REQUESTS FOR ADDITIONAL INFORMATION

1. Probabilistic Risk Assessment Approach (ADAMS Accession No. ML060950275)

[PRA-1] How are the similarities and differences between the PBMR PRA and those that are conventional for LWRs accounted for in establishing the acceptability of the PBMR PRA? What is the process for assessing the applicability of LWR data to the PBMR? How will needed data be obtained if either no LWR data is available or existing LWR data is not applicable? Provide some examples of LWR data that is considered applicable to the PBMR, and some examples of LWR data that is not considered applicable to the PBMR.

1. Additionally, referring to p. 52, Table 6, what is the technical basis for applying LWR piping experience and pipe reliability models to the PBMR? Will this basis be justified based on a fundamental metallurgical and mechanical understanding of PBMR HPB degradation mechanisms and degradation rates, etc.? Provide Reference 29 for review.

[PRA-2] Define the terms "regulatory design criteria," "design acceptance criteria," and "deterministic safety criteria," and provide a few specific examples of each term. For example, do "design acceptance criteria" include items such as safety limits, limiting conditions for operation, or limiting safety system settings (i.e., items that would be included in technical specifications) or SSC design limits (i.e., items that would be included in a design document)? How are "regulatory design criteria," "design acceptance criteria," and "deterministic safety criteria" determined (e.g., through consideration of regulations, regulatory guidance, codes and standards, and/or good engineering practices)? Are these criteria determined exogenously or are they estimated as requirements on the basis of the PRA? What are the interrelations among these terms?

[PRA-3] It is stated (p.10) that "The PRA itself incorporates key inputs and analyses from deterministic codes and analytical methods covered by other papers." Will the PRA consequences use the deterministic codes and analysis methods results directly, or are they indirectly derived from the deterministic codes and analysis methods results?

[PRA-4] It is stated (p.14) that "The PRA that is performed to derive the LBEs for the PBMR DCA will be updated as necessary to reflect changes in the plant design, construction and operational stages to the extent needed to ensure that conclusions derived from the PRA in support of the licensing basis and design certification remain valid." Discuss how licensing stability can be achieved given that PRA updates may impact the selection of LBEs and the classification of SSCs. Specifically describe when and how possible new information from technology research and development, such as potential changes to the mechanistic source term calculation arising from research into fuel performance and fission product transport test results, will be incorporated into the PRA.

[PRA-5] It is stated (p.20) that "PBMR-specific risk metrics that are described in Chapter 3 of this paper will be used in lieu of the LWR-specific metrics CDF and LERF…" What are the PBMR risk metrics that reflect the PBMR safety design approach and how are they defined? Explain how the PBMR-specific risk metrics can be used to assess the risk importance of operational events, planned design changes, and proposed license amendments.

[PRA-6] It is stated (p.23) that sensitivity analyses will be performed "…to address other sources of uncertainty that are more difficult to quantify." Provide illustrative examples of PBMR-specific uncertainties that will be difficult to quantify wherein sensitivity analyses will be used.

Enclosure

2. **[PRA-7]** How will incompleteness due to lack of knowledge associated with the PBMR PRA be addressed for LBE selection?

[PRA-8] With reference to Table 1 (p.28): Summarize the methodology used in the PRA to calculate accident consequences. Specifically, discuss the overall computational approach to determine the mechanistic source terms and accident consequences, and identify the computer codes used. With respect to the lower right entry in Table 1, explain what is meant by a "conservative bounding treatment."

[PRA-9] Please give examples of PBMR success criteria and discuss how they were developed. In particular, discuss the analyses method(s) used to develop the success criteria.

[PRA-10] What approach/criteria (e.g., failure modes and effects analysis) have been developed for concluding that a failure does not impact a safety function of another SSC or cause an initiating event? What documentation will be developed that describes the screening process and its results, including the list of retained SSCs and screened-out SSCs along with the technical basis for screening? For SSCs within the HPB, provide examples of SSCs that were screened out and the basis for these decisions. How will the screening process and its results be peer reviewed, especially for PBMR-unique SSCs?

[PRA-11] It is stated (p.34) that “The PBMR safety design philosophy utilizes inherent safety features and passive SSC to perform the required safety functions.” For AOOs, it appears that active SSCs perform the safety functions. Explain why these safety functions are or are not considered to be “required” safety functions? For example, is the reactor protection system, which contains active SSCs, considered a required safety function?

[PRA-12] Please provide a table (similar to p.45, Table 5) for Cs-137 for the 60th year of operation of a PBMR, particularly with regard to dust-borne radioactivity plated out on HPB surfaces.

[PRA-13] It is stated (p.47) that “Event sequences with frequencies less than $1x10^{-8}$ per plant-year are not developed in terms of a quantitative consequence analysis…” Is the $1x10^{-8}$ per plant-year truncation limit considered on a mean value basis? How is event sequence uncertainty considered when comparing to the truncation limit? Explain how the truncation limit of $1x10^{-8}$ per plant-year is consistent with the ASME PRA Standard (ASME RA-Sb-2005) Supporting Requirements QU-B2 and QU-B3.

[PRA-14] In Figure 7 (p.48), the assessment of event sequences with frequencies less than $1x10^{-8}$ per plant-year appears to be made in the column corresponding to event sequences for only one reactor rather than for an eight- reactor module plant. Additionally, when event families are aggregated, event sequences may fall above the $1x10^{-8}$ per plant-year truncation limit. How are these factors being considered in deciding whether event sequences should be developed as possible LBEs?

[PRA-15] In the event tree for MPS (p.49, Figure 8) heat exchanger break for single module, no branch is provided for the safety function for “control chemical attack.” Why do the PBMR event trees (especially those that involve a loss of HPB integrity) not include a branch for the SSC(s) that provide this function and its assumed functional reliability? What are the relevant success criteria for control chemical attack for air and water ingress events?

[PRA-16] It is stated (p.50) that “Given the reduced reliance on active SSCs to perform safety functions, it is reasonable to expect that safety function failures will be dominated by events and conditions that exceed the design basis envelope for passive SSC…” Discuss the implications of this statement for BDBEs. Provide a few examples of safety function failures for BDBEs that identify which passive SSCs have failed. Provide similar information for events having a lower frequency than BDBEs.

[PRA-17] The largest pipe in a PBMR is much smaller than the largest LWR RCS pipe. Are HPB cross sectional area openings larger than the area associated with the largest HPB pipe considered for LBEs? Explain. How will uncertainty in the LWR data be addressed in developing break size vs. frequency for the PBMR. What is the sensitivity of the accident source term to the HPB break size and/or location?

[PRA-18] Do the initiating event frequencies include the failure of passive SSCs internal to the HPB? Operational experience (e.g., Fermi 1, AVR, Monju, THTR) indicates that failures of passive SSCs internal to the HPB are potentially risk-significant for non-LWR designs.

[PRA-19] How will uncertainty in the pipe rupture size for a given frequency be addressed in the PRA, specifically when an LBE transitions from one event frequency category (e.g., DBEs) to the next lower event frequency category (e.g., BDBEs).

[PRA-20] It appears that extensive use is made of iterative processes during design of the PBMR. For example, it is stated on Page 59 that "The safety design philosophy itself is rooted in deterministic safety assessment principles. Key design parameters…are based on the principle of preventing core damage and large releases from the fuel using deterministic methods and means." On Page 60, it is stated that "…once the LBEs are selected based on input from the PRA, the safety classification of SSC is performed using a deterministic method. This safety classification is then subjected to a rigorous and conservative deterministic safety analysis to demonstrate that the safety classified SSC are sufficient to ensure that the dose criteria for the DBA are met with sufficient margin." What rules are applicable to this iterative process, including the closure rules for declaring an end to iteration? For example, it appears that the chosen LBEs should dominate the risk associated with the initial PRA. How is this assured?

[PRA-21] What process was used to identify the sources of uncertainty (parametric, model, and completeness) in the PRA? Characterize the sources of uncertainty in terms of their impact on the PRA results and uses.

3. **[PRA-22]** It is stated (p.61) that "By design, PBMR has no damage states analogous to LWR core damage state in which a large fraction of the fission product inventory is released from the fuel as is postulated to occur in more severe core damage events that are modeled in typical LWR PRAs." Are there damage states for the PBMR that are analogous to the LWR core-damage state with frequencies less than $1x10^{-8}$ per plant-year? Explain why or why not.

[PRA-23] The paper indicates that PBMR PRA will be equivalent primarily to what is defined in the ASME PRA Standard (ASME RA-Sb-2005) as Capability Category II. For each Supporting Requirement, identify the planned or expected Capability Category and justify, giving due consideration to the planned uses of the PRA, the use of less than Capability Category III.

[PRA-24] Describe the process used to adapt the ASME PRA Standard (ASME RA-Sb-2005) to the PBMR PRA. Specifically, identify which requirements are applicable as-is, which requirements need modification or augmentation, and proposed new requirements. Explain the rationale behind the process, e.g., how the unique aspects of the PBMR design were considered, how passive systems are modeled in a PRA, how the requirements can be meaningfully applied to a PRA of a nuclear power plant that is in the design phase (as opposed to the operational phase), and how the standard will be revised to ensure that the PRA has a higher degree of technical acceptability to support its use in establishing a probabilistic approach to the licensing basis.

2. Licensing Basis Event Selection

(ADAMS Accession No. ML061930123)

[LBE-1] The PBMR approach refers to top level regulatory criteria (TLRC), particularly 10 CFR Part 20 and 10 CFR Part 50.34, to establish a frequency-consequence (fC) limit, at least in the region for frequencies >10^{-4}/plant-yr. However, the TLRC do not themselves associate particular dose limits with particular event frequencies, leaving the selection of fC curve anchor points subject to interpretation. Please clarify how the proposed curve provides enhanced safety and sufficient margin.

[LBE-2] For frequencies below 10^{-4}/plant-yr, it is not clear what basis was used to limit consequence, in other words, for a consequence bound on the fC curve. The first paragraph on page 33, referring to Figure 8, states, “There is not a corresponding dose limit shown on the figure for BDBEs….” Is the design capability of the PBMR such that BDBE doses will also meet accident public dose requirements? Please identify an appropriate limit on the consequence of BDBEs and the basis for selecting it.

[LBE-3] The prediction of performance of advanced reactor concepts, such as PBMR, involve greater uncertainties than the performance of existing LWRs. Categorizing events only on the basis of a computed mean value could lead to an inadequate treatment of the uncertainties inherent to the risk calculations and therefore an inadequate margin for uncertainty. Please clarify how uncertainties in event categorization will be addressed.

[LBE-4] In establishing the “transition break size” between the DBA and BDBA event regions, the mean frequency of pipe breaks is used to categorize HPB failure event sequences. A conservative upper bound of the pipe break frequency would be more consistent with current regulatory approaches. Please contrast this with the approach taken to risk-informing the break size for 50.46(a) and explain how it provides similar assurance of adequate protection.

[LBE-5] In the PBMR approach, no explicit deterministic criteria are imposed on the LBEs in any of the frequency regions. The calculation of DBAs in the DBE region (which show that the fC limit is not challenged by the DBEs they comprise even when only safety significant SSCs are credited) are not deterministic in the sense of the term as it has been used in LWR licensing. Deterministic LBEs could demonstrate defense-in-depth (e.g., provision of a low leakage barrier) and support siting considerations. Please identify appropriate deterministic criteria (e.g., redundant shutdown means, barrier retention) to compensate for completeness uncertainty.

[LBE-6] Cumulative limits on initiating events, monitored during plant operation, could help to confirm that assumptions in the PRA on initiating event frequencies are valid. Please identify appropriate limits on cumulative initiating event frequencies.

[LBE-7] With regard to the LBE categories, paragraph no. 4 on page 9 of the white paper lists the acceptable consequence limits. (Note that the 10 CFR 20 TEDE limit is an annual limit of 100 mrem, while the 10 CFR 50.34 limit is 25 rem per event.) The paragraph refers to “realistically calculated” Total Effective Dose Equivalent (TEDE) for AOOs and “conservatively calculated” TEDE for DBEs. Please elaborate on differences between realistic and conservative calculation methods (other than crediting non-safety-related SSCs) and provide examples.

[LBE-8] In SECY-03-0047, the staff recommended that for *deterministic* safety analysis of DBA and AOO event sequences, “deterministic engineering judgment to bound uncertainties” associated with PRA would be applied (for meeting SSC top level design criteria). (The Commission approved that recommendation in a subsequent staff requirements memorandum

[ML072190282].) As described in the white paper, deterministic safety analysis does not directly address the issue of supplementing the PRA with deterministic engineering judgment to select events. Please describe how this is to be accomplished.

[LBE-9] It appears that the basic margins for safety rely on relatively few calculations. Please provide examples of such calculations for each kind of event (AOO, DBE, DBA, and BDBE) describing the assumptions, inputs, methods, tools, and acceptance criteria employed in evaluating the results. Show which deterministic safety analysis rules (i.e., best estimate, conservative) would be applied for BDBEs with a 95%-tile that extends into the DBE region. It is expected that this will clarify the meaning of “mechanistically conservative” DBA calculation and the distinction between “realistically calculated” Total Effective Dose Equivalent (TEDE) for AOOs and “conservatively calculated” TEDE for DBEs.”

[LBE-10] The staff must ensure that the LBE selection process yields a robust set of LBEs (a set that encompasses a wide range of accident phenomenology, not just on the basis of probability). Please discuss how the proposed LBE selection methodology achieves this goal.

[LBE-11] Because of the relatively large uncertainties involved in estimating doses (or other accident consequences), their use as the only acceptance criterion seems insufficient for regulatory purposes. For example, PBMR-specific fuel-related licensing criteria for LBEs (i.e., AOOs, DBAs) such as a TRISO particle fuel failure fraction limit would be analogous to fuel damage. Please identify engineering acceptance criteria (expressed in terms of physical parameters such as temperature, pressure, fuel oxidation, etc.) to provide reasonable assurance of adequate protection.

[LBE-12] LWR accidents that result in substantial meltdown of the core have been used as DBAs for purposes of ensuring that the design of the reactor radiological containment system is sufficient to meet 25 rem at the EAB. Regulations state (10 CFR 50.34(a)(1)(ii)) that (among other things) the Commission will take into consideration:

> (D) … an applicant shall assume a fission product release[6] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated.

The associated footnote 6 states

> The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

Please discuss how such a “major accident” will be established for the PBMR.

[LBE-13] How will the uncertainty distributions for the event frequencies be used with the criteria for screening out events with a probability of less than about 10^{-8} per year. Please provide a range of examples of events that are just below 10^{-8}/plant-yr from the PBMR PRA that were screened out as candidate BDBEs along with the mean, 95%-tile, and 5%-tile for frequency of these events (e.g., as a table similar to Tables 1 and 2 for the BDBEs in Figure 8).

[LBE-14] Are the design criteria for SSCs associated with AOOs developed and assessed on a mean-value basis or a conservative basis?

[LBE-15] Describe the general magnitude of the added margins provided by the non-safety-related SSCs in meeting the dose acceptance criteria. To support this, please provide a comparison of DBA vs. DBE dose consequences (e.g., as a table or graph) for several of the limiting DBEs shown in Figure 5.

[LBE-16] Large breaks in the HPB are not identified as DBEs/DBAs. Why does the uncertainty associated with the frequency of large break probabilities not result in one or more large breaks in the HPB being a DBE/DBAs?

[LBE-17] The PBMR approach evaluates the AOOs, DBEs, and BDBEs in a cumulative manner to compare individual risk against the NRC Reactor Safety Goals. However, the method described on page 35 of the white paper for evaluating the QHOs ("adjust dose from EAB to midway in annular region") is not consistent with traditional probabilistic consequence models accepted by the NRC (e.g., MACCS). Please clarify the basis for concluding that this model permits assessment of the individual risk discussed in the QHOs.

3. Structures, Systems, and Components Classification (ADAMS Accession No. ML062400070)

[SSC-1] Referring to Item 2 (p.8), clarify the relationship between SSC classification and the role that SSCs provide in satisfying defense-in-depth needs. It seems reasonable that SCCs that are "significant" in providing defense-in-depth should be classified as safety-related and receive special treatment.

[SSC-2] Referring to Item 2 (p.8), explain why there is not a third category of safety-related SSCs under the heading “Safety Related” that are relied upon to perform required safety functions to mitigate Beyond Design Basis Events (BDBEs) with potential consequences greater than the 10 CFR 50.34 dose limits in order to prevent 10 CFR 50.34 dose limits from being exceeded.

[SSC-3] Referring to Item 3 (p.8), it is stated that “The special treatment for the safety-related (SR) category of classification is commensurate with that needed for the SSCs to perform their capability and reliability requirements during DBEs and high consequence BDBEs to meet the 10 CFR §50.34 dose limits.” For BDBEs, why are such SSCs not identified as safety-related under Item 2?

[SSC-4] Discuss how deterministic engineering judgment elements are used to supplement the probabilistic elements for the SSC safety classification scheme to avoid having a safety classification approach that is probability-based to the exclusion of other considerations.

[SSC-5] Referring to Figure 2 (p.18), explain why the safety functions “maintain control of radionuclides,” “maintain core and reactor vessel geometry,” and “maintain reactor building structural integrity,” (listed on Page 18) do not appear in Figure 2. Explain why the safety function “maintain reactor building structural integrity” is not included in the safety functions defined for the PRA (refer to Page 35 of the PRA white paper).

[SSC-5] It is stated (p.18) that “Safety analyses have been performed to determine which are the safety functions for the reactor sources.…” What is the basis for the mechanistic source term used in these “safety analyses” (e.g. circulating activity, plate-out activity, and fission products adsorbed in fuel matrix dust)?

[SSC-6] In general, is Figure 2 (p.18) representative of current PBMR understanding of fission product transport mechanisms that must be modeled in dose calculations? For example, what is the basis for concluding that control of transport from HPB and the reactor building are not needed to keep the DBEs within the offsite dose limits of 10 CFR 50.34? Does this conclusion imply that the HPB and the reactor building have small risk reduction factors, as defined in the DID white paper?

[SSC-7] Referring to Figure 2 (p.18) the staff understands that the shaded safety functions shown on the figure to identify those that are needed to keep the DBEs within the offsite dose limits of 10 CFR 50.34. It is stated that “The functions shown without shading are not required…” What are the required safety functions for AOOs (which are related to 10 CFR Part 20) and BDBEs (which are related to the latent QHO)? Will Figure 2 and the list of safety functions for AOOs, DBEs, and BDBEs be revisited after the source term calculation basis is finalized?

[SSC-8] It is stated on (p,22) that “If there is more than one alternative set of SSC that is determined to be available and sufficient, the set that is selected as the safety-related set is the one that reflects the highest level of confidence that it will perform its required safety function.”

Provide additional details about the process used to decide among alternative sets of SSCs. Specifically, discuss the roles of quantitative PRA results, deterministic considerations, and engineering judgment in the decision making process.

[SSC-9] Please provide tables similar to Tables 3 and 4 for all of the safety functions defined for the PBMR and also similar tables for NSRST SSCs for all event categories.

[SSC-10] Table 5 (p.25) indicates that the only safety-related SSC needed to control chemical attack is the reactor vessel. In contrast, Table 3 (p.38) of the PRA white paper indicates that other SSCs are also needed to control chemical attack (e.g., HPB piping, interfacing systems at lower pressure than the MPS, etc.). Explain this apparent discrepancy.

[SSC-11] Referring to Table 5 (p.25) explain why the reactor building and citadel are only needed to maintain core geometry, and how they do so. Are these SSCs also needed to support the safety function "maintain reactor building structural integrity," which is listed on Page 18?

[SSC-12] Referring to Section 3.3.6 (p.27) identify all SSCs that support the AOO safety functions whose failure leads to consequences in excess of the 10 CFR 50.34 criteria. How does the proposed classification scheme ensure these SSCs are classified as safety-related and receive appropriate special treatments?

[SSC-13] Referring to Section 3.3.7 (p.27), for SSCs that support DBE and BDBE safety functions, how is the selection made regarding SSC classification for safety-related versus NSRST? Could any of these SSCs actually be classified as NSRST? How does the process ensure that these SSCs are appropriately classified considering associated uncertainties in the supporting PRA?

[SSC-14] Referring to Figure 5 (p.31) discuss the significance of a hypothetical "Challenge D" located just to the right of the BDBE TLRC and with a frequency just above $5x10^{-7}$ per plant-year with respect to safety classification and special treatment of SSCs. Discuss the safety classification and special treatment of SSCs that prevent potentially very high consequence LBEs from increasing in frequency into the BDBE or DBE range. How are these SSCs identified and treated if there is a cutoff of $1x10^{-8}$ per plant-year for evaluation as proposed in the paper? Does the cutoff become valid only <u>after</u> the evaluation and special treatments are established?

[SSC-15] Referring to Table 7 (p.33) what is the basis or rationale (regulation, regulatory guide, code or standard, etc.) for each of the special treatments of safety related and NSRST categories? How do the special treatments for NSRST relate to the staff's philosophy for considering the regulatory treatment of non-safety systems (RTNSS), e.g., SECY-95-132 (ADAMS Accession No. ML003708005) and Section C.IV.9 of Regulatory Guide 1.206?

[SSC-16] Referring to Table 7 (p.33) what is the basis for the difference in level of treatment for safety-related SSCs versus NSRST SSCs for those situations where the needed capability and reliability for those labeled safety-related and NSRST are commensurate?

4. Defense-In-Depth
(ADAMS Accession No. ML063470549)

[DID-1] It is stated (p.12) that “Programmatic Defense-in-Depth is also germane to the selection of design codes and standards for the PBMR…” However, Page 33 indicates that the selection of codes and standards is part of the plant capability defense-in-depth. Please explain the different characterizations of codes and standards treatment.

[DID-2] It is stated (p.26) that “…off-site emergency planning is a licensing consideration that is outside the scope of a design certification.” Explain how the defense-in-depth framework and the PRA will be used to examine plant siting and emergency planning during applications for early site permits and combined licenses, given that the PRA only expresses accident consequences in terms of dose at the site boundary (PRA White Paper, Page 27).

[DID-3] It is stated (p.28) that “Defense-in-depth is an established philosophy in which multiple lines of defense and safety margins are applied to the design, operation, and regulation of nuclear plants to assure that the public health and safety are adequately protected.” This definition is vague and not particularly helpful in understanding the PBMR approach to defense-in-depth because it does not provide a basis for deciding the necessity and sufficiency of defense-in-depth (e.g., How many multiple lines of defense are needed? What safety margin is adequate?). Please clarify your approach by revising your definition of defense-in-depth to include a basis for determining necessity and sufficiency. For reference purposes, the Commission-approved definition for defense-in-depth, as presented in the NRC’s Strategic Plan is as follows:

> “Defense-in-Depth: an element of the NRC’s Safety Philosophy that employs successive compensatory measures to prevent accidents or lessen the effects of damage if a malfunction or accident occurs at a nuclear facility. The NRC’s Safety Philosophy ensures that the public is adequately protected and that emergency plans surrounding a nuclear facility are well conceived and will work. Moreover, the philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility.”

[DID-4] The PBMR proposed approach for DID apparently credits practically all engineering good practices and plant attributes (see Tables 2, 3, 6, and 9). How does the PBMR approach separate the additional plant features provided for defense-in-depth from the plant features needed to assure that the TLRC, RDC, and other acceptance criteria are met?

[DID-5] It is stated (p.30) that the “Risk-Informed Evaluation of Defense-in-Depth is the structured use of information provided by the PRA to identify the roles of SSCs in the prevention and mitigation of accidents, to identify and evaluate uncertainties in the PRA results, to devise deterministic approaches to address these uncertainties, and to guide and provide risk insights to support deterministic judgments on the adequacy and sufficiency of defense-in-depth.” However, the discussion on Pages 39–45 is limited to evaluating accident prevention and mitigation. Explain how deterministic judgment is used in the establishment of DID to compensate for uncertainties (in particular, completeness uncertainties) in the PRA.

[DID-6] The white paper indicates an aspiration (p.30) for a risk-informed and performance-based approach to defense-in-depth. The Commission’s definition of a performance-based approach (NRC’s Strategic Plan, NUREG-1614, Vol.3) has a key provision that states (in part): “…a framework exists or can be developed in which the failure to meet a performance criterion, while undesirable, will not in and of itself constitute or result in an immediate safety concern." What performance criteria other than dose at the site boundary exist to support defense-in-depth? Significant radiological dose at the site boundary implies that an immediate safety concern could occur if this criterion is the only measure used.

[DID-7] Discuss the "multiple barrier" approach defined by LWRs vs. HTGRs and their implications to defense in depth against fission product releases. Figures 7 and 8 indicate that the MHTGR (i.e., surrogate for the PBMR) barriers are multiple orders of magnitude different in their capabilities. All do not appear to be fully capable to meet dose acceptance criteria but are required multiple barriers that are all needed to meet acceptance criteria.

[DID-8] The PBMR approach (p.40) relies heavily on the defense-in-depth principles from SRP Chapter 19. These are principles that are to be maintained given a licensing change (e.g., design change) to the plant. These principles assumed that there is already adequate DID, and as such, does not define what is meant by DID. For a new design, an explicit definition of DID is needed, and therefore, the criteria for determining when there is enough DID (i.e., the basis for defining the minimal needed DID). Please provide such a definition and basis.

[DID-9] It is stated (p.40) that "Prevention strategies as defined as those strategies that are employed to reduce the frequency of accidents by improving the reliability of SSCs whose failure would cause initiating events and/or adversely affect the ability to mitigate an event sequence. Mitigation strategies are those that are employed to improve the capability of SSCs that serve to mitigate the consequences of events and event sequences that may challenge them." Figure 7 (p.53) makes these definitions explicit by indicating that, in the PBMR, "prevention" refers to reducing the accident sequence frequency and "mitigation" refers to reducing the accident sequence consequences. Explain why these definitions are chosen over the ones traditionally used for LWRs, e.g., "accident prevention" refers to preventing core damage and can be assessing by examining the core-damage frequency (CDF), and "accident mitigation" refers to preventing offsite releases, and can be assessed by examining the large early release frequency (LERF). Would it be helpful to define "prevention" in terms of frequency of significant releases from the fuel, and "mitigation" in terms of frequency of significant releases to the environment?

[DID-10] Referring to Table 5 (p.43), how is the table used in practice? Will the design certification application include documentation that identifies the plant features that provide defense-in-depth, classifies them as either plant capability or programmatic, provides their risk reduction factors, demonstrates that they meet the defense-in-depth principles listed in Table 5, etc.? For example, how will it be demonstrated that there is not over-reliance on programmatic approaches to compensate for design weaknesses (i.e., the principle in Table 5). How does engineering judgment factor into this demonstration?

[DID-11] Discuss how the PBMR defense-in-depth approach addresses aspects of the plant design that have unknowingly been omitted from the risk analysis (i.e., the "unknown unknowns").

[DID-12] The staff finds that Figure 5 (p.44) is not consistent with parts of the discussion in the paper. Please revise or clarify Figure 5 to make it consistent with the discussion in the white paper.

[DID-13] What is the role of the single failure criterion in the PBMR's defense-in-depth framework? Is it the intent of the white paper to propose a risk-informed alternative to the single failure criterion? If so, please articulate and justify that alternative. As context for developing the responses, it may be useful to consider the alternatives the staff has identified in SECY-05-0138, “Risk-Informed and Performance-Based Alternatives to the Single-Failure Criterion.”

[DID-14] For each LBE in each event category, do the SSCs that provide defense in depth have sufficient capability to meet the TLRC when the credited SSCs (e.g., the safety-related SSCs for DBAs) are assumed to fail? For LBEs in general, discuss the level of performance capability that will be required by the SSCs performing a defense-in-depth role to ensure that the TLRC and other criteria are met.

List of Acronyms

AOO	abnormal operational occurrence
AVR	Arbeitsgemeinshaft Versuchsreactor
ASME	American Society of Mechanical Engineers
BDBE	beyond design basis events
CDF	core damage frequency
CFR	Code of Federal Regulations
DBA	design basis accident
DBE	design basis event
DCA	design certification application
DCD	design control document
DID	defense-in-depth
EAB	exclusion area boundary
fC	frequency-consequence
FHSS	fuel handling and storage system
FMEA	failure modes and effects analysis
HPB	helium pressure boundary
HTGR	high-temperature gas-cooled reactor
LBE	licensing basis event
LRF	large release frequency
LWR	light-water reactor
MSPI	mitigating system performance indicator
MHTGR	modular, high-temperature gas-cooled reactor
NSR	non-safety-related
NSRST	non-safety-related with special treatment
PBMR	pebble-bed modular reactor
PCS	primary coolant system
PRA	probabilistic risk assessment
QHO	quantitative health objective
ROP	reactor oversight process
RCS	reactor coolant system
SDP	significance determination process
SR	safety-related
SSC	structures, systems, and components
THTR	Thorium High Temperature Reactor
TLRC	top-level regulatory criteria
WP	white paper