Industry Responses to NRC Questions on NEI's White Paper "Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone"

1. On page i of the white paper, it is stated that the approach is based on some key assumptions, including "the expectation of enhanced safety inherent in the design of Small Modular Reactors (SMRs) (e.g., increased safety margin, reduced risk, smaller and slower fission product accident release, and reduced potential for dose consequence to population in the vicinity of the plant)." What are the key design features and operational programs relied upon for this to be a good assumption, particularly the slower fission product accident release and reduced potential for dose consequence?

Response

As stated on page 5 of the white paper, "[T]he SMR designs are different from traditional, large [light water reactor] LWR plants in ways which significantly reduce the potential for offsite fission product release and dose consequences (e.g., smaller core fission product inventories, improved design features, slower accident sequence evolution)." The following provides additional details about these enhanced design features:

Smaller Core Fission Product Inventories – *SMRs are by design physically smaller than the nuclear reactors currently operating and under construction. Therefore, the SMR cores are physically smaller in both the radial and vertical directions and contain fewer fuel assemblies as compared to traditional reactors. This smaller size results in a smaller core thermal power. Accordingly, the amount of radiological material available for potential release is greatly reduced, resulting in smaller source terms and lower decay heat levels following a reactor trip, and leads to a corresponding reduction in assumptions associated with accident initiation and progression.*

Improved Design Features – *The white paper is focused on integral pressurized water reactor (iPWR) SMRs. The following are some of the more common safety enhancements incorporated into the iPWR designs – note not all of these features are applicable to all designs:*

- *An integral reactor vessel design eliminates large-break Loss-of-Coolant Accidents (LOCAs) by eliminating large-bore piping in the design and no reactor vessel penetrations below the top of the core.*
- *Fuel and reactor components are largely or completely located below ground level, resulting in a reduction in postulated release paths.*
- *A large water volume relative to the thermal power is available for cooling and shielding. The larger volume-to-break-size ratios also result in slower accident progression assumptions. The volume of water is sufficient that fuel failure is not postulated to occur for many hours or days following the onset of an accident condition.*
- *The postulated short term (accident) release path goes through pools surrounding the containment.*
- *A greater number of reactor coolant pumps result in a smaller impact from loss of a single pump, minimizing or precluding of historically postulated accidents such as locked rotor.*
- *Use of natural circulation for the primary loop during both normal operations and post-accident conditions results in lower core flow rates, lower core heat flux, larger margins to departure from nucleate boiling (DNB), lower peak cladding temperatures, and reductions in stored energy and fuel centerline temperatures.*
- *Internally mounted control rod drive mechanisms, obviating historically postulated control rod ejection accidents.*

Slower Accident Progression – *iPWR designs, such as those described above, result in a lower likelihood of fuel damage, a longer time to any fuel failure, and significant reduction in accident consequences. In addition, the ability to mitigate or accommodate long term consequences and beyond design basis events is enhanced – power and water demands to maintain the plant in a safe condition are greatly reduced or eliminated and easily accommodated.*

These fundamental design advantages are part of the iPWR designs under development and their effectiveness in the form of enhanced safety will be confirmed as part of design certification (DC) reviews.

The methodology in the white paper is technology-neutral. A specific SMR design may incorporate some, or many, of the enhanced design features. However, use of all of the design features, or any specific feature, is not a prerequisite condition to use the methodology in the NEI white paper. When utilizing the generic methodology, the actual enhanced features incorporated into a specific design will be considered when determining an appropriately sized emergency planning zone (EPZ) for that design.

2. On page iv, the white paper states that "…establishing an acceptable methodology and criteria early via this white paper is essential to support SMR design certification applications expected to be submitted beginning in 2014."
   a. Will SMR combined license (COL) applicants use this white paper to arrive at conclusions which are sufficient to propose a plume exposure emergency planning zone (EPZ) for their specific sites? If so, what guidance will be provided to address the impact of different structural and reactor core designs on the source terms?

Response
Industry's objective is to develop a technology-neutral, dose-based, consequence-oriented emergency preparedness framework for light water SMRs. The primary purpose of the white paper is to provide DC and combined license (COL) applicants with a generic methodology and criteria for plume exposure EPZ that can be adopted and used to develop a design-specific and site-specific technical basis for an appropriately sized EPZ.

It is expected that each DC applicant will submit one or more topical or technical reports describing the methodology for calculating accident source terms for their design. Proposed system and reactor core designs would be addressed in these design-specific submittals.

COL applicants would use the white paper methodology, supplemented by design-specific or site-specific methodologies as necessary, to develop the technical basis to define a plume exposure EPZ for the site. This may include design specific source term calculations or other supporting technical information to provide a technical basis for an appropriately sized EPZ for their site.

   b. On page i, the white paper has a disclaimer that it "is limited to the consideration of plume exposure EPZ." However, the paper states on page 10 that "the EPZ size decision should be made in context with decisions on the SMR planning standards and confirmation of a substantial base for expansion of response." On page 25, Section 4.4, the paper reiterates that "these planning elements should be integrated with the decision on the EPZ." Please explain how this integration will inform the plume exposure EPZ size, since the paper states on page 2 that planning standards "are not addressed in the paper."

<u>*Response*</u>
Industry's objective is to develop a technology-neutral, dose-based, consequence-oriented emergency preparedness (EP) framework for light water reactor SMRs that provides "reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency."[1] *The plume EPZ is only one of several EP elements that constitute "complete and integrated emergency plans."*[1] *While the white paper is limited to establishing the technical basis for the plume exposure EPZ, the paper notes on pages 10 and 25, and in other places, the need to integrate the EPZ sizing methodology with the other EP planning standards and requirements appropriate for SMRs. Industry plans to evaluate the additional elements of emergency preparedness in order to determine whether and how they should be adapted for SMRs to develop a proposed generic emergency preparedness framework for SMRs. SMR applicants may also evaluate these elements for their designs.*

It is not expected that other EP elements will inform the determination of the plume EPZ, but it is possible that the plume EPZ may inform other EP elements. The intent of the white paper's statements on need for considering other EP elements in an integrated emergency plan is as follows:

- *Acknowledge that COL applicants also need to address the 16 EP planning standards, and the related requirements in 10 CFR 50, Appendix E.*
- *Recognize that implementation of EP planning standards and requirements cannot be performed independently from EPZ size and the technical basis for this size, including fission product release magnitude and the "time between the onset of accident conditions and the start of a major release."*[2] *That is, there will necessarily be a relationship between the EPZ size and the technical basis for the size, and the manner in which EP planning standards and requirements are implemented for a given site.*
- *Acknowledge the need for SMR EP to provide capabilities for expansion of response efforts should events warrant such an expansion, as is the case for current operating nuclear power plants.*

As noted above, industry will develop a proposed generic EP framework for SMRs, which would include a review and, where necessary, adaptation of the 16 EP planning standards listed in 10 CFR 50.47(b) and the associated requirements in 10 CFR 50, Appendix E. We will also determine if development of an SMR appropriate EP framework should be facilitated by changes to, or deviations from, guidance documents such as NUREG-0654, the Federal Emergency Management Agency (FEMA) Radiological Emergency Preparedness (REP) Program Manual and NEI 99-01 (on EALs), or through the creation of new guidance. Industry plans to communicate additional details on these activities in the near future for discussion with the staff. In addition, individual design certification and/or COL applicants may recommend exemptions from current requirements and guidance where such changes are dependent upon the specific design and technical basis for the EPZ.

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[1] 10 CFR 50.47(a)
[2] NUREG-0654, page 13

c. If the Nuclear Energy Institute (NEI) intends for planning standards to be generic for SMRs as stated on page iv of the white paper, how will the 16 planning standards currently found in 10 CFR 50.47(b) inform all plume exposure SMR EPZs?

Response
The paper did not intend that the 16 planning standards currently found in 10 CFR 50.47(b) will inform all plume exposure SMR EPZs. As noted in the response to 2.b, above, NEI and industry intend to engage the staff on the development of a generic EP framework for SMRs. This would involve a review of the current 16 EP planning standards in 10 CFR 50.47(b), related requirements 10 CFR 50, Appendix E, and implementation guidance (e.g., NUREG-0654) to identify those elements that may require changes to better support application to SMRs. It is expected that the EP planning standards and requirements applicable to SMRs would recognize and accommodate the potential for differences in EPZ size that could result from design-specific application of the methodology described in the EPZ white paper.

3. On page 2, the white paper states that it is "a first step to reflect" on lessons learned from the Fukushima Dai-ichi accident. On page 6, it mentions NRC's Near Term Task Force (NTTF) review of insights from the Fukushima Dai-ichi accident.
   a. Provide information on how the lessons learned from the Fukushima Dai-ichi accident and NRC's NTTF review may inform the size of plume exposure EPZs for SMRs.

Response
The following lessons learned from the Fukushima Dai-ichi accident are reflected in the white paper methodology:

- *The need to consider risk from less probable external events and multi-module events is addressed in the Section 3.5 methodology for implementing Criterion C.*
- *Provision of an operationally-focused mitigation capability is addressed in Section 4.1 to address PRA completeness uncertainty and the need to maintain basic safety functions in the face of extreme events.*
- *SFP accident risks are addressed in Section 3 as part of the PRA-based evaluation.*
- *In addition, as part of development of EP planning standards, the white paper identifies the need to confirm the capability for expansion of response exists if necessary.*

b. In light of the Fukushima Dai-ichi multi-unit accident, how does SMR modularity impact the described methodology and criteria for assessing the size of plume exposure EPZs for SMRs?

Response
The 2011 accident at Fukushima Dai-Ichi did not involve modular nuclear power plants, but it did involve common cause failure of emergency ac and dc power from the emergency diesel generators and battery system for each of the affected reactors due to the beyond design basis tsunami. SMR modularity will be taken into consideration in analyzing extreme external hazards and the potential impact on reactor modules that have common or shared systems. Determination of an appropriate plume EPZ will

explicitly account for SMR CCF, interdependence between modules, and effects on the source term. In the white paper methodology, multi-module considerations are addressed in Section 3.5 for the PRA-based evaluation. It is noted that the NRC is planning to issue review criteria on multi-module considerations. The response to Question 6 provides more information on how DC applicants intend to address multi-module risk.

4. On page 7 of the white paper, Methods for Estimation of Leakages and Consequences of Releases (MELCOR), Modular Accident Analysis Program Version 5 (MAAP5), and the State-of-the-Art Reactor Consequence Analysis (SOARCA) are examples given of advanced tools and models. These tools and models are designed for large light water reactors (LWRs). Although SMRs are conceptually similar to LWRs, there is little operational or experimental data. Please provide an explanation of how these tools and models can be extrapolated for SMR analysis to inform plume exposure EPZ size.

Response
The white paper methodology is based upon the premise that extrapolation of MELCOR, MAAP5 and the SOARCA tools and models used for large LWRs for SMR analysis is acceptable. It is recognized that additional discussion with the NRC, and justification for this premise may be necessary; however, this work is outside the intended scope of the white paper. It is anticipated that if additional justification is necessary, it will be provided through design-specific pre-application meetings and submittals to the NRC.

MELCOR and MAAP have been developed to provide integrated analyses of severe accident progression including

- *transient thermal-hydraulic response of the compartments within structures affected by flow from failed fluid systems,*
- *the prediction of challenges to containment integrity and the resulting flow rates out of the containment and into and out of other structures or systems that may be present in release paths to the environment,*
- *core damage progression and the impact of core debris or hot gases from the core on reactor components and containment structures,*
- *radioactive releases from the damaged reactor core,*
- *deposition and transport of those radioactive releases within coolant systems, containments, and other structures, and*
- *the release of radioactivity to the environment including the temperature and composition of those releases.*

The phenomenological models existing in the large LWR versions of these integrated codes are generally applicable to SMRs, as well. However, the models of both MELCOR and MAAP, as applicable, will be examined by the SMR developer to ensure that new or significantly changed design features (SMR vs. currently operating large LWRs) are reflected in the modeling. In fact, specialized versions of MELCOR and MAAP that have such modifications fully incorporated are already in use by SMR developers.

The role of the SOARCA project in helping to inform the selection of a SMR plume-exposure EPZ size is concentrated in two areas:

- *an approach for identifying specific, risk-significant accident sequences for detailed analysis that, in aggregate, are able to adequately characterize the potential risk posed by the facility to the public, and*
- *a basis for selecting* MELCOR Accident Consequence Code System *(MACCS) input parameters for dose consequence analysis that are neither design/sequence-specific nor site-specific*

More information is contained in the responses to RAIs 7a through 7c with respect to SOARCA influence on the application of MACCS.

With respect to sequence selection, the SOARCA effort concentrated on the potentially high-consequence, risk-significant accident sequences for the two plants studied. Therefore, it has no bearing on meeting EPZ size selection Criterion a in the NEI white paper. However, as discussed in Sections 3.4 and 3.5 of the white paper, SOARCA processes do have a role in meeting Criteria b and c.

The role of SOARCA is specifically acknowledged in Sections 3.4 and 3.5 of the white paper with respect to Criteria b and c, but the white paper also calls for an SMR-specific adaptation and expansion of the SOARCA approach in meeting those criteria. The adaptation/expansion includes for Criterion b addressing frequency per plant year instead of per reactor year and addressing potential basemat melt-through accidents, and for Criterion c the potential impact on reactor modules that have some common or shared systems, particular attention to accident scenarios involving extreme seismic and other external hazards, and consideration of fuel handling and spent fuel pool accidents.

5. On page 10, it is stated that the intent of the methodology is to be "part of an integrated, decision-making process for SMR EPZ sizing which uses risk-informed judgment...such that the technical basis for EPZ size is insights, not just numbers or criteria." Please clarify what is meant by "risk-informed judgment" and how the insights will be used.

<u>*Response*</u>
Risk-informed judgment is a process that utilizes both risk insights and defense-in-depth engineering insights as inputs to a deliberative decision-making process. Figure 2 in NRC Regulatory Guide (RG) 1.174 is a good example of such a process as applied to licensing basis changes for operating plants. The specifics of the deliberative, decision-making process for SMR EPZ will be developed and refined as part of actual design-specific applications of the methodology in Section 3 of the NEI white paper. Risk insights on the determination of appropriate SMR EPZ size would be based in part on PRA-developed accident sequences, associated source terms, and concomitant dose-versus-distance calculations in accordance with the stated criteria.

Defense-in-depth engineering insights would be based on the enhanced plant capabilities described in Section 4 of the white paper. The enhanced plant capabilities are a complement to PRA-based effort, are primarily deterministic, and are less numerically-driven and more qualitative. There are four areas to be addressed by applicants on enhanced plant capabilities:

- *Completeness uncertainty including developing a diverse, operationally-focused accident mitigation strategy, not based on probabilities, but addressing risks from the Section 3 accident sequences as well as being flexible and adaptable so as to maintain basic safety functions in the face of extreme, site-wide situations*
- *Potential risks that are not fully addressed in the PRA such as security events*
- *Risk impact of lower frequency accidents (cliff edge effects) which are judged to be credible (physically plausible)*
- *Balancing accident prevention, accident mitigation, and protective actions, including specifying an emergency plan consistent with providing a base for expansion of response if necessary*

6. The white paper indicates that a number of technical issues with establishing the proposed generic methodology and criteria have not been resolved. For example on page 15, "one acceptable way" of using the probabilistic risk assessment (PRA) for accident scenario selection is discussed. In the footnote on that page, it states that "other approaches" to accident sequence selection and grouping could be used. The footnote on page 17 provides another example. The discussion of accident source term evaluation on page 18 suggests that a method for determining source terms for multi-module core damage events needs to be developed. When and how will these and perhaps other parts of the methodology be determined? What process is envisioned for completing development of the methodology?

Response
The intent of the white paper is to provide one acceptable way to determine an appropriately sized EPZ for SMRs. It was not intended for the white paper to preclude other acceptable approaches, nor to identify all approaches that are acceptable. The footnotes on page 15 and page 17 identify two areas where alternative approaches could also be considered.

With regard to accident selection, the SOARCA-like process, adapted for SMRs as discussed on page 15 of the white paper, is to be used for informing sequence selection. The "one acceptable way" phrase and the "other approaches" footnote convey that an EPZ applicant would have the option to use a different method for informing sequence selection, similar to the option that applicants in general have to take exception to NRC regulatory guidance. It is recognized that an applicant may need to provide additional justification in order to use alternative approaches.

Regarding multi-module events, there are no plans to expand the discussion of multi-module considerations in white paper methodology. The methodology for determining source terms for such events will be specified in design-specific submittals (e.g., pre-application topical or technical report(s), DC application, or COL application), and the schedule for these submittals will be determined by each applicant. In a public meeting on June 26, 2014, the NRC proposed criteria for consideration of multi-module risk for SRP 19.0, which the NRC plans to issue in interim staff guidance (ISG). Multi-module risk consideration will include: a systematic approach to identify core damage or large release accident sequences including human error; selected alternative features, operational strategies and/or design options to prevent these sequences from occurring; demonstration that these multi-module sequences are insignificant to contributors to risk; and operational strategies to provide reasonable assurance that *there is sufficient ability to mitigate multiple core damage accidents. The process for SMR applicants to consider multi-module risk is expected to be in conformance with this NRC guidance.*

7. The paper states that the MELCOR Accident Consequence Code System (MACCS) code is an appropriate tool to calculate consequences for the analyses. The paper also states that insights from the NRC's use of MACCS for the SOARCA will be used to inform use of the code for this purpose.
   a. Considering that MACCS has not previously been used for SMR analyses to support EPZ sizing, will NEI provide more information or guidance on the use of the MACCS code for this purpose? Topics that could be different for this proposed use of MACCS are the determination of the basis for code input and model assumptions (including appropriate nodalization of the near-field), sources of information for input, use of conservatism, addressing uncertainty, and input and assumptions for the local area and population.

Response
Additional information on the use of the MACCS code is outside the intended scope of the NEI White Paper. NEI also does not have plans to develop additional information and guidance on use of the MACCS code for SMRs. It may be possible to establish generic guidance for those MACCS inputs that are neither design/sequence-specific nor site-specific (e.g., near-field nodalization schemes), and for the proper translation of MELCOR or MAAP-generated source terms to the source term specification form used in MACCS. However, additional guidance would not be necessary in cases where existing NRC guidance will suffice. For example, site-specific inputs for dose consequence analysis such as the compilation of meteorological data can be developed in accordance with existing NRC guidance. If generic guidance is unavailable, then justification for using these codes for SMRs will be performed by the individual applicants and provided in design-specific submittals.

Some considerations for the use of MACCS for SMRs include:

- *Regarding use of conservatism, simple conservatism (such as that as employed in design basis accident analysis) is generally not appropriate for developing a plan for emergency response where an excessive or premature response can have serious negative consequences. A better approach is to use realistic analyses and to identify uncertainties, and either compensate for those uncertainties in a qualitative way (through, for example, enhanced plant capabilities) or quantitatively assess those uncertainties and make associated judgments on how much the quantified uncertainty affects some given aspect of the emergency plan; e.g., the decision regarding the necessary plume-exposure EPZ size.*

- *Regarding addressing uncertainty, the major sources of dose consequence uncertainty in MACCS (for a given accident sequence radioactive release) are atmospheric dispersion and related deposition from the plume (i.e., wind direction, wind speed, atmospheric stability, and precipitation). These effects are explicitly accounted for in the MACCS calculations. MACCS also addresses the probability distribution of doses to be used for comparison with the white paper dose criteria. The meteorological sampling scheme to be used when running MACCS (METCOD = 2, same as that used for SOARCA) is identified in the response to RAI 7b below.*

Other sources of uncertainty are considered secondary and are not included in the dose assessments. The selection of MAACS inputs addressing other uncertain parameters (e.g., protection and shielding factors) is discussed further in the response to RAI 7b.

- *Regarding inputs and assumptions for the local area and population, uncertainties are not considered important since only individual doses (assuming a population uniformly distributed in the vicinity of the facility) are calculated. That is, the actual population distribution of the site in question has no impact on the individual doses being calculated for comparison to white paper Criteria a, b, and c. No evacuation or sheltering will be assumed for the purpose of these MACCS2 dose calculations to an individual.*

b. Which information from SOARCA is proposed to be used, and how will it be used?

Response
The specific MELCOR modeling schemes found in the SOARCA study for the two, large, current-generation LWRs, Surry and Peach Bottom, have little direct applicability to the accident progression analyses that will support the determination of the necessary sizes for SMR EPZs. However, as broadly described in the white paper, a modified version of the approach used to define the risk-dominant accident sequences in the SOARCA study will be used in meeting Criteria b and c for plume-exposure EPZ size selection. The white paper also refers to a process for evaluating operator mitigation strategies for more severe, less probable accident scenarios which is based on similar work performed in the SOARCA project.

With respect to the dose consequence analysis, the application of MACCS in the SOARCA study was directed towards calculating the plume exposure early fatality and latent cancer risk posed by the two plants studied as a means of updating earlier risk studies. As such, emergency response was modeled, and the health effects models of EARLY (the EARLY module of MACCS2 models Emergency Phase Calculations) were employed. For purposes of evaluating White paper Criteria a, b, and c, on the other hand, ATMOS (the ATMOS module of MACCS2 models Atmospheric Transportation and Deposition) and EARLY modules will be modeled, but the EARLY input is greatly simplified compared to that of SOARCA because no health effects are to be calculated.

The following MACCS inputs for the ATMOS and EARLY modules can be taken from SOARCA and can be used generically for the SMR EPZ size assessment. Individual applicants may choose design-specific values for these inputs. Justification for the input values should be provided.

ATMOS

- *Radionuclide Data (IS)*
- *Wet Deposition Data (WD)*
- *Dry Deposition Data (DD)*
- *Dispersion Parameter Data (DP)*
- *Plume Meander Data (PM)*
- *Plume Rise Data (PR)*
- *Release Description Data (RD)*
- *Output Control Data (OC)*

- *Meteorological Sampling Specification (M1)*
- *Boundary Weather Data (M2) (except LIMSPA)*
- *Meteorological Bin Sampling Data (M4) (except IRSEED)*

EARLY

- *Miscellaneous Data (MI)*
- *Shielding and Exposure Data (SE)*

c. Will a test case or pilot case be provided for demonstrating how MACCS and SOARCA information would be applied in a realistic situation?

<u>Response</u>
It is anticipated that the demonstration of how MACCS and SOARCA information are applied in a realistic situation will be through a pilot application for a SMR design certification applicant using site parameters assumed by that applicant. The pilot application is expected to illustrate, clarify as needed, and gain NRC acceptance of the generic methodology as applied to a particular SMR design.

8. Are emergency response actions, such as evacuation modeled for all three criteria, using MACCS? If evacuation is modeled in the analyses, provide a discussion on why this is appropriate.

<u>Response</u>
Relocation on an ad-hoc basis may be considered, but evacuation is not modeled. For the comparison against the PAGs, a 4-day in-place exposure time is used.

9. In the evaluation of accident consequences against Criteria b and c, will each selected accident scenario have a separate consequence analysis or will the scenarios be grouped? If scenarios are grouped, what is the basis for the grouping (e.g., core damage frequency, release frequency, release characteristics)?

<u>Response</u>
Page 15 of the white paper discusses that accident sequences will be grouped into accident scenarios, and the basis for this grouping of sequences into scenarios is similar timing to core damage and similar equipment availabilities. The scenarios would not be grouped. The intent is that each scenario will have a separate consequence analysis.

10. The discussion of the scenario selection process for Criterion b is not clear in some areas. For example, is the white paper stating that using the SOARCA process as adapted to SMRs results in a cut-off frequency of 1E-8/plant-year for steps 1 and 2? Also, define an "intact containment" and explain how "intact containment severe accident scenarios" contribute to dose in the EPZ. Finally, provide examples of what is meant on page 15 by "precluded by design."

<u>Response</u>

Regarding the first part of the question on applying a SOARCA-like process, the scenario selection process in the white paper uses 1E-8 per plant year as an initial step but not as a cut-off frequency. The process does the following:

- *The initial step is accident sequences with mean CDF greater than 1E-8 per plant year are selected and grouped into accident scenarios.*
- *Then for Criterion b, recognizing that SMR CDFs are typically quite low and that some SMR designs may not have any accident sequences greater than 1E-8 per plant year, there is an additional step in Section 3.4 to address even lower frequency intact containment sequences.*
- *Finally, as described in Section 4.3, it is specified that the 1E-8 per plant year accident sequence frequency be extended to lower frequencies to assess potential cliff-edge effects.*

Regarding the second part of the question on intact containment, such scenarios are those with core damage where containment functions as designed (i.e., isolates and remains intact). These scenarios would tend to be the more probable, less severe accidents to be addressed in Criterion b. They contribute to dose due to design basis (technical specification) leakage, and are to be compared to the EPA 1 rem and 5 rem TEDE PAGs.

Regarding "Precluded by design", this generally means a situation in a given design where a certain phenomenon or contributor to risk has been addressed in the design in such a manner as to prevent the occurrence of the phenomenon or to mitigate its risk impact. Referring to the third bullet on page 15 of the white paper on basemat melt-through, an example would be flooding up around the reactor vessel such that the molten core would not penetrate the lower reactor vessel head. Another example would be providing water and appropriate geometry on the containment floor to cool the debris. Phenomena precluded by design will have to be justified by the applicant.

11. With respect to the analysis done against Criterion c, the proposed methodology states that fuel handling accidents and spent fuel pool accidents will be considered. How are these accident scenarios determined?

<u>Response</u>

Each SMR vendor will determine the appropriate fuel handling and spent fuel pool accident scenarios for their specific design. Safety analyses for fuel handling and spent fuel storage, PRA, and other, more deterministic information are mechanisms that could be used to identify which beyond design basis accident scenarios are credible, and are to be addressed in establishing the technical basis for EPZ size. Examples of design basis accident scenarios that have been included in large light water reactor designs include: dropping a fuel assembly, dropping a cask loaded with fuel assemblies, and dropping a heavy load on the spent fuel storage racks. Some SMR designs could incorporate features that preclude these accident scenarios, or introduce different scenarios. The DC applicant will include the appropriate fuel handling and spent fuel accident scenarios in their submittal.

12. On page 15, it is stated that SMRs would need to use per plant year, rather than per reactor year, indicating that accidents occurring on more than one reactor coincidentally will be

considered. How many coincident core damage events will be assumed to develop the design basis accident offsite dose estimates to compare against the Protective Action Guides?

Response
Use of "per plant year" is specified in the white paper to account for the potential impact on the core damage frequency of a single module from the multi-module design of the plant (e.g., use of common or shared systems), <u>not because a single initiating event is postulated to cause simultaneous core damage accidents in more than one reactor</u>. As noted in Section 3.5, multi-module accidents are expected to be addressed as part of Criterion c, and the industry response to NRC question 13 provides additional information on addressing multi-module accidents. It is expected that the more probable, less severe core damage scenarios addressed for Criterion b (versus the PAGs) would not involve multi-module accidents due to the very low frequency of multi-module events. This would need to be confirmed on a design-specific basis.

Use of per plant year may not be appropriate for all SMR designs. If use of per plant year is not appropriate, the individual design certification applicant should provide justification.

13. On page 18, it is stated that an accident source term evaluation must consider multi- module accidents, if they are credible. Please describe the basis for the conclusion that "source terms and associated dose would not be expected to be additive" for multi- module SMRs.

Response
It is expected that SMRs incorporate features, operational strategies and design options to prevent multiple core damage sequences from occurring and demonstrate that these accident sequences are not significant contributors to risk. Applicants will also demonstrate that operational strategies provide reasonable assurance that there is sufficient mitigation ability in the unlikely event of multiple core damage accidents. This is consistent with draft technical criteria for evaluating multi-module risk that the NRC has proposed and plans to issue in guidance (see response to Question 6).

In the unlikely event of a multi-module accident at an SMR, source terms and doses would not be additive because core damage progression would not be expected to be the same from one module to another (i.e., it would be staggered – not coincident). Individual SMR applicants will provide additional design-specific information on this.

Individual applicants will provide additional details to demonstrate that these expectations are met.

14. Page 18 lists five documents that provide the basis for accident sequence selection for operating reactor designs. How will these be adapted for SMRs?

Response
It is not intended that the documents which were listed in the white paper be "adapted for SMRs". The documents were cited only for information so as to note precedents where frequency (1E-7 per year in

these cases) was used as a basis for sequence selection or was used, or proposed as reasonable for use, to inform an evaluation or decision. As described in the white paper, industry intends to use frequency to inform scenario selection as part of the EPZ sizing basis. A SOARCA-like process, adapted for SMRs, will be applied. Further clarification on this is provided in response to Question 10.

15. The paper states that specifics of the methodology for determining the probability of dose exceedance will need to be defined as part of implementation. Can more details on this topic be provided?

Response
As discussed on page 19 of the white paper, the intent is to use the NUREG-0396 methodology for determining the probability of dose exceedance. While individual applicants may propose variations to adapt this methodology to their design, the following provides an example of calculating probability of dose exceedance using a conditional probability approach. Absolute probability could also be used, as discussed further in the response to NRC question 17.

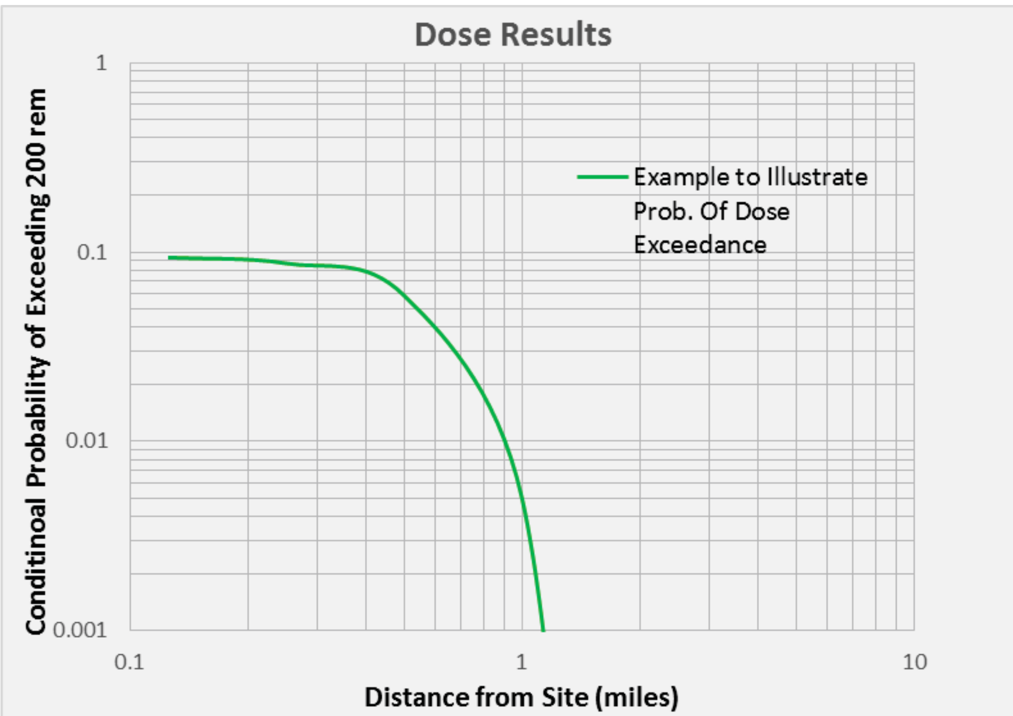
If conditional probability is used, in equation form the probability of dose exceedance is as follows:

Consider n scenarios, with core damage frequencies $f_1, f_2, \ldots, f_n$. Let the conditional probability of dose exceedance (given core damage) for scenario i at distance j be p_{ij}. Then summed over all scenarios, the conditional probability of dose D exceeding a given dose D_0 at distance j is

$$p_j(D > D_0) = \sum_{i=1}^{n} f_i p_{ij} / f_i$$

A simple numerical example illustrating the steps for determining conditional probability of dose exceedance is as follows. Consider 3 scenarios, S_1, S_2, and S_3 with frequencies (CDF per plant yr) as shown in the table below, across the top. The total CDF is shown in the top right hand cell. The conditional probability (given core damage) of dose exceeding 200 rem whole body acute (the NUREG-0396 dose for substantial early health effects) for each of the 3 scenarios is given for 5 distances from the plant, 0.125 miles to 1.5 miles. The conditional probability of dose exceeding 200 rem summed over all scenarios at a given distance is in the right hand column. From these values for the five distances in the numerical example, a curve similar to NUREG-0396, Figure I-11 can be plotted as shown in the figure below. In this example, the dose exceedance declines rapidly after 1 mile, and thus an EPZ of 1 mile provides for substantial reduction in early severe health effects in the event of more severe core melt accidents.

		Scenarios			
		S1	S2	S3	Total CDF
	CDF per plant yr	8.00E-06	5.00E-07	5.00E-08	8.55E-06
	Distance (mi)	Cond. prob. of exceeding 200 rem for scenario i at distance j			Total cond. prob. of exceeding 200 rem at distance j
1	0.125	0.05	0.7	1	9.36E-02
2	0.25	4.00E-02	7.50E-01	1.00E+00	8.71E-02
3	0.5	2.00E-02	6.00E-01	9.00E-01	5.91E-02
4	1	0.00E+00	8.00E-02	6.00E-02	5.03E-03
5	1.5	0.00E+00	0.00E+00	4.00E-04	2.34E-06



As noted in the white paper, the design specific methodology for determining the probability of dose exceedance will need to be defined as part of implementation including possible updates to the approach described in NUREG 0396. Further information on use of MACCS2 is provided in the response to NRC question 7.

16. What is the proposed probability basis for Criterion c (probability of dose exceedance)? Is it probability over weather trials; over scenarios; over accident classification (frequent, infrequent, severe); over type (internal, external, low power and shutdown, internal flood, internal fire, other); over release categories; or something else?

Response

Referring to the response to NRC question 15, the probability basis for Criterion c is probability over weather trials and probability over scenarios. The p_{ij} are the conditional probabilities of dose exceedance *(given core damage) for scenario i at distance j. The p_{ij} decrease with distance from the reactor due to there being fewer and fewer weather trials that have plume concentration high enough to sustain a given dose as distance increases. The f_i are the core damage frequencies of the scenarios. To get the conditional probability of dose D exceeding a given dose D_0 at distance j summed over all scenarios, the p_{ij} are weighted by the respective scenario CDFs and summed, and then normalized by dividing by total CDF.*

17. In the discussion of "Comparison against Early Severe Health Effect Risk" on page 19, SMR applicants are offered the option of using conditional or absolute probability of exceeding a whole body acute dose of 200 rem for showing that the EPZ size provides for a substantial reduction of early severe health effects. This is a fundamental parameter; therefore, explain why a consistent approach is not used. Clarify how the use of absolute probability does not contradict the concept that layers of defense-in- depth should be as independent as possible.

Response
There are valid reasons for an applicant to consider use of absolute probability in evaluating the criterion for substantial reduction in early severe health effects. These include:

- *Accident frequencies in SMRs are expected to be significantly lower than in large plants, and use of absolute accident frequencies will provide a better representation of risk than use of conditional probabilities.*
- *Prevention is traditionally an important layer of defense-in-depth in U.S. LWR designs. In SMR designs incorporating defense-in-depth principles, prevention will be based on features which are diverse and largely independent from mitigation features. In such designs, use of absolute probability would not contradict the concept of layers of defense-in-depth being independent.*

Use of conditional probability (NUREG-0396) has a visible, historical precedent, and is a way to address potential uncertainties in core damage frequency.

Industry believes that both approaches should be available to applicants, and that individual applicants should be able to use the approach most appropriate for their site. It is recognized that applicants would need to justify the approach selected. We would welcome a discussion of factors for EPZ applicants to consider in determining which approach to use.

18. On page 20, Regulatory Guide 1.200 and American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa 2009 are identified as the necessary guidance for demonstrating sufficient technical adequacy of the base PRA. This guidance does not address all relevant initiating events and operating modes, nor does it address Level 2 PRA. How will the lack of guidance in these additional important areas be addressed?

Response
NRC recently issued draft ISG-28 with guidance on expectations for DC/COLA PRA. Industry plans to comment on ISG-28 and applicants are expected to apply the resulting guidance to develop the PRAs required for DC. In support of the EPZ methodology, applicants are expected to develop design-specific

methods for relevant initiating events, operating modes, and Level 2 PRA and engage NRC on these methods, as well as refer to trial use guidance on Level 2 PRA, pending completion of pilot applications and NRC endorsement.

19. On page 21, the need for acceptance values for the level of uncertainty in plant core damage frequency (CDF) and large early release frequency (LERF) is discussed. The guidance given for establishing the acceptance values is not clear; please provide additional explanation. Also, provide the rationale for the suggestion that the smallness of a single SMR core should be credited when the metrics put forth are for a plant (i.e., per plant-year) and a plant may contain several cores.

Response

Regarding acceptance values for level of uncertainty in the proposed cumulative plant risk design objectives (total mean CDF < 1E-5 per plant year, mean LERF < 1E-6 per plant year), the PRA community has a long history of addressing uncertainty which indicates that the uncertainty on these metrics typically spans two orders of magnitude (i.e., ± factor of 10). Thus an acceptable way to establish acceptance values for level of uncertainty would be for the applicant to demonstrate that the mean values for the design are less than 10% of the risk design objectives. In addition, the potential for larger uncertainties can be addressed by demonstrating that the 95% values for the design are less than the risk design objectives.

It is expected that applicants will formulate the details for implementing the concept of factoring in the small core thermal power in the acceptance values of the risk metrics. The following provides a high level discussion that could be considered by applicants:

- *Risk is the product of likelihood and consequences. Thus consequences are a key part of risk metrics and determining risk significance. Due to the smaller core thermal power, SMR accidents are expected to result in smaller consequences.*
- *CDF is a widely used surrogate metric for risk in the U.S. commercial nuclear power industry for two reasons: (1) its relative simplicity and availability; and (2) operating U.S. nuclear plants have cores within roughly a factor of 2 on thermal power which means given core damage, consequences would tend to be comparable. Although SMRs use per plant year, instead of per reactor year, the CDF for SMRs are expected to be smaller due to the nature of iPWR designs.*

  In order for CDF to be interpreted as a measure of risk, the consequences need to be taken into account. For SMR applications, the CDF may need to be adjusted or normalized. Each applicant will develop an approach based upon the details of their design.

20. On page 21, while proposing cumulative plant risk design objectives for CDF and LERF, the paper states that "[T]he acceptance values should also factor in the smaller core power for SMRs." Please explain how a smaller core correlates with or changes acceptance values for CDF and LERF.

Response

See the response to NRC Question 19. As noted, the details for implementing the concept of factoring in the small core thermal power in the acceptance values of the risk metrics have not yet been formulated. Industry believes that individual SMR applicants should develop and apply the concept on a design-specific basis. An underlying consideration would be the fact that CDF and LERF should be normalized to the relative source term, as indicated by its surrogate core power, of the SMR designs.

21. Regarding the first bullet in Section 4.1 on page 22, would the design features of the facility that "facilitate application of [regional] assets" be described in the final safety analysis report? Would specific special treatment requirements be assigned to them?

Response
If required by the SMR design, features that facilitate the application of off-site assets will be described in the FSAR or equivalent license basis document as appropriate. It is expected this will be documented through the review and approval of the Design Certification Application and/or the Part 50/52 license application process. The treatment of these features in specific SMR designs will be commensurate with their role in addressing the PRA completeness uncertainty.

22. Regarding the second bullet in Section 4.1 on page 22, would the modeling of mitigation strategies in the Level 2 PRA (including use of severe accident management guidelines and extreme damage mitigation guidelines) and results of analysis with that Level 2 model be discussed in Chapter 19 of the final safety analysis report? How would the availability and reliability of the onsite portable equipment and regional assets be factored into the analysis?

Response
A level 2 PRA will be used by a COL applicant as part of the determination of the appropriate EPZ size. In doing this, mitigation strategies utilized in the analysis to achieve reduction in health effects in the event of more severe core melt accidents as well as less severe core melt accidents will be determined. The location of the description of the Level 2 PRA assumptions and results for the EPZ size is a subject for the COL applicant and NRC to determine.

If the SMR design relies on portable equipment and/or regional assets as part of the basis for the EPZ size, then the use of these inputs will be described in sufficient detail, either qualitatively or quantitatively as appropriate, to support the staff's safety review. Availability and reliability of offsite assets for use in the PRA will be well-established by the time the Level 2 PRA is completed and used for EPZ sizing. On-site asset availability and reliability analyses will be design and even site specific, hence will be addressed during the COL application stage.

23. The paper is not clear on how risk insights and defense-in-depth considerations will collectively inform the size of the plume exposure EPZ. Provide an example with explanations on how

these concepts are merged to inform the methodology and criteria for determining plume exposure SMR EPZ size.

<u>*Response*</u>
Risk insights from the PRA and insights from more qualitative, deterministic evaluations of defense-in-depth features will be weighed so as to develop SMR emergency planning that meets the 10 CFR 50.47 requirement to provide "reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency." The objective is that SMR emergency planning will provide this reasonable assurance, just as the current 10-mile EPZ does for the existing large LWR operating fleet.

Specific examples with explanations on how risk insights and defense-in-depth considerations are considered in the methodology will be developed and refined as part of actual design-specific applications of the methodology. These examples will be discussed with the NRC during pre-application interactions.

24. Regarding the third bullet in Section 4.2 on page 23, please describe how the plant simulator would be used to address uncertainty associated with control room layout, shift staffing, emergency response, and operating procedures.

<u>*Response*</u>
The plant simulator will be used to address uncertainty associated with the CR layout using principles and guidance provided by the NRC in the Human-System Interface (HSI), Task Analysis Design Elements of HFE as described in NUREG-0711 Revision 3, design as well as functional analysis guidance in IEC 61839 and IEC 60964. The uncertainty in shift staffing will be evaluated with the plant simulator using the principles and guidance that are delineated in NRC guidance including the HFE elements of operating experience review, Information Notice 95-48, and integrated system validation (NUREG-0711 Revision 3, Sections 6.4, 11.4.3.2, 11.4.3.4). Uncertainty in emergency response and operating procedures will be evaluated with the plant simulator using NRC guidance including the HFE element of Human System Interface (HSI) as described in NUREG-0711 Revision 3 Chapter 9 as well as guidance in NUREG-0654 Revision 1 and NUREG-0696.

25. Several options for addressing low frequency/high consequence events (cliff-edge effects) are identified in Section 4.3. Will these options be evaluated further to determine pros and cons and under what conditions one option would be better than another?

<u>*Response*</u>
The intent of the white paper is that the applicant evaluate potential cliff-edge effects in a technically sound manner based on the guidance in the white paper, addressing all credible (i.e., physically plausible) accident sequences. Each applicant will evaluate the options and implement the methodology accordingly. Referring to Section 4.3, bullets 1 or 2, bullet 3 and bullet 4 are considered necessary to address, with other bullets optional.

26. Throughout the paper, enhanced plant capabilities are referenced. Please clarify what is meant by this phrase.

<u>*Response*</u>
Enhanced plant capabilities are those features that are not required to address design basis events, but rather provide defense-in-depth. Page 2 of the white paper states that, "…The enhanced plant capabilities in the methodology addresses this by: providing a balance between deterministic, defense-in-depth considerations and risk considerations." Page 21 states that, "… Section 4 discusses additional steps, in the form of enhanced plant capabilities to account for uncertainties... As noted in Section 1 this is a complement to the PRA-based evaluation, and in large part is a deterministic, defense-in-depth approach."

There are four areas where enhanced plant capabilities are relied upon for the determination of an appropriately sized EPZ:

- *Address completeness uncertainty including developing a diverse and flexible operationally-focused strategy addressing both prevention and mitigation on a design-specific basis.*
- *Address potential risks that are difficult to quantify or not fully addressed in the PRA*
- *Assess potential impact on risk of lower frequency accidents (cliff edge effects)*
- *Provide balance between accident prevention, accident mitigation, and protective actions (an essential property of defense-in-depth).*

Enhanced plant capabilities are largely design-specific and are expected to be the subject of discussion during review of specific applications.

27. Please expand on aspects of this proposed methodology as they are specifically related to qualitative and quantitative approaches that contribute to decreasing the current 10 mile plume exposure EPZ.

<u>*Response*</u>
Quantitative approaches to the SMR EPZ determination in the proposed methodology are primarily risk-based as described in Section 3 of the white paper. Section 4, Additional Steps to Account for Uncertainties, is mainly deterministic and is more qualitative, emphasizing additional layers of defense-in-depth. As noted in the response to Question 23, specific examples with explanations on how a balance between risk insights and defense-in-depth considerations are considered in the methodology will be developed by SMR vendors and discussed with the NRC in pre-application interactions. These examples will expand on aspects of the proposed methodology as they are specifically related to qualitative and quantitative approaches.