Risk-Informed Approach for Regulatory Approval of Microreactor Transport

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Abstract: Pacific Northwest National Laboratory (PNNL) is addressing the challenges associated with safe transport of microreactors including the development and evaluation of regulatory options. PNNL developed a risk-informed regulatory framework for the licensing of the transportation of microreactors in which irradiated nuclear fuel is part of the microreactor transportation package. The framework lays out a viable regulatory pathway, including decision points for regulatory options and the supporting technical evaluations for those options in phases from near to long term.

A microreactor and its contents will likely not be able to meet all the federal regulatory requirements as a Type B or fissile material transportation package under 10 CFR Part 71 ("Packaging and Transportation of Radioactive Material"). However, the regulatory framework developed by PNNL lays out a viable risk-informed licensing options that are safe and feasible. Risk assessment such as probabilistic risk assessment (PRA) can be used to show comparable safety to that provided by a Type B or fissile material package for surface transport.

The framework includes guidance on applicable regulations and discusses historical precedence in using risk information for transportation licensing. The framework includes guidance for performing a microreactor transportation PRA, use and development of risk evaluation criteria, and factors such of defense-in-depth and safety margin concepts. Key advantages of using the approach are (1) increasing the likelihood of successfully obtaining regulatory transportation package approval, (2) informing the design on the relative risk significance of microreactor containment and shielding, and (3) informing the need for transportation compensatory measures as well as identification of appropriate measures. This paper focuses on primary elements of the framework which include development of a transportation PRA for these packages and risk acceptance guidelines to assess the results of the PRA for regulatory decision-making.

1. INTRODUCTION

Microreactors are very small nuclear reactors of 20 megawatts electric (MWe) or less, designed to be factory-built and may be transportable. The microreactor designs considered in this proposed approach are based on tri-structural isotropic (TRISO) fuel using high-assay low-enriched uranium (HALEU) with enrichments of 5 percent to 20 percent. At the time of this writing, two vendors were developing competing final engineering designs for a demonstration microreactor under the Department of Defense's (DOD) Strategic Capabilities Office (SCO) Project Pele, with a down-select decision to be made this year. Both microreactor designs are high-temperature gas-cooled reactors that utilize accident-tolerant TRISO fuel. The first unit are expected to be demonstrated in the USA and to be transported over public highways. Transportation of radioactive materials in the USA is regulated by the Nuclear Regulatory Commission (NRC). A microreactor and its irradiated fuel contents will likely not be able to meet all the regulatory requirements as a Type B or fissile material transportation package under NRC regulations.

Pacific Northwest National Laboratory (PNNL) in work for the National Reactor Innovation

Center (NRIC) a national Department of Energy (DOE) program within the Office of Nuclear Energy research (NE) developed a regulatory framework for transporting a microreactor in PNNL-31867, "Proposed Risk Informed Regulatory Framework for Approval of Microreactor Transportation Packages." [1] The framework identifies potential viable regulatory pathways for the licensing of future microreactor transportation packages including the licensing of a demonstration one-time shipment that utilizes risk information to show comparable safety to that provided by a Type B or fissile material package.

Based on and as a follow-on to that study, PNNL has developed a plan to be used by Project Pele microreactor vendors for the development and application of a risk assessment approach to support a risk-informed pathway for regulatory approval of a one-time domestic highway shipment of a microreactor transportation package. Demonstration of the plan includes the development of a risk assessment methodology, risk evaluation guidelines, technical information, data, and example analyses that provide a potential template for a vendor to follow when making a request to the NRC for approval to transport a microreactor containing irradiated fuel.

The paper summarizes describes the most feasible regulatory approach for transportation of the demonstration microreactor, proposed risk evaluation guidelines, and an applicable risk assessment approach.

2. REGULATORY APPROCH

This section discusses the regulatory pathway that was identified to be the most feasible for the licensing transportation of the demonstration microreactor and why a risk-informed approach is needed to support this option. It also summarizes applicable federal regulations to support later discussions of the proposed risk evaluation guidelines and microreactor transportation risk assessment approach.

The 10 CDF Part 71 ("Packaging and Transportation of Radioactive Material") exemption process was determined in an evaluation of potential regulatory approval options performed by PNNL to be the most feasible approach for licensing transportation of the demonstration microreactor package. Identification of possible regulatory options, evaluation of those options for both the demonstration and production stages of the Pele Project, and selection of the most feasible option for each stage is discussed in PNNL-31867 [1]. Though evaluation of the 10 CFR Part 71 requirements which require an exemption could potentially be qualitative or semi-quantitative, there are significant challenges associated with using qualitative evaluation to demonstrate that transport of an microreactor can be performed at an acceptable level of risk. Concerns include the fact that transport of a microreactor will occur with irradiated fuel which will be a first-of-a-kind endeavour, design and modelling uncertainties, and the potential risk to the public if a transportation accident occurs. Accordingly, for the demonstration phase, the 10 CFR Part 71.12 ("Specific Exemptions") process should be supported by quantitative risk assessment.

According to 10 CFR Part 71, four regulatory options are available for the approval of a microreactor transportation package. The following subsections discuss for the U.S.:

- demonstration of compliance with environmental test conditions,
- demonstration of compliance with alternate environmental test conditions,
- request for special package authorization, and
- request for Specific Exemptions.

These current regulatory pathways or options for obtaining NRC approval of shipments involving Type B quantities of radioactive materials are discussed in detail in PNNL31867 [1]. The preferred regulatory pathway was determined to be through the exemption process (10 CFR 71.12) because exemption(s) (1) can be applicable to multiple shipments (unlike the special package authorization approach under 10 CFR 71.41(d)), (2) provide for greater flexibility in deviating from the deterministic requirements of 10 CFR Part 71 (compared to the alternative environmental and test conditions approach under 10 CFR 71.41(c)), and (3) have historical precedent. It appears infeasible and cost-prohibitive to acquire a Certification of Completion for a microreactor package that it meets federal requirements for Type B

package including severe test conditions for the prototype demonstration unit that may only be used a few times before being retired.

Demonstrating compliance with all environmental and test conditions in 10 CFR 71.41(a) and all leak rate and shielding requirements in 10 CFR 71.51 or 10 CFR 71.55 after hypothetical accident conditions will prove challenging for the transportation of microreactor packages. As stated above, irradiated fuel will be shipped as an integrated component of the package (e.g., loaded in the microreactor). A risk-informed approach will be used to address the fact that elements of the deterministic federal requirement cannot be met creating uncertainty about the level risk from transportation of the package.

Based on insights from past applications, a 10 CFR part 71 exemption process will need to include the following, among the other standard contents of a transportation package approval request:

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

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

3. PROPOSED RISK EVALUATION GUILINES

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

3.1. NRC Suggested Risk Evaluation Guidelines

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

For the accident risk associated with this type of activity, NRC proposes guidance in a report titled *Risk-Informed Decisionmaking for Nuclear Material and Waste Applications* [2] (referred to hereafter as the RIDM report) for accepting the risk associated with transportation of nuclear material based on a risk assessment approach such as a probabilistic risk assessment (PRA). The approach involves the use of

quantitative health guidelines (QHGs) that are based on the same safety goals that the risk evaluation guidance for nuclear power plants is derived. However, the risk evaluation guidance presented in the RIDM report for the transportation of nuclear material has not been endorsed by NRC, and there remains challenges to approving and applying the approach. The RIDM report itself cautions that development of risk evaluation guidelines based on QHGs needs discussion and is ultimately a policy decision. None-the-less, as a starting point to developing risk evaluation guidance for the transportation of a microreactor transportation package, the following is a summary of the proposed QHG approach.

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

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

- "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed."
- "The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."

Based on these QHOs, the RIDM report proposes the following QHGs to define the threshold for negligible accident risk for use as risk evaluation guidelines for the risk associated with transportation of nuclear material as shown in Table 1:

Table 1: NRC Proposed QHGs Based on Interpretation of Safety Policy Statement [2]

Receptor	Acute Fatality	Latent Cancer Fatality	Serious Injury (Cancer Illness)
Public	QHG-1 - Public individual risk of acute fatality is negligible if it is less than or equal to 5×10^{-7} fatality per year.	QHG-2 - Public individual risk of a LCF is negligible if it is less than or equal to 2×10^{-6} fatality per year or 4 mrem per year	QHC-3 - Public individual risk of serious injury is negligible if it is less than or equal to 1×10^{-6} injury per year.
Worker	QHG-4 - Worker individual risk of acute fatality is negligible if it is less than or equal to 1×10^{-6} fatality per year.	QHG-5 - Worker individual risk of LCF is negligible if it is less than or equal to 1×10^{-5} fatality per year or 25 mrem per year.	QHG-6 - Worker individual risk of serious injury is negligible if it is less than or equal to 5×10^{-6} injury per year.

There are no specific guidelines in the NRC Safety Goal Policy statement for workers. However, based on considerations discussed in the RIDM report proposed qualitative and quantitative criteria. The report proposed for workers that the additional risk of prompt fatality from accidents involving acute exposure should be small in comparison to the same risk faced by U.S. workers in general but not as small as members of the public who are not formally trained in radiation protection. The report uses similar rationale for latent cancer fatality (LCF) and serious injury (cancer illness) and quantitative measures consistent with those criteria as shown in Table 1.

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

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

- It may not be necessary and would reduce calculational burden to express the release from the accident sequences in terms of rems to worker or the public without conversion to health effects. This is particularly true if many scenarios or sensitivity studies are anticipated as in the case for demonstration microreactor transportation PRA study.
- If the QHGs are expressed as pairs of acceptable likelihood and consequence values in which the consequence is expressed as radiological dose without combining the values, then comparisons can made of these radiological dose threshold limits to the radiological dose limits in relevant federal and international regulations and guidance. This comparison can be used to validate dose threshold limits derived from the QHGs.
- Additionally, if the accident sequence results of a PRA were determined as pairs of likelihood and consequence values, then the PRA results will provide a greater level of information that can be useful for decisionmaking or development of applicable design changes or compensatory actions.

This substitute risk measure of pairs of likelihood and consequence values can be thought of as a surrogate to the proposed QHGs. Even nuclear power plant PRAs, for which the PRA technology is mature and well-accepted by NRC, are not typically taken Level III to determine public health impacts. Rather, PRAs used to support risk-informed licensing decisions produce results in terms of Core Damage Frequency (CDF) and Large Early Release Fraction (LERF) because the risk evaluation guidelines established using these metrics are much more attainable and practical to use than QHGs. Accordingly, CDF and LERF are used as surrogate measures to the QHGs. NRC has issued guidance in Regulatory Guide (RG) 1.174, Revision 3 [3] that stipulates CDF and LERF levels at which a change in a plant's operating license would not be allowed using a risk informed approach. RG 1.174 states that it uses the Safety Goal Policy Statement and QHOs to define an acceptable level of risk based on "subsidiary objectives" derived from the safety goals and QHOs. RG 1.174 refers to CDF and LERF risk evaluation criteria (e.g., 1E-04 per year for total CDF and 1E-05 per year for total LERF) as "surrogates" based on the Safety Goal Policy Statement and QHOs. In support of these surrogates for the current fleet of light-water nuclear power plants, NRC has demonstrated that these are acceptable metrics for the latent and early QHOs using calculations presented in an NRC memo entitled *Transmittal of Technical Work to Support Possible Rulemaking on a Risk-Informed Alternative to 10 CFR 50.46/GDC 35*, Appendix C, "Quantitative Guidelines from the Framework for Risk-Informing 10 CFR Part 50" [4].

[*] Level I PRA determines the CDF and LERF and other release categories, Level II PRA determines the quantity and activity of the radioactive material released from the plant, and Level III determines the health consequences to the public.

3.2. Development of Risk Evaluation Guideline Surrogates from Safety Goals QHOs

Development of risk evaluation guidelines surrogates from the Safety Goal QHOs was performed by first examining comparable dose limits stipulated or referenced by federal and international regulations and associated guidance. Then, the selected dose threshold limits were paired with applicable likelihood limits based on this examination. These limits were then tested and refined to demonstrate that they are equivalent or more conservative than the QHGs proposed in the RIDM report.

The regulations and guidance examined to support development of the surrogate risk evaluation guidelines are the (1) the Q System developed for the International Atomic Energy Agency (IAEA) to support regulation of transport of radioactive materials, (2) risk evaluation guidelines used by DOE for nuclear safety basis development, and (3) NRC endorsed evaluation guidelines in support of licensing advanced reactor design.

The Q System was developed by United Kingdom researchers by MacDonald and Goldfinch [5] in 1983 for the International Atomic Energy Agency (IAEA) to support regulation of transport of radioactive materials. The Q system defines "quantity" limits, in terms of so-called A_1 and A_2 values for radionuclides that are allowed in a Type A package [6, 7]. These limits are also used for several other purposes in the Transport Regulations [], such as in specifying package activity leakage limits for other packages (e.g., Type B(U), Type B(M) or Type C packages). The content limits are set to ensure that the radiological consequences of severe damage to a Type A package are acceptable and design approval by the competent authority is not required, except for packages containing fissile material. The more robust Type B(U) or Type B(M) packages require testing against a range of accidents which expose packages to severe dynamic forces.

Under the Q system, a series of exposure pathways is considered, each of which might lead to persons in the vicinity of a Type A package involved in a severe transport accident receiving external or internal radiation exposure. The effective dose to a person exposed in the vicinity of a transport package following an accident was set not to exceed 50 mSv. This value of 50 mSv or 5 rem corresponds to the annual dose limit for radiation workers and is also the occupational dose limit per year for general employees in the U.S. per 10 CFR 835.202 (Occupation dose limits for general employees) of 10 CFR Part 835 ("Occupational Radiation Protection"). The determination of A_1 and A_2 values do not explicitly take into consideration the likelihood of accident but implies this release from a damaged Type B package is highly unlikely. From this examination, it was concluded that a radiological dose greater than 5 rem is unacceptable if the likelihood of the accident is more than highly unlikely.

The DOE uses the concept of risk evaluation threshold values to support the nuclear safety basis for non-reactor nuclear facilities. The maximum radiological dose to the nearest member of the public as well as the onsite worker are calculated and then evaluated according to defined risk evaluation guidelines. The risk assessment approach used to support the allocation of nuclear safety controls at a DOE nuclear facility is typically not a PRA but rather a qualitative or semi-quantitative risk informed hazard analysis supported by accident analysis, and if needed, by event and fault trees modelling (like the event and fault trees modelling is used in PRA). DOE refers to its process as "risk ranking" in DOE-STD-3009-2014, *DOE Standard – Preparation of Nonreactor Nuclear Facility Documented Safety Analysis* [8]. Their concept of risk ranking is based on characterizing the risk of an activity or facility in terms of the consequence and likelihood of possible accident sequences.

DOE-STD-3009-2014 establishes a Total Effective Dose (TED) of 25 rem as the evaluation guideline for the maximally exposed offsite individual (MOI) and a TED of 100 rem as the threshold for designation of safety controls for the co-located worker CW. It defines the following consequence and likelihood categories for risk-ranking:

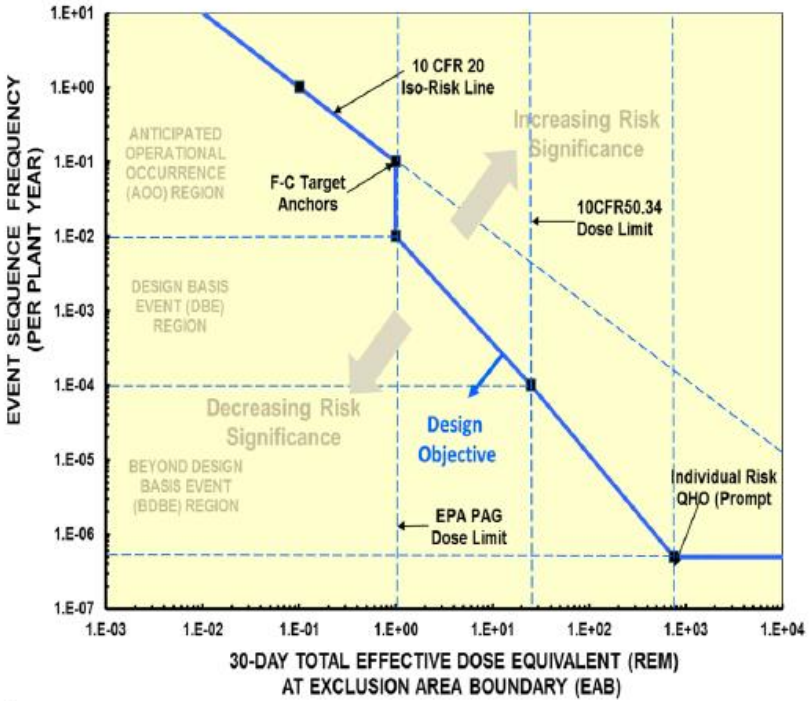
- High consequences for the MOI to be a TED ≥25 rem.
- Moderate consequences for the MOI to be a TED <25 rem but ≥5.

- Low consequences for the MOI to be a TED <5 rem TED.
- High consequences for the CW to be a TED ≥100 rem.
- Moderate consequences for the CW to be a TED <100 rem but ≥25 rem.
- Low consequences for the CW to be a TED <25 rem TED.
- Beyond Extremely Unlikely accidents as having a likelihood $<1\times10^{-6}$ per year.
- Extremely Unlikely accidents having a likelihood of between 1×10^{-4} and 1×10^{-6} per year.
- Unlikely accidents as having likelihood of between 1×10^{-2} and 1×10^{-4} per year.
- Anticipated accidents as having likelihood of $> 1\times10^{-2}$ per year.

Examination of this guidance suggests that a radiological dose of 25 rem to the public and 100 rem TED to workers could be defined as unacceptable if the likelihood of the accident is more than 1×10^{-6} per year. Also, that a radiological dose of 5 rem to the public and 25 rem TED to workers could be defined unacceptable if the likelihood of the accident is more than 1×10^{-4} per year. Of the possible pairs of consequence and likelihood categories from the defined, the lower likelihood, higher consequence criteria will be the most important, given the anticipated robustness of the microreactor package.

The nuclear industry has produced guidance for a risk-informed performance-based and technology-inclusive process to inform licensing of advanced non-light water reactor (LWR) designs. This involves a risk informed approach for selection of license basis events; safety classification of structures, systems, and components and associated risk-informed special treatments; and determination of defense-in-depth adequacy. This approach which has been endorsed by NRC is described in Nuclear Energy Institute (NEI) 18-04, Revision 1, *Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development* [9]. As shown in Figure 1, the approach uses a set of frequency-consequence criteria to define target risk significance based on federal requirements and guidance about acceptable dose for different applications. The likelihood-consequence pairs used to define the blue line which separates the acceptable from the unacceptable regions are complimentary to criteria previously discussed. Accordingly, this set of likelihood-consequence limits were used along with those previously discussed to derive proposed risk evaluation guidelines for microreactor transportation.

Figure 1: Frequency-Consequence Targets from NEI 18-04, Revision 1 [2]



3.3. Proposed Risk Evaluation Guideline Based on Safety Goals QHOs

The risk evaluation guidance discussed above from the three sources was considered and used develop the proposed risk evaluation guidance shown in Table 2 which has adjustments to ensure that it is at least as conservative as the proposed QHGs proposed in the RIDM report. The maximally exposed offsite individual (MOI) is an adult located at the point of maximum exposure to the accident. The term "worker" needs to be defined but is, in general, is more apt to near the accident. The conversion from radiological dose to health effects can be performed using 2002 guidance from the DOE Office of Environmental Policy and Guidance [10]. The guidance in the memo is based on a technical report attached to the memorandum by the Interagency Steering Committee on Radiation Standards (ISCORS[†]). These criteria could be applied to the risk of bounding representative accidents or total risk of all accidents

Table 2: Example Proposed Radiological Dose Evaluation Guidelines

Annual Accident Frequency (per year)[(a)]	Radiological Dose Consequence to the MOI[(b)]	Radiological Dose Consequence to the Worker[(b)]	Risk Acceptability
$\leq 5\times10^{-7}$[(c)]	>750 rem TED [(c)]	>750 rem TED [(c)]	Acceptable
$<1\times10^{-6}$ and $>5\times10^{-7}$	≤750 and >25 rem TED	≤750 and >100 rem TED	Unacceptable
$\leq 1\times10^{-6}$ and $>5\times10^{-7}$	≤ 25 and >5 rem TED	≤100 and >25 rem TED	Acceptable
$<1\times10^{-4}$ and $>1\times10^{-6}$	≤ 25 and >5 rem TED	≤100 and >25 rem TED	Unacceptable
$\leq 1\times10^{-4}$ and $>1\times10^{-6}$	≤ 5 and >1 rem TED	≤25 and >1 rem TED	Acceptable
$<1\times10^{-3}$ and $>1\times10^{-4}$	≤ 5 and >1 rem TED	≤25 and >1 rem TED	Unacceptable
$\leq 1\times10^{-3}$ and $>1\times10^{-4}$	≤1 and >100 mrem TED	≤1 and >100 mrem TED	Acceptable
$>1\times10^{-3}$	≤1 and >100 mrem TED	≤1 and >100 mrem TED	Unacceptable
$\leq 1\times10^{-3}$	≤100 mrem	NA	Acceptable
$\leq 1\times10^{-2}$	NA	≤100 mrem TED	Acceptable

Notes:

(a) Determination of the accident frequency should consider whether more than one shipment a year is considered.

(b) The radiological dose consequence as presented as TED, which is based on Integrated Committed Dose to all organs thereby accounting for direct exposure as well the 50-Year Committed Effective Dose Equivalent.

(c) If the accident frequency is $<5\times10^{-7}$ per year, then the risk of the accident scenario is acceptable, but further analysis may be warranted if the consequences are expected to be exceptionally large.

4. PROPOSED RISK ASSESSMENT APPROACH AND CHALLENGES

This section describes the proposed risk assessment approach for performing a microreactor transportation PRA and the basis for the approach. It also describes key technical challenges and how the PRA can be used to address those challenges.

4.1. Proposed Risk Assessment Approach

The development of a microreactor transportation risk assessment is a novel endeavor, because until now there has not been a need to develop a risk-informed licensing bases for a mobile reactor, and

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

therefore, a technical basis has not been seriously investigated. The licensing of transportation of a microreactor package does not fit cleanly into the existing licensing categories for transportation of nuclear material in an approved containers, casks, or packages or for operation of a stationary nuclear power plant. However, the risk assessments performed for the transportation of radiological material in approved containers is commonly performed and provides some insight. Likewise, the use of PRA for risk-informed applications associated with amending the operating license of light water reactors in the U.S. has become common and the development and review of the PRA models that support such applications is now very mature. However, it is unlikely that the microreactor package will be able to meet the requirements for a Type B package for transporting material with high levels of radioactivity, and hazards exist associated with microreactor transport that are not encompassed by the typical safety basis or the PRA of a stationary nuclear power plant.

NRC Commission paper SECY-99-100 [11], "Framework for Risk-Informed Regulation in the Office of Nuclear Material and Safety and Safeguards," describes the results of an effort to scope the development of a framework for applying risk assessment methods to the regulation of nuclear material uses and waste disposal and makes recommendations to the NRC Commission for how to proceed. This paper and the proposed guidance in RIDM report indicates that for transportation of nuclear material the most appropriate risk assessment method is either a PRA or an Integrated Safety Analysis (ISA). As the RIDM report explains, ISAs are normally qualitative or semi-quantitative assessments, and therefore, are not as effective in producing the quantitatively derived benefits discussed above such as demonstrating that the risk of transport meets accepted safety goals and quantitative risk evaluation guidelines. PNNL-31867 [1] describes historical examples of using risk information and insights to develop the technical basis for regulatory approval of transportation packages. However, none of these past cases were as technically challenging as transport of an microreactor. In fact, past risk informed approvals primarily consisted of showing that transportation accidents leading radiological consequence of any significance were incredible, especially in consideration of stipulated compensatory actions. Accordingly, the quantitative risk assessment approach presented here is a PRA performed to be consistent with the units of measure used in risk evaluation guidelines presented in Section 3.3.

The term "risk" is defined by Kaplan and Garrick (1981), who are two pioneers in PRA especially as it pertains to high-risk highly engineered systems such as nuclear power plants, as a risk triplet that defines the set, $<s_i, f_i, x_i>$, in which S_i, represents the i^{th} scenario (sequence or progression), f_i is the associated frequency, and x_i is the resulting consequence. In simple language, risk is the determination of what can go wrong, how likely is it, and what are the consequences? PRA modelling of accident scenarios typically involves two types of logic analyses: fault and event tree analyses. Fault-tree analysis is a deductive process used for determining combinations of system failures and human errors that could result in the occurrence of defined undesired events. Event-tree analysis, by comparison, uses inductive logic to define accident sequences starting with specific initiating events and the mapping possible subsequent events that lead to different outcomes.

For the most part, complex system analysis (e.g., failure of the control rods to SCRAM or Emergency Diesel Generators to start) using fault trees is not required or beneficial in microreactor transportation PRA. The failures that are considered in the microreactor transportation accident scenarios are primarily the result of the initiating event itself as opposed to subsequent random failures. Also, even though event tree models have been used in past transportation risk assessments, the benefit of their use for this application is seen as limited. The event trees like those shown in NUREG-2125 [13] are useful for sorting out the different kinds of highway vehicle accidents that can occur during transport opposed to defining the course of accident scenarios based on the success or failure of different nodes that correspond to various prevention and mitigation systems functions (e.g., the course of an accident after a large pipe break at a nuclear powerplant). That said, the results of the accident analysis identification process should be reviewed against NUREG-2125. Accordingly, as described above the proposed microreactor transportation PRA approach is not performed with fault and event tree models but focuses on: (1) identification of accident scenarios, (2) development of the likelihood of those scenarios, and (3) development of the consequences of those scenarios.

Accident scenarios begin with initiating event which can be a system upset or failure, a human error, or an external event (e.g., an event outside the system or activity of interest like a natural phenomenon event). Identification of initiating events requires a systematic search across the range of events that can affect the system of interest. There are multiple methods for identifying initiating events for PRA including inductive, and deductive approaches and searching through event data. Inductive approaches (i.e., bottom-up) include use of a hazard identification and analysis are particularly useful for understanding the broad range of possibilities and for when event data may be incomplete. The accident scenarios defined for microreactor transport are not complex in terms of requiring consideration of multiple active prevention and mitigation systems like for nuclear power PRAs. However, a microreactor package transportation PRA is a first-of-its-kind endeavour, and therefore, such an effort appears warranted. Past risk assessments for more typical transportation packages have focused on severe road accidents, but other possibilities for containment failure seem possible such as improper disassembly and packaging. Accordingly, hazards analysis is used as a systemic way to identify and define microreactor transportation accidents that are important contributors to risk and need to be evaluated in detail.

After potential accident scenarios are identified, bounding representative accidents will be defined to reduce the number of accidents and sensitivity studies that will need to be evaluated in detail for their likelihood and consequence. The likelihood analysis will be based on available data and the consequence analysis will follow conventional guidance for determining radiological dose to humans.

4.2. Addressing Risk Assessment Challenges

An important input to any nuclear risk assessment is the radiological material inventory that could be at risk in an accident. The radiological material inventory considered in the microreactor transportation PRA includes irradiated TRISO fuel, but also includes radiological material that has diffused into or has been deposited on reactor structures and systems. A low percentage of defective TRISO fuel particles can make a meaning contribution to radiological material that accumulates in the reactor core structure and gas cooling system as discussed in INL/EXT-16-40784, “A Summary of the Results from the DOE Advanced Gas Reactor (ATR) Fuel development and Qualification Program.” [12] On one hand, TRISO fuel is designed to maintain its integrity during rector core accidents to minimize the risk to the public, on the other hand the microreactor is disassembled and could be subject severe mechanical impacts during transport in a way not considered for typical non-mobile reactors.

The microreactor radionuclide inventory is used in the microreactor transportation PRA to define the Material at Risk (MAR) in a transportation accident that could become the source term in an accident leading to release of radiological material or as the sources of direct radiation exposure. In general, the radiological material that could be released includes the TRISO fuel itself and radiological material that has diffused into reactor core, system, and structures or plated-out in the gas cooling system. The same radiological material along with material that has become radiologically activated could be the source of direct radiation exposure in accident (e.g., control rods and motors, the Reactor Pressure Vessel, copper wires). Direct radiation exposure is a separate radiological dose pathway to workers and the public from material that is released and is inhaled or ingested.

The accident analysis and estimate of dose consequences should consider the following potential contributors to the MAR:

1. Nongaseous fission products from TRISO fuel and heavy metal contamination damaged an accident,
2. Gases from TRISO fuel and heavy metal contamination damaged an accident,
3. Radioactive material that has diffused and is held up the compact and other core structures
4. Radioactive material that has plated-out in the Primary Cooling system,
5. Noble gases in the pressure boundary of cooling system assuming (it’s not evacuated),
6. Clean-up system inventory (if applicable),
7. Contamination outside the reactor.

8. Contamination on and outside the microreactor transportation package

In addition to addressing the uncertainty associated with the possible contributors above, the radiological material inventory is, of course, a function of irradiation time in the reactor and time after shutdown, and therefore is variable. Other key uncertainties concern the damage ratios, release fractions of radiological material, and airborne release fractions associated with various microreactor transportation accidents such as such high energy collisions, rollovers, and fires. Meeting federal deterministic package requirements for hypothetical accident test conditions prescribed in 10 CFR Part 71 ("Packaging and Transportation of Radioactive Material") is likely not feasible, and so, assessing performance of the package in terms of factors that are important to the estimation of radiological dose consequences will likely be a source of modelling uncertainty

In general, the PRA can be used to sort out the risk associated with different kinds of microreactor transportation accidents, and therefore, certain sources of modelling uncertainty may be determined to be irrelevant. Importantly, the PRA can identify microreactor transportation accidents that are important contributors to risk, so that resources can be used to refine risk calculations, identify upgrades the design to reduce risk, or develop prevention and mitigation strategies that reduce risk. It may provide a lot of insight, for example, to know the risk differences between high-consequence low-likelihood accident scenarios such as a collision with a heavy tanker full of flammable or combustible material and lower consequence but higher-likelihood accident scenarios such a loss of microreactor package containment due to shock and vibration during road travel.

Additionally, sensitivity studies using the PRA model can be used to address sources of modelling uncertainty associated with PRA inputs such as the uncertainty associated with determining the damage ratios, release fractions and airborne release fractions discussed above. Sensitivity studies can be performed to examine the impact that variation in radiological material inventory has on risk. Sensitivity studies can also provide important insights about design improvements that could reduce risk and the possible trade-offs in design and risk. Sensitivity studies could be used to examine the risk impact of emergency response factors such as the distance at which barriers to the public at constructed.

Of course, the primary purpose of the microreactor transportation PRA results is to demonstrate that the risk associated with microreactor transport is acceptably low. If risk cannot be shown to be acceptably low compared to risk evaluation guidelines such as those proposed in Section 3.3, then design modifications or compensatory actions will need to be considered.

5. CONCLUSIONS

This paper discusses the regulatory pathway identified to be the most feasible for the licensing transportation of the demonstration microreactor and why a microreactor transportation PRA is needed to support the 10 CDF Part 71 exemption process. The paper explains how PRA can be used to address the challenges associated this first-of-its-kind transportation of a microreactor with irradiated fuel by determining the potential risk to the public if a transportation accident occurs. It also discusses how PRA can be used to provide design insights, help identify compensatory actions to reduce risk consistent with the guidance in 10 CFR 71.12, and how the PRA, itself, can be used to address modelling challenges.

This paper explains that regulatory risk evaluation guidelines do not exist for transportation of nuclear material as they do for nuclear power plants but proposes an approach based on examination of applicable NRC, DOE, and international guidance. The paper proposes a workable approach consistence with regulatory safety goals and philosophy, that still needs to be vetted with regulatory authorities such as NRC. Accordingly, this paper describes an approach for determining the risk associated with microreactor transportation that can be used to show the risk acceptability of a 10 CFR 71 exemption.

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