Methodology Evaluation Report

Docket No. 71-9396

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

Background

In its request dated February 20, 2023 (Agencywide Documents Access and Management System Accession No. ML23066A201), as supplemented on September 18, 2023 (ML23268A328), on behalf of the Strategic Capabilities Office (SCO) within the U.S. Department of Defense (DoD), Pacific Northwest National Laboratory (PNNL) asked the U.S. Nuclear Regulatory Commission (NRC) to review PNNL’s document titled “Development and Demonstration of a Risk Assessment Approach for Approval of a Transportation Package of a Transportable Nuclear Power Plant for Domestic Highway Shipment[1],” hereafter referred to as “the PNNL document.” In the PNNL document, PNNL provided the NRC a risk-informed methodology for review and a numerical demonstration (of the how the risk-informed methodology could be implemented.[2] The risk-informed methodology in the PNNL document consists of PNNL’s frequency-consequence (F-C) plots in figures 4.7, “Proposed Offsite Public Risk Evaluation Guidelines Chart for Transport of a TNPP [Transportable Nuclear Power Plant] Package,” and 4.8, “Proposed Worker Risk Evaluation Guidelines Chart for Transport of a TNPP Package,” and the steps to develop the transportation probabilistic risk assessment (PRA). The numerical demonstration includes the values and assumptions for the values that PNNL used to develop the transportation PRA and it is not intended to represent the actual values and assumptions that would be used in a future application.

The demonstration transportable micro-reactor design, known as Project Pele, consists of multiple modules, including the reactor module and three other modules (an intermediate heat exchange module, a control module, and a power conversion system module) to support the micro-reactor. The four modules are intended to be transported and contained in separate International Organization for Standardization (ISO)-compliant container express CONEX box-like structure boxes[3] with dimensions of about 8 feet wide by 8 feet high by 20 feet long. In its risk-informed methodology, PNNL evaluated transport of the micro-reactor module and not the ancillary equipment needed to support the micro-reactor operations.

In parallel with the reactor design development by BWX Technologies Inc., SCO tasked PNNL with developing a risk-informed methodology using risk insights to support a future application

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1 In its September 23, 2023, supplement, PNNL changed the title of its document. The original title in PNNL’s February 20, 2023, submittal was “Development and Application of Risk Assessment Approach for Transportation Package Approval of a Transportable Nuclear Power Plant for Domestic Highway Shipment.”

2 PNNL is not seeking endorsement of the numerical values used for estimating the risks as part of its demonstration PRA.

3 The Project Pele modules will be transported in custom-developed ISO containers that resemble CONEX boxes. A CONEX box is a large metal, weather-resistant container used to store or ship items. SCO is planning to use a CONEX box-like structure that is about 8 feet wide by 8 feet high by 20 feet long.

for a transportation package. PNNL developed the risk-informed methodology[4] in the event that an application for either the Project Pele demonstration transportable micro-reactor and/or a future DoD tri-structural isotropic (TRISO)-based transportable micro-reactor module[5] cannot meet all the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, “Packaging and Transportation of Radioactive Material,” for hypothetical accident conditions and therefore requires an exemption, as authorized in 10 CFR 71.12, “Specific exemptions.”

Because SCO determined that exemption(s) from the NRC regulations may be needed, SCO developed a two-phase approach to a package application. First, SCO contracted for the development and submittal of the risk-informed methodology to determine whether its proposed approach would be appropriate for use in its package application for the Project Pele TRISO-based demonstration micro-reactor. The second phase would have been an application for the Project Pele demonstration transportable micro-reactor. However, in its letter dated April 15, 2024 (Agencywide Documents Access and Management System [ADAMS] ML24113A069), SCO stated that it would be discontinuing work on the Project Pele transportation application and expressed its continued interest in the NRC endorsement of the risk-informed methodology.

PNNL developed the risk-informed methodology using the TRISO-based Project Pele demonstration reactor characteristics, however, the risk-informed methodology could be applied to any TRISO-based DoD reactor since the methodology does not depend on the specific design of the Project Pele reactor. The DoD views the NRC endorsement of the risk-informed methodology as setting a pathway for developing an application to meet the requirements of 10 CFR Part 71, if the DoD pursues TRISO-based transportable micro-reactors in the future.

The risk-informed methodology proposes to use a PRA to demonstrate the dose to the maximally exposed individual resulting from postulated accidents would meet the specified risk safety guidelines in figures 4.7 and 4.8 in the PNNL document. The risk-informed methodology would be used only if the transportable micro-reactor module package cannot meet the containment and dose rate criteria in 10 CFR 71.51(a)(2). Thus, the risk-informed methodology provided in the PNNL document is only intended to demonstrate that the DOD’s transportable micro-reactor package can meet the requirements for specific exemptions from hypothetical accident conditions and some post-test criteria. PNNL stated that, based on its current view, an application for package approval would not need exemptions from the remaining applicable package approval requirements in 10 CFR Part 71.

PNNL’s PRA results are not based on the actual route that might be used for the transportable micro-reactor or an engineering evaluation of the final design attributes of the Project Pele micro-reactor module. The risk-informed methodology estimates the frequencies on a hypothetical route and uses engineering judgment, rather than quantitative engineering evaluations, to estimate package damage and release of radioactive material. Therefore, the NRC’s endorsement of the risk-informed methodology applies only to the approach used to estimate risks using PRA methods and the guidelines for acceptable levels of risk; it does not

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4 For the purposes of the NRC review, the PNNL risk-informed methodology includes the attributes and methods used to develop a PRA used for estimating the risks from a transportation accident and the frequency and consequence plots used as surrogate guidelines for acceptable levels of risk. The risk-informed methodology does not include the approaches and values used to determine inputs to the transportation PRA as part of the ‘demonstration’ of how the risk-informed methodology would be implemented.

5 The terms “future DoD reactors” or “future DoD transportable micro-reactors” denote any DoD transportable micro-reactor that is TRISO-based. The reactor design may differ from the Project Pele demonstration reactor design, but are still TRISO-based designs.

extend to the basis and adequacy of numerical values and assumptions used to develop inputs to the risk assessment and the estimated values from the risk assessment.

Review Summary

Because there is no specific NRC staff guidance to inform the staff's review of a risk-informed methodology of this nature, the NRC staff reviewed PNNL's methodology considering existing regulatory approaches and methods where relevant to evaluate the risk of the transportation accident scenarios[6] discussed in the following:

- 10 CFR Part 71
- Regulatory Guide 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," issued December 2020 (ML20238B871)
- International Atomic Energy Agency (IAEA) Specific Safety Guide No. SSG-26, Revision 1, "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2018 Edition)," issued June 2022
- Integrated safety analyses in NUREG-1520, Revision 2, "Standard Review Plan for Fuel Cycle Facilities License Applications," issued June 2015 (ML15176A258)
- NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," issued August 2020 (ML20234A651)

The risk-informed methodology, which is limited to road transport of TRISO fuel, includes F-C plots and an approach to perform the risk calculations in a future application for a DoD transportable micro-reactor package approval. The risk-informed methodology includes two F-C plots, one for the maximally exposed individual who is a member of the public and another for workers who may receive an occupational dose.[7] Hereafter, the term "workers" includes individuals who are part of the protection program and could receive an occupational dose.

The risk-informed methodology provides the steps to be used as a basis for developing a PRA for an application for a future DoD transportable micro-reactor package. The risk-informed methodology also includes a logical and structured approach to identify the safety or risk significance of a transportable micro-reactor and potential compensatory measures to be exercised by the shipper. While the F-C plots show PNNL's demarcation between acceptable and unacceptable regions on the plots (figures 4.7 and 4.8 in the PNNL document) based on accident frequency and the total effective dose equivalent (TEDE) for members of the public and workers, understanding both the sensitivity of the analysis results and what is credited for defense-in-depth (DID) will play a role in a future package approval.

[6] An accident scenario is one that leads to unexpected or unintended exposures. Exposures from normal conditions of transport are not considered to be unexpected or unintended.

[7] The definition of *occupational dose* in 10 CFR 20.1003, "Definitions," states that it is a dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. Occupational dose does not include doses received from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released under 10 CFR 35.75, "Release of individuals containing unsealed byproduct material or implants containing byproduct material," from voluntary participation in medical research programs, or as a member of the public.

In addition to development of the F-C plots, the NRC focused its review on integral steps of a transportation PRA including defined relationships among accident scenarios, package evaluation, compensatory measures during shipment, uncertainty analysis, and assessments of DID. An applicant for a future DoD transportable micro-reactor package should perform the evaluations in an iterative fashion as it develops the design and package approval strategies:

A. Development of PRA for package transport

    1. Initiating event and accident sequence analysis

        i. Initiating events

        ii. Accident sequence analysis (evaluation of accident effects on the package)

    2. Source term analysis (i.e., “Level 2”)/how much of what gets released

        i. Subsection: Associated particle sizes

    3. Consequence analysis (i.e., “Level 3”)

        i. Dispersion of any material released to evaluate individual uptake

        ii. Determination of TEDE

    4. Human reliability analysis

    5. Uncertainty analysis (including sensitivity analyses)

B. Defense-in-depth

Consistent with the risk-informed, performance-based[8] principles in the NRC’s approach to regulation and decision-making, approval of a future application will be based on engineering analysis of the final design, risk insights (such as those obtained from the implementation of the risk-informed methodology), and engineering judgment. NRC approval of a package application will depend on the implementation of each of these steps in the risk-informed methodology and the assumptions used along with their justifications. A package application should identify and provide the appropriate level of information needed to satisfy the regulatory requirements for approval of an exemption under 10 CFR 71.12, “Specific exemptions.”

Based on the statements and representations in SCO’s request, as supplemented, the staff determined that the risk-informed methodology provides an acceptable, systematic approach to form the basis for an exemption request. The NRC will review whether the implementation of the risk-informed methodology complies with the requirements in 10 CFR Part 71 in a future transportation package application for a future DoD transportable micro-reactor.

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[8] As stated in SECY-98-0144 issued on March 1, 1999 (ML003753601), a risk-informed, performance-based regulation is an approach in which risk insights, engineering analysis and judgment (including the principle of DID and the incorporation of safety margins), and performance history are used to (1) focus attention on the most important activities, (2) establish objective criteria for evaluating performance, (3) develop measurable or calculable parameters for monitoring system and licensee performance, (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes, and (5) focus on the results as the primary basis for regulatory decision-making.

Chapter 1: Risk-Informed Methodology Overview

PNNL developed the risk-informed methodology to serve as one element for the applicant for a transportable micro-reactor package approval to demonstrate an acceptable level of safety for exemptions from the dose rate and containment criteria for hypothetical accident conditions. The risk-informed methodology, as developed by PNNL, serves as a tool to risk-inform (1) a vendor application for package approval, (2) the design of the relative risk significance of transportable micro-reactor containment and shielding, and (3) the need for transportation compensatory measures. The risk-informed methodology includes establishing risk targets that are intended to be consistent with the level of risk from ionizing radiation to workers and the public from both operating nuclear power plants and fuel cycle facilities that are generally found to be acceptable by the NRC, as discussed more fully below.

A peer review for the PRA that uses the risk-informed methodology in an application for a future DoD reactor package approval would be useful as part of the NRC's risk-informed review. Peer reviews are part of the codes and standards for PRA development as endorsed by Regulatory Guide 1.200 and Regulatory Guide 1.247 (For Trial Use), "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities," issued March 2022 (ML21235A008). The NRC staff also notes that there is no standard for a PRA for transportation package approval or guidance for performing a peer review for this type of PRA; however, NRC staff notes general principles for conducting a peer review would likely apply. Given that this application is the first of its kind, an independent review of the PRA would support an efficient review by the NRC staff.

Chapter 2: Comparison of the Proposed Frequency-Consequence Plot(s) to Existing Risk Criteria

In section 4.0, "Safety Goals and Risk Evaluation Guidelines," PNNL noted that risk evaluation guidelines do not exist for transportation of nuclear material as they do for nuclear power plants that are assembled at the site of operations. Section 4.1, "NRC-Suggested Risk Evaluation Guidelines," discusses the NRC report "Risk-Informed Decisionmaking for Nuclear Material and Waste Applications," Revision 1, issued February 2008, known as the RIDM report (ML080720238), which proposed quantitative health guidelines (QHGs) for determining the acceptability of risk. These QHGs are based on the quantitative health objectives (QHOs) from the 1986 NRC Safety Goal Policy statement (published in Volume 51 of the *Federal Register* (FR), page 30028 (51 FR 30028)) developed for nuclear power plants.

In section 4.2, "Development of Risk Evaluation Guideline Surrogates for Safety Goals," PNNL proposed establishing surrogate measures for the QHGs by selecting likelihood-consequence pair limits based on existing guidance from the U.S. Department of Energy (DOE), the NRC, and the IAEA. Section 4.2.1, "Risk Evaluation Guidelines Used by DOE for Nuclear Safety," discusses the risk evaluation guidelines the DOE uses for nuclear safety. Section 4.2.2, "NRC Performance Criteria for the Integrated Safety Analysis of Nuclear Fuel Cycle Facilities," discusses the performance criteria the NRC uses, as delineated in Subpart H, "Additional Requirements for Certain Licensees Authorized to Possess a Critical Mass of Special Nuclear Material," of 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," and NUREG-1520. Section 4.2.3, "Risk References in the IAEA Q System," discusses the IAEA Q System described in appendix I, "The Q System for the Calculation and Application of A_1 and A_2 Values," to IAEA SSG-26. Section 4.2.4, "NRC-Endorsed Risk-Informed Methodology to Support the Licensing of Advanced Reactor Designs," discusses the risk-informed approach for selecting licensing basis events described in the Nuclear Energy Institute (NEI) technical report NEI-18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” issued August 2019 (ML19241A472). The NRC endorsed the risk-informed approach from NEI 18-04 in Regulatory Guide 1.233, Revision 0, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certification, and Approvals for Non-Light-Water Reactors,” issued June 2020 (ML20091L698).

Based on information from the above sources, section 4.2.5.4, “Selection of the Likelihood-Consequence Pair Limits for the Surrogate Risk Evaluation Guidelines,” proposes likelihood-consequence pair limits that are more conservative than the RIDM QHGs. These limits are converted to health effects and compared to the applicable RIDM QHGs in table 4.6, “Comparison of Selected Dose-Consequence Limit Surrogates to the Limiting QHGs.”

Finally, section 4.3, “Proposed Surrogate Risk Evaluation Guidelines Established to Meet the Safety Goal QHOs,” uses the likelihood-consequence pair limits to propose risk evaluation guidelines in the form of F-C curves against which bounding representative accidents (BRAs) will be compared for acceptability. PNNL suggested that the BRAs will be conservative enough to be compared to the QHGs separately, without summation. The proposed guidelines for the public and workers are shown in table 4.7, “Proposed Radiological Risk Evaluation Guidelines.” Additionally, figure 4.7, “Proposed Offsite Public Risk Evaluation Guidelines Chart for Transport of a TNPP Package,” shows the proposed guidelines for the public compared to the F-C target from NEI-18-04. The guidelines for workers are shown in figure 4.8, “Proposed Worker Risk Evaluation Guidelines Chart for Transport of a TNPP Package.” The F-C curves are formulated in units of accident frequency (per year). In this case, the accident frequencies are based on an accident during transport of one micro-reactor module shipment per year, so the frequencies are calculated per trip multiplied by one trip per year to produce units of accidents per year.

<u>NRC Review</u>

The PNNL document uses multiple references to inform the development of risk criteria to evaluate the acceptability of transporting a transportable micro-reactor module package and specifically ties the proposed guidelines to the QHGs defined in the RIDM report. Because the NRC developed or approved several of these references, including the RIDM report, they provide a reasonable basis for creating and evaluating the proposed risk evaluation guidelines.

Section 4.2.5.4 of the PNNL document describes the comparison of the proposed risk evaluation guidelines to the QHGs, which are also summarized in table 4.6. There are limitations to this comparison. The QHGs, as stated in the RIDM report, are intended to be calculated as the risk to an “average individual” within “the relevant population at significant risk” from the licensed activity. Conversely, the risk-informed methodology suggests calculating the risk to the maximally exposed individual for each BRA. In section 4.3, the PNNL document suggests that BRAs will be calculated in a manner that is sufficiently conservative that aggregation is unnecessary and would be overly conservative. The potential nonconservatism of not aggregating the BRAs can be limited by not overly subdividing into an excessive number of BRAs. The subdivision performed for the risk-informed methodology description was not excessively subdivided and is therefore reasonable.

The PNNL document’s comparison to the QHGs is conservative in that it suggests calculating the risk to the maximally exposed individual rather than averaging the risk over a population. The difference in the risk to the maximally exposed individual and the average individual among a population will depend on many factors, including (1) the area/population averaged over,

(2) the amount of dispersion, and (3) protective actions limiting maximum dose (and risk) more than average dose. As noted in section 4.3, in the case of a transportation package, the population is spread out along the route. The difference between the maximally exposed individual at 25 meters and the average among a population spread over the entire route (1289 miles) would be quite large. The difference may not be as large for workers, depending on how the population is defined. Considering these differences, for purposes of a transportable micro-reactor module package, the staff considers that the proposed risk evaluation guidelines are bounded by the QHGs.

The risk criteria in NRC-endorsed guidance for fuel cycle facilities (NUREG-1520) and advanced reactor designs (NEI-18-04, endorsed by Regulatory Guide 1.233) both consider sequences separately and doses at the site boundary rather than doses averaged across a population. For the public, the risk evaluation guidelines proposed by the risk-informed methodology are less than these criteria, except for a small portion of the proposed guidelines exceeding the NEI-18-04 risk line, as shown in figure 4.7. For workers, the risk evaluation guidelines proposed by the risk-informed methodology are about an order of magnitude lower than the likelihoods identified in NUREG-1520, to the extent that they overlap.

In its review of the risk-informed methodology, the NRC expected an application for the Project Pele demonstration reactor consistent with the demonstration PRA, which was for two transports, one per year, one away from and one back to INL. The NRC review did not consider the risk associated with multiple shipments in a year (e.g., the higher risk to persons living along a route for which there are multiple shipments a year). NRC endorsement is limited to use of the method in a scenario with limited transports. The method could possibly be used in scenarios where there are multiple shipments per year, with additional considerations and/or justifications.

The NRC reviewed PNNL’s proposed risk evaluation guidelines illustrated by figures 4.7 and 4.8 which are on a per year basis for two trips, one per year, and finds the F-C targets to be consistent with NRC risk-informed approaches and, therefore, acceptable for forming the basis of an exemption request under 10 CFR 71.12, as part of a transportation package application for a future DoD transportable micro-reactor.

Chapter 3: Elements of the Probabilistic Risk Assessment

3.1 Initiating Events and Accident Sequence Analysis

The risk-informed methodology describes the process for identifying hazardous conditions and grouping them into accident scenarios for further analysis.

3.1.1 Initiating Events

To identify initiating events, the risk-informed methodology first defines the safety functions of the TNPP package. Similar to other transportation packages, the safety functions are identified in section 5.2, “Identification of TNPP Package Safety Functions,” as containment of radiological material, radiation shielding, and maintaining criticality safety. In addition to these safety functions, that can be considered primary safety functions, other supporting safety functions, such as structural analyses, heat dissipation, or materials analyses may contribute to the dose rate, containment, and criticality safety and should be identified. As discussed in section 5.3.1, “Approach to Development of Accidents Scenarios,” PNNL expected that the accident scenarios would not be complex and thus uses hazard analysis to explore a broad range of possibilities rather than complicated system interactions. PNNL compared the results to NUREG-2125, "Spent Fuel Transportation Risk Assessment," issued January 2014 (ML14031A323), to review the comprehensiveness of the process.

Section 5.3.2, "Identification and Assessment of TNPP Transportation Hazardous Conditions," discusses the details of the hazard identification. PNNL subject matter experts filled out worksheets to generate a catalogue of hazardous conditions (grouped by category and including descriptions) as well as preliminary estimations of frequency, consequence, risk, and other factors for each identified scenario. Appendix B, "Evaluation of TNPP Package Transportation Hazardous Conditions," to the PNNL report documents the results of this process.

3.1.2 Accident Sequence Analysis (Evaluation of Accident Effects on the Package)

The risk-informed methodology includes evaluations of accident sequence analysis, the potential for fuel release and dispersion, and dose consequences to a worker or bystander. The numerical demonstration in the PNNL document does not include engineering analysis to determine damage to the packaging or contents as such analysis would be developed when the risk-informed methodology is applied in an application for a transportable micro-reactor package approval. As discussed in section 5.3.3.1, "Delineation of Accident Scenarios from the Identified Hazardous Conditions," the demonstration of the risk-informed methodology initially assigns six accident consequence groups based on expected (not calculated) contributors to the material at risk (MAR). The NRC reviewed the approach PNNL used to determine the MAR, but not the numerical assumptions and estimated results for the accident consequence groupings. The NRC expects that an applicant for a transportable micro-reactor package approval using the risk-informed methodology would either provide an analysis of the expected MAR for each accident or justify the consequence groupings.

Consistent with the risk-informed methodology, for each accident sequence, an applicant for a transportable micro-reactor package approval would use appropriate initial and boundary conditions and evaluate damage to the package based on the accidents, which would include any material degradation such as corrosion, oxidation, or radiation. These conditions could include the impact energy in the event due to truck speed and any other impacting traffic, the most damaging orientation of the package (unless only one specific orientation is possible), and thermal energy input in the case of a fire. The accident sequence evaluation should include a structural evaluation of the package either as it would be prepared for shipment, or in a damaged state if any of the normal conditions of transport affect the package performance during an accident. The results of these structural and thermal evaluations should include damage to the package, including amount of degraded shielding, leak paths for radioactive material release, and the quantity of damaged fuel, if any, to determine the amount of radioactive material released for the source term analysis.

3.1.3 Estimating the Likelihood of Accident Scenarios

Section 6, "Development of Estimates of the Likelihood of Occurrence of the TNPP Transportation Accident Scenarios", of the PNNL document describes the development of estimates of the likelihood of accidents occurring, including the bases for those estimates. The NRC considers the estimates of the likelihood of accident scenarios as inputs used in PNNL's demonstration of the methodology. Although these accident scenario inputs are necessary for the demonstration of how the risk-informed methodology would be applied, these inputs are not being considered as part of the NRC's endorsement of the risk-informed methodology. The technical basis and adequacy of the numerical values and assumptions used in estimating transportation risks will be an important part of an NRC review, should an applicant for a DoD transportable micro-reactor package approval submit a request to the NRC using the proposed risk-informed methodology. NRC's endorsement of the risk-informed methodology does not represent approval of the basis and adequacy of numerical values and approaches used in demonstrating the risk-informed methodology in the PNNL document.

The staff notes that an application for a package approval for a future DoD transportable micro-reactor using the risk-informed methodology will need to provide adequate justification to support the numerical values used to estimate risk. As such, likelihood estimates will need to justify specific values and ranges, including any specific approaches used to estimate accident frequency data, conditional factors applied to obtain likelihoods, and its applicability to specific route considerations. Additionally, as discussed later in this review, an uncertainty and sensitivity analysis conducted as part of the PRA should further inform the significance and importance of specific numerical values. The NRC expects the level of justification of the approaches and numerical values in an application for a future DoD transportable micro-reactor package will be commensurate with its importance to safety.

3.2 Source Term Analysis

3.2.1 Radionuclide Inventory

The risk-informed methodology includes determining the source term potentially available for release during an accident and the remaining radioactive material to determine the TEDE to the maximally exposed individual. The source term and remaining radioactive material can be determined by assessing the following:

- radionuclide inventory of the package
- MAR
- quantity of material in various components of the package that could be released in an accident as it migrates out of the TRISO fuel towards the containment boundary

In its source term analysis in the numerical demonstration, PNNL calculated the radionuclide inventory based on potential reactor operations. In section 5.1 of the PNNL document, PNNL provided its basis for estimating the radionuclide inventory, which includes the operational characteristics, operating history of the reactor and the cooling time before shipment. In appendix A, PNNL estimated the radionuclide inventory from 30 days to 2 years after shutdown following 3 years of operation.

In section 5.1.3, "Estimated Radiological Inventory," of the numerical demonstration, PNNL performed a two-phase screening of radionuclides to identify the radionuclides with nonnegligible contributions to dose rates in the event of potential accidents. The first phase screened out all radionuclides with an inventory less than 1 millicurie.

For the second phase, PNNL used the A_2 value[9] to screen out small quantities of radionuclides with low dose potential. PNNL stated that any radionuclide without a corresponding A_2 value in

[9] As defined in 10 CFR 71.4, "Definitions," the A_2 value "means the maximum activity of radioactive material, other than special form material, LSA [low specific activity], and SCO [surface contaminated object] material, permitted in a Type A package." The radionuclides with smaller A_2 values have a larger hazard, because by the Q System radionuclides are normalized based on radiological hazard to a human receptor. The derivation of the A_2 values in Appendix A to 10 CFR Part 71 is determined using the Q System, as discussed in appendix I to IAEA SSG-26.

table A-1 in appendix A, "Determination of A_1 and A_2," to 10 CFR Part 71 was included in the list of radionuclides if "its activity was greater than or equal to 0.001 (0.1%) of the appropriate A_2 value from 10 CFR 71, appendix A Table A-3." This means that the activity of the radionuclide would be greater than 5.4×10^{-4} curies for beta- or gamma-emitting radionuclides and 2.4×10^{-6} curies for pure alpha emitters and neutron emitters. Based on its screening analysis of radionuclides at a 90-day cooling time, PNNL stated that more than 99.99999 percent of the radionuclide inventory would be included in the consequence analysis.

3.2.2 Material Available for Release

In section 5.1.4.1, "Development of Estimates of Material at Risk for Different Accidents," PNNL stated that there are two types of MAR. The first is radioactive material that migrates out of the TRISO fuel during normal reactor operations, anticipated operational occurrences, and potential transportation accidents. The second type of MAR is activation products in the reactor components. Radioactive material released from the TRISO due to a transportation accident will contribute as accident-related MAR. In section 5.1.4.1, PNNL named the primary (radioactive material released from TRISO) and secondary (activated components) contributors to the MAR but stated that the secondary contributors would be negligible. To determine the MAR in the numerical demonstration, PNNL developed release fractions from the TRISO fuel based on TRISO fabrication errors, and fuel failure in an accident and attenuation factors as material migrate from the TRISO kernel through the fuel compact.

As described in section 7.1, "Source Term Methodology for Transportation Accident Scenarios," PNNL used the following equation from the DOE Handbook DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities," issued December 1994 (reaffirmed 2013), to determine the airborne release of the source term:

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

where: MAR = the material at risk
DR = damage ratio (i.e., fraction of MAR affected by stresses due to the accident)
ARF = airborne release fraction
RF = respirable fraction
LPF = leak path factor

PNNL developed the MAR for three regions within the transportable micro-reactor module package to account for radioactive material that migrates out of the TRISO and its reduction due to deposition on surfaces within the package as the radioactive material migrates toward the exterior of the packaging. The three regions in the package include the TRISO particles, the compact and other core structures (fuel assemblies in coolant channels, moderator blocks, and control rods), and the reactor pressure boundary (reactor vessel and primary cooling system). The numerical demonstration in the PNNL document estimates diffusion of radioactive material out of the TRISO particles during normal operation and releases from the TRISO due to manufacturing defects. PNNL used the fabrication and failure parameters along with the attenuation factors to develop the TRISO release fractions. PNNL then used the release fractions to calculate the MAR that would be available for release from the package in an accident to contribute to inhalation/ingestion dose and the remaining material to determine the direct dose.

The numerical demonstration in the PNNL document contains an estimate of the fraction of material that is respirable based on a particle size with an aerodynamic equivalent diameter of 10 microns or less. For each accident, an estimate for the damage ratio is based on the energy in the accident, the physical phenomena that causes the release, and the physical and chemical forms of the MAR. While the numerical demonstration provides the estimate of damaged fuel and airborne releases for accident-generated stresses for each of the three regions in table 7.2, "Damage Ratios for the Bounding Represented Accidents," and table 7.3, "Combined Airborne Release Fractions and Respirable Fractions for Represented Accidents," respectively, the estimates are not based on structural and thermal evaluations performed on the package, and therefore the NRC did not review these numerical values and assumptions. The NRC expects that an applicant for a DoD transportable micro-reactor package approval would either develop these release estimates, along with the leak path factors, based on structural and thermal evaluations on the package due to an accident or otherwise justify their assumed factors.

In section 7.2, "Description of the Source Term for Each Bounding Representative Accident," in the numerical demonstration, PNNL estimated the source term (overall release fraction for the MAR) for each accident using the values it provided earlier. For the same reasons discussed related to the development of the factors developed in section 7.1, the NRC did not review the overall numerical assumptions and estimated results for the source term.

<u>NRC Review</u>

Screening radionuclides that contribute less than a millicurie may be appropriate because the smallest A_2 value listed in appendix A to 10 CFR Part 71 is 2.4 millicuries for actinium-227; any quantity less than a millicurie would be authorized for shipment in a Type A package. Type A packages are self-certified under the U.S. Department of Transportation regulations and are not required to withstand accidents. In the Q System, for evaluation of a Type A quantity, the entire contents are assumed to be released in an accident and the effective or committed effective dose to an individual would be less than 5 rem after an accident, which for 1 millicurie would be less than 2 rem for a radioisotope with an A_2 greater than 2.4 curies. In NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," issued March 2007 (ML071340012), the NRC staff used screening of radionuclides based on A_2. However, if the TEDE for an accident is close to a dose threshold, an applicant for a future DoD transportable micro-reactor package approval should perform a sensitivity analysis on all radionuclides eliminated to show that their contribution is negligible.

The NRC reviewed PNNL's process for determining the quantity of MAR, not the values and assumptions, and finds it acceptable. The process developed by PNNL provides an approach that an applicant for a future DoD transportable micro-reactor package approval could use to develop the quantity of MAR available for release from the package based on the inventory of material that migrates out of the TRISO core and containment boundary. However, an applicant for package approval for a future transportable micro-reactor package should justify their assumptions for radioactive material that is not included in the dose analyses, such as secondary contributors from irradiated reactor components. Additionally, the release fraction approach used in the risk-informed methodology is consistent with other documented approaches used in estimating releases from containerized spent fuel, such as in NUREG-1864 (dry cask storage risk assessment), and NUREG-2125 (transportation package risk assessment).

3.3 Consequence Analysis

PNNL proposed basing the dose consequences in risk-informed methodology on the Q System, as described in appendix I to IAEA SSG-26, to determine the dose to an individual. The IAEA uses the Q System to determine the maximum quantity limit for material in a Type A package and establish leakage rate limits for Type B packages based on radiation dose consequences for human receptors. The risk-informed methodology uses the same exposure pathways as the Q System because these exposure pathways were agreed upon by the IAEA Member States to be the dominant pathways for the public and workers who could be exposed to radioactive material released during a transportation accident. The Q System exposure pathways include external photon dose, external beta dose, inhalation dose, skin contamination and ingestion dose, and dose from submersion in a cloud.

In the risk-informed methodology, consistent with IAEA SSG-26, PNNL did not include the ingestion dose or the dose due to submersion in a cloud. SSG-26 does not include the ingestion for a transportation accident because, as stated in item I.45 of appendix I to SSG-26, “the inhalation pathway will normally be limiting for internal contamination and, therefore, explicit consideration of the ingestion pathway is unnecessary.” It is likely that, after an accident contaminates land, there would either be remediation of the land to levels for unrestricted access or the land would be cordoned off. The submersion doses calculated in appendix I to SSG-26 are for rapid depressurization indoors, so PNNL did not include the submersion dose because the accident is assumed to take place outside and release from the package would be a puff release.

3.3.1 External Photon Dose

For direct photon dose to a worker, PNNL included both the material released that is too large to be inhaled and is therefore deposited on the ground, and the remaining radioactive material left in the damaged package. For the external photon dose, PNNL stated that it increased the source term in section 7.2, “Description of the Source Term for Each Bounding Representative Accident,” by a factor of 100 to account for total material released rather than just the respirable material. PNNL used dose coefficients from IAEA SSG-26, which does not account for dispersion for material released from the package and assumes the receptor is 1 meter away from any released material. For external dose to a member of the public, PNNL assumed that the person is located 25 meters from the package compared to 1 meter for the worker. PNNL based this assumption on emergency responders isolating the area around the release for 25 meters in all directions, according to the U.S. Department of Transportation and the Pipeline and Hazardous Materials Safety Administration’s “2020 Emergency Response Guidebook[10],” issued September 2020. While the Emergency Response Guidebook does include an isolation distance of at least 25 meters, this is after emergency response personnel arrive on the scene. An applicant for a future DoD transportable micro-reactor package approval using the risk-informed methodology should justify the distance to a member of the public, as it is possible that such an individual could receive the maximum direct photon dose before emergency response personnel could cordon off the area. The risk-informed methodology calculates external dose due to unreleased material for an individual standing 1 meter from the package with potentially degraded shielding.

3.3.2 External Beta Dose

In section 7.3.3, “External Dose Due to Beta Radiation,” PNNL stated that, like in IAEA SSG-26, the external beta dose in the risk-informed methodology is based on skin contamination due to material released from the package for a person 1 meter away from the accident. The risk-informed methodology’s dose coefficients include a range of shielding factors that depend on beta energy and are based on an absorber thickness of approximately 150 milligrams per square centimeter. PNNL did not include explicit calculations of annihilation radiation as it is

[10] The Emergency Response Guidebook is a joint venture between U.S. DOT PHMSA, Transport Canada’s CANUTEC, and Mexico’s Secretaría de Comunicaciones y Transportes.

expected to be a small contribution to potential doses. PNNL included the 0.51 megaelectron volt gamma rays in the photon energy per disintegration in the derivation of the photon dose coefficients for the radionuclides. The risk-informed methodology includes conversion electrons as monoenergetic beta particles. Similar to the external gamma dose, the dose coefficients do not account for dispersion. In addition, PNNL stated that it increased the source term in section 7.2, “Description of the Source Term for Each Bounding Representative Accident,” by a factor of 100 to account for the total material released rather than just the respirable material.

3.3.3 Inhalation Dose

As discussed in section 7.3.4, “Inhalation Dose,” the risk-informed methodology used the estimated airborne source term and a human uptake value of 1×10^{-3} based on the discussion in item I.36 in appendix I to IAEA SSG-26 about determination of a receptor within 10 meters of an outdoor release. SSG-26 estimates that human uptake at 100 meters is increased by a factor of approximately 30 at a 1-meter distance rather than 100 meters from the package. PNNL used this information to develop a scaling function to determine the uptake factor at 25 meters for the public. SSG-26 then converts the uptake to dose to a human receptor based on the effective dose coefficient for inhalation as provided in table II.2, “Dose and Dose Rate Coefficients of Radionuclides,” of appendix II, “Half-Life and Specific Activity of Radionuclides, Dose and Dose Rate Coefficients of Radionuclides and Specific Activity,” to SSG-26.

3.3.4 Skin Contamination Dose

PNNL proposed calculating the dose due to skin contamination using the source term from section 7.2 in the risk-informed methodology along with the equivalent skin dose rate per unit activity per unit area of the skin found in table II.2 of appendix II to SSG-26. While the risk-informed methodology does calculate the dose to the skin, the risk-informed methodology presumes that workers handling radioactive material after an accident would be wearing appropriate protective clothing. Therefore, the risk-informed methodology calculates the skin dose in the radiation dose consequences; however, PNNL neither included it in the risk results nor used it for comparison with the risk evaluation guidelines.

3.3.5 Neutron Dose

The Q System includes neutron dose due to californium-252; however, there are additional radioisotopes in spent fuel that decay due to spontaneous fission, such as some plutonium, uranium, and curium radioisotopes. PNNL estimated the dose due to these isotopes and stated that for these other isotopes, the dose is dominated by curium-242 and curium-244, consistent with studies completed on light-water reactor spent fuel. Based on its assessment, PNNL estimated that the dose due to neutrons is less than one half percent of the dose due to photons. In addition, based on its literature review, PNNL estimated that the dose due to alpha reactions causing neutron emission is less than 10 percent of the dose due to spontaneous fission. The NRC staff’s experience with evaluating spent fuel packages for light-water reactor fuel is that areas with significant amounts of dense materials such as steel and lead, which may have small amounts of material to stop neutrons, can have a neutron dose rate contribution greater than or equal to the gamma dose rate. Therefore, an applicant for a future DoD transportable micro-reactor package approval should justify whether the neutron dose is negligible or evaluate the neutron dose to the maximally exposed individual.

3.3.6 Exposure Pathways Not Addressed by the Q System

PNNL proposed not including the pathways excluded in the Q System for exposure to radioactive material. The exposure pathways used in the Q System were judged by the IAEA Member States to be the dominant pathways for the public and worker exposure to radiation resulting from a transportation accident involving radioactive materials. While other exposure pathways (e.g., resuspension, skyshine, drinking water ingestion) could be evaluated using IAEA SSG-26, PNNL stated that it did not anticipate significant exposure pathways for transportation accidents involving irradiated fuel, as it expected such incidents to be mitigated by emergency response to a transportation accident.

3.3.7 Radionuclides Without Dose Coefficients in IAEA SSG-26

Based on its screening analyses, PNNL determined that IAEA SSG-26 does not include an effective dose coefficient for six radionuclides: barium-136, tritium, yttrium-89m, promethium-146, antimony-127, and terbium-161. The analysis includes barium-136 and yttrium-89m since they are decay products of cesium-136 and strontium-89, respectively. As discussed in item I.51 of appendix I to SSG-26, progeny with a half-life less than 10 days "are assumed to be in secular equilibrium with the longer lived parent; however, the progeny's contribution to each Q value is summed with that of the parent." PNNL deemed the radionuclides tritium, promethium-146, antimony-127, and terbium-161 to have a negligible source term and dose contributor.

<u>NRC Review</u>

The NRC staff reviewed the approach for consequence calculations in the PNNL document but did not review the numerical assumptions and estimated results in the numerical demonstration for the dose calculations because, as discussed above, the effects of an accident on the package were not explicitly evaluated. The NRC staff determined that an applicant for a future DoD transportable micro-reactor package using the risk-informed methodology for package approval would likely determine all appropriate doses to an individual, whether a worker involved in the shipment or a member of the public, and that the equations used to calculate dose are representative or bounding. However, the staff notes that exclusion of submersion dose is based on the accident occurring outdoors and a puff release from the package. If an accident were to happen that is a prolonged release or were to take place in a tunnel, then an applicant for a future DoD transportable micro-reactor package should justify neglecting the submersion dose.

The NRC notes that the IAEA uses the Q System to determine A_1[11] and A_2 values, along with release limits for Type B packages containing normal form material. The A_1 and A_2 values in the Q System are based on limiting the dose to 5 rem to an individual, assuming the package releases all of its contents. Similarly, the maximum quantity of material that may be released from a Type B package after the tests for hypothetical accident conditions is also limited to an A_2, which may be a small fraction of the material in the package. These limits set by the Q System and codified in 10 CFR 71.51(a)(2) presume a limited amount of material is released by a Type A or Type B package. Using the risk-informed methodology could mean that the package releases more than the allowed limit in 10 CFR 71.51(a)(2) after the tests for hypothetical accident conditions. Therefore, an applicant for a future transportable micro-reactor package approval should evaluate whether exposure pathways that are neglected in the

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[11] As defined in 10 CFR 71.4, the A_1 value "means the maximum activity of special form radioactive material permitted in a Type A package." Special form material is also defined in 10 CFR 71.4.

Q System (e.g., resuspension, skyshine, ingestion) provide a significant dose to a member of the public.

The NRC staff also notes that the risk-informed methodology did not include dose consequences for some scenarios with a frequency less than 5×10^{-7} per year; however, a package application for a future DoD transportable micro-reactor using the risk-informed methodology should address doses to a worker and member of the public for these scenarios to ensure that there are no cliff edge effects, such as a change that could result in an inadvertent criticality.

The NRC staff found that, if followed with appropriate justifications, as discussed above, the risk-informed methodology provides an acceptable basis to form the technical content of an exemption request in an application for a future DoD transportable micro-reactor package in accordance with the regulations in 10 CFR Part 71. While the NRC agrees that the risk-informed methodology provides a systematic approach that could be used to form the basis for an exemption request, there remains a substantial number of unknowns or unjustified numerical assumptions in the numerical demonstration that would need to be determined (either by testing, analysis, or justifiable assumptions) prior to submission of a package application for a future DoD transportable micro-reactor. In addition, the applicant for package approval should justify any dose pathway that it neglects in calculation of the TEDE.

Chapter 4: Human Reliability Analysis

Unlike operating reactors, there are no active systems associated with the transportable micro-reactor module that require human action to mitigate an accident. In the risk-informed methodology, PNNL considered two possible human errors and calculates the probability for each: during reactor disassembly for transport and during transportation package preparation. The only other human error would be driver error, which would be included in accident rates. The NRC assumes that the applicant would follow a quality assurance program during fabrication and package preparation, which should help reduce the potential for human error. The estimates provided by PNNL provide a conservative estimate of the frequency of human errors that may affect the package during an accident. A package application for a future DoD transportable micro-reactor should describe the human actions that could affect the package performance during transport and determine whether they are sufficiently limited such that reasonable estimates of the human actions do not require more detailed human reliability analysis methods than those used in the PNNL document.

Chapter 5: Uncertainty Analysis (Including Sensitivity Analyses)

In section 10.1, “Role of Uncertainty in PRA,” the PNNL document references relevant discussions of uncertainty in NRC guidance documents such as Regulatory Guide 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” issued January 2018 (ML17317A256); Regulatory Guide 1.200; NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” issued March 2017 (ML17062A466); and the RIDM report, as well as requirements found in national PRA consensus standards as endorsed by the NRC (such as in Regulatory Guide 1.200). The risk-informed methodology addresses model uncertainty and the variability of very large truck (semitrailer) crash data.

PNNL analyzed modeling uncertainty using sensitivity studies, which are primarily discussed in section 9, “Sensitivity Study on PRA Modeling Inputs.” As a first step, PNNL considers lists of assumptions found in the PRA and determines whether they could be candidates for a sensitivity study. PNNL considered assumptions for the hazardous condition evaluation, the accident likelihood determination, the consequences determination, and the compensatory measures to reduce or mitigate risk. In the second step, PNNL further screened the candidate sensitivities for feasibility, as some may be difficult to study because of a lack of data. The results of the sensitivity studies are compared to the risk evaluation guidelines, particularly whether the risk conclusions are sensitive to the assumptions. Finally, PNNL discussed insights from the sensitivity studies in section 9.3, “Insights Gained from the Sensitivity Studies.”

The NRC reviewed the steps taken to define, perform, and analyze the sensitivity studies and determined that the process PNNL used is acceptable and sufficient for identifying, characterizing, and understanding the impacts of the key sources of model uncertainty.

For parametric uncertainties, PNNL noted that for reactor PRAs, NRC guidance recommends using mean values, which necessitates the development of probability distributions. PNNL stated that the development of probability distributions would be difficult for a transportable micro-reactor module PRA, as it is the first of its kind and the data may not be available to develop the probability distributions for some likelihoods. In section 10.2.2, “Evaluation of the Impact of the Variability in the Very Large Truck Crash Data,” PNNL estimated the variability of very large truck crash rates by considering subsets of the available data. While it is not necessary to develop full parametric uncertainty distributions for a package application for a future DoD transportable micro-reactor, the NRC would expect the applicant would use the risk-informed methodology to examine other key parameters to establish high and low estimates to demonstrate that there are no cliff edge effects, unless the risk results (likelihood and consequences) are demonstratively bounding.

Chapter 6: Defense-in-Depth

In section 11, “Defense-in-Depth and Safety Margin Concerns,” PNNL articulates a DID approach for the transportation of packages containing a transportable micro-reactor module based on the following: (1) the TRISO fuel and package containment is robust, (2) the support of safety functions during transport do not rely on active systems, (3) the transportable micro-reactor module package transportation risk is quantified and shown to be low, (4) sensitivity studies show that most sources of uncertainty in PRA modeling assumptions and inputs do not impact the conclusions about risk from accidents, and (5) compensatory measures will be administered but were not credited in the transportable micro-reactor module package PRA to reduce risk to workers and the public as well as reducing the risk uncertainties through preventive and mitigative actions and features. The following are key aspects of the DID approach in section 11 associated with the demonstration design:

- Passive elements of the design provide safety functions (e.g., they are not reliant on alternating current power or operator intervention):
  - TRISO fuel limits release and is designed to tolerate the elevated temperatures that can occur in nuclear power plant accidents.
  - The reactor coolant boundary acts as a containment boundary, including isolation devices for disconnected piping.
  - The reactor module will absorb much of the energy of an impact crash, protecting the reactor coolant boundary from more significant damage.

- – The CONEX-like box provides another barrier for the release of radionuclides.
- – Rod-locking mechanisms prevent reactivity insertion.
- – Radiation shielding is provided by the reactor and supplemented by transport shielding in the walls of the CONEX-like box.

- The PRA appears to indicate that the risk of the safety systems failing is low, and the package application for a future DoD transportable micro-reactor will include verification of the performance of the safety features and components in (e.g., release of radiological material from a transportation accident involving significant mechanical forces is very small).

- Administrative transport controls not credited in the risk-informed methodology's PRA will be applied to limit potential accidents and help assure the reliability of the safety systems:

    - Escort vehicles travel in the front of and behind the truck carrying the transportable micro-reactor module package.
    - Routes are selected that avoid bodies of water and other measures are taken, as necessary, such as inspecting bridges over bodies of water, closing bridges to other traffic, and scheduling shipments to avoid high winds while on bridges.
    - Shipments are transported at night to avoid high traffic.
    - Fire detection and suppression systems are installed on the transport vehicle.

- Recovery plans will be in place in the event of transportation incidents and accidents:

    - – The transportation personnel are trained in emergency response procedures (e.g., setup of a safety perimeter).
    - – PRA sensitivity studies are used to assess worker and public safety (distance from accident location and duration of exposure) to enhance recovery response.

<u>NRC Review</u>

The NRC articulated its approach to applying the principles of DID for ensuring safe use of radioactive material for both reactor and materials applications. Key aspects of this DID approach are detailed in "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," dated March 1, 1999 (ML003753601):

> The concept of defense-in-depth* has always been and will continue to be a fundamental tenet of regulatory practice in the nuclear field, particularly regarding nuclear facilities. Risk insights can make the elements of defense-in-depth clearer by quantifying them to the extent practicable. Although the uncertainties associated with the importance of some elements of defense may be substantial, the fact that these elements and uncertainties have been quantified can aid in determining how much defense makes regulatory sense. Decisions on the adequacy of or the necessity for elements of defense should reflect risk insights

gained through identification of the individual performance of each defense system in relation to overall performance.

*Defense-in-depth is an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance, and operation is that the facility or system in question tends to be more tolerant of failures and external challenges.

NUREG-2150, "A Proposed Risk Management Regulatory Framework," issued in April 2012 (ML12109A277), provides additional perspectives regarding the application of DID in transportation:

> While the term "defense-in-depth" is not explicitly used, the current regulatory approach for approving and inspecting radioactive shipping packages follows the risk-informed and performance-based defense-in-depth approach in a general sense. For example, the safety requirements for different types of shipping packages become more stringent with the quantity (radioactivity), or hazard, contained.

NUREG-2150 also made the following finding regarding the risk assessments and DID:

> Finding 2.2: Risk assessments provide valuable and realistic insights into potential exposure scenarios. In combination with other technical analyses, risk assessments can inform decisions about appropriate defense-in-depth measures.

NUREG-1520 generally discusses safety at a fuel cycle facility in terms of items relied on for safety (IROFS) and the integrated safety analysis, whereas the approach for a transportation package of a transportable micro-reactor module uses terms of safety features and components and PRA to convey the same concepts. The NRC staff used NUREG-1520 to assist its evaluation of the DID approach for transporting the transportable micro-reactor module package. DID in the context of accident sequences that could result in undesired consequences at a fuel cycle facility are similar to the undesired consequences in transporting a transportable micro-reactor module package (i.e., material releases or loss of containment and criticality). NUREG-1520 provides the following concepts that are directly relatable to the transportable micro-reactor module risk framework in the risk-informed methodology:

> An integrated safety analysis identifies potential accident sequences in the facility's operations, designates items relied on for safety (IROFS) to either prevent such accidents or mitigate their consequences to an acceptable level, and describes management measures to provide reasonable assurance of the availability and reliability of IROFS.
>
> Defense in depth is the degree to which multiple IROFS or systems of IROFS must fail before the undesired consequences (e.g., criticality, chemical release) can result. IROFS that provide for defense in depth may be either independent or dependent, although IROFS should be independent whenever practical because of the possibility that the reliability of any single IROFS may not be as great as anticipated. This will make the results of the risk evaluation more tolerant of error.

The NRC staff has determined that the DID provided in the risk-informed methodology would provide an acceptable approach for forming the basis of an exemption request for the following reasons:

- The approach describes the multiple layers of defense that are not reliant on a single feature or component.
- The passive safety features and components provide reasonable assurance of availability and reliability.
- Administrative transport controls not credited in the PRA will be applied to limit potential accidents and help to further ensure the reliability of the safety systems.
- Recovery plans will be in place in the event of transportation incidents and accidents as a final safety protection.

Taken together, these protections are consistent with the NRC's statements regarding DID and statements in NUREG-1520 regarding DID with respect to fuel cycle facilities.

Chapter 7: Conclusion

The NRC staff found that the risk-informed methodology proposed in "Development and Demonstration of a Risk Assessment Approach for Approval of a Transportation Package of a Transportable Nuclear Power Plant for Domestic Highway Shipment" outlines an approach that, if followed with appropriate justifications and additional information, as discussed above, could be used as the basis for demonstrating that a future transportation package for a micro-reactor will not endanger life or property nor the common defense and security, such that the NRC could grant an exemption from certain portions of the regulations under 10 CFR 71.12 in a future DoD transportation application package. This conclusion is based on the fact that the risk-informed methodology contains appropriate F-C targets for both members of the public and radiation workers, and the appropriate elements for a transportation package PRA.