Enclosure 1

PNNL Response to
Request for Additional Information Transmitted by NRC April 19, 2023
Docket No. 71-9396
Project Pele

Response to Request for Additional Information

The following is the PNNL response to NRC request for additional information (RAIs) on a draft report regarding demonstration of a risk-informed approach to support transportation of a transportable microreactor with irradiated fuel using Probabilistic Risk Assessment (PRA). The request is available in the NRC Agencywide Documents Access and Management System (ADAMS) as ML23087A109 (the transmittal letter) and ML23087A110 (the questions). The questions along with the lead-in paragraphs provided below are extracted directly from the NRC file (ML23087A110) and are followed by responses by PNNL in a different font as marked. The questions are organized by section number of the report and question within that section. In some cases, the question was broken down into parts as determined by PNNL to provide an interim response to a request or observation. Some of the responses also address relevant observations made in the Observation attachment to the request (ML23087A111) as a way to provide clarification to the report.

<u>Questions by NRC</u>
By request dated February 20, 2023 (Agencywide Documents Access and Management System ML23066A202), on behalf of the Strategic Capabilities Office (SCO) within the Department of Defense, the Pacific Northwest National Laboratory (PNNL) requested U.S. Nuclear Regulatory Commission (NRC) review of PNNLs document titled "Development and Application of Risk Assessment Approach for Transportation Package Approval of a Transportable Nuclear Power Plant [TNPP] for Domestic Highway Shipment."

This request for additional information identifies information needed by the NRC staff in connection with its review of the request for endorsement. Since, the regulations in 10 CFR Part 71 are prescriptive requirements, the NRC staff evaluated the request for endorsement against some regulatory approaches and methods discussed in Regulatory Guide 1.200 "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities" and integrated safety analyses in NUREG-1520, "Standard Review Plan for Fuel Cycle Facilities License Applications," as they are applicable to evaluating risk of transportation accidents. The requested information is listed by chapter number and title in the report. Each question describes information needed and the staff's justification for asking the question.

1.0 INTRODUCTION

1. Clarify whether the risk assessment approach for transport of a TNPP will be used only for accidents or will it be used for normal conditions of transport. If the approach is used for normal conditions of transport, then different dose criteria may be needed than for accidents.

   <u>PNNL Response to Section 1, Question 1, Part 1:</u>
   The proposed risk assessment approach for supporting approval of transport of a TNPP was developed to be used only for evaluating accidents. The assumption was made that the design would meet the deterministic requirements like CFR 71.71 for normal conditions of transport (NCT). If a TNPP package cannot meet the regulatory requirements for NCT, that would suggest a different package robustness than assumed in the demonstration PRA, and the package performance would need to be reconsidered against hypothetical accident conditions (HAC) (i.e., the assumptions made in the consequence analyses supporting the transportation PRA would need to be reconsidered). A PRA approach could be developed to evaluate the risk

associated with normal conditions of transport (NCT) if the requirements for NCT are not met but that condition is not addressed in the current study.

A discussion of NCT and HAC that explains their relevance to the TNPP PRA has been added to new section ("Overview of the Risk Assessment Approach") that was moved to Section 2 (which is now Section 3 in the updated report). Section 3.2 of the updated report refers to the definition of "accident" used in the PRA. Also, a discussion was added to Section 3 (now Section 4 of the updated report) that explains how NCT was considered in developing the proposed risk evaluation guidelines. This discussion is provided in Section 4.2.5.3 of the updated report and described later in response to other NRC questions.

Section 1.3 states that the dose rate regulatory limits will be met during transport, which is in conflict with the statement in section 2.1 which states that "Compliance with all environmental and test conditions in 10 CFR [Title 10 of the *Code of Federal Regulations*] 71.41(a) and all leak rate and shielding requirements in 10 CFR 71.51 ("Additional requirements for Type B packages") or 10 CFR 71.55 ("General requirements for fissile material packages") after hypothetical accident conditions (HAC) will likely prove challenging for TNPP transportation packages." In addition, 10 CFR 71.43 requires no substantial reduction in the effectiveness of the packaging under the conditions specified in normal conditions of transport (10 CFR 71.71); it does not state an acceptable probability.

<u>PNNL Response to Section 1, Question 1, Part 2</u>:
The cited statement from Section 1.3 of the report is meant to refer to regulatory limits for normal conditions of transport. The cited statement from Section 1.3 (now Section 2.1 of the updated report) has been modified to state: "It is assumed that the suite of TNPP containers will meet the NRC and DOT regulatory dose rate limits during shipment for normal conditions of transport (NCT), although the distance at which dose rate limits are met may be part of the exemption request."

However, as indicated, it is assumed that not all 10 CFR 71.73 deterministic tests can be met. In Section 2.1 of the updated report, a sentence was added stating: "However, it is also assumed that the TNPP Package will not meet all environmental and test conditions in 10 CFR 71.41(a) and subsequent leak rate and shielding requirements in 10 CFR 71.51 ("Additional requirements for Type B packages") or 10 CFR 71.55 ("General requirements for fissile material packages") after subjection to the postulated hypothetical accident conditions specified in 10 CFR 71.73 (HAC)." As a follow-on to this statement and to address an observation made in the NRC Observations attachment (ML23087A111), a statement was added to Section 3.2 of the updated report stating "When it is known which 10 CFR 71.73 deterministic tests can be met and which cannot, it is possible that certain accidents could be excluded from consideration in the TNPP transportation PRA. However, in practice, it would likely be difficult to align the crash conditions with conditions created by the 10 CFR 71.73 tests. The fact that the hypothetical accident condition tests are performed sequentially as specified in 10 CFR 71.73 to determine their cumulative effect on the package make this comparison even more difficult."

We agree with the last statement ("it does not state an acceptable probability") and take it to mean that per 10 CFR 71.43 there is no probabilistic element to meeting normal conditions of transport. We note that the current requirements for packaging under 10 CFR Part 71 are deterministic.

The transportation regulations in 10 CFR Part 71, have different dose rate and containment criteria for normal conditions of transport and hypothetical accident conditions in 10 CFR 71.47 and 10 CFR 71.51(a). (The dose rate criteria are located in 10 CFR 71.41 and containment criteria in 10 CFR 71.51(a)(1) for normal conditions of transport and the dose rate and containment criteria is in 10 CFR 71.51(a)(2) for hypothetical accident conditions.) This recognizes the fact that the impact of radiological material to the public should be lower during normal transport conditions than in an accident. In addition, development of the approach recognizes this in section 3.1 where it states, "For routine and chronic exposures, 10 CFR Part 20 ["Standards for Protection Against Radiation"] provides regulatory limits and constraints that must be considered in decisionmaking." However, some of the "accidents" in table 4-5 appear to be similar to the tests and conditions for normal conditions of transport in 10 CFR 71.71, such as items 7e, 8b, 9a, 9c and potentially 11c.

<u>PNNL Response to Section 1, Question 1, Part 3</u>
We acknowledge that the following cited events from Table 4-5 (now Table 5-5 of the updated report) appear to be caused by normal condition of transport:

- Accident 7e (Extreme cold that fails containment),
- 8b (High ambient air temperature and containment failure),
- 9a (Radiolysis and possible hydrogen accumulation),
- 9c (Random vibration or human error),
- 11c (Adverse weather that causes delay).

These events (and others) were identified in the hazard analysis as hazardous conditions that could potentially be considered accidents and are included in Table 4-5 (now Table 5-5 of the updated report) and evaluated for completeness. As stated in Note (d) and (e) at the bottom of new Table 5-5, certain events were evaluated and considered not to be accidents.

Accident 7e and 8b are grouped with other accidents to create bounding representative accidents BRA 7 and BRA 8 as described in Sections 5.3.4.4 and 5.3.4.5 which pertain to breach of the reactor containment boundary (pressurized and unpressured) caused by failure or error and in some cases facilitated by an environmental condition.

However, potential Accidents 9a,9b, 9c and 9d involve only release of contamination from the package but outside the reactor cooling boundary and are therefore considered to be covered by the Radiation Safety program. A note was added to Table 4-5 (now Table 5-5 of the updated report) stating that "Events that cause spread of contamination that affect the worker during transport due to events, environmental conditions, or phenomena that can occur during transport were identified as important hazardous conditions and are shown here for completeness but are not carried forward as accidents that contribute to bounding representative accidents for the reasons explained in Section 5.3.4.6." (Section 5.3.4.6 in the updated report was originally Section 4.4.3.2.6).

Likewise, potential Accidents 11a, 11b, and 11c potentially result in increased unplanned additional exposure to normal radiation from the package and are therefore considered to be covered by the Radiation Safety program. A note was added to Table 4-5 (now Table 5-5) stating that "Events that cause additional radiation exposure to the worker during transport due to delays caused by environmental conditions or technical problems were identified as important hazardous conditions and are shown here for completeness but are not carried forward as accidents to contribute to bounding representative accidents for the reasons explained in Section 5.3.4.7." (Section 5.3.4.7 in the updated report was originally Section 4.4.3.2.7).

2.0 DEFINITION OF REGULATORY APPROACH

No questions

3.0 DEFINITION OF SAFETY GOALS AND RISK EVALUATION GUIDELINES

1. Clarify the discussion on document No. DOE-STD-3009-2014, "Preparation of Nonreactor Nuclear Facility Documented Safety Analysis," relating to dose acceptance criteria for an accident.

Section 3.2.1 discusses that the acceptance criteria in DOE-STD-3009-2014 includes "[a] radiological dose of greater than 25 rem to the public and 100 rem to workers are acceptable, if the likelihood of the accident that produces this consequence is 1E-06 per year or less; and is unacceptable if the likelihood of the accident is more than 1E-06 per year." However, on page 17, it also states: "The standard [DOE-STD-3009-2014] states that if the unmitigated offsite release consequence of an accident exceeds the "Evaluation Guideline (EG)" of 25 rem total effective dose (TED) per year, then controls shall be applied to prevent the accident or mitigate its consequences to below the EG." These appear to be in conflict with one another for a member of the public.

<u>PNNL Response to Section 3, Question 1, Part 1</u>
The cited statement from Section 3.2.1 (which is in a bullet point on page 18) concerns "a hypothetical risk evaluation scheme" postulated based on "investigation of potential risk evaluation concepts" in DOE-STD-3009-2014 as explained in the preceding paragraphs. (The cited statement is now in Section 4.2.1 of the updated report.)

We acknowledge that the purpose of DOE-STD-3009-2014 is not to define acceptable risk but rather to define when controls are needed. Appendix A.10 of the DOE-STD-3009-2014 standard states that the 25 rem TED Evaluation Guideline is not a safety standard because it does not define acceptable or unacceptable dose.

However, DOE-STD-3009-2014 does endorse the concept of risk ranking in the Hazard Evaluation as discussed in Appendix A.4 and elsewhere of the standard. Based on the frequency and consequence categories defined in the standard and other regulatory guidance we hypothesized criteria shown in Table 3-1 of the report (which are like schemes widely used in the hazardous condition evaluations supporting Documented Safety Analyses for DOE nuclear nonreactor facilities). This is now presented in Table 4-1 of the updated report.

Also, on page 18, last paragraph the DOE-STD-3009-2014 states "This analysis shall demonstrate how SC [safety class] mitigative SSCs [structures, systems, and components] and/or SACs [specific administrative controls] reduce consequences below the EG and how SC (if identified) and SS [safety significant] mitigative SSCs and/or SACs reduce co-located worker consequences below 100 rem," which appears to state that dose to co-located worker should be mitigated to less than 100 rem, which is not what the document shows in its tabular form for the DOE Standard in tables 3-1 and 3-2, as both tables show that, for accidents with a frequency less than 10^{-6}, there is no upper limit on the dose.

PNNL Response to Section 3, Question 1, Part 2
Please refer to the discussion above in response to Part 1 of this section. Table of 3-1 (now Table 4-1 of the updated report) presents a "hypothetical risk evaluation scheme" postulated based on "investigation of potential risk evaluation concepts" in DOE-STD-3009-2014 as indicated in the title of the table. In like vein, Table 3-2 (now Table 4-2 of the updated report) is "a hypothetical risk evaluation scheme" based on our examination of guidance in 10 CFR Part 70 and NUREG-1520.

2. Clarify whether the results of an accident that meets the dose rate and containment criteria in 10 CFR 71.51, will also have to meet the quantitative health guidelines (QHGs.)

PNNL Response to Section 3, Question 2, Part 1
We propose that accidents meet only the suggested risk evaluation guidelines like those presented in Table 3-7 (now Table 4-7 of the updated report). However, we show (as discussed below) that if bounding representative accidents meet the proposed surrogate risk evaluation guidelines, then the RIDM acute fatality QHGs would also be met. Concerning the requirements in 10 CFR 71.51, we assume that the criteria for normal condition of transport (NCT) are met but that the criteria for hypothetical accent conditions (HAC) are not met.

Discussion at end of section 3.2, "Development of Risk Evaluation Guidelines Surrogates for Safety Goal QHOs [quantitative health objectives]," notes that some of the lower consequence higher likelihood bins violate the QHGs but seems to argue that the package will be designed to prevent these events anyway. That's more an argument that the package will easily meet these limits rather than an argument that the limits are acceptable.

PNNL Response to Section 3, Question 2, Part 2
The purpose of Table 3-6 at the end of Section 3.2 (now Table 4-6 at the end of Section 4.2 in the updated report) is to present an evaluation of the selected likelihood-dose limits and compare those limits to the RIDM QHGs based on a conversion from dose limits to health effects. Adjustments were made to the proposed risk evaluation guidelines presented in Table 3-7 (now Table 4-7 of the updated report), to ensure the resulting health effects meet the RIDM QHGs. The cited sentence about lower-consequence higher-frequency events being addressed by safety programs was deleted. (Though the statement may be true it caused confusion about the purpose of the table.)

We refined Table 3-6 (Table 4-6 of the updated report) to address Item 2 of Section 3 of the NRC Observations attachment (ML23087A111) that notes that determination of health effects from the proposed dose limits are based only on the lower end of defined frequency intervals for acceptable dose. We adjusted the determination of health effects to be based on the upper end of the defined frequency intervals for an acceptable dose. Accordingly, the conversion to health effects and subsequent comparison to RIDM QHGs is based on using highest allowed frequency of the accident frequency interval and the highest allowed consequence of the consequence interval to calculate health effects. Use of this approach and other conservations described in Section 4.2.5.4 of the updated report results in conservative determination of risk evaluation guidelines limits.

After updating Table 3-6 (now Table 4-6 of the updated report), it was necessary to adjust our proposed risk evaluation guidelines presented in Table 3-7 (now Table 4-7 of the updated report) to ensure the proposed risk evaluation guidelines meet the RIDM QHGs.

3. Clarify whether terminology such as accidents, anticipated occurrences, etc., are defined in a manner that is consistent with NRC regulations or provide a definition of the terms.

PNNL Response to Section 3, Question 3, Part 1
Terms such as "accidents" and "Anticipated Operational Occurrences (AOOs)" were to be used in a manner that is consistent with NRC regulations. However, clarifications were made in the report about the way the term "accident" is used in the study. The term "anticipated occurrences" is not used in the draft report. The term AOO is used in Section 3.2.4 (now Section 4.2.4) of the report along with terms like Beyond Design Basis Events (BDBEs) which are defined in Section 3.2.4 (now Section 4.2.4) of the report based on their use in NEI 18-04. The guidance in NEI 18-04 has been endorsed by NRC in RG 1.233 which noted no clarifications of terminology. The accident frequency category of "Anticipated" used in the Hazardous Condition Evaluation is defined in Section 4.4.2.1 (Section 5.3.2.1).

The last paragraph of Section 4.4.1, "Approach to Development of Accident Scenarios" (now Section 5.3.1 of the updated report) states that the accidents of interest in the TNPP transportation PRA are events that lead to release of radiological material into the environment or direct radiation exposure to workers or the public. Such events were identified using hazard analysis as a comprehensive approach to identifying potential accident events evaluated for applicability. A definition of accidents was added to the report in Section 3.3 (which is now Section 4.3 of the updated report) titled "Proposed Surrogate Risk Evaluation Guidelines Established to Meet the Safety Goal QHOs" as follows:

> "For the risk evaluation guidelines for TNPP transportation accidents to be appropriately applied, the term "accident" must be clearly defined and used in the PRA in a way that is compatible with criteria used in the risk evaluation guidelines. Given that the safety functions that must be preserved by the TNPP Package during transport as discussed in Section **Error! Reference source not found.** are containment, shielding, and prevention of criticality, the accidents of interest addressed in the PRA are (1) a release of radiological material to the environment, (2) direct radiation exposure from unreleased radiological material (e.g., due to degraded internal or external shielding), or (3) a criticality that potentially involves both direct radiation and release of radiological material."

Section 3, page 13, "discusses potential risk evaluation guideline approaches and presents proposed risk evaluation guidelines for TNPP transportation package risk that are consistent with the U.S. NRC's safety goal philosophy, guidance, and historical practice."

Although the classification of "not unlikely" as greater than 10^{-4} per year could imply the scale could go all the way to 1 or more per year, the NRC has typically used other terms such as "anticipated occurrences," or "off-normal conditions" to describe events or occurrences that can be anticipated to occur rather than referring to these conditions/events as “accidents”. In contrast, figures 3-1 and 3-2 in the SCO report provide accident frequency versus consequences based on DOE-STD-3009-2014 that cuts off the curve at a frequency of 0.01 per year, which is consistent with an accident as an event that does not have a frequency of 1.

In summary, discussion regarding the intent of calling items that are expected to occur as accidents versus what NRC might consider normal conditions of transport should be included in order to distinguish them from hypothetical accident conditions, see question 1 above, in section 1, “INTRODUCTION.” The term ‘normal conditions’ is not used in 10 CFR Part 70 to represent such things as expected conditions and events. Importantly the regulatory requirements are different for accidents than for normal conditions (e.g., the occupational dose limit in 10 CFR Part 20.1201 for adults) only apply to normal operating conditions and are the primary guidelines in emergencies – see 56 FR 23365; May 21, 1991; the 10 CFR Part 20 dose limit of 100 mrem for the public is the limit for normal operations. Discussion of the use of the term ‘accidents’ versus normal conditions (e.g., is the intention to apply guidelines/limits for accidents to normal conditions of transport?) may be helpful.

Figure 3-5 (taken from NEI 18-04) uses the term ‘event sequence’. Regardless of the term used discussion regarding the limits/requirements for normal conditions of transport (e.g., events expected to occur) and accidents (e.g., those events with a low likelihood of occurrence) – including how this is represented in figures and tables in the document may be helpful. In discussing TNPP Safety Functions (section 4.3, page 61) there is an identification of ‘normal conditions of transport’ and ‘hypothetical accident conditions,’ but figures just reflect all events as accidents. Additionally, the term ‘anticipated’ is provided on page 73 as a frequency greater than or equal to 0.01 as an accident likelihood category - accidents should not be considered normal operating conditions but this implies they are.

<u>PNNL Response to Section 3, Question 3, Part 2</u>
Discussion of the deterministic requirements in 10 CFR Part 71 concerning normal conditions of transport (NCT) and hypothetical accident conditions (HAC) was added to the Section 2 (now Section 3 titled “The Risk-Informed Regulatory Approach” of the updated report). This discussion provided in Section 3.2 “Overview of the Risk Assessment Approach,” makes reference to the definition of accidents used in the PRA. This discussion points out that for the most part, the consequences of TNPP transportation accidents are less likely and more consequential than conditions that might be assumed to be normally encountered during transport, but this is not always the case. For example, as determined by the PRA, fire-only events during transport of TNPP, which clearly must be considered accidents, lead to no or very minimal radiation dose consequence.

We note that the term “NCT” is deterministic as there is no probabilistic element to the NCT requirements but for releases of radioactive material, a maximum radiological leak rate (10E-06 A_2 per hour) after a set of conditions and test are met is specified. (These include high and low temperature, high and low external pressure, vibration, water spray, a free drop (from about 1 meter depending on the weight), a corner drop, compression, and a penetration test.)

A new section was added to Section 3 (now Section 4) that explains how NCT was considered in developing the proposed risk evaluation guidelines. Section 4.2.5.3 of the updated report cites

IAEA SSG-26, Appendix I.64 through I.70 which explains that the acceptable NCT leak rate of 1E-06 A_2 per hour is derived from limiting the effective dose to 2-rem for a worker spending time in a transport vehicle with the package with a specified air space volume, air exchange rate, and breathing rate for one year. This condition differs from a release caused by an accident which is not expected to be continuous and will be over in far less than a year (e.g., in minutes). None-the-less, limiting the effective radiation dose to 2 rem per year for NCT suggests this dose limit may also be acceptable for a worker for accidents with frequencies from the most likely side of the frequency consequence matrix. Accordingly, given the package design meets the requirements for NCT, development of risk evaluation guidelines was performed in way that avoids defining pairs of likelihood-dose threshold limits as unacceptable when the limit is comparable to the risk to workers from NCT.

4. Revise section 3.2.2, to clarify that not all accidents with a dose less than 5 rem are acceptable.

Section 3.2.2 “NRC Performance Criteria for Integrated Safety Analyses of Nuclear Fuel Cycle Facilities” depicts acceptable and unacceptable accident risk regions for the offsite public (Figure 3.3) and workers (figure 3.4) based on information in 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” and NUREG-1520, “Standard Review Plan for Fuel Cycle Facilities License Applications.” Although NRC understands how these figures were constructed from portions of NRC’s regulations and NUREG-1520, the construct of these figures need clarification to accurately represent key elements of NRC’s regulations and NUREG-1520.

In particular, figure 3-3 depicts all accidents below 5 rem are acceptable for a 10 CFR Part 70 license. Although this is consistent with the development of the Integrated Safety Analysis (ISA) under 10 CFR Part 70, there are other requirements in 10 CFR Part 70 and expressed in NUREG-1520 that would make unacceptable a ‘blanket’ approval of a 5 rem dose to the public (e.g., 10 CFR 70.61(c) requires controls to ensure an event with a dose of 5 rem is unlikely). The regulations provide requirements for unlikely and highly unlikely accidents at 10 CFR 70.61 and NUREG-1520 provides guidance regarding a numerical definition of unlikely and highly unlikely (i.e., unlikely is less than 10^{-4} per event per year and highly unlikely is less than 10^{-5} per event per year; page 3-32) and identify dose limits with respect to high and intermediate consequences (e.g., greater than 25 rem and 5 rem, respectively, for the offsite public; page 3-A-2). As explained in NUREG-1520, this construct was done to “identify accidents for which the consequences and likelihoods yield an unacceptable risk index and to which items relied on for safety must be applied” (page 3-A-3). The report by the SCO appears to interpret this information as a public dose less than 5 rem is acceptable in all situations including accidents with a probability of 1 (there is a later discussion on concern with considering high probability events as accidents). The SCO report ignores the fundamental aspect of the 10 CFR Part 70 and NRC regulations, in general, that the ‘acceptability’ of the ‘not unlikely’ accidents is evaluated under the radiation protection program as described in NUREG-1250, “Report on the Accident at the Chernobyl Nuclear Power Station,” (section 4):

> “[T]he reviewer should be aware that accident sequences considered ‘not unlikely’ in the ISA summary are constricted, under the ALARA requirement in 10 CFR Part 20, to minimize exposure to personnel and the public” (NUREG-1520; page 4-13). The not unlikely category includes those accidents with a probability

greater than 10^{-4} per event per year (NUREG-1520; page 3-A-6). Thus, the dose for the 'not unlikely' accidents are subject to additional constraints that would be expected to reduce the dose especially for those accidents with a high likelihood of occurring (events greater than 10^{-2}). Additionally, 10 CFR 70.62(c)(i-v) requires the licensee/applicant to identify all credible accident sequences including those that are "not unlikely."

Although the identification of the "not unlikely" accident sequences are not required to be submitted to the NRC, the licensee is required to maintain the analysis of these events onsite including the consequences and likelihood. This information is reviewed by the NRC staff during the initial horizontal and vertical slice review and can be reviewed by the NRC inspectors during routine inspection.

<u>PNNL Response to Section 3, Question 4</u>
The investigation of the NRC performance criteria for integrated safety analyses (ISA) of nuclear fuel cycle facilities was performed (along with investigation of other concepts) to arrive at the proposed risk evaluation guidelines presented in Table 3-7 and Figures 3-7 and 3-8 (now Table 47 and Figures 4-7 and 4-8 of the updated report). The hypothetical radiation dose evaluation guidelines postulated in Table 3-2 and Figures 3-3 and 3-4 (now Table 4-2 and Figures 4-3 and 4-4 of the updated report) is a derivation based on reviewing guidance in 10 CFR 70.61 and NUREG-1520.

However, concerning the caveat noted in the NRC observation above, an explanation on events defined to be greater than 1E-04 being subject to additional restraints was added to Section 3.2.2 (Section 4.2.2 in the updated report) for perspective and completeness.

5. Revise section 3.2.3 to clarify the intent of the use of the Q system from the International Atomic Energy Agency's Specific Safety Guide No. SSG-26 (Rev. 1), "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2018 Edition)."

Section 3.2.3 states "The analysis of accidents that could damage a package uses the reference dose of 5 rem to judge when a Type A package is insufficient to limit the transportation risk of the package." The Q system isn't based on analyses of accidents to determine when Type A package is insufficient, as the Q system uses dose to an individual, without regard to evaluation of specific accidents. For special form radioactive material, the Q system uses calculation of a whole-body dose limit of 30 mSv (3 Rem) assuming a distance of 3 m over a period of 3 h. For normal form material, the dose limit for A_2 is set based on a release of $10^{-6}A_2$, which is a "median accident". The median accident is defined as one which leads to complete loss of shielding and to a release of 0.1% of the package contents in such a manner that a bystander subsequently received an intake of 0.1% of this released material, hence the $10^{-6}A_2$ release. Based on this calculation the A_2 value is set to limit the dose to a radiation worker to half the annual limit on intake for each specific radionuclide.

<u>PNNL Response to Section 3, Question 5</u>
We understand the guidance in IAEA SSG-26 Edition 2018 to say that the assumption you refer to "a whole-body dose limit of 30 mSv (3 Rem) assuming a distance of 3 m over a period of 3 h" is an old assumption that has been superseded in the current version of the safety guide. Appendix I.7 discusses this original assumption but goes on to state that the current basis includes consideration of a broader range of exposure pathways than the earlier A_1/A_2 system.

The parameters of those exposure pathways were used or referenced as a starting point as explained in Section 4.6.3 of the draft report (now Section 7.3). SSG-26 states that for external dose due to photons a reference dose of 5 rem (0.05 Sv) is used. For external dose to beta emitters, a reference dose of 50 rem (0.05 Sv) is used. For internal dose due to inhalation a reference dose of 5 rem (0.05 Sv) is used. For submersion dose due to gas, a reference dose of 5 rem (0.05 Sv) and 50 to the skin is used. The consequence analysis performed in support of the TNPP transportation PRA starts from this more current basis.

However, as far as Section 3 (now Section 4) is concerned, it is the reference dose of 5 rem that is the primary contributor to development of the risk evaluation guidelines.

6. Clarify what appear to be errors/typos in the following:

a. Table 3-4 contains many acceptable/unacceptable, more than/less than phrases which appear to be reversed. e.g., "A …dose…is acceptable if the likelihood is more than…"

<u>PNNL Response to Section 3, Question 6, Part a</u>
The "not applicable" blocks are intended to indicate that the dose limit in the first column did not come from the source identified in the column header. Concerning Table 3-4 (now table 4-4 of the updated report), the report has been updated to explain the meaning of the "not applicable" blocks in the body of text where the table is discussed.

We recognize that the use of "more and less than phrases" in the table can be confusing, perhaps because the likelihood intervals have an upper and lower value. Table 3-4 (now table 4-4 of the updated report) has been updated for clarity.

b. In table 3-6:

- Numbers for worker (last row) appear to be incorrect
- Risk columns don't have any unit labels (fatalities/yr)
- QHG for acute fatality would not apply for the lower consequence bins

<u>PNNL Response to Section 3, Question 6, Part b</u>
Concerning the first bullet, we agree that values for the worker in the last row needed to be updated. That said, Table 3-6 (now Table 4-6 in the updated report) has been updated for other reasons described above and correction of these values has been superseded by other changes. (These other changes include those associated with adjusting surrogate risk evaluation guidelines to meet RIDM QHGs as described in the response to Question 2 of Section 3 and addressing how NCT should be considered for the worker as described in the end of the response to Question 3 of Section 3.)

Concerning the second bullet, unit labels were added to the column headers (i.e., "fatalities/year" and "injuries/year") of Table 3-6 (now Table 4-6 of the updated report) also, differentiation of acute from latent and fatalities for 750 rem is explained in Note (a) of updated Table 4-6.

Concerning the third bullet, Table 3-6 (now Table 4-6 in the updated report) has been updated to meet RIDM QHGs as described in the response to Question 2 of Section 3.

c. Section 3.2.4, "NRC Endorsed Risk-Informed Methodology in Support of Licensing Advanced Reactor Design," should the phrase "None-the-less, the guidance document presents the frequency-consequence evaluation plot shown in Figure 3-3 …" really be Figure 3-5, since the caption for Figure 3-5 states "Frequency-Consequence Targets from NEI 18-04, Revision 1"? The caption for Figure 3-3 states "Frequency Consequence Chart for Offsite Public Based on 10 CFR Part 70 and NUREG-1520."

PNNL Response to Section 3, Question 6, Part c
This error had been internally noted. The reference was updated to be Figure 3.5 (now Figure 4.5) of the report (which is Figure 3.1 of the NEI 18-04 report).

7. Revise the statement in the first paragraph in section 3.2.3, "Risk Reference Used in Developing the IAEA Q System," regarding Type B(U) and Type B(M) package testing.

The statement in the first paragraph of section 3.2.3: "The more robust Type B(U) or Type B(M) packages require testing that takes into account a large range of accidents which expose packages to severe dynamic forces" is incorrect. Hypothetical accident conditions were not designed to represent an actual accident the package would experience during transport but, as stated in the proposed rulemaking dated December 21, 1965 (30 FR 15748), was "chosen that satisfactory performance of a package exposed to them may be considered to give reasonable assurance of satisfactory performance in accidents likely to occur in transportation."

PNNL Response to Section 3, Question 7
The cited statement in the first paragraph of Section 3.2.3 (now Section 4.2.3) has been replaced with the statement quoted from the Federal Register as suggested as it provides a more accurate articulation of the purpose of the HAC tests.

8. Clarify what is meant by the "not applicable" blocks in table 3-4.

Table 3-4, "Summary of Relevant Risk Limits from Other Applications," shows a number of dose rate blocks labeled "not applicable." It is not clear what "not applicable" means in this context.

PNNL Response to Section 3, Question 8
Concerning Table 3-4 (now Table 4-4 of the updated report), the report has been updated to explain the meaning of the "not applicable" blocks in the body of text where the table is discussed. The "not applicable" blocks are intended to indicate that the dose limit in the first column did not come from the source identified in the column header.

9. Clarify the following statement in section 3.3, "a TNPP package will be designed to remain intact for most hazards and initiating events that can cause accidents particularly if the event is not highly unlikely."

The term “remain intact” is a vague description. Does this mean, no release of radioactive material from the package? Also, it appears that these events seem to be normal conditions of transport; however, the acceptance criteria in Part 71 for normal conditions of transport (dose rates in 10 CFR 71.41 and containment criteria in 10 CFR 71.51(a()1)) are lower than the acceptance criteria for hypothetical accident conditions (dose rate and containment criteria in 10 CFR 71.51(a)(2). The dose criteria listed in the document appear to be for accidents, not normal conditions of transport. (See question 1, above, in the Introduction.)

PNNL Response to Section 3, Question 9
The term “remain intact” is admittedly a vague description and has been removed given that its use in reference to Table 3-6 (which is now Table 4-6 in the update) is no longer needed.

As described earlier we provide an explicit definition in Section 3.3 (now Section 4.3 of the updated report) for the term “accident” for the purposes of this report as “accidents of interest addressed in the PRA are: (1) a release of radiological material to the environment, (2) direct radiation exposure from unreleased radiological material (e.g., due to degraded shielding), or (3) a criticality that potentially involves both direct radiation and release of radiological material.” Section 4.2.5.3 was added to the updated report to explain how NCT is considered in the development of the risk evaluation guidelines. Additionally, the report has been updated to more clearly explain that certain kinds of events were not carried forward to become part of bounding representative accidents, but rather are seen as operational upsets that are managed using normal programs such a radiation safety programs (see Table 5-5, Note (d) and (e), Section 5.3.4.6 and Section 5.3.4.7).

4.0 TNPP TRANSPORTATION PRA METHODOLOGY, DATA, AND RESULTS

1. Clarify the following provided in the risk assessment approach for a TNPP:

   a. Screening Analysis

      Provide information on when a screening analysis will be performed and what screening criteria is used, including, if applicable, what types of scenarios will be screened out. The basis for screening and how it will be performed and documented as part of the framework should be further described/explained. This should include an initial list of events, the screening process, and a final list of events. Currently, it is not clear which, if any, scenarios have been excluded and is also not clear how several scenarios that have no, or minimal consequences survived the screening process.

      PNNL Response to Section 4, Question 1, Part a, Subpart 1
      Item #14 of Section 4.4.2.2 (now Item 16 of Section 5.3.2.2 of the updated report and second paragraph of Section 4.4.3.1 (now Section 5.3.3.1 of the updated report) explain that accidents with “conditions estimated to be of low risk were screened out because (1) the likelihood was determined to be “Beyond Extremely Unlikely” (which is defined for the hazard analysis in Section 4.4.3.1 (now Section 5.3.3.1 of the updated report) or (2) the consequences were determined not to significantly impact any of the TNPP radiological inventory contributors listed in Section 4.4.2 (now Section 5.3.2 in the updated report). In the hazardous condition evaluation tables presented in Tables 8.2-1 through 8.2-9 (Now Appendix B, Tables B.1 through B.9 of the updated report) the low-

risk hazardous conditions are identified in the 6th column from the left. These conditions were evaluated as low risk during the hazard analysis process based on the criteria above and then screened out during the accident identification process. The remaining hazardous conditions were converted into accident events which are listed in Table 4-5 (now Table 5-5) and described in Sections 4.4.3.1.1 through 4.4.3.1.31 (which is now Section 5.3.3.2 through 5.3.3.33 of the updated report)

The accident scenarios that were qualitatively screened out based on the criteria cited above are not explicitly listed in the draft report other than in the Appendix 8 hazardous condition evaluation tables (now Appendix B of the updated report).

Section 4.4.2.2 (Page 70, Item 14) states: “Hazardous conditions qualitatively evaluated to be low risk were not carried forward for detailed accident analysis. Low risk scenarios were screened out because the likelihood was determined to be “Beyond Extremely Unlikely” or the consequences were determined not to significantly impact any of the TNPP radiological inventory contributors…”; however, table 4-26 presents a risk summary of the bounding representative accidents (BRA) and includes accidents that have no release of radiological material and no loss of shielding (presenting 0 consequences in the table) for BRA 1, BRA 4L; consequences on the order of a microrem for BRA 7; and consequences on the order of a millirem or less for BRA 2 and BRA 8 (5 of the 12 BRAs have very low consequences).

<u>PNNL Response to Section 4, Question 1, Part a, Subpart 2</u>
We acknowledge that Table 4-26 (now Table 7-6 in the updated report) shows bounding representative accidents that were determined after evaluation to result in no or very low dose consequences. Accident scenarios were defined based on identification of hazardous conditions that were qualitatively judged to produce enough damage to the TNPP package to result in non-zero dose consequences as described Section 4.4 (now Section 5.3 of the updated report) on accident identification and development. During, the consequence analysis stage described in Section 4.6 (now Section 7.0 of the updated report), detailed evaluations of the radiological consequences were performed that determined the dose to be very low or zero for certain bounding representative accidents. The bases (i.e., source term development) for these determinations are described in Section 4.6.2 (now Section 7.2 of the updated report). For example, Section 4.6.2.1 (now Section 7.2.1 of the updated report) describes why fires that originate with within the transport module do not produce enough damage to the package to cause a release. These consequence results for these scenarios were, none-the-less, presented with the other results to provide perspective and to show completeness, (though they might have been screened).

b. Clarify the definition of fission products used in the document.

A wide range of radionuclides (e.g., Pu isotopes, which are actinides) are included as a class of fission products, for example, in table 4-1; however, in section 4.2.4.1 (page 56) the text indicates that fission products and actinides are separate groupings. Consider defining what the term “fission products” includes in the document and then use it consistently throughout the document.

PNNL Response to Section 4, Question 1, Part b

The term "radionuclides," as it is used to define Material at Risk and Source Term released, encompasses actinides, fission products and activation products, as appropriate. As such, this is consistent with the way the term "fission products" is used in the first sentence of Section 4.2.4.1 (now Section 5.1.4.1 of the updated report). The report was updated for consistent use of the term "fission products" which involved revising the title of Table 4-1 (now Table 5-1 in the updated report) to be "Radionuclide Classification," and revising the first column of this table to be "Radionuclide Grouping" consistent with the terminology used in NUREG-1465.

c. Clarify the release fractions in section 4.4.3.1.1, "Accident 1(a) – Collision with a Light Vehicle," for collision with a light vehicle.

Section 4.4.3.1.1 (Page 79) states: "The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set lower than values used for collision with a light vehicle." This sentence states that the values should be set lower than the values used for collision with a light vehicle; however, this section is for "collision with a light vehicle".

PNNL Response to Section 4, Question 1, Part c

We acknowledge this misstatement (which we had also internally identified). The cited sentence in Section 4.4.3.1.1 (now Section 5.3.3.2 of the updated report) revised to state: "The damage ratio, airborne release fraction, and respirable fraction used in the consequence analysis for this accident should be set lower than the values used for collision with a heavy vehicle."

2. With respect to section 4.4.2.2, "Hazardous Condition Evaluation Assumptions," provide clarifications for the following:

a. In item number 11 (page 70) it states that the hazards analysis assumed no prohibition of transport in extreme weather, and that "this assumption was reconsidered in the accident analysis." It is not clear what was meant by that statement.

PNNL Response to Section 4, Question 2, Part a

The cited statement in Item 11 of the assumption list in Section 4.4.2.2 (now Item 13 in Section 5.3.2.2 of the updated report) has been updated for clarity. We intended to indicate that an assumption was made (later) that a shipment would not deliberately be made in severe weather conditions. Therefore, we deleted the cited statement (i.e., "this assumption was reconsidered in the accident analysis") and replaced it with an updated version of the second part of the item (i.e., "With the exception that it was assumed a shipment would not deliberately be made in severe weather conditions.").

However, in practice it should be noted that the way extreme or inclement weather events are included in the PRA are as contributors to the crash events counted to derive the very large truck accident frequency as explained in Item #12 of the assumption list in Section 4.4.2.2 (now item 14 of the assumption list in Section 5.3.2.2 of the updated report).

3. With respect to section 4.4.3.1.10, “Tornado or High Wind Event,” provide clarifications for the following:

   a. Additional information is necessary to describe how the likelihood of these events are defined. The section states that “the highest frequency along the route could be used to be conservative.” Considering the potential that the highest frequency event may result in lower consequences due to lower wind speeds vs. a lesser frequency event that could result in higher consequences due to the higher wind speeds, clarify how these differences in events levels and their associated frequencies are being considered when defining likelihood and consequences in the accident progression analysis (e.g., 40-mph wind event could be more likely to occur (i.e., higher frequency) than a 90-mph event; however, one level of high wind event may fall into anticipated event with low consequences vs. the other be an unlikely with higher consequences; or for tornadoes an F4 tornado may just be considered as extremely unlikely, vs. an F1 tornado may just be unlikely).

   PNNL Response to Section 4, Question 3, Part a

   As explained in Item #12 of the assumption list in Section 4.4.2.2 (now item 14 of the assumption list in Section 5.3.2.2), the way extreme or inclement weather events are included in the PRA are as contributors to the crash events counted to derive the very large truck accident frequency. However, there is not enough information in the large truck datasets used to separate large truck accident events by different wind speeds. Furthermore, deriving an event frequency using location dependent wind speed data for a moving vehicle that causes a crash would be overly complex. Therefore, the risk impact of high frequency lower wind speed versus low frequency higher consequence speed events is not explicitly addressed. However, the frequency of high wind correlated to crashes is accounted for in the data used to calculate crash frequencies.

   b. The frequency of a tornado or a high wind event will vary from one type of event vs. another. Because of this, for similar wind intensity or wind loads each event will be associated with a different frequency. Clarify how these differences will be captured and evaluated for each type of event if there are evaluated under the same accident condition.

   PNNL Response to Section 4, Question 3, Part b

   As indicated in the response above extreme weather events that can contribute to the occurrence of highway accidents that damage the TNPP package are included in the large truck data, and therefore, are not treated considered in separate scenarios for wind events. To further clarify, we also updated Item 12 of the assumption list in Section 4.4.2.2 (now Item #14 in assumption list in Section 5.3.2.2 of the updated report) to state: “Moreover, the mechanical impact associated with very large truck crashes was assumed to dominate the accident phenomena, and as a result weather phenomena were not factored into determination of source terms factors (e.g., High wind was not assumed to increase the impact or dilute the concentration of released material.)”

4. Clarify the accident progression described in section 4.4.3.2.8, “Criticality Accidents,” of the report regarding criticality under accident BRA 9.

This section of the report states that the accident consists of "drop into a body of water (e.g., from a bridge) and enough impact to cause a change in core geometry." Clarify whether the change in geometry is a necessary precursor to package criticality, or if flooding with water alone is enough to initiate criticality. This information may affect the frequency determination of accident BRA 9.

<u>PNNL Response to Section 4, Question 4</u>
The wording used to define BRA 9 (now BRA 9A) includes the criterion of "enough impact to cause a change in core geometry." However, the vendor clarified that the "prototype unit will not preclude criticality during a water immersion and inundation event." Therefore, it is assumed in the TNPP transportation PRA that if the TNPP is submerged a criticality will occur. That said, an alteration in core geometry is possible from the drop event and could contribute to criticality. The description of criticality events in Section 4.4.3.2.8 (now Section 5.3.4.8 of the updated report) was updated to state: "For the demonstration TNPP design, the change in core geometry is not required to cause criticality if the core is inundated but an alteration in core geometry is possible as a result of a drop event and could contribute to criticality."

5. Clarify the basis for the frequency of accident BRA 9, "Criticality Event Involving Drop into a Body of Water."

Section 4.7.11 of the report states, regarding the frequency of criticality events involving a drop of the package into a body of water:

> "The actual rate is judged to be between 2.1E-06 per year and 5.1E-09 per year and likely less than 5E-07 per year as presented in Table 4-37."

The basis for the conclusion that the frequency is less than 5E-07 per year is not clear. The 2.1E-06 estimate is well within the frequency range considered in figures 3-1 and 3-2 for the maximally exposed offsite individual and co-located worker, respectively. The applicant does not give a reason for assuming that the frequency is lower than the 2.1E-06 estimate developed in section 4.5.3.1.2, "Frequency of Highway Accidents that Could Result in a Criticality Event," of the report.

<u>PNNL Response to Section 4, Question 5</u>
We acknowledge that Section 4.7.11 (now Section 8.1.11 of the updated report) and Section 4.5.3.1.2 (now Section 6.3.1.2 of the updated report) did not provide the full basis for determining that the frequency of this flooded criticality event is less than 5E-07 per year. The summary in Section 4.5.3.1.2 (now Section 6.3.1.2 of the updated report) of this flood criticality event involving a drop into a body of water has been updated to provide a more complete explanation of the basis for the accident frequency estimate of this accident.

As summarized in Section 4.5.3.1.2 (now Section 6.3.1.2) of the report two approaches were used to estimate the frequency of this accident, one using GIS information and the other using nationwide very large truck data. A detailed description of the GIS approach is provided in Section 4.5.1.4 (now Section 6.1.3 of the updated report). The summary in Section 6.3.1.2 of the updated report, as described below, draws from the detailed description of this GIS approach and from Section 4.5.3.1.2 (now Section 6.3.1.1 in the updated report) on nationwide data for submersion events.

The description in Section 4.5.1.4 (now Section 6.1.3 of the updated report) of the report explains that the estimate for the frequency of a flooded criticality event using geographic information system (GIS) data is based on the following conditions:

- an accident occurs within a 100-foot segment of the route near a sufficiently deep body of water (i.e., five feet or greater for at least part of the year) that is 50 meters or so closer from the highway, and

- where there is sufficient slope from the highway to the body of water (i.e., sufficient slope is assumed to be a 1-to-4 slope), so

- that as a result of any accident, the truck and tailor (with the reactor vessel) will always slide or roll into the body of water from the crash.

The probabilities of these conditions are multiplied by the very large truck accident rate for the five states along the assumed route to produce the 2.1E-06 per year estimate. However, in reality there are other conditions needed that would reduce the estimated frequency of this accident if credited. Though difficult to estimate the probabilities of these other conditions include the fact that: 1) the required conditions do not necessarily exist simultaneously along the 100 foot segment, 2) many accidents would not leave the road enough to be caused to slide or roll down the adjacent slope, 3) the truck, trailer and Reactor Module may come to rest short of the body of water depending on the circumstances of the cash, the ground surface between the roadway and body of water, and the presence of rocks, shrubs and trees that may impeded their slide or roll, and 4) and the water may not be sufficiently deep during the time of year the accident occurs or at the point in the stream, river, or other body of water were the Reactor Module ends up.

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

Using the other approach to estimating this accident frequency, as explained in Section 4.5.3.1.2 (now Section 6.3.1.1 of the updated report) on nationwide data for submersion events (the database refers to these as “immersion events”) are estimated to occur at a frequency of 5.12E-09 per year (as shown in Table 6-13 of the updated report). This estimate is based on immersion events identified in the dataset as the most harmful events (MHE). The 2016-2019 subcategories of first harmful event (FHE) included immersion or partial immersion, motor vehicle in-transport, and collision with a guardrail face. A total of 12 immersion or partial immersion MHEs were reported during this period for large trucks on interstate highways and all resulted in fatalities and there were no injury-only or property-damage-only events that involved immersion events. It is judged that immersion or partial immersion events for large trucks are likely to be clearly identified and reported because of their uniqueness.

Even allowing for possible underreporting using the data approach because there may be uncounted non-MHEs for BRA 9A, it is judged that the frequency of a flooded criticality event is less than 5E-07 per year. Given, this estimate along with the conservatism explained above for estimating the frequency of this accident using the GIS approach, the final estimated is judged to be less than 5E-07 per year. (The updated report now has two flooded criticality events - BRA 9A and BRA 9B.)

6. The probabilistic risk assessment (PRA) should consider a less than completely flooded package or fire scenarios for criticality under accident scenario BRA 9.

The TNPP core is significantly moderated by graphite as designed and built, such that small amounts of water added to the system could significantly increase system k_{eff}, and result in criticality. The reactor pressure vessel may not need to be fully flooded to achieve criticality. Bodies of water with less depth than required to completely submerge the package may still result in criticality. Criticality analyses of the package with varying levels of water moderation will be necessary to determine the depth for bodies of water to be included in the frequency determination for criticality under accident BRA 9.

PNNL Response to Section 4, Question 6, Part 1
The estimation of the frequency for BRA 9 (i.e., flooded criticality) would not change for bodies of water of less depth based on two separate approaches to estimating this frequency as described in response to Item 5 above.

Additionally, fires in or near TNPP packages are likely to be aggressively suppressed to prevent radionuclide release. In the event the containment is failed, due to impact or other event, water or other hydrogenous fire suppression materials may enter the core in sufficient quantities to cause criticality. Criticality analyses of the package with varying amounts of water or other hydrogenous fire suppression materials in the core may be necessary to determine the frequency of criticality under this accident scenario.

This information may affect the frequency determination of accident BRA 9.

PNNL Response to Section 4, Question 6, Part 2
It is acknowledged that if fire suppression water enters the reactor vessel, it could potentially cause criticality. Therefore, a criticality caused by fire suppression water (or other hydrogenous material) was added as a second variation of a flooded criticality and is identified as bounding representative accident BRA 9B. The original variation of this accident involving a drop into a body of water was re-identified as BRA 9A. This second possibility was added as a hazardous condition in the worksheet presented in Table 8.2-7 (now Appendix B, Table B.7 in the updated report) with a note stating the entry was made as the result of review after the hazards analysis sessions were completed. This scenario was also added to Table 4-5 of Section 4.4.3.1 (now Table 5-5 of Section 5.3.3.1 in the updated report) as potential accident and corresponding discussion was added as Section 5.3.3.32 of the updated report. Description of the likelihood development of this bounding representative accident (i.e., BRA 9B) was added to Sections 4.5.3.1.2 (now Section 6.3.1.2 of the updated report) and a detailed explanation of the development of the accident frequency was added to Section 4.5.5.9 (now Section 6.5.10 of the updated report).

The radiation dose consequences of BRA 9B (like BRA 9A) are not developed but an entry was added to the radiation dose consequence summary in Table 4-26 (now Table 7-6 of the updated report). Also, a description of the overall risk results for BRA 9B were added as Section 8.1.12 of the updated report.

The following provides the key elements the detailed description of the development of the accident frequency added to 4.5.5.9 (now Section 6.5.10 of the updated report):

There are five bounding representative accidents that involve fire (i.e., BRA 1, BRA 2, BRA 5-M, BRA 5H, and BRA 6). BRA 1 and BRA 2 are fire-only events that do not involve a crash, and therefore, the containment is intact, and no water intrusion occurs.

BRA 5H (hard impact and fire) and BRA 6 (crash with tanker carrying combustible liquids and fire) have accident frequencies well below the risk evaluation guideline frequency of 5E-07 per year. The risk of accident below 5E-07 per year are acceptable regardless of consequence using the proposed risk evaluation guidelines.

The remaining bounding representative accident - BRA 5M (medium impact and fire) - has an accident frequency of 5.9E-07 per year just over the risk evaluation guideline frequency of 5E-07 per year. However, there are other conditions besides a crash and fire suppression response needed to produce a flooded criticality event that would decrease the estimated frequency of this accident. It is difficult to estimate the probabilities of these other conditions, but they can be characterized in the following way. The crash would need to cause opening in both the Reactor Module and TNPP that would allow water to run into the reactor core. Fire suppression water (or other hydrogenous material) would be directed at the fire which would likely be associated with the engine, wheels or tires, or a fuel spill near under the diesel fuel tanks rather than at the Reactor Module which is carried by the trailer behind the truck. However, a fuel pool could form below the Reactor Module and there could be an opening in the both the Reactor Module and reactor vessel caused by the impact of the crash that allows fire suppression water to enter and inundate the reactor vessel. It is judged that the probabilities of these conditions (though not quantitatively estimated) are enough to reduce the frequency of a flood criticality from fire suppression water due to BRA 5M to below the risk evaluation guideline frequency of 5E-07 per year even without quantitative estimation.

7. Justify using the Q system is appropriate to calculate doses during an accident.

The methodology in SSG-26 was developed to calculate A_1 and A_2 values for individual radionuclides to determine the maximum quantity of material in a package that is not evaluated for hypothetical accident conditions. While the methodology includes external photon dose, external beta dose, inhalation dose, skin, and ingestion dose due to contamination transfer and submersion dose, it does not include neutron sources, except for Cf-252, and does not include interactions that may generate neutrons, such as alpha, neutron (α,n) reactions.

<u>PNNL Response to Section 4, Question 7</u>
The following explanation was inserted into the report after Section 4.6.3.5, “Exclusion of Ingestion and Submersion Dose” (now Section 7.3.6 of the updated report) as Section 7.3.7 “Exposure Pathways Not Addressed by the Q System.”

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

With specific regards to exposure to neutrons, SSG-26 (2018 Edition) states: "In the case of neutron emitters, it was originally suggested under the Q system that there were no known situations with (α,n) or (γ,n) sources or the spontaneous neutron emitter Cf-252 for which neutron dose would contribute significantly to the external or internal radiation pathways considered earlier [I.4]. However, neutron dose cannot be neglected in the case of Cf-252 sources." As noted in the RAI, neutron dose from Cf-252 sources (spontaneous fission) is now specifically considered in the development of the Q_A and A_1 values, but this is done to explicitly address the neutron dose risk associated with Cf-252 sources (special form material); the quantity of Cf-252 in irradiated fuel is insignificant and so is a negligible contributor to dose.

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

Because of the complexity with assessing the neutron emission rate for alpha, neutron (α,n) reactions, PNNL did not perform a separate evaluation of the neutron dose contribution from these reactions. However, significant evidence is provided in the literature that the neutron dose contribution from these reactions is less than 10% of the dose contribution from spontaneous fission neutrons for the time periods of interest (e.g., transportation of the TNPP package within a few years after reactor shutdown). See, for example, https://publications.jrc.ec.europa.eu/repository/bitstream/JRC112361/report_eur_29301en.pdf These results are for irradiated LWR uranium oxide (UO_2) fuel. The UCO TRISO fuel used in Project Pele, in addition to having significant quantities of oxygen, also has significant quantities of carbon. While the presence of carbon is not expected to significantly alter these results for LWR fuel, this difference is a source of uncertainty that may need to be addressed by an applicant utilizing the risk-informed methodology for transporting irradiated UCO TRISO fuel.

8. Clarify section 4.2.2 with regard to 10 CFR 50.71, "Maintenance of records, making of reports", referencing A_2 values.

Section 4.2.2 states: "This approach is consistent with 10 CFR 50.71 ["Maintenance of records, making of reports"] which specifies that an A_2 value from Table A-3 of this regulation may be used if an A_2 value for the radionuclide is not provided in Table A-1 of this regulation." However, 10 CFR 50.71 does not reference 10 CFR Part 71, Appendix A for isotopes that do not have a specified A_2 value. It is unclear what the statement is conveying.

PNNL Response to Section 4, Question 8
The citation in Section 4.2.2 (now Section 5.1.3 in the updated report) was corrected to say:

> 10 CFR 71, "Packaging and Transportation of Radioactive Material", Appendix A "Determination of A_1 and A_2".

9. Clarify whether the following statement in section 4.2.3.1 is discussing in-reactor operations or during transport:

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

PNNL Response to Section 4, Question 9
The cited sentence in Section 4.2.3.1 (now Section 5.1.3.1 of the updated report):

> "However, transportation of an TNPP that has experienced a DBE or BDBE is beyond the scope of this assessment"

was revised to state that

> "However, for the demonstration TNPP transportation PRA, it was assumed that the TNPP being transported has not experienced a DBE and BDBE during operation which would have affected diffusion rates during operation."

This assumption was also added to the end of the list of hazard analysis assumption list in Section 4.2.2 (now Section 5.3.2.2, Item 17 list of assumptions in the updated report)

10. Clarify table 4-5, item 6, and section 4.4.3.1.12, "Accident 6(b) – Diesel Fuel Fire Only Event," to indicate whether any of the fires include any other combustible components of the truck such as tires. If it does not, justify not including combustible portions of the truck.

PNNL Response to Section 4, Question 10
It was assumed that the only external fire of sufficient magnitude to propagate into the Reactor module from the outside is a diesel fuel fire. Other external truck fires such engine fires and wheel or tire fires were assumed not to be of sufficient magnitude to propagate into the TNPP Package. This assumption was also added to the list of hazard analysis assumptions presented in Section 4.2.2 (now Section 5.3.2.2 as Item 11).

11. Clarify the language in section 4.4.3.1.1, “Accident 2(a) – Collision with a Fixed Object.”

In several places there is language like the following “If a worst-case collision with an object is **rare** and the consequences are **high**, then...” Consider reviewing the document for language consistency. Is rare considered unlikely, highly unlikely? Is high considered “High”- Consequence group A or “Very High” – Consequence group B. Several places use the term high consequences that cover both groups.

<u>PNNL Response to Section 4, Question 11</u>
In the cited case, a general likelihood term was used because we were not trying to associate its use with a defined likelihood category like those defined in Section 4.4.3.1 (now Section 5.3.2.1 of the updated report) used for hazardous condition evaluation. Also, concerning the proposed risk evaluation guidelines presented in Table 3-7 (now Table 4-7 of Section 4.3 of the updated report), we purposely did not label the likelihood and consequence intervals to avoid confusion with the likelihood and consequence categories defined in Section 4.4.3.1 (now Section 5.3.3.1 of the updated report). In the updated report use of accident likelihood and consequence terms such as in the one cited was reviewed for consistency and adjustments were made in some cases.

12. Clarify whether any of the accidents in section 4.4.3.1, “Identification and Description of the Full Set of Important Accident Scenarios,” includes drop onto a lower elevation which could be caused by another accident, such as impact with a light or heavy vehicle or fixed object, jackknife, or rollover.

It appears that the discussion of a drop onto a lower elevation result appears to be a single event or due to a fire, however, it appears that a drop onto a lower surface could be caused by another initiating event.

<u>PNNL Response to Section 4, Question 12</u>
Events that involve a drop to a lower elevation are grouped into the bounding representative accidents (see BRA 3, BRA 5H and BRA 9) as indicated in Table 4-6 of Section 4.4.3.2.9 (now Table 5-6 of Section 5.3.4.9 of the updated report). As such, they contribute to “hard impacts” events and are aggregated with other hard impacts into BRA 3 (crash only) and BRA 5H (crash and subsequent fire). These other hard impacts include collision with a heavy vehicle or an unyielding object with have their own event data as shown in Table 4-16 (now Table 6-10 of Section 6.3.1.1 of the updated report) and own frequencies as shown in Table 4-19 (now Table 6-13 of Section 6.3.1.1 of the updated report). BRA 9A, which is a drop into a body of water that results in criticality, can also be considered a drop to a lower elevation. However, estimation of the accident frequency of BRA 9A has already been described earlier in response to Question 5 of Section 4.

In the nationwide datasets, data fields that could indicate a crash involved “a drop to a lower elevation” were not found. Therefore, a GIS approach used was to estimate the frequency contribution of this possibility (i.e., “drops to a lower elevation”) to BRA 3 and BRA 5H as described in detail in Section 4.5.1.5 (now Section 6.1.4 of the updated report) and summarized in Section 4.5.3.1.3 (now Section 6.3.1.3 of the updated report).

As described in Section 4.5.1.5 (now Section 6.1.4 of the updated report), GIS was used to identify topography along the route where there is a drop to a lower surface just off the roadway (e.g., on a bridge or overpass, or near a steep embankment). This was assumed to be a drop-off or a slope of at least 1-to-3 within 25 meters of the edge of the roadway. The assumption was also made that if the truck has an accident at these locations and leaves the road, then significantly more damage could occur to the TNPP package if the vehicle drops to a lower elevation. It is conservatively assumed that every crash leaves the roadway at the defined locations and drops to a lower elevation.

As described in Section 4.5.3.1.3 (now Section 6.3.1.3 of the updated report), the probability that segments of the route meet the criteria discussed above was multiplied by the general crash rate for very large trucks for the five states along the assumed route to yield a frequency of 2.3E-06 per year. This frequency was then added with other contributors to estimate the accident frequency of BRA 3 as described in Section 4.5.5.3 (now Section 6.5.3 of the updated report) and BRA 5H as described in the later part of Section 4.5.5.5 (now Section 6.5.5.1 of the updated report).

13. Justify the statement in section 4.6.3.1, "External Dose Due to Photons," that the distance to the closest member of the public is 25 meters from the accident.

The report assumes that the closest member of the public is 25 meters from the accident based on U.S. Department of Transportation isolation and protective action distance for high level radiological material emergency response. There is no justification for why a member of the public cannot be closer than 25 meters during an accident. While the U.S. Department of Transportation Emergency Response Guide states that a cordon of 25 meters surrounding a spill or leak of radioactive material, this would occur after the accident. The report includes two consequence-probability curves to account for public and worker dose for accidents; however, it is not clear why there needs to be two curves if a member of the public can be located closer than a worker.

PNNL Response to Section 4, Question 13

The uncertainty associated with the cited assumption is recognized (i.e., that the public is assumed to 25 meters from the accident), and it is acknowledged that a member of the public could potentially be closer than 25 meters after an accident until a barrier is established of at least 25 meters surrounding accident site. Therefore, the update of the draft report includes a sensitivity study described in the second half of Section 9.2.2 supported by results presented in Table 9-15 through Table 9-20 that addresses this concern. In the sensitivity study a member of the public assumed to be at the same distance from the accident as a worker (which 10 meters for inhalation dose and 1 meter for other dose pathways based the SSG-26 approach). The results of the sensitivity study show that the overall conclusions about risk from two bounding representative accidents are changed from the baseline case given this change in assumptions. This result is described in Section 9.2.2 of the updated report.

For the baseline case, we judged that the public will be 25 meters or further from point of a release and would be quickly evacuated to a safe location if they are closer than 25 meters. The transport is expected to include an escort vehicle in the front and back of the vehicle carrying the Reactor Module. The escort vehicles create buffer space in front of and behind the truck carrying Reactor Module where other vehicles are expected to be prevented from occupying. The escort vehicles are expected also to provide emergency responders in case of an accident. Also, the assumed route will be entirely on interstate highways which significantly limits crashes involving oncoming traffic.

The question at the end the question concerns why separate risk evaluation risk calculations for the worker and the public given the observation in the question. In defining the two receptors, we are following the lead of the RIDM report (*Risk-Informed Decisionmaking for Nuclear Material and Waste Applications*) and (2) the guidance provided by nuclear non-reactor facility safety approaches such as the approach discussed in DOE-STD-3009. Moreover, by differentiating the risk evaluation guidelines for the worker from the public, we are recognizing that the "worker" has made a choice to be nuclear worker or support nuclear activities and would be expected to be trained in in radiation safety. Also, workers travelling with the transport are at greater risk than a member of public because workers are with the transport for the duration of the transport.

5.0 DEFENSE-IN-DEPTH AND SAFETY MARGIN CONCERNS

1. Revise this section with a focus on what will be developed/presented to describe what is relied on for safety, including the uncertainties with estimating the performance of those items relied on for safety.

   While the overall methodology contains the main topics to be addressed in a PRA approach for estimating risk, it appears that the treatment of some of the topic areas, such as defense-in-depth and uncertainty, may not be to the appropriate level of detail or possibly do not address the primary regulatory aspect of the topic. NRC notes that an applicant for package approval should address both areas in much greater detail in its application.

   PNNL Response to Section 5, Question 1
   It is acknowledged that the defense-in-depth discussion presented in the draft report did not provide the level-of-detail needed for an application and did not sufficiently address many of the unique challenges presented to the demonstration application by applying the defense-in-depth and safety margin philosophies. Therefore, the discussion of the application of each defense-in-depth philosophy in Section 5 (which is now Section 11 in the updated report) has been expanded beyond its original focus.

   The updated primary elements of the defense-in-depth philosophy for this application of the exemption process (10 CFR 71.12 are: (1) the robustness of the TRISO fuel and containment, (2) support of safety functions during transport do not rely on active systems, (3) the TNPP transportation risk is quantified and shown to be low, (4) sensitivity studies show that most sources of uncertainty in PRA modeling assumptions and inputs do not impact the conclusions about risk from accidents, and (5) because compensatory measures will be administered and not credited in the TNPP PRA to reduce risk to the worker and the public and uncertainty about risk through preventive and mitigative actions and features.

   a. Defense-in-Depth

      A number of statements in section 5 appear to imply that defense-in-depth is not really needed due to the low risk. Although a low-risk value may be estimated for a certain activity, NRC's regulatory approach does not dismiss a need for defense-in-depth simply based on risk. NRC considers risk insights gained from conducting a PRA to promote an improved understanding of the system in support of the appropriate level of defense-in-depth:

> "Risk insights can make the elements of defense-in-depth more clear by quantifying them to the extent practicable. Although the uncertainties associated with the importance of some elements of defense may be substantial, the fact that these elements and uncertainties have been quantified can aid in determining how much defense makes regulatory sense. Decisions on the adequacy of or the necessity for elements of defense should reflect risk insights gained through identification of the individual performance of each defense system in relation to overall performance." (NRC White Paper on Risk- Informed and Performance-Based Regulation; March 11, 1999; [ML17348B272]).

> "Defense in depth is invoked primarily as a strategy to ensure public safety given the unquantified uncertainty in risk assessments. The nature and extent of compensatory measures should be related, in part, to the degree of uncertainty." (Letter to Chairman Meserve from B. John Garrick [Chairman Advisory Committee on Nuclear Waste] and Dana A. Powers [Chairman Advisory Committee on Reactor Safeguards]); Use of Defense in Depth in Risk-Informing NMSS [Office of Nuclear Materials Safety and Safeguards] Activities; May 25, 2000; [ML003718610]).

Additionally, in Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement (60 FR 42627; August 16, 1995) the NRC has made clear that a defense-in-depth approach is appropriate to all its programs:

> "Deterministic-based regulations have been successful in protecting the public health and safety and PRA techniques are most valuable when they serve to focus the traditional, deterministic-based, regulations and support the defense-in-depth philosophy."

Below are some of the statements that need further consideration regarding the defense-in-depth approach and how the PRA would support an understanding of the defense-in-depth approach appropriate for the TNPP:

Section 5.1, "Defense in Depth Philosophy"

"The primary element of the defense-in-depth philosophy for this application of the exemption process (10 CFR 71.12, "Specific exemptions") is the fact that the TNPP transportation risk is quantified and shown to be low, but in addition compensatory actions will be administered that reduce the risk to the worker and the public and associated uncertainty through preventative and mitigative features."

<u>Comment</u>: It would appear the defense-in-depth approach for design is based solely on a low-risk estimate rather than an articulation of the design basis for the low risk (e.g., the compensatory measures represent operational constraints and not attributes of the design – such as ship at night to avoid other traffic, escort provided forward and aft, etc.). Defense-in-depth precludes a complete reliance

on one single safety component for safety of the design. The question to be answered is why is the risk low? – what are the safety components that are relied on for safety of the transportation system?

PNNL Response to Section 5, Question 1, Part a, Subpart 1

The cited summary sentence from Section 5.1 (now Section 11.1 of the updated report) was expanded to include other elements of the defense-in-depth philosophy (as primary elements) for this application as shown below:

"The primary elements of the defense-in-depth philosophy for this application of the exemption process (10 CFR 71.12, "Specific exemptions") are: (1) the robustness of the TRISO fuel and containment, (2) support of safety function during transport do not rely on active systems, (3) the TNPP transportation risk is quantified and shown to be low, (4) sensitivity studies show that most uncertainties in modelling assumptions and inputs do not impact the conclusions about risk from accidents, and (5) because compensatory measures will be administered and not credited in the TNPP PRA to reduce risk to the worker and the public and uncertainty about risk through preventive and mitigative actions and features."

Page 192, Item 1, "Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures…"

"Also, the design will be robust and though it may not meet all the requirements in 10 CFR 71.55 ("General requirements for fissile material packages") after HAC, it is expected to meet many or most of the requirements."

Comment: It is unclear what is being conveyed in stating that the Project Pele design "may not meet all" of the safety requirements. The question to be answered is: what are the different design aspects that reduce the risk (such as the tri-structural isotropic (TRISO) fuel, the reactor containment, etc.)? Defense-in-depth is about describing the various 'safety' components and explaining the limits of their functionality with respect to reducing risk.

PNNL Response to Section 5, Question 1, Part a, Subpart 2

The question refers to Item 1 and cites text from the report associated Item 1 but cites defense-in-depth philosophy associated with Item 2 ("Preserve adequate capability…."). Therefore, in the response we describe the enhancement of our assessment of the defense-in-depth philosophy for both Item 1 and Item 2 of the first list from RG 1.200.

Concerning Item 1 (Preserve a reasonable balance among the layers of defense), the discussion was amended to discuss the layers of defense (which includes design features as the robustness of the TRSIO fuel to heat and pressure and the robustness of the containment system against release) and explain that no given layer by itself is relied on primarily for nuclear safety. (In the original version, we intended to convey that even though the TNPP package is not expected to meet all 10 CFR 71.55 (b) tests associated with hypothetical accident conditions, it is expected, none-the-less, to possess a very high level of robustness.)

Concerning Item 2 (Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures), the discussion was amended to state that sensitivity studies show that most sources of uncertainty in PRA modelling assumptions and inputs do not impact the conclusions about risk from transportation accidents.

Page 192, Item 3, "Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences…"

"Redundancy, independence, and diversity are concepts that are more relevant to an operating reactor with redundant active systems."

Comment: While a TNPP is a reactor, it is not an operating reactor. Section 5 does not appear to focus on what makes transporting a microreactor safe (e.g., the TRISO fuel limits release, the reactor core limits release, and the container, express (CONEX) box offering some protection) and how there are protections beyond just the CONEX box.

PNNL Response to Section 5, Question 1, Part a, Subpart 3
The cited text in Item 3 of Section 5.1 (now Section 11.1) has been rewritten to focus on preservation of redundancy, independence, and diversity. The primary safety functions of containment, shielding and maintaining criticality safety are performed by different design features and components. None of the features and components that perform these functions rely on (i.e., are dependent on) active AC power. Shielding and containment are afforded by a combination of the robust design of the TNPP itself independent from the metal CONEX-box-like Reactor Module. Though the Reactor Module does not provide a strong containment function (e.g., it is not leak tight), it would absorb much of the energy of a crash involving impact, and therefore, protects the reactor coolant boundary (which does provide a containment function) from more significant damage. Much of the radiation shielding is bult into and afforded by the reactor vessel itself but is augmented by lead plates in the walls of the Reactor Module. Criticality safety is maintained by a completely different set design features that help prevent reactivity insertion.

Page 193, item 1, "Ensure key safety functions do not depend on a single element of design or operation."

"For TNPP transport, this is a possible weakness of the TNPP design if damage from a severe impact (e.g., collision with a heavy truck) leads to a significant release of radiological material. Another weakness is that the current design of the demonstration unit does not include transportation poison rods as an additional mechanism to prevent a criticality event from a control insertion event as a result of severe impact. However, the PRA shows that the likelihood of TNPP accidents that produce the highest consequences are beyond extremely unlikely."

Comment: The tone of this statement is that there could be reliance on a single component, which is contrary to a defense-in-depth approach. This does not align with other statements in this section that identify such items as the fuel itself and the reactor vessel as significant barriers to release.

PNNL Response to Section 5, Question 1, Part a, Subpart 4

The cited text in Item 1 (of the second list) of Section 5.1 (now Section 11.1 of the updated report) has been rewritten to focus on how there is not a focus of key safety functions on a single element of design or operation. Again, the safety functions that need to be protected are containment, shielding and maintaining criticality safety. There is no single element of the design or operation that is relied on to ensure key safety functions. Shielding and containment are afforded by a combination of the robust design of the TNPP itself independent from the metal CONEX-box-like Reactor Module and complement each other in the way described above. Criticality safety is maintained by the design features that help prevent of reactivity insertion largely independent from features created for shielding and containment (e.g., rod locking mechanisms). The PRA results demonstrate that the risk one of these safety systems failing is low. Additionally, the reliability of these safety functions will be supported by administrative transportation controls (e.g., is expected that there will an escort vehicle in the front and behind the truck carrying the Reactor Module) and other compensatory measures that are not credited in the TNPP PRA.

Page 192, item 5, “provide time for recovery operations,” includes a statement regarding the fuel itself and the reactor vessel as key safety barriers; however, the safety significance of these barriers is completely undermined by items 3 and 4 and emphasis on the lack of redundancy and potential common-cause failures continues on the page 193 with a new set of points (items 1 and 2).

PNNL Response to Section 5, Question 1, Part a, Subpart 5

The question refers to Item 5 from the first list based on RG 1.200 but cites defense-in-depth philosophy associated with Item 5 from the second list from the NRC RIDM report (“Provide time for recovery actions”). Therefore, in the response we describe the update of our assessment of the defense-in-depth philosophy for both Item 5 of the first list from RG 1.200 and Item 5 of the second list from NRC RIDM.

Concerning assessment of Item 5 from the first list from RG 1.200 (Maintain multiple fission product barriers), the question asserts that assessment of safety barriers is “undermined” by the original assessment of Item 3 on redundancy, independence, and diversity and the original assessment of Item 4 on CCF. The assessment of the defense-in-depth philosophies addressed in Item 3 and Item 4 have been significantly updated as described above which now act to support the assessment in Item 5 (which has not been changed.)

Concerning Item 5 of the second list from the NRC RIDM report, the assessment of the philosophy of “Provide time for recovery” has been updated. It is expected that TNPP transportation will include a recovery plan for possible transportation accidents and the transportation workers and personnel should be trained on the transportation plan. Quick recovery actions that minimize the risk of release to the public should be included in the transportation plan (e.g., setup of a safety perimeter to keep the public away from the point of release). It is expected that that the TNPP would be transported using and escorts in the front and back of the truck carrying the TNPP and that personnel in these vehicles would be trained in the emergency response procedures. Results of the TNPP PRA and associated sensitivities studies can be used to enhance recovery response. For example, sensitivity studies were explicitly performed that address assumptions made about the distance of the worker and public from the location of the accident and the duration that they were exposed to the release (or direct radiation from unreleased material).

The end of section 5 summarizes by stating defense-in-depth was applied consistent with NRC guidance and available information; however, the summary does not appear consistent with a number of the points made in section 5 that seem to state precisely the opposite. The application should demonstrate that the principle of defense-in-depth is satisfied.

If the package itself is insufficient for this, then the application should identify other attributes of the design that are relied upon or provide a compelling basis for reliance on administrative measures, such as those identified, and compensatory measures in the document that are not explicitly credited in the PRA.

PNNL Response to Section 5, Question 1, Part a, Subpart 6
We acknowledge that the discussion does not address the points made by NRC in this section and that it cannot be concluded that defense-in-depth principles are met based on the original discussion. Accordingly, the section has been updated as described above.

2. Provide a description of a more robust treatment of uncertainty, which could be based on extensive sensitivity analyses.

The proposed framework appears to lack a formal treatment of uncertainties with the exception of proposing sensitivity studies. This is a reasonable approach to characterize the uncertainty and should focus on key parameters that could significantly increase or decrease the estimated risk. If sensitivity studies are the primary method of characterizing the uncertainty, one would expect the number and level of detail of the sensitivities to be robust.

PNNL Response to Section 5, Question 2
An assessment of uncertainty has been performed and is presented in the updated report in Section 10 which includes identification and evaluation of key sources of uncertainty and assessment of parametric uncertainty (to the extent it can be addressed). The examination PRA modelling assumptions to identify and disposition sources of uncertainty that could impact the conclusions about risk is extensive (see Section 10.2.1 of the updated report). It is based on assessment performed for the sensitivity analyses described in Section 9.1.1 in the new report) which informs the sensitivity analyses as well as the uncertainty analysis.

6.0 TECHNICAL ADEQUACY OF TRANSPORTATION RISK ASSESSMENT

No Questions

7.0 CONCLUSIONS

No Questions