POLICY ISSUE
(Information)

February 7, 2016 SECY-16-0012

FOR: The Commissioners

FROM: Victor M. McCree
Executive Director for Operations

SUBJECT: ACCIDENT SOURCE TERMS AND SITING FOR SMALL MODULAR REACTORS AND NON-LIGHT WATER REACTORS

PURPOSE:

To inform the Commission of the status of staff activities related to accident source terms for small modular reactors (SMRs) and non-light water reactors (non-LWRs), and the staff's current assessment of potential policy issues associated with use of mechanistic source terms (MSTs) in design-basis accident (DBA) dose analyses and siting.

This paper responds to Commission direction in Staff Requirements Memorandum (SRM)-M110329, "Staff Requirements – Briefing On Small Modular Reactors, 9:00 A.M., Tuesday, March 29, 2011, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance)," dated April 14, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110880535). This paper does not address any new commitments or resource implications.

SUMMARY:

As a result of pre-application activities over recent years and earlier work by the U.S. Nuclear Regulatory Commission (NRC) staff and the Commission, the NRC staff previously identified a number of potential policy and licensing issues for SMRs and non-LWRs. One issue is associated with the determination of source terms and the resulting dose calculations and siting evaluations. Current regulations and guidance, while maintaining the requirement to assume

CONTACTS: Arlon Costa, NRO/DEIA
(301) 415-6402

Michelle Hart, NRO/DSEA
(301) 415-1265

substantial meltdown of the core and substantial release into containment, permit the use of MSTs to account for the design-specific accident scenarios and accident progression in developing DBA radiological source terms. These same source terms are then used as a basis for evaluating equipment qualification, vital area access, some shielding calculations, and as part of the site suitability determinations. Near-term SMR applicants will likely propose to follow the existing regulations and guidance to develop design-specific DBA scenarios, coupled with more realistic mechanistic methods to model the accident progression and develop the source terms for these scenarios. Evaluation of the mechanistic methods will be an important part of the staff's review, but the approach proposed by near-term SMR design certification applicants does not currently appear to raise any policy issues.

Use of MSTs for DBAs for SMRs, given the expected smaller amount of fuel and unique and passive nature of these designs, is expected to result in reduced source terms when compared to large LWRs. These reduced source terms could form the basis for an applicant request to establish emergency planning zones that are smaller than what is currently required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.47(c)(2). In addition, the reduced source terms could result in smaller exclusion areas and low population zones (LPZs) as defined in 10 CFR 100.3, as determined in accordance with the safety assessment and dose criteria in 10 CFR 50.34(a)(1). Any NRC-approved reduction in the size of the LPZ could, in turn, allow such a reactor to be sited in closer proximity to a large population center as compared to large LWRs, as provided under 10 CFR 100.21(b). Any proposed site would also need to be consistent with other NRC requirements including 10 CFR 100.21(h), which limits, in qualitative terms, how close to the large population center a site can be.

The staff is informing the Commission of this potential policy issue in accordance with Commission direction to the staff to think expansively about upcoming SMR issues and engage the Commission early on matters of policy.[1] Some industry organizations have expressed a desire to deploy SMRs to replace existing coal plants, and acknowledge that to do so would require siting SMRs closer to population centers than large LWRs typically have been. For example, the Electric Power Research Institute's Utility Requirements Document states that "[T]he deployment of [SMRs] to replace retired coal-fired power stations in critical locations for electric grid stability requires acceptance of small nuclear plants closer to population centers." Staff intends to further engage stakeholders to understand whether proposed industry plans include siting of SMR facilities closer than has typically been allowed for large LWRs, and whether a clarification of the NRC's position on siting is warranted.

DISCUSSION:

Calculations of Source Terms

Quantitative determination of radioactive material that could potentially escape from the fuel and subsequently from the reactor core during normal operation or as a result of an accident, and ultimately be released outside the containment structure, plays a critical role in the facility's design and the agency's requirements to protect public health against radiation hazards. The analysis of the details of the timing and type of accident that could occur and the related amount of radioactive material that could be released in the event of that accident involve some

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1 SRM, "Briefing on Small Modular Reactors," dated April 14, 2011, (ADAMS Accession No. ML110880535).

uncertainty; therefore, DBA dose assessments typically are performed using methods that yield conservative dose estimates that are then compared to the regulatory requirements. The quantification of the accidental release of fission products into the containment atmosphere, or accident radiological "source term," affects the design of plant systems and is one element used to determine site suitability by meeting the dose criteria established for power reactor siting and safety system design in 10 CFR 50.34(a)(1)(ii)(D). A method to develop an MST is applicable to both SMRs and non-LWRs; however, there will be differences in the details of the analysis resulting from differences in the designs.

Significant progress has been made through the years in understanding reactor accident behavior for LWRs, including fission product release and transport. This increased technical understanding results in more detailed mechanistically based assessments of source terms to estimate the release and behavior of these fission products, which can substantially attenuate the magnitude of the release as compared to a more deterministic approach. The use of a mechanistic analysis includes accounting for fission product retention and removal processes.

A DBA source term is intended to be representative of a major accident scenario involving significant core damage, with radiation released to the environment through expected leakage from an intact containment. 10 CFR 50.34(a)(1)(ii)(D), which is referenced by 10 CFR 100.21, "Non-Seismic Siting Criteria," requires that the total effective dose equivalent (TEDE) from this source term to an individual at the exclusion area [2] boundary (EAB) and at the outer boundary of the LPZ not exceed 25 rem TEDE within any 2-hour period of a release and for the duration of the release, respectively. Among other siting criteria, one criterion contained in 10 CFR 100.21(b) requires that the distance to the nearest population center of at least 25,000 people is at least one and one-third times the distance from the reactor to the outer boundary of the LPZ. Other NRC requirements (10 CFR 100.21(h)) address the Commission's stated policy to site reactors "away from very densely populated centers" with a preference for "areas of low population density." More information on the specific NRC reactor siting and safety analysis requirements can be found in the enclosure.

A staff memorandum [3] issued on June 20, 2014, documented the status of this issue through the previous years. That memorandum included information on staff interactions with stakeholders and vendors such as the Nuclear Energy Institute, Generation mPower LLC, NuScale Power, and Department of Energy/Idaho National Laboratory. Communications and meetings with potential applicants were focused on design-specific activities associated with the DBA scenario. Since the 2014 memorandum, the staff has continued interactions with NuScale and, to a lesser degree, with the other stakeholders mentioned. In December 2015, NuScale submitted a licensing topical report[4] for staff review and approval of the report's accident source term methodology. The enclosure provides more background information on the evaluation of MSTs.

The staff is well prepared to review DBA source terms for SMRs, including MST analyses. The staff has the tools and technical expertise to evaluate the appropriate topics, such as scenario

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[2] See 10 CFR 100.3, "Definitions."

[3] Commission Memorandum, "Status of Mechanistic Source Term Policy Issue for Small Modular Reactors" June 20, 2014 (ADAMS Accession No. ML14135A482).

[4] NuScale Power, Topical Report TR-0915-17565, "Accident Source Term Methodology," Revision 0, dated December 2015 (ADAMS Accession No. ML16004A217).

selection, design-basis accident progression, fission product transport and removal processes, and the treatment of uncertainty. Although the staff has not yet developed source term tools and technical expertise for non-LWRs to the same level as that for SMRs, the staff believes a mechanistic approach could also be applied to non-LWR designs, subject to the availability of adequate tools and analysis approaches. In pre-application discussions with SMR designers, an overview and outline of proposed methods have been discussed. The methods proposed by the potential applicants in those interactions appear to generally build on currently approved methods. The staff did not note any specific area of concern during those interactions, nor any potential policy issues regarding implementation of mechanistic modeling of DBA source terms for SMRs.

Potential Effects of Source Terms on Siting

As discussed above, SMRs and non-LWRs may be designed with smaller cores and passive safety features, which may result in the calculation of smaller releases to the environment for a DBA and the resulting hypothetical dose calculation. This in turn would provide additional margin with respect to the dose related portions of the current NRC siting requirements focused on radiological safety. For example, future SMR applicants may be able to show that an individual at the EAB would not receive a dose that exceeds 1 rem TEDE within any 2-hour period of a release. This calculated dose is much lower than the current regulatory requirement that the dose not exceed 25 rem TEDE.

This results in the potential for the EAB and LPZ to be at the same distance around a very small site, potentially at a few hundred meters from the center of the reactor location or facility. The dose criteria which would allow for a smaller LPZ would also potentially allow the reactor to be considered for a location at a distance that is relatively close to a population center with at least 25,000 people. This reduced distance would be considered in combination with other siting requirements, including 10 CFR 100.21(h), which states that:

> Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable.

Although 10 CFR 100.21(h) could allow siting a reactor closer to a “densely populated center,” it also requires a plant to be sited away from a “very densely populated area.”

It is important to note that dose to individuals is only one element in the considerations of siting and proximity to densely populated centers. Siting requires considerations of other factors as cited in 10 CFR 100.20, “Factors to be considered when evaluating sites.” These include population density and use characteristics of the site environs, nature and proximity of man-related hazards, and physical characteristics of the site, which includes seismology, meteorology, geology, and hydrology.

There are currently no combined license (COL) applications scheduled that employ an SMR or non-LWR design, nor have any pre-application discussions to date specifically indicated that an applicant plans to site its facility closer to densely populated centers. However, the staff understands that future applicants may request to site SMRs at retired fossil fuel plants or other industrial sites. For example, one SMR vendor has stated that its design “can be sited next to population centers without any threat to the local environment or populace.” Therefore, the staff will proactively engage with stakeholders to determine whether clarification on siting is needed. If so, the staff will inform the Commission and request approval to develop a draft regulatory basis for Commission consideration that can be subsequently included in regulatory guidance. The staff anticipates that development of the draft regulatory basis will take approximately 18 months and, if approved by the Commission, another year will be needed to finalize the regulatory guidance. In light of resource constraints and other higher priorities, the staff believes this approach to develop guidance is the most prudent means of handling this issue. Of course, the staff will ensure that it remains abreast of industry plans so that the staff can complete the regulatory guide ahead of the time it will be needed.

CONCLUSIONS:

The Commission has previously provided guidance to the staff on the use of MST analysis and the staff has implemented it in appropriate stages, such as the development of guidance for justifiable mechanistic analysis for an alternative accident source term. SMR and non-LWR applicants can employ modern analysis tools to demonstrate quantitatively the safety features of those designs. MST analysis methods can also be used by applicants to demonstrate the ability of the enhanced safety features of plant designs to mitigate accident releases. This would also allow future COL applicants to consider reduced distances to EABs and LPZs, and potentially increased proximity to population centers.

All currently known near-term applicants are utilizing sites that appear to comply with the directive in 10 CFR 100.21(h) to locate away from very densely populated areas. However, the staff may need additional direction from the Commission if an applicant were to propose a site that is significantly closer to a very densely populated center than previously approved.

Staff will explore these issues with interested stakeholders over the next 12 to 18 months. Engaging stakeholders on this issue relatively early in the licensing process for these designs will provide time for the staff to propose any changes to the Commission. Any policy issues and the staff’s proposed path forward will be provided to the Commission in a future paper.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection.

/RA/

Victor M. McCree
Executive Director
for Operations

Enclosure:
Background Information Regarding
Accident Source Terms and Siting

COORDINATION:

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Victor M. McCree
Executive Director
for Operations

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OFFICE	NRO/DARR:PM	NRO/DSEA:TR	NRO/DARR:BC	NRO/DARR:D
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OFFICE	NRO/DSEA:D	RES:D	NRR:D	NSIR:D
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Background Information Regarding Accident Source Terms and Siting

Relevant Documents Summary

The following is a short summary of relevant Nuclear Regulatory Commission (NRC) documents related to accident source term analyses, which are discussed in more detail later in this enclosure:

- TID-14844 (1962):[1] Used a conservative and deterministic approach to estimate light-water reactor (LWR) off-site doses from a postulated core melt accident to account for site suitability in compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria."

- SECY-90-016 (1990):[2] Provided recommendations concerning departures from regulations for evolutionary LWRs while recognizing that deviations from practices at the time could be warranted in future designs. This led to the development of a more physics-based design-basis accident (DBA) source term than the TID-14844 source term.

- SECY-93-092 (1993):[3] Defined mechanistic source terms (MSTs) and recommended that scenario-specific (mechanistic) source terms be allowed. This included a caveat such that there should be sufficient understanding of fuel performance, fission product behavior, and accident selection to bound uncertainties. The Commission approved this staff recommendation in a July 30, 1993, staff requirements memorandum (SRM) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003760774).

- NUREG-1465 (1995):[4] Provided insights from severe accident research on fission product release and transport that were used in developing a revised source term for large LWRs, which was expressed in terms of times and rates of appearance of radioactive fission products into the containment, the types and quantities of the species released, and other important attributes such as the chemical forms of iodine, given a severe core-melt accident.

- RG 1.183 (2000):[5] Provided elements of a mechanistic approach with the deterministic analysis and, for regulatory purposes, a more realistic estimate of the amount of fission

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1 Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," Atomic Energy Commission, March 23, 1962 (ADAMS Accession No. ML021750625).

2 SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," January 12, 1990 (ADAMS Accession No. ML003707849).

3 SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," dated April 8, 1993 (ADAMS Accession No. ML040210725).

4 NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995 (ADAMS Accession No. ML041040063).

5 Regulatory Guide RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

products present in the containment from a postulated severe accident for large LWRs than was included in TID-14844.

- SECY-03-0047 (2003):[6] Retained previous Commission's guidance in an SRM [7] (consistent with previous Commission and Advisory Committee on Reactor Safeguards (ACRS) views contained in SRM-SECY-93-092) that allowed the use of scenario-specific source terms for non-LWRs, provided that there were sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgement used to bound uncertainties. It also provided support for moving this issue toward performance-based regulations and away from prescriptive regulations.

- SECY-05-0006 (2005):[8] Proposed a framework for the "Use of Scenario-Specific Source Terms for Licensing Decisions" and licensing approach for new plant licensing, in support of SRM-SECY-03-0047.

- NRC Policy Statement (2008):[9] Defined along with NUREG-1226 [10], advanced reactors as those with innovative designs for which licensing requirements will be significantly different from the existing LWR requirements. These documents also provide guidance for developing new regulatory requirements supporting licensing advanced reactor designs.

- Commission Memorandum (2011):[11] Provided the status on planned activities to address methods for determining a MST and to describe the circumstances in which a source term determined by such methods could be appropriate.

- Commission Memorandum (2013):[12] Provided the status of activities for the siting source term and emergency plan (EP) issues for small modular reactors (SMRs).

- Commission Memorandum (2014):[13] Documented the status of this issue since issuance of SECY-10-0034 and also described staff interactions with stakeholders, such as public meetings during the 2010–2014 period.

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6 SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," March 28, 2003 (ADAMS Accession No. ML030160002).

7 SRM-SECY-03-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs," dated June 26, 2003 (ADAMS Accession No. ML031770124).

8 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 (ADAMS Accession No. ML043560093).

9 73 FR 60612, "Policy Statement on the Regulation of Advanced Reactors," October 14, 2008.

10 NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," June 1988.

11 Commission Memorandum, "Status of Staff Activities to Address Mechanistic Source Term Methodology and Its Application to Small Modular Reactors," December 29, 2011 (ADAMS Accession No. ML113410366).

12 Commission Memorandum, "Current Status of the Source Term and Emergency Preparedness Policy Issues for Small Modular Reactors," May 30, 2013 (ADAMS Accession No. ML13107A052).

13 Commission Memorandum, "Status of Mechanistic Source Term Policy Issue for Small Modular Reactors" June 20, 2014 (ADAMS Accession No. ML14135A482).

DBA Source Term and MST Requirements

The regulations in 10 CFR 50.2, "Definitions," define source term as "the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release." The safety analysis described in 10 CFR 50.34, "Contents of Applications; Technical Information," requires consideration of an accident that results in core melting and release of fission products to the containment. For large LWRs, the loss-of-coolant accident (LOCA) is the DBA that most fits the required analysis as described in 10 CFR 50.34(a)(1), and it models an arrested core melt with releases from the reactor coolant system (RCS) to the intact containment, fission product transport, and removal in the containment and other engineered safety features (ESFs), and release to the environment through design containment leakage at the leak rate demonstrated through technical specification surveillance. DBA dose analyses for accidents other than the LOCA are also evaluated in the safety analysis report. Non-LOCA DBA dose analyses also have radiological source terms, but in general terms they are based on calculated bounding fuel failure rates (if applicable to the accident), or the reactor coolant may be the only source of radioactivity for release through design-specific calculated coolant steaming and leak rates.

The deterministic LOCA source term in TID-14844 was used by most of the operating reactors in initial licensing, and is the technical basis to calculate distance requirements originally used for 10 CFR Part 100 since 1962. The TID-14844 LOCA assumed an instantaneous and homogenous release of 100 percent of the noble gases and 50 percent of the iodines from the core to the containment, 50 percent of which was assumed to deposit on surfaces. These assumptions were applied to all large LWR designs.

In 2000, guidance in Reg Guide 1.183 for an alternative source term (AST) originated from severe accident research [14] and physically based (mechanistic) models. It contains one LOCA source term for all pressurized-water reactors (PWRs) and a second one for all boiling-water reactors (BWRs). The AST LOCA accident scenario similarly deterministically assumed initial failure of core cooling, but the progression and release to containment was mechanistically modeled to be applicable to large LWRs (either PWRs or BWRs). The AST release to containment occurs over time instead of instantaneously, as in TID-14844.

Earlier discussions regarding SMRs and non-LWR designs focused on a full mechanistic approach, meaning that the "mechanistic source term," or MST, uses scenario-specific mechanistic or physically based modeling of accident progression to estimate a design-specific DBA radiological source term release from the RCS to the containment to demonstrate compliance with reactor siting and safety assessment regulations. It is a further evolution of the DBA radiological source term.

The "Policy Statement on the Regulation of Advanced Reactors" states that the Commission expects that "advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions." The staff understands that SMR and non-LWR designers are including safety margins or using simplified inherent, passive, or other innovative means to accomplish safety functions in their

[14] Because of SECY-90-016 and the related SRM, the revised source term for LWRs in NUREG-1465 was developed to provide a more physically based DBA source term to replace the TID-14844 source term.

specific reactor design features. One such feature is the passive emergency core cooling function. In general, the designs rely on limited amounts of equipment, rely on natural circulation, and do not call for operator action to initiate the function. Under current NRC regulations, SMR and non-LWR designers are required to provide in their applications the appropriate technical bases for the assumptions and methods they use to model the DBA scenarios and system response, including the bases for any passive safety functions that are credited in fission product release mitigation. This is in line with the evaluations of large LWR passive design features as has already been done for the certified reactor designs of AP600 and AP1000, as well as the Economic Simplified Boiling-Water Reactor.

SECY-02-0139 [15] framed the proposal to use MSTs that are plant-specific, by stating:

> Current LWRs use site-specific parameters (e.g., exclusion area boundary) and a predetermined source term into containment to analyze the effectiveness of the containment and site suitability for licensing purposes. These source terms are described in documents TID-14844 and NUREG-1465 and are based upon enveloping the fission product releases that would be predicted to occur given a core melt accident. On the other hand, future plants, particularly non-LWRs, propose not to use a predetermined source term for assessing the effectiveness of plant mitigation features or site suitability, but rather to use plant-specific accident source terms corresponding to each of the AOOs [anticipated operational occurrences] and DBEs [design basis events] defined for the plant. Such an approach puts a burden on the applicant and staff to understand the fission product release characteristics and uncertainties associated with a variety of accident scenarios.

SECY-93-092 and its associated SRM also provided the following definition and staff recommendation on MST for advanced non-LWR designs:

> **[Definition:]**
> A mechanistic source term is the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.
>
> **[Recommendations:]**
> Advanced reactor and CANDU 3 source terms should be based upon a mechanistic analysis and will be based on the staff's assurance that the provisions of the following three items are met:
>
> - The performance of the reactor and fuel under normal and off normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through

[15] SECY-02-0139, "Plan for Resolving Policy Issues Related to Licensing Non-Light Water Reactor Designs," dated July 22, 2002 (ADAMS Accession No. ML021790610).

the research, development, and testing programs to provide adequate confidence in the mechanistic approach.

- The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.

- The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties.

The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design.

The staff indicated that this approach provides, for regulatory purposes, a more realistic estimate of the amount of fission products present in the containment from a postulated severe accident than was included in TID-14844. It was recognized that this approach to source term modelling can have implications on issues related to 10 CFR Part 100, equipment qualification, control room habitability, and assessments of severe accident risks in plant environmental impact statements. The SRM further stated "that a mechanistic 'scenario specific' source term for each reactor concept warrants further consideration before evaluating the acceptability of the design."

In SECY-03-0047, the staff reaffirmed the earlier approach in SECY-93-092, recommending the following, which was approved by the Commission in its associated SRM dated June 26, 2003:

- Retain the Commission's guidance contained in the July 30, 1993, SRM that allows the use of scenario-specific [mechanistic] source terms, provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgement is used to bound uncertainties[2].

Footnote 2 above states: "This represents a fundamental change in practice from that used in LWRs, in that the source term used for siting considerations may not be that associated with a core melt accident."

- This recommendation…would allow credit to be given for the unique aspects of plant design and builds[sic] upon the recommendation under Issue 4[[16]]. Furthermore, this approach is consistent with prior Commission and ACRS views. However, this approach is also dependent upon understanding fuel and fission product behavior under a wide range of scenarios and on ensuring fuel and plant performance is maintained over the life of the plant. This approach is also very dependent on the event selection process.

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[16] Probabilistic Event Selection, Safety Classification and Reliability Criteria.

The staff updated the Commission on the use of scenario-specific source terms in SECY-05-0006.[17] Among other topics within SECY-05-0006, it specifically addressed how the staff proposed to integrate scenario-specific source terms into the proposed regulatory structure for new plant licensing. The staff proposed using a flexible, performance-based approach to establish scenario specific licensing source terms. The key features of the staff's approach were as follows:

- Scenarios are to be selected from a design-specific probabilistic risk assessment (PRA).
- Source term calculations are based on verified analytical tools.
- Source terms for compliance should be 95 percent confidence level values based on best estimate calculations.
- Source terms for emergency preparedness should be mean values based on best estimate calculations.
- Source terms for licensing decisions should reflect scenario-specific timing, form, and magnitude of the release. This approach puts the burden on the applicant to develop the technical basis. An applicant could, however, propose to use a conservative source term.

Based on the documents listed at the beginning of this enclosure, the approach used by applicants for MST methodologies for SMRs and non-LWRs, should be, at a minimum, as consolidated in the following list:

1. The reactor must be considered to be an advanced reactor in accordance with the description in the NRC's Policy on Advanced Reactors.

2. An MST used in siting and DBA dose analyses is defined as the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, resulting in release into the environment. The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design. In developing an MST, the following conditions apply:

    a. The performance of the reactor and fuel under normal and off-normal conditions must be sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through research, development, and testing programs to provide adequate confidence in the mechanistic approach. Deterministic engineering judgment should be used to bound uncertainties.

[17] 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 (ADAMS Accession No. ML043560093).

b. The transport of fission products can be adequately modeled for all barriers and pathways to the environment, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.

c. Credit may be given for the unique aspects of plant design (i.e., performance-based), which builds upon probabilistic event section, safety classification, and reliability criteria to determine the appropriate accident scenario and progression.

d. The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties. Scenarios are to be selected using information from a design-specific PRA.

e. Source term calculations are based on verified analytical tools.

3. Source terms for licensing decisions should reflect scenario-specific timing, form, and magnitude of the release. This approach puts the burden on the applicant to develop the technical basis. An applicant could, however, propose to use a conservative source term.

a. Source terms for compliance related to safety assessment and siting should be 95 percent confidence level values based on best estimate calculations.

b. By comparison, source terms for emergency preparedness should be mean values based on best estimate calculations.

In discussions with SMR and non-LWR (e.g., Next Generation Nuclear Plant) designers, the designers indicated that they believed that the current siting regulatory framework is applicable to their designs. As described previously, the current regulatory approach requires the analysis of a large release to containment. To fulfil this requirement, the SMR or non-LWR designer would need to develop accident scenarios, specific to each design, using a mechanistic approach to developing the DBA large radiological source term to containment.

The selection of appropriate scenarios for the purposes of developing design-specific DBA source terms is crucial and will require close review and coordination with the reviews of event classification and PRA. Furthermore, mechanistic modeling of fission product transport and mitigation in the core, RCS, containment, and ESFs will require adequate technical basis. The staff already has been evaluating similar aspects within reviews of new large LWR designs. As an example, the staff has previously evaluated and given approval of design-specific mechanistic modeling of aerosol deposition in containment for the AP600 and AP1000.[18]

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18 NUREG-1512, "Final Safety Evaluation Report Related to the Certification of the AP600 Standard Design Certification Docket No.52-003," September 1998 (ADAMS Accession No. ML081020331) and NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," September 2004 (ADAMS Accession Nos. ML043450344, ML043450354, ML043450284, ML043450290, and ML043450274).

Reactor Siting and Safety Analysis Regulatory Requirements

Using MST analysis affects the application of the siting criteria contained in 10 CFR Part 100 and the calculation of the DBA radiological consequences when comparing the results to the dose criteria in 10 CFR 50.34(a)(1). The regulations in 10 CFR 100.21, "Non-Seismic Siting Criteria," require that the site have an exclusion area surrounding the reactor in which there are no permanent residents and the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from that area. An low-population zone (LPZ) immediately surrounding the exclusion area is also required.[19] The number and density of residents within the LPZ are to be such that there is a reasonable probability that protective measures could be taken on their behalf in the event of a serious accident. Specific distances to the outer boundary of the exclusion area, called the exclusion area boundary (EAB), and to the outer boundary of the LPZ are not prescribed by regulation, so they may vary from reactor site to reactor site. As a siting criterion, 10 CFR 100.21(b) requires that the distance to the nearest population center of at least 25,000 people is at least one and one-third times the distance to the outer boundary of the LPZ.

The siting regulation in 10 CFR 100.21 also requires that the consequences of postulated accidents meet the radiological dose criteria in 10 CFR 50.34(a)(1), considering the site's atmospheric dispersion characteristics. Specifically, the analysis described by 10 CFR 50.34(a)(1) is a description and safety assessment of the site and a safety analysis of the facility, including analysis of the ESFs and those barriers that must be breached because of an accident that can result in a release of radioactive material to the environment, with special attention to plant design features that are intended to mitigate the radiological consequences of accidents. The regulation in 10 CFR 50.34(a)(1)(ii)(D) states that the safety assessment by the applicant:

> …shall assume a fission product release[6] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:
>
> (1) An individual located at any point on the boundary of the exclusion area 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).
>
> (2) 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

[19] The regulations in 10 CFR 100.3, "Definitions," provide the regulatory definitions of the exclusion area and low population zone, including more specific detail on the residents and population surrounding the site.

fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

Footnote 6 of this regulation describes the assumed release to containment for this analysis 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.

Safety analysis requirements in 10 CFR 52.17(a)(1)(ix), 10 CFR 52.47(a)(2), and 10 CFR 52.79(a)(1)(iv), for new reactor applications for an early site permit, standard design certification, and combined operating license, respectively, describe the same type of analysis and use the same evaluation criteria of 25 rem TEDE at the EAB and LPZ as given in 10 CFR 50.34(a)(1). For DBAs other than LOCA (or bounding scenario as applicable to SMR and non-LWR designs), the regulatory guidance [20] is that the dose acceptance criteria are fractions of the 25 rem TEDE criterion, either 25 percent (6.5 rem TEDE) or 10 percent (2.5 rem TEDE). These dose acceptance criteria are to be generally commensurate with the higher likelihood of the accident scenario, such that the more likely scenarios (e.g., rupture of one steam generator tube with the coolant at equilibrium activity concentration) are compared to a lower dose criterion (2.5 rem TEDE). For the set of DBAs evaluated for offsite consequences, doses are also calculated within the control room to show compliance with the 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 radiological habitability dose criterion of 5 rem TEDE.

The siting dose criteria are expected to be evaluated through DBA dose analyses on a per-reactor basis, even for multimodule plants. This is because of the design protection against external events that may affect more than one module concurrently, separation and independence of the modules' systems, structures and components and safety functions, and design against common cause failures among modules, in accordance with GDCs 2, 4, and 5. This means that the siting of a multimodule plant, including the determination of the EAB, LPZ and population center distances, is currently expected to be based upon the evaluation of a single reactor.

Implications for SMRs and Non-LWRs

The siting regulations require that residents in a densely populated center of 25,000 people or more be located at minimum specified distances from the reactor relative to the LPZ outer boundary. For currently operating reactors and recently licensed new reactors, EABs and LPZ boundary distances may vary from site to site, and reactor to reactor, because the distance determination is partly a function of the capability to meet the siting dose acceptance criteria at the chosen distance. In general, large LWR EABs are around 0.8 km (0.5 miles), and the LPZs may be around 4.8 to 8 km (3 to 5 miles). The 10 CFR 100.21(b) requirements on reactor location relative to population centers result in the closest population center to the reactors lying

[20] NUREG-0800 (SRP) Section 15.0.3, "Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors" (ADAMS Accession No. ML070230012).

outside the LPZ, but the population center could be inside the 10-mile plume exposure EPZ used for large LWRs.

In addition as stated in 10 CFR 100.21(h):

> Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable.

Regulatory Guide 4.7 [21] provides guidance related to location of LWR sites relative to major population centers. Additionally, regarding considerations of a low likelihood of severe accidents, the NRC has previously provided the following discussion in the *Federal Register* regarding "Siting Away From Densely Populated Centers":[22]

> In summary, next-generation reactors are expected to have risk characteristics sufficiently low that the safety of the public is reasonably assured by the reactor and plant design and operation itself, resulting in a very low likelihood of occurrence of a severe accident. Such a plant can satisfy the QHOs [quantitative health objectives] of the Safety Goal with a very small exclusion area distance (as low as 0.1 miles). The consequences of design basis accidents, analyzed using revised source terms and with a realistic evaluation of engineered safety features, are likely to be found acceptable at distances of 0.25 miles or less. With regard to population density beyond the exclusion area, siting a reactor closer to a densely populated city than is current NRC practice would pose a very low risk to the populace.
>
> Nevertheless, the Commission concludes that defense-in-depth considerations and the additional enhancement in safety to be gained by siting reactors away from densely populated centers should be maintained.

As is reflected in the above, the Commission has previously considered whether ESFs could allow for siting of reactors near densely populated cities. Further discussion of past metropolitan siting issues can be found in "Containing the Atom, Nuclear Regulation in a Changing Environment," 1963-1971, J. Samuel Walker, Chapter 4, "Reactors Downtown? The Debate over Metropolitan Siting," as well as in NUREG-0478.[23] The staff anticipates examining this history more closely as it engages with stakeholders over the next several months.

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21 Regulatory Guide 4.7, Rev. 3, "General Site Suitability Criteria for Nuclear Power Stations," March 21, 2014 (ADAMS Accession No. ML12188A053).

22 Federal Register, 61 FR 65157, RIN 3150-AD93, "Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants," December 11, 1996.

23 NUREG-0478, "Metropolitan Siting - A Historical Perspective," D.F. Bunch, October 31, 1978 (ADAMS Accession No. ML12187A192).

MST Relationship to the Basis for EPZ Size Evaluations

In SECY-11-0152,[24] the staff discussed possible consideration of potential changes to the emergency planning and preparedness framework for SMRs. Among the considerations for establishing the size of EPZs for SMRs, the staff stated its expectation that the dose assessments that provide the basis for the EPZ distances would evaluate a spectrum of accidents, using the plant design PRA, as well as including current insights on severe accident progression.

Subsequently in SECY-15-0077,[25] the staff proposed a consequence-based approach to establishing requirements for offsite emergency planning, as determined to be necessary for SMRs and other new technologies. In the related SRM dated August 4, 2015 (ADAMS Accession No. ML15216A492), the Commission approved the staff's proposal to revise NRC regulations and guidance through rulemaking to "demonstrate how their proposed facilities achieve U.S. Environmental Protection Agency (EPA) Protective Action Guide (PAG)[26] dose limits at specified EPZ distances, which may include the site boundary." During the upcoming rulemaking process, the technical basis for the emergency planning framework for SMRs and non-LWRs will be developed, including guidance and criteria for radiological consequence analyses performed by the applicant to justify an EPZ size commensurate with the potential offsite radiological risk posed by the facility.

The DBA dose analyses performed by an applicant to show compliance with the siting safety analysis regulations and their related radiological releases to the environment are expected to be included in the spectrum of analyses that form the technical basis for the EPZ distance. However, the EPZ size basis dose analysis would be done for the appropriate exposure period for comparison to the EPA PAG dose criterion of 1 rem TEDE, which is comparably less than the siting dose criterion of 25 rem TEDE. In this case, EABs and LPZ boundaries, which are located at shorter distances, may coincide with the plume exposure EPZ. Satisfying a 1 rem TEDE dose criterion for an EPZ of the same size as the EAB and LPZ would likely also demonstrate compliance by a large margin if compared with the 25 rem TEDE siting requirement in 10 CFR 50.34(a)(1)(ii)(D). Additionally, the technical basis for both emergency plans and EPZs needs to include consideration of beyond design basis accidents.

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24 SECY-11-0152, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," October 28, 2011 (ADAMS Accession No. ML112570439).

25 SECY-15-0077, "Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies," May 29, 2015 (ADAMS Accession No. ML15037A176).

26 EPA PAGs, are given in EPA-400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." The PAG is defined as the dose at which public protective actions should be considered and undertaken.