Parametric Study of Factors that Affect Calculated Dose from TRISO Fueled Microreactor Transportation Accident

November 2022

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Pacific Northwest
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U.S. DEPARTMENT OF ENERGY

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Summary

Microreactors are very small nuclear reactors with a power output up to 50 megawatts electric (MWe) which also meet the standards of an “advanced nuclear reactor.” Microreactors are designed to be factory-built, modular in nature, and may be transportable. These compact small-sized reactors in some design configurations are small enough to be transported by truck which could help solve certain energy challenges. This report refers to these reactors as transportable microreactors or transportable nuclear power plants (TNPPs). Transportable microreactors are being considered for a wide range of independent operation applications.

However, TNPP transportation packages have never been approved by the U.S. Nuclear Regulatory Commission (NRC), which could especially be a challenge if the TNPP contains irradiated fuel. In a previous study documented in PNNL-31867 (PNNL 2021), it was concluded 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 transportation package is unlikely to meet NRC requirements in Title 10 of the Code of Federal Requirements (10 CFR) Part 71 for a Type B package. Based on this conclusion, a risk-informed regulatory framework was proposed in PNNL-31867 for demonstrating that a TNPP transportation package and associated shipment process and controls provides equivalent safety to that of a Type B package. In PNNL-31867, it was observed that such an undertaking would require a realistic assessment of the release of radionuclides to the environment from a severe transportation accident involving a TNPP serving as a transportation package, which requires knowledge of: (i) postulated release of fission products and actinides from the irradiated fuel into the reactor cooling system and other reactor systems as a result of normal reactor operations and (ii) postulated releases from the irradiated fuel and the released fission products and actinides from reactor systems as a result of hypothetical transportation accidents. The report noted the lack of such information and the need to characterize the uncertainty associated these releases when assessing the risk from TNPP transportation accidents using Probabilistic Risk Assessment.

This study investigates this uncertainty in three parts: (i) conducting a literature search on the current understanding and postulates related to releases from TRi-structural ISOtropic (TRISO) fuel during hypothetical transportation accident conditions, (ii) conducting a parametric study of the potential radiological exposure consequences that might result from hypothetical transportation accidents involving a micro high-temperature gas-cooled (HTGR) utilizing TRISO fuel, and (iii) evaluating the radiological consequence results of the parametric study for insights.

As a follow-on to PNNL-31867, this report provides the results of a literature review that investigated the availability of release fractions for TNPPs utilizing HTGR technology fueled by TRISO fuel. In addition, recognizing the potential sensitivity of transportation accident consequences to release fraction assumptions used in a risk-informed assessment, this report also presents the results of a parametric study that addresses the modelling uncertainty with these assumptions. To achieve this, a consequence model was developed to calculate the radiological dose consequences to a nearby member of the public from hypothetical accidents involving transportation of a TRISO fueled representative 20-megawatts thermal (MWt) microreactor. Key factors in this assessment included evaluation of fractions of material released from the TRISO fuel during normal operations that reside in the compacts, core structure, and cooling system (i.e., release fraction) and the fraction of material that is subsequently released during a transportation accident based on accident conditions.

The focus of the literature review for releases during normal reactor operations was on the results of two fuel irradiation tests, designated AGR-1 and AGR-2, conducted under the aegis of the U.S. Department of Energy's (DOE's) Advanced Gas Reactor Fuel Development and Qualification (AGR) Program. The purpose of this program was to demonstrate oxycarbide TRISO fuel performance during irradiation and in post-irradiation high-temperature accident safety tests. The reason for focusing the review on just these irradiation tests is that the NRC has reviewed this test data and concluded in its safety evaluation report that this test data can be used to support safety analyses referencing the unique design features of the TRISO fuel particle, subject to certain limitations and conditions.

The failure metrics of TRISO fuel particles in the AGR-1 and AGR-2 tests were developed for normal operations and for certain accident conditions. For normal operations, fuel failure was reported for three different metrics: (i) TRISO fuel particle failure fraction based on the number of partially-failed TRISO particles and on the number of full TRISO particle failures, (ii) release-rate-to-birth-rate ratio for two fission product gases released from the TRISO particles, and (iii) fraction of the total inventory of five non-gaseous fission products released from the TRISO particles. The failure rates were reported either as a mean and a 95 percent confidence level (the first metric) or as a range from which minimum and maximum values could be obtained (the second and third metrics). These results are summarized in this report. However, this study only addresses transportation of a TNPP that has not experienced an accident, and so only test results for normal reactor operations are discussed in detail here.

For the parametric study, release and attenuation parameters were developed for about 50 radionuclides (fission products and actinides) using the methodology documented in INL/EXT-11-24034, Revision 1, "Scoping Analysis of Source Term and Functional Containment Attenuation Factors." A comparison of the release and attenuation parameters used in this methodology with the data reported for the AGR-1 and AGR-2 tests show that, generally, the parametric study would result in values that were more conservative than those reported in the AGR tests or were bounded in the sensitivity studies.

The literature search did not identify any testing performed on irradiated TRISO fuel under transportation conditions. Therefore, for the parametric study, estimates of release parameters for the TNPP package were developed based on engineering-based assumptions and judgement. To ensure that the results of the parametric study are bounding of expected actual TNPP package performance under accident conditions, these estimates were developed with the intent to be conservative and sensitivity studies were performed to try to bound the radiological exposure results. Nevertheless, the release parameters (i.e., source term factors) under transportation accident conditions are a significant source of uncertainty.

No federal risk-informed radiological dose evaluation limits exist for transporting microreactors, therefore a 25 roentgen-equivalent-man (rem) dose evaluation limit for the parametric study was selected based on federal siting guidelines for reactors and performance requirements that must be met when more than the minimum critical mass of fissile nuclear material is involved in certain facilities such as nuclear fuel cycle facilities, and DOE nuclear safety guidance for nonreactor nuclear facilities. Based on the baseline radiological dose consequences analyses performed for selected transportation accidents, assuming the TNPP has been shutdown and cooling for five years, the dose to an individual human receptor located close to the accident appears to be acceptability low (i.e., less than 25-rem) in-spite of conservatisms in the analysis. One of the accidents selected for this study, a fuel tanker collision with the TNPP and subsequent fire, is considered to be bounding because it is judged to result in the greatest potential radiological dose consequences of reasonable hypothetical accidents. This judgment

is based on hazard analysis principles and previous transportation risk assessment studies. An example 20 MWt microreactor core inventory was chosen because it is judged to represent the upper end of the power range for microreactors that might be transported by trucks. Therefore, the radiological dose consequences from a transportation accident involving a lower power reactor would be expected to be less.

However, the sensitivity studies show that when the key consequence analyses are set to be more conservative, the radiological dose consequences from the fuel tanker collision and subsequent fire event can well exceed the 25-rem dose evaluation limit. This is particularly true for cases in which aggregate source term factors were increased by an order of magnitude. For the sensitivity case in which the TNPP is transported just one year after the shutdown, the dose to the human receptor is about 35-rem, which marginally exceeds the 25-rem limit. On the other hand, a sensitivity case in which the release fraction from normal operations is increased from the mean of an estimated probability distribution to the 95th percentile shows that the dose consequences from the fuel tanker collision and fire accident would remain below the evaluation limit.

The results of this parametric study appear to indicate that the radiological dose consequences from a worse-case TNPP transportation accident could be acceptably low based on using best judgment assumptions. However, given the lack of testing and analysis of the performance of microreactors as transportation packages, it is not clear these results are generalizable. This modelling uncertainty would need to be addressed in order to perform a TNPP transportation Probabilistic Risk Assessment. Of particular concern is the assumed damage that occurs as a result of a transportation accident which can in turn have a significant effect on the estimated source term factors. This is of particular concern for the source term factors associated with the TRISO fuel itself as the radiological dose consequences from the TRISO fuel (opposed to material diffused into the reactor core internals or plated out in the primary system) dominate the dose consequence results.

In all sensitivity cases, removal of the TRISO fuel (compact stacks) prior to transportation of the microreactor reduces the potential dose consequence well below 25-rem, including in the most extreme cases. So, transporting the TRISO fuel separately in a standard certified spent fuel package configuration dramatically reduces dose since diffused fission and activation products in the reactor core components and within the primary circuit pressure boundary have much smaller contributors to dose consequences.

Additional post-irradiation examination of the mechanical properties of TRISO fuel particles, compacts, and graphite core components under impact loads could provide data that may reduce conservatism in postulated release fractions by informing transportation package design and performance in bounding accident circumstances.

Acknowledgments

The work performed in support of this report was funded by the National Reactor Innovation Center, a National Department of Energy program led by Idaho National Laboratory for the Office of Nuclear Energy Research and Development which supports demonstration of microreactor technology.

The authors also gratefully acknowledge X-energy's commitment to develop and provide radionuclides inventories at different reactor shutdown cooling times for a representative high temperature gas microreactor for use in the parametric study of factors that affect calculated dose from a TRi-structural ISOtropic fueled microreactor transportation accident.

Acronyms and Abbreviations

°C	degree(s) Celsius
AGR	Advanced Gas Reactor Fuel Development and Qualification
ARF	airborne release fraction
BISO	bistructural isotropic particle fuel
CFR	*Code of Federal Regulations*
Ci	curie(s)
CoC	Certificate of Compliance
DBE	design basis events
DoD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DR	damage ratio
EFPD	effective full power days
EIS	environmental impact statement
FIMA	fissions per initial metal atom
FP	fission product
GPa	gigapascal(s)
HALEU	high assay low enriched uranium
HTGR	high-temperature gas-cooled
IAEA	International Atomic Energy Agency
in	inch(s)
INL	Idaho National Laboratory
kg	kilogram(s)
kW	kilowatts
kWe	kilowatt(s)-electric
kWth	kilowatt(s)-thermal
LPF	leak path factor
MAR	material at risk
µm	micrometer(s)
m	meter(s)
MeV	Mega electron-volt
mg/cm^2	milligram per square centimeter
MPa	megapascal (s)
mph	mile(s) per hour
mSv	millisievert(s)
MWd/MTHM	megawatt-day(s) per metric ton of heavy metal

MW(e)	megawatt(s) electric
MW(t)	megawatt(s) thermal
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NRIC	National Reactor Innovation Center
PIE	post irradiation examination
PNNL	Pacific Northwest National Laboratory
ppm	parts(s) per million
psi	pound-force per square inch
psig	pounds per square inch gauge
PWR	pressurized water reactor
PyC	specific pyrolytic carbon
R/B	release-rate-to-birth-rate ratio
rem	roentgen equivalent man
RF	respirable fraction
RG	regulatory guide
SCO	DoD Strategic Capabilities Office
SiC	silicon carbide
Sv	sievert
Sv/Bq	sievert per becquerel
$Sv\ s^{-1}\ TBq^{-1}\ m^2$	sievert per terabecquerel second square meter
TED	total effective dose
TEDE	total effective dose equivalent
TRISO	Tri-structural Isotropic particle fuel
TNPP	Transportable Nuclear Power Plant
UCL	upper confidence limit
UCO	oxycarbide

Contents

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1.0 Introduction

Microreactors are very small nuclear reactors with a power output up to 50 megawatts electric (MWe) which also meet the standards of an "advanced nuclear reactor."[1] Microreactors are designed to be factory-built, modular in nature, and may be transportable. These compact small-sized reactors in some design configurations are small enough to be transported by truck which could help solve certain energy challenges. This report refers to these reactors as transportable microreactors or transportable nuclear power plants (TNPPs). These reactors are suitable for a wide range of independent operation applications, including power for remote locations, mobile backup power, mining operations, military installations, space missions, desalination, and emergency power supplies in support of disaster relief operations.

The U.S. Department of Energy (DOE) supported National Reactor Innovation Center (NRIC) of the Idaho National Laboratory (INL) is supporting the U.S. advanced reactor industry through funding, legislation, and regulatory development to accelerate the demonstration and deployment of advanced nuclear energy, including TNPPs. Remote rural communities in the U.S. that rely on diesel generators for electricity are also considering microreactors as a source of reliable, zero-carbon energy capable of operation for several years without refueling. The U.S. Department of Defense (DoD) also shares great interest in microreactor design concepts as its military operations become more energy-intensive and require portable, dense power sources. For example, the DoD Strategic Capabilities Office (SCO) is currently funding Project Pele, with technical support from NRIC, to design, construct, and demonstrate a prototype TNPP.

In fiscal year 2021, Pacific Northwest National Laboratory (PNNL) with funding from the NRIC, evaluated regulatory options for transportation of microreactors and developed a proposed regulatory framework for transporting microreactors within the U.S. This regulatory framework, documented in PNNL-31867 (PNNL 2021), assumes a previously operated microreactor, containing irradiated fuel would be transported in accordance with U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Transportation (DOT) regulations for ground transportation of radioactive material. (This report is hereafter referred to as the "framework report") The authors concluded in this report that, because of the expected radioactive inventory in the irradiated fuel, an NRC-approved Type B package (or spent nuclear fuel cask) would be required for shipment of the microreactor but that a microreactor transportation package is unlikely to meet NRC requirements in Title 10 of the *Code of Federal Requirements* (10 CFR) Part 71 for a Type B package. Based on this conclusion, a risk-informed regulatory framework was proposed for demonstrating that the microreactor transportation package and associated shipment process and controls provides equivalent safety to that of a Type B package.

To support implementation of the proposed regulatory framework, PNNL-31867 observed that a realistic assessment of the release of radionuclides to the environment from a severe transportation accident involving a microreactor transportation package requires knowledge about (i) release of fission products and actinides from the irradiated fuel into the reactor cooling system and other reactor systems as a result of normal reactor operations and (ii) releases from the irradiated fuel and the reactor systems as a result of hypothetical transportation accidents. During fiscal year 2022 PNNL was funded by NRIC to investigate the

[1] Public Law No: 117-58, Infrastructure Investment and Jobs Act, Subtitle C—Nuclear Energy Infrastructure.

availability of release fractions for TNPPs utilizing high-temperature gas-cooled (HTGR) technology fueled by TRi-structural ISOtropic (TRISO) particle fuel. This report documents the results of this investigation.

1.1 Purpose and Scope

The purpose of the research conducted for this report is to address a technical modeling uncertainty that was identified in the framework report (PNNL 2021) to have significant impact on the determination of risk for the transportation of a microreactor package. Specifically, the amount of radioactive material that could be released in a severe transportation accident is a critical parameter in understanding the significance of the potential radiological exposure of transportation workers and members of the public due to the accident. This study investigates this uncertainty in three parts: (i) conducting a literature search on the current understanding and postulates related to releases from TRISO fuel during hypothetical transportation accident conditions, (ii) conducting a parametric study of the potential radiological exposure consequences that might result from hypothetical transportation accidents involving a micro HTGR TRISO fuel, and (iii) evaluating the radiological consequence results of the parametric study for insights. This modelling uncertainty would need to be addressed to perform a TNPP transportation probabilistic risk assessment. Based on the results of this research, recommendations are proposed for future modeling and testing of TRISO fuel particles and compacts to fill data gaps and reduce the uncertainty in the estimated release of radioactive materials under transportation accident conditions.

The amount of radiological material potentially released because of a severe transportation accident is predominantly influenced by three release mechanisms: (i) material released from the TRISO fuel particles during reactor operations and are retained within the reactor core structures and/or portions of the reactor coolant pressure boundary system during transportation of the microreactor package, (ii) material released from reactor core structures and the reactor coolant pressure boundary due to damage to the transportation package confinement system caused by the transportation accident, and (iii) additional material released due to failure of TRISO fuel particles that are damaged during the transportation accident.

A primary concern in a TRISO fueled reactor accident as it pertains to the accident source term is that fuel could be heated to the point where it fails or releases fission products via defects. However, a concern in a transportation accident is the possibility of the mechanical rupture of the TRISO fuel due to energy involved in an event, such as a vehicle collision or other significant impact and the release from the compact around the fuel, core components, or the coolant boundary of the radioactive material released or activation products produced during reactor operations.

Since most of the diffusive species (other than gases), tend to get held up in the core graphitic components, any process that could generate fines could potentially contain radioactive material. Vibration and impact could also generate and release graphitic fines into the reactor coolant pressure boundary that contain diffused radioactive material. These graphite fines and other radioactive material may be released into the environment because of failures during transportation.

This report summarizes the results of the literature search on each of these release mechanisms. This report also presents the results of a parametric study that addresses the modelling uncertainty for factors associated with release of radiological material for use in

calculating the radiological dose consequences of an accident involving transportation of a TRISO fueled microreactor. The factors include fractions of material released from the TRISO fuel during normal operations that reside in the core, reactor structure, and cooling system (i.e., release fraction) and the fraction of material that is released during a transportation accident based on accident conditions.

Using sensitivity cases, the study addresses the impact that the uncertainty of the factors described above and other key assumptions for transportation specific accidents have on calculated radiological dose consequences. The purpose of the study is to: (i) gain generalizable information about the values to use for release-related factors that could be used in a dose consequence analysis for a transportation accident involving a TRISO fueled microreactor, (ii) provide insights about how the uncertainty associated with the release-related values used in a dose consequence analyses impacts the estimated radiological dose, and (iii) generate insights about future analyses and tests that could be performed to address the modelling uncertainty having the greatest impact on estimating the radiological dose consequences.

The parametric study was performed for selected road transportation accident scenarios involving a representative 20-megawatts thermal (MWt) microreactor that has operated at full power for three years and allowed to cool for a period of one to five years prior to being shipped. The radiological dose to an individual human receptor located close to the accident was calculated using the approach discussed in Section 5.3.4.

The case of a TNPP containing an integral sealed core and with reloadable fuel that is removed prior to transportation of the microreactor were considered since commercial applications may benefit economically from a “site” reloadable core. In the case of the reloadable core, the radioactive material potentially available for release during a transportation accident only considered that which was previously released from the TRISO fuel particles during the reactor operations and were retained within the reactor core structures and/or reactor coolant pressure boundary system. The spent TRISO fuel in this case would be transported using packages approved under the codified regulations. The radiological inventories associated with a defueled reactor can be deduced from the existing parametric analysis by removing the contribution from the TRISO fuel.

1.2 Report Content and Organization

The ensuing sections of this report are organized as follows:

Section 2.0 summarizes the results of the literature review of testing and analyses performed to identify the mechanisms for release of radionuclides from TRISO fuel and to characterize the relative significance of each of these release mechanisms. Specifically addressed is: (i) release of radionuclides during normal reactor operations and anticipated operational occurrences that are retained in core structures and the reactor coolant pressure boundary and (ii) releases of radionuclides from the reactor system due to a transportation accident. Releases from the TRISO which occur as the result of an operating reactor accident are not included in this study.

Section 3.0 summarizes historical experience with transportation of TRISO fuel.

Section 4.0 summarizes the radionuclide release fractions and parameters assumed in the dose consequence analysis and parametric analysis. The factors include fractions of material

released from the TRISO fuel during normal operations that reside in the core, reactor structure, and cooling system (i.e., release fraction) and the fraction of material that is released during a transportation accident based on accident conditions.

Section 5.0 provides a description of the dose consequence analysis approach used to perform the parametric study. This section also provides a description and results of the parametric study addressing the modelling uncertainty for factors associated with release of radiological material for use in calculating the radiological dose consequences of an accident involving transportation of a TRISO fueled microreactor.

Section 6.0 provides a summary of the report, including the key conclusions drawn from the results of the parametric study and key recommendations for addressing data gaps to increase confidence in the dose consequence analysis and results.

Section 7.0 provides a listing of the references.

Appendix A provides the initial reactor radionuclide inventory used in the dose consequence analysis.

Appendix B summarizes the model used to develop the release fractions for normal reactor operations and anticipated operational occurrences.

2.0 TRISO Fuel Release Fraction Literature Review

This study assumes that the TNPP reactor core consists of cylindrical TRISO fuel compact stacks in channels adjacent to, but separate from parallel coolant channels, encased within graphite moderator blocks. Control rods are located in certain interspersed parallel coolant channels and control drums may be located at the radial periphery of the core. The fuel compacts are composed of high assay low enriched uranium (HALEU) oxycarbide (UCO) TRISO particles contained within a graphite matrix. A cross-sectional view of a single spherical TRISO fuel particle is shown in Figure 2-1. An example of fuel compacts is shown in Figure 2-2.

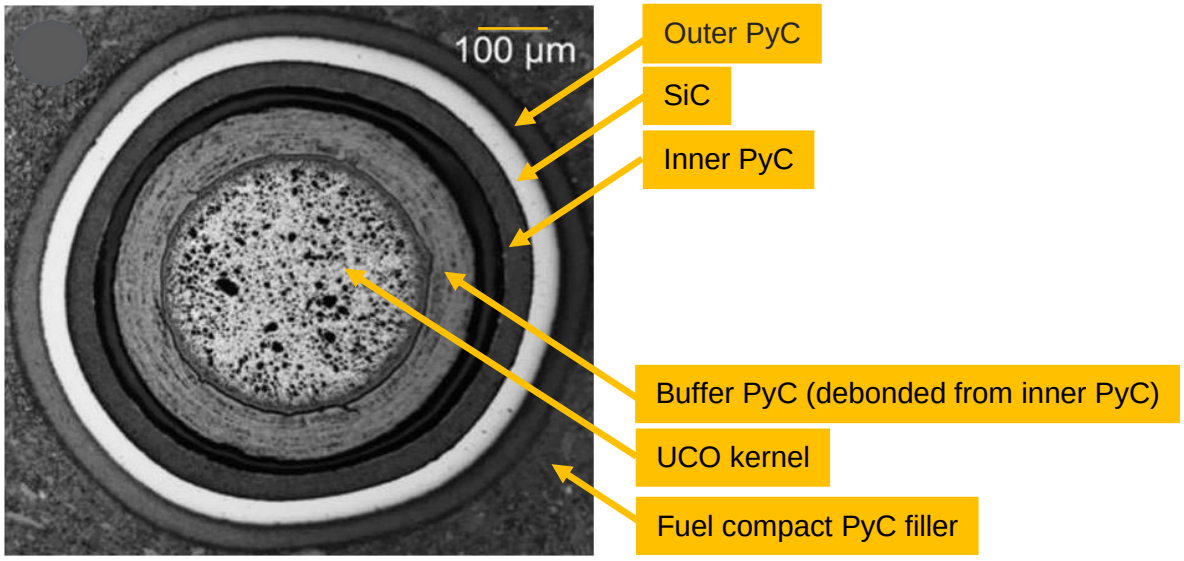

Figure 2-1. Cross Section of Irradiated Oxycarbide TRISO Fuel Particle (Electric Power Research Institute 2020).

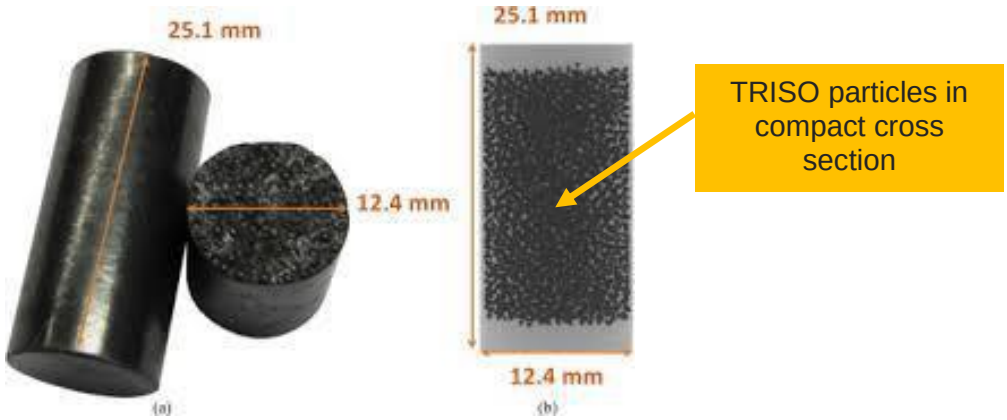

Figure 2-2. Typical Oxycarbide TRISO Fuel Particle Compact Containing Fuel Particles (Harp 2014).

It is assumed that the TRISO fuel and compact for this study is designed to ensure that its characteristics are bounded by DOE's advanced gas-cooled reactor (AGR-1 and AGR-2) fuel qualification program for TRISO fuel. Typically, the overall design of an HTGR ensures that the operating and passive decay heat cooling scenarios do not allow the fuel to exceed its expected

maximum allowable temperatures, which ensures the TRISO fuel will maintain its structural integrity necessary for retention of fission products

The UCO TRISO fuel particles and fuel compacts release fractions of certain fission products during normal operations, anticipated operational occurrences, and design basis events (DBE) of TRISO fueled reactors. Isotopes of silver and a few other metal species are predominantly released by diffusion that is sufficiently high within the temperature range of the fuel compacts under normal operation conditions. This involves the diffusion of fission product species through the specific pyrolytic carbon (PyC) coating layers (porous PyC buffer layer, 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 than silver.

Various mechanisms for the diffusion are involved and strongly dependent on the specific microstructure of the UCO fuel kernel (a blend of HALEU UO_2 and UC_2) and of the four coating layers. These complex diffusion mechanisms are most often modeled for the large ensemble of TRISO particles as simple Fickian diffusion with strongly temperature dependent effective diffusion coefficients and the assumption of negligible counter diffusion effects (a reasonable assumption at low release rates into a medium very far from saturation by competing diffusive species). The most common model for the coefficients varies exponentially with temperature. These models are used because different fission product species diffuse at widely different rates, and most do not readily diffuse within the normal operation and anticipated operational occurrences envelop by design. In fact, many do not significantly diffuse even under DBE conditions.

In addition to diffusion at normal operating temperature, which occurs even if all TRISO particles have perfect quality, a certain number of TRISO particles may have *apriori* manufacturing defects such as an incomplete or missing SiC layer or a major defect that exposes the fuel kernel. In manufacturing following best practices, the fraction of TRISO particles with *apriori* defects are extremely limited (~0.0001 – as small as can be practically verified by an implemented quality control inspection process, see Barnes 2006). However, direct detection of such particles becomes increasingly difficult as the fraction of defects becomes very small. These defects can also be inferred later by detection of certain fission product isotopes during irradiation that cannot be otherwise explained as occurring via diffusion or other mechanisms.

A certain amount of heavy metal contamination may also occur in the TRISO particle manufacturing process. Efforts are made to limit such contamination in manufacturing processes, and such contaminations can be very low (~20 parts per million [ppm], see EPRI 2020). In the case of UCO TRISO, some of this tramp uranium will show up in the outer PyC layer of the TRISO particles and possibly even in the compact PyC material as well. As such, in some cases, it is impossible to distinguish fission product isotopes that arise from this contamination and from potential *apriori* manufacturing coating defects unless the degree of contamination is accurately and independently monitored by trace analysis of residues from the manufacturing processes (e.g., equipment holdup, etc.). Qualified TRISO fuel fabrication processes have quality control specifications on allowable threshold of contamination and *apriori* defects as mentioned above in addition to other manufacturing flaws.

Finally, at higher burnups, certain TRISO particles will have in-service failures that may occur due to unknown reasons since post irradiation examination (PIE) cannot be performed on such an extraordinary number of TRISO particles. Such mechanisms have been observed by PIE

and are being studied with a view to minimizing (or predicting) them further. These in-service failure mechanisms can sometimes be observed in instrumented capsules as a stochastic and unexpected small release event indicating a single particle SiC layer failure occurring at a temperature and burnup that is too low to explain via other mechanisms such as SiC decomposition or SiC layer over pressurization, or other microscopic flaw.

2.1 Non-Accident Release Test Data

This section reviews the results of two fuel irradiation tests, designated AGR-1 and AGR-2, conducted under the aegis of the DOE's Advanced Gas Reactor Fuel Development and Qualification (AGR) Program. The purpose of this program was to demonstrate UCO TRISO fuel performance during irradiation and in post-irradiation high-temperature accident safety tests. The results of this testing program are documented in Topical Report EPRI-AR-1(NP)-A (EPRI 2020). The NRC reviewed this report and in its safety evaluation report concluded that the data in this EPRI 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). The research results presented in the EPRI report bound the fission product release fractions and fuel failure fractions used in the parametric risk analysis presented later in this report to estimate releases during reactor operations. These released fission products could be retained within the TNPP reactor coolant pressure boundary components included as a part of the transport and therefore available for release in the event of a transportation accident.

2.1.1 Advanced Gas-cooled Reactor-1 and 2 Test Configuration for Oxycarbide TRISO Particle Fuel Compacts

To orient the reader, Figure 2-3 represents the design of an AGR series test capsule. These capsules were stacked in a single housing (a string) and lowered into irradiation positions in the Advanced Test Reactor at the INL site. Each capsule had dedicated sweep gas lines to continuously sample noble gas fission products released and thermocouples positioned as indicated.

The gray material is PyC as a surrogate for structures such as the outer layer of a moderating pebble, a PyC prismatic block moderator, a PyC fuel compact sleeve, etc. The green material is a stack of TRISO bearing compacts in the shape of a right circular cylinder. Capsule number refers to one of these gadgets. The compacts within the stacks within each capsule are also individually numbered.

The embedded thermocouples provided an experimental value on temperature, in the moderator near to the compact stack at a few different levels, to compare with a detailed finite element modelling result. The uncertainties between measured and calculated values were roughly ±5 percent.

The thermocouples occasionally failed at the high temperatures and high radiation doses associated with these long running tests. There were also occasional failures associated with the gas line and analysis system. Data that could not be qualified were mentioned, but not presented, in the EPRI qualification report (EPRI 2020). More details appear in the specific AGR programmatic data reports (Collin 2014 and Collin 2015).

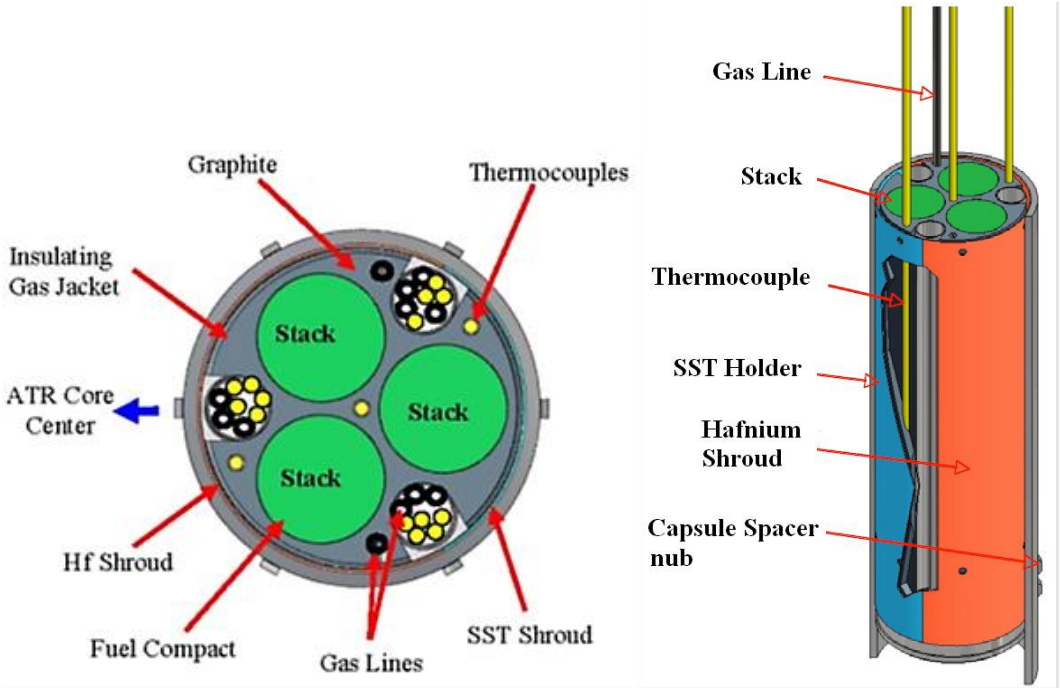

Figure 2-3. Single Advanced Gas-cooled Reactor-1 Test Capsule (Collin 2015).

2.1.2 Advanced Gas-Cooled Reactor-1 and 2 Data on Oxycarbide TRISO Silicon Carbide Layer and Whole Particle Failures

The TRISO fuel failure fraction based on the results of the AGR-1 and AGR-2 fuel irradiation testing under normal reactor operating conditions are summarized in Table 2-1. The table lists the total number of compacts and particles included in each test, the number of observed failures of each failure type based on the best estimate values from irradiation data, the apparent failure fraction, and the 95 percent upper confidence limit (UCL) on the failure fraction. Data for AGR-1 and AGR-2 are provided both separately and in combination.

Table 2-1. Apparent (inferred) Post Irradiation Failure Data for Advanced Gas-Cooled Reactor-1 and Advanced Gas-Cooled Reactor-2.

			Silicon carbide Layer Failures (Missing Layer)			Full TRISO Particle Failures (Exposed Kernel)		
Test String	Number of Compacts	Number of Particles	Number of Particle Failures	Failure Fraction	95% Upper Confidence Limit Failure Fraction	Number of Particle Failures	Failure Fraction	95% Upper Confidence Limit Failure Fraction
AGR-1*	72	298,000	4	1.3×10^{-5}	$<3.1 \times 10^{-5}$	0	0	$<1.1 \times 10^{-5}$
AGR-2**	36	114,336	4	3.5×10^{-5}	$<8.1 \times 10^{-5}$	<4	$<3.5 \times 10^{-5}$	$<8.1 \times 10^{-5}$
AGR-1 + AGR-2	108	412,336	8	1.9×10^{-5}	$<3.6 \times 10^{-5}$	<4	$<9.7 \times 10^{-6}$	$<2.3 \times 10^{-5}$

AGR-1 = advanced gas-cooled reactor-1, AGR-2 = advanced gas-cooled reactor-2, TRISO = Tri-structural Isotropic particle fuel.
* Time-average maximum fuel particle temperature less than 1200 degrees Celsius (°C), time-averaged volume average fuel particle temperature less than 1090 °C, and peak compact burnup less than 19.6% fissions per initial metal atom (FIMA).
**Time-average maximum fuel particle temperature less than 1360 °C, time-averaged volume average fuel particle temperature less than 1250 °C, and peak compact burnup less than 13.2% FIMA.
Adapted from EPRI 2020.

There was increased trace uranium contamination of the PyC materials used in some of the AGR-2 test string. This contamination resulted in additional production of fission products outside the TRISO SiC barriers. The somewhat lower performance (three to eight-fold) on apparent failure rates is partly the result of this contamination and not attributed to actual particle failures. Reported SiC layer failures and exposed TRISO kernels are inferred by comparison of certain fission product releases into and beyond the compact in the test string. These reported failures of TRISO particles are not the result of detailed PIE analyses of many tens of thousands of individual particles. The additional contamination level was within the allowed limits of the current required specifications for manufacturing commercial UCO TRISO particles[2,3] (Barnes 2006, 2009) and compacts. The releases associated with particle failures is on the same order as the increase in average trace level uranium contamination of PyC in some of the AGR-2 TRISO compacts.

[2] INL/MIS-05-00238-Revision-1, AGR-1 Fuel Product Specification and Characterization Guidance.
[3] SPC-923, "AGR-2 Fuel Specification," Rev.3, January 2009.

2.1.3 Representative Noble Gas Fission Product Releases

Table 2-2 summarizes the fission gas release results for the AGR-1 and AGR-2 tests. The table provides the minimum and maximum release-rate-to-birth-rate ratio (R/B)[4] results for Krypton-85m (Kr-85m) and Xenon-138 (Xe-138). The R/B ratio for Krypton-88 (Kr-88) was comparable to, though slightly less than, the R/B ratio for Kr-85m. These results show that release of noble gas fission products is extremely low under conditions of normal reactor operations. Fission gas releases are constrained by three TRISO layers and failures are prerequisite to any significant release of noble gas from the TRISO particles. Fission product release is greater for AGR-2 than for AGR-1. A significant contributor to fission product release in AGR-2 was from additional uranium contamination of PyC materials in the fuel form matrix located outside the SiC layers as opposed to gas released through actual TRISO failures. This uranium contamination was within the specifications for fabrication of TRISO fuel, though in some lots of particles, contamination was roughly 10 times higher, based on destructive analysis.

Table 2-2. Maximum and Minimum Release-Rate-To-Birth-Rate Ratios For Noble Gases.

	Minimum And Maximum Release-Rate-To-Birth-Rate Ratio (R/B Ratio)			
	Krypton-85m		Xenon-138	
Test String	Minimum	Maximum	Minimum	Maximum
AGR-1	2×10^{-9}	$2 \times 10^{-7*}$	6×10^{-10}	$2 \times 10^{-8*}$
AGR-2	3×10^{-8}	$2 \times 10^{-6**}$	1×10^{-8}	$3 \times 10^{-7**}$

AGR-1 = advanced gas-cooled reactor-1, AGR-2 = advanced gas-cooled reactor-2.
* These two max figures occurred in AGR-1 Capsule 5, during a run with the highest irradiation temperature.
** These two max figures occurred in AGR-2 Capsule 2, during a run with the highest irradiation temperature.
Adapted from EPRI 2020.

In AGR-1, the Kr-85m R/B ratio ranged from 2×10^{-9} to 2×10^{-7} for irradiation times ranging from zero up to 600 effective full power days (EFPD). Noble gas fission products diffuse from TRISO particles out to the sweep gas locations at the periphery of the capsules and Kr-88 may be slightly less mobile at a given temperature than the lighter isotope because of additional atomic mass. Isotopic diffusion differences drop quickly as atomic mass rises. For gas into solid diffusion (and gas into pore diffusion), the physics is complex and depends specifically on the diffusing species and the details of the solid material matrix and its microscopic defects.

Noble gases from irradiated uranium contamination of PyC materials outside the TRISO particles (that is, outside the TRISO SiC layers – indistinguishable in PIE from contamination in the compact matrix) also effects the measured release. The Xe-138 R/B ratio ranged from 6×10^{-10} to 2×10^{-8} over the same irradiation time span. Figures as low as 6×10^{-10} likely indicate some momentary change in the test/measurement configuration since they occur suddenly (a single day-averaged figure) in all the capsules on specific intervals across all the measured isotopes. The maximum R/B ratio is two orders of magnitude lower than the 95 percent UCL for number of exposed TRISO kernel failures reported in Table 2-1 for AGR-1. This would seem to indicate that the 95 percent UCL is conservative. For AGR-1 this indicates that there were zero exposed kernel failures in that particular lot of TRISO particles.

[4] R/B is an indicator of initial fuel quality and fuel performance; it is defined as the ratio of the release rate (measured) over the birth rate (calculated) of short-lived fission gases that are released from exposed kernels (because of defective or failed coating layers) or dispersed uranium contamination outside of the coating layers (EPRI 2020).

In AGR-2, R/B ratios also span a factor of about 100 between minimum and maximum values but are also roughly 10 times larger overall than in AGR-1. A complication with the AGR-2 data is due to a damaged fission gas line that caused unintentional intermixing of gas flows between capsules resulting in confusing differences between certain capsule trends (in essence, resulting in an averaging of the test data). In the data that could be qualified, the Kr-85m R/B ratio ranged from about 3×10^{-8} to 2×10^{-6} for irradiation times from zero up to 150 EFPD. For the reasons just explained, the data above 150 EFPD in AGR-2 could not be qualified. As with AGR-1, the Kr-88 R/B ratios were comparable though slightly less than that for Kr-85m. The Xe-138 R/B ratios ranged from about 1×10^{-8} to 3×10^{-7} over the same 150 EFPD irradiation time span, up to almost a factor of 10 lower than those for Kr-85m.

Even though all the R/B ratios are very small, it is important to point out that all the maximum values that occurred in the AGR data sets had the highest irradiation temperatures. All other things being equal, higher maintained irradiation temperature has a potent effect. Total burnup in AGR-2 is significantly less than AGR-1, yet AGR-1 has lower R/B values. The highest releases of several isotopes occurred in AGR-2 capsule 2 which also had the highest maintained irradiation temperature across both series of tests. This infers a diffusion release mechanism.

The data for noble gas release during normal operating conditions demonstrates that a significant portion of the tiny amounts of noble gas fission product release can stem from uranium contamination allowed within the fuel fabrication specifications for PyC matrix materials in the TRISO manufacturing process. Assays showed that the difference in trace uranium contamination was about a factor of 10 higher in AGR-2. However, the AGR-2 series had a few recorded exposed kernel particle failures so the increased release in AGR-2 is likely a combination of multiple causes. Regardless, the noble gas fission product releases are still very small.

2.1.4 Other Representative Fission Product Releases

Silver (Ag) fission product isotopes in TRISO are well known to diffuse and escape TRISO particles at normal fuel operation temperatures – at much lower temperatures than in heat soak accident conditions. The Ag in its metallic form melts at a lower temperature than typical TRISO operating temperatures. Silver oxide melts at a much lower temperature than Ag metal and decomposes at a comparatively low temperature as well. Since Ag does not remain oxidized or in solid form at comparatively low temperatures, and has a modest atomic mass, it has optimum characteristics that contribute to its rather high mobility in the fuel and core materials via diffusion.

In TRISO fuel Ag crosses both PyC and SiC layers, but Ag has been demonstrated to remain in the larger and cooler PyC matrix of the compacts and especially in the pebble or prism components in common TRISO fuel and core designs. Silver does not form compounds with silicon (Si) and its dicarbide form has a very low melting point and is unstable, decomposing readily to Ag and carbon (C) since the bonds are weak ionic bonds. Silver cannot scavenge C from SiC or graphite structures since those are much more tightly bound. It is free to roam by diffusion if the temperature is high enough.

Silver is less mobile at the temperature of the reactor PyC core materials near the coolant flows and higher adjacent to hot TRISO particles and TRISO compacts. Silver can be expected to accumulate in cooler parts of the PyC core structure over time, but those hosting materials must include higher temperatures for the Ag to diffuse prior to encountering lower temperatures and

becoming more stationary depending on the local Ag concentration gradient in the host material matrix. This expected behavior is demonstrated in the AGR test series. The amount of diffused Ag fission products is a function of time at a given temperature in a host material – PyC in this case.

The relative distance between the coolant gas and the TRISO bearing compacts is dependent on the prismatic core design. Pebble beds are a different configuration, but still the effective distance between the higher densities of TRISO particles and the encompassing PyC matrix material effects fission product diffusion in those core designs as well. Silver and other diffusing isotopes are not volatile fission products unless a much higher temperature is achieved such that Ag has a sufficiently high vapor pressure. However, the low Ag vapor pressure at operating temperature is sufficient to allow circulation at a low partial pressure. This indicates that over a sufficient period of time, Ag can plate out on cooler surfaces in the primary coolant circuit and that a small quantity of Ag vapor is present in the coolant during operation. Once operation ceases and temperatures drop, Ag will fully condense out of the coolant gas onto interior surfaces, including on graphitic fines that accumulate electrostatically on some surfaces (General Atomics 2008).

The amount of fission products that might potentially spread into the broader reactor coolant system components due to contamination of the coolant as it passes through the reactor core is unknown, apart from conducting PIE and post-shutdown examination of reactor coolant system components. Condensable fission products can plate out or settle out on the interior surfaces within the reactor system, or they may be retained in the coolant and removed via the coolant cleanup system. In certain core designs, which may involve fuel element hydraulic chatter in prismatic cores or in other designs such as a pebble bed reactor where pebbles move and tribology occurs at contact points, graphitic fines may be produced from fuel element surfaces and possibly circulate in the reactor coolant system. If circulation occurs, encapsulated fission products, that previously diffused into that graphitic material, would presumably follow those fines.

In the dry helium environment, potential transport of Ag isotopes at normal reactor operating temperatures would likely be minimized since there are few chemistry pathways. If maintenance of a suitable gas fill condition and low core temperature are a requirement for transportation of the reactor, then the source term of Ag isotopes should be very low, perhaps only occurring if there were loose dust composed of PyC core materials bearing diffused or later plated out Ag – particularly those directly associated with or near TRISO bearing elements. The existence of such particulates, dust, and aerosols, may initially exist associated with manufacturing and assembly. Additional dust would seem to imply tribology or other fracture mechanics occurring in the core from a combination of chatter during reactor operations (or other relative motions) plus any impacts and vibration associated with reactor module movements or transportation.

Silver release fractions from compacts into the surrounding PyC matrix can be quite high under normal fuel operating conditions. In the AGR-1 tests, up to 70 percent of the Ag inventory produced in certain capsules escaped both the TRISO particles and the fuel compacts by diffusion into the capsule PyC material. The capsule PyC mimics the surrounding PyC core component materials in a prismatic or pebble core design.

Up to roughly 80 percent of the total Ag inventory escaped the TRISO particles (diffused through the SiC layer) and 10 percent remained within the fuel compacts. This is likely because the fuel compacts and nearby materials are hotter, and Ag is more mobile at higher

temperatures. As a result, it is expected that it would accumulate further away where temperatures are lower, or another barrier intercedes and blocks diffusion (e.g., the metal capsule wall that is adjacent to cooling water in the AGR tests).

Plainly the effect of the temperature field in the diffusion problem can be potent over time. Mobile trace species will tend to migrate away from higher temperature areas (such as TRISO particles and a given “hot” compact) and toward cooler areas in the capsule where that species is at a lower concentration. If there is a “cold spot” in the capsule, it could act as a “cold finger” and accumulate a species around that spot though the species may not penetrate to the center of a cold spot if it is sufficiently cool. Distribution of diffused species concentrations in host materials can be very complex since they are highly temperature dependent, time dependent, and concentration gradient dependent.

If the primary circuit were to contain humid air (predominantly nitrogen [N], oxygen [O], hydrogen [H]), then other compounds of Ag could potentially form. Typically, in the case of Ag, these compounds have low melting points and low decomposition temperatures such that they are likely to be more mobile than elemental Ag at somewhat lower temperatures. Decomposition reactions for Ag species can occur below 500 °C. Maintaining a dry inert environment sets the least reactive conditions that could somehow contribute to mobilizing Ag. Maintaining a clean dry helium environment would be far better than allowing any uncontrolled air ingress and especially so if temperatures are elevated far above ambient during transportation.

The range of Ag release data from TRISO particles into compacts is 500-fold, measured on a compact-by-compact basis. The range of Ag release from fuel compacts into the capsules is 70-fold. Deconvolving the effects of TRISO layer failures on slowing diffusion out of TRISO particles (PyC density, SiC crystal boundary morphology, and various layer debonding and fractures) and the strong effect of local material temperature on mobility and accumulation beyond the particles, is a complex matter that is not fully illuminated in the irradiation studies to date.

Table 2-3 summarizes the non-gaseous fission product release results for the AGR-1 and AGR-2 tests. The table provides the minimum and maximum fractional inventory results for Ag-110m, Cerium-144 (Ce-144), Cesium-134 (Cs-134), Europium-154 (Eu-154), and Strontium-90 (Sr-90) in two different regions of the core, specifically the compacts and the capsules.

Table 2-3. Maximum and Minimum Inventory Fractions from Advanced Gas-Cooled Reactor-1 and Advanced Gas-Cooled Reactor-2 Test Strings.

Isotope	Holdup Location	Maximum Inventory Fraction	Minimum Inventory Fraction
Silver-110m	Compacts	1×10^{-1}	2×10^{-4}
	Capsules	7×10^{-1}	1×10^{-2}
Cerium-144	Compacts	2×10^{-3}	1×10^{-6}
	Capsules	1×10^{-5}	3×10^{-9}
Cesium-134	Compacts	6×10^{-4}	4×10^{-7}
	Capsules	9×10^{-5}	3×10^{-8}
Europium-154	Compacts	*1×10^{-1} (2×10^{-2})	3×10^{-4}
	Capsules	*4×10^{-2} (4×10^{-4})	**1×10^{-6} (3×10^{-5})
Strontium-90	Compacts	*8×10^{-2} (3×10^{-3})	2×10^{-6}
	Capsules	*2×10^{-2} (9×10^{-5})	9×10^{-7}

AGR-1 = advanced gas-cooled reactor-1, AGR-2 = advanced gas-cooled reactor-2.
* The first value is the result for AGR-2, Capsule 2 data, which had higher average irradiation temperature plus additional uranium contamination than the other AGR-2 capsules. The parenthetical value is the maximum value among all the other AGR-2 capsules and the AGR-1 capsules.
** The first value is at the limit of detectability for the measurement method used. The parenthetical value is the minimum value among data with reliable measurement accuracy.
Adapted from EPRI 2020.

The inventory fraction is defined as the inventory measured divided by the calculated inventory (total expected inventory based on Monte Carlo transport code [MCNP]-ORIGEN2[5] calculations for the specific fuel irradiation case). Inventory is measured using various destructive analyses (e.g., deconsolidation leach-burn-leach followed by direct measurements of the significant species in leached solutions).

The “compacts” holdup location includes all the material between the TRISO SiC layer outer boundary and the compact outer boundary. The OPyC layers on the involved TRISO particles are included as “capsules” inventory since this material is in close diffusive communication with the capsule matrix material and cannot be reliably distinguished experimentally in the PIE measurements. The “capsules” holdup location includes all the PyC material from the outer compact boundary to the inner capsule boundary (a sealed metal container).

The maximum inventory fraction is defined, for the purposes of this study, as a one significant figure value rounded up from the highest value in the AGR-1 and AGR-2 data (a conservative value for these experimental data). The minimum inventory fraction is presented only for the purpose of illustrating the total range of the data in each case.

Though a broader span of isotopes of the indicated elements are produced, the listed should be considered as indicative for all the isotopes of the given element since mobility of the isotopes through various materials is largely determined by physicochemical processes with isotopic separation effects being small compared to accuracy of the data. The approach taken in

[5] ORIGEN2 is a point-depletion and radioactive-decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions (Croff 1983).

measurement was to select the indicating isotopes based on which one would potentially produce the most accurate data using the available quantitative methods. As indicated in data sets, in some cases the inventory measured was at the limit of detection for the method used.

The case of AGR-2, Capsule 2 represents a slightly lower burnup case with a significantly higher irradiation temperature than any of the other data in the AGR-1 and AGR-2 data sets. All things considered, including the burnup and level of additional trace uranium contamination of PyC materials, it appears the higher irradiation temperature has a much stronger effect than other factors with respect to diffusing metal isotopes. In the case of the europium (Eu) and strontium (Sr) bounding maximum inventory fractions have a substantial effect and the data range for those higher temperature irradiation data do not overlap with any other capsule data. As a result, the maximum values for the non-AGR-2 Capsule 2 data are presented since they are significantly lower. Both the AGR-2 Capsule 2 data and data for all of the other AGR-1 and AGR-2 capsules are presented separately because the values are perturbed by at least three different independent variables: level of trace uranium contamination, burnup, and irradiation temperature.

Uranium contamination might increase the maximum values but taken alone it cannot explain the two orders of magnitude difference in some of the maximum values (difference in trace contamination is 10-fold, not 100-fold). Burnup is higher for the AGR-1 data, but not dramatically. The slightly lower burnup does not explain the larger differences since it should move contrary to the observed behavior. Higher irradiation temperature may be the most prominent cause since these two isotopes have diffusion behaviors comparable to, but not as mobile as, Ag in the given temperature range. Hypothetically, the appreciably higher irradiation temperature may allow greater diffusion of Eu and Sr in AGR-2 Capsule 2 in addition to some augmentation by uranium contamination in the PyC materials within that AGR-2 series capsule. It does not seem possible to deconvolve these effects in these data sets.

The non-parenthetical maximum inventory fractions can be considered an initial bounding case. In the Eu and Sr data, these maximum inventory fractions may prove to be quite conservative as they are roughly 10 to 100-fold higher than the highest among the non-AGR-2 Capsule 2 data. In the case of Ag, the higher irradiation temperature only appears to double the maximum inventory fraction in the capsules from about 30 percent to 70 percent and the data overlaps across AGR-1 and AGR-2. At these high fractions of total inventory, mobility may be more sensitive to species concentration gradient since the bulk value is more rapidly reducing in the TRISO particles and compact matrix. In the small fractional values, most of the inventory still remains either in the particles or compacts and acts as a nearly constant source term for the tiny capsule inventory that is slowly building up. The details of where diffused species will arrive, and in what concentration, is complex and sensitive to several time dependent variables.

Silver should stay mostly bound within the core materials as long as the temperatures are not extremely high, which would facilitate broader diffusion and increase the vapor pressure of Ag sufficiently to allow more evaporation from the surface into the coolant or other gases (presuming no air ingress allowing chemistry with N, O, H, or other potentially available materials exposed to Ag within the core might produce Ag species that have other pathways at lower temperatures).

This suggests that the plausible dispersal mechanisms of concern for Ag, following core cool down, involve tribology and fracture mechanics of Ag bearing host materials such as PyC, etc. Given proper data, these mechanisms should be possible to evaluate by bounding analysis. It may be that these mechanisms can be ruled out entirely or constrained by imposition of

compensatory measures applied to the transportation modalities as a requirement for the license of the transportation package.

2.2 Accident Release Test Data

Available test data on TRISO fuel failures, and associated release of radioactive material, due to an accident during operation of a TNPP (DBE or beyond DBE) is not applicable in this risk informed transportation analysis since a reasonable condition set for an exemption on package certification may be that an accident condition has not previously occurred during reactor operations. It is expected that any accident condition would require detailed analyses to understand the condition of the reactor before a specific approach to disassembly and transportation could be developed. However, it may be useful for the purpose of this study to consider the most extreme AGR-1 and AGR-2 series heat soak tests as exemplary cases setting expectations for fission product mobility from TRISO, into and through compacts, and into near compact core structural components.

Experimental data that simulates a loss of coolant flow accident are presented below to illustrate how a severe accident effects failure data. The upper bounds of these heat soak testing circumstances simulate a condition well beyond any expected loss of flow accident in typical commercial scale HTGR reactor designs utilizing an inert gas coolant. A subset of compacts from each irradiation string were irradiated and then heat soaked under fully inert conditions (dry helium) at 1600 °C, 1700 °C, and 1800 °C. These tests all ran up to 300 hours. This heat soak period is far longer than calculations indicate as an expected accident heat soak period given expected decay heat transients. As such they are considered to be conservative representations of these accident conditions.

Certain prototype and production type microreactors may vary from this expectation because of potentially significant design differences from much larger stationary power reactor designs. For example, the expected peak temperatures in a TNPP may be lower because the smaller geometry of the reactor core is expected to allow more effective passive heat removal. However, variable external hazards of TNPPs introduces additional uncertainty (e.g., potential burial in a landslide or inundation in a flood or other transportation related accidents).

Table 2-4 provides the combined statistical performance of AGR-1 and AGR-2 test strings at 1800 °C heat soak for 300 hours. This presents a very conservative accident case with no significant air or water vapor ingress.

As reported in Table 2-1, under normal reactor operation conditions, the 95 percent UCL on the failure fraction for UCO TRISO fuel manufactured in the U.S., using methods that supplied fuel to the AGR-1 and AGR-2 fuel tests, may contribute to less than 4 failures per 100,000 fuel particles during a full fuel duty cycle. Under severe (with exception of inert helium) accident conditions, bounded by 1800 °C up to 300 hours (implausible in many passively cooled reactor designs), this figure may rise to less than 3 failures per 2000 fuel particles. That is roughly a 40-fold increase in TRISO failure fraction in a severe heat soak accident when compared to normal operations.

Table 2-4. Apparent (Inferred) Post Irradiation Failure Data Simulating Heat Soak Accident (300 hours at 1800 °C).

Test String	Number of Compacts	Number of Particles	Silicon carbide Layer Failures (Missing Layer)			Full TRISO Particle Failures(Exposed Kernel)		
			Number of Particle Failures	Failure Fraction	95% Upper Confidence Limit Failure Fraction	Number of Particle Failures	Failure Fraction	95% Upper Confidence Limit Failure Fraction
AGR-1 + AGR-2 (Heat soaked at 1800 °C)	7	26,028	24	9.2×10^{-4}	$<1.3 \times 10^{-3}$	3	1.2×10^{-4}	$<3.0 \times 10^{-4}$

AGR-1 = advanced gas-cooled reactor-1, AGR-2 = advanced gas-cooled reactor-2, TRISO = Tri-structural Isotropic particle fuel.
Adapted from EPRI 2020.

However, accident condition failures are very sensitive to increases in temperature above 1800 °C since the SiC layer chemically decomposes as it approaches and rises above 2000 °C. Particles should be conservatively expected to eventually release all volatile and certain diffusive fission products somewhat above this sustained heat soak temperature. Temperature uncertainty is factored into these 95 percent probability values. The time average temperature uncertainty is roughly ±5% between measured and calculated values for the test strings.

Additional information on accident tolerance of the TRISO fuel is not included here. A reactor that has endured an accident either in operation or in prior transportation should not be further transported apart from a specific assessment and related regulatory efforts prior to licensing for that specific package condition. The impact of accident conditions and recovery should be considered as part of any transportable reactor operations contingency analysis.

2.3 Other Applicable TRISO Fuel Test Data

In typical HTGR stationary plant deployments in the future, a TRISO fueled nuclear power plant will meet design requirements for external hazards comparable to other stationary nuclear power plant. However, a TNPP is subject to a different and broader range of hazards because the reactor is being transported which makes it subject to different hazards such a collision, impact, vibration, and other external hazards along its transportation route. Inherent in the design of TNPPs are tradeoffs between transportability considerations (e.g., maximum weight of a shippable unit, etc.) and the shielding and containment safety functions. Also, hazards are introduced because of the need to assemble and dismantle the TNPP to deploy and to transport the unit. Hazards associated with disassembly and re-assembly of the TNPP are important activities to evaluate for site-based licensing but are not included within the scope of certification for a transportation package. In this report, the focus is on normal transportation of a reactor module within the conterminous United States, with NRC approval of the TNPP transportation package and DOT regulation of its shipment during transit.

A principal difference posed by the user requirement of transportability are hazards to the reactor, its irradiated fuel, and contaminated systems and components (e.g., portions of the reactor coolant system) resulting from accidents during transport of the TNPP. Transportation accidents could involve kinetic hazards (such as impacts with vehicles or other objects), fire hazards, random failures, or human errors (e.g., human error in preparing the TNPP package for transport), natural hazards (such as earthquakes, landslides or burial, and tornados), and potential submersion in water. Assessment of each of these hazards during transportation of the TNPP package is described later in this report.

In general, because the TNPP reactor vessel and portions of reactor coolant system also provides the primary means of maintaining containment during transportation (in addition to providing this function during reactor operation), it seems unlikely that TNPP packages, containing irradiated TRISO fuel, will be able to meet the Hypothetical Accident Conditions requirements of 10 CFR 71. Tests and analysis would be required to prove otherwise.

With respect to the TRISO fuel and compacts, and 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 disassembly and reconfiguration of fuel and near fuel materials in addition to potential failures of fission product barriers and production of fine particulates that represent a potential source of released radioactive material.

Since no transportation assessments have yet been completed on the performance of irradiated TRISO fuel and compacts that are not transported in a certified Type B package, or a “cask,” an engineering judgement must be made in this study regarding the consequence of impacts and vibration during anticipated transportation conditions and in bounding accident conditions. Do these dynamic loads imply a significant hazard to the fission product barriers in the fuel and near fuel structures? To make a judgement requires knowledge of fresh and irradiated material mechanical properties. These properties are generally determined as part of the efforts underlying fuel qualification and reactor design certification.

At high burnup there is weakening of fuel material strength and considerable fission gas pressure behind the SiC barrier. The SiC layer acts as a pressure vessel for gaseous and volatile fission products in a TRISO particle. This is a complex fuel performance problem since layer interactions typically unload the SiC layer in the initial stages of the burnup. At some point in a mid-burnup range, the dynamics change and the SiC barrier begins to load until eventually a barrier failure occurs at very high burnup. The SiC barrier is essentially a pressure vessel that can fail given sufficient fission gas pressure combined with reductions in material strength.

The AGR-1 and AGR-2 qualification studies demonstrate burnup as high as 19% FIMA with no obvious broad failures of SiC layers. At the expected lower burnups in certain TNPP demonstration designs, which may be ~10% FIMA or below, gas pressure is not expected to fail the SiC layers. If insulted by a sudden and potent thermal or mechanical impact, when in a condition of high burnup, some SiC barriers might fail indicating some enhancement of release of gaseous material at risk (MAR). More likely of concern in low burnup cases is vibration spectra and consequence that is critical to understand fatigue tolerance of the fuel compacts and near fuel materials and tribology that could release fine particles that represent a source of MAR.

Similarly, with fire hazards, the irradiated thermal material properties are important to understand maximum temperatures that may occur in the fuel and core materials that hold up diffusing fission product species. Time at high temperature is one of the common sources of

release of radionuclides from TRISO fuels. In this case, the combination of fire applying heat to the exterior of the reactor and decay heat from the core would determine the outcome. Since the TRISO fuel typically performs very well up to 1400 °C, remaining decay heat is a significant factor in a consequence analysis. The codified regulatory pool fire test is set at 800 °C, which is conservative for a liquid fossil fuel pool fire. In the absence of sufficient decay heat, the fire hazard would seem to be related to potential enhancement of barrier failures other than TRISO fuel and near fuel core component barriers such as failures of seals on the reactor coolant system barrier that could release plated out and particulate radioactive material released from TRISO fuel during normal reactor operations.

Because the temperature of the graphite needed to produce a self-sustaining graphite fire is not expected to be reached during plausible transportation accidents, involving limited air ingress though a failed seal (considering the contribution from both decay heat and an 800 °C fire for 30 minutes), a so-called “graphite fire” is not plausible for conditions during bounding transportation accidents. Rather, a process of very slow surface oxidation might occur that proceeds depending strongly on temperature of the graphite (strongly implying high available decay heat or high injected air temperature both of which seems implausible).

Extended time at temperature, sufficient oxygen, and possibly water vapor adjacent to graphitic materials must be maintained for over 100 hours at well over 1000 °C simply to oxidize the fuel sleeves and compact graphite (Moormann 2011). However, a small amount of oxidation (if any occurred) could produce gaseous reaction products and heat could potentially release aerosols that contain radioactive material that might have been previously plated in the primary circuit (especially near a failed seal where hot moist air might enter), but a self-sustaining fire, based on the reactor’s graphitic materials acting as an oxidizing fuel, seems implausible and the TRISO fuel particles themselves resist air oxidation caused releases up to 1400 °C for well over 100 hours at that temperature. Hence, any increase in releases from a transportation accident involving a fire would at worst be associated with materials that had previously plated or settled in the primary circuit.

Moreover, storage of significant Wigner energy in these graphitic structures in this reactor should not pose a significant risk in transportation since the core graphite operates at a high temperature and is therefore annealed. Other heat sources would have to raise the temperature of the graphite above its operating temperature. Thus, any stored energy from irradiation of the graphite should not significantly contribute to any fire hazard during plausible transportation accidents (Schweitzer 1987).

The material properties of TRISO fuel and close core materials are most relevant to potential release of radioactive material during expected bounding accidents, other than those linked to fire hazards, are strength of materials properties. The ability of TRISO fuel, compacts, and close core structures to tolerate impact loads and vibration spectra and how that may influence the potential to release radioactive material during a bounding transportation accident has not been examined in detail in the literature to date.

The TRISO fuel particles themselves are composite structures composed of layers as shown in Figure 2-1. The UCO kernel and SiC layer are ceramic materials and the PyC layers are strongly anisotropic graphite materials, all of which tend to display brittle failure characteristics and comparatively high failure strengths. Moreover, there is an inner porous PyC layer deposited on the UCO kernel surface that acts as a buffer to allow expansion of the UCO kernel under irradiation, in addition to providing a gas plenum that contains fission gases as they evolve up to very high burnup.

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

The TRISO particle geometry and degree of bonding between the deposited layers and their behavior under irradiation results in the overall observed composite properties of an individual particle and the ensemble of all the particles taken together determines the composite properties of the compact or pebble fuel form. This is the case for nearly all the ensemble properties and particularly true for the thermal and mechanical properties. For the mechanical properties that matter most under transportation accident conditions, evaluation of the fuel form is rather complex.

Some direct measurements of hardness of the containment layers inside the TRISO particles have been made. These measurements were made on unirradiated materials, so they represent the case without radiation damage. Irradiated materials are expected to be somewhat weaker. However, this is a fuel that operates at very high temperature and so annealing may limit the influence of radiation damage and especially for the SiC pressure boundary since that material begins to decompose at about 500 °C beyond the typical fuel operating temperature of up to about 1400 °C. The PyC materials and UCO kernel (composed of UO_2 and UC_2) do not decompose until far higher temperatures than SiC.

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

In any event, some very precise and stringent preparations and measurements on actual unirradiated TRISO particle materials have been made (Byun 2008, Hosemann 2013). To perform these measurements, individual particles are abraded to expose a cross section and then indentation hardness measurements are made. The hardness measurements demonstrate that the SiC layer has over 10 times the hardness of the typical high density PyC material in the adjacent layers. The high density PyC layer material is ~3 gigapascal (GPa) whereas the SiC material is ~40 GPa hardness. From these hardness measurements, it is clear that the SiC layer is a rather dominant element in the overall TRISO particle strength. The hardness of the porous PyC buffer layer is not given in those reports since it doesn’t 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 megapascal (MPa) and ~1000 MPa. Local fracture stress is roughly twice the magnitude. In British units, this corresponds to a range between ~29,000 pound-force per square inch (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 measurement would have to be done in a hot cell presumably. Fracture of the particle could be detected easily 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 properties.

Moreover, crushing of whole irradiated compacts would be useful to understand the strength of the fuel form as a whole under various impact load circumstances. 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 (converted to an impact load[6]), then expectations of release fractions may not include fission products still contained inside the TRISO particle SiC layers (most of the inventory).

Similarly, if models show that expected impact loads would not fracture compacts, then contained radioactive material in the compacts may also be unavailable for release during bounding accidents. Understanding the relative strength of irradiated core structural components (graphitic), TRISO compacts, and TRISO particles in comparison to expected impact loads under various bounding accident scenarios is essential to judging the degree of conservatism in release models.

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 are transferred to the core structures).

In this fuel type, the concern would seem to be diffused fission products that may be held up in graphitic materials that may be released into the primary circuit 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 (e.g., if a particular design concept

[6] Conversion to an impact load in a composite material is complex since specific geometry of the application and time duration of loading, plus the specific geometry and mechanical properties of the component materials in the composite (including the nature of their bonding at dissimilar material interfaces) determines how much energy can be absorbed in a given volume of the composite. The static fracture stress may be lower than an impact fracture stress under many conditions, but to understand the relationship between the two often requires measurements since many factors are involved and each factor has uncertainties that may or may not be known.

requires opening of the primary circuit to prepare for transportation of TNPP modules or to unload and assemble them for operation) and much less with respect to reactor operations or during transportation.

3.0 Historical Experience with Transportation of TRISO Fuel

Significant quantities of both unirradiated and irradiated TRISO fuel have been shipped in the U.S. In all cases these shipments were performed using NRC-approved transportation packages. Section 3.1 provides examples of several of the approved transportation packages and provides examples of TRISO shipments that have been made using these packages. Section 3.2 summarizes release assumptions that have been made in transportation accident analyses performed for environmental impact statements (EISs) involving the shipment of irradiated TRISO fuel using approved transportation packages.

3.1 Approved TRISO Fuel Transportation Packages

There have been several transportation packages for irradiated and unirradiated TRISO fuel approved by the NRC. Examples of these transportation packages include:

- the FSV-1 legal weight truck transport cask;
- the TN-FSV legal weight truck transport cask;
- the FSV-3 transportation package;
- the Versa-Pac transportation package; and
- the ES-3100 transportation package.

The FSV-1 transportation cask (NRC Docket No. 71-9277 and Docket No. 71-6346) was certified to ship irradiated HTGR TRISO fuel elements from the Fort St. Vrain nuclear power plant. For the Fort St. Vrain fuel, the maximum enrichment allowed by the certificate of compliance (CoC) was 93.5 percent. The fuel elements consisted of a graphite body that was hexagonal in horizontal cross section, approximately 31.2-inches (in) high and 14.2-in across the flats. The FSV-1 transportation cask had a capacity of six Fort St. Vrain irradiated fuel elements. The CoC did not specify a maximum burnup but did specify a maximum decay heat of 4.1 kilowatts (kW). The FSV-1 was a Type B(U)F[7] package and the CoC expired on October 1, 2008. The FSV-1 was used to make 123 truck shipments of irradiated Fort St. Vrain fuel elements from Fort St. Vrain to the INL (NRC 2010).

The TN-FSV transportation cask (NRC Docket No. 71-9523) is currently certified to ship irradiated HTGR TRISO fuel elements from the Fort St. Vrain nuclear power plant and bi-structural isotropic (BISO) fuel elements from Peach Bottom Unit 1, Core 2. For the Fort St. Vrain fuel, the maximum enrichment allowed by the CoC is 93.5 percent, the maximum burnup is 70,000 megawatt-days per metric ton of heavy metal (MWd/MTHM), and the minimum cooling time is 1600 days. For the Peach Bottom fuel, the maximum enrichment allowed by the CoC is

[7] The NRC Type B(U)F package designation, and other designations such as Type AF, is defined in 10 CFR Part 71. Generally, Type B packages are used to transport materials with high levels of radioactivity, such as spent nuclear fuel. The U means that the Type B package, together with its radioactive contents, for international shipments requires unilateral approval only of the package design and of any stowage provisions that may be necessary for heat dissipation. The F means the Type B(U) package is approved for shipment of fissile material.

93.15 percent, the maximum burnup is 73,000 MWd/MTHM, and the minimum cooling time is 27 years. The TN-FSV transportation cask had a capacity of six Fort St. Vrain irradiated fuel elements and is a Type B(U)F package. The CoC for the TN-FSV does not authorize the air transport of fissile material.

The FSV-3 transportation package (Docket No. 71-6347) was certified to ship unirradiated HTGR TRISO fuel elements from the Fort St. Vrain nuclear power plant. The CoC allowed enrichments up to about 93 percent. The CoC expired on October 1, 2008. The FSV-3 was a Type AF package. The CoC for the FSV-3 did not authorize the air transport of fissile material.

The Versa-Pac transportation package (Docket No. 71-9342) is certified to ship unirradiated TRISO fuel in the form of fuel kernels or compacts. The Versa-Pac is available in two configurations, the VP-55 (a 55-gallon drum) and the VP-110 (a 110-gallon drum) and is a Type AF package. For enrichments less than 20 percent, the Versa-Pac is limited to 410 grams of U-235. The CoC for the Versa-Pac authorizes transport by air.

The ES-3100 transportation package (Docket No. 71-9315) is certified by the DOE to ship bulk highly enriched uranium unirradiated TRISO. Up to 2 kilograms (kg) of TRISO fuel containing 1.815 kg of uranium-235 (U-235) is authorized. The ES-3100 is a Type B(U)F package. The NRC CoC for the ES-3100 does not allow transport of irradiated TRISO fuel.

In summary, approved transportation packages for irradiated and unirradiated TRISO fuel were/are either: (i) Type AF packages in which the total quantity of radioactive material in the package cannot exceed a Type A quantity that will likely be exceeded by a TNPP package with irradiated fuel or (ii) Type B packages that have requirements the TNPP package is unlikely to meet. Accordingly, none of these transportation packages appear to provide any important insights into transport of a TNPP package.

3.2 Transportation Accident Release Fractions Used for EIS

Environmental Reports and EISs have generally used the accident severity categories, conditional probabilities and release fractions[8] developed in NUREG/CR–6672 (NRC 2000) to estimate the radiological impacts of transportation accidents involving spent nuclear fuel. These release fractions (or source term factors) are for boiling water reactor and pressurized water reactor (PWR) spent fuel shipped in truck and rail transportation packages that are certified by the NRC. In the Yucca Mountain Final EIS (DOE 2002) and the Repository Supplemental EIS (DOE 2008a), these source term factors were modified for TRISO fuel. Table 3-1 lists the assumptions used in these EISs to estimate transportation accident releases for TRISO fuel.

Table 3-2 provides the impact speed versus velocity matrix for transportation accidents involving TRISO fuel based on the matrix from NUREG/CR–6672 (NRC 2000), modified based on expert judgement, to account for the attributes of TRISO fuel. This matrix provides the conditional probability given an accident and transportation accident release fractions (i.e., source term

[8] NUREG/CR–6672 (NRC 2000) uses the term “release fraction” to mean the fraction of radiological material released from containment that becomes the “source term” in a radiological dose consequence calculation. The term “release fraction” is also used in this report to refer to the fraction of radiological material that is released from the TRISO fuel during normal reactor operation and diffuses into the reactor core elements. Therefore, the term “source term factors” is used here in conjunction with the term “release fraction” to clarify that it pertains to the fraction of material released during a transportation accident opposed to material released during normal operation.

factors) for 19 accident scenarios. In DOE (2002) and DOE (2008a), the irradiated TRISO fuel was assumed to be shipped in a Type B transportation package and so all releases were from failed cask seals. For this reason, the aerosolized fractions and respirable fractions associated with the releases were assumed to be 1.0 for all the scenarios. Release fractions are provided for krypton (Kr), cesium (Cs), ruthenium (Ru), and particulates for each cell in the matrix. In DOE (2002) and DOE (2008a), the irradiated TRISO fuel was assumed to be shipped in a Type B transportation package and releases were from failed seals. For this reason, the aerosolized fractions and respirable fractions associated with the releases were assumed to be 1.0.

Table 3-1. Transportation Accident Release Calculation Assumptions for TRISO Fuel.

Model Element	Modeling Assumption and Basis
Release fraction from impact	The release fraction for damaged particles is assumed to be the same as the pressurized water reactor (PWR) oxide fuel release fractions. Not all particles are assumed damaged. For Tri-structural Isotropic (TRISO) particle fuel, the fraction of the fuel particles damaged on impact is assumed to be 0.1% for cask impacts with an unyielding surface ranging from 60-90 miles per hour (mph), and 1% for impacts greater than 90 mph.
Fuel release fraction from thermal stress	The PWR thermal release fractions were used if the fuel particles were damaged by impact. It was assumed that particles not damaged on impact would not fail from thermal heating. They are designed to maintain their integrity in a higher-temperature reactor environment.
Clad or canister release	No canister is assumed. The pressure inside the cask at the time of the release is based on the assumption that the cask was closed at room temperature and atmospheric pressure. The fuel temperature during normal transport was assumed to be 300 °C. Thus, the pressure inside the cask at the time of cask failure by impact was assumed to be 1.91 atmospheres (13.3 pounds per square inch gauge [psig]).
Fraction of radionuclides deposited in cask inner surfaces	Same as PWR deposition fraction.
Cask damage	Same as cask loaded with PWR fuel.
Fraction of deposited material volatilized in fire when two areas of the cask are breached	Same as PWR volatilization fraction.
Release fraction from cask	Gas pressure in cask from thermal heating is the only driving force for release. Pressure is based on closure at 27 °C at atmospheric pressure.

Source: Jason Technologies (2001), Table 5-19.

Table 3-3 lists the accident severity categories, conditional probabilities, and transportation release fractions (i.e., source term factors) used for truck transportation accidents involving TRISO fuel combined into six accident severity categories. These accident severity categories, conditional probabilities, and release fractions were also used in the Global Nuclear Energy Partnership Programmatic EIS (DOE 2008b and Dimsha 2008).

The baseline transportation accident release fractions (i.e., source term factors) used in the parametric study to determine the dose consequences from the selected transportation accidents are presented in Section 5.3.3 and Section 5.4.1 of this report. The selected sensitivity studies performed to provide insights about the impact of the modeling uncertainty associated with these values are defined in Section 5.4.2.

Table 3-2. Truck Transportation Accident Temperature-Impact Matrix for TRISO Fuel.

Equivalent Impact onto an Unyielding Surface (mph)						
	>120	1 Seal Failure on Impact Part 3.57E-9 Ru 3.57E-9 Cs 3.57E-9 Kr 4.76E-3 Prob 1.53E-8	11 Seal Failure on Impact Part 3.89E-9 Ru 3.89E-9 Cs 3.89E-9 Kr 5.18E-3 Prob 1.49E-10	12 Seal Failure on Impact Part 5.30E-9 Ru 5.30E-9 Cs 5.30E-9 Kr 7.07E-3 Prob 2.41E-14	13 Seal Failure on Impact Part 5.73E-9 Ru 5.73E-9 Cs 6.32E-8 Kr 7.64E-3 Prob 0	17 Failure by Shear/Puncture Seal Failure by Fire Part 5.73E-9 Ru 6.32E-8 Cs 6.32E-8 Kr 7.64E-3 Prob 0
	90-120		8 Seal Failure by Fire Part 3.89E-9 Ru 3.89E-9 Cs 3.89E-9 Kr 5.18E-3 Prob 1.17E-8	9 Seal Failure by Fire Part 5.30E-9 Ru 5.30E-9 Cs 5.30E-9 Kr 7.07E-3 Prob 1.90E-12	10 Seal Failure by Fire Part 5.73E-9 Ru 5.73E-9 Cs 6.32E-8 Kr 7.64E-3 Prob 0	16 Failure by Shear/Puncture Seal Failure by Fire Part 5.73E-9 Ru 6.32E-8 Cs 6.32E-8 Kr 7.63E-3 Prob 0
	60-90		5 Seal Failure by Fire Part 3.89E-10 Ru 3.89E-10 Cs 3.89E-10 Kr 5.18E-4 Prob 4.89E-7	6 Seal Failure by Fire Part 5.30E-10 Ru 5.30E-10 Cs 5.30E-10 Kr 7.07E-4 Prob 9.22E-11	7 Seal Failure by Fire Part 5.73E-10 Ru 5.73E-10 Cs 6.32E-9 Kr 7.64E-4 Prob 3.30E-12	15 Failure by Shear/Puncture Seal Failure by Fire Part 5.73E-10 Ru 6.32E-9 Cs 6.32E-9 Kr 7.64E-4 Prob 3.30E-15
	30-60		2 Seal Failure by Fire Part 0 Ru 0 Cs 0 Kr 0 Prob 6.19E-5	3 Seal Failure by Fire Part 0 Ru 0 Cs 0 Kr 0 Prob 2.81E-7	4 Seal Failure by Fire Part 0 Ru 0 Cs 0 Kr 0 Prob 6.99E-8	14 Failure by Shear/Puncture Seal Failure by Fire Part 0 Ru 0 Cs 0 Kr 0 Prob 6.99E-11
	No Impact	19 No Releases Prob 0.99993			18 Seal Failure by Fire Part. 0 Ru 0 Cs 0 Kr 0 Prob 5.59E-6	
		No Fire	300-350 °C	300-750 °C	300-1000 °C	300-1000 °C
		Initial and Final Temperature Associated with Cells				

Part = particulates, Ru = ruthenium, Cs = cesium, Kr = krypton, Prob = conditional probability.
Source: Jason Technologies (2001).

Table 3-3. Accident Severity Categories, Conditional Probabilities, and Release Fractions for Truck Accidents Involving TRISO Fuel.

Accident Severity Category	Conditional Probability	Release Fraction			
		Inert Gas	Cesium	Ruthenium	Particulates
1	0.99993	0.0	0.0	0.0	0.0
2	6.22×10^{-5}	0.0	0.0	0.0	0.0
3	5.59×10^{-6}	0.0	0.0	0.0	0.0
4	5.16×10^{-7}	7.50×10^{-4}	5.63×10^{-10}	5.63×10^{-10}	5.63×10^{-10}
5	6.99×10^{-8}	0.0	0.0	0.0	0.0
6	2.24×10^{-10}	3.52×10^{-3}	2.72×10^{-9}	2.64×10^{-9}	2.64×10^{-9}

Aerosolized fraction and respirable fraction = 1.0.
Source: Jason Technologies (2001), Table 5-40.

4.0 Development of Reasonably Bounding Release, Airborne Release, and Respirable Fractions

4.1 Considered Generic Reactor Design Concepts

To support this evaluation, a conceptual design was selected based on a HALEU UCO TRISO fueled prismatic HTGR using clean dry helium coolant. The inventory used for this evaluation was based on a preliminary scoping study performed by X-energy for a 20 MWt reactor operated for three effective full power years with selected cooling (decay) times of one and five years. These initial inventories are provided in Appendix A. For transport, the reactor vessel and part of the reactor coolant pressure boundary piping out to a flanged/quick opening connection will be part of the overall reactor transport package. It is also assumed that this boundary will be pressurized at a low level to prevent air in leakage into the system during transport.

To aid in modelling the radionuclide transport within the HTGR core and reactor coolant pressure boundary, the radionuclides comprising the inventory were assigned into various bins developed to address discrete release and retention characteristics associated with the TRISO kernels and coatings, TRISO compact fuel form matrix, graphite moderator prismatic blocks, and reactor coolant pressure boundary. This binning was principally based on the chemical and physical properties of the radionuclides and was informed by past work related to consequence assessment (NUREG-1435, NUREG/CR-6410; SAND2021-11073). Table 4-1 presents the classification of various radionuclides used in the binning and subsequent assignment of the release groups consistent with INL/EXT-11-24034 (INL 2012).

Table 4-1. Classes of Radionuclides of Interest for High-Temperature Gas-Cooled Reactor Transportation.

Class/Group1	Idaho National Laboratory Key Nuclides[1]	Modeled Inventory Nuclides
Noble Gases (Noble Gases)	Kr-85, Kr-88, Xe-133	Kr, Xe
Halogens (I, Br, Te, Se)	I-131, I-133, Te-132,	Br, I, Se, Te
Alkali Metals (Cs, Rb)	Cs-134, Cs-137	Cs, Rb
Alkaline Earths (Sr, Ba, Eu)	Sr-90	Ba, Eu, Ga, Gd, Sr, Tb
Noble Metals (Ag, Pd)	Ag-110m Ag-111	Ag, Ge, In, Pd, Sn
Metalloid (Sb)	Sb-125	As, Cd, Sb, Zn
Transition Metals (Mo, Ru, Rh, Tc)	Ru-103	Mo, Nb, Rh, Ru, Tc

Class/Group1	Idaho National Laboratory Key Nuclides[1]	Modeled Inventory Nuclides
Lanthanides (La, Ce)	Ce-144 La-140	Ce, La, Nd, Pm, Pr, Sm, Y, Zr
Actinides (Pu, actinides)	Pu-239	Am, Cm, Np, Pa, Pu, Th, U

I = iodine, Br = bromine, Te = tellurium, Se = selenium, Cs = cesium, Rb = rubidium, Sr = strontium, Ba = barium, Eu = europium, Ag = silver, Pd = palladium, Sb = antimony, Mo = molybdenum, Ru = ruthenium, Rh = rhodium, Tc = technetium, La = lanthanum, Ce = cerium, Kr = krypton, Xe = xenon, Nd = neodymium, Pm = promethium, Pr = praseodymium, Sm = Samarium, Y = Yttrium , Zr = zirconium, Am = americium, Cm = curium , Np= neptunium, Pa = protactinium, Pu = plutonium, Th = thorium, U = uranium.

4.2 Next Generation Nuclear Power Plant TRISO Fuel Release Fraction Modeling

A key element of an operating HTGR safety approach that is applicable to the transportation of a spent reactor is the evaluation of radionuclide releases from the core during normal plant operation. For this report a mechanistic source term was developed, following the approach and release characteristics documented in EPRI TR-1009382 *A Review of Radionuclide Release from HTGR Cores During Normal Operation* and INL/EXT-11-24034, *Scoping Analysis of Source Term and Functional Containment Attenuation Factors* and utilized to estimate the releases from the TRISO during normal operations.

4.2.1 Releases from TRISO Fuel Under Normal Operations

The release of fission products and gases during normal operation are from four major sources as follows:

- heavy metal contamination in the fuel;
- releases from particles with SiC coating defects;
- release due to incremental in-service fuel failures that occur under normal operation; and
- diffusive release through intact coatings at elevated temperatures.

Based on the current database on fission product transport through the kernel and coating layers, very little release of most fission products is expected via diffusion at the peak normal operations fuel temperatures anticipated for a HTGR and a three year fuel life (Petti, 2013). For reactors with a fuel life longer than three years, this assumption should be revisited.

For this evaluation, releases during normal operations were determined utilizing the approach and model presented in INL/EXT-11-24034. The release from TRISO is a function of the initial fuel quality, fuel performance and failure information and assumed reactor operating (outlet) temperature. This modeling assumes a prismatic core with outlet temperature in the range of 900 °C and the fuel release fractions conservatively set consistent with the 50% and 95% values contained in INL/EXT-11-24034. These values have been tabulated in Appendix B Table B-1.

4.2.2 Attenuation of Releases During Normal Operations

Attenuation factors represent the capability of an identified barrier to prevent or reduce the release of fission products. The attenuation factor estimates are also associated with normal operation and are also dependent on temperature. As noted in EPRI TR-1009382, for TRISO fueled reactors the following barriers are typically credited with an attenuation factor: (i) the fuel kernel, (ii) the particle coatings, particularly the SiC coating, (iii) the fuel element and structural graphite, and (iv) the reactor coolant pressure boundary. For some transportation accidents, a fifth barrier (associated with the transportation package) may be available but is generally not credited in this evaluation.

The attenuation factors used in this evaluation are set consistent with the 50% and 95% values contained in INL/EXT-11-24034. These values have been tabulated in Appendix B Table B-2.

4.2.3 Representative Release Fractions for Normal Operations

Utilizing the modelling approach discussed in INL/EXT-11-24034 with the input parameters provided in Appendix B a set of standardized release fractions under normal operations (mean and 95% values) were developed of each of the Grouping Classifications as presented in Table 4-1. The resulting release fractions for fission products and gases attenuated by the fuel element/core material and for those released into the pressure boundary are tabulated in Table 4-2. These release fractions when combined the individual inventories provide an estimate of the MAR within the fuel matrix and core material and pressure boundary (see Section 5.3.2).

Table 4-2. Representative Release Fractions for Radionuclide Classifications.

		Mean Release Fraction		95% Release Fraction	
Classification/ Group	Representative Radionuclides	Fission Products Release into Fuel Matrix/ Core Material	Fission Products Release into Pressure Boundary	Fission Products Release into Fuel Matrix/ Core Material	Fission Products Release into Pressure Boundary
Noble Gases[1] (Noble Gases)	Xe-133	0.00E+00	8.01E-06	0.00E+00	3.16E-05
	Kr-85	0.00E+00	8.13E-06	0.00E+00	3.25E-05
	Kr-88	0.00E+00	8.18E-06	0.00E+00	3.22E-05
Halogens (I, Br, Te, Se)	I-131	0.00E+00	8.01E-06	0.00E+00	3.17E-05
	I-133	0.00E+00	8.02E-06	0.00E+00	3.21E-05
	Te-132	0.00E+00	8.23E-06	0.00E+00	3.24E-05
Alkali Metals (Cs, Rb)	Cs-137	1.45E-04	1.64E-04	4.99E-04	5.55E-04
	Cs-134	1.47E-04	1.63E-04	5.07E-04	5.50E-04
Alkaline Earths (Sr, Ba, Eu)	Sr-90	3.35E-03	1.89E-05	9.89E-03	6.72E-05
Noble Metals (Ag, Pd)	Ag-110m	0.00E+00	8.39E-03	0.00E+00	2.52E-02
	Ag-111	0.00E+00	1.98E-03	0.00E+00	5.41E-03
Metalloid (Sb)	Sb-125	2.58E-04	1.03E-04	8.63E-04	4.47E-04
Transition Metals	Ru-103	3.29E-05	2.12E-07	1.09E-04	8.65E-07

Classification/ Group	Representative Radionuclides	Mean Release Fraction		95% Release Fraction	
		Fission Products Release into Fuel Matrix/ Core Material	Fission Products Release into Pressure Boundary	Fission Products Release into Fuel Matrix/ Core Material	Fission Products Release into Pressure Boundary
(Mo, Ru, Rh, Tc)					
Lanthanides (La, Ce)	Ce-144	3.27E-05	2.12E-07	1.09E-04	8.67E-07
	La-140	3.28E-05	2.14E-07	1.10E-04	8.73E-07
Actinides (Pu, actinides)	Pu-239	2.94E-05	1.52E-08	1.04E-04	6.62E-08

I = iodine, Br = bromine, Te = tellurium, Se = selenium, Cs = cesium, Rb = rubidium, Sr = strontium, Ba = barium, Eu = europium, Ag = silver, Pd = palladium, Sb = antimony, Mo = molybdenum, Ru = ruthenium, Rh = rhodium, Tc = technetium, La = lanthanum, Ce = cerium, Kr = krypton, Xe = xenon, Pu = plutonium.

[1] It is assumed that noble gases released into the pressure boundary are removed prior to transport of the reactor and are not included in subsequent dose analysis.

4.3 Development of Source Term Factors

This section discusses the development of the material that could be released into the environment referred as the “source term” during an accident that degrades or fails the containment system where MAR resides (e.g., TRISO particle, compact, or reactor coolant system). Radiological material released into the environment can impact a human receptor through different dose pathways. The principal radiological dose pathway is usually airborne and the dose from the inhalation typically dominates the overall dose. Radiological material that is released produces a direct exposure dose in addition to inhalation dose. Dose pathways such as submersion or ingestion are discussed in Section 5.3.

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

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

where:

- MAR = material at risk
- DR = damage ratio
- ARF = airborne release fraction
- RF = respirable fraction
- LPF = leak path factor

The five-factor formula, while traditionally developed for non-reactor nuclear facilities, is applicable to a microreactor transportation accident analysis. The following discusses the development of the individual elements making up the source term calculation as applied to the TNPP transportation accident analysis except the MAR, which is discussed in detail in Section 5.3.2.

The MAR is the material potentially subject to release due to accident stressors. For a TNPP transportation accident the MAR is generally classified by locations and includes the following:

- fission products and gases contained within TRISO particles;

- fission products and gases from heavy metal contamination and from defective or failed particles and now assumed to be held up in the compact or core structure; and
- fission products and gases from heavy metal contamination and from defective or failed particles and now assumed to be within the primary pressure boundary.

The damage ratio represents the fraction of the MAR that is impacted by the accident generated stresses. For this evaluation, the damage ratios are estimates based on the transportation accident stresses transmitted to containment of the MAR and is developed individually for the three primary locations of MAR. The damage ratio is primarily a function of the energy involved in the accident and physical phenomena that can cause release.

The airborne release fraction (ARF) represents the estimate of the total amount of a radioactive material that can be suspended in air and made available for airborne transport under an accident specific set of induced physical stresses. These fractions are developed individually for the three primary locations of MAR. The ARF is primarily a function of the energy involved in the accident, physical phenomena that can cause release, and the form of the MAR.

The respirable fraction (RF) 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-micrometre (μm) aerodynamic equivalent diameter and less. Again, for this evaluation, the RF estimates are based on the both the material forms and accident stresses and is developed individually for the three primary locations and forms of MAR.

The leak path factor (LPF) represents the attenuation (including deposition, holdup) of the airborne materials as they are transported from source to the surrounding environment where is it subjected to atmospheric dispersion. The LPF is primarily a function of the pathway that is created by the damage caused by the transportation accident to containment and containment systems.

5.0 Parametric Study of Factors that Affect Release

5.1 Purpose

The Section 5.0 provides a description and results of a parametric study addressing the modelling uncertainty for factors associated with release of radiological material for use in calculating the radiological dose consequences of an accident involving transportation of a TRISO fueled microreactor. The factors include fractions of material released from the TRISO fuel during normal operation that reside in the core, reactor structure, and cooling system (i.e., release fraction) and the fraction of material that is released during a transportation accident based on accident conditions. An example of these factors is the ARF and respirable fractions that are created (Table 3-1 refers to these as "release fractions," but these are from accident conditions rather than normal operation and are described below as the five factors used in the five-factor formula to determine the source term).

The study addresses the impact that the uncertainty of the factors described above and other key assumptions for transportation specific accidents have on calculated radiological dose consequences. The purpose of the study is to: (i) gain generalizable information about the values to use for release related factors that could be used in a dose consequence analysis for a transportation accident involving a TRISO fueled microreactor, (ii) provide insights about how the uncertainty associated with the release related values used in a dose consequence analyses impacts the estimated radiological dose, and (iii) generate insights about future analyses and tests that could be performed to address the modelling uncertainty having the greatest impact on estimating the radiological dose consequences.

Section 5.2 of this section of the report discusses the TRISO fueled microreactor transportation accident scenarios selected for investigation in the parametric study. Section 5.3 describes the radiological dose consequences methodology that will be used in the parametric study. Section 5.4 describes the baseline and sensitivity studies selected for the parametric study and the bases their selection. Section 5.5 presents the results and insights from the parametric study.

5.2 Accident Scenarios Selected for the Parametric Study

This section describes the TRISO fueled microreactor transportation accident scenarios selected for investigation in the parametric study. There are multiple methods for identifying initiating events for further analysis in risk informed approaches 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).

The kinds of hazards that could be considered in hazard analysis performed to support the identification and qualitative assessments of microreactor transportation accident include:

- fire hazard events;
- explosion events;
- kinetic energy events;
- potential energy events;

- loss of containment hazard events;
- direct radiological exposure events;
- criticality events;
- man-made external events; and
- natural phenomena hazards.

Given that the safety functions of concern during microreactor transportation with irradiated fuel are very likely confinement of radiological material, shielding, and prevention of criticality, the scenarios of interest are likely to: (i) a release of radiological material to the environment, (ii) direct radiation exposure (or an increase worker radiation exposure), or (iii) a criticality which potentially involves both direct radiation and release of radiological material. Accordingly, hazardous condition created by the hazards identified above are of interest if they can lead to one the three cited outcomes.

Insights from earlier transportation risk assessment studies can be used to inform the selection of accidents that could be important from a radiological dose consequence point of view. The NUREG-2125, "Spent Fuel Transportation Risk Assessment" (NRC 2014) and NUREG/CR-6672, "Reexamination of Spent Fuel Shipment Risk Estimates," (NRC 2000) seek to characterize the safety transporting spent fuel in qualified casks provide and present the types of truck accident events that could challenge the integrity of the shipping package. Amongst other accidents, NUREG-2125 identifies "collision with a gasoline tanker" which was judged in this study to be the accident leading to the greatest possible impact to the shipping package.

This accident concerns release of potential radiological material from the microreactor package to the environment caused by damage due to collision with a tanker carrying gasoline which could lead to a large fire (a degradation is possible but is not postulated here). There would appear to a strong possibility of significant mechanical damage to the reactor and radiological material confinement. Fire is a second physical phenomenon that can create additional release mechanisms. It can: (i) create release pathways by damaging or weakening material that might have kept the radiological material contained, (ii) 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 (iii) create a convective current that causes the material to be airborne. The significance of this accident is that it likely produces the highest dose consequences of any highway accident because it involves impact with a heavy vehicle in combination with a hot and long-lasting fire due the large quantity of flammable material. In addition to causing a release of radioactive material the accident could also severely damage installed shielding causing increases in direct radiation.

A candidate accident for the parametric study that is not addressed in the published transportation risk studies for qualified packages is a low energy event that results in release of radiological material without a highway accident involving a hard impact. This kind of accident may be applicable to TRISO fuel microreactor transportation because the reactor package likely cannot be qualified as a Type B package under 10 CFR 71. If the microreactor power plant cannot be transported as a single unit, then it will need to be dismantled and broken into segments, so that it fits into transportation containers. If this is the case, and the primary loop is divided between segments, then the piping ends will need to be closed-up to contain radiological material that may exist in the reactor or piping. Failure of containment could occur due to human error in packaging or due to a random failure of a seal, joint or connection, or a failure induced by typical vibration and limited impact forces from road surface irregularities, or

other phenomena. It is assumed that the reactor will be pressurized with a “cover gas” to minimize potential for air in-leakage. Also, if there is sufficient decay heat remaining, then the reactor and primary cooling system could pressurize to some extent creating a motive force to expel loose or semi-loose radioactive material from containment if it fails. This kind of accident may be more likely than an accident the leads to the greatest consequence such a collision with a gasoline tanker. Accordingly, this kind of low energy event makes it a good accident candidate for the parametric study because it is different from transportation accidents that have been evaluated in the past, it may be more likely than other transportation accidents, and it involves completely different physical phenomena.

A criticality event is also an accident that hypothetically could occur during transportation of a TRISO fueled microreactor which is completely different from the two accidents discussed above and could lead to significant dose consequences. However, consideration of criticality accidents is not a focus of this study and would be overly complex to address in this study. Moreover, it judged very likely that criticality events during transportation will be designed away for microreactors manufactured for commercial use.

Accordingly, the two accident types selected for the parametric study are defined below:

> Accident #1: Release of radiological material from a TRISO fueled microreactor package to the environment caused by damage due to collision with a tanker carrying gasoline and subsequent large gasoline fire.
>
> Accident #2: Release of radiological material from a TRISO fueled microreactor package to the environment caused by failure of the reactor coolant (gas) pressure boundary caused by human error or random failure of a seal, joint or connection, or other failure induced by vibration or other phenomena. It is assumed there is sufficient pressure to create motive force to expel loose or semi-loose condensed or plated-out radioactive material, if this containment fails.

5.3 Dose Consequence Analysis Approach

This section provides a description of the dose consequence analysis approach used to perform the parametric study. The major elements of the radiological dose consequence analysis are: (i) the assumed radionuclide inventory of the microreactor during transport, (ii) the MAR that could be released, (iii) the quantity of material that is released into the environment referred to as the “source term” and defined by the elements of the five-factor formula approach, and (iv) the calculation of radiological dose based on the released material. Material that is released can cause internal and external dose consequences to humans through various radiological dose pathways including direct radiation effects.

The radiological dose pathway that typically dominates the consequences of an accident in which radiological material is released from a contained environment is the dose from inhalation because it is an internal dose. However, other dose pathways such as submersion or ingestion are possible and might in some cases be important considerations. However, those dose pathways are not considered in this study; as due to the release scenarios these pathways are considered unlikely to significantly impact the calculated dose.

Additionally, radiation shielding could be lost or degraded in a transportation accident resulting in a direct radiation dose to human receptors who are near enough to the degraded shielding to receive a dose. However, investigation of the uncertainty in modeling direct radiation impact is

not the focus of this study and is it highly dependent on the design of reactor and its shielding which must address the tradeoff between weight and radiation protection. It is presumed that the reactor vessel itself will largely remain intact after a transportation accident, and so shielding is unlikely to be entirely lost. The most likely degradation is believed to be fissures that create streaming pathways that are not addressed in this study. That said, radiological material that is released has a direct radiation impact on the human body, and therefore, this radiological dose pathway is included in the dose calculations performed for a transportation accident as described below in Section 5.3.4.

Section 5.3.1 of this section of the report presents the assumed radionuclide inventory of the microreactor during transport and accompanying assumptions such as assumption about decay time. Section 5.3.2 defines the MAR that could become released and, as discussed above, defines the different forms of MAR. Section 5.3.3 discusses the quantity of material that could be released into the environment (i.e., the source term). This section does not present the five-factor formula values that are used in the base case or sensitivity studies, but rather presents a description of these terms. Section 5.3.4 describes the calculation of radiological dose based on the source term, as well as the approach used for performing these calculations.

5.3.1 Radionuclide Inventory of Microreactor During Transport

The section identifies the radionuclides of interest to the transport evaluation and the base case release fractions associated with normal operation-based releases from TRISO fuel.

Table 5-1 summarizes the results of the binning process from Section 4.1 applied to the radionuclides with greater than 0.1 Curie (Ci) of activity after a 5-year decay interval and identifies the mean value of the assumed release fraction associated with the fuel and core material and pressure boundary.

Table 5-1. Classes of Radionuclides and Representative Release Fraction for High-Temperature Gas-Cooled Reactor Transportation.

Classification/ Group	Inventory Radionuclides, >0.1 Ci After 5 Years	Representative Release Fraction, Mean Values	
		Fuel/Core Material	Pressure Boundary
Noble Gases (Kr, Xe)	Kr-85	0.00E+00	8.18E-06
Halogens (I, Br, Te, Se)	Te-125m	0.00E+00	8.23E-06
Alkali Metals (Cs, Rb)	Cs-134, Cs-135, Cs-137	1.47E-04	1.64E-04
Alkaline Earths (Sr, Ba, Eu)	Ba-137m, Eu-152, Eu-154, Eu-155, Sr-90	3.35E-03	1.89E-05
Noble Metals (Ag, Pd)	Ag-110, Ag-110m, Sn-119m, Sn-121, Sn-121m, Sn-126	0.00E+00	8.39E-03
Metalloid (Sb)	Cd-113m, Sb-125, Sb-126m	2.58E-04	1.03E-04
Transition Metals	Nb-93m, Rh-106, Ru-106, Tc-99	3.29E-05	2.12E-07

Classification/ Group	Inventory Radionuclides, >0.1 Ci After 5 Years	Representative Release Fraction, Mean Values	
		Fuel/Core Material	Pressure Boundary
(Mo, Ru, Rh, Tc)			
Lanthanides (La, Ce)	Ce-144, Pm-147, Pr-144, Pr-144m, Sm-151, Y-90, Zr-93	3.28E-05	2.14E-07
Actinides (Pu, actinides)	Am-241, Am-242, Am-242m, Am-243, Cm-242, Cm-243, Cm-244, Cm-245, Np-237, Np-239, Pa-233, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, U-236, U-237	2.94E-05	1.52E-08

Ci = Curie, Kr = krypton, Xe = xenon, I = iodine, Br= bromine, Te = tellurium, Se = selenium, Cs =cesium, Rb= rubidium, Sr = strontium, Ba = barium, Eu = europium, Ag= silver, Pd = palladium, Sb = antimony, Mo = molybdenum, Ru = ruthenium, Rh = rhodium, Tc = technetium, La = lanthanum, Ce = cerium, Pu = plutonium, Sn = tin, Nb = niobium, Pm = promethium, Pr = praseodymium, Sm = samarium, Y = yttrium, Zr= zirconium, Am = americium; Cm = curium, Np = neptunium, Pa = protactinium, U =uranium.

5.3.2 Material at Risk

The section provides the MAR used for the base case analysis. As discussed in Section 4.2.3 of this report, this radioactive material consists of different forms and exists in different locations of the reactor. Development of the MAR is based upon the release of fission products and gases from the individual TRISO particles during normal reactor operations. As discussed in Section 4.3 of this report, MAR has been developed for three locations within the reactor vessel:

- 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; and
- fission products and gases that have diffused from the TRISO fuel and have condensed or plated-out in the reactor coolant pressure boundary. It is conservatively assumed than all of this MAR is located within the transportation package envelope.

Based on the discussion in Section 4.2.3 the MAR is calculated from the radionuclide inventory provided in Appendix A and release fractions provided in Table 4-2. Table 5-2 provides the MAR for the base case 5-year cooling and mean release fraction. The MAR for all analyzed scenarios is contained in Section 4.3.

Table 5-2. Mean Value Material at Risk, 5-year Cooling.

Radionuclide	Classification	Material At Risk in TRISO fuel (Ci)	Material At Risk in Fuel Element/Core Material (Ci)	Material At Risk in Pressure Boundary (Ci)
Ag-110	Noble Metals (Ag, Pd)	1.00E-01	0.00E+00	8.47E-04
Ag-110m	Noble Metals (Ag, Pd)	7.36E+00	0.00E+00	6.23E-02
Am-241	Actinides (Pu, actinides)	4.80E+02	1.41E-02	7.29E-06
Am-242	Actinides (Pu, actinides)	3.84E+00	1.13E-04	5.82E-08

Radionuclide	Classification	Material At Risk in TRISO fuel (Ci)	Material At Risk in Fuel Element/Core Material (Ci)	Material At Risk in Pressure Boundary (Ci)
Am-242m	Actinides (Pu, actinides)	3.86E+00	1.14E-04	5.86E-08
Am-243	Actinides (Pu, actinides)	8.88E+00	2.61E-04	1.35E-07
Ba-137m	Alkaline Earths (Sr, Ba, Eu)	7.77E+04	2.61E+02	1.47E+00
Cd-113m	Metalloid (Sb)	1.29E-01	3.34E-05	1.34E-05
Ce-144	Lanthanides (La, Ce)	9.76E+03	3.19E-01	2.07E-03
Cm-242	Actinides (Pu, actinides)	1.38E+01	4.06E-04	2.09E-07
Cm-243	Actinides (Pu, actinides)	7.42E+00	2.18E-04	1.13E-07
Cm-244	Actinides (Pu, actinides)	1.08E+03	3.19E-02	1.64E-05
Cm-245	Actinides (Pu, actinides)	1.41E-01	4.14E-06	2.14E-09
Cs-134	Alkali Metals (Cs, Rb)	2.04E+04	3.00E+00	3.33E+00
Cs-135	Alkali Metals (Cs, Rb)	6.25E-01	9.21E-05	1.02E-04
Cs-137	Alkali Metals (Cs, Rb)	8.23E+04	1.20E+01	1.35E+01
Eu-152	Alkaline Earths (Sr, Ba, Eu)	4.23E+00	1.42E-02	8.02E-05
Eu-154	Alkaline Earths (Sr, Ba, Eu)	2.22E+03	7.47E+00	4.21E-02
Eu-155	Alkaline Earths (Sr, Ba, Eu)	9.59E+02	3.22E+00	1.82E-02
Kr-85	Noble Gases (2)	7.04E+03	0.00E+00	5.75E-02
Nb-93m	Transition Metals (Mo, Ru, Rh, Tc)	4.54E-01	1.49E-05	9.63E-08
Np-237	Actinides (Pu, actinides)	2.42E-01	7.11E-06	3.67E-09
Np-239	Actinides (Pu, actinides)	8.89E+00	2.61E-04	1.35E-07
Pa-233	Actinides (Pu, actinides)	2.43E-01	7.15E-06	3.69E-09
Pm-147	Lanthanides (La, Ce)	4.16E+04	1.37E+00	8.88E-03
Pr-144	Lanthanides (La, Ce)	9.75E+03	3.20E-01	2.08E-03
Pr-144m	Lanthanides (La, Ce)	9.34E+01	3.07E-03	1.99E-05
Pu-238	Actinides (Pu, actinides)	2.12E+03	6.23E-02	3.22E-05
Pu-239	Actinides (Pu, actinides)	1.21E+02	3.55E-03	1.83E-06
Pu-240	Actinides (Pu, actinides)	1.80E+02	5.29E-03	2.73E-06
Pu-241	Actinides (Pu, actinides)	4.63E+04	1.36E+00	7.02E-04
Pu-242	Actinides (Pu, actinides)	1.11E+00	3.26E-05	1.68E-08
Rh-106	Transition Metals (Mo, Ru, Rh, Tc)	7.01E+03	2.30E-01	1.49E-03
Ru-106	Transition Metals (Mo, Ru, Rh, Tc)	7.04E+03	2.31E-01	1.49E-03
Sb-125	Metalloid (Sb)	1.40E+03	3.63E-01	1.45E-01
Sb-126m	Metalloid (Sb)	1.56E-01	4.03E-05	1.61E-05
Sm-151	Lanthanides (La, Ce)	3.34E+02	1.10E-02	7.14E-05
Sn-119m	Noble Metals (Ag, Pd)	1.29E+00	0.00E+00	1.09E-02
Sn-121	Noble Metals (Ag, Pd)	9.45E+00	0.00E+00	8.00E-02
Sn-121m	Noble Metals (Ag, Pd)	1.22E+01	0.00E+00	1.03E-01
Sn-126	Noble Metals (Ag, Pd)	1.55E-01	0.00E+00	1.31E-03
Sr-90	Alkaline Earths (Sr, Ba, Eu)	6.97E+04	2.34E+02	1.32E+00

Radionuclide	Classification	Material At Risk in TRISO fuel (Ci)	Material At Risk in Fuel Element/Core Material (Ci)	Material At Risk in Pressure Boundary (Ci)
Tc-99	Transition Metals (Mo, Ru, Rh, Tc)	1.26E+01	4.13E-04	2.66E-06
Te-125m	Halogens (I, Br, Te, Se)	3.44E+02	0.00E+00	2.83E-03
U-236	Actinides (Pu, actinides)	3.62E-01	1.06E-05	5.50E-09
U-237	Actinides (Pu, actinides)	1.13E+00	3.33E-05	1.72E-08
Y-90	Lanthanides (La, Ce)	7.03E+04	2.31E+00	1.50E-02
Zr-93	Lanthanides (La, Ce)	1.78E+00	5.85E-05	3.81E-07

Ci = Curie, TRISO = Tri-structural Isotropic particle fuel, Am = americium; Ba = barium, Cd = cadmium, Ce = cerium, Cm = curium, Cs = cesium, Eu = europium, Kr = krypton, Nb = niobium, Np = neptunium, Pa = protactinium, Pm = promethium, Pr = praseodymium, Pu = plutonium, Ru = ruthenium, Sb = antimony, Sm = samarium, Sn = tin, Sr = strontium, Tc = technetium, Te = tellurium, U = uranium, Y = yttrium, Zr= zirconium.

5.3.3 Source Term

The section discusses development of the source term components. The Source Term is a function of the MAR (Section 5.3.2), accident-related stresses causing release of the material, and attenuation of the material prior to entering the surrounding environment.

For airborne releases, the source terms will be estimated using the following five-factor formula approach equation (DOE 2013) as described in Section 4.3 of this report:

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

As discussed in the next section, the DR, ARF, RF values are estimated based on the accident release mechanism, the physical stresses induced by the accident, and the form and physical properties of the material containing the radiological material and the radiological material itself. The remainder of this section discusses the development of the DR, ARF, RF and LPF for the specific accidents selected for analysis.

5.3.3.1 Accident Scenario 1 – Tanker Collision and Fire

This scenario addresses the release of radiological material from a TRISO fueled microreactor package to the environment caused by damage due to a collision with a tanker carrying gasoline and subsequent large gasoline fire. This accident is evaluated in two parts. First, the impact of the tanker into the TNPP and resultant impact stresses and releases and then secondly, the fire results in thermal stresses with additional material being released. The stresses from this accident are developed independently as the stresses will depend on the location and form of the MAR.

MAR contained in the pressure boundary. It is assumed all MAR in the pressure boundary is in particulate form. This MAR is expected to occur in one of three forms: (i) incorporated within the oxide layer, (ii) plated out (chemisorbed) on metallic surfaces, and/or (iii) in the form of contaminated dust particles loosely bound or tightly attached to the surfaces.

Impact Stresses.

Damage Ratio: It is assumed that all the MAR is subjected to the impact stresses, DR=1. This is a bounding assumption.

Airborne Release Fraction: The ARF is assumed to be consistent with a shock-vibration impact (loose surface contamination) with an ARF= 1E-03 (NUREG/CR-6410 Section 3.3.4.12a).

Respirable Fraction: The RF is assumed to be consistent with a shock-vibration impact (loose surface contamination) with an RF= 0.3 (NUREG/CR-6410 Section 3.3.4.12a).

The aggregate of the ARF and RF is expected to be conservative based on the form of the MAR including material which includes dust as well as both an oxide layer and plateout (chemisorbed) material.

Leak Path Factor: The LPF was set to be 0.5. This reflects the distribution of the material within the primary pressure boundary with respect to an open pathway to the environment, it is not plausible that all the MAR would be located near the opening. In addition, there would be insufficient transport forces to overcome adhesion of smaller particles and settling of larger particles that contain the MAR.

Thermal Stresses.

Damage Ratio: It assumed that all the MAR is subjected to the thermal stresses, DR=1. This is a bounding assumption.

Airborne Release Fraction: The ARF is assumed to be consistent with burning of nonreactive compounds and the release under thermal stress for contaminated noncombustible solids; ARF= 6E-03 (NUREG/CR-6410 Section 3.3.2.10).

Respirable Fraction: The RF is assumed to be consistent with burning of nonreactive compounds and the release under thermal stress for contaminated noncombustible solids; RF= 1E-02 (NUREG/CR-6410 Section 3.3.2.10).

Leak Path Factor: The LPF was set to be 0.5. This reflects the distribution of the material within the primary pressure boundary with respect to an open pathway to the environment, it is not plausible that all the MAR would be located near the opening. In addition, there would be minimal motive force as thermal entrainment would be limited due to leak path configuration, it is not assumed that a significant chimney effect could be established.

MAR contained in the Fuel/Core Matrix. It is assumed all MAR in the Fuel/core matrix is in particulate form. This MAR is expected primarily be fission metals as fission gases are not significantly retained.

Impact Stresses.

Damage Ratio: It assumed that 10% of the MAR is subjected to the impact stresses, DR=0.1. This is judged to be a conservative assumption given the structure of the core and the need to transmit stresses through several barriers prior to reaching the core.

Airborne Release Fraction: As discussed above, the ARF is assumed to be consistent with a shock-vibration impact (loose surface contamination) with an ARF= 1E-03 (DNUREG/CR-6410 Section 3.3.4.12a).

Respirable Fraction: The RF is also assumed to be consistent with a shock-vibration impact (loose surface contamination) with an RF= 0.3 (NUREG/CR-6410 Section 3.3.4.12a).

Leak Path Factor: The LPF was set to be 0.1. This reflects material impeding the pathway to the environment. It is judged that the actual damage will be to minor failures in the reactor coolant pressure boundary (vessel, piping, and seals) and some fracturing of the core components, so this LPF value is considered to be conservative.

Thermal Stresses.

Damage Ratio: It assumed that 10% of the MAR is subjected to the thermal stresses, DR=0.1. This is judged to be a conservative assumption given the structure of the core and the need to transmit stresses through several barriers prior to reaching the core.

Airborne Release Fraction: The ARF is assumed to be consistent with burning of nonreactive compounds and the release under thermal stress for contaminated noncombustible solids; ARF= 6E-03 (NUREG/CR-6410 Section 3.3.2.10).

Respirable Fraction: The RF is assumed to be consistent with burning of nonreactive compounds and the release under thermal stress for contaminated noncombustible solids; RF= 1E-02 (NUREG/CR-6410 Section 3.3.2.10).

Leak Path Factor: The LPF was set to be 0.1. This reflects material impeding the pathway to the environment. It is judged that the actual damage will be to minor failures in the reactor coolant pressure boundary (vessel, piping, and seals) and some fracturing of the core components, so this LPF value is considered to be conservative.

MAR contained in the TRISO. It is assumed the MAR in the TRISO includes both particulate and gaseous forms.

For the MAR in gaseous form, it is assumed to be completely released with no attenuation on the initial impact failure of the fuel, ARF=1, RF=1, LPF=1.

Impact Stresses.

Damage Ratio: It assumed that 0.1% of the MAR is subjected to the impact stresses, DR=0.001. This is consistent with the values used in DOE (2002) and DOE (2008) for TRISO fuel in cask impacts with an unyielding surface ranging from 60-90 mph (Section 2.3) and is judged to be a conservative assumption given the TRISO particles are embedded in a graphite matrix and the need to transmit stresses through several barriers prior to reaching the TRISO.

Airborne Release Fraction: As discussed above, the ARF for particulates is assumed to be consistent with a shock-vibration impact (loose surface contamination) with an ARF= 1E-03 (NUREG/CR-6410 Section 3.3.4.12a).

Respirable Fraction: The RF for particulates is also assumed to be consistent with a shock-vibration impact (loose surface contamination) with an RF= 0.3 (NUREG/CR-6410 Section 3.3.4.12a).

Leak Path Factor: The LPF for particulates is set to be 0.05. This reflects that portions of the core structure must fail to impact the fuel elements resulting in additional material impeding the pathway to the environment compared to previous LPFs discussed for this accident.

Thermal Stresses.

Damage Ratio: It assumed that 0.1% of the MAR is subjected to the impact stresses, DR=0.001. This is consistent with the values used in DOE (2002) and DOE (2008) for TRISO fuel in cask impacts with an unyielding surface ranging from 60-90 mph (Section 2.3) and is judged to be a conservative assumption given the TRISO particles are embedded in a graphite matrix and the need to transmit stresses through several barriers prior to reaching the TRISO. No additional fuel failures are expected due to the temperature effects of the fire.

Airborne Release Fraction: The ARF is assumed to be consistent with burning of nonreactive compounds and the release under thermal stress for contaminated noncombustible solids; ARF= 6E-03 (NUREG/CR-6410 Section 3.3.2.10).

Respirable Fraction: The RF is assumed to be consistent with burning of nonreactive compounds and the release under thermal stress for contaminated noncombustible solids; RF= 1E-02 (NUREG/CR-6410 Section 3.3.2.10).

The LPF for particulates is set to be 0.05. This reflects that the portions of the core structure must fail to impact the fuel elements resulting in additional material impeding the pathway to the environs compared to previous LPFs for this accident.

5.3.3.2 Accident Scenario 2 – Containment Failure (Random)

This scenario addresses the release of radiological material from a TRISO fueled microreactor package to the environment caused by failure of the containment (reactor coolant pressure boundary) of the microreactor. This accident is assumed to only involve MAR located in the reactor pressure boundary. It is assumed that the impact of normal road travel (vibration, minor impacts) will be included in the design specifications and as such do not result in the significant generation of additional loose material or failure of TRISO particles.

MAR contained in the pressure boundary. It is assumed all MAR in the pressure boundary is in particulate form. This MAR is expected to occur in one of three forms: (i) incorporated within the oxide layer, (ii) plated out (chemisorbed) on metallic surfaces, and/or (iii) in the form of contaminated dust particles loosely bound or tightly attached to the surfaces.

Impact Stresses.

Damage Ratio: It assumed that only a portion of the MAR is subjected to the stresses associated with the venting, DR=0.2. This is consistent with Japanese testing of liftoff fractions of 10-20% for HTGR pipe rupture testing at high pressure (EPRI 1003387). This is judged to be a conservative assumption given the low pressure and small equivalent leak opening for this event.

Airborne Release Fraction: The ARF is assumed to be consistent with a low-pressure release for a container of powder, ARF= 2E-03 (NUREG/CR-6410 Section 3.3.1.11).

Respirable Fraction: The RF is assumed to be consistent with a low-pressure release for a container of powder, RF= 0.4 (NUREG/CR-6410 Section 3.3.1.11).

The aggregate of the ARF and RF is expected to be conservative based on the form of the MAR including material which includes dust as well as both an oxide layer and plate-out (chemisorbed) material.

Leak Path Factor: The LPF was set to be 0.1. This value is consistent with NUREG/CR-6672 and NUREG-1864 for mechanically closed canisters and is judged to be conservative given the relationship of the MAR to a specified opening.

5.3.4 Calculation of Transportation Accident Radiological Dose

This section discusses the approach for transportation accident radiological dose consequences developing based on the source term as discussed in Section 5.3.3 (i.e., any radiological material that is released from its containment). The dose calculations are based on the methodology of the International Atomic Energy Agency (IAEA) Q system described in IAEA SSG-26 (IAEA 2014) and is the basis of the A_1 and A_2 values included in 10 CFR Part 71. The specific methodology for calculating radiological dose consequences for human receptors is presented in Appendix I of SSG-26.

The IAEA Q system is a way to define quantity limits for material in a Type A package as well as applications in transport regulations and establishing leakage limits in Type B(U), Type B(M), or Type C package activity leakage limits, low specific activity, and excepted package contents limits, and contents limits for low dispersible radioactive material and special form and non-special form radioactive materials (IAEA 2014). The IAEA Q system methodology was chosen for the dose calculations for this activity based on its wide acceptance and adoption both within the U.S. transportation regulations as well as the international community. The Q system includes exposure pathways for someone in the vicinity of a type A package involved in a severe transportation accident. The pathways used to determine a series of Q values are external photon dose, external beta dose, inhalation dose, skin, and ingestion dose due to contamination transfer and submersion dose. For this effort ingestion dose and submersion dose will not be included; ingestion will not be included (consistent with IAEA SSG-26 findings that explicitly mention that consideration of the ingestion pathway is unnecessary) and submersion dose will not be included because the assumption is made that the exposure will take place outside and this will limit the time that a receptor might stand in a gaseous cloud of radionuclides. The Q value analyses do not consider the content limits for special form alpha and neutron emitters or tritium. The A_2 values are defined by the lowest of the Q values (for the exposure pathways) or the A_1 value if it is lower than the Q values. The Q values are derived based on the following radiological criteria in IAEA SSG-26 (IAEA 2014):

- The effective dose or committed effective dose to a person exposed in the vicinity of a transport package following an accident should not exceed a reference dose of 50 millisievert (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 sievert (Sv), or in the special case of the lens of the eye, 0.15 Sv.

- A person is unlikely to remain at 1 meter (m) from the damaged package for more than 30 minutes (Appendix I, page 273, IAEA 2014).

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

The following discussions describe how the radiological dose calculations are performed for the different radiological dose pathways:

- external dose due to photons;
- external dose due to beta;
- inhalation dose;
- skin contamination beta dose; and
- ingestion and submersion are excluded.

5.3.4.1 External Dose Due to Photons

The external dose due to photons is determined by evaluating the external radiation dose due to gamma or X rays to the whole body of a person standing 1 m from the edge of the unshielded radioactive material. The external dose coefficients used in this report are from SSG-26 (IAEA 2014). This calculation does not account for dispersion so it will likely be overly conservative because it assumes that the receptor is 1 m away from any released material. The source term developed in Section 5.3.3 was multiplied by 100 so that the amount of material used to calculate external dose was more reflective of the total material released rather than the amount of material that is respirable.

5.3.4.2 External Dose Due to Beta Radiation

The external dose due to β radiation is evaluated based on the potential for beta dose to the receptors skin. The IAEA SSG-26 methodology is for beta emitters that are unshielded but includes a concept of residual shielding for β emitters which has been retained in this dosimetry analysis. The previous β emitter shielding in the Q system was associated with the materials such as the β window protector, package debris, etc. and was assumed to be a very conservative shielding factor of three for β emitters of maximum energy (≥2 Mega electron-volt [MeV]) (IAEA 2014). The IAEA SSG-26 methodology and associated dose coefficients used in this analysis extended this shielding methodology to include a range of shielding factors depending on the β energy based on an absorber of approximately 150 milligram per square centimeter (mg/cm^2) thickness. In the case of annihilation radiation, this has not been included in the evaluation of β 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 β particles.

The dose rate coefficients used in this report are from SSG-26 (IAEA 2014). The use of the dose coefficients for external dose due to β radiation are for a person standing 1 m away from the released contamination. This calculation does not account for dispersion so it will likely be overly conservative because it assumes that the receptor is 1 m away from any released material. The source term developed in Section 5.3.3 was multiplied by 100 so that the amount

of material used to calculate external dose was more reflective of the total material released rather than the amount of material that is respirable.

5.3.4.3 Inhalation Dose

The inhalation dose is calculated using the effective dose coefficient for inhalation (sievert per becquerel [Sv/Bq]) listed in the Appendix I of IAEA SSG-26 (IAEA 2014). The human uptake value of 10^{-3} 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 10^{-3} was derived based on work related to conservative dispersion and human uptake assumptions for a downwind distance of 100 m. Extrapolation of these models to shorter distances is unreliable, but IAEA SSG-26 estimates that uptake values at 10 m would increase by a factor of about 30 compared to those at 100 m which would put uptake factors in the range of 10^{-4} to 10^{-3}. For the purposes of this dose evaluation uptake factors for the source term calculated in Section 5.3.3 of this report will be assumed to be 10^{-3} for a person standing approximately 10 m from the release point. This uptake value represents the amount of material taken up into a human receptor following a release and is separate from the estimate of what material was released as calculated in Sections 5.3.3 of this report. Inhalation doses for this effort were calculated using the inhalation dose coefficients found in Appendix I of IAEA SSG-26 (IAEA 2014).

5.3.4.4 Skin Contamination Dose

The skin contamination dose from β emitters is estimated for a person that has been contaminated with non-special form radioactive materials from the release. For this dose assessment, the dispersed radionuclides (source term) will be evaluated by the criteria set in Section 5.3; this is a deviation from the methodology of IAEA SSG-26 which has a set assumption for amount of material released from the package. The IAEA SSG-26 assumptions are related to ungloved work with debris leading to 10% of radioactive material released getting on the hands and remaining there for five hours. The skin contamination dose is based on the source term calculated in Section 5.3.3 which is a respirable release fraction; while the actual amount of material release is higher than the RF it is also unlikely that a worker would be handling debris around this accident and so it is assumed that the 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 (sievert per terabecquerel second square meter [Sv TBq^{-1} s^{-1} m^2]) found in Appendix I of IAEA SSG-26 (IAEA 2014).

5.3.4.5 Ingestion and Submersion Dose - Excluded

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 explicitly mention consideration of the ingestion pathway is unnecessary. Internal dose via the inhalation pathway will normally be limiting for internal contamination for both β and α emitters under the Q system. Submersion dose is not included because the assumption is made that the exposure will take place outside with high potential for effective dilution and conditions that limit the time that a receptor might stand in a gaseous cloud of radionuclides. Submersion dose is considered in IAEA SSG-26 only for gaseous radionuclides that do not become incorporated into the body. These include certain isotopes of argon, krypton, xenon, and radon. Only 3 radionuclides identified in Section 5.3.1 and 5.3.2 would be excluded: Kr-85, Xe-131m, and Xe-133.

5.3.4.6 Radionuclides Not Included

The inventory of radionuclides that are included in the consequence assessments are defined in Section 5.3.2 and are limited to those radionuclides that have the most impact on the radiological dose consequence calculated from a TRISO fueled transportation accident The radionuclides that were provided in the source term but that do not have corresponding dose conversion factors from SSG-26 are silver-110 (Ag-110), americium-242 (Am-242), barium-137m (Ba-137m), hydrogen-3 (H-3), praseodymium-144 (Pr-144), praseodymium- 144m (Pr-144m), rhodium-106 (Rh-106), antimony-126m (Sb-126m), tin-121 (Sn-121), and uranium-237 (U-237). All of these radionuclides except H-3 are part of decay chains captured in the radionuclide source term and are generally short-lived. The dose contributions from these radionuclides are captured in the dose conversion factors of their parent radionuclides which are included in this analysis. The H-3 is not included but will have no external dose contribution due to its decay characteristics.

5.4 Baseline and Sensitivity Studies Selected for the Study

This section discusses the baseline and sensitivity studies selected for the parametric study and the bases for that selection. The basis for selecting the two different TRISO-fueled transportation accidents that will be evaluated in the parametric study is described in Section 5.3 above. This section identifies the assumptions that are made for those two accidents in baseline and sensitivity studies for the dose consequence calculations. The baseline assumptions reflect the research's team judgment about the most reasonable assumptions to make. This section also defines sensitivity studies that were performed and for each sensitivity study and how the assumptions changed.

The assumptions primarily concern: (i) how long the reactor core has decayed since shutdown, (ii) fractions of material released from the TRISO fuel during normal operations that reside in the core, reactor structure, and cooling system (i.e., release fraction), and (iii) the DRs, ARFs, RFs, and LPFs that dictate the source term.

5.4.1 Baseline Accident Scenarios and Assumptions

This section describes the assumptions that are made for two accidents selected for the parametric study used in baseline case for the dose consequence calculations. It provides a narrative description of the assumptions for each of the two accident scenarios and provides the factors, fractions and ratio assumed. For both accidents, it is assumed:

1. The radionuclide inventory is representative of 20 MWt microreactor that has operated for three effective full power years.
2. The microreactor core is assumed to be a prismatic design and therefore contains graphite core compact that that has captured radionuclides during normal operation.
3. It is assumed that there has been five years of radionuclide decay of reactor core.
4. It is assumed that the packaging of the microreactor (or the reactor vessel portion of the microreactor) will primarily consist of the reactor vessel itself, limited shielding, and limited structural reinforcement due the weight restrictions for transporting a microreactor by road.
5. It is assumed that some disassembly of the microreactor (e.g., separation of the reactor vessel and reactor coolant system piping from the heat exchanger and power conversion equipment) is needed and creates the need for temporary closure devices.

6. It is assumed that the portion of reactor coolant pressure boundary transported contains all the condensed or plated-out radioactive material, e.g., that released fission product and condensed gases in this system has not been removed before transport.
7. It is assumed that a radioactive material cleanup system and/or the resulting radioactive waste material is not transported with the microreactor.
8. It is assumed that the possibility of a criticality event as a result of a transportation accident has been designed away.

One of the selected accidents for the parametric study is collision with a gasoline tanker and subsequent fire. The other accident is loss of containment caused by error or random failure. Specific assumptions made for these accidents are:

1. It was assumed that a collision with gasoline tanker and subsequent fire would lead to the greatest possible dose consequences.
2. It is assumed the tanker collision will cause significant mechanical damage to the microreactor package given impact with a very heavy moving tanker.
3. It is assumed that the tanker holds gasoline and the collision leads to a large fire that engulfs the microreactor package for greater than 30 minutes. No explosion or deflagration is assumed.
4. It is assumed that the heat of the fire is not hot enough to challenge the integrity of intact TRISO fuel after the heavy impact that damages a small fraction of the kernels.
5. It is assumed that the reactor will be pressurized with a “cover gas” to minimize potential for air in-leakage and there could be pressure build-up from decay heat.
6. For the loss of containment accident, it is assumed that the reactor coolant pressure boundary is pressurized at a low level and that failure of containment occurs because of human errors or failures of the containment surfaces of a seal, joint or connection occurred randomly or because of transport conditions such as road vibration. It is also assumed that the pressure creates a motive force to expel loose or semi-loose radioactive material.
7. The dose consequences of direct radiation exposure of released material is addressed, but the dose consequence of direct radiation exposure from material that has not been released (including activated material) is not addressed though reactor shielding could be degraded in the transportation accident. The focus of this study is on release fractions, and in any event, the shielding damage would highly design specific and difficult to generalize.
8. It is assumed for the baseline case that the release fractions from normal operations for material residing in the core, reactor structure, and coolant system and the source term factors described in Section 5.3.3.1 represent best judgment but conservative estimates.
9. It is assumed that using a standard transportation accident consequence analysis approach based on guidance from IAEA SSG-26 provides reasonable and comparable results without refining the approach for specific considerations.

Table 5-3 provides a summary of fractions, ratios, and factors assumed for the two baseline accident scenarios evaluated in the parametric study.

Table 5-3. Summary of Baseline Inputs.

	Normal Operations Release Fraction	Source Term Factors		
Release from the TRISO Particle Fuel Decayed for Five Years				
	Percentile of distribution	**Damage Ratio**	**ARF × RF[1]**	**LPF**
Baseline Accident #1 Tanker collision	Mean	1E-3 (FP/gases)	3E-4 impact 6E-5 thermal (FP)[2]	0.05 (FP)[2]
Baseline Accident #2 Loss of containment	Mean	0	-	-
Release from the Compact and Other Core Structures				
Baseline Accident #1 Tanker collision	Mean	1E-01 (FP)	3E-4 impact 6E-5 thermal (FP)	0.1 (FP)
Baseline Accident #2 Loss of containment	Mean	0	-	-
Release Form Condensed or Plated-out Material in the Reactor Coolant Boundary				
Baseline Accident #1 Tanker collision	Mean	1 (FP)	3E-4 impact 6E-5 thermal (FP)	0.5 (FP)
Baseline Accident #2 Loss of containment	Mean	0.2 (FP)	8E-4 (FP)	0.1 (FP)

TRISO = Tri-structural Isotropic particle fuel, ARF = airborne release fraction, RF = respirable fraction, FP= fission product, LPF = leak path factor.
[1] The ARF and RF are combined because the submersion and ingestion radiological dose pathways which do not require a RF are negligible contributors to dose and are not calculated.
[2] ARF, RF and LPF are 1 for gases.

5.4.2 Sensitivity Studies and Assumptions

This section describes the sensitivity cases that were performed for the parametric study and defines the changes in the input values from the values used for the two baseline accident scenarios for each sensitivity case. The purpose of the first sensitivity case is to show how an increase in the decay time before the microreactor transport impacts the estimated radiological dose consequences for the selected accident scenarios. The purpose of the second sensitivity case is to show how an increase in the fraction of material released from the TRISO fuel during normal operation that reside in the core, reactor structure, and cooling system (i.e., release fraction) impacts the radiological dose consequences for the selected accident scenarios. The purpose of the third sensitivity case is to show how an increase in the DR, ARF, RF, and LPF values impact the radiological dose consequences for the selected accident scenarios. These source term factors dictate how much of the MAR becomes a source term based on the physical phenomena

Table 5-4 provide a summary of fractions, ratios, and factors assumed for the two baseline accident scenarios evaluated in Sensitivity Case 1. This sensitivity case was performed to show

how much the potential dose from a transportation accident increases by decreasing the delay time before transportation of the microreactor with its irradiated fuel from two years to one year.

Table 5-4. Summary of Sensitivity Case 1 Inputs.

	Normal Operations Release Fraction	Source Term Factors		
Release from the TRISO Fuel Decayed for One Year				
	Percentile of distribution	**Damage Ratio**	**ARF × RF**	**LPF**
Accident #1 Tanker collision	Mean	1E-3 (FP/gases)	3E-4 impact 6E-5 thermal (FP)[2]	0.05 (FP)[2]
Baseline Accident #2 Loss of containment	Mean	0	-	-
Release from the Compact and Other Core Structures				
Accident #1 Tanker Collison	Mean	1E-01 (FP)	3E-4 impact 6E-5 thermal (FP)	0.1 (FP)
Baseline Accident #2 Loss of containment	Mean	0	-	-
Release From Condensed or Plated-out Material in the Reactor Coolant Boundary				
Accident #1 Tanker Collison	Mean	1 (FP)	3E-4 impact 6E-5 thermal (FP)	0.5 (FP)
Accident #2 Loss of containment	Mean	0.2 (FP)	8E-4 (FP)	0.1 (FP)

TRISO = Tri-structural Isotropic particle fuel, ARF = airborne release fraction, RF = respirable fraction, FP= fission product, LPF = leak path factor.

[1] The ARF and RF are combined because the submersion and ingestion radiological dose pathways which do not require a RF are negligible contributors to dose and are not calculated.

[2] ARF, RF and LPF are 1 for gases.

Table 5-5 provide a summary of fractions, ratios, and factors assumed for the two baseline accident scenarios evaluated in Sensitivity Case 2. This sensitivity case was performed to show how an increase in the estimated fractions of material released from the TRISO fuel during normal operation that reside in the core, reactor structure, and cooling system impacts the radiological dose consequences for the selected accident scenarios. The radiological dose consequences from each form of the MAR are provided in the results tables (i.e., TRISO fuel, core, compact and other core structures, and condensed or plated-out material in the reactor coolant boundary). As described in Section 2.0, release fraction from normal operation estimates are made as a percentile of the estimated probability distribution.

Table 5-5. Summary of Sensitivity Case 2 Inputs.

	Normal Operations Release Fraction	Source Term Factors		
Release from the TRISO Fuel Decayed for Five Years				
	Percentile of distribution	**Damage Ratio**	**ARF × RF**	**LPF**
Accident #1 Tanker collision	95%	1E-3 (FP/gases)	3E-4 impact 6E-5 thermal (FP)[2]	0.05 (FP)[2]
Accident #2 Loss of containment	95%	0	-	-
Release from the Compact and Other Core Structures				
Accident #1 Tanker collision	95%	1E-01 (FP)	3E-4 impact 6E-5 thermal (FP)	0.1 (FP)
Accident #2 Loss of containment	95%	0	-	-
Release Form Condensed or Plated-out Material in the Reactor Coolant Boundary				
Accident #1 Tanker collision	95%	1 (FP)	3E-4 impact 6E-5 thermal (FP)	0.5 (FP)
Accident #2 Loss of containment	95%	0.2 (FP)	8E-4 (FP)	0.1 (FP)

TRISO = Tri-structural Isotropic particle fuel, ARF = airborne release fraction, RF = respirable fraction, FP= fission product, LPF = leak path factor.

[1] The ARF and RF are combined because the submersion and ingestion radiological dose pathways which do not require a RF are negligible contributors to dose and are not calculated.

[2] ARF, RF and LPF are 1 for gases.

Table 5-6 provides a summary of fractions, ratios, and factors assumed for the two baseline accident scenarios evaluated in sensitivity Case 3 and 3b. This sensitivity cases are performed to show the impact that the assumptions about the estimated fractions of MAR released during the accidents have on the radiological dose consequences (i.e., uncertainty in the source term factors). Sensitivity Case 3a assumes that aggregate product of the Source Term Factors increases by an order of magnitude, while Case 3b assumes it is decreased by an order of magnitude. The increase in source term was does not to apply to gases (as the LPF, AR, RF are set to 1), because the uncertainty was assumed to be related to the ARF, RF, and LPF and not to the DR.

Table 5-6. Summary of Sensitivity Case 3a and 3b Inputs.

	Release Fraction from Normal Operations	Source Term Factors			
Release From the TRISO Fuel Decayed for Five Years					
	Percentile of distribution	**Damage Ratio**	**ARF × RF[1]**	**LPF**	**Sensitivity Cases**
Accident #1 Tanker collision	Mean	1E-03 (FP/gases)	3E-4 impact 6E-5 thermal (FP)[2]	.05 (FP)[2]	Case 3a: × 10 Case 3b: × 0.1
Accident #2 Loss of containment	Mean	0	-	-	Case 3a: × 10 Case 3b: × 0.1
Release from the Compact and Other Core Structures					
Accident #1 Tanker Collison	Mean	0.1 (FP)	3E-4 impact 6E-5 thermal (FP)	0.1 (FP)	Case 3a: × 10 Case 3b: × 0.1
Accident #2 Loss of containment	Mean	0	-	-	Case 3a: × 10 Case 3b: × 0.1
Release Form Condensed or Plated-out Material in the Reactor Coolant Boundary					
Accident #1 Tanker collision	Mean	1 (FP)	3E-4 impact 6E-5 thermal (FP)	0.2 (FP)	Case 3a: × 10 Case 3b: × 0.1
Accident #2 Loss of containment	Mean	0.2 (FP)	8E-4 (FP)	0.1 (FP)	Case 3a: × 10 Case 3b: × 0.1

TRISO = Tri-structural Isotropic particle fuel, ARF = airborne release fraction, RF = respirable fraction, FP= fission product, LPF = leak path factor.

[1] The ARF and RF are combined because the submersion and ingestion radiological dose pathways which do not require a RF are negligible contributors to dose and are not calculated.

[2] ARF, RF and LPF are 1 for gases.

Table 5-7 provides a summary of fractions, ratios, and factors assumed for the two baseline accident scenarios evaluated in Sensitivity Case 4 and 4b. The sensitivity cases are performed to show the impact changing both: (i) the fraction of MAR released during the accidents and (ii) the fraction of material released from the TRISO fuel during normal operation that reside in the core, reactor structure, and cooling system are higher than the base case. In Sensitivity Case #4a, it is assumed that the release from normal operation is increased to the 95th percentile value and the aggregate product of the Source Term Factors increased by an order of magnitude. In Sensitivity Case 4b, it is assumed that the release from normal operation is increased to the 95th percentile value and the aggregate product of the Source Term Factors decreased by an order of magnitude. Like Table 5-6, the increase in source term was does not to apply to gases (as the LPF, AR, RF are set to 1), because the uncertainty was assumed to be related to the ARF, RF, and LPF and not to the DR.

Table 5-7. Summary of Sensitivity Case 4a and 4b Inputs.

	Release Fraction from Normal Operations	Source Term Factors			
Release from the TRISO Fuel Decayed for Five Years					
	Percentile of distribution	**Damage Ratio**	**ARF x RF[1]**	**LPF**	**Sensitivity Cases**
Accident #1 Tanker collision	95%	1E-03 (FP/gases)	3E-4 impact 6E-5 thermal (FP)[2]	0.05 (FP)[2]	Case 4a: × 10 Case 4b: × 0.1
Accident #2 Loss of containment	95%	0	-	-	Case 4a: × 10 Case 4b: × 0.1
Release from the Compact and Other Core Structures					
Accident #1 Tanker Collison	95%	0.1 (FP)	3E-4 impact 6E-5 thermal (FP)	0.1 (FP)	Case 4a: × 10 Case 4b: × 0.1
Accident #2 Loss of containment	95%	0	-	-	Case 4a: × 10 Case 4b: × 0.1
Release Form Condensed or Plated-out Material in the Reactor Coolant Boundary					
Accident #1 Tanker Collison	95%	1 (FP)	3E-4 impact 6E-5 thermal (FP)	0.5 (FP)	Case 4a: ×10 Case 4b: × 0.1
Accident #2 Loss of containment	95%	0.2 (FP)	8E-4 (FP)	0.1 (FP)	Case 4a: × 10 Case 4b: × 0.1

TRISO = Tri-structural Isotropic particle fuel, ARF = airborne release fraction, RF = respirable fraction, FP= fission product, LPF = leak path factor.

[1] The ARF and RF are combined because the submersion and ingestion radiological dose pathways which do not require a RF are negligible contributors to dose and are not calculated.

[2] ARF, RF and Leak Path Factor are 1 for gases.

5.5 Presentation of Study Results and Insights

This section presents the radiological dose consequences for two accidents included in this for the baseline case and sensitivity studies selected for the parametric study. The assumptions used in baseline case reflect the research's team judgment about the most reasonable assumptions to make. The baseline case assumed five-year decay time, a median release fraction, and best judgment DRs, AFRs and RFs, and LPFs. The sensitivity studies consist of the following:

- Sensitivity Case #1 – Decrease the decay time down to one year.
- Sensitivity Case #2 – Increase of the release fraction to the 95th percentile from the mean.
- Sensitivity Case #3a and 3b – Increase of the combined RF, RF, and LPF by an order of magnitude and decrease by an order of magnitude.

- Sensitivity Case #4a and 4b – Increase of the release fraction to the 95th percentile from the mean combined with increase of the combined DR, ARF, RF, and LPF by an order of magnitude and decrease by an order of magnitude.

The radiological dose consequences were determined for the public using the methodology from SSG-26 (IAEA 2014) described in Section 5.3.4 above. This methodology assumes that an individual could be one meter from the point of the release. In this study the dose consequence of direct radiation exposure of released material is addressed, but the dose consequence of direct radiation exposure from material that has not been released (including activated material) is not addressed though the associated shielding could be degraded in the accident. The focus of this study is on release fractions. Moreover, the shielding damage would be highly design specific.

The assumption is that the dose consequences from inhalation dose will be the primary contributor to dose consequences and direct radiation dose from unreleased material for a member of the public not immediately near the radiation source is negligible.

Table 5-8 presents the radiological dose consequences from the selected accidents for the baseline case. For the tanker collision and subsequent fire accident, the table shows two aggregate source term factors, one each for the fire and impact phenomena. It shows total the radiological dose for each accident to be at an acceptable level compared to a 25-rem dose evaluation limit. Though there are currently no risk evaluation guidelines for the dose consequences from release of radiological material in a transportation accident, a 25-rem evaluation limit is used in other nuclear safety applications. For example, siting guidelines from 10 CFR Part 100, “Reactor Site Criteria,” Section 100.11, establish an exclusion zone around a commercial nuclear plant to prevent a total radiation dose to the whole body in excess of 25 rem and the associated acute health risk. The DOE guidance for nuclear nonreactor facilities in DOE-STD-3009-2014 (2014 DOE) states that if the unmitigated release consequence of an accident exceeds the 25-rem total effective dose, then controls shall be applied to prevent the accident or mitigate its consequences to below this risk evaluation guideline limit. Section 61 of 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” defines the performance requirements that must be met when more than a critical mass of special nuclear material is involved which includes certain nuclear fuel cycle facilities. High-consequence events that result in an acute dose of 25 rem or greater total effective dose equivalent (TEDE) to any individual located outside the controlled area will be limited using controls.

Table 5-8 shows for the tanker collision and subsequent fire, based the source factor used for the baseline case, that the radiological dose consequences are below the 25-rem dose evaluation limit. The table also shows that the contribution from the TRISO fuel itself to be the most significant contributor to the dose to a human receptor based on the assumptions used from IAEA SSG-26 (IAEA 2014) described above in Section 5.3.4. However, the release from the compact and other core structures as well as from the condensed or plated-out material in the reactor coolant boundary could be significant if these contributions are significantly underestimated for the baseline case. The table shows for the loss of containment event from a non-road accident that the radiological dose consequences are very low compared to the 25-rem dose evaluation limit discussed above. Accordingly, the uncertainty associated with the inputs to this dose calculation are not likely to be very important. Based on the source term factors, it appears that the release associated with impact phenomenon is a greater contributor to radiological dose that the contribution from the fire phenomenon given that the aggregate source term factors are order of magnitude greater for impact than for fire and all other factors are the same. This likely due to the qualification of TRISO fuel to very high temperatures.

Table 5-8. Summary of Baseline Radiological Dose Consequence Results.

Accident / Release Location	Normal Operations Release Fraction (Percentile of Distribution)	Source Term Factors, (DR × ARF × RF × LPF)		TEDE to an Individual (rem)
Release from the TRISO Particle Fuel Decayed for Five Years				
Accident # 1: Tanker Collison				
		Fire Phenomenon	**Impact Phenomenon**	
TRISO fuel (five-year decay)	Mean	3E-09	1.5E-08	20.5
Reactor core structures	Mean	6E-07	3.0E-06	0.56
Coolant pressure boundary	Mean	3E-05	1.5E-04	0.25
Total				21.3
Accident # 2: Loss of Containment				
TRISO fuel (five-year decay)	Mean	0		0
Reactor core structures	Mean	0		0
Coolant pressure boundary	Mean	2E-08		1.21E-02
Total				1.21E-02

TRISO = Tri-Structural Isotropic Particle Fuel (TRISO), DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor, TEDE = total effective dose equivalent, rem = roentgen equivalent man.

Table 5-9 presents the radiological dose consequences from the accident for Sensitivity Case #1 in which decrease of the decay time down from five years to one year. The results of this sensitivity case show an increase in radiological dose from the tanker collision and subsequent fire accident of 62% which due almost entirely to the contribution from the TRISO fuel. For the containment loss accident, the results show a small increase and that the absolute dose is very low. The total radiological dose consequence from the tanker collision and fire is a little above the 25-rem dose evaluation limit discussed in this section above. The total radiological dose consequence from the containment loss accident is well below the 25-rem dose evaluation limit.

Table 5-9. Summary of Sensitivity Case #1 Radiological Dose Consequence Results.

Accident / Release Location	Normal Operations Release Fraction (Percentile of Distribution)	Source Term Factors, (DR × ARF × RF × LPF)		TEDE to an Individual (rem)
Release from the TRISO Fuel Decayed for One Year				
Accident # 1: Tanker Collison				
		Fire Phenomenon	**Impact Phenomenon**	
TRISO fuel (five-year decay)	Mean	3E-09	1.5E-08	33.5
Reactor core structures	Mean	6E-07	3.0E-06	0.64
Coolant pressure boundary	Mean	3E-05	1.5E-04	0.42

Accident / Release Location	Normal Operations Release Fraction (Percentile of Distribution)	Source Term Factors, (DR × ARF × RF × LPF)	TEDE to an Individual (rem)
Total			34.6
Accident # 2: Loss of Containment			
TRISO fuel (five-year decay)	Mean	0	0
Reactor core structures	Mean	0	0
Coolant pressure boundary	Mean	2E-08	1.76E-02
Total			1.76E-02

TRISO = Tri-Structural Isotropic Particle Fuel Decayed (TRISO), DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor, TEDE = total effective dose equivalent, rem = roentgen equivalent man.

Table 5-10 present the radiological dose consequences from the accident for Sensitivity Case #2 in which the release fraction from normal operation that reside in the core, reactor structure, and cooling system was increased to its 95th percentile. The results of this sensitivity case shows only a small increase of 8.5% in the radiological dose results for the tanker collision and subsequent fire accident which is due to the increase in dose from the reactor core structures and coolant sealant boundary. For the containment loss accident, the results show about a 6.4 factor increase in radiological dose, but the absolute dose is still very low. The radiological dose consequences from both accidents are below the 25-rem dose evaluation limit discussed in this section above.

Table 5-10. Summary of Sensitivity Case #2 Radiological Dose Consequence Results.

Accident / Release Location	Normal Operations Release Fraction (Percentile of Distribution)	Source Term Factors (DR × ARF × RF × LPF)		TEDE to an Individual (rem)
Release from the TRISO Fuel Decayed for Five Years				
Accident # 1: Tanker Collison				
		Fire Phenomenon	**Impact Phenomenon**	
TRISO fuel (five-year decay)	95%	3E-09	1.5E-08	20.5
Reactor core structures	95%	6E-07	3.0E-06	1.71
Coolant pressure boundary	95%	3E-05	1.5E-04	0.86
Total				23.1
Accident # 2: Loss of Containment				
TRISO fuel (five-year decay)	95%	0		0
Reactor core structures	95%	0		0
Coolant pressure boundary	95%	2E-08		7.71E-02
Total				7.71E-02

TRISO = Tri-Structural Isotropic Particle Fuel (TRISO), DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor, TEDE = total effective dose equivalent, rem = roentgen equivalent man.

Table 5-11 presents the radiological dose consequences from the accident for Sensitivity Case 3a in which the combined ARF, RF, and LPF was increased by an order of magnitude, while Table 5-12 presents the radiological dose consequences from the accident for Sensitivity Case #3b in which the combined ARF, RF, and LPF was decreased by an order of magnitude. The results of Sensitivity Case 3a shows a significant increase of a factor of 7.3 in the radiological dose results for the tanker collision and subsequent fire accident for a total dose of 155 rem. For the containment loss accident, the results show about an order of magnitude increase radiological dose, but the absolute dose is still very low. The increase in source term was does not to apply to gases (as the LPF, AR, RF are set to 1) which results in the radiological dose increasing by only a factor of 3 for the tanker collision and subsequent fire accident. In other words, the uncertainty was assumed to be related to the ARF, RF, and LPF and not to the DR. The radiological dose consequences from the tanker collision and fire accident are well above the 25-rem dose evaluation limit and the dose consequences from the containment loss accident are significantly below the evaluation limit discussed earlier in this section.

Table 5-12 shows presents the results of Sensitivity Case 3b that shows 62.8% decrease in the radiological dose results for the tanker collision and subsequent fire accident. For the containment loss accident, the results show about an order of magnitude decrease in radiological dose, but the absolute dose before the decrease was very low. The radiological dose consequences from both accidents in both cases are below the 25-rem dose evaluation limit discussed above in this section.

Table 5-11. Summary of Sensitivity Case 3a Radiological Dose Consequence Results.

Accident / Release Location	Normal Operations Release Fraction (Percentile of Distribution)	Source Term Factors (DR × ARF × RF × LPF)		TEDE to an Individual (rem)
Release from the TRISO Fuel Decayed for Five Years				
Accident # 1: Tanker Collison				
		Fire Phenomenon	**Impact Phenomenon**	
TRISO fuel (five-year decay)	Mean	3E-08	1.5E-07	147
Reactor core structures	Mean	6E-06	3.0E-05	5.64
Coolant pressure boundary	Mean	3E-04	1.5E-03	2.49
Total				155
Accident # 2: Loss of Containment				
TRISO fuel (five-year decay)	Mean	0		0
Reactor core structures	Mean	0		0
Coolant pressure boundary	Mean	2E-07		1.21E-01
Total				1.21E-01

TRISO = Tri-Structural Isotropic Particle Fuel (TRISO), DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor, TEDE = total effective dose equivalent, rem = roentgen equivalent man.

Table 5-12. Summary of Sensitivity Case 3b Radiological Dose Consequence Results.

Accident / Release Location	Normal Operations Release Fraction (Percentile of Distribution)	Source Term Factors (DR × ARF × RF × LPF)		TEDE to an Individual (rem)
Release from the TRISO Fuel Decayed for Five Years				
Accident # 1: Tanker Collison				
		Fire Phenomenon	**Impact Phenomenon**	
TRISO fuel (five-year decay)	Mean	3E-10	1.5E-09	7.83
Reactor core structures	Mean	6E-08	3.0E-07	0.06
Coolant pressure boundary	Mean	3E-06	1.5E-05	0.02
Total				7.91
Accident # 2: Loss of Containment				
TRISO fuel (five-year decay)	Mean	0		0
Reactor core structures	Mean	0		0
Coolant pressure boundary	Mean	2E-09		1.21E-03
Total				1.21E-03

TRISO = Tri-Structural Isotropic Particle Fuel (TRISO), DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor, TEDE = total effective dose equivalent, rem = roentgen equivalent man.

Table 5-13 presents the radiological dose consequences from the selected accidents for Sensitivity Case 4a in which: (i) the release fraction from normal operation that reside in the core, reactor structure, and cooling system is increased to its 95th percentile and (ii) the combined ARF, RF, and LPF is increased by an order of magnitude. Table 5-14 presents the radiological dose consequences from the accident for Sensitivity Case 4b in which: (i) the release fraction from normal operation that reside in the core, reactor structure, and cooling system was increased to its 95th percentile and (ii) the combined ARF, RF, and LPF was decreased by an order of magnitude.

The results of Sensitivity Case 4a shows a significant increase of a factor of 8.1 in the radiological dose results for the tanker collision and subsequent fire accident for a total dose of 173 rem which exceeds the 25-rem dose evaluation limit discussed above in this section. For the containment loss accident, the results show about a factor of 6.4 increase, but the absolute dose is still very low.

Table 5-14 presents the results of Sensitivity Case 4b which shows about 62% decrease in the radiological dose results for the tanker collision and subsequent fire accident. For the containment loss accident, the results show about a 36% decrease in radiological dose, but the absolute dose before the decrease was extremely low. The radiological dose consequences from both accidents are well below the 25-rem dose evaluation limit discussed above.

Table 5-13. Summary of Sensitivity Case 4a Radiological Dose Consequence Results.

Accident / Release Location	Normal Operations Release Fraction (Percentile of distribution)	Source Term Factors (DR × ARF × RF × LPF)		TEDE to an Individual (rem)
Release from the TRISO Fuel Decayed for Five Years				
Accident # 1: Tanker Collison				
		Fire Phenomenon	**Impact Phenomenon**	
TRISO fuel (five-year decay)	95%	3E-08	1.5E-07	147
Reactor core structures	95%	6E-06	3.0E-05	17.1
Coolant pressure boundary	95%	3E-04	1.5E-03	8.68
Total				173
Accident # 2: Loss of Containment				
TRISO fuel (five-year decay)	95%	0		0
Reactor core structures	95%	0		0
Coolant pressure boundary	95%	2E-07		7.71E-01
Total				7.71E-01

TRISO = Tri-Structural Isotropic Particle Fuel (TRISO), DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor, TEDE = total effective dose equivalent, rem = roentgen equivalent man.

Table 5-14. Summary of Sensitivity Case 4b Radiological Dose Consequence Results.

Accident / Release Location	Normal Operations Release Fraction (Percentile of Distribution)	Source Term Factors (DR × ARF × RF × LPF)		TEDE to an Individual (rem)
Release from the TRISO Fuel Decayed for Five Years				
Accident # 1: Tanker Collison				
		Fire Phenomenon	**Impact Phenomenon**	
TRISO fuel (five-year decay)	95%	3E-10	1.5E-09	7.83
Reactor core structures	95%	6E-08	3.0E-07	0.17
Coolant pressure boundary	95%	3E-06	1.5E-05	0.09
Total				8.09
Accident #2: Loss of Containment				
TRISO fuel (five-year decay)	95%	0		0
Reactor core structures	95%	0		0
Coolant pressure boundary	95%	2E-09		7.71E-03
Total				7.71E-03

TRISO = Tri-Structural Isotropic Particle Fuel (TRISO), DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor, TEDE = total effective dose equivalent, rem = roentgen equivalent man.

6.0 Conclusions and Recommendations

Based on the radiological dose consequences analyses performed for selected transportation accidents involving an example 20 MWt microreactor core inventory that has decayed for five years and best-judgment, assumptions, the dose to an individual human receptor close to the accident appears to be acceptability low in-spite of conservatisms in the analyses. One of the accidents selected for this study, the tanker collision and subsequent fire accident, is considered to be bounding because it judged result in the greatest radiological dose consequences. This judgment is based on hazard analysis principles and previous transportation risk assessment studies as described in Section 5.2. An example 20 MWt microreactor core inventory was chosen because it is judged to represent the upper end of the power range for microreactors that might be transported by trucks. Therefore, the radiological dose consequences from a transportation accident involving a lower power reactor should be even lower, because the radionuclide inventory should be less (if all other factors remain about the same or proportional.) Also, though the baseline assumptions are best judgment estimates, they were chosen conservatively given the uncertainties identified in this report such as the source term factor values to use in a transportation accident in which the package is subject to high energy impact.

However, the sensitivity studies show that when the key consequence analyses are set to be more conservative, the radiological dose consequences from the tanker collision and subsequent fire event can well exceed the 25-rem dose evaluation limit. This is particularly true for Sensitivity Case 3a and 4a in which aggregate source term factors (i.e., the ARF, RF, and LPF) were increased by an order of magnitude. Conversely Sensitivity Cases 4a and 4b, in which aggregate source term factors (i.e., the ARF, RF, and LPF) were decreased by an order of magnitude show that the total dose for the tanker collision and fire accident is well below the evaluation limit used in this study. Sensitivity Case 1, which provides the dose consequences for the same core decayed for one year opposed to five, indicates that the radiological dose consequence from the tanker collision and fire accident would exceed the evaluation limit but not by the margins that Sensitivity Case 3a and 4a exceed the evaluation limit. However, Sensitivity Case 2, which provides the dose consequences when the release fraction from operations is increased from the mean of an estimated probability distribution to the 95th percentile, shows that the dose consequences from the tank collision and fire accident would be below the evaluation limit.

The results of this parametric study indicate that the radiological dose consequences from a worse-case TNPP transportation accident could be acceptably low based on using best judgment assumptions. However, given the lack of testing and analysis of the performance of microreactors as transportation packages, it is not clear if these results are generalizable. This modelling uncertainty would need to be addressed in order to perform a TNPP transportation Probabilistic Risk Assessment. Of particular concern is the assumed damage that occurs as a result of a transportation accident which can in turn have a significant effect on the estimated source term factors (i.e., DR, ARF, RF, and LPF). This is of particular concern for the source term factors associated with the TRISO fuel itself as the radiological dose consequences from the TRISO fuel (opposed to material diffused into the reactor core internals or plated out in the primary system) dominate the dose consequence results.

In all sensitivity cases, removal of the TRISO fuel (compact stacks) prior to transportation of the microreactor reduces the potential dose consequence well below 25-rem, including in the most extreme cases. So, transporting the TRISO fuel separately in a standard certified spent fuel

package configuration dramatically reduces dose since diffused fission and activation products in the reactor core components and within the primary circuit pressure boundary are much smaller contributors to dose consequence.

Additional PIE and testing of the mechanical properties of TRISO fuel particles, compacts, and graphite core components under impact loads could provide data to reduce conservatism in postulated release fractions by informing the transportation package design and validating its performance under bounding accident circumstances.

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Appendix A
Radionuclide Inventory

In support this evaluation, X-energy provided a scoping inventory for a representative 20 MWt reactor operated for three effective full power years with selected cooling (decay) times of one and five years. This initial listing contained over 1300 radionuclide fission products and gases. The list was subsequently screened to identify the radionuclides with greater than 0.1 Ci after five years of cooling (Base Case). This resulted in 49 radionuclides which accounted for over 99% of the total curies. These 49 radionuclides also account for over 95% of the total Ci for the one year cooling sensitivity case.

Table A-1. Radionuclide Inventory for a Representative 20 MWt High Temperature Gas Reactor at Three Effective Full Power Years.

Radionuclide	Inventory 1 year cooling (Ci)	Inventory 5 year cooling (Ci)
Ag-110	5.80E+00	1.01E-01
Ag-110m	4.28E+02	7.43E+00
Am-241	1.55E+02	4.80E+02
Am-242	3.92E+00	3.84E+00
Am-242m	3.95E+00	3.86E+00
Am-243	8.90E+00	8.88E+00
Ba-137m	8.55E+04	7.80E+04
Cd-113m	1.58E-01	1.29E-01
Ce-144	3.41E+05	9.76E+03
Cm-242	5.30E+03	1.38E+01
Cm-243	8.18E+00	7.42E+00
Cm-244	1.26E+03	1.08E+03
Cm-245	1.41E-01	1.41E-01
Cs-134	7.78E+04	2.04E+04
Cs-135	6.26E-01	6.26E-01
Cs-137	9.03E+04	8.23E+04
Eu-152	5.20E+00	4.25E+00
Eu-154	3.08E+03	2.23E+03
Eu-155	1.72E+03	9.62E+02
H-3	3.78E+02	3.02E+02
Kr-85	9.11E+03	7.04E+03
Nb-93m	2.12E-01	4.54E-01
Np-237	2.42E-01	2.42E-01
Np-239	8.89E+00	8.89E+00

Radionuclide	Inventory 1 year cooling (Ci)	Inventory 5 year cooling (Ci)
Pa-233	2.41E-01	2.43E-01
Pm-147	1.20E+05	4.16E+04
Pr-144	3.40E+05	9.75E+03
Pr-144m	3.26E+03	9.34E+01
Pu-238	2.17E+03	2.12E+03
Pu-239	1.21E+02	1.21E+02
Pu-240	1.80E+02	1.80E+02
Pu-241	5.63E+04	4.63E+04
Pu-242	1.11E+00	1.11E+00
Rh-106	1.07E+05	7.01E+03
Ru-106	1.07E+05	7.04E+03
Sb-125	3.84E+03	1.40E+03
Sb-126m	1.56E-01	1.56E-01
Sm-151	3.45E+02	3.34E+02
Sn-119m	4.09E+01	1.30E+00
Sn-121	1.02E+01	9.53E+00
Sn-121m	1.31E+01	1.23E+01
Sn-126	1.56E-01	1.56E-01
Sr-90	7.71E+04	7.00E+04
Tc-99	1.26E+01	1.26E+01
Te-125m	9.35E+02	3.44E+02
U-236	3.62E-01	3.62E-01
U-237	1.38E+00	1.13E+00
Y-90	7.74E+04	7.03E+04
Zr-93	1.78E+00	1.78E+00

Ci = Curie, Ag = silver, Am = americium; Ba = barium, Cd = cadmium, Ce = cerium, Cm = curium, Cs = cesium, Eu = europium, H = hydrogen, Kr = krypton, Nb = niobium, Np = neptunium, Pa = protactinium, Pm = promethium, Pr = praseodymium, Pu = plutonium, Rh = rhodium, Ru = ruthenium, Sb = antimony, Sm = samarium, Sn = tin, Sr = strontium, Te = tellurium, U = uranium, Y = yttrium, Zr= zirconium.

Appendix B
INL/EXT-11-22034 Model

As discussed in Section 4.2.1, the release of fission products and gases and subsequent attenuation during normal operation defines the material at risk (MAR) in the three locations (reactor coolant pressure boundary, core structure and compact graphite, and Tri-structural Isotropic [TRISO] fuel). These various MAR are key inputs into subsequent dose modeling for transportation accidents.

The release from TRISO is a function of the initial fuel quality, fuel performance and failure information and assumed reactor operating (outlet) temperature. This model assumes a prismatic core with outlet temperature in the range of 900 °C. The fuel release fractions, conservatively set consistent with the 50 percent and 95 percent values contained in INL/EXT-11-24034, are presented in Table B-1.

Attenuation represents the capability of an identified barrier to prevent or reduce the release of fission products. The attenuation factor estimates associated with normal operation and are also dependent on temperature. The attenuation factors assumed for this evaluation include: (i) the fuel kernel, (ii) the particle coatings, particularly the silicon carbide (SiC) coating, (iii) the fuel element and structural graphite, and (iv) the reactor coolant pressure boundary. The attenuation factors, conservatively set consistent with the 50 percent and 95 percent values contained in INL/EXT-11-24034, are presented in Table B-2.

Using the inputs of Table B-1 and Table B-2, release fractions were calculated. The process for developing the release fractions utilized a Monte Carlo analysis of 100,000 trials for the radionuclides in each of the fission product classes as defined in Table 4-1 consistent with the simplified equations presented below for releases to the reactor coolant pressure boundary. Releases fractions to the core structure and compact graphite were determined following a similar approach. These release fractions multiplied by the initial inventories provide the MAR specific to a location. Finally, the fission product and gas inventory MAR remaining with the TRISO fuel are defined as:

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

For additional information see INL/EXT-11-24034.

Normal operations Release ($MAR_{i,RCPB}$)

$$= Normal\ Ops\ HMC + Normal\ Ops\ SiC\ Defect + Normal\ Ops\ In\ Service\ Failure$$

Normal operations heavy metal contamination released to the RCB

$$= Inv_i \cdot \frac{FP_{i,HMC}}{AF_{i,HMC} \cdot AF_{i,G}}$$

Normal operations SiC released to the RCB

$$= Inv_i \cdot \frac{FP_{i,ISF}}{AF_{i,K} \cdot AF_{i,G}}$$

Normal operations In-Service Failure released to the RCB

$$= Inv_i \cdot \frac{1}{AF_{i,K} \cdot AF_{i,DR} \cdot AF_{i,G}}$$

where:

AF = Attenuation Factor
DR = diffusive release through coating
G = graphite
HMC = heavy metal contamination
i = radionuclide index
IF = incremental failure under accident
Inv = Inventory
FP = fission product
ISF = in-service failure
K = fuel kernel

Table B-1. TRISO Fuel Release Parameters — Normal Operations; 900 °C, Prismatic.

	Sources of Fission Product Release								
	Fraction Heavy Metal Contamination			Fraction Silicon Carbide Defects			In-Service Failure		
Nuclide	50%	95%	Sigma	50%	95%	Sigma	50%	95%	Sigma
Xe-133	1.00E-05	1.00E-04	5.472E-05	NA	NA		1.40E-05	7.00E-05	3.405E-05
Kr-85	1.00E-05	1.00E-04	5.47E-05	NA	NA		1.40E-05	7.00E-05	3.405E-05
Kr-88	1.00E-05	1.00E-04	5.472E-05	NA	NA		1.40E-05	7.00E-05	3.405E-05
I-131	1.00E-05	1.00E-04	5.472E-05	NA	NA		1.40E-05	7.00E-05	3.405E-05
I-133	1.00E-05	1.00E-04	5.472E-05	NA	NA		1.40E-05	7.00E-05	3.405E-05
Te-132	1.00E-05	1.00E-04	5.472E-05	NA	NA		1.40E-05	7.00E-05	3.405E-05
Cs-137	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
Cs-134	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
Sr-90	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
Ag-110m	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
Ag-111	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
Sb-125	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
Ru-103	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
Ce-144	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
La-140	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
Pu-239	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	2.10E-04	1.05E-03	5.11E-04
H-3	1.00E-05	1.00E-04	5.472E-05	1.00E-05	3.00E-05	1.216E-05	1.40E-05	7.00E-05	3.405E-05

Xe = xenon, Kr = krypton, I = iodine, Te = tellurium, Cs = cesium, Sr= strontium, Ag = Silver, Sb= antimony, Ru = ruthenium, Ce = cerium, La = lanthanum, Pu = plutonium, H = hydrogen.

Table B-2. Radionuclide Attenuation Parameters — Normal Operations; 900 °C, Prismatic.

	Barriers to Fission Product Release (Attenuation Factors)											
	Heavy Metal Contamination			Kernel			Diffusive Release Through coating			Graphite		
Nuclide	AF 50%	AF 95%	Sigma	AF 50%	AF 95%	Sigma	AF 50%	AF 95%	Sigma	AF 50%	AF 95%	Sigma
Xe-133	5	1.5	7.32E-01	25	8.3333333	6.68E-01	5.00E+07	5.00E+06	1.40E+00	1	1	0
Kr-85	5	1.5	7.32E-01	25	8.3333333	6.68E-01	5.00E+07	5.00E+06	1.40E+00	1	1	0
Kr-88	5	1.5	7.32E-01	25	8.3333333	6.68E-01	5.00E+07	5.00E+06	1.40E+00	1	1	0
I-131	5	1.5	7.32E-01	25	8.3333333	6.68E-01	5.00E+07	5.00E+06	1.40E+00	1	1	0
I-133	5	1.5	7.32E-01	25	8.3333333	6.68E-01	5.00E+07	5.00E+06	1.40E+00	1	1	0
Te-132	5	1.5	7.32E-01	25	8.3333333	6.68E-01	5.00E+07	5.00E+06	1.40E+00	1	1	0
Cs-137	1	1	0	1.2	1	0.1108436	1.00E+07	1.00E+05	2.80E+00	2	1	4.21E-01
Cs-134	1	1	0	1.2	1	0.1108436	1.00E+07	1.00E+05	2.80E+00	2	1	4.21E-01
Sr-90	1	1	0	3	1	0.6679088	500	100	9.78E-01	100	30	7.32E-01
Ag-110m	1	1	0	1	1	0	200	40	9.78E-01	1	1	0
Ag-111	1	1	0	1	1	0	1000	200	9.78E-01	1	1	0
Sb-125	1	1	0	1	1	0	5.00E+07	5.00E+05	2.80E+00	5	1	9.78E-01
Ru-103	1	1	0	250	15	1.71E+00	1.00E+07	1.00E+06	1.40E+00	200	60	7.32E-01
Ce-144	1	1	0	250	15	1.71E+00	1.00E+07	1.00E+06	1.40E+00	200	60	7.32E-01
La-140	1	1	0	250	15	1.71E+00	1.00E+07	1.00E+06	1.40E+00	200	60	7.32E-01
Pu-239	1	1	0	500	50	1.40E+00	1.00E+07	1.00E+06	1.40E+00	5.00E+03	500	1.40E+00
H-3	2	1	4.21E-01	1	1	0	2	1	4.21E-01	100	10	1.40E+00

AF= attenuation factors, Xe = xenon, Kr = krypton, I = iodine, Te = tellurium, Cs = cesium, Sr= strontium, Ag = silver, Sb= antimony, Ru = ruthenium, Ce = cerium, La = lanthanum, Pu = plutonium, H = hydrogen.

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