OFFICE OF NEW REACTORS

SUMMARY FEEDBACK ON FOUR KEY LICENSING ISSUES

NEXT GENERATION NUCLEAR PLANT
PROJECT 0748

INTRODUCTION

The U.S. Department of Energy (DOE) and its Idaho National Laboratory (INL) (hereafter referred to collectively as DOE/INL) established the Next Generation Nuclear Plant (NGNP) Project as required by Congress in Subtitle C of Title VI of the Energy Policy Act of 2005 (EPAct). The mission of the NGNP Project is to develop, license, build, and operate a prototype high-temperature gas-cooled reactor (HTGR) plant that generates high-temperature process heat for use in hydrogen production and other energy-intensive industries while also generating electric power. To fulfill this mission, DOE/INL is considering a modular HTGR with either a prismatic block or pebble bed core.

As stipulated by the EPAct, DOE/INL and the U.S. Nuclear Regulatory Commission (NRC) have been engaged in prelicensing interactions on technical and policy issues that could affect the design and licensing of the NGNP prototype. Such early interactions are encouraged by the Commission's policy statement on advanced reactors.[1]

As outlined by the NRC in a letter to DOE dated February 15, 2012, the NRC staff has since focused its NGNP interactions with DOE/INL on the further assessment of technical and policy issues in key areas previously highlighted in the NGNP Licensing Strategy Report that NRC and DOE jointly issued to Congress in 2008.[2] This document discusses these issues under the following four headings:

(1) Licensing basis event selection
(2) Source terms
(3) Containment functional performance
(4) Emergency preparedness

DOE/INL has engaged the NRC staff on its proposed approaches to such issues primarily through a series of white paper submittals. In February 2012, the NRC provided its preliminary

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[1] "Policy Statement on the Regulation of Advanced Reactors," Volume 73 of the *Federal Register*, page 60612 (73 FR 60612); October 14, 2008

[2] "Next Generation Nuclear Plant Licensing Strategy – A Report to Congress," August 2008, (NRC Agencywide Documents Access and Management System (ADAMS) Accession No. ML082290017)

feedback to DOE/INL in the form of two initial assessment reports (ML120240671).[3] Subsequent interactions have largely focused on addressing issues and follow-up items identified in those initial assessment reports. DOE/INL brought further focus to these interactions in its letter to NRC dated July 6, 2012 (ML121910310).

The remainder of this NRC staff document summarizes and consolidates the staff's views in terms of the July 6th letter's specific requests for feedback under each of the four key issue headings. Specifically, this document provides staff feedback under bulleted subheadings that quote or paraphrase the specific requests from the July 6th letter. More detailed NRC staff comments on these and related issues are provided in the following two updated white paper assessment reports:

(1) "Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms," Revision 1, [insert date of issuance when available] (ML13002A162).

Note that the issue discussions that follow refer to this NRC assessment report on fuel qualification (FQ) and mechanistic source terms (MST) as the "FQ-MST assessment report." The discussions refer to the subject DOE/INL white paper submittals as the "FQ white paper" (ML102040261) and the "MST white paper" (ML102040260), respectively.

(2) "Assessment of White Paper Submittals on Defense in Depth, Licensing Basis Event Selection, and Safety Classification of Structures, Systems, and Components," Revision 1, [insert date of issuance when available] (ML13002A168).

Note that the three subject white paper submittals collectively describe DOE/INL's proposed risk-informed, performance-based (RIPB) licensing approach. For brevity, the issue discussions that follow refer to this NRC assessment report as the "RIPB assessment report." The DOE/INL white paper submittals on defense in depth (DID), licensing basis event (LBE) selection, and safety classification of structures, systems, and components (SSC) are referred to respectively as the "DID white paper" (ML093480191), the "LBE white paper" (ML102630246), and the "SSC white paper" (ML102660144).

RESPONSES TO SPECIFIC REQUESTS FOR NRC FEEDBACK

The responses provided here reflect the NRC staff's evolving interest in pursuing risk-informed, performance-based approaches for licensing advanced reactors. In an August 2012 report to Congress on advanced reactor licensing, the NRC staff indicated its initiative to streamline its review of new reactor licensing applications.[4] In that report, the NRC discussed its approach to licensing light-water small modular reactor (SMR) designs (also known as integral pressurized water reactor (iPWR) designs) and non-light-water reactor (non-LWR) advanced reactor designs. The approach includes: "(1) use a more risk-informed and integrated review framework for staff preapplication and application review activities pertaining to iPWR design applications; and, (2) develop, over the longer term, a new risk-informed, performance-based regulatory structure for licensing non-LWR advanced reactor designs (e.g., high-temperature, gas-cooled reactors (HTGRs) and liquid-metal reactors (LMRs))."

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[3] Note that this and subsequent references to ADAMS omit "ADAMS Accession No." for brevity.

[4] *Report to Congress: Advanced Reactor Licensing*, August 2012, (ML12153A014)

The NRC staff statements that follow, as with those in the NRC staff assessment reports on related NGNP white paper submittals, do not provide a final regulatory decision on any aspect of the NGNP design concepts because such conclusions would be provided in the NRC staff's safety evaluation of a future license or design certification submittal. Final licensing decisions are based on well defined designs described and analyzed in legally binding application submittals. Although these views represent the engineering and licensing judgment of a subset of staff members, they are based on generic modular HTGR design concepts rather than a specific design, and they were not developed in the context of a docketed license application. Therefore, these NRC staff statements are advisory. Moreover, since many of the technical and policy issues cannot be addressed or resolved until more specific and detailed information about the NGNP design is available, the staff views presented here are subject to change and may require future Commission approval.[5] The views presented here are based on consideration of previous NRC staff recommendations and Commission policy precedence. The staff identifies certain issues as potential Commission policy issues, meaning that the staff would likely ask the NRC Commissioners for policy guidance in resolving such issues.

Lastly, although DOE/INL has framed some of its feedback requests using words like "accept," "acceptable," and "endorse," it bears noting that such words have legal/regulatory connotations that would not be appropriate in this context. The staff instead addresses such requests in terms of whether DOE/INL's proposed approaches to the respective issues are "reasonable."

1. Licensing Basis Event Selection

History of Pertinent NRC Staff and Commission Positions

In SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU-3 Designs and Their Relationship to Current Regulatory Requirements," dated April 8, 1993, the NRC staff provided positions for the Commissioners to consider in providing policy guidance on a risk-informed licensing structure that would be acceptable. Included was a discussion of accident analysis and licensing basis event evaluation. Note that SECY-93-092 was based, in part, on the NRC staff's preapplication review efforts, as documented in NUREG-1338, "Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor [MHTGR]" issued March 1989, for a proposed modular HTGR design and licensing approach very similar to those now proposed for NGNP.

The following statements from SECY-93-092 describe the evaluation approach that the NRC staff proposed for all advanced reactor designs:

- Events and sequences will be selected deterministically and will be supplemented with insights from probabilistic risk assessments of the specific designs.

- Categories of events will be established according to expected frequency of occurrence. One category of events that will be examined is accident sequences of a lower likelihood than traditional light-water reactor (LWR) design-basis accidents. These accident sequences would be analyzed without applying the conservatisms used for design-basis

[5] The term "Commission," as used in this document, refers to the five appointed NRC Commissioners, whereas the term "staff" refers to NRC career staff.

accidents. Events within a category equivalent to the current design-basis accident category will require conservative analyses, as is presently done for LWRs.

- Consequence acceptance limits for core damage and onsite/offsite releases will be established for each category to be consistent with Commission policy guidance.

- Methodologies and evaluation assumptions will be developed for analyzing each category of events consistent with existing LWR practices.

- Source terms will be determined as approved by the Commission in Section B [SECY-93-092, Section B "Mechanistic Source Term"]

- A set of events will be selected deterministically to assess the safety margins of the proposed designs, to determine scenarios to mechanistically determine a source term, and to identify a containment challenge scenario.

- External events will be chosen deterministically on a basis consistent with that used for LWRs.

- Evaluations of multi-module reactor designs will be considered as to whether specific events apply to some or all reactors on site for the given scenario for all operations permitted by proposed operating practices.

In the staff requirements memorandum (SRM) dated July 30, 1993, to SECY-93-092, the Commission approved these evaluation principles for advanced reactors.[6] The NRC staff then reviewed these principles and refined them in SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," dated March 28, 2003.

In SECY-03-0047, the NRC staff proposed to place greater emphasis on the use of risk information by allowing the use of a probabilistic approach for identifying events to be considered in an applicant's design bases, provided that plant and fuel performance are sufficiently understood and deterministic engineering judgment is used to bound uncertainties in the applicant's analysis. Specifically, the staff recommended in SECY-03-0047 that the Commission should take the following three actions to define the extent to which a probabilistic approach can be used to establish the licensing basis:

(1) Modify the Commission's guidance, as described in the SRM of July 30, 1993, to SECY-93-092, to put greater emphasis on the use of risk information by allowing the use of a probabilistic approach in the identification of events to be considered in the design, provided there is sufficient understanding of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties.

(2) Allow a probabilistic approach for the safety classification of structures, systems, and components.

(3) Replace the single failure criterion with a probabilistic (reliability) criterion.

[6] As noted subsequently in this document, the Commission's approval regarding MHTGR containment challenge scenarios stated that the staff should also consider the potential for "chimney effect" air-ingress events with graphite oxidation.

The Commission then approved these recommendations without revision in the SRM to SECY-03-0047. Note that these approved recommendations are consistent with a risk-informed approach in that they extend the use of probabilistic risk assessment (PRA) into forming part of the basis for licensing and thereby place greater emphasis on PRA quality, completeness, and documentation. Additionally, the staff provided updates to the Commission on the development of a regulatory structure for new plant licensing in SECY-04-0157, "Status of Staff's Proposed Regulatory Structure for New Plant Licensing and Potentially New Policy Issues," dated August 30, 2004, and in SECY-05-0006, "Second Status Paper on the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing," dated January 7, 2005.

In December 2007, the NRC staff published NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," Volume 1, "Main Report," and Volume 2, "Appendices A through L," which explored the feasibility of developing a risk-informed and performance-based regulatory structure for the licensing of future nuclear power plants. As such, NUREG-1860 documents a framework that provides an approach, scope, and criteria that could be used as a guide to develop a set of new regulations to serve as an alternative to current regulations for licensing future nuclear power plants.

In August 2008, the NRC and DOE jointly issued to Congress the NGNP Licensing Strategy Report. The strategy report describes four options for adapting existing NRC regulatory requirements. These options range from a deterministic approach similar to that used for current reactors to a new set of risk-informed and performance-based regulatory requirements. DOE and the NRC endorsed Option 2, a risk-informed and performance-based approach that uses deterministic engineering judgment and analysis, complemented by NGNP design-specific PRA information, to establish the licensing basis, including the selection of licensing-basis events (LBEs) and licensing technical requirements. Use of PRA would be commensurate with the quality and completeness of the PRA presented with the application.

On July 12, 2011, the NRC published the report, "Recommendations for Enhancing Reactor Safety in the 21st Century – The Near-Term Task Force [NTTF] Review of Insights from the Fukushima Dai-ichi Accident." NTTF Recommendation 1, the first of the report's twelve overarching recommendations, is to establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations. The NRC staff is now developing options for addressing this recommendation.

In April 2012, the NRC staff published NUREG-2150, "A Risk Management Regulatory Framework," which describes the results of a task force study on a proposed a risk management regulatory approach that could be used to improve consistency among the NRC's various programs. Commissioned by NRC Chairman Gregory Jaczko and headed by Commissioner George Apostolakis, the task force's charter was to develop a strategic vision and options for adopting a more comprehensive, holistic, risk-informed, performance based regulatory approach for reactors, materials, waste, fuel cycle, and transportation that would continue to ensure the safe and secure use of nuclear material. The proposed risk management regulatory framework builds upon well-established practices, such as the NRC's defense-in-depth philosophy and its policies to incorporate risk-informed and performance-based approaches into the agency's regulation and oversight of byproduct, source, and special nuclear materials.

DOE/INL proposes a new risk-informed and performance-based approach for NGNP similar to those that have been or may be considered for NUREG-1860, NUREG-2150, and NTTF Recommendation 1. A revised or new framework resulting from these other efforts may change the current NRC staff positions discussed in this document and the FQ-MST and RIPB assessment reports.

Responses to DOE/INL Requests Concerning Licensing-Basis Event Selection

- ***DOE/INL Request*: Agree on key terminology and naming conventions for event categories.**

DOE/INL first described its proposed approach to event selection and categorization in its subject white paper submittals and subsequently clarified and refined the approach during assessment interactions with the NRC staff. Through those interactions, the staff understands that DOE/INL currently proposes the following set of event category names and descriptions:

- Anticipated Events (AEs): AEs encompass planned and anticipated event sequences. The doses from AEs are required to meet normal operation public dose requirements. AEs are utilized to inform operating conditions for normal operation modes and states.

- Design Basis Events (DBEs): DBEs encompass unplanned, off-normal event sequences not expected in the plant's lifetime, but which might occur in the lifetimes of a fleet of plants. The doses from DBEs are required to meet accident public dose criteria for reactor siting. DBEs are the basis for the design and construction of all SSCs and their operation during accidents.

- Design Basis Accidents (DBAs): DBAs are DBEs with only safety-related equipment successfully responding. The doses from DBAs are required to meet accident public dose criteria for reactor siting. DBAs are the basis for the design and construction of safety-related SSCs and their operation during accidents.

- Beyond Design Basis Events (BDBEs): BDBEs are rare, off-normal event sequences of lower frequency than DBEs. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public.

The NRC staff concludes that the above category names and descriptions are generally reasonable but should be clarified and supplemented in view of the discussion that follows.

In its white paper submittals, DOE/INL initially named the first event category "anticipated operational occurrences" (AOOs). At a public meeting with the NRC staff in August 2012, DOE/INL decided to change the name to AEs when the staff noted that the proposed event category differs from the AOOs traditionally considered for LWR licensing. DOE/INL has stated its intention to use this new event category name in any future submittals for NGNP. In subsequent assessment interactions with the staff, DOE/INL clarified that AEs would be considered in establishing operating conditions and associated administrative controls (i.e., plant operating procedures, Technical Specification limits, etc.) intended to ensure that the plant responds as expected and that event releases are compliant with annual public dose limits.

The NRC staff has found it useful to understand the distinction been the proposed AE category and the current AOO category for LWRs. Specifically, an AOO for LWRs is an event of moderate frequency (e.g., a loss of all offsite power (LOOP)) with only safety-related equipment responding successfully. As proposed by DOE/INL, all AEs, DBEs, and BDBEs are complete event sequences that include the plant's full response to an initiating event. For example, a relatively frequent LOOP sequence with all available safety-related and non-safety-related SSCs responding as intended would be categorized by DOE/INL as an AE if the frequency of that sequence were shown to lie within the proposed AE frequency range. By the same token, a less frequent LOOP sequence with one or more non-safety-related SSCs failing to respond would be categorized as a DBE if that sequence's frequency were shown to lie within the proposed DBE frequency range.

AEs, DBEs, and BDBEs collectively comprise the events that DOE/INL's white paper submittals refer to as licensing basis events (LBEs). Although portions of DOE/INL's LBE white paper submittal describe DBAs as distinct from LBEs, DOE/INL subsequently explained that DBAs are intended to be a subset of LBEs and that DBA evaluation would comprise an essential part of the licensing basis. DOE/INL's proposal to define DBAs as PRA-derived DBEs in which only safety-related equipment is assumed to be available represents a substantial departure from the postulated single-failure DBAs currently used in LWR licensing.

The NRC staff would likely seek Commission guidance in selecting a set of DBAs for NGNP licensing. The NRC's final selection of DBAs would be based in part on reviewing the DBE-derived DBAs and other risk-informed LBEs proposed by DOE/INL and may include additional sequences from the proposed BDBE frequency range as well as physically plausible deterministic event sequences postulated by the staff. All DBAs would assume that only SSCs classified as safety related are available. Dose consequence compliance criteria for all DBAs would be evaluated at the 95% upper confidence bound of the best-estimate mean value. Approval of specific reliability criteria to replace single-failure criteria in DBAs would likely necessitate a Commission policy decision. As discussed in the FQ-MST assessment report, the NRC staff notes that rod ejection events are among the postulated accidents analyzed for LWRs and that such events would likely have to be analyzed for NGNP unless they are precluded by specific design features (i.e., unless they are not physically plausible).

- ***DOE/INL Request*: Agree with the placement of top-level regulatory criteria (TLRC) on a frequency-consequence (F-C) curve.**

The NRC staff concludes that the proposed approach to placing top level regulatory criteria (TLRC) on a frequency-consequence (F-C) curve is reasonable. Figure 1, a copy of DOE/INL's proposed F-C curve, identifies the regulatory bases for the respective TLRC and displays the TLRC-based dose limits as a function of event sequence mean frequency over the proposed event category regions.

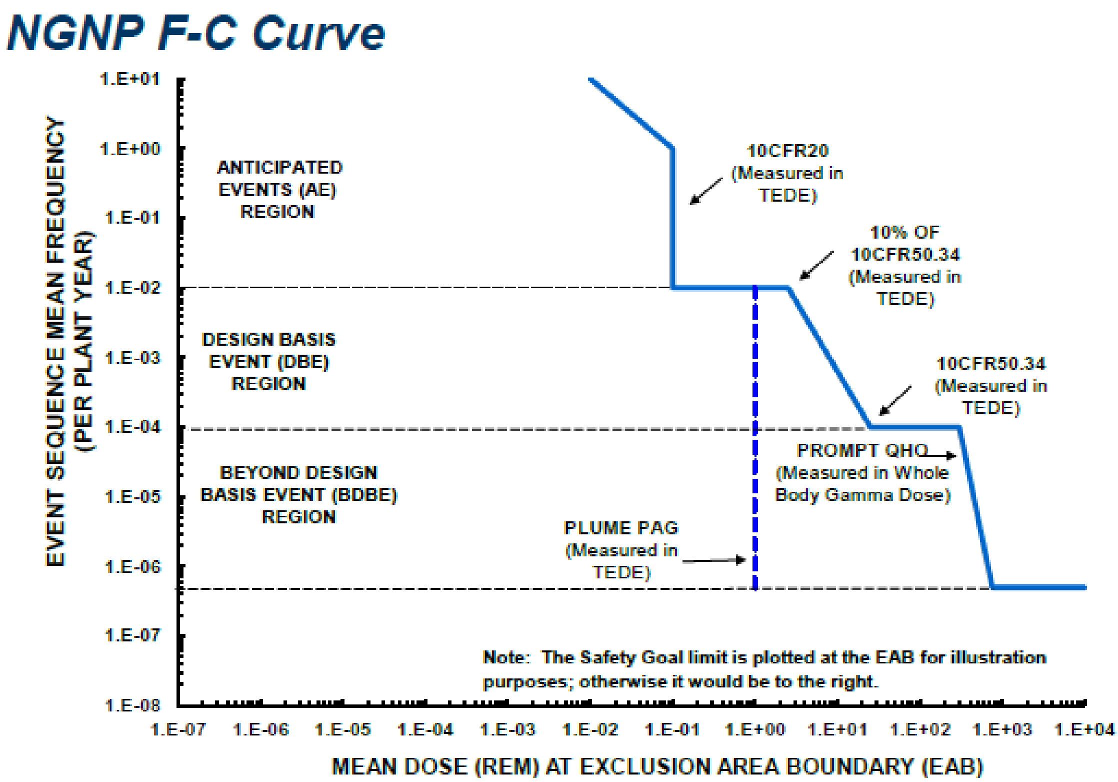

Figure 1. NGNP Frequency-Consequence curve proposed by DOE/INL (ML12262A090)

The proposed FC-curve's event category regions and their respective TLRC bases are summarized as follows:

- The TLRC for the AE region would be based on limiting the annual dose to an individual member of the pubic to 100 millirem total effective dose equivalent (TEDE) as required in 10 CFR Part 20, "Standards for Protection against Radiation." DOE/INL proposes that AE source terms should be mechanistically modeled and that dose consequences should be realistically calculated at the exclusion area boundary (EAB).

- The TLRC for the DBE region would be based on an event-based public dose criterion of 25 rem TEDE as required in 10 CFR 50.34, "Contents of Applications; Technical Information."[7] DOE/INL proposes that DBE source terms should be mechanistically modeled and that dose consequences should be conservatively calculated at the EAB.

- The TLRC for the BDBE region would be based on cumulative public dose limits derived from the quantitative health objectives (QHOs) of the NRC safety goals, whereby the QHO for prompt fatality is the most limiting. DOE/INL proposes that BDBE source terms should be mechanistically modeled and that offsite dose consequences should be realistically calculated.

[7] DOE/INL's white papers cite 10 CFR 50.34. The same requirements appear in 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report." The latter would be the appropriate citation for submittals under 10 CFR Part 52.

The TLRC are used with the F-C curve to establish limits on the frequencies of event sequences and their radiological consequences. The F-C curve and its TLRC are then used in the categorization and evaluation of LBEs and in characterizing the treatment of SSCs. The overall approach provides a logical technical basis for ensuring and demonstrating that the design meets applicable regulatory criteria for public health and safety. It further bears noting that this approach is largely consistent with the staff's approved recommendation in SECY-03-0047 regarding Issue 4 in that it places greater emphasis on the use of risk information by allowing a probabilistic approach in the identification of events to be considered in the design.

- ***DOE/INL Request**: Establish frequency ranges based on mean event sequence frequency for the LBE categories.*

The NRC staff concludes that the approach that DOE/INL has proposed for categorizing each event sequence based on its mean frequency is reasonable. As outlined in DOE/INL's LBE white paper (ML102630246), uncertainties would be considered in deriving both the mean frequency and mean consequence of an event sequence. The mean frequency would then be used to categorize an event sequence as AE, DBE, or BDBE, based on where the mean frequency falls in relation to the respective event category frequency ranges.

Additionally, the LBE white paper states that the upper and lower bounds of the 95% confidence intervals of the event frequency uncertainty distribution will be compared against the frequency boundaries of the LBE categories. If the upper or lower bounds of the confidence intervals straddle the frequency boundaries between LBE categories, the white paper states that the consequences of the event sequence will be compared against the consequence criteria for each LBE category. The staff finds this approach reasonable.

- ***DOE/INL Request**: Endorse the "per-plant-year" method for addressing risk at multi-reactor module plant sites.*

The commercial modular HTGR concept proposed for NGNP involves a plant with multiple reactor modules. The staff concludes that DOE/INL's "per-plant-year" approach for addressing integrated risk in PRA modeling is reasonable. The staff believes that an integrated risk approach is more conservative and comprehensive than the treatment of modules on an individual basis. Addressing risk metrics, such as event frequency, with an integrated approach (i.e., per plant-year or per multi-module reactor package) would enable the risk assessment to include event sequences that involve source terms from one reactor module or multiple reactor modules and ultimately would provide a more robust assessment of the overall risk profile of the plant. However, the staff believes that the Commission may need to provide policy direction for this topic.

- ***DOE/INL Request**: Agree on the frequency cutoffs for the DBE and BDBE regions.*

The NRC staff concludes that DOE/INL's proposed frequency cutoffs are reasonable for use in the proposed context of modular HTGR licensing. Although DOE/INL states that the proposed F-C curve with its associated frequency cutoffs is intended to be technology neutral, the staff

has considered the eventual use of the F-C curve only in the technology-specific context of modular HTGRs and not in the context of other reactor types, such as LWRs.

DOE/INL defines DBEs as having sequence frequencies between 1×10^{-2} and 1×10^{-4} per plant-year. The staff believes this lower frequency cutoff is reasonable as long as the PRA used in the LBE selection process assesses multiple failures from common-cause events and as long as it accounts for both operating and shutdown modes and internal and external plant hazards.

DOE/INL defines BDBEs as having sequence frequencies between 1×10^{-4} and 5×10^{-7} per plant-year. The QHO of the prompt fatality safety goal in NUREG-0880, "Safety Goals for Nuclear Power Plants: A Discussion Paper," issued February 1982, limits the increase in an individual's annual risk of accidental death to 0.1 percent of 5×10^{-4} per year or an incremental increase of 5×10^{-7} per year. Consistent with this QHO, the NRC staff concludes that the lower frequency cutoff for the BDBE region of 5×10^{-7} per plant-year is reasonable.

DOE/INL further proposes that the PRA should consider event sequences down to 1×10^{-8} per plant-year to ensure that there are no any "cliff-edge" sequences that could result in high dose consequence below the BDBE cutoff of 5×10^{-7} per plant-year. This proposal is likewise reasonable. However, the staff notes that the regulatory treatment of these lower cutoff frequencies will likely require consideration by the Commission.

- ***<u>DOE/INL Request</u>*: Endorse the overall process for performing assessments against TLRC, including issues with uncertainties and the PRA, the calculational methodologies to be employed (conservative versus best estimate), and the adequate incorporation of deterministic elements.**

The NRC staff concludes that that DOE/INL's proposed overall process for performing assessments against the TLRC, including the use of engineering judgment to address uncertainties, is a reasonable approach for identifying and analyzing LBEs in a risk-informed manner. Specific attributes of the LBE evaluation process include the use of plant-specific PRA for the identification and evaluation of LBEs with respect to an applicant's proposed F-C curve

Within DOE/INL's proposed evaluation process, the LBE frequency would be the frequency of the full event sequence, including initiating event and subsequent plant responses. LBEs with frequency uncertainty distributions that straddle two event category regions at the 95% confidence level would be analyzed using the dose acceptance criteria of each region. Dose consequences for AEs, DBEs, and BDBEs would be evaluated with explicit modeling of uncertainty in performing realistic (i.e., best estimate or 50% confidence level) analysis and conservative (i.e., 95% upper confidence level) analysis as follows:

- <u>AEs</u>: Consequences would be realistically analyzed and assessed against the requirements in 10 CFR Part 20.

- <u>DBEs</u>: Consequences would be conservatively analyzed and assessed against the requirements in 10 CFR 50.34.

- <u>BDBEs</u>: Consequences would be realistically analyzed and summed together with other LBEs to demonstrate compliance with cumulative dose criteria derived from the QHOs of the NRC's safety goals.

DOE/INL further states that design basis accidents (DBAs) would be deterministically derived from DBEs by assuming that only safety-related SSCs are available and that DBA dose consequences would be conservatively analyzed (95% upper confidence level) to show compliance with the offsite dose limits in 10 CFR 50.34.

The staff's position on Issue 6 in SECY-05-0006, which discusses the use of scenario-specific source terms for licensing decisions, is that source terms for compliance should be conservative (95% upper confidence level) values based on best-estimate calculations, which is consistent with DOE/INL's proposal for DBEs and DBAs. For AE and BDBE TLRC compliance, the staff views DOE/INL's proposed use of realistic (50% confidence level) source term calculations as meriting further consideration but notes that it would involve new regulatory interpretations likely to require consideration by the Commission.

- ***DOE/INL Request**: Endorse the proposed process and categorizations for SSC classification.*

DOE/INL's approach to risk-informed safety classification of SSCs for NGNP is generally reasonable. The approach blends the strengths of probabilistic and deterministic methods in accordance with the NRC's policy statement on PRA.[8] The proposed risk-informed safety classification categories and the criteria for SSC classification within each category are also generally reasonable. DOE/INL's proposed classification of safety-related SSCs applies a risk-informed approach while still staying within the traditional deterministic definition of safety-related SSCs in 10 CFR 50.2, "Definitions." At public meetings with the NRC staff in 2012, DOE/INL clarified its intention to not deviate from the definition in 10 CFR 50.2. Additionally, DOE/INL has indicated that the special treatments for the safety-related and non-safety-related with special treatment (NSRST) categories of SSC classification will be commensurate with those necessary for ensuring that SSCs can perform their required safety functions for LBEs, provide significant DID, and ultimately meet the TLRC for the respective LBE categories. DOE/INL proposes that capability requirements would be derived from accident mitigation considerations, whereas reliability requirements would be derived from accident prevention considerations. Special treatment measures would focus on both the capability of SSCs to mitigate DBEs and the reliability of SSCs to prevent high-consequence BDBEs.

In SECY-11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated February 18, 2011, the staff focused on the selection of SSCs for iPWRs but also included an approach that could be used to refine the SSC classification for a modular HTGR.

During public meeting discussions with DOE/INL, the NRC staff clarified the current regulatory treatment of frequently occurring events analogous to those that DOE/INL would classify as AEs. Current regulatory practice for large LWRs requires conservative analysis of AOOs with regard to their potential effect on fuel integrity and safety limits. In particular, the regulation of LWR AOOs is based in part on specified acceptable fuel design limits (SAFDLs) that cannot be exceeded by design. Therefore, by definition, LWR AOOs would not challenge TLRC dose

[8] "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," (1995, 60 FR 42622).

consequence limits. For modular HTGRs, DOE/INL has not yet developed design limits analogous to the SAFDLs used for LWRs but has acknowledged the need to do so in support of NGNP design development. The staff believes that events should be established for modular HTGRs that are analogous to AOOs for LWRs.

When specified acceptable design limits have been established for modular HTGR fuel and core designs, it will be necessary to determine whether they can be challenged by operational occurrences, anomalies, or event sequences with frequencies potentially in the AE range. The staff believes that any operational events or AEs that have a potential to challenge the such design limits for modular HTGRs should be evaluated conservatively and should credit only safety-related SCCs for mitigation.

2. Mechanistic Source Terms

History of Pertinent NRC Staff and Commission Positions

For power reactor combined licenses, 10 CFR 52.79(a)(1)(vi) requires a description and safety assessment of the site, including an evaluation of the major SSCs that “bear significantly on the acceptability of the site” under the radiological consequence evaluation factors. This assessment should assume a postulated fission product release from the core into the containment with the facility operating at the ultimate power level contemplated. The regulations at 10 CFR 100.21, “Non-Seismic Siting Criteria,” require that each applicant for a construction permit or operating license on or after January 10, 1997 (new reactors/advanced reactors), comply with 10 CFR 50.34(a)(1)(ii), which provides similar requirements.

The following site radiological consequence evaluation factors appear in 10 CFR 52.79(a)(1)(vi) and 10 CFR 50.34(a)(1)(ii)(D)(1)(2):

- An individual located at any point on the EAB for any 2-hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 25-rem TEDE.

- An individual located at any point on the outer boundary of the low population zone who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25-rem TEDE.

Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” establishes minimum requirements for the design criteria for water-cooled nuclear power plants. General Design Criterion 19, “Control Room,” states, for new reactors, that “adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) TEDE as defined in 10 CFR 50.2 for the duration of the accident.”

Footnote 6 to 10 CFR 50.34 describes the source term assumed for these postulated events as follows:

> 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.

The licensing approach for large LWRs assumes that the major accident used for siting purposes is a severe accident, generally a large break loss-of-coolant accident that results in a full core melt and large fission product release to containment. In particular, this deterministic source term is used to evaluate the release mitigation effectiveness of the engineered safeguards systems, including the containment and safety-related filtration and ventilation systems.

The NRC developed NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 15, "Transient and Accident Analysis," Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," issued March 2007, to help the staff in licensing reviews of new large LWRs. Section 15.0.3 of NUREG-0800 states that the guidance on DBA source terms in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," is acceptable for use at LWRs as it applies to the design. The alternative source term (as it is commonly known) provides guidance on modeling assumptions for fission product release, transport, and mitigation for the accidents evaluated in DBA and siting analyses.

The intent of the postulated fission product release described in 10 CFR 50.34(a)(1)(ii)(D) is to provide a bounding analysis for plant siting purposes in accordance with 10 CFR Part 100, "Reactor Site Criteria." However, the accident described in Footnote 6 in 10 CFR 50.23 is not representative of the wide spectrum of possible events that make up the planning basis of EP; therefore, it is not sufficient by itself for that purpose. In Regulatory Guide 1.183, the NRC staff states that "the NRC staff does not preclude the appropriate use of the insights of the alternative source term in establishing emergency response procedures, such as those associated with emergency dose projections, protective measures, and severe accident management guides." In addition, SECY-97-020, "Results of Evaluation of Emergency Planning for Evolutionary and Advanced Reactors," dated January 27, 1997, states, among other criteria, the following criterion for determining the generic distance for the plume exposure pathway emergency planning zone (EPZ):

> The EPZ should encompass those areas in which projected dose from design-basis accidents could exceed the EPA [U.S. Environmental Protection Agency] PAGs [protective action guidelines].

In SECY-93-092, the NRC staff recommended that source terms for modular HTGRs should be based on a bounding mechanistic analysis that meets certain performance and modeling criteria supported by research and test data. In its SRM to SECY-93-092, the Commission approved the staff's recommendation. The technical basis for, and the uses of, such source terms are the subject of DOE/INL's FQ and MST white papers and the NRC's FQ-MST assessment report.

SECY-03-0047 includes the NRC's recommendation that the Commission should take the following action:

> Retain the Commission's guidance contained in the July 30, 1993, SRM that allows the use of scenario-specific [event-specific] source terms, provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties.

This staff recommendation allows credit to be given for the unique aspects of plant design (i.e., performance based) and builds upon the following recommendation in Issue 4 of SECY-03-0047:

> This approach is consistent with prior Commission and ACRS views. However, this approach is also dependent upon understanding fuel and fission product behavior under a wide range of scenarios and on ensuring [that] fuel and plant performance is maintained over the life of the plant. This approach is also very dependent on the event selection process. For the purpose of siting and containment/confinement decisions, the staff recommends that conservative source terms for AOOs and DBEs be used. For EP purposes, a best-estimate source term would be reasonable.

In reiterating the concept of mechanistic source terms, the NRC staff stated in SECY-10-0034, "Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs," dated March 28, 2010, that "the staff will assess what will be necessary to establish the basis for a scenario-specific approach and how uncertainties should be taken into account. In addition, design and license applicants and the NRC will need to establish appropriate bounding source terms for high-temperature gas-cooled reactors (HTGRs)."

The staff noted a need to establish a technical basis for the mechanistic modeling of modular HTGR source terms in SECY-93-092 and again in SECY-03-0047. Specifically, Issue 5 of SECY-03-0047 considered the following question:

> Under what conditions, if any, should scenario-specific accident source terms be used for licensing decisions regarding containment and site suitability?

The staff's recommendation, which was approved by the Commission, was that use of scenario-specific source terms should be allowed "provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties." However, the staff cautioned that "this approach is also dependent upon understanding fuel and fission product behavior under a wide range of scenarios and on ensuring fuel and plant performance is maintained over the life of the plant."

The Commission may need to consider regulatory or policy issues in determining whether the site boundary dose acceptance criteria and associated dose calculations for use in the evaluation of site suitability and emergency planning for SMR designs should be revised or whether new requirements for SMRs should be established. Current regulatory practice employs the siting dose criteria in 10 CFR 50.34 and 10 CFR 52.79 in conjunction with deterministic DBA analyses as the key input parameters for analyzing the effectiveness of the containment, for determining site suitability, and for preparing site emergency plans.

In addition to considering appropriate accident source terms for specific advanced reactor designs, the evaluation of site suitability would include consideration of the population density; use of the site environs, including proximity to man-made hazards; and the physical characteristics of the advanced reactor site, including seismology, meteorology, geology, and hydrology.

Responses to DOE/INL Requests Concerning Mechanistic Source Term

- ***DOE/INL Request*: Endorse the proposed definition of NGNP mechanistic source terms (i.e., the quantities of radionuclides released from the reactor building to the environment during the spectrum of LBEs, including the timing, the physical and chemical forms, and the thermal energy of the release).**

DOE/INL's proposed definition of mechanistic source terms describes what is being released to the environment for assessing offsite dose consequences from accidents, normal operations, and other operational occurrences. Although DOE/INL defines source terms as releases from the reactor building, the proposed mechanistic analysis of source terms considers all barriers to release and therefore necessarily includes the calculation of releases from the primary system to the reactor building.

DOE/INL's accident source term definition is different from the traditional LWR accident source term in that it is not based on a severe core damage event. At a public meeting held in 2012, DOE/INL stated that the reference to substantial core melt in Footnote 6 of 10 CFR 50.34 does not apply to modular HTGRs. For the NGNP, releases to the reactor building would instead be based on a spectrum of limiting, mechanistically evaluated, risk informed LBEs supplemented by insights from credible bounding event sequences. Such bounding event sequences would take into account the safety behavior of the plant, and the associated fission product releases would be evaluated mechanistically.

DOE/INL's proposed definition generally aligns with the NRC staff's associated recommendation in SECY-93-092, which defined a mechanistic source terms a follows:

> The result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.

In summary, the NRC staff concludes that DOE/INL's proposed definition of NGNP mechanistic source terms aligns with the current staff position on the treatment of advanced reactor mechanistic source terms and is thus reasonable for use in DOE/INL's proposed approach to determining licensing parameters for modular HTGRs.

- ***DOE/INL Request*: Agree that NGNP source terms are event specific and determined mechanistically using models of radionuclide generation and transport that account for fuel and reactor design characteristics, passive features, and radionuclide release barriers.**

The NRC staff concludes that DOE/INL's proposed mechanistic approach to evaluating event-specific release source terms and resulting offsite dose consequences provides a reasonable basis for determining the licensing parameters for modular HTGRs. This approach is consistent with the Commission-approved staff positions on source terms in SECY-93-092 and SECY-03-0047.

- ***<u>DOE/INL Request</u>: Agree that NGNP has adequately identified the key HTGR fission product transport phenomena and has established acceptable plans for evaluating and characterizing those phenomena and associated uncertainties.***

The NRC staff's FQ-MST assessment report concludes, with caveats, that DOE/INL's ongoing and planned testing and research activities for NGNP fuel qualification and mechanistic source terms development appear to constitute a reasonable approach to establishing a technical basis for the identification and evaluation of key HTGR fission product transport phenomena and associated uncertainties. The staff expects more information on release and transport phenomena through event-specific pathways to be developed as DOE/INL's activities in these areas proceed. The discussion below on containment functional performance includes additional NRC staff comments on DOE/INL's approach to NGNP fuel qualification and mechanistic source terms development.

3. Containment Functional Performance

<u>History of Pertinent NRC Staff and Commission Positions</u>

In SECY-93-092, the staff recommended that containment designs should be evaluated against a functional performance standard instead of a prescriptive criterion, stating that functional containment designs must be adequate to meet the specified onsite and offsite radionuclide release limits for the event categories within their design envelope. The Commission approved the staff's recommendation in the SRM to SECY-93-092.

In SECY-03-0047, the staff recommended that the Commission approve the use of functional performance requirements to establish the acceptability of containment (i.e., a non-pressure-retaining building may be acceptable, provided that performance requirements can be met). If approved by the Commission, the staff would develop the functional performance requirements using guidance contained in the Commission's SRM of July 30, 1993, and the Commission's guidance on the other issues discussed in SECY-03-0047. In the resulting SRM dated June 26, 2003, the Commission stated that it did not have sufficient information to determine the best options and to make a decision on the viability of a non-pressure-retaining building. The Commission directed the staff to develop containment functional performance requirements and criteria working closely with industry experts (e.g., designers, Electric Power Research Institute, etc.) and other stakeholders regarding options in this area, taking into account such features as the core, fuel, and cooling systems design. The Commission further directed the staff to pursue the development of containment functional performance standards and then submit options and recommendations to the Commission on this policy issue.

In SECY-05-0006, the staff discussed many of the concepts developed in previous communications between the staff and Commission on the topic of functional containment performance and, as directed in the SRM to SECY-03-0047, outlined the attributes for a functional containment. The NRC staff concludes these attributes are applicable to the functional containment proposed by DOE/INL. Specifically, the functional containment should do the following:

- Protect risk-significant SSCs from internal and external events.
- Physically support risk-significant SSCs.
- Protect onsite workers from radiation.
- Remove heat to prevent risk-significant SSCs from exceeding design or safety limits.
- Provide physical protection (i.e., security) for risk-significant SSCs.
- Reduce radionuclide releases to the environs (including limiting core damage).

Additionally, consistent with options recommended in SECY-05-0006, the NRC staff would be open to evaluating containment functional performance based on a risk-informed analysis and mechanistic evaluation of selected credible licensing basis events for off-site dose analysis purposes and, with the caveats noted in SECY-05-0006, to establish credible events for emergency planning zone (EPZ) considerations.

Responses to DOE/INL Requests Concerning Containment Functional Performance

➢ ***DOE/INL Request*: Confirm that the plans being implemented under the Advanced Gas Reactor (AGR) Fuel Development and Qualification Program (hereafter referred to as the AGR Fuel Program) are generally acceptable and that they provide reasonable assurance of the capability of coated particle fuel to retain fission products in a controlled and predictable manner. Identify any additional information or testing needed to provide adequate assurance of this capability, if required.**

Among the defining features of the modular HTGR design concept is its use of inert helium gas to cool a graphitic reactor core containing billions of tristructural-isotropic (TRISO) ceramic coated fuel particles. The design concept is further defined by its predominant use of inherent and passive design features (e.g., low power density, negative temperature coefficient, slender core geometry, passively cooled reactor vessel) to keep fuel operating and accident conditions within defined limits and by a safety case that emphasizes the resulting ability to limit radionuclide releases from the fuel over a broad spectrum of off-normal events. The high-temperature radionuclide retention capability of the TRISO coated fuel particle is therefore recognized as a key element in the design and licensing of modular HTGRs.

The Commission has found the concept of functional containment generally acceptable, as indicated in the SRMs to SECY-93-092 and SECY-03-0047. However, approval of DOE/INL's proposed approach to functional containment for the modular HTGR concept, with its emphasis on passive safety features and radionuclide retention within the fuel over a broad spectrum of off-normal conditions, would necessitate that the required fuel particle performance capabilities be demonstrated with a high degree of certainty.

In its FQ-MST assessment report, the NRC staff provides detailed feedback on DOE/INL's ongoing and planned activities in the AGR Fuel Program. In summary, the staff views the proposed high-level approaches to NGNP fuel qualification and mechanistic source terms as generally reasonable. The staff observes that the fuel development and testing activities completed to date in the AGR Fuel Program have been conducted in a rigorous manner and with early results that show promise towards realizing the desired retention capabilities of the TRISO particle fuel developed for NGNP. In assessing the comprehensiveness of the AGR Fuel Program, the staff has nevertheless identified a gap in the planned scope of fuel qualification testing. As discussed in the following paragraphs, additional testing to address this

gap would be needed for providing reasonable assurance that the coated particle fuel developed for NGNP has the capability to retain fission products in a controlled and predictable manner consistent with DOE/INL's stated performance goals for NGNP operating and accident conditions.

The AGR Fuel Program proposes to derive TRISO fuel performance data solely from accelerated fuel sample irradiations in the Advanced Test Reactor (ATR), a water-cooled materials test reactor located at INL. The NRC staff concludes that the data provided by the AGR Fuel Program should be verified and supplemented by additional data from real-time fuel irradiations in an HTGR-like neutron environment. Fuel irradiated in an HTGR neutron energy spectrum breeds and fissions more plutonium than in the ATR test spectra used in the AGR Fuel Program. Plutonium fission is known to be the predominant source of certain fission products (e.g., palladium, silver) that can penetrate TRISO particle coatings and potentially degrade their retentiveness and integrity. Moreover, the planned test irradiations in the AGR Fuel Program are accelerated up to three times, thereby further reducing the potential for coating degradation from time-at-temperature effects of plutonium fission products.

The staff acknowledges that the AGR Fuel Program nevertheless includes significant ongoing and planned research efforts to investigate the poorly understood phenomenology of silver and palladium interactions with TRISO coating layers. DOE/INL has stated that these research efforts may include examinations on fuel samples irradiated in the ATR at temperatures significantly above those normally expected during irradiation in an NGNP core. The staff would consider new insights emerging from such investigations in evaluating the potential fuel performance uncertainties associated with the AGR Fuel Program's current lack of plans for real-time fuel irradiations in an HTGR neutron spectrum.

The staff believes that supplemental fuel testing is necessary to address this issue and potentially other issues concerning fuel qualification and fuel service conditions as discussed in the staff's FQ-MST assessment report. The FQ-MST assessment report comments on the potential roles of special fuel testing and surveillance programs in the NGNP prototype reactor (i.e., the first -of-kind NGNP reactor module) in verifying and supplementing the technical bases for NGNP fuel service conditions and fuel performance. The staff believes that invoking special prototype requirements and license conditions in accordance with 10 CFR 50.43(e)(2)[9] would provide a particularly effective approach to NGNP licensing in view of the identified gap in the fuel qualification testing plans of the AGR Fuel Program.

In cases where a prototype plant is used to meet testing requirements, the regulations at 10 CFR 50.43(e)(2) allow the NRC to impose additional requirements for the prototype plant to protect the public and plant staff from the possible consequences of accidents during the testing period. For NGNP prototype licensing, the NRC would use conservatively evaluated pre-prototype-test fuel performance uncertainties as a basis for determining any additional requirements on prototype design features, siting, or operating limits during the testing period. Enclosure 1 to SECY-11-0112, "Staff Assessment of Selected Small Modular Reactor Issues Identified in SECY-10-0034," dated August 12, 2011, further discusses prototype licensing approaches.

[9] Note that 10 CFR 52.79(e)(24) incorporates the requirements of 10 CFR 50.43(e).

- ***<u>DOE/INL Request</u>: Establish options on containment functional performance standards as requested by the Commission in the SRM to SECY-03-0047 and as discussed further in SECY-05-0006.***

The concept of performance-based containment acceptability for a modular HTGR has been well established by the Commission in response to SECY-93-092 and SECY-03-0047. The Commission-approved performance-based containment concept specifically does not require a pressure-retaining shielded containment structure similar to that used in current large LWR plants. In its SRM to SECY-03-0047, the Commission directed the staff to pursue the development of containment functional performance standards and to submit options and recommendations to the Commission for a future policy decision.

SECY-05-0006 is a policy issue information paper that describes the staff's work on several issues that were considered in the development of a future technology-neutral framework for reactor licensing, including the Commission-requested efforts on containment functional performance. However, as with the other issues discussed in SECY-05-0006, the staff did not submit the technology-neutral containment functional performance requirements and criteria options outlined in SECY-05-0006 for a Commission policy decision. The Commission would likely need to review the specific criteria applied to evaluate a modular HTGR functional containment concept for both a prototype plant and subsequent standard plants.

Consistent with the positions presented in SECY-05-0006, the staff agrees with the following DOE/INL description of a performance standard for a functional containment:

> The upper tier performance standard for the functional containment for the NGNP should be to ensure the integrity of the fuel particle barriers (i.e., the kernel and coatings of the TRISO-coated fuel particles) rather than to allow significant fuel particle failures and then need to rely extensively on other mechanistic barriers (e.g., the reactor coolant pressure boundary and the reactor building). This standard should be characterized by [the following]:
>
> - [Ensuring] radionuclide retention within fuel during normal operation with relatively low inventory released into the helium pressure boundary (HPB).
> - Limiting radionuclide releases to the environs to meet the onsite and offsite radionuclide dose acceptance criteria (i.e., 10 CFR 50.34 and EPA PAGs) at the EAB with margin for a wide spectrum of off-normal events.
> - Maintaining the capability to establish controlled leakage and controlled release of delayed accident source term radionuclides.

An additional set of containment functional performance standards that the staff already accepted in SECY-05-0006 is to directly or indirectly accomplish the following accident prevention and mitigation safety functions:

- Protect risk-significant SSCs from internal and external events.
- Physically support risk-significant SSCs.
- Protect onsite workers from radiation.
- Remove heat to prevent risk-significant SSCs from exceeding design or safety limits.
- Provide physical protection (i.e., security) for risk-significant SSCs.

➢ ***DOE/INL Request*: Establish a staff position to support a final determination on how LBEs will be considered for making plant siting and functional containment design decisions, taking into consideration the staff's previous position in SECY-95-299, "Issuance of the Draft of the Final Preapplication Safety Evaluation Report (PSER) for the Modular High-Temperature Gas-Cooled Reactor (MHTGR)," dated December 19, 1995, that improved fuel performance is a justification for revising siting source terms and containment design requirements. (In particular, DOE/INL asks the staff to provide an adaptation of the guidance that has generally been applied to LWRs for compliance with 10 CFR 100.21.)***

Compliance with 10 CFR 100.21 would require either interpreting part of the related footnote[10] in 10 CFR 50.34 as not directly applicable to modular HTGRs or alternatively requesting an exemption. Footnote 6 in 10 CFR 50.34 was established for large LWRs based on Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," issued March 1962, and has existed since the initial issuance of 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," on April 12, 1962 (27 FR 3509). Although 10 CFR 100.11 no longer applied to the licensing of power reactors after January 10, 1997, the NRC included the siting source term concept in similar footnotes attached to the portions of 10 CFR 50.34 and 10 CFR 52.79 that give siting dose reference values.

Current reactor siting criteria primarily encompass separate regulations for seismic and non-seismic considerations. A regulatory action taken in 1996[11] relocated source term and dose requirements from 10 CFR 100, "Reactor Site Criteria," to 10 CFR Part 50.34(a) for plant applications after 1997. Siting source term and dose criteria therefore apply to plant designs as required by 10 CFR Part 50.34. Notwithstanding the nexus that exists with atmospheric dispersion characteristic requirements in 10 CFR 100.21, "Non-Seismic Siting Criteria," an applicant for a reactor site may obtain an early site permit (ESP) under 10 CFR Parts 52.17(b) or 52.79(a). The requirements of 10 CFR Part 50.34(a) must be addressed in the plant design referenced in an application for a construction permit or combined license. This understanding of the applicability of 10 CFR Part 50.34(a) in relation to future applications for an ESP is maintained throughout these documents whenever "siting criteria" are cited.

The Statements of Consideration for 10 CFR 100.11 state that "applicants are free and indeed encouraged to demonstrate to the Commission the applicability and significance of considerations other than those set forth in the guides." Given advanced reactor designs for which core melt events are not physically credible, as purported for the modular HTGR design concept, such a demonstration may be useful to show the Commission that some event other

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[10] See Footnote 6 in 10 CFR 50.34.

[11] "Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants," 65 FR 65157; December 11, 1996.

than a “core melt” scenario would be sufficient to meet the intent of 10 CFR 52.79 in a combined license application.

Alternatively, an exemption to this aspect of the footnote(s) in 10 CFR 52.79 may be justified based on research, testing, analysis, and validation. DOE/INL has proposed the following interpretation to address the intent of the footnotes for siting source terms and Footnote 7 in 10 CFR 52.79(a)(2)(IV) for the engineered safety features of modular HTGRs:

> 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 will be based on a spectrum of limiting, mechanistically evaluated, risk-informed LBEs supplemented by insights from credible (i.e., physically plausible) bounding event sequences. Such bounding event sequences will take into account the safety behavior of the plant, and the associated fission product release will be evaluated mechanistically.

The staff agrees that this interpretation of the footnotes on the siting source term and the design of engineered safety features is reasonable for modular HTGRs.

The staff’s preapplication review of the MHTGR, as documented in NUREG-1338, followed an approach for including the evaluation of a set of staff-selected bounding events. The staff now believes that similar sets of bounding events should be further evaluated for current modular HTGR designs. In addition, the SRM to SECY-93-092 indicates a need to better explore the potential for cliff-edge effects associated with the possibility of air and moisture ingress events that could result in significant graphite oxidation in the core and support structures. In this regard, the SRM specifically directs the staff to consider “chimney-effect” air ingress events (i.e., with concurrent helium pressure boundary breaks above and below the core). Considerations for the selection of bounding event sequences for plant siting and functional containment design evaluations should be informed by “safety terrain” insights from such exploratory studies and should reflect the Commission’s PRA policy statement by blending the strengths of probabilistic and deterministic methods.

Events with moisture ingress or large breaks in the primary pressure boundary may be found to maximize the pressure-driven prompt releases from the modular HTGR functional containment system. The selection of large break sizes and locations for use in siting analyses should be informed by critical examination of the plausibility of gross vessel failure in the modular HTGR conceptual designs under consideration for NGNP. The evaluation of longer term siting releases to the reactor building and environs should be based on a plausible large break event selected to bound the potential for air ingress into the primary system and the resulting air oxidation of graphitic core and support structures. The progression and consequences of such long-duration oxidation events should be evaluated in terms of the release of activity previously bound in the affected graphitic materials and any potential to overheat fuel particles (due to the addition of exothermic oxidation energy) or expose fuel particle coatings to oxidation by air. Factors that significantly affect the long-term progression of such oxidation events may include the rate of air in-leakage into the reactor building and the ability of passive design features of the building and primary system to delay or limit oxygen transport to the core and support structures.

In summary, the staff believes that siting source term events for modular HTGRs should be deterministically selected to bound both the short-term and long-term releases of radionuclides

beyond the primary helium pressure boundary. The selected siting events should be physically plausible event sequences, and the resulting event-specific siting source terms should be mechanistically analyzed.

4. Emergency Preparedness

History of Pertinent NRC Staff and Commission Positions

Emergency Preparedness (EP) is a significant aspect of the NRC's DID approach to nuclear regulation designed to protect public health and safety and the environment.

The NRC's predecessor, the U.S. Atomic Energy Commission (AEC), required nuclear power plant licensees to address EP starting in 1958. The AEC published TID-14844 to establish a computational method for distances and exposures associated with a general class of reactors. The AEC used TID-14844 to establish zones defined in 10 CFR Part 100, which required licensees to establish an exclusion area, low population zone, and population center distance around nuclear power plants.

NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light-Water Nuclear Power Plants" (EPA 520/1-78-016), issued November 1978, introduces the conceptual basis for EPZs that could provide dose savings for a spectrum of accidents that could be associated with the PAGs described in the EPA-400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," issued 1992.

The NUREG-0396 task force, which included staff from the NRC and EPA, also established EPZ distance criteria, issued in December 1978, based on the following elements:

- The EPZ should encompass those areas in which projected dose from DBAs could exceed the EPA PAGs.

- The EPZ should encompass those areas in which consequences of less severe Class 9 (core melt) accidents could exceed the EPA PAGs.

- The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of the more severe Class 9 accidents.

The NRC incorporated these EPZ definitions into Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. Specifically, one of the 16 standards in 10 CFR 50.47(b)(11) states the following:

> Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

In the SRM to SECY-93-092, the Commission stated the following:

> The staff should remain open to suggestions to simplify the emergency planning requirements for reactors that are designed with greater safety margins. To that end, the staff should submit to the Commission recommendations for proposed technical criteria and methods to use to justify simplification of existing emergency planning requirements.

The Commission further stated that work on EP should be closely correlated with work on accident evaluation and source terms to avoid unnecessary conservatism. In addition, the work on EP for advanced reactors should be coordinated with the approach for evolutionary and passive advanced reactors.

Subsequently, SECY-97-020 presents the staff's review of NUREG-0396 rationale, criteria, and methods and the evaluation of the rationale, criteria, and methods for EP for evolutionary and passive advanced LWRs. This review and evaluation enabled the staff to recognize the following statement as one of their conclusions:

> Changes to EP requirements may be warranted if the technical criteria for EP requirements were modified to account for the lower probability of severe accidents or the longer time period between accident initiation and release of radioactive material for most severe accidents associated with evolutionary and passive advanced LWRs.

SECY-11-0152, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," issued October 28, 2011, includes the following consideration for establishing the size of EPZs for SMRs:

> The staff considers it appropriate to be open to applicant requests for establishing SMR technology-neutral, variable distance, plume exposure EPZs. However, the staff recognizes that the burden would be on the applicant to provide a well-justified basis for this section.

The NRC staff recognizes that new and advanced reactors may incorporate enhanced margins of safety or may use simplified, inherent, passive, or other innovative means to accomplish safety and security functions and to thereby address the expectations stated in the 1986 Commission policy statement on the regulation of advanced reactors (updated in 2008). To the extent that such safety and security improvements can be demonstrated for advanced designs, potentially including modular HTGRs, the NRC staff is open to considering alternative treatment of EP for advanced reactors.

<u>Responses to DOE/INL Requests Concerning Emergency Preparedness and Planning</u>

- ***<u>DOE/INL Request</u>: Propose a new policy or revised regulations on how EPZ sizing can be scaled to be commensurate with the accident source term, fission product release, and associated dose characteristics.***

The key issues in this request are as follows:

- the use of the DOE/INL-proposed RIPB approach to calculate the frequency of exceeding the PAG values as a function of distance from the plant for a spectrum of accidents

- the establishment of criteria for determining that the point at which the frequency of exceeding the PAG values is acceptably low

In SECY-11-0152, the staff indicated a willingness to consider alternative EP requirements and frameworks for SMR facilities. SECY-11-0152 describes a PAG-based dose-distance scalable approach that could be considered for determining EPZs on a case-by-case basis for modular HTGRs.

The staff recognizes that design-specific policy issues may be associated with the approach suggested by DOE/INL for proposing a combined low population zone and EAB (or a scaled or reduced EPZ) partly based on event-specific release source terms calculated mechanistically for a spectrum of LBEs. For instance, one of the modular HTGR design goals is to not have any identified credible LBEs that result in severe core damage and associated large offsite radiological releases. Although the NRC staff may consider these issues in future prelicensing or licensing interactions, issue resolution will likely require the Commission's approval. The staff does not plan to propose additional new EP policy or to revise guidance for specific changes to EP requirements at this time.

- ***<u>DOE/INL Request</u>: Establish guidance on how the specific emergency planning requirements in 10 CFR Part 50 can be applied with a graded approach (when compared to current emergency plans for LWRs) that allows for the development of onsite and offsite emergency plans commensurate with the NGNP design and a plume exposure EPZ at a distance from the plant (e.g., approximately 400 meters from the reactor centerline) to demonstrate that it meets the PAG values.***

The NRC staff states in SECY-11-0152 that it considers it appropriate to be open to applicant requests for establishing technology-neutral, variable distance, plume exposure EPZs for SMRs. SECY-11-0152 describes a dose-distance scalable approach that could be emulated for determining SMR EPZs. In addition, SECY-10-0034 states that HTGR facilities belong to a technology group of SMRs that may be likely to submit a license application to the NRC.

The staff does not plan to provide additional guidance for specific changes to EP requirements in the absence of specific proposals from the NGNP applicant or nuclear industry. The staff expects that the license application would provide sufficient design information for the review of the proposed NGNP EP framework approach. The NRC does not expect that changes to regulations will be necessary to adopt a graded approach to EP requirements. However, any proposed changes to established EP policy and guidance will require the Commission's consideration.

- ***<u>DOE/INL Request</u>: Propose guidance on how issues related to the modularity of the designs and the collocation of multiple-module plants near industrial facilities should be considered in emergency planning.***

The DOE/INL expects to collocate the NGNP with industrial facilities. The plant would provide energy in the form of electricity and process heat to the collocated industrial facilities. Examples include petrochemical, oil refinery, chemical processing, coal liquefaction, hydrocarbon extraction, and hydrogen production industrial facilities.

EP issues related to licensing nuclear plants that are collocated with industrial facilities could be similar to those currently evaluated for the LWRs that are near industrial facilities. However, a policy issue requiring the Commission's consideration would be necessary if the intended usage differs significantly from existing practices, such as the Waterford 3 Steam Electric Station (Waterford) located near an industrial park in Killona, LA. The proximity of the industrial park requires Waterford to address NRC regulations related to the impact of potential industrial hazards, such as industrial chemical releases. Response plans incorporate this type of assessment to ensure the protection of nuclear plant safety systems, plant personnel, and the public.

License applications must consider the following issues that involve, among others, the use of nuclear process heat by collocated industrial facilities:[12]

- safety implications and equipment protection associated with shared industrial facility SSCs
- standoff considerations of potential explosions and missiles or fires at the collocated industrial facilities
- external events, such as aircraft impact, flooding, and seismic events, that affect the collocated industrial facilities
- the effect of chemicals, gases, and radioactive hazards from industrial facilities
- response coordination with the collocated industrial facility and with State, Federal, and county agencies and resolution of jurisdictional issues
- radioactive material monitoring and plant security at the collocated industrial facilities

SUMMARY AND CONCLUSIONS

As required by the EPAct, DOE/INL and the NRC staff have been engaged in a series of prelicensing interactions on the NGNP project since 2007. Prelicensing activities conducted since late 2009 have included the NRC staff's review of a series of DOE/INL white paper submittals that describe elements of DOE/INL's proposed approach for implementing the NGNP licensing strategy that DOE and NRC jointly developed and reported to Congress in 2008. Since February 2012, the latest set of interactions has focused on further resolving issues in four key areas for licensing the NGNP prototype. As clarified by DOE/INL in its letter of July 6, 2012, these four key issue areas are:

- Licensing basis event selection
- Source terms
- Containment functional performance
- Emergency preparedness.

[12] See the section entitled, "Industrial Facilities Using Nuclear-Generated Process Heat," in SECY-11-0112.

The NRC staff had previously provided DOE/INL with incremental feedback on its proposed NGNP licensing approach through public meeting interactions and public correspondence, including two preliminary NRC assessment reports on five related white paper topics. This summary feedback document and the more detailed feedback in the NRC staff's updated white paper assessment reports conclude that the proposed risk-informed framework and performance-based criteria for licensing the NGNP prototype present a generally reasonable approach for implementing the framework outlined in the joint NGNP Licensing Strategy Report of 2008, which includes the following major conclusion:

> The best approach to establish the licensing and safety basis for the NGNP will be to develop a risk-informed and performance-based technical approach that adapts existing NRC LWR technical licensing requirements in establishing NGNP design-specific technical licensing requirements. This approach uses deterministic engineering judgment and analysis, complemented by probabilistic risk assessment (PRA) information and insights, to establish the NGNP licensing basis and requirements. As discussed in this report, the selected approach provides significant advantages in meeting the schedule for licensing an NGNP while providing consistency with Commission policy guidance on the use of probabilistic risk information and insights.

Accordingly, the focus of the NRC staff's review has been primarily on (1) the general approach for applying the RIPB criteria to NGNP licensing and (2) a determination of how such an approach could be adaptable to current licensing requirements. A future application for NGNP licensing should specify the details of these adaptations, some of which may entail specific regulatory exemptions and policy issues for Commission consideration.

The joint Licensing Strategy Report includes the use of the prototype testing provisions under 10 CFR 50.43(e). The prelicensing activities that the NRC conducted with DOE/INL since 2008 have reinforced and refined the staff's early views on the regulatory necessity and technical importance of testing and surveillance for the NGNP prototype. Special design features, siting considerations, and operating limits may be necessary for the NGNP prototype that, subject to the successful completion of the required programs for testing and surveillance, may not be necessary for a standard NGNP design.

The NRC staff will generally determine what information the NGNP applicant must provide as part of the license application. The applicant will be responsible for providing any additional research data needed to support the NGNP safety case. The NRC will use the agency's resources if it believes that the research is important to independently assess the applicant's submittals or to provide the technical bases needed to develop the regulatory requirements.

As noted in the preceding sections and as further discussed in the staff's revised assessment reports, the NRC staff believes that DOE/INL's proposed approaches to the respective key issues are generally reasonable and are responsive to the Commission's Policy Statement on advanced reactors. The NRC staff has further identified technical and regulatory issues, such as EP, that could have policy implications that would require future direction from the Commission. Although the staff will consider alternative EP requirements and frameworks for advanced reactors and SMR facilities, it does not plan to propose additional new EP policies or to revise the existing guidance for addressing EP requirements at this time.

Lastly, the NRC staff notes that further insights gained from future NGNP licensing efforts should benefit ongoing efforts to further risk-inform the existing reactor licensing framework and related longer term efforts to develop a new technology-neutral framework for reactor licensing.