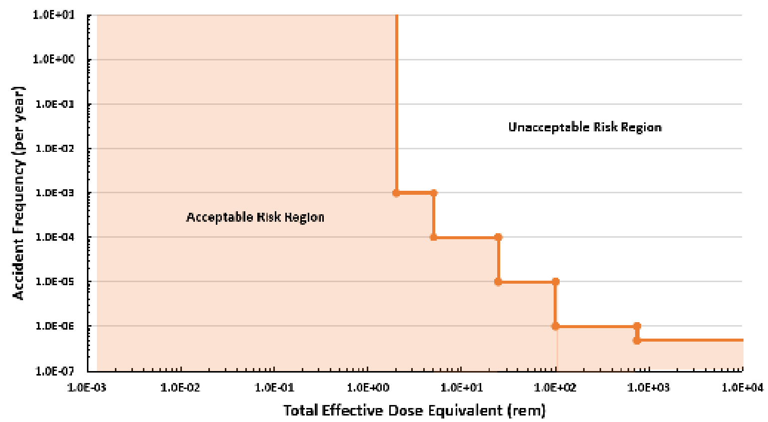
Proposed Risk Evaluation Guidelines

Proposed risk evaluation guidelines compatible with NRC nuclear safety goals, Qualitative Health Objectives, and NRC-proposed QHGs in the NRC 2008 RIDM report

For the Maximum Exposed Member of the Public



For the Worker



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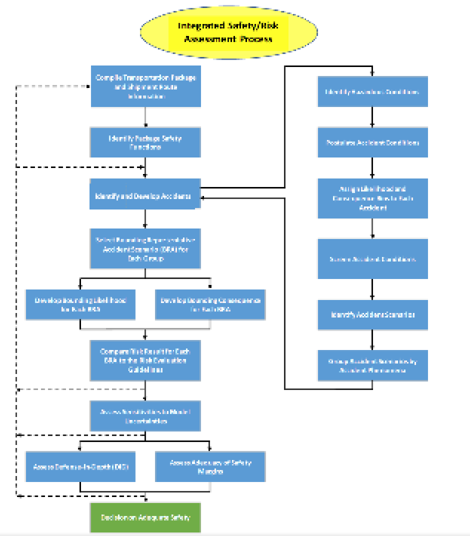
Integrated Risk Assessment Process

- Uses probabilistic risk assessment (PRA) approaches and methods to the level of Bounding Representative Accidents
- A transportation PRA on the Project Pele TNPP was performed as a demonstration of the approach, applying the following steps:
  1. Compilation of the TNPP design and Shipment Route Information – which is information intensive and should be started early
  2. Identification of the Package Safety Functions
  3. Identification and Development of Accidents Scenarios
  4. Definition of Bounding Representative Accidents
  5. Development of Likelihood for Bounding Representative Accidents
  6. Development Bounding Consequences for Bounding Representative Accidents
  7. Comparison of Risk Results to Proposed Risk Evaluation Guidelines
  8. Assessment of Sensitivities and Model Uncertainty
  9. Assessment of Defense-in-Depth and Safety Margin



Probabilistic Risk Assessment Development Process

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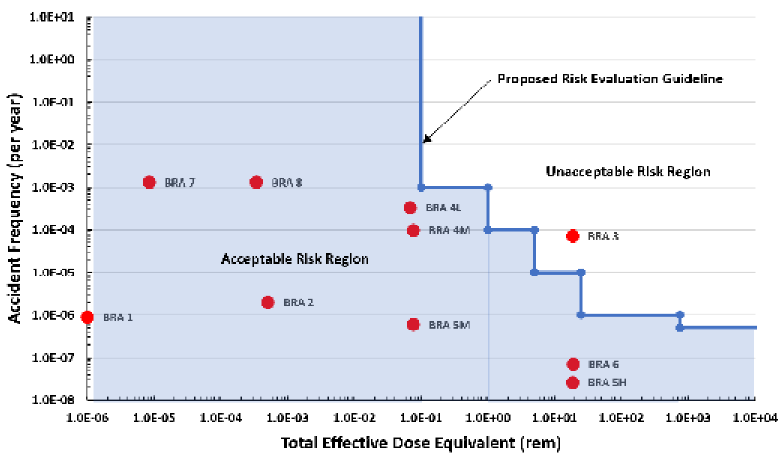
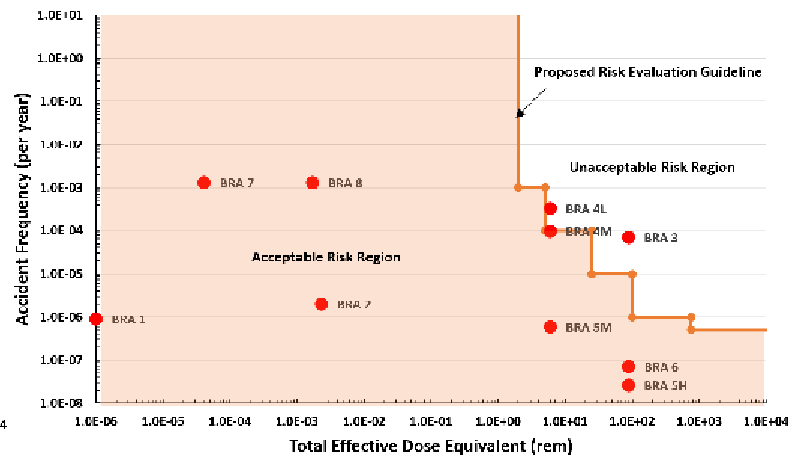
Summary of Demonstration TNPP PRA Risk Results

- Risk for the Bounding Representative Accident Results Shown Graphically

For the Maximum Exposed Member of the Public



For the Worker



Note: BRA 9A and 9B - two kinds of flooded criticality events - are not shown here because their consequences were not calculated given that their likelihoods were determined to be extremely low.
BRA 10 – reactivity insertion caused by crash impact leading to criticality was not developed because it was anticipated the demonstration design will preclude (or design against) this possibility (e.g., using locking mechanisms)

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Follow-up Topics From the Last Meeting

1. Cliff edge effects
   - Criticality is an example because even though it occurs less than a frequency of 5E-07 per year it introduces a new phenomenon and might produce a dose greater than 750 rem dose.
   - Other factors that could impact risk are seen as having just incremental effects.
2. Modeling accident recovery
   - The purpose of recovery is to mitigate occupational exposure
   - Reduction of accident risk using recovery action (if possible) is not credited in the PRA
   - Recovery can involve increased occupational dose for radiation workers. However, this dose is managed by a recovery plan under a radiation protection program and not under the Risk Evaluation Guidelines
3. Risk Evaluation Guidelines for multiple shipments
   - The demonstration was done for a shipment in one year. It could be applied per TNNP for multiple shipments per year by increasing the accident frequencies proportionately.
4. Generic Applicability
   - PNNL believes that the approach has generic applicability and in fact used a similar approach for DOE National Nuclear Security Administration for a package not meeting the codified requirements.
   - However, the approach should be demonstrated for other modes of transport (e.g., barge, rail, maritime) and other types of packages,… because there would no doubt be differences and knowledge transfer.

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Follow-up Topics From the Last Meeting

5. Use of state-level accident rates versus route segment-specific accident rates

- Accident rate data are for trucks > 26,000 lbs. and legal weight trucks have a maximum weight of 80,000 lbs.
  - ✓ Trucks that would carry a Reactor Module would weigh approximately 150,000 lbs. and would require a state-issued permit to be operated on interstates, highways, and roads
  - ✓ These permits are issued by a state to provide permission for an oversize or overweight vehicle and load to travel on a specific route, and potential hazards are mitigated through permit conditions such as requiring route surveys, the time of day and day of week during which travel is allowed, the number or spacing of the vehicle's tires to distribute the weight of the load, the speed that the load can travel, the use of warning signs and lights, the use of escorts, etc.
- Vehicle-mile data are not reported for specific locations
  - ✓ Reported at the functional system level (interstates, other freeways and expressways, etc.) for each state, for rural and urban areas (DOT Table VM-2)
  - ✓ Reported by vehicle type for road type level (interstates, other arterials, other roads), for each state, for urban and rural areas (DOT Table VM-4)
- Existing analyses are not designed to estimate accident risks at specific locations along the routes; rather, they are designed to integrate the risk over the entire route in a probabilistic manner consistent with the Risk Evaluation Guidelines
  - ✓ SNF routing requirements in 49 CFR Part 397, Subpart D are specifically designed to minimize radiological risks, and 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 to determine the level of radiological risk
  - ✓ Base vehicle accident rates are combined with conditional probabilities of specific accidents to yield annual frequencies for entire route
  - ✓ This is the same approach to accident frequency that was used in the Yucca Mountain EIS (DOE/EIS-0250), NUREG/CR-4829, NUREG/CR-6672, and NUREG-2125
  - ✓ Correlations between accident rate, target hardness, presence of rivers and streams, and other potential route hazards could be dealt with by a route survey and appropriate compensatory measures
- That said, an uncertainty analysis or sensitivity study is needed to explore the impact of the variability in the base vehicle accident rate on the risk.

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Applying the Proposed Risk Informed Licensing Methodology to a Draft NRC Safety Evaluation/Safety Analysis Report (SAR) Application

For an applicant to receive transportation package licensing approval, they must develop a complete transportation package safety basis as part of their application that demonstrates reasonable assurance of adequate safety to the public, worker, and environment is provided. This would involve:

- An assessment of all influencing physical, chemical, and environmental loading conditions that would adversely affect package performance when considering all disciplines (structural, thermal, containment, shielding, criticality, operations, and acceptance) to verify maintenance of subcriticality, retention of radionuclide inventory, and adequate shielding and thermal management
- Application of all applicable consensus standards (e.g., ASME Codes and Standards), NRC Transportation (Division 7) Regulatory Guides (e.g., Regulatory Guide 7.1 - 7.13), NRC Standard Review Plans (e.g., NUREG-2216), etc., and using Regulatory Guide 7.9 as standard format and content guidance of Part 71 applications to:
  - Deterministically demonstrate TNPP package compliance with dose rate and containment success criteria after Normal Conditions of Transport (NCT) as outlined in 10 CFR 71.71
  - Deterministically demonstrate TNPP package compliance with dose rate and containment success criteria after Hypothetical Accident Conditions (HAC) tests as outlined in 10 CFR 71.73 or fully exploit the design to determine the level of robustness and capacity to meet these requirements
  - Develop legitimate compensatory measures while employing quantitative risk assessment using an integrated assessment process based on PRA methods which includes use of sensitivities and uncertainty analysis and consideration of DID and SM to reestablish equivalent safety only for those challenges identified through a rigorous screening of HAC related assessments
  - Request that NRC consider an exemption following the process outlined in 10 CFR 71.12 and leverage the substantiating information from the previous step to protect the basis of exemption and demonstrate acceptable risk

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Applying the Proposed Risk Informed Licensing Methodology to a Draft NRC Safety Evaluation/Safety Analysis Report (SAR) Application

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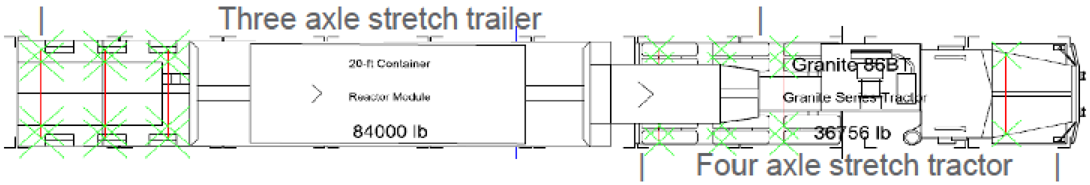
How would a shipment of a TNPP be conducted?



A Reactor Module shipment would be conducted in a manner similar to current SNF shipments. This requires that:

1. The shipment is made in accordance with U.S. Department of Transportation Hazardous Materials Regulations (49 CFR Parts 171-180)
2. A state permit has been obtained for the oversize/overweight vehicle (about 150,000 lb.) that carries the reactor module
   - This includes an evaluation of the proposed route and any alternative routes to verify that the transportation infrastructure can accommodate the truck and its load
   - Alternative routes included in case of weather, road closures, etc.
3. The proposed transportation route and any alternative routes must meet the requirements of 49 CFR Part 397, Subpart D
   - These routes are chosen to minimize radiological risks
   - Interstates, and interstate bypasses or beltways around a city
   - States may also designate preferred routes as alternatives or in addition to interstates, and interstate bypasses or beltways
4. The vehicle and its load have received a Commercial Vehicle Safety Alliance (CVSA) Level VI inspection
   - This requires the vehicle and the load to be defect-free prior to departure.

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How would a shipment of a TNPP be conducted?

Also required are that:

5. The proposed transportation route and any alternative routes have been evaluated from a security perspective
   - NUREG-0561, Physical Protection of Shipments of Irradiated Reactor Fuel, or the Army equivalent
   - Includes coordination with local law enforcement agencies and identification of safe parking areas in case the shipment is delayed en route due to mechanical problems, bad weather or hazardous road conditions or other unanticipated problems
6. All identified and specified compensatory measures are in place
   - Time of day and day of week restrictions, rolling road closures, escorts fore and aft, etc.
7. Conducted only after the proposed shipment is coordinated with all affected States and Tribes as part of planning and communication
   - Advance notification of the States and Tribes along the route, shipment tracking, and shipment status
   - Emergency response plans and procedures in place
8. Ensuring that the shipment avoids bad weather and hazardous roads through constant communication with drivers and by carefully monitoring road and weather conditions and restricting travel when adverse conditions pose a threat to shipment safety
   - Any delays (traffic, weather, mechanical issues, etc.) are coordinated in real time with downstream States and Tribes.

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Pacific Northwest National Laboratory

902 Battelle Boulevard
P.O. Box 999
Richland, WA 99354
1-888-375-PNNL (7665)

www.pnnl.gov