OFFICE OF NEW REACTORS

ASSESSMENT OF WHITE PAPER SUBMITTALS ON DEFENSE-IN-DEPTH; LICENSING-BASIS EVENT SELECTION, AND SAFETY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS (REVISION 1)

NEXT GENERATION NUCLEAR PLANT PROJECT 0748

1. INTRODUCTION

The U.S. Department of Energy (DOE) and 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 DOE/INL 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 generating electric power at the same time. To fulfill this mission, DOE/INL is considering a modular HTGR with either a prismatic block or pebble bed core and safety features described by DOE/INL as follows:[1]

> To achieve the safety objectives for the NGNP Project, the HTGR relies on inherent and passive safety features. Modular HTGRs use the inherent high temperature characteristics of TRISO-coated fuel particles, graphite moderator, and helium coolant, along with passive heat removal capability of a low-power-density core with a relatively large height-to-diameter ratio within an uninsulated steel reactor vessel to assure sufficient core residual heat removal under loss-of-forced cooling or loss-of-coolant-pressure conditions.
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> The primary radionuclide retention barrier in the HTGR consists of the three ceramic coating layers surrounding the fissionable kernel to form a fuel particle. As shown in Figure 4, these coating layers include the inner pyrocarbon (IPyC), silicon carbide (SiC), and outer pyrocarbon (OPyC), which together with the buffer layer constitute the TRISO coating. The coating system constitutes a miniature pressure vessel that has been engineered to provide containment of the radionuclides and gases generated by fission of the nuclear material in the kernel. Thousands of these TRISO-coated particles are bonded in a carbonaceous material into either a cylindrical fuel compact for the prismatic HTGR or a spherical fuel element for the pebble bed HTGR. These fuel particles can withstand extremely high temperature without losing their ability to retain radionuclides under all accident conditions. Fuel temperatures can remain at 1,600 ºC for several hundred hours without loss of particle coating integrity (INL 2010a). This high temperature radionuclide retention capability is the key element in the design and licensing of HTGRs.

[1] INL/EXT-11-22708, "Modular HTGR Safety Basis and Approach," NGNP information paper submitted September 6, 2011, Project 0748, ML11251A169, excerpt page 8.

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 an NGNP prototype. Such early interactions are encouraged by the NRC's policy statement on advanced reactors, which states, in part, the following:[2]

> During the initial phase of advanced reactor development, the Commission particularly encourages design innovations that enhance safety, reliability, and security…and that generally depend on technology that is either proven or can be demonstrated by a straightforward technology development program. In the absence of a significant history of operating experience on an advanced concept reactor, plans for the innovative use of proven technology and/or new technology development programs should be presented to the NRC for review as early as possible, so that the NRC can assess how the proposed program might influence regulatory requirements.

In accordance with the provisions of the EPAct, DOE and the NRC prepared a report (hereafter referred to as the "Licensing Strategy Report") that describes the NGNP licensing strategy and submitted it to Congress in August 2008 (ADAMS Accession No. ML082290017[3]). The 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 probabilistic risk assessment (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.

The Licensing Strategy Report describes this approach as the "preferred option" to complete licensing within the timeframe identified by the EPAct. DOE and the NRC considered Option 2 to be the most viable option and expected it to limit regulatory and licensing uncertainty. The agencies considered the other options to be less viable than Option 2 and believed that they would increase this uncertainty.

DOE/INL has prepared a series of white papers on aspects of the HTGR design and safety basis to obtain NRC feedback on design, safety, technical, and licensing process issues that could affect NGNP deployment. Three of these white papers describe the approach that DOE/INL intends to use to implement the Option 2 risk-informed and performance-based approach. These three papers are as follows:

(1) INL/EXT-10-19521, "Next Generation Nuclear Plant Licensing Basis Event Selection White Paper," dated September 16, 2010 (ML102630246, hereafter referred to as the licensing basis event (LBE) white paper)

(2) INL/EXT-09-17139, "Next Generation Nuclear Plant Defense-in-Depth Approach," dated December 6, 2009 (ML093480191, hereafter referred to as the defense-in-depth (DID) white paper)

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

[3] Note that subsequent references to ADAMS herein will omit the phrase "ADAMS Accession No." for brevity.

(3) INL/EXT-10-19509, “Next Generation Nuclear Plant Structures, Systems, and Components Safety Classification White Paper,” dated September 21, 2010 (ML102660144, hereafter referred to as the structures, systems, and components (SSC) white paper)

These papers discuss a series of outcome objectives that describe specific areas for which DOE/INL is seeking NRC feedback and agreement on the proposed approach that the eventual NGNP license applicant will apply. Outcome objectives for the LBE, DID, and SSC white papers appear in Appendices A, B, and C, respectively.

This assessment addresses each of the papers’ outcome objectives along with any other issues associated with these topics that the staff believes may be relevant to licensing the NGNP. The topics of the three white papers are closely interrelated, and they are integral parts of the proposed risk-informed, performance-based approach. Therefore, this assessment addresses all three topics together.

This assessment does not provide a final regulatory conclusion on any aspect of the NGNP licensing approach or design. Completion of the NGNP design in accordance with the principles proposed by the white papers will not be sufficient justification for design approval or certification of a standard design. A safety evaluation of a future combined license (COL), design approval, or design certification submittal will provide conclusions on design approval or design certification, upon the staff’s determination that the proposed design meets all current NRC regulations and that it is consistent with NRC guidance for the review of such applications and with relevant Commission policy.

Similarly, the staff’s feedback on these papers is preliminary because many issues identified by the staff cannot be addressed or resolved until more information about the NGNP design is available. However, the staff believes that identifying these issues is valuable for prospective designers because it allows them to incorporate relevant insights into their design efforts.

2. ASSESSMENT PROCESS

To develop the requested NRC feedback, an assessment process was conducted and documented in two phases, as described in the two subsections below. Submittals, correspondence, meeting materials, and meeting summaries pertinent to the assessment process and other NGNP prelicensing activities are available in ADAMS under Docket No. PROJ0748.

DOE/INL did not submit revisions to the white papers during the assessment process; however, it did indicate that any future NGNP prelicensing or licensing submittals related to the topics in the white papers would incorporate revisions and clarifications based on consideration of NRC assessment comments.

2.1 Initial Assessment Phase

For the initial phase of the assessment process, the NRC assembled an assessment working group comprising several personnel from the NRC Office of New Reactors and Office of Nuclear Regulatory Research. Routine biweekly conference calls between NRC and DOE/INL facilitated continuing coordination of all interactions related to NGNP, including those for the assessment of the subject white papers.

The DID white paper submittal was received several months before the closely related LBE and SSC submittals. Based on its initial examination of the DID paper, the NRC working group issued a limited set of requests for additional information (RAIs) on July 26, 2010 (ML102020580). DOE/INL responded to this letter on September 30, 2010 (ML102770386). The staff issued a more extensive set of RAIs addressing all three white papers on August 3, 2011 (ML112140336). The staff received responses to this second set of RAIs on October 14, 2011 (ML11290A188).

Additionally, in September 2011, DOE/INL submitted a white paper titled, "Next Generation Nuclear Plant Probabilistic Risk Assessment White Paper" (ML11265A082, hereafter called the PRA white paper). The working group members reviewed this white paper during the final months of the initial assessment phase. However, because DOE/INL submitted the white paper relatively late in the assessment process, the group agreed that the NRC would not issue RAIs or formally assess the paper in terms of its stated outcome objectives; instead, it would discuss the PRA white paper in subsequent public meetings with DOE/INL and would consider its content in completing the follow-on assessment phase.

This assessment phase culminated with the issuance of an initial version of this NRC assessment report (ML120240671 and ML120170084) and an associated NRC letter to DOE on February 15, 2012 (ML120240682). The letter to DOE includes a brief discussion under DOE's request for continued preapplication interactions on the DID approach, LBE selection, SSC safety classification, and other topics as they relate to four key licensing issues highlighted in the NGNP Licensing Strategy Report to Congress.

2.2 Follow-on Assessment Phase

The follow-on assessment phase was conducted through a series of public working meetings and conference calls with DOE/INL and included reviews of additional DOE/INL submittals that addressed comments and issues discussed in the initial version of this NRC assessment report. As in the initial assessment phase, continuing assessment interactions were facilitated and coordinated through routine conference calls between NRC and DOE/INL. These routine calls were generally conducted on a biweekly basis, shifting to a weekly basis during the closing weeks of the process.

Further discussions on the PRA white paper occurred during the follow-on phase; these discussions provided the staff with background information for use in further assessing the DID, LBE, and SSC white papers. In a letter dated July 6, 2012 (ML121910310), DOE/INL clarified its overall objectives with respect to the following four key issues acknowledged in the NRC's letter dated February 15, 2012, to DOE:

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

DOE/INL's letter dated July 6, 2012, thus provided a useful framework for coordinating and integrating the continuing assessment interactions for the DID, LBE, and SSC white papers with those for the NGNP white papers on fuel qualification and mechanistic source terms.

The staff noted during the assessment process that the word "acceptable," as used in the DOE/INL-stated outcome objectives, carries regulatory/legal connotations that would not be appropriate for the white paper assessments. Therefore, in completing the assessments, the NRC has instead assessed the proposed approaches in terms of whether they are reasonable, thereby effectively replacing "acceptable" with "reasonable" in DOE/INL's feedback requests.

Appendix D of the report lists the NRC staff who participated in the initial and follow-on assessment phases for the DID, LBE, and SSC white papers. Participants in the follow-on phase included additional staff from appropriate NRC program offices. This updated white paper assessment report provides the NRC staff's views that revise, clarify, and supplement the agency's working group views presented in the initial report.

3. ASSESSMENT RESULTS

The discussion below provides the staff assessment of each outcome objective identified by the three white papers. In addition, the staff provides feedback on implementation of the strategy described in the NGNP Licensing Strategy Report along with feedback on other issues that it believes may be useful as the future NGNP reactor design is developed. Certain issues discussed below are Commission policy issues. In this context, a Commission policy issue is an issue for which the staff presently believes the Commission would have to make a specific policy determination.[4]

3.1 Licensing Basis Event Selection

The LBE white paper proposes to use a combination of deterministic and probabilistic methods to establish the NGNP licensing basis events. In general terms, Section 3.1 of the LBE white paper describes the proposed approach as follows:

- A deterministic approach is used to select an initial event set providing a starting point for a given phase of the design process. For example, a set of initial events developed from conceptual design provides the starting point for preliminary design.

- The LBEs are updated as the design and analysis progress. The PRA is developed and revised as the design matures. This begins to risk inform the LBE event sequences with insights gained from the PRAs conducted during the design phase as the design continues to develop.

- A review of the LBEs is performed at the end of the design phase to evaluate conservatisms in the selected events.

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

The paper outlines a process in which PRA models are improved as the scope and detail of design information increase. Those models are used in an iterative fashion to identify potential LBEs and possible measures to mitigate those events.

Designers are expected to examine the results and to determine whether design changes are desirable. An updated model would reflect any changes, and the process would be repeated until the designer is satisfied that adequate performance has been achieved.

3.1.1 LBE Outcome Objective 1 – Structured Process for Licensing Basis Event Selection

The first LBE outcome objective states that "the structured process for selecting LBEs is an acceptable approach for defining the LBEs."

The LBE white paper defines LBEs "as the events derived from the HTGR technology and plant design that are considered by the licensing process and are used to derive design-specific performance requirements for SSCs." The paper also states that "a combination of deterministic and probabilistic analysis is used to identify these events and evaluate the event sequences. The LBE selection process will identify event sequence families based on an identified set of initiating events and will establish the frequency of each of these event sequences." The intent is to establish a risk-informed and performance-based process.

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In addition to deterministic selection of the kinds of initiating events, the LBE white paper proposes to use PRA to establish the envelope of event sequences that the staff must consider for licensing the NGNP. This is a new application of a plant-specific PRA in that the licensing basis events will come from PRA event sequences. For current operating reactors and new reactors, deterministic judgment has been used to establish most events in the licensing basis, rather than event sequences selected from the plant PRA. As such, requirements and guidance for the technical adequacy of the plant PRA must be established. These requirements will be different and will likely be more demanding than those for a "design PRA" that supports certification in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Design PRAs have been developed with a level of quality commensurate with their application, which to date has ranged from identifying enhancements to the design to risk-informing technical specifications and surveillance requirements. The increased emphasis on risk for the NGNP approach would require a corresponding enhancement in the level of PRA quality.

The NGNP Licensing Strategy Report states that the technical approach for establishing the NGNP licensing basis and requirements should involve the "selection of licensing-basis events using deterministic engineering judgment complemented by insights from the NGNP PRA." The Licensing Strategy Report further states that "once the NGNP technology is demonstrated through successful operation and testing of the NGNP prototype, and a quality PRA including data becomes available, greater emphasis on design-specific PRA to establish the licensing basis and requirements will be a more viable option for licensing a commercial version of the NGNP reactor."

The approach in the Licensing Strategy Report for use of the plant PRA follows the staff requirements memorandum (SRM) for 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 (ML040210725), on accident selection and evaluation. The Commission approved the staff's recommendations (1) to select events and sequences deterministically, (2) to ensure that they use conservative assumptions, and (3) to supplement them with insights from the PRA for the specific design.

An important aspect of the staff's assessment of the proposed NGNP LBE selection process focused on determining whether the proposed approach for applying deterministic engineering judgment to complement the insights from the NGNP PRA to select NGNP LBEs was consistent with the approach for applying conservative engineering judgment.

The LBE white paper states that "a combination of deterministic and probabilistic analysis is used to identify initiating events and evaluate the event sequences. The LBE selection process will identify event sequence families based on an identified set of initiating events and will establish the frequency of each of these event sequences." This process is intended to be risk-informed and performance-based. The PRA white paper defines a deterministic process as an approach that evaluates predetermined fixed scenarios based on physical principles and states. A deterministic process is prescriptive (in that elements of it may be imposed) and may incorporate bounding assumptions, criteria, or regulations imposed to compensate for related uncertainties. The PRA white paper states that a probabilistic element is associated with an evaluation that explicitly accounts for the likelihood and consequences of possible accident sequences in an integrated fashion.

The staff agrees that it is appropriate to identify initiating events deterministically. In this regard, DOE/INL stated in response to RAI LBE-3 that it used engineering judgment to determine that design-basis accidents (DBAs) would include pressurized and depressurized loss of forced cooling events (conduction cooldowns). In addition, DOE/INL will use deterministic judgment in LBE selection to incorporate lessons learned in HTGR design and operations and in consideration of additional event challenges that have been postulated during previous HTGR licensing efforts. However, assessing the adequacy of the implementation of the approach is not possible until DOE/INL provides considerably more detail on the proposed NGNP design, processes used to conduct the deterministic evaluation, and the outcome of the design effort by the reactor vendor.

Section 3.6.1 of the PRA white paper also describes the approach that will be used to identify potential initiating events. The paper describes this approach as a structured, step-by-step process similar to that used in light-water reactor (LWR) PRAs to define SSC failure modes and to identify initiating events, including challenges posed during all operating and shutdown modes of operation applicable to NGNP, as well as those posed by internal and external hazards. The process focuses on identifying events that are specific to the NGNP design. For completeness, it will also include review of experience with LWRs and other gas-cooled reactors. This general approach is consistent with the approaches used for new LWR designs that the staff has been reviewing and is reasonable. The staff will review the implementation of this approach for specific HTGR designs as part of its review of an application for a license or design certification.

The LBE and SSC white papers, respectively, state that DOE/INL will use deterministic judgment to address uncertainties as follows:

- In categorizing an event as an anticipated event (AE),[5] design-basis event (DBE), or beyond-design-basis event (BDBE), DOE/INL will use deterministic judgment to address uncertainties in the PRA event sequence frequency (e.g., uncertainties in the reliability of SSCs in the event sequence).

- In evaluating predicted consequences against the regulatory dose acceptance criteria, DOE/INL will use deterministic judgment to address uncertainties in the performance of SSCs (i.e., uncertainties in the capability of the SSCs).

The LBE white paper further states that, in distinguishing a DBA sequence from a DBE sequence, DOE/INL will use deterministic judgment in choosing the SSCs from the list of SSCs that can perform a required safety function.

The LBE white paper describes an approach that places significant emphasis on the NGNP design-specific PRA and that uses some degree of deterministic judgment for selecting the NGNP LBEs. However, the LBE white paper does not clearly describe how DOE/INL will use deterministic engineering judgment to select LBEs other than to (1) identify the kinds of initiating events and the equipment failures that the PRA will include, (2) bound the uncertainty in the LBE sequence frequencies for purposes of categorizing events, (3) bound the uncertainty in LBE-predicted dose consequence, and (4) select safety-related (SR) equipment for distinguishing the DBAs from the DBEs.

For the first-of-a-kind NGNP, the LBE white paper does not clearly describe how DOE/INL will use engineering judgment to deterministically select LBEs in the event categories that are conservative with respect to the calculated dose consequences relative to the LBEs developed from the NGNP PRA. For example, such LBE sequences would involve conservative assumptions that would potentially involve a combination of aspects, such as conservative initiating events (i.e., initiating event severity) with respect to the resulting mechanistic source term, conservative SSC performance characteristics associated with fission product barriers and accident heat removal, and conservative core thermal fluid characteristics resulting in conservative fission product releases and distributions during normal operation (e.g., conservative bypass flow during normal operation). The intent of using deterministic engineering judgment in these ways is to address, in part, the "unknown unknowns" associated with the safety performance of the NGNP (i.e., completeness of the NGNP PRA) and to address the intent of the risk-informed approach, including the selection of DBAs, that DOE and the NRC selected in the NGNP Licensing Strategy Report.

As stated above, the LBE white paper indicates that NGNP will use a deterministic event selection process that may impose prescriptive elements and may incorporate bounding assumptions, criteria, or imposed aspects to compensate for uncertainties. The staff believes that this is an important aspect of event selection that is not adequately described in the LBE white paper.

[5] During the follow-on assessment phase, DOE/INL decided to change the name of this event category to AE in response to NRC feedback provided at public meetings. This name replaces the event category name "anticipated operational occurrences" (AOOs) that the assessed NGNP white papers used for LBEs, DID, SSCs, and PRAs. The staff understands that DOE/INL intends to use the event category name AE in place of AOO in all future submittals for the NGNP. The staff acknowledges this understanding by using AE as the event category name in place of AOO throughout this report.

The staff believes that DOE/INL should use deterministic engineering judgment to select additional events that credibly bound the source terms for the event families identified and should use the proposed approach described in the LBE white paper for the selection of such events. The additional deterministically selected events would conservatively and credibly envelop the transport and eventual release of fission products across the NGNP containment barrier system (i.e., fuel system, helium pressure boundary (HPB), and reactor building) for each of the event families. Selecting additional event sequences that involve a loss of HPB integrity is particularly important because they result in a pathway for the transport of fission products from the HPB into the reactor building and, potentially, to the environment. Such events could also potentially enhance the release of fission products from the core due to chemical attack arising from the effects of air or moisture ingress along with increased transport of fission products from the core and HPB through natural circulation. Other examples of deterministically selected events used to conservatively calculate the siting source term include bounding the degraded performance of a passive safety system, such as the fuel barrier or decay heat removal system, which would then be used to inform the emergency planning zone (EPZ) requirements.

The NGNP license applicant and the NRC will need to agree on a set of deterministically selected LBEs in the event categories consistent with the NGNP design and safety characteristics of the design. The staff recognizes the importance of ensuring that any additional measures that might be considered or needed to mitigate such deterministically selected events do not inadvertently result in increased plant risk. The application of deterministic judgment in the selection of LBEs should account for data and modeling uncertainties associated with the proposed approach and should address "unknown unknowns," or the lack of adequate or directly applicable data. For example, a cross-vessel break may be an event that is deterministically evaluated that could become a bounding event sequence for the siting source term and could be used in the bounding analysis for potential scaled EPZ considerations.

The staff did not conduct an extensive review of the PRA white paper, but did examine that paper to gain insight regarding how DOE/INL expects to apply PRA to the topics addressed in this assessment. The PRA white paper summarizes how the PRA would include external events (e.g., fires, earthquakes, floods, high winds, transportation accidents, and nearby industrial facility events) as the causes of initiating events. The response to RAI LBE-4 states that the PRA white paper describes how DOE/INL plans to apply a draft American Society of Mechanical Engineers (ASME)/American Nuclear Society standard titled, "Technology Neutral Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants," issued July 2011, and supporting LWR standards, draft standards, and regulatory guides that were used to develop the advanced non-LWR PRA standard to the NGNP design. DOE/INL states that it will select the event sequences resulting from external hazards as DBEs, BDBEs, and the deterministically selected DBAs (described in Section 3.1.7). The analysis of seismic events will be done in the context of a seismic PRA.

The PRA white paper states that events and conditions that exceed the design-basis envelope for passive SSCs (e.g., extreme external hazards) are expected to dominate failures in safety functions because of the reduction in the reliance on active SSCs to perform their safety functions. DOE/INL states that the justification for screening out any external hazards will be made in accordance with the requirements in the PRA standards.

The LBE white paper states, in part, that BDBEs ensure that adequate emergency planning is in place to address these highly improbable events. BDBEs are selected from those families of events whose mean frequency falls within the BDBE region, which is identified as the emergency planning basis event region.

Emergency response planning for large LWRs is based on a deterministic 10-mile EPZ that is informed by the accident described in Footnote 6 to 10 CFR 50.34, “Contents of Applications; Technical Information,” and in Footnote 7 to 10 CFR 52.70. This postulated accident is evaluated in Chapter 15, “Accident Analysis,” of the licensing application for LWR plants licensed before 1997 under 10 CFR 100.11, “Determination of Exclusion Area, Low Population Zone, and Population Center Distance,” and for those licensed after 1997 under 10 CFR 100.21, “Non-Seismic Site Criteria.” This DBA is used to evaluate the engineered safety features of a plant to ensure that dose consequences (e.g., magnitude, timing, and chemical form of releases from containment) do not exceed the values specified in Footnote 2 to 10 CFR 100.11 for plants licensed before 1997 or Footnote 7 to 10 CFR 50.34 for plants licensed after 1997 at the site exclusion area boundary (EAB) and the outer boundary of the low population zone. This approach to selecting emergency planning basis events (rather than other approaches, such as using risk significance, event probability, or cost/benefit) has been used because the consequences of a spectrum of events bounded by the postulated fission product release discussed in 10 CFR 50.34(a)(1)(ii)(D) provide the best means of identifying adequate planning standards and establish conservative bounds for emergency planning.

The submittal of a COL or design certification application to the NRC for the NGNP will require the staff to assess whether emergency planning basis events identified on the basis of event sequence probability would provide an adequate spectrum of events for emergency response planning for the NGNP. In this regard, additional deterministically selected event sequences in the BDBE region may be necessary to provide an adequate spectrum of accident dose consequences. A bounding event sequence for siting and emergency planning purposes may be deterministically selected and evaluated mechanistically using conservative upper bound evaluation criteria, as specified 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. Additionally, lessons learned from the effects of the earthquake and tsunami at Fukushima Dai-ichi may result in additional regulatory requirements related to selection of beyond design basis external events for the licensing of advanced reactors, including the NGNP prototype. It is likely that Commission consideration will be necessary to allow increased emphasis on the use of the plant PRA and event sequence probabilities (including screening out selected external hazards) as the basis for establishing emergency planning requirements.

SUMMARY OF CONCLUSIONS

The staff believes that the DOE/INL proposed approach for selecting the applicable LBEs is reasonable. This performance-based approach uses a combination of deterministic and probabilistic evaluation criteria for LBE selection throughout the design phase and into the licensing phase. However, the staff concludes that the following licensing issues may require further consideration:

- DOE/INL needs to identify the deterministic elements of the proposed approach that meet the criteria outlined in the 2008 Licensing Strategy Report to ensure conservative selection of bounding events that would become the DBAs and postulated accidents used for the siting source term and to justify proposed emergency response measures. The selection of DBAs and postulated accidents for the NGNP will likely require Commission consideration.

- There is currently insufficient design detail available to allow the staff to fully interpret or understand how an applicant will select the events that lead to DBAs (i.e., how it will apply engineering judgment regarding initiating event severity, determine conservative SSC performance, calculate bounding reactor thermal-hydraulic characteristics, and identify and address uncertainties). The staff will review these implementation process details during subsequent licensing activities with an applicant.

3.1.2 LBE Outcome Objective 2—Comprehensive Spectrum of Events

The second LBE outcome objective states that "LBEs cover a comprehensive spectrum of events from normal operation to rare, off-normal events."

The LBE white paper describes the following three proposed LBE categories:

(1) Anticipated events (AEs). AEs are proposed to encompass planned and anticipated events. The doses from AEs are required to meet normal operation public dose requirements. AEs are proposed to be utilized to set operating limits for normal operation modes and states. AEs are distinct from LWR AOOs. An AOO is a deterministic initiating event assumed to occur within the lifetime of the plant (i.e., a loss of all offsite power (LOOP)) with only safety-related equipment responding, whereas an AE represents the entire plant response to an event (i.e., a LOOP and the response of all available SR and non-safety-related (NSR) SSCs).

(2) Design Basis Events (DBEs). DBEs are proposed to encompass unplanned, off-normal events that are not expected to occur in the lifetime of the plant but that might occur in the lifetime of a fleet of plants. Like AEs, DBEs are event sequences including the entire plant response. The consequences from DBEs are required to meet the offsite dose limits in 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report." DBEs are the basis for the design, construction, and operation of the SSCs during accidents. For example, a LOOP with one or more NSR SSCs that fail to respond might be a DBE.

(3) Beyond Design Basis Events (BDBEs). BDBEs are defined as rare, off-normal events of lower frequency than that of DBEs. The evaluation of BDBEs is done to ensure that these events do not pose an unacceptable risk to the public.

The LBEs in all three categories will be evaluated individually to support the assessment of the performance of SSCs with respect to their safety functions in response to initiating events and collectively to demonstrate that the integrated risk of a multi-module plant design meets the NRC's safety goals.

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.

> In addition to the event classifications above, DOE/INL proposed the following definition for DBAs during public meeting discussions: DBAs should be deterministically selected from a review of LBEs by assuming that only SSCs relied on to meet 10 CFR 50.34 (those classified as SR) are available. Consequence acceptance criterion should be evaluated at the 95% upper bound of the mean value used to meet 10 CFR 52.79 offsite dose limits.

DOE/INL acknowledges that its DBE-derived definition of DBAs differs from how the term DBA has been defined and used for LWR licensing.

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The NGNP Licensing Strategy Report states that the technical approach for establishing the NGNP licensing basis and requirements is expected to involve the establishment of LBE categories (i.e., AOOs, DBAs, and beyond-design-basis accidents) based on the expected probability of event occurrence within each category selection of LBEs using deterministic engineering judgment complemented by insights from the NGNP PRA.

The NGNP categorization of LBEs into AEs, DBEs, and BDBEs is a reasonable approach for the classification of LBEs. These LBEs would include the LBEs selected using the proposed event selection process and the additional deterministically selected LBEs for inclusion in each category. The proposed event categories appear to be consistent with LWR event categorization practices.

The regulations in 10 CFR 52.79(a)(1)(vi)[6] require applicants for power reactor COLs to provide a description and safety assessment of the site that includes an evaluation of the major SSCs that “bear significantly on the acceptability of the site” under the radiological consequence evaluation factors. This assessment must assume a postulated fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The regulations in 10 CFR 100.21 require each applicant for a construction permit or operating license on or after January 10, 1997 (new reactors/advanced reactors), to comply with the similar requirements in 10 CFR 50.34(a)(1)(ii).

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 total effective dose equivalent (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.

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[6] The NGNP white papers generally refer to requirements in 10 CFR 50.34. For the proposed COL application, this assessment refers to 10 CFR 52.79 because it is the relevant regulation. However, the dose requirements discussed here are identical in the two regulations; therefore, this administrative detail has no effect on the technical requirements.

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,” in Appendix A to 10 CFR Part 50 for new reactors states 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 safety objective of 10 CFR 52.79 and 10 CFR Part 100, “Reactor Site Criteria,” is, in part, to establish the DID accident mitigation capability of the LWR containment. Among other things, the intent of the requirement is to provide reasonable assurance that the LWR containment can meet the dose guidelines in 10 CFR Part 100 even for a “major accident” or “postulated event” that results in the release of appreciable quantities of fission products into the containment.

SECY-93-092 describes the approaches proposed by non-LWR designers for the selection of events for consideration in the design and for safety classification. The approach proposed in SECY-93-092 included the following aspect: “A set of events would be selected deterministically to identify a containment challenge scenario.” In its response to SECY-93-092, the Commission issued an SRM on July 30, 1993, which approved the staff’s proposals.

The principle that the containment system must provide DID to prevent unacceptable fission product releases for the unknown or unexpected events is presented in a technology neutral manner in NUREG-1860, Volume 1, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing.” Section 4.3, “Defense-in-Depth Objectives and Principles,” of NUREG-1860 includes Principle 5, which states that “the plant design has containment functional capability to prevent an unacceptable release of radioactive material to the public.” NUREG-1860 states that the purpose of this principle is to “protect against unknown phenomena and threats, i.e., to compensate for completeness uncertainty affecting the magnitude of the source term.” Principle 5 further states the following:

> The design of the controlled leakage barrier should be based upon a process that defines a hypothetical event representing a serious challenge to fission product retention in the fuel and the coolant system. The applicant and NRC should agree upon a hypothetical event, consistent with the technology and safety characteristics of the design. The principle recognizes that the particular means used to retain or control the release will depend on the reactor technology.

Principle 5 describes the containment system as an essential aspect of the NRC’s DID philosophy and provides a design-basis approach and criteria for the DID capability of the containment system. The applicant must show that the containment system can prevent an unacceptable release of radioactive material to the public for a hypothetical event that represents a serious challenge to fission product retention in the fuel and the coolant system.

The NGNP mechanistic source terms white paper submitted on July 21, 2010 (ML102040260), describes the NGNP functional containment system as comprising several barriers that limit the release of radionuclides to the environment (defined as the source term) for each postulated event, including normal operating conditions, abnormal operating conditions, and accident conditions. The proposed functional containment and associated performance standards are addressed in part in the NRC staff's assessment report on fuel qualification and mechanistic source terms and have been further discussed between NRC and DOE/INL in public meetings during the follow-on assessment phase. The staff generally 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 assure 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:
>
> - 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 Exclusion Area Boundary 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.

The staff believes that the NGNP prototype should meet the siting dose guidelines in 10 CFR Part 100 for a set of physically plausible bounding events that represent a serious challenge to functional fission product retention within the overall functional containment system. The staff also believes that event sequences in the frequency range of $1x10^{-5}$ to $1x10^{-7}$ per plant-year in identifying hypothetical events for this purpose. In addition events in the BDBE frequency range should also be considered to ensure that the NGNP functional containment system provides sufficient DID to meet the intent of dose limits in 10 CFR Part 100 for a "major accident" or "postulated event" that results in the release of appreciable quantities of fission products, as required by 10 CFR 52.79.

The staff believes that any final NRC determination on whether and how estimated event sequence frequencies can be considered in defining hypothetical events for siting source term analysis will likely require guidance from the Commission. In particular, because current reactors have used deterministic engineering judgment to identify LBEs, the explicit consideration of frequency criteria for NGNP LBEs involves a new interpretation of regulations (e.g., 10 CFR 50.34(a)(1)(ii)(D) or 10 CFR 52.17(a)(1)(ix)) and, therefore, presents policy issues that the Commission would have to determine.

SUMMARY OF CONCLUSIONS

The staff agrees that categorization of events as AEs, DBEs, and BDBEs is a reasonable approach. However, the staff believes that it may be necessary to consider bounding credible events that could fall within the BDBE region, as defined by DOE/INL. These deterministically selected events would satisfy the criteria in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Chapter 15, “Transient and Accident Analysis,” for postulated DBAs and the hypothetical dose consequence source terms to ensure adequate DID for containment of fission products in accordance with regulatory requirements. The staff also expects that Commission consideration will be necessary to determine criteria for excluding events in the BDBE range and to determine which events should be evaluated for cliff-edge considerations.

3.1.3 LBE Outcome Objective 3 – Licensing Basis Event Frequency Ranges

The LBE white paper proposes that the frequencies of LBEs be expressed in units of events per plant-year, where a plant is defined as a collection of reactor modules having certain shared systems. The proposed frequency ranges for the LBE categories are as follows:

- AEs: Event sequences with mean frequencies greater than $1x10^{-2}$ per plant-year
- DBEs: Event sequences with mean frequencies less than $1x10^{-2}$ per plant-year and greater than $1x10^{-4}$ per plant-year
- BDBEs: Event sequences with mean frequencies less than $1x10^{-4}$ per plant-year and greater than $5x10^{-7}$ per plant-year

STAFF ASSESSMENT

The staff believes that the proposal in the LBE white paper is reasonable. To account for multi-module plants, the paper proposes expressing the frequencies of LBEs in units of events per plant-year, where a plant is defined as a collection of reactor modules that have selected shared systems. The guidelines for the upper and lower frequency bounds for categorizing events should also be on a per-plant-year basis.

The NGNP Licensing Strategy Report states that the technical approach to establishing the NGNP licensing basis and requirements will probably involve the establishment of LBE categories based on the expected probability of event occurrence. Within each category, the selection of LBEs uses deterministic engineering judgment complemented by insights from the NGNP PRA. For events that involve a single reactor module, the frequency ranges per reactor module per year would be the proposed frequency ranges divided by the number of reactor modules that comprise the plant. For example, for such events (as stated above) for an eight-reactor-module plant design, the proposed lower frequency cutoff of $1x10^{-4}$ per plant-year for DBEs that affect only one of the eight reactors would result in a lower frequency cutoff guideline of about $1x10^{-5}$ per reactor-year. However, for a four-reactor-module plant design, the proposed lower bound frequency cutoff of $1x10^{-4}$ per plant-year for DBEs that affect only one of the four reactors would result in a lower frequency cutoff of about $2.5x10^{-5}$ per reactor-year. Thus, the cutoff frequencies on a per-reactor-year basis would vary depending on the number of reactor modules in the plant.

SECY-03-0047, “Policy Issues Related to Licensing Non-Light-Water Reactor Designs,” dated March 28, 2003 (ML030160002), includes Issue 4, “Probabilistic Event Selection, Safety Classification and Reliability Criteria.” The discussion of Issue 4 includes a table titled “Example of Event Selection Criteria,” that provides an example of LBE categories, the dose acceptance criteria for each category, and the frequency range for each category. The table presents examples of event selection criteria for which AEs may have a frequency greater than $1x10^{-2}$ per plant-year, DBEs may range in frequency from $1x10^{-2}$ per plant-year to $1x10^{-6}$ per plant-year, and emergency planning basis events (i.e., BDBEs) may range in frequency from $1x10^{-6}$ per plant-year to $1x10^{-8}$ per plant-year. A footnote in the table proposes that the frequency range for each event category should apply to the initiating event frequencies or event scenario (i.e., event sequence) frequencies. The footnote is based, in part, on current LWR regulatory practices that generally consider the initiating event frequency instead of the event sequence frequency for categorizing events.

Consistent with this guidance, the staff believes that DBEs involve event sequences with mean frequencies ranging from $1x10^{-2}$ per plant-year to $1x10^{-4}$ per plant-year. This frequency range would apply to any plant regardless of the number of modules. For example, the frequency range of DBEs would be the same for a one-module plant and a four-module plant. Therefore, this frequency range per reactor-year for multi-module plants would be lower (more conservative) with each additional module.

Additionally, the staff believes that BDBEs should involve event sequences with mean frequencies ranging from $1x10^{-4}$ per plant-year to $5x10^{-7}$ per plant-year. This frequency range would apply to any plant regardless of the number of modules. For example, the frequency range of BDBEs would be the same for a one-module plant and a four-module plant. Therefore, this frequency range per reactor-year for multi-module plants would be lower (more conservative) with each additional module.

The staff believes that the above frequency ranges are only guidelines and are not sharp demarcations of the event category frequency boundaries. In the categorization of events, the staff believes that the applicant will need to apply conservative engineering judgment to address the uncertainty in the LBE frequency.

In developing its views on frequency ranges, the staff has followed a principal of maintaining consistency with approaches that it has taken in the selection of values for this metric in LWR applications. This approach is consistent with a “technology neutral” approach, such as that documented in NUREG-1860. The staff has chosen this approach because any difference in the level of safety associated with a modular HTGR design, as compared to an LWR design, has yet to be demonstrated. If it becomes evident that, after the presentation and evaluation of a design for the NGNP, an inherently positive difference exists, the staff may be able to justify a relaxation in its position. However, such a change would be a Commission policy issue.

Finally, RAI LBE-17 asked DOE/INL to clarify whether the events that would be used for the design basis for SR SSCs designed to national codes and standards (e.g., Section III of the ASME Boiler and Pressure Vessel Code (ASME Code)) would be based on events in a specific frequency range (e.g., DBE frequency range) or would be based on events and conditions defined by the codes. DOE/INL responded that the selection and application of codes and standards and the associated design rules will not be solely based on event frequencies, as defined for DBEs. For example, DOE/INL states that the construction rules for the reactor coolant pressure boundary in Section III of the ASME Code would be applied. The approach described in the RAI LBE-17 response is reasonable.

SUMMARY OF CONCLUSIONS

The staff believes that DOE/INL’s proposed frequency ranges for DBEs and BDBEs are reasonable. The ranges for the proposed event sequence frequencies are as follows:

- DBE frequencies range from 1×10^{-2} to 1×10^{-4} per plant-year.
- BDBE frequencies range from 1×10^{-4} to 5×10^{-7} per plant-year.

The approval of frequency ranges for the various event categories is a Commission policy issue because it involves a new interpretation of the regulations and associated guidance for demonstrating compliance.

3.1.4 LBE Outcome Objective 4—Event Consequence Acceptance Limits

The LBE white paper proposes the following acceptable limits on the event sequence consequences and the analysis basis for the LBE categories:

- The limit for AOOs (later renamed to AEs), in accordance with 10 CFR Part 20, "Standards for Protection against Radiation," is 100-millirem (mrem) TEDE, mechanistically modeled and realistically calculated at the EAB.

- The limit for DBEs, in accordance with 10 CFR 50.34, is 25-rem TEDE, mechanistically modeled and conservatively calculated at the EAB.

- The limit for BDBEs is based on NRC safety goal quantitative health objectives (QHOs) and is mechanistically and realistically calculated at 1 mile (1.6 kilometers (km)) and 10 miles (16 km) from the plant.

The LBE white paper states that the proposed acceptable public consequences are based on the existing regulations and policy, as described in Section 2 of the paper. The annual dose limits in 10 CFR Part 20 would apply to AEs, and the 2-hour dose limit in 10 CFR 50.34 would apply to DBEs and the DBAs. The safety goal QHOs would apply to BDBEs and would be applied to all the LBEs in an integrated manner. The paper states that the bases for the analyses will follow conventional practice for each of the LBE categories and the respective top-level regulatory criteria (TLRC).

STAFF ASSESSMENT

DOE/INL proposes dose acceptance limits on the LBE consequences and the analysis basis for the LBE categories, which are generally consistent with the NGNP LS with several significant exceptions. Figure 1, a reproduction of Figure 3 from the NGNP LBE white paper, depicts the proposed dose acceptance limits for the NGNP safety analyses.

The event frequency versus dose consequence acceptance line shown in Figure 1 is generally referred to as an "F-C curve." For events in the AE frequency region, construction of the proposed F-C curve is based on the dose limits in 10 CFR Part 20. For events in the DBE frequency region, construction of the proposed F-C curve is based on the dose limits in 10 CFR 50.34.[7] For events in the DBE region, the proposed F-C curve has been constructed in such a manner that would allow a reduced dose as DBE frequency increases. For the DBAs (which are selected and derived from the DBEs), DOE/INL proposes that dose criteria in 10 CFR 50.34 apply (i.e., without reduction). For events in the BDBE frequency region,

[7] As noted above, the requirements in 10 CFR 50.34 are the same as those in 10 CFR 52.79, which is the regulation pertinent to a COL. The COL is the planned application type for the NGNP, as described in the Licensing Strategy Report.

construction of the proposed F-C curve is based on the prompt mortality safety goal. DOE/INL proposed that the dose limits be associated with, and apply at, the NGNP EAB.

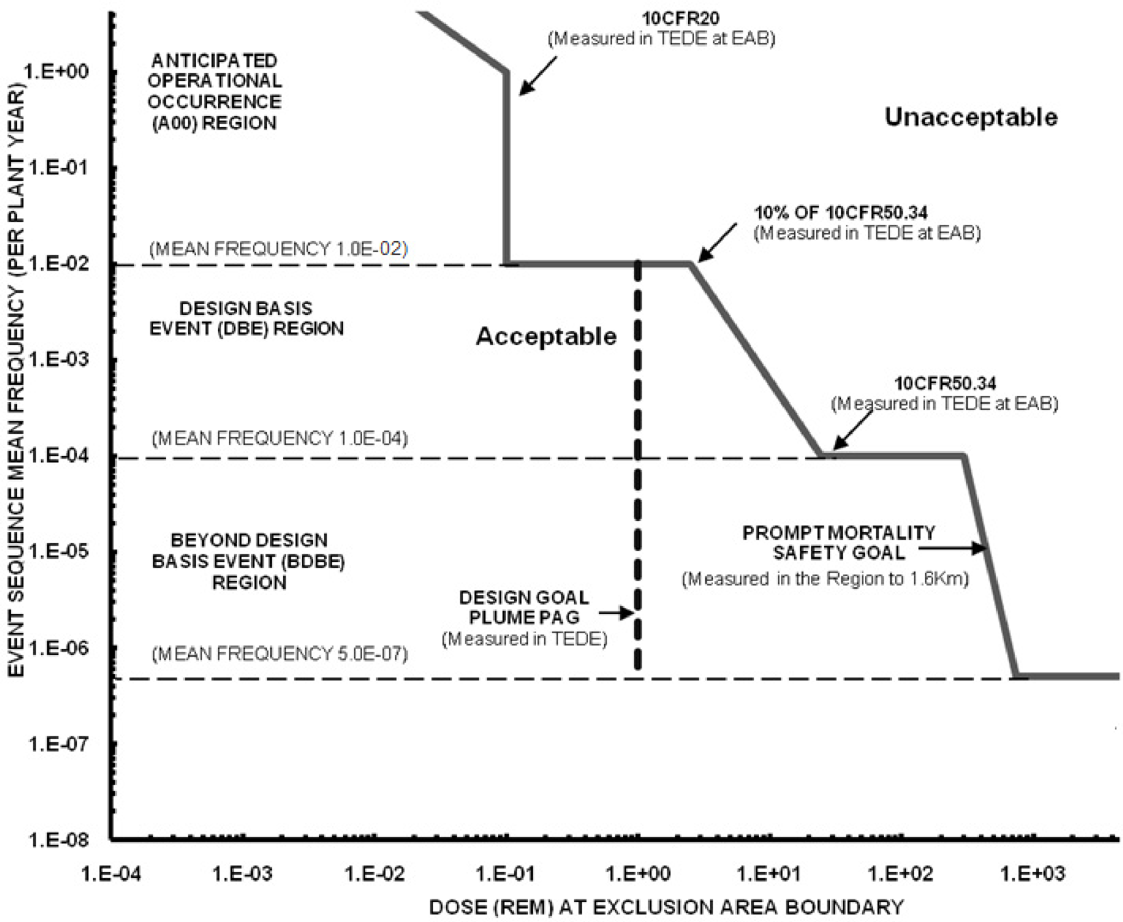

Figure 1 Proposed dose acceptance limits for the NGNP safety analyses (Source: Figure 3 in the NGNP LBE white paper)

The overall construction of the proposed F-C curve is intended to generally follow an iso-risk curve to allow low doses for relatively frequent events and higher dose limits for rare events. The intent of the construction of the F-C curve is to ensure that the overall NGNP risk for all LBEs combined will meet the NRC's safety goals.

The Licensing Strategy Report states that the technical approach for establishing the NGNP licensing basis and requirements is expected to involve the use of consequence acceptance limits for onsite or offsite releases for LBEs that are consistent with current dose limits for LWRs in 10 CFR Part 20 and 10 CFR 50.34.[8] The dose requirements in 10 CFR 50.34 and 10 CFR 52.79 are equivalent; therefore, these requirements have no technical distinction for purposes of the F-C curve discussion. The NGNP Licensing Strategy Report also states that

[8] As noted above, the Licensing Strategy Report also describes a plan for a COL application for which the pertinent regulation is 10 CFR 52.79.

the assessment of radiological consequences for LBEs would be done using event-specific mechanistic source terms.

The staff recognizes that there are a large number of ways to construct an F-C curve that use the dose acceptance criteria of existing NRC regulations and applies them to inferred frequency ranges for the event categories to meet the above stated objectives. For example, the proposed frequency consequence curve shown in Figure 3-3 in NUREG-1860 was developed in connection with the NRC staff's feasibility study for a risk-informed and performance-based regulatory structure for future plant licensing. The development of the F-C curve shown in Figure 3-3 of NUREG-1860 was done using the same LWR regulations that DOE/INL used to develop its proposed F-C curve. The development of the F-C curve in NUREG-1860 also includes consideration of additional dose acceptance criteria associated with national and international radiological health standards.

The NRC staff evaluated whether the specific F-C curve proposed for modular HTGRs associated the agency's top-level regulatory requirements (i.e., dose criteria) with event frequency ranges in an appropriate and reasonable manner. The staff's evaluation also considered whether the proposed analysis rules (e.g., best estimate and conservative) were appropriate for performing the deterministic safety analysis for the events in each category. The staff did evaluate the proposed F-C curve in an effort to establish an F-C curve that it could apply to all reactor technologies on a technology-neutral basis for future plant licensing.

Acceptance Criteria for Anticipated Events

The staff believes that, as proposed for AEs by the LBE white paper, the regulatory limits in 10 CFR Part 20 should apply (i.e., 100-mrem TEDE) at the EAB. The LBE white paper proposes that the calculation of the dose at the EAB for each AE should be mechanistically modeled and realistically calculated.

SECY-03-0047 proposes that the dose consequences of AOOs should be calculated on a conservative basis. More recently, as documented in Table 6-3 in NUREG-1860, the staff proposed that, with the exception of the mechanistic source term calculation, realistic calculations be conducted to obtain the mean and uncertainty distribution for estimating the consequences. That is, while the source term calculation should model all SSCs that have a role in determining AOO consequences, the staff proposed that the mechanistic source term calculation should use a conservative 95-percent probability value.

In meetings with the staff in May 2012 and December 2012, DOE/INL stated that the NGNP event sequences classified as AOOs relate to the types of events that are normally described in Chapters 11 and 12 of the final safety analysis report for LWRs—not those that require (1) mitigation to prevent a challenge to the integrity of the reactor coolant pressure boundary, (2) the reactor to be shut down and maintained in a safe shutdown condition, or (3) mitigation to prevent potential offsite exposures from exceeding the applicable guideline exposures in 10 CFR 50.34(a)(1) or 10 CFR 100.11—and that should be evaluated realistically on an expected mean basis versus the offsite dose limit in 10 CFR Part 20.

The staff believes that the dose calculation model for AEs should include all the SSCs that have a role in the safety analysis of the event sequence and that a mechanistic source term should be used to demonstrate that the plant has met the dose limits in 10 CFR Part 20 as long as the source term analysis is bounded where insufficient data are available through conservative deterministic assumptions.

Additionally, LWR safety requirements include the establishment of safety limits on the principal fission product barriers, such as the fuel barrier, and direct plants to not exceed the established safety limits for AOOs. Regulatory guidance provides conservative calculation methods and assumptions that plants should use to demonstrate that they have not exceeded the established safety limits. Ensuring barrier integrity for LWRs is considered an element of DID for AOOs.

The DID principles in NUREG-1860 also include the expectation that appropriate safety limits will be placed on the key barriers, such as the fuel barrier. The safety limit is established to ensure that there is very low probability of loss of the barrier safety function and that the applicant demonstrated that it has met the appropriate limits with high confidence. NUREG-1860 advocates that the applicant use the 95-percent probability value of the design distribution to show that it has met the regulatory safety limit.

The NGNP fuel qualification white paper does not identify (or identify a need for) a safety limit for the NGNP fuel barrier for AEs or any other event category. As such, DOE/INL has not proposed to provide separate calculations to demonstrate that a required level of fuel integrity is met for AEs. DOE/INL has instead proposed a concept that involves the use of multiple barriers to show that the combined effectiveness of all barriers is sufficiently effective to meet the top-level regulatory requirements, such as 10 CFR Part 20.

The staff believes that DOE/INL should pursue the development of an appropriate regulatory limit (e.g., safety limits and limiting conditions for operation) to ensure the required level of integrity of the fuel barrier during normal operation and AEs. As a minimum, the staff believes that the deterministic safety analyses for AEs should include a demonstration that the plant has not exceeded the fuel design limits (i.e., the maximum design conditions, such as the maximum fuel irradiation temperature, associated with the NGNP fuel qualification test program) during any AE.

During public meeting discussions with DOE/INL, the staff clarified the treatment for evaluating frequently occurring events classified as AEs. Current regulatory practice requires a conservative analysis of AOOs in regard to their effect on fuel integrity and safety limits. Large LWRs cannot exceed the specified acceptable fuel design limits (SAFDLs) by design. Therefore, AOOs in LWRs would not, by definition, challenge TLRC dose consequence limits.

The staff notes with particular interest that DOE/INL included in a December 2011 report on NGNP project status the following task in a list of required future technical support activities:

> Develop specified acceptable fuel design limits (SAFDLs) for the HTGR particle fuel type, since the SAFDL structure that's been established for LWR fuel can't be applied to particle fuel.

The staff would expect any SAFDLs (or analogous fuel and core design limits) for a modular HTGR to ensure substantial margin to dose limits in the AE region. In any case, the staff would expect the licensee to conservatively evaluate such limits for operational events or for AEs that have a potential to challenge the SAFDLs and to credit only SR SCCs for mitigation. The evaluation of AE dose consequences against the criteria in 10 CFR Part 20 could then be based on a realistic best-estimate evaluation using the mean value from of a mechanistic source term uncertainty analysis.

Acceptance Criteria for Design-Basis Events

The staff believes that for DBEs, the regulatory limits in 10 CFR 50.34 and 10 CFR 52.79 apply (i.e., 25-rem TEDE) at the EAB for the 2-hour dose and the 30-day low population zone dose. The LBE white paper proposes that the doses be mechanistically modeled and conservatively calculated. The proposal to conservatively calculate the mechanistic source term and the dose consequences for DBEs is consistent with the earlier staff views documented in the table in Issue 4 of SECY-03-0047 and Table 6-3 in NUREG-1860 and is considered reasonable.

Acceptance Criteria for Beyond-Design-Basis Events

The LBE white paper proposes that the NRC's safety goal QHOs for BDBEs should apply and that these dose consequences of BDBEs should be mechanistically modeled and realistically calculated at 1 mile (1.6 km) and 10 miles (16 km) from the plant.

As stated in LBE Outcome Objective 2, the staff believes that BDBEs should be evaluated to ensure that they do not pose an unacceptable risk to the public and that they should be considered in the development of the accident source term for conducting the NGNP siting analysis under 10 CFR 50.34 or 10 CFR 52.79 and for ensuring DID in the capability for containment of fission products. The staff believes that Commission guidance may be necessary to resolve these issues.

SUMMARY OF CONCLUSIONS

The staff believes that the proposed F-C curve and the associated dose calculation framework are generally reasonable with a few exceptions, as follows:

- Regulatory controls should be established to ensure that fuel integrity is maintained throughout the normal operation envelope and for AEs.

- Bounding events that would otherwise fall within the BDBE region should be evaluated to ensure adequate DID for the containment of fission products in accordance with regulatory requirements.

- Deterministic elements of the proposed approach should be strengthened to ensure conservative selection of bounding events, including events used to justify the siting source term and the proposed emergency response measures.

Like the conclusions in Section 3.1.3, the definition of dose acceptance criteria for the various event categories is a Commission policy issue because it involves a new interpretation of the regulations and associated guidance for demonstrating compliance.

3.1.5 LBE Outcome Objective 5—Lower Bound of Event Frequency

The LBE white paper proposes that the frequency below which events are not selected as LBEs is 5×10^{-7} per plant-year. The PRA examines events to 1×10^{-8} per plant-year to ensure that there are none just below this de minimus frequency. The evaluation of BDBEs will be done to ensure that they meet the NRC safety goals at the prescribed distances from the plant. DOE/INL proposes 5×10^{-7} per plant-year because it claims that lower frequency events meet the NRC safety goal QHO for acute individual risk of fatality.

STAFF ASSESSMENT

As discussed in LBE Outcome Objectives 2 and 4, the staff believes that BDBEs should be evaluated to ensure that they do not pose an unacceptable risk to the public and that they should be considered in the development of the accident source term for conducting the NGNP siting analysis under 10 CFR 50.34 or 10 CFR 52.79 and for ensuring DID for containment of fission products.

As documented in the table in Issue 4 of SECY-03-0047 and Table 6-3 in NUREG-1860, the staff provided examples of event category frequency ranges, including frequency ranges for events in the lowest frequency category. SECY-03-0047 provides the staff's preliminary view that events in the lowest frequency category (i.e., events considered for emergency planning) should consider initiating events or event sequences down to $1x10^{-8}$ per plant-year. In the SRM for SECY-03-0047, the Commission approved the staff's recommendations for Issue 4.

Table 6-1, "Proposed Dose/Frequency Ranges for [the] Public," in NUREG-1860 extends the F-C curve dose consequence limits for events down to a frequency of $1x10^{-7}$ per year. Furthermore, Section 6.4, "LBE Selection Process and LBE Criteria," of NUREG-1860 documents a proposed LBE selection process and provides the lower bound frequency for events that will be included in the event selection process. Step 2 of the proposed process states that all PRA sequences with a point estimate frequency of less than $1x10^{-8}$ per year should be dropped from consideration as potential LBEs. Step 3 of the proposed process states that, for sequences that have a point estimate frequency equal to or greater than $1x10^{-8}$ per year, the mean and 95^{th} percentile frequency should be determined. Finally, Step 4 of the proposed process states that all PRA event sequences with a 95^{th} percentile frequency greater than $1x10^{-7}$ per year should be identified for inclusion in the event categories.

As previously stated in its assessment of LBE Outcome Objective 3, the staff believes that BDBEs should include event sequences or initiating events with mean frequencies down to $5x10^{-7}$ per plant-year, which would be equivalent to event sequences or initiating events with mean frequencies of approximately $1x10^{-7}$ per reactor-year for a plant multi-modular plant with four reactor modules. The staff believes that $5x10^{-7}$ per plant-year provides a reasonable cutoff for assessing whether the NGNP meets the NRC safety goals. The staff's preliminary view is that PRA event sequences with a point estimate below $1x10^{-8}$ per plant-year can be dropped from consideration as potential LBEs.

The PRA event sequence frequency cutoff for establishing the spectrum of events that will be considered for siting and the associated emergency planning requirements is a Commission policy issue.

SUMMARY OF CONCLUSIONS

The staff believes that BDBEs should include events with a mean frequency of $5x10^{-7}$ per plant-year or greater. A Commission policy decision will be necessary to define the event frequency used to establish emergency planning requirements.

3.1.6 LBE Outcome Objective 6 Events, Failures, and Natural Phenomena Evaluated

The LBE white paper proposes that the kinds of events, failures, and natural phenomena that will be evaluated should include the following:

- multiple, dependent, and common-cause failures to the extent that these contribute to LBE frequencies
- events that affect more than one reactor module
- internal events (including transients and accidents) and internal and external plant hazards that occur in all operating and shutdown modes and that potentially challenge a plant's capability to satisfactorily retain any source of radioactive material

The PRA that supports the application will involve a full-scope evaluation (including all operating modes). The PRA white paper provides additional information on this approach.

STAFF ASSESSMENT

The NGNP Licensing Strategy Report states that the technical approach to establishing the NGNP licensing basis and requirements is expected to involve the "selection of licensing-basis events using deterministic engineering judgment complemented by insights from the NGNP PRA." The Licensing Strategy Report further states that "once the NGNP technology is demonstrated through successful operation and testing of the NGNP prototype, and a quality PRA, including data, becomes available, greater emphasis on [a] design-specific PRA to establish the licensing basis and requirements will be a more viable option for licensing a commercial version of the NGNP reactor."

The PRA white paper addresses the approach used for developing the NGNP PRA, including the kinds of events, failures, and natural phenomena that will be evaluated. As discussed earlier in LBE Outcome Objective 1, Section 3.6.1of the PRA white paper describes how a deterministic approach will be used to identify potential initiating events. The PRA white paper also describes the approach used for modeling multiple, dependent, and common-cause failures. The staff agrees that the NGNP PRA will need to include an evaluation of multiple, dependent, and common-cause failures. The staff agrees that the evaluation of multiple, dependent, and common-cause failures by the PRA will contribute to both LBE frequencies; however, such kinds of failures can also contribute to the LBE consequences. Events affecting more than one reactor module will need to be considered for a plant with multiple reactor modules.

The staff agrees that the NGNP PRA should include internal events (including transients and accidents) and internal and external plant hazards that occur in all operating and shutdown modes and that potentially challenge a plant's capability to satisfactorily retain any source of radioactive material. The NGNP PRA would include both internal events initiated by faults and failures in the balance of plant, including a process heat facility, and external events initiated by a process heat facility near the NGNP.

The staff agrees that the kinds of events and failures and natural phenomena that should be evaluated should include those proposed by DOE/INL and that a full-scope PRA for all NGNP modes (i.e., operating and shutdown modes) and for all sources of radioactive material (e.g., reactor core, spent fuel storage, and waste cleanup and handling systems) should be used for the assessment.

Finally, as stated in its assessment of LBE Outcome Objective 1, the lessons learned from Fukushima Dai-ichi may require additional regulatory requirements on the analysis of external events (e.g., natural phenomena) and the associated event sequences that will be considered for NGNP licensing. Such additional requirements have yet to be established.

SUMMARY OF CONCLUSIONS

The staff agrees with the proposed scope of events and phenomena that should be considered; however, it notes that additional requirements or guidance may arise from the NRC's evaluation of lessons learned from Fukushima Dai-ichi.

3.1.7 LBE Outcome Objective 7 – Design Basis Accidents

The LBE white paper states that DBAs for Chapter 15 of the license application are derived from the DBEs by assuming that only SSCs classified as SR are available to mitigate the consequences. The public consequences of DBAs are based on mechanistic source terms and are conservatively calculated. The upper bound consequence of each DBA must meet the consequence limit in 10 CFR 50.34 at the EAB.

STAFF ASSESSMENT

The staff believes that the proposed NGNP approach for deriving DBAs from the DBEs by demonstrating success pathways for the DBAs relying solely on the response actions of SR SSCs follows regulatory requirements and practice. DBAs are selected from LBE sequences that have a proposed event sequence frequency of between $1x10^{-2}$ per plant-year and greater than $1x10^{-4}$ per plant-year (LBE Outcome Objective 3). Furthermore, the staff agrees that the consequences of DBAs can be based on event-specific mechanistic source terms and that these source terms should be calculated on a conservative basis. However, the DOE/INL has not yet provided the details of the conservative mechanistic source term calculation methodology; therefore, the staff has not been able to evaluate whether the margin associated with such calculations is reasonable. The staff expects to make a determination on this issue during interactions with the NGNP license applicant.

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.
For example, selected BDBEs that involve the failure of the HPB may potentially cause significant and prolonged air ingress into the reactor vessel and significant oxidation of the core graphite and fuel carbonaceous material (e.g., degradation of fuel particle coating layers). Such events may potentially cause substantial releases of fission products from the core, HPB, and reactor building unless SSCs are provided to limit air ingress and to limit the magnitude and duration of fission product transport from the reactor building. If additional SSCs are necessary for such events to ensure that the plant meets the F-C curve acceptance criteria, the staff believes that it should include such events as DBAs and should classify any such additional SSCs necessary to meet the F-C curve acceptance criteria as SR. The process and criteria that should be used for the selection of DBAs to demonstrate compliance with 10 CFR 52.79 are Commission policy issues.

SUMMARY OF CONCLUSIONS

The staff believes that DOE/INL must address the following issues before the NRC can endorse the proposed approach for evaluating DBAs:

- As discussed in Section 3.1.2 and elsewhere, the staff believes that postulated bounding events that would otherwise fall within the BDBE region must be considered DBAs to ensure adequate DID for the containment of fission products in accordance with regulatory requirements. The staff expects that a Commission policy decision will be necessary if such events are to be excluded from consideration.

- The staff believes that additional design information will be necessary to determine whether the margins in DBA dose calculations provide adequate conservatism.

3.1.8 LBE Outcome Objective 8—Event Frequency and Consequence Uncertainties

The LBE white paper states that uncertainty distributions are evaluated for the mean frequency and the mean consequence for each LBE. The mean frequency is used to determine whether the event sequence family is an AE, DBE, or BDBE. If the upper or lower bound on the LBE frequency straddles two or more regions, the LBE is compared to the consequence criteria for each region. The mean, lower, and upper bound consequences are explicitly compared to the consequence criteria in all applicable LBE regions. The upper bound for the DBE and DBA consequences must meet the regulatory criteria for EAB doses.

STAFF ASSESSMENT

NUREG-1860 proposes an approach to the treatment of event frequency uncertainties and event consequence uncertainties for establishing whether the event sequence family should be considered an AOO, DBE, or BDBE. Section 6.4.1, “Probabilistic LBE Selection,” of NUREG-1860 describes the process proposed by the staff. The staff’s approach made it unnecessary to consider whether the upper and lower bound of the LBE frequency straddled two LBE categories (i.e., regions). The more conservative approach (i.e., higher LBE frequency) would result in an LBE that would have to meet the same or potentially more restrictive dose criteria than the approach proposed by DOE/INL. However, the event categorization process proposed by the LBE white paper includes an additional step to compare the LBE consequences to the consequence acceptance criteria of the more restrictive region if the upper or lower bound on the LBE frequency straddles two regions.

The staff believes that the evaluation approach proposed by the LBE white paper to account for uncertainty distributions in the LBEs (i.e., event sequence frequency distributions) in selecting the appropriate category for LBEs (i.e., AE, DBE, or BDBE) is reasonable as a guideline. For this reason, the staff views the frequency ranges for categorizing LBEs as guidelines, not sharp break points in the categorization of events. Considering a BDBE that falls near the upper frequency boundary for such events could ensure that such an event is addressed in an appropriately conservative manner. Any final staff decisions on the adequacy of LBE categorization would be made during the NGNP licensing review.

However, as discussed in the assessment of LBE Outcome Objective 1, the staff believes that the selection of additional events in the event categories should be done using deterministic judgment complimented by insights from the NGNP PRA. The categorization of these events should be based on deterministic engineering judgment complemented by risk insights. For example, a rare event that might otherwise be considered a BDBE might be selected as a postulated event to be evaluated in addition to the DBAs to ensure adequate DID, as discussed above.

SUMMARY OF CONCLUSIONS

The staff believes that the basic approach for addressing frequency and consequence uncertainties appears reasonable. However, evaluation of the adequacy of the NGNP LBE categorization cannot be done until more design information is available. As noted previously, the staff also believes deterministic elements of the LBE categorization process should be strengthened.

3.2 Defense-in-Depth Approach

The DID white paper defines DID as a safety philosophy that is based on multiple lines of defense, safety margins, and compensatory measures that are applied to the design, construction, operation, maintenance, and regulation of nuclear plants to prevent and mitigate accidents and to ensure adequate protection of public health and safety.

The staff notes that the SRM dated September 10, 2007 (ML072530501), on SECY-07-0101, "Staff Recommendations regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50," dated June 14, 2007, directs the NRC staff to develop a draft policy statement on DID for future plants. The NRC expects future NGNP designers or applicants to address the provisions of that policy statement if it is approved by the Commission prior to the submittal of an NGNP license or design application.

3.2.1 DID Outcome Objective 1—Adequacy of Defense-in-Depth Definition

The proposed risk-informed and performance-based framework for DID has three major elements: (1) plant capability DID, (2) programmatic DID, and (3) risk-informed evaluation of DID. These elements are viewed as complementary for ensuring that a design is tolerant to uncertainties in the knowledge of plant behavior, component reliability, or operator performance that might compromise safety.

The intent of plant capability DID is to ensure that the designer has incorporated multiple lines of defense in designing the functional capability of the physical plant, including conservative design approaches for the barriers and SSCs that perform safety functions associated with the prevention and mitigation of accidents.

The intent of programmatic DID is to ensure that the programmatic actions for designing, constructing, operating, testing, maintaining, and inspecting the plant are adequate to ensure that the plant's capabilities are maintained throughout the life of the plant.

The risk-informed evaluation of DID should be a structured, logical process for assessing the adequacy and sufficiency of the plant's capabilities and programmatic DID and should be based on risk insights from the PRA in defining LBEs, identifying the roles of SSCs in the prevention and mitigation of accidents, and addressing uncertainties.

STAFF ASSESSMENT

The NGNP RIPB staff agrees with the high-level definition of DID as “a safety philosophy that is based on multiple lines of defense, safety margins, and compensatory measures that are applied to the design, construction, operation, maintenance, and regulation of nuclear plants to prevent and mitigate accidents and to ensure adequate protection of public health and safety” (Section 3.2.1). An instructive discussion of the DID philosophy also appears in the Fukushima Near-Term Task Force Report,[9] which describes DID as encompassing the following five criteria:

(1) Require the application of conservative codes and standards to establish substantial safety margins in the design of nuclear plants.

(2) Require high quality in the design, construction, and operation of nuclear plants to reduce the likelihood of malfunctions and to promote the use of automatic safety system actuation features.

(3) Recognize that equipment can fail and operators can make mistakes and, therefore, require redundancy in safety systems and components to reduce the chance that malfunctions or mistakes will lead to accidents that release fission products from the fuel.

(4) Recognize that, in spite of these precautions, serious fuel damage accidents may not be completely prevented and, therefore, may require containment structures and safety features to prevent the release of fission products.

(5) Further require that comprehensive emergency plans be prepared and periodically exercised to ensure that actions can and will be taken to notify and protect citizens near a nuclear facility.

Of the above criteria, the staff does not clearly understand the NGNP approach to Criterion 4. NRC regulations address this criterion in the requirements for evaluating a hypothetical dose consequence based on the postulated fission product release outlined in 10 CFR Part 100. The regulatory requirement is based on a fission product release from a major credible accident at a large LWR; the NRC imposes this requirement to ensure that plants consider the mitigation of consequences and the prevention of such very low probability but potentially high consequence events during the licensing process.

The staff believes that the plant should consider a spectrum of deterministically selected credible events that adequately represent the intent of the fission product release described in 10 CFR Part 100 for siting purposes as part of the DID measures in the safety design of a modular HTGR.

Events with moisture ingress and large breaks in the primary pressure boundary may maximize the pressure-driven prompt releases from the modular HTGR functional containment system. The selection of large break sizes and locations to be used in siting analysis should be informed by critical examination of the plausibility of gross vessel failure in the modular HTGR conceptual designs being considered 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

[9] See the report entitled, “Recommendations for Enhancing Reactor Safety in the 21st Century, The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” issued July 2011 (ML111861807).

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 in the building and primary system to inherently delay or limit oxygen transport to the graphitic core and support structures.

Although the DID white paper correctly notes that the LWR severe accident definition is not applicable to HTGRs, the staff believes that a severe accident definition more pertinent to a modular HTGR could be a very low probability event that results in a significant release of radionuclides from the primary pressure boundary.

In regard to the DOE/INL assertion that modular HTGR bounding source terms should be calculated for a spectrum of BDBEs that are of comparable likelihood to those of LWRs (i.e., > $1x10^{-6}$ per reactor-year), the staff believes that this approach may not provide adequate DID. The mechanistic source term white paper seems to convey the message that, for all BDBEs considered in that white paper, the peak fuel temperature does not rise significantly. However, the staff does not understand what “comparable likelihood” means in this context.

DID for NGNP is intimately related to several other white paper topics (e.g., LBE selection (discussed above), safety classification and treatment of SSCs, mechanistic source term, and PRA). The staff considers the DOE/INL proposed approach for selecting the LBEs using deterministic judgment complemented by insights from the NGNP PRA, in theory, reasonable.

To ensure that no “cliff-edge effects” (i.e., large consequence spikes) occur from events just beyond the proposed NGNP LBE spectrum and to understand the ultimate safety capability of the NGNP design, the NRC staff should select a set of events for exploratory evaluation of the modular HTGR “safety terrain,” with the understanding that such events should be physically plausible, and should consider passive characteristics and safety behavior intrinsic to modular HTGRs. “Safety terrain” insights from such exploratory studies should inform considerations for the selection of bounding event sequences for plant siting and functional containment design evaluations. These exploratory studies should reflect the Commission’s PRA Policy Statement by blending the strengths of probabilistic and deterministic methods. The staff believes that conservative deterministic inputs to the selection of LBEs could ensure adequate DID for the NGNP prototype.

The concept of risk metrics is an important consideration in assessing the adequacy of the DID approach for the NGNP. The DID white paper states that the conventional risk metrics, such as core damage frequency as it is used in the context of LWRs, are not applicable to the NGNP. The white paper and the response to RAI DID-1 suggest instead that the specific risk metrics that will be used for the NGNP will be a product of the PRA and will consider the following factors to ensure that an adequate set of controls for public protection and DID is provided:

- The release categories will be sufficient to address the integrated risk of a multiple-reactor module facility.

- Some release categories will involve source terms from two or more reactor modules, whereas others will involve releases from non-core sources of radioactive material, such as the spent fuel storage.

- Risk metrics will include event sequence frequencies associated with LBEs, release categories, and a quantification of the offsite radiological consequences to facilitate comparisons to the TLRC and the QHOs.

- Risk significance will be defined in terms of release category frequencies and will meet the requirements of the supporting PRA standards. Risk significance will also address the margins relative to the TLRC and QHOs.

- The risk metrics used in typical LWR PRA applications, such as core damage frequency and large early release frequency, are based on Level 1 PRAs with modest extensions in the Level 2 domain to address large early release frequency.

- A comparison of the risk metrics in the NGNP PRA to the risk metrics of a full-scope Level 3 PRA for an LWR is more useful than evaluating them at an intermediate level that is specific to LWRs.

The PRA white paper further elaborates on the above statements, although the staff believes that the white paper does not provide a clear definition of the risk metrics for the NGNP. The staff makes the following observations in this regard:

- Risk metrics should include event sequence frequencies associated with LBEs that would result in bounding source terms.

- A consensus PRA standard, which is essential for assessing risk significance, has yet to be approved.

- A full-scope Level 3 PRA, which is essential for establishing the NGNP risk metrics, has not yet been developed.

In RAI DID-13, the staff asked for clarification of the role of cross-cutting issues, such as emergency planning, and design codes and standards. Based on the information provided in the RAI response, the staff agrees with the NGNP approach that selection of codes and standards supports both the plant capability DID and the programmatic DID. Likewise, the staff agrees that emergency planning is also a cross-cutting element because it includes elements of plant capability DID and programmatic DID.

The staff notes that the NGNP DID approach describes multiple concentric barriers to fission product release, which is a concept similar to LWR multiple barriers; however, the approach does not elaborate on the containment functional performance issue as part of this barrier concept. The white papers do not address containment functional performance as an issue within the scope of this assessment.

SUMMARY OF CONCLUSIONS

Demonstrating the adequacy and sufficiency of the proposed DID approach requires a thorough understanding of, and proper implementation of, event selection (including a stronger deterministic element, as discussed in the assessment of the LBE white paper), safety classification and treatment of SSCs, source term, emergency planning, and scope and applicability of PRA methodologies (including risk metrics). Because of the current lack of detailed design information on these topics, the staff cannot make a more definitive determination on the adequacy and sufficiency of the details of the proposed DID approach. The NRC staff would make such a determination only after reviewing a specific detailed NGNP design, for example, in a topical report that it could review before it receives a license application.

3.2.2 DID Outcome Objective 2—Appropriate Plant Capability

The DID white paper explains that plant capability DID reflects the decisions made by the designer to incorporate DID into the functional capability of the physical plant. These decisions include the use of multiple barriers, diverse and redundant means to perform safety functions to protect the barriers, conservative design approaches for the barriers and SSCs, safety margins, siting, and other physical and tangible elements of the design that use multiple lines of defense to protect the public. The decision making is systematically evaluated in a risk-informed manner by using the PRA and a parallel set of deterministic evaluations. The PRA is based on plant design, including the specification of the capabilities of the plant's SSCs.

The results of the PRA depend on the safety margin and reliability of each SSC modeled in the PRA. The reliability of the SSCs responsible for the plant capability DID is to be ensured by their design and the elements of the programmatic DID. The PRA and the parallel deterministic evaluations include, as part of the modeling and quantification of the scenarios, models of the plant's capabilities and should explain how the plant is operated and maintained under the programmatic controls. Information from the PRA and the deterministic evaluations is used to support the design, to provide input to the formulation of process requirements, and to provide information to evaluate the adequacy and sufficiency of the DID strategies. The PRA also provides critical input for the identification and evaluation of the uncertainties that the plant capability and programmatic DID elements address.

STAFF ASSESSMENT

The plant capability DID is based on a number of important DID principles. The framework appears to be logical; however, several items of the framework defer substantial discussion to the license application stage, which limits the staff's ability to assess the proposal. For example, the NGNP safety design claims to include multiple robust barriers to radioactive material release. However, addressing the independence of barrier concept and challenges to barrier integrity cannot be done until the licensee submits the PRA and other detailed design information to support the license application. Preapplication interactions with a prospective licensee can provide additional information on the approach for each of the plant's capability DID principles to ensure their proper consideration during the development of the design.

The staff notes that all three major components of the approach described in the DID white paper (i.e., plant capability DID, programmatic DID, and risk-informed evaluation) have specific roles to play in addressing uncertainties. The DID white paper and related documents claim that uncertainties in the definition of LBEs and the safety classification of SSCs are explicitly taken into account in defining the plant capability DID. However, because of the lack of more information on the specific NGNP design, it is not possible to determine how the uncertainties are accounted for. This topic is addressed in the LBE white paper discussion above.

The staff also notes that the definition of adequate plant capability is highly dependent on conservative LBE selection. The staff believes that DOE/INL should address the conclusions associated with that topic to ensure adequate DID.

SUMMARY OF CONCLUSIONS

The staff believes that the definition of plant capability DID appears adequate. However, the staff cannot conclude that this definition will yield an acceptable outcome until the applicant provides detailed design information for NRC review.

3.2.3 DID Outcome Objective 3—Appropriate Programmatic Capability

The DID white paper explains that programmatic DID reflects the programmatic actions for designing, constructing, operating, testing, maintaining, and inspecting the plant. These programmatic actions allow the licensee to provide a greater degree of assurance that it can maintain the DID factored into the plant capabilities during the design stage throughout the life of the plant. The DOE/INL approach to programmatic DID includes the application of conservative safety margins and deterministic elements in the definition of the F-C curve, the selection of LBEs, the safety classification of SSCs, and the formulation of special treatment requirements for SSCs.

As in the case of plant capability DID, the decision making for programmatic DID is systematically evaluated in a risk-informed manner by using PRA and a parallel set of deterministic evaluations. As part of the modeling and quantification of the scenarios, the PRA and the parallel set of deterministic evaluations explain how the plant is operated and maintained under the programmatic controls. The PRA also provides critical input to the identification and evaluation of the uncertainties that are addressed in the programmatic DID elements.

STAFF ASSESSMENT

The scope of topics covered by the programmatic DID element are reasonable except for measures necessary to address catastrophic events, as discussed below. The efficacy of the programs is indeterminate at this time because of the lack of specific information about those programs. Therefore, no conclusions can be made until specific programs are proposed by a license applicant and reviewed by the NRC staff. For example, some NSR SSCs will have special treatment to provide DID. However, as discussed in the SSC white paper assessment below, it is not clear how the plant will identify and apply the appropriate treatment for this purpose. In addition, quantifying the effects of treatment in a PRA is difficult. The staff believes that conservative engineering judgment is necessary to ensure the adequacy of treatment of SSCs that provide DID.

The staff also believes that the scope of programmatic DID topics does not adequately consider programmatic elements for managing catastrophic events. Especially in regard to the related recommendations from the NRC's near-term review of insights from the Fukushima Dai-ichi accident (e.g., Recommendation 8),[10] the staff believes that DOE/INL should give broader consideration to how the plant will apply such programmatic DID measures as emergency operating procedures, severe accident management guidelines, and extensive damage mitigation guidelines to limit the progression and consequences of hypothetical catastrophic events (i.e. severe accident or security BDBEs).

[10] See the report titled, "Recommendations for Enhancing Reactor Safety in the 21st Century, NRC Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," issued July 2011 (ML111861807).

Hypothetical events resulting in major damage to both the primary loop and the reactor building may be among the catastrophic events that are deterministically selected and mechanistically analyzed in this DID context for modular HTGRs. RAI ARP DD-6 suggests that events that lead to massive ingress of air or oxygen[11] into the primary system may require programmatic DID coping measures and mitigation strategies aimed at terminating air/oxygen ingress into the damaged reactor building and reactor system. By acting to limit the amount of oxygen that enters the reactor building and subsequently the primary system, the goal of such measures would be to prevent or limit the radioactive releases that could otherwise result from the exothermic oxidation of structural and core graphite and the eventual heat up and oxidation-induced failure of TRISO-coated fuel particles. The time available for implementation was one of the factors considered in the evaluation of such DID coping measures and mitigation strategies.

The DOE/INL response to RAI ARP DD-6 dated September 15, 2010, cites the DID white paper's listing of several potentially relevant programmatic DID elements, including emergency operating procedures and severe accident management guidelines. The response further notes, as stated in Section 3.2.3 of the DID white paper, that DOE/INL will develop such strategies in the context of the NGNP approaches for using PRA, selecting LBEs, and classifying and treating SSCs. However, the staff notes that the LBE white paper does not address catastrophic events appropriate to this DID element. The regulatory context for such considerations should become more clear as guidance and regulations evolve in response to the Fukushima task force's recommendations and other relevant insights.

SUMMARY OF CONCLUSIONS

The staff believes that DEO/INL should strengthen programmatic DID to address the mitigation of hypothetical events.

3.2.4 DID Outcome Objective 4—Acceptable Balance of Deterministic and Probabilistic Criteria

The DID white paper states that the risk-informed evaluation combines plant capability DID and programmatic DID in a manner that provides an acceptable balance between the deterministic and the probabilistic safety evaluations that will be performed to support the NGNP licensing application. The risk-informed evaluation of DID, which is partly based on a review of the PRA results, identifies and evaluates the strategies (including the role of SSCs) for preventing and mitigating accidents. Prevention and mitigation are defined in regard to limiting the release of significant amounts of radioactive material as a result of event sequences selected for a given design. Finally, the intent of the balanced use of deterministic and probabilistic evaluations is to provide a logical process to establish the adequacy and sufficiency of DID.

STAFF ASSESSMENT

The DID white paper and related submittals describe the intent to integrate PRA into the NGNP design process at an early stage. Integrating PRA into the design process may be interpreted as using PRA to perform, for example, LBE selection and SSC classification. Alternatively, it may mean traditional deterministic analysis for this purpose, complemented, as appropriate, by

11 NUREG/CR-6944, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)," Volume 6, "Process Heat and Hydrogen Co-Generation PIRTs," issued March 2008, states that plumes of cold oxygen that hug the ground pose potentially significant hazards to the reactor plant from events at collocated process heat user facilities.

PRA (i.e., the use of PRA to refine the otherwise deterministically selected LBEs based on a more rigorous approach to SSC reliability). The applicant must ensure the adequacy and quality of the PRA if it is to rely on such an assessment for LBE selection and SSC classification. Section 3.11 of the PRA white paper discusses the approach for achieving technical adequacy in the NGNP PRA. DOE/INL indicates that technical adequacy in the NGNP PRA will be achieved by satisfying appropriate standards and by undergoing a peer review. This general approach is consistent with the approach applied to LWR PRAs that support the licensing of LWRs under 10 CFR Part 52. DOE/INL acknowledges that current standards for LWR PRAs are not appropriate for use in developing and reviewing the NGNP PRA. Section 3.11 of the PRA white paper discusses the main areas of LWR standards that must be revised to develop a standard for developing a PRA for the NGNP. However, even if an appropriate standard is available and even if a peer review demonstrates that the NGNP PRA meets this standard, the NGNP PRA will likely contain a large amount of uncertainty because of issues, such as the inclusion of many assumptions and the lack of operational data. Consequently, consideration of some deterministically selected bounding events for LBEs and the assurance of adequate safety margins would be prudent.

The DID white paper states that the derivation of the F-C curve considered DID. Specifically, dose criteria embodied in the TLRC and the F-C curve are lower than the surrogate dose criteria associated with the NRC's safety goal QHO for prompt fatality risk. The DID white paper further states that experience has shown that designs that meet the TLRC and that include appropriate design and analysis margins (including DID) also meet the safety goal often by orders of magnitude.

As stated above in its assessment of the LBE Outcome Objective 3, the staff believes that BDBEs should include event sequences with mean frequencies down to $5x10^{-7}$ per plant-year, which provides a reasonable cutoff for assessing whether the NGNP meets the NRC safety goals. The staff believes that this is a Commission policy issue.

SUMMARY OF CONCLUSIONS

As discussed above, the staff believes that DOE/INL should incorporate addition deterministic elements in the LBE white paper assessment to strengthen the proposed approach.

<u>3.2.5 DID Outcome Objective 5—Adequacy of Information to Be Provided</u>

The DID white paper states that the following information should be sufficient to allow the NRC to determine the adequacy of the DID provisions:

- definition of DID that is appropriate for modular HTGRs
- the roles of each barrier to radioactive material retention for each significant inventory of radionuclides in providing the plant capability DID
- explanation of how the reliability, capability, and independence of each barrier are defined and evaluated as they relate to the plant capability DID
- explanation of how the safety functions are defined and how they support the integrity of each barrier in providing the plant capability DID

- the roles of diverse combinations of inherent and passive design features, SSCs, and active engineered systems that perform the safety functions as part of the plant capability DID

- explanation of how the reliability, capability, and independence of each SSC that provides a safety function are defined and evaluated as they relate to the plant capability DID

- explanation of how the principles of design margins, redundancy, diversity, and independence have been applied to provide the plant capability DID

- an appropriate definition of accident prevention and mitigation and a means for evaluating the affect that DID strategies have on maintaining acceptable risk levels

- the roles and effectiveness of specific barriers and SSCs in the prevention and mitigation of accidents

- the role that design safety margins in the applied codes and standards play in providing a robust design with DID

- explanation of how compensating measures and other aspects of programmatic DID are applied to address uncertainties

- explanation of how a set of deterministic principles derived from the regulatory foundation is applied in the risk-informed evaluation of the adequacy and sufficiency of DID for modular HTGRs

- explanation of how the elements of the safety design approach are used to evaluate plant design features in an integrated manner as part of an overall risk management approach in which risk analysis is used to broadly improve operational and engineering decisions by identifying and maximizing opportunities to reduce risk

STAFF ASSESSMENT

The staff considers the scope of topics covered in DID Outcome Objective 5 reasonable. However, because of the current lack of NGNP design-specific information on these topics, the staff cannot make a determination on the adequacy and sufficiency of the details that the proposed topics intend to cover. The NRC staff would make such a determination only after reviewing a specific detailed NGNP design, for example, in a topical report that it could review before it receives a license application. Such a review would determine whether the proposed approach demonstrates compliance with regulatory requirements.

SUMMARY OF CONCLUSIONS

Although the scope of topics that this outcome objective addresses is reasonable, additional design information is necessary to assess the adequacy of the implementation of these topics.

3.3 Safety Classification of Structures, Systems, and Components

The SSC white paper describes a proposed approach to classify SSCs according to their safety significance with the intent of focusing resources on the most significant SSCs.

3.3.1 SSC Outcome Objective 1—Acceptable Approach

The SSC white paper asks the NRC to agree with its first outcome objective and states the objective as follows:

> The NGNP approach to risk-informed safety classification and special treatment that blends the strengths of probabilistic and deterministic methods is acceptable.

The SSC white paper states that the NGNP fuel will be classified as SR because it is the most important barrier to the release of radionuclides to the environment. The paper also states that the SSCs that ensure safe shutdown of the NGNP reactor will also be classified as SR. Additional SSCs designated as SR will include (1) SSCs that the plant relies on to perform “required safety functions” to prevent or mitigate the consequences of DBEs for compliance of the DBE dose consequence with the TLRC for DBEs and (2) SSCs that the plant relies on to perform required safety functions to prevent the frequency of BDBEs with consequences greater than the dose limits in 10 CFR 52.79 and 10 CFR 50.34 from increasing into the DBE region. The NGNP designer will select the specific SR SSCs to meet these two criteria.

The SSC white paper states that the selection of SR SSCs will begin by identifying the required safety functions (i.e., “required safety functions”) for each DBE to meet the DBE TLRC, which are derived from the siting dose consequence reference values in 10 CFR Part 100 as described in the LBE white paper above. To meet the DBE TLRC, required safety functions will generally include reactor shutdown, removal of core heat and containment of radionuclides, and control of any chemical attack (e.g., fuel oxidation and graphite oxidation). For each DBE, the NGNP SSCs that are provided to perform each required safety function are reviewed to determine which SSCs have sufficient capability and reliability. A determination of the SSCs that can perform each required safety function for each DBE allows for the selection and classification of a set of SSCs as SR that can provide the required safety functions with the required capability and reliability for all DBEs. The SSCs that the designer selects as SR should be generally passive rather than active (e.g., SSCs involved in passive accident core heat removal and passive radionuclide barriers, such as the fuel). These SR SSCs should be provided with a full range of special treatments to ensure that the SSCs have the required capability and reliability to perform their required safety functions. The NGNP DBA sequences are defined as the DBE accident sequences with the assumption that only the SR SSCs are available (i.e., credited) for prevention and mitigation.

The SSC white paper states that the process for selecting SR SSCs continues by analyzing each BDBE with all the plant SSCs modeled in the plant PRA. The reliability and availability of the SSCs are assumed to be consistent with those assumed in the NGNP PRA. If a BDBE meets the BDBE dose consequences acceptance criteria but has a dose consequence that is above the DBE dose consequence acceptance criteria (i.e., the reference values in 10 CFR Part 100), the plant must provide assurance that the BDBE frequency will remain below the lower frequency cutoff of the DBE region. Any BDBE sequence that has a dose consequence that is higher than the dose limits in 10 CFR 52.79[12] and 10 CFR 50.34 must be reviewed to determine the safety functions necessary to prevent the frequency of the BDBE from increasing into the DBE region. The SSCs that are available and sufficient to perform the required functions to keep the frequency of the BDBE from increasing into the DBE region are identified. The SSCs that are selected to perform these safety functions are classified as SR.

The SSC white paper also defines a category of SSCs that are NSR with special treatment (NSRST). These SSCs are defined as SSCs that the plant relies on to perform safety functions to (1) mitigate the consequences of AEs[13] to comply with the TLRC and (2) prevent the frequency of DBEs with consequences greater than the offsite dose limits in 10 CFR Part 20 from increasing into the AE region.

DOE/INL notes that special treatments can enhance the capability and the reliability of NSRST SSCs, thereby shifting the locations of the LBEs on the frequency versus dose chart (i.e., reduce the frequency of the LBE or reduce the dose consequences of the LBE or both). This approach will ensure that any NSR SSCs that the plant relies on during an AE to meet the offsite annual dose criteria in 10 CFR Part 20 are subject to adequate special treatment. Any SSCs that are classified as SR in accordance with 10 CFR 50.2, "Definitions," and that mitigate or prevent AEs will have special treatment specific to both the AEs and DBEs in which they are involved. Therefore, the requirement that SR SSCs must be capable of mitigating AEs will be satisfied. NSRST SSCs that the plant relies on during AEs will be subject to the requirements in 10 CFR Part 20, whereas SR SSCs that the plant relies on during DBEs will be subject to the requirements in 10 CFR 52.79.

STAFF ASSESSMENT

The SSC white paper describes an approach for classifying and treating SSCs that includes specific criteria for specifying SR SSCs, assesses SSCs against the criteria in a risk-informed manner, and provides DID through the identification of NSR SSCs that will receive treatment to ensure that their capability and reliability are consistent with the assumptions in the PRA. The NRC staff generally supports a risk-informed approach for classifying SSCs and for determining the appropriate levels of treatment for the SSCs under different classifications. Such an approach is currently available to operating reactor licensees through voluntary application of the requirements in 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," although no licensee has taken advantage of this rule as of the date of this report. In addition, the SRM dated May 11, 2011, for SECY-11-0024, "Use of Risk Insights To Enhance the Safety Focus of Small Modular Reactor Reviews," dated February 18, 2011, directs the NRC staff to complete a feasibility study for using risk information in categorizing SSCs as SR or NSR for the design-specific small modular reactor review plans. The staff's response to the SRM discusses long-term activities that the

[12] Like the other papers discussed in this assessment, the SSC white paper cites 10 CFR 50.34. The proper regulatory citation is 10 CFR 52.79 for a COL application; however, as noted previously, the dose requirements of these two regulations are identical.

[13] In a meeting with the staff on May 16, 2012, DOE/INL stated that the NGNP event sequences classified as AOOs relate to the types of events that are normally described in Chapters 11 and 12 of the final safety analysis report for LWRs—not those that require (1) mitigation to prevent a challenge to the integrity of the reactor coolant pressure boundary, (2) the reactor to be shut down and maintained in a safe shutdown condition, or (3) mitigation to prevent potential offsite exposures from exceeding the applicable guideline exposures in 10 CFR 50.34(a)(1) or 10 CFR 100.11—and that should be evaluated realistically on an expected mean basis versus the offsite dose limit in 10 CFR Part 20.

NRC can implement to enhance the use of risk insights in categorizing SSCs for small modular reactors. The NRC's long-term activities may include a pilot study to test the concepts of developing a new risk-informed regulatory framework, as described in SECY-11-0024 and in NUREG-1860, Volume 1, and NUREG-1860, Volume 2, "Appendices A through L," issued December 2007.

SUMMARY OF CONCLUSIONS

Although the staff generally supports the NGNP's use of a risk-informed approach for classifying SSCs, it requires more detailed design information to conclude that it would find any proposed risk-informed approach to the classification and treatment of SSCs acceptable. Information on the NGNP design or on details of the implementation of the approach is currently insufficient to support a conclusion that it will yield an acceptable outcome. However, the staff acknowledges that a risk-informed, performance-based approach can result in an acceptable design if it is properly implemented. For example, the SSC white paper does not explain how the NGNP will implement the Commission's policy on the regulatory treatment of non-safety systems (RTNSSs).

<u>3.3.2 SSC Outcome Objective 2—Acceptable Classification Categories</u>

The white paper states the second SSC outcome objective as follows:

> The NGNP risk-informed safety classification categories and the bases for SSC classification within each category are acceptable.

The SSC white paper states that DOE/INL has adopted the COL application process in 10 CFR Part 52 that the Licensing Strategy Report recommends as the foundation for the NGNP licensing strategy. DOE/INL has proposed to classify the NGNP SSCs into one of two basic safety categories. These safety categories are SR and NSR.

SSCs that are classified as SR must be given the full scope of special treatment. Special treatment would include design requirements to ensure that the SSC can perform its safety function during the most severe conditions of the LBE and that the SSC includes safety margin and design conservatism, technical specification requirements, inservice inspection requirements, inservice testing requirements, quality assurance requirements, and other regulatory requirements.

SSCs are classified as SR from the set of SSCs that are necessary and sufficient to prevent or mitigate DBEs from exceeding the TLRC for DBEs. Additionally, any LBE in the BDBE region that has a dose consequence above dose reference values in 10 CFR Part 100 is reviewed to determine which safety functions and associated SSCs would be necessary to prevent the frequency of the BDBE from increasing into the DBE region. For any such BDBE, a sufficient set of SSCs that can perform these safety functions is identified. The SSCs in this set are classified as SR if not previously selected to be SR. DOE/INL states that it has applied deterministic engineering judgment by designating the NGNP fuel as SR and by designating those SSCs necessary to safely shut down and cool down the NGNP reactor as SR.

The SSC white paper states that SSCs classified as NSR must be provided with special treatment commensurate with their safety significance. NSR SSCs that meet the criteria discussed in Section 3.3.1 above are designated as NSRST. NSRST SSCs are to be provided with less than the full scope of special treatment. Special treatment must be commensurate with the safety importance and the required capability and reliability of the SSC modeled in the NGNP PRA. An NSR SSC may have no special treatment if it is not important for the prevention or mitigation of any LBEs.

STAFF ASSESSMENT

Categories for Classification

The SSC safety classifications for the NGNP are SR, NSR without special treatment, and NSRST. SSCs that are classified as SR must be given the full scope of special treatments. Some SSCs that do not meet the criteria for an SR classification may serve in a backup safety role and will therefore receive special treatments that are commensurate with their safety significance. An NSR SSC may have no special treatments if it is not important for the prevention or mitigation of any LBEs.

Consistent with 10 CFR 50.2 and 10 CFR 50.49(b), SR SSCs would be available to prevent and mitigate the effects of DBEs and DBAs. NSRST SSCs would prevent and mitigate certain initiating events considered AOOs for LWRs (i.e., a LOOP) when such events are in the AE region. A LOOP would be in the DBE/DBA region if it occurred when no NSRST SSCs were available; SR SSCs would be available for such an event. Therefore, the staff concludes that these safety classifications are reasonable and that they are consistent with current regulations.

The staff concludes that the three-category approach proposed for the NGNP is reasonable. Similar approaches have been used for the certification of passive reactor designs and by licensing applicants that have referenced those designs. In addition, the NRC expects to use a graded approach for its review of applications for certification and licensing of integral pressurized-water reactor designs.

The staff believes that (to the extent that the Commission may require the inclusion of events in the BDBE frequency region in the accident source term for evaluating site suitability (i.e., 10 CFR Part 100)) the SSCs that the plant relies on to perform required safety functions to prevent or mitigate the consequences of BDBEs, in accordance with 10 CFR Part 100, should be SR. As noted by the SSC white paper, NGNP SSCs will need to conform to ASME Code requirements that have not yet been developed and to existing ASME Code requirements that are applicable and adaptable to the NGNP.

SUMMARY OF CONCLUSIONS

The staff believes that the DOE/INL approach to SSC classification is generally reasonable with the following exceptions:

- The classification categories should clearly address all fission product release barriers, including the HPB and the reactor building.
- SSCs that mitigate events in the BDBE region that would be designated as DBAs may need to be SR.
- SSCs will need to conform to ASME Code requirements that have not yet been developed.

3.3.3 SSC Outcome Objective 3—Special Treatment of Safety-related SSCs

The white paper states the third SSC outcome objective as follows:

> The special treatment for the SR category of classification is commensurate with ensuring the SSCs' ability to perform their safety function for DBEs and high consequence BDBEs.

The SSC white paper proposes that the fuel and SSCs that ensure safe shutdown and cooldown be classified as SR. The paper also proposes that (1) SSCs that the plant relies on to perform required safety functions to prevent or mitigate the consequences of DBEs to comply with the TLRC and (2) SSCs that the plant relies on to perform required safety functions to prevent the frequency of BDBEs with consequences greater than the dose limits in 10 CFR 50.34 from increasing into the DBE region also be classified as SR. SSCs that are classified as SR are to be given the full scope of special treatments. As noted earlier, special treatments would include design requirements to ensure that the SSC can perform its safety function during the most severe conditions of the LBE and that the SSC includes safety margin and design conservatism, technical specification requirements, inservice inspection requirements, inservice testing requirements, quality assurance requirements, and other regulatory requirements. The determination of specific treatment requirements will be made as part of the deterministic safety analysis.

STAFF ASSESSMENT

The staff believes that the establishment of the full scope of special treatment requirements for the SSCs classified as SR should be done to ensure that the SSCs have the capability and capacity to perform their required safety functions. For this reason, SR SSCs will be subject to SR design conditions (i.e., temperatures, stresses, and heat loads) that they must meet for each DBE. The design, fabrication, and operational requirements for the capability and reliability of SR SSCs are dependent on LBE sequences for which these SSCs must perform and would be reviewed and assessed as part of the NGNP licensing application review. For this reason, the NGNP design should include a number of SR SSCs with innovative design elements, such as the fuel, passive decay heat removal, and safe shutdown components. As such, these features should involve new and innovative special treatment requirements (e.g., a fuel qualification program and reactor cavity coolant system testing). The NRC staff would evaluate the special treatment requirements of these SR SSCs as part of the NGNP license application review.

The SSC white paper's description of special treatment states that the reliability and capability of SR SSCs are derived from the frequency and consequences of the LBEs that those SSCs mitigate. This description is incomplete because frequency and consequences do not account for the SSC's ability to function adequately within the environmental conditions under which it may be subjected in the event of an accident, for its ability to withstand a seismic event, or for other performance attributes unrelated to the frequency and consequences of the event. Although Table 1 in the SSC white paper appears to recognize these additional factors, the text of the SSC white paper is incomplete because it implies a more limited set of special requirements. The DOE/INL response to RAI SSC-9 proposes a revision to the SSC white paper that states that the LBE definition will define the loading conditions and environmental conditions, including any harsh environments under which the SSC must fulfill its safety function(s); this revision will facilitate a full definition of the special treatment requirements. The staff reviewed the proposed revision to the white paper and concludes that it resolves the concerns in RAI SSC-9.

The NGNP approach involves the use of both the PRA and the plant design safety analyses to identify the capability and reliability requirements that an SSC must meet through special treatment. The approach acknowledges that aspects of LBEs other than event frequency and event consequences (e.g., performance in an adverse environment) must be addressed through special treatment. The SSC white paper lists the areas of special treatment that will apply to SR SSCs, and the staff concludes that this list is complete. The SSC white paper also notes that treatment requirements established through codes and standards that have not yet been developed will be applied. For these reasons, the staff concludes that the third outcome objective (i.e., that the special treatment for the SR category of classification is commensurate with ensuring that the SSCs can perform their safety functions for DBEs and high consequence BDBEs) is reasonable. However, because of the current lack of design detail, the staff notes that its conclusion does not address the acceptability of any specific NGNP design feature or SSC performance or treatment. The NRC staff will make determinations on the acceptability of the specific NGNP design in its review of a future license application that provides the necessary design detail. As discussed above, the staff also notes that the scope of SSCs classified as SR is incomplete. This incomplete scope could therefore affect the scope of special treatment attributes addressed in this category.

SUMMARY OF CONCLUSIONS

The staff concludes that the special treatment described for SR SSCs is reasonable. However, the staff cannot make a determination on whether the treatment of those SSCs meets relevant regulatory requirements until it receives a complete license application.

3.3.4 SSC Outcome Objective 4—Special Treatment of NSRST SSCs

The SSC white paper states the last outcome objective as follows:

> The special treatment for the NSRST category is commensurate with ensuring the SSCs' ability to perform their safety function of providing significant DID.

The SSC white paper proposes that special treatment requirements for NSR SSCs should be commensurate with the requirements necessary to enable the SSCs to perform with the capability and reliability requirements during AEs. NSR SSCs do not have as many special treatment requirements as compared to those for SR SSCs. The paper states that the reduced special treatment requirements are reasonable because the required reliability of NSR SSCs is lower than that of SR SSCs.

The SSC white paper defines NSRST SSCs as SSCs that the plant relies on to perform safety functions to mitigate the consequences of AEs to comply with the TLRC or as SSCs that the plant relies on to perform safety functions to prevent the frequency of DBEs with consequences greater than the offsite dose limits in 10 CFR Part 20 from increasing into the AE region.

The SSC white paper proposes that the NGNP PRA be used to establish the required capabilities and reliabilities of NSR SSCs similar to those of the SR SSCs necessary to meet the TLRC. The establishment of special treatment requirements for the SR and NSR SSCs will be done in a manner that ensures that SSC capability and reliability are consistent with the capability and reliability modeled in the PRA. The special treatment requirements for each NSR SSC should be based on the specific LBE sequences in which the NSR SSCs are modeled and credited to perform their safety functions in the LBE sequences. Special treatments of the NSRST SSCs are used to adjust the ordinates of the AE LBEs on the F-C chart and are used to reduce the uncertainties associated with the AE LBE sequence frequencies and consequences. If the dose consequence ordinate of the AE sequence or the frequency ordinate of the AE sequence must be reduced (to make it a DBE sequence), the special treatment requirements of one or more NSR SSCs are increased as necessary.

The DOE/INL response to RAI SSC-1 states that the NGNP approach for special treatment is consistent with the criteria in 10 CFR 50.69(d) for deriving special treatment requirements for SR and NSRST SSCs. Consistent with the principles in 10 CFR 50.69, the guiding principle for the NGNP is to establish a necessary and sufficient set of special treatment requirements to ensure that each classified SSC has sufficient capability and reliability to perform its required safety function. Although a different process is used to establish "the safety significance of SSCs," DOE/INL has stated that the principles are the same.

STAFF ASSESSMENT

RAI SSC-13 notes that the white paper does not clearly state that SSCs described by this section (i.e., NSRST SSCs) will meet regulatory requirements, such as 10 CFR 50.55a(a)(1), which requires that SSCs be "designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed." In RAI SSC-13, the staff asked DOE/INL to clarify its intent for these SSCs.

The DOE/INL response to RAI SSC-13 states that DOE/INL expects to conform to future code requirements developed for HTGR plants, including HTGR codes developed and incorporated into NRC regulations, such as 10 CFR 50.55a, "Codes and Standards." Code requirements associated with design, fabrication, erection, inspection, testing, operation, and maintenance would be applicable to those SSCs that require special treatment to the extent necessary to provide reasonable assurance that such SSCs, as determined through a plant-specific PRA, will perform their intended safety functions with sufficient capability and reliability to meet the TLRC. The staff concludes that this information provides adequate clarification.

The SSC white paper indicates that the plant will adjust special treatment as necessary (presumably during plant operation) to maintain the reliability assumptions in the PRA. This adjustment will involve a performance monitoring program that was not described in the SSC white paper. However, the staff notes that "operational performance monitoring" is an area of special treatment that the plant will provide to NSR SSCs. The concern with this feature of the approach is that the relationship between the level of a particular type of treatment (e.g., quality assurance or maintenance) and the reliability parameters assumed in the PRA is not well defined. The staff recognized this issue in preparation for its review of large passive LWR applications. Accordingly, neither the reliability assurance program nor the program for RTNSSs claims to directly enforce the assumptions in the PRA.

The proposed use of the NSRST classification increases the scope of review for a prospective license application and introduces additional regulatory uncertainty. The staff notes that considerable challenges were associated with the development of 10 CFR 50.69. The regulations at 10 CFR 50.69 provide alternative risk-informed classification and special treatment requirements and were informed by many years operating and licensing experience with over 100 operating LWRs. Because the NRC has far less relevant experience with HTGRs, the staff anticipates that there could be additional uncertainty associated with the NGNP NSRST classification. The staff also notes that, as of the date of this assessment, no licensees have implemented 10 CFR 50.69.

Another regulation that is relevant to the proposed NSRST classification is 10 CFR Part 21, "Reporting of Defects and Noncompliance." This regulation requires entities that construct, own, operate, or supply components to a nuclear power plant to immediately inform the NRC of defects in the facilities, "which could create a substantial safety hazard." The regulations define attributes of "basic components" that are subject to these reporting requirements. For currently operating reactors, components that mitigate fuel safety limits for AOOs are classified as SR and clearly fall within the scope of the requirements in 10 CFR Part 21. The staff has yet to apply these requirements to SSCs as defined in 10 CFR 50.69 (i.e., NSR SSCs that perform significant safety functions).

SUMMARY OF CONCLUSIONS

The staff believes that the DOE/INL NSRST classification is reasonable; however, it notes the following challenges to SSC Outcome Objective 4:

- The ability to adjust special treatment and to reflect those adjustments in the PRA has not been established.

- Relevant regulatory experience with HTGRs and in the use of 10 CFR 50.69 is generally lacking.

- DOE/INL has not addressed the applicability of 10 CFR Part 21 and other regulatory requirements.

4. SUMMARY

This section provides a summary of the staff's key conclusions. Readers are referred to the text above for the bases of these conclusions and for other observations that may be relevant to future NGNP licensing.

The staff has identified two cross-cutting issues that substantially affect many of the conclusions described in this preliminary assessment. First, sufficient design detail is often not available to allow the staff to interpret or understand how the proposed framework will be implemented. Second, the staff believes that DOE/INL should strengthen the proposed approach through increased use of deterministic elements. The assessment above discusses how conservative deterministic selection of bounding events can clearly demonstrate compliance with regulatory requirements and improve DID by ensuring that plant equipment can mitigate a wide range of events.

For LBE selection, the staff summarizes its principal conclusions as follows:

- The approach to the categorization of LBEs is generally reasonable, although the staff believes that deterministic elements should have a stronger role. The NGNP license applicant and the NRC staff will need to agree on the use of deterministic engineering judgment complemented by NGNP design-specific PRA to deterministically select LBEs in the event categories that are consistent with the NGNP design and safety characteristics and the Commission's policy decisions.
- The NGNP applicant and NRC staff will need to agree on the event categories and the events necessary to demonstrate that the NGNP complies with the requirements in 10 CFR 52.79 and 10 CFR 50.34 and the plant capability DID to retain fission products.

- The limits on the event frequency ranges that define the AE, DBE, and BDBE categories are reasonable.
- The determination on whether PRA initiating events and event sequences below $1x10^{-8}$ per plant-year can be dropped from consideration for establishing emergency planning requirements is a Commission policy decision.
- The proposed approach to events, failures, and natural phenomena is generally reasonable. Although the DOE/INL risk-informed, performance-based approach has apparently addressed some of the extended licensing-basis considerations from the Fukushima Dai-ichi earthquake and tsunami, the staff may require consideration of additional external events (e.g., natural phenomena) and the associated event sequences for NGNP licensing as it continues to acquire more information.
- The approach for the development of the NGNP DBAs is reasonable with some modifications, and the use of a conservative calculation for the DBA mechanistic source term is appropriate.

For DID, the staff highlights the following issues:

- DID is a topic that potentially will require Commission deliberation.
- DID is closely linked to other potential technical/policy issues (e.g., mechanistic source term, containment functional performance, and emergency planning).

For SSC classification and treatment, the staff has identified the following key issues:

- Key fission product barriers should be SR.
- The proposed approach does not address the role of RTNSSs in DID.

Based on its review of the proposed risk-informed, performance-based licensing approach, the staff believes that the approach proposed in the white papers (discussed above) may not be consistent with the intent of Option 2 of the Licensing Strategy Report. The staff has identified a number of issues that challenge the effective implementation of a risk-informed, performance-based licensing framework for the NGNP. These issues may increase regulatory uncertainty for a future license application and, therefore, may increase the level of effort and time by the NRC to complete a review. Resolution of these issues during preapplication interactions with a prospective designer or licensee or both may reduce regulatory uncertainty for a subsequent licensing review. The staff reiterates that these conclusions and observations are preliminary because of the lack of detailed design information available. The NRC staff will be better able to determine the adequacy of the proposed risk-informed, performance-based framework by examining its implementation as more design information is made available.

APPENDIX A

NEXT GENERATION NUCLEAR PLANT LICENSING-BASIS EVENT SELECTION WHITE PAPER OUTCOME OBJECTIVES

The September 16, 2010, submittal discusses the outcome objectives outlined below.

(1) The structured process for selecting license-basis events (LBEs) is an acceptable approach for defining the LBEs.

<u>Next Generation Nuclear Plant (NGNP) Approach</u>. An acceptable approach starts with a deterministically selected initial event list. It then includes a design-specific PRA with the following elements: (1) an engineering analysis of the plant response to each initiating event using verified computer models, (2) deterministic success criteria, and (3) conservative deterministic safety analyses of design-basis accidents (DBAs) in Chapter 15 of the license application. Both the deterministic and probabilistic analysis will be supported by a comprehensive and systematic search for initiating events, including internal events and internal and external plant hazards that could occur during all operating and shutdown modes, and will cover the sources of radioactive material.

(2) LBEs cover a comprehensive spectrum of events from normal operation to rare, off-normal events.

The following three categories of LBEs apply:

(1) Anticipated events (AEs) encompass planned and anticipated events. 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.

(2) Design-basis events (DBEs) encompass unplanned, off-normal events not expected in the plant's lifetime, but that might occur in the lifetimes of a fleet of plants. The doses from DBEs are required to meet accident public dose requirements. DBEs are the basis for the design, construction, and operation of the structures, systems, and components (SSCs) during accidents.

(3) Beyond-design-basis events (BDBEs) are rare, off-normal events of lower frequency than DBEs. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public.

The LBEs in all three categories will be evaluated individually to support the tasks involved in assessing the performance of SSCs in regard to safety functions in response to initiating events and collectively to demonstrate that the integrated risk of a multi-module plant design meets the safety goals of the U.S. Nuclear Regulatory Commission (NRC).

<u>NGNP Approach</u>. The three categories of LBEs and their purposes adhere to existing NRC regulations and policies. The LBEs include the AEs, DBEs, and BDBEs used in NRC regulatory policies and guidance. DBEs, AEs, and BDBEs are selected through the use of the PRA and are based on a realistic response of the entire plant. This foundation is necessary for understanding the plant's safety functions and the SSCs available to perform these functions. This understanding leads to the safety

classification of SSCs, which is the subject of another paper. Once the SSC safety classification is known, the deterministically selected initial event list is risk-informed by DBAs that are derived from the DBEs by demonstrating success pathways for the DBAs relying solely on their response of safety-related SSCs, as in the conventional regulatory practice.

(3) The frequencies of LBEs are expressed in units of events per plant-year, where a plant is defined as a collection of reactor modules that have certain shared systems. The limits on the frequency ranges for the LBE categories are as follows:

- AEs—event sequences with mean frequencies greater than $1x10^{-2}$ per plant-year
- DBEs—event sequences with mean frequencies less than $1x10^{-2}$ per plant-year and greater than $1x10^{-4}$ per plant-year
- BDBEs—event sequences with mean frequencies less than $1x10^{-4}$ per plant-year and greater than $5x10^{-7}$ per plant-year

NGNP Approach. DOE/INL proposes a lower frequency limit of $1x10^{-2}$ per plant-year for AEs, a lower frequency range for event sequences of $1x10^{-4}$ per plant-year for DBEs that meets the NRC safety goals and that is consistent with the regulatory practice for light-water reactors, and a lower limit of $5x10^{-7}$ per plant-year for BDBEs.

To account for multi-module concepts, DOE/INL proposes that the frequency be stated on a per-plant-year basis. For example, the frequency for a $1x10^{-4}$ per plant-year event that affects only one module of an eight reactor module facility is $1.25x10^{-5}$ per plant-year. For events such earthquakes that affect more than one of, and up to all, eight reactor modules, the frequency is $1x10^{-4}$ per plant-year, and the consequences will account for all eight reactor modules. By setting the lower bound of the DBE region at $1x10^{-4}$ per plant-year, the NGNP commits to design for all events whether these event affect one reactor module or all the reactor modules.

(4) Acceptable limits on the event sequence consequences and the analysis basis for the LBE categories are as follows:

- The limit for AEs, in accordance with 10 CFR Part 20, “Standards for Protection against Radiation,” is 100-millirem total effective dose equivalent (TEDE), mechanistically modeled and realistically calculated at the exclusion area boundary (EAB).
- The limit for DBEs, in accordance with 10 CFR 50.34, “Contents of Applications; Technical Information,” is 25-rem TEDE, mechanistically modeled and conservatively calculated at the EAB.
- The limit for BDBEs is based on NRC safety goal quantitative health objectives (QHOs) and is mechanistically and realistically calculated at 1 mile (1.6 kilometers (km)) and 10 miles (16 km) from the plant.

NGNP Approach. The acceptable public consequences have been derived from the existing regulations and policy in Section 2. In summary, the limits in 10 CFR Part 20 are applied to AEs, and the limits in 10 CFR 50.34 are applied to the DBEs and the DBAs. The safety goal QHOs are applied to all the LBEs in a cumulative manner. The bases for the analyses follow the conventional practice for each of the LBE categories and respective top-level regulatory criteria.

(5) The frequency below which events are not selected as LBEs is $5x10^{-7}$ per plant-year. The PRA examines events to $1x10^{-8}$ per plant-year to ensure that those just below this de minimus frequency do not occur.

NGNP Approach. BDBEs will meet the NRC safety goals at the prescribed distances from the plant. DOE/INL proposes $5x10^{-7}$ per plant-year because lower frequency events, by definition, meet the NRC safety goal QHO for acute individual risk of fatality.

(6) The kinds of events, failures, and natural phenomena that are evaluated include the following:

- multiple, dependent, and common-cause failures to the extent that these contribute to LBE frequencies
- events that affect more than one reactor module
- internal events (including transients and accidents) and internal and external plant hazards that occur in all operating and shutdown modes and that potentially challenge the plant's capability to satisfactorily retain any source of radioactive material

NGNP Approach. The PRA that supports the application will conduct a full-scope evaluation that will include all operating modes. A future NGNP white paper will discuss this topic in greater detail.

(7) The DBAs in Chapter 15 of the license application are derived from the DBEs by assuming that only SSCs classified as safety-related are available to mitigate the consequences. The public consequences of DBAs are based on mechanistic source terms and are conservatively calculated. The upper bound consequence of each DBA must meet the consequence limit in 10 CFR 50.34 at the EAB.

NGNP Approach. The DBAs will be derived from the DBEs by considering only the response of SSCs classified as safety-related. The consequences of DBAs will be based on mechanistic source terms and will be conservatively calculated. The upper bound consequence of each DBA will meet the consequence limit in 10 CFR 50.34 at the EAB.

(8) Uncertainty distributions are evaluated for the mean frequency and the mean consequence for each LBE. The mean frequency is used to determine whether the event sequence family is an AE, DBE, or BDBE. If the upper or lower bound on the LBE frequency straddles two or more regions, the LBE is compared to the consequence criteria for each region. The mean, lower, and upper bound consequences are explicitly compared to the consequence criteria in all applicable LBE regions. The upper bound for the DBE and DBA consequences must meet the dose limit in 10 CFR 50.34 at the EAB.

NGNP Approach. Uncertainty distributions will be evaluated for the mean frequency for each LBE. The mean frequency will be used to determine whether the event sequence family is an AE, DBE, or BDBE. If the upper or lower bound on the LBE frequency straddles two regions, the LBE will be compared to the consequence criteria for each region.

APPENDIX B

NEXT GENERATION NUCLEAR PLANT DEFENSE-IN-DEPTH WHITE PAPER OUTCOME OBJECTIVES

The December 9, 2009, submittal discusses the outcome objectives outlined below.

(1) The definition of defense in depth (DID) presented in Section 3 of this paper, which recognizes the three elements of a DID approach (i.e., plant capability DID, programmatic DID, and the risk-informed evaluation of DID), is consistent with available definitions in the regulatory foundation and is appropriate for the Next Generation Nuclear Plant license application.

(2) The plant capability DID element, which includes multiple independent and diverse barriers to radionuclide transport; the use of inherent features and passive and active structures, systems, and components (SSCs) necessary to perform the required safety functions; and conservative design strategies, is appropriate for the license application.

(3) The programmatic DID element represents an acceptable approach for incorporating DID principles into the definition of programs that will provide assurance that (1) the plant's capabilities to ensure safety and DID will have sufficient reliability and will be maintained throughout the lifetime of the plant and (2) compensatory actions adequately address uncertainties.

(4) The risk-informed evaluation of DID element provides an acceptable balance of deterministic and probabilistic assessments and evaluation criteria. Furthermore, this element includes an acceptable event sequence framework for the definition of accident prevention and mitigation and for the evaluation of the roles of design features and SSCs responsible for the prevention and mitigation of accidents to demonstrate the safety case. Finally, the balanced use of deterministic and probabilistic evaluations provides a logical process for establishing the adequacy and sufficiency of DID for modular high-temperature gas-cooled reactors (HTGRs).

(5) If the Next Generation Nuclear Plant license application applies the approach described in this paper, the NRC will have sufficient information to determine the adequacy of the DID provisions. This information will include the following items:

- definition of DID that is appropriate for modular HTGRs
- the roles of each barrier to radioactive material retention for each significant inventory of radionuclides in providing the plant capability DID
- explanation of how the reliability, capability, and independence of each barrier are defined and evaluated as they relate to the plant capability DID
- explanation of how the safety functions are defined and how they support the integrity of each barrier in providing the plant capability DID

- the roles of diverse combinations of inherent and passive design features, SSCs, and active engineered systems that perform the safety functions as part of the plant capability DID

- explanation of how the reliability, capability, and independence of each SSC that provides a safety function is defined and evaluated as it relates to the plant capability DID

- explanation of how the principles of design margins, redundancy, diversity, and independence have been applied to provide the plant capability DID

- an appropriate definition of accident prevention and mitigation and a means for evaluating the affect that DID strategies have on maintaining acceptable risk levels

- the roles and effectiveness of specific barriers and SSCs in the prevention and mitigation of accidents

- the role that design safety margins in the applied codes and standards play in providing a robust design with DID

- explanation of how compensating measures and other aspects of programmatic DID are applied to address uncertainties

- explanation of how a set of deterministic principles derived from the regulatory foundation is applied in the risk-informed evaluation of the adequacy and sufficiency of DID for modular HTGRs

- explanation of how the elements of the safety design approach are used to evaluate plant design features in an integrated manner as part of an overall risk management approach in which risk analysis is used to broadly improve operational and engineering decisions by identifying and maximizing opportunities to reduce risk

APPENDIX C

NEXT GENERATION NUCLEAR PLANT STRUCTURES, SYSTEMS, AND COMPONENTS SAFETY CLASSIFICATION AND TREATMENT WHITE PAPER OUTCOME OBJECTIVES

The September 21, 2010, submittal discusses the outcome objectives outlined below.

(1) The Next Generation Nuclear Plant approach to risk-informed safety classification and special treatment that blends the strengths of probabilistic and deterministic methods is acceptable.

(2) The Next Generation Nuclear Plant risk-informed safety classification categories and the bases for the classification of structures, systems, and components (SSCs) within each category are acceptable.

(3) The special treatment for the safety-related category of classification is commensurate with ensuring the ability of the SSCs to perform their safety functions for design-basis events and high consequence beyond-design-basis events.

(4) The special treatment for the nonsafety-related with special treatment category is commensurate with ensuring the ability of the SSCs to perform their safety function for providing significant defense in depth.

APPENDIX D

U.S. NUCLEAR REGULATORY COMMISSION PARTICIPANTS IN THE ASSESSMENT OF THE NEXT GENERATION NUCLEAR PLANT WHITE PAPERS ON THE DEFENSE-IN-DEPTH APPROACH, SELECTION OF LICENSING-BASIS EVENTS, AND SAFETY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

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