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POSITION PAPER

SMALL MODULAR REACTOR SOURCE TERMS

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This NEI Position Paper was developed by the NEI Small Modular Reactor Licensing Task Force. Authors and reviewers of this paper include Task Force representatives from Holtec, Generation mPower, NuScale Power, and Westinghouse all of whom are in the process of designing integrated Pressurized Water Reactor (iPWR) Small Modular Reactors (SMRs).

I. Introduction and Purpose

The subject of radiological source terms associated with small modular reactors (SMRs) has been identified by the Nuclear Regulatory Commission (NRC) in SECY 10-0034, and by the nuclear industry, as a licensing topic for which it may be appropriate to utilize a different treatment from that currently used for large light water reactors (LWRs). The Nuclear Energy Institute (NEI) SMR Licensing Task Force has developed this position paper to identify unique SMR source term issues within the context of current regulations and to delineate paths forward to address these issues. Radiological source terms are important aspects of a wide range of SMR licensing, design, and operations issues. Although the principal purpose of this paper is to address source terms that would be used for accident analyses as presented in Chapter 15 of a safety analysis report (SAR) or design certification document (DCD), aspects of this paper are expected to be applicable to: shielding design, emergency planning, security, control room design, post-accident monitoring and recovery, equipment qualification, probabilistic risk assessment (PRA) severe accident analyses, and siting. NEI is preparing a separate paper on emergency planning and other source term related topics are expected to be addressed by individual SMR design organizations.

Although generally intended to be applicable to all SMR designs, the focus of this paper is primarily on integral pressurized water reactor (iPWR) SMRs because they have the most mature designs and are currently involved in pre-application interactions with the NRC. An iPWR is defined as a pressurized water reactor design in which the primary coolant system and all (or most) of its components (i.e. pressurizer, steam generators, and reactor coolant pumps) are enclosed in one pressure vessel. The iPWR SMR designers also have announced specific near term schedules[1] for NRC design licensing review either as part of 10 CFR 52 in a Design Certification Application (DCA) or under 10 CFR 50 in a Construction Permit Application (CPA). This paper is directed at addressing the iPWR radionuclide inventories associated with the reactor core, primary coolant system, secondary coolant system, and spent fuel pool, but does not address dry spent fuel storage.

The purpose of this paper is to present an evaluation of existing regulations and regulatory guidance that are pertinent to SMR source terms, delineate SMR source term applications, discuss unique SMR features that may affect source terms, identify SMR accident dose analysis source term regulatory issues and propose a resolution path forward for these issues within the existing NRC regulatory framework.

This paper is divided into the following five sections: (1) introduction and purpose, (2) source term regulation overview, (3) source term attributes and iPWR design-operational differences, (4) assessment of SMR regulatory issues, and (5) summary. Appendix A presents a comprehensive tabulation of source term considerations for each safety analysis report chapter and Appendix B presents a brief discussion of several policy papers that describe aspects of a mechanistic source term definition.

[1] Four iPWR SMR design companies, Holtec, mPower, NuScale Power, and Westinghouse have publicly announced their planned NRC license application submittal dates (either 10 CFR 52 or 10 CFR 50) to be in the 2013-2014 time frame.

II. Source Term Regulation Framework Overview

The assessment of radionuclide source terms involves a number of safety and environmental review areas associated with normal operations, anticipated operational occurrences, and postulated accidents. The specific use of *source term* phraseology in the regulations themselves, however, is predominantly directed at accident analyses and focuses on licensing decisions relating to containment performance and plant siting.

Regulatory Requirements

10 CFR § 50.2 defines *source term* as referring to:

> "…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 regulatory text that follows in 10 CFR Part 50 is then limited to using the phrase "accident source term" in relation to design basis radiological consequence analyses". For example, 10 CFR § 50.34(f)(2), in addressing additional TMI-related requirements, states that an application shall provide sufficient information relative to:

> "(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)"

This focus on accidents appears in several additional requirement statements in 10 CFR § 50.34. The next use is in 10 CFR § 50.67 which details requirements should an operating plant license holder wish to revise their current accident source term used in their design basis radiological analyses. Finally, 10 CFR Part 100 has one use of the phrase in describing considerations for determining the size of the exclusion area, low population zone and population center distance for sites with multiple reactor facilities.

Of note is that in each regulatory statement on accident source terms, a footnote is entered that states:

> "The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These 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."

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The regulatory history is mixed on the use of the phraseology mechanistic source term, sometimes describing accident source terms as "new", "modern", or "advanced". Appendix B presents a brief discussion of several policy papers that describe aspects of a mechanistic source term definition.

Regulatory Guidance

A review of NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)*, is useful in identifying the broader set of review areas where source terms are considered within the context of a Safety Analysis Report (SAR). The safety review guided by

NUREG-0800 (and referenced regulatory guides) is directed at demonstrating acceptability of plant systems and components as well as the protection of plant workers and members of the public during normal operation, anticipated operational occurrences, and accident events.

Appendix A presents a summary of the areas associated with review of source terms within a typical SAR, using as a guide the detailed review sections that comprise NUREG-0800. The identified areas are listed by SAR chapter and type of safety consideration. In several areas, aspects of the analysis methodology to be used are described more directly in the regulatory guidance. For these areas, predominantly the accident analysis area, the approach for determining which radionuclides are to be considered and how they are to be analyzed is specifically stated. Appendix B summarizes the regulatory and policy definition of mechanistic source3 term.

Modularity

The topic of multiple reactors at a single site is addressed in 10 CFR § 52.47(c)(3), which states:

> "An application for certification of a modular nuclear power reactor design must describe and analyze the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions. The final safety analysis must also account for differences among the configurations, including any restrictions that will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating.

Specific to the topic of source terms, 10 CFR § 100.11(b) states[2]:

> "(b) For sites for multiple reactor facilities consideration should be given to the following:
>
> > (1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.
> >
> > (2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the Commission the basis for such a reduction in the source term."

In its memorandum, "Status of Staff Activities To Address Mechanistic Source Term Methodology and Its Application to Small Modular Reactors," dated December 29, 2011, the NRC staff presented to the

[2] § 100.11(b) is within 10 CFR Part 100, subpart A, "Evaluation Factors for Stationary Power Reactor Site Applications Before January 10, 1997 and for Testing Reactors." In the absence of comparable language in subpart B to Part 100, "Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997," the text in § 100.11 is informative.

Commission a brief background on the historical basis and noted several areas where a mechanistic source term might contribute to the safety evaluation for SMRs, e.g., "siting, control room habitability, emergency preparedness, and security considerations. The memorandum stated:

> "The staff will remain engaged with SMR stakeholders regarding applications of a mechanistic source term, review preapplication white papers and topical reports concerning source term issues that it receives from potential SMR applicants, discuss design-specific proposals to address this matter, and consider research and development in this area. If necessary, the staff will propose changes to existing regulations or regulatory guidance or propose new guidance concerning the source term for an SMR to support development of review standards for iPWRs or other SMR designs."

In summary, several sections within 10 CFR Part 50 and Part 100, along with policy papers and NUREG-0800 provide regulatory requirements, guidance, and background for source term development and application that can be used by SMRs. The NRC has recognized that the use of appropriately developed mechanistic source terms for accident analysis can assure regulatory compliance. Moreover, the NRC has stated that unique design features of SMRs may warrant a different approach to source terms than that used for large LWRs.

III. Source Term Attributes and iPWR Design-Operational Differences

In order to address source term applications for SMRs, this section provides a description of source term related attributes (e.g., physical and chemical forms), evaluates the different purposes of source terms in safety analyses, and highlights those design and operational features that potentially affect source terms in iPWRs.

Source Term Related Attributes

The potential or postulated release of radioactive substances from a nuclear power plant to the environment (the source term) depends on the following factors:

- the inventory of fission products and other radionuclides in the core,
- the progression of core damage[3],
- the fraction of radionuclides released from the fuel and the physical and chemical forms of released radioactive materials,
- the retention of radionuclides in the primary coolant system, and
- the performance of the means of confinement (e.g., emergency system ventilation rate, filter efficiency, containment leakage rate, liquid effluent release rate, radioactive decay due to time delay of release, deposition on surfaces, and resuspension).

In addition, the doses associated with the radionuclides released depend on the release mode (single puff, intermittent, or continuous) and the release point (stack, ground level, confinement bypass).

Radionuclide Source Term Use in Power Reactor Safety Analyses

Radionuclide source terms, typically discussed in terms of accident analyses, also have significance in a number of design, environmental, and safety areas. For power reactors, 10 CFR Parts 50 and 52 require that a SAR be submitted "that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole" (see §52.47(a) as an example). The format and content of today's SARs are guided by Regulatory Guides 1.70 and 1.206 and the Standard Review Plan (NUREG-0800). A review of the SAR format shows that radionuclide source terms are an element in most of the 19 chapters. Table A-1 (see Appendix A) briefly describes each of these uses of source terms in the SAR and presents several considerations where SMR designs and operations have the potential to alter the current usage as applied to large LWRs. Note that these considerations were derived based on currently available public information for the Holtec, mPower, NuScale, and Westinghouse iPWR SMR designs and are applicable to one or more of these SMRs.

The considerations associated with iPWR SMR radionuclide source terms identified in Table A-1 have been categorized as noted below.

1. Radionuclide inventory of the reactor core and spent fuel

   Typical PWR fuel designs will be used in all four iPWRs, except there will be fewer fuel assemblies in the core. Three of the iPWR designs use shorter fuel assemblies while the Holtec

[3] Core damage is defined as any condition that results in fuel cladding breach and fission product release to the primary coolant system.

design has full length fuel assemblies. The iPWR fuel cycles range from two to four years and either will employ fuel shuffling or will completely replace all fuel after each cycle. Lower power levels will result in lower fission product inventories than for typical large PWRs since the core inventory is roughly proportional to power. The net effect of these differences in iPWR design and operations will be a reduction in the amount of radionuclides available for release during postulated accidents that damage the fuel in the core or spent fuel pool.

The core source terms will be calculated using currently accepted practice and computer codes (e.g., ORIGEN2 for core fission product inventory).

2. Impact on shielding and minimizing radiation exposure of personnel – containments, equipment layouts, operations different for SMRs

   The integration of the reactor vessel, steam generator, and pressurizer may result in increased radiation exposure to components that are internal to the vessel rather than external (e.g., control rod drive mechanisms). Each of the iPWR designs uses a free-standing steel vessel as the primary containment. These containments are considerably smaller than the primary containment of a large PWR. As such, some components will be placed closer together and there is less room available for shielding, potentially resulting in higher equipment dose rates than are found for large LWRs (and possibly resulting in higher dose rates outside the containment vessel). As a result, access to the containment vessel during power operation will be limited or, in the case of the Westinghouse and NuScale designs, precluded since they are both submerged during normal power operation.

   The nuclear island portion of the iPWRs is also much smaller than typical large PWRs. This may impact the ability to provide adequate separation from/shielding of access routes to vital areas or areas requiring access during normal operations or following an event.

3. Plant discharges to the environment

   Plant discharge during normal operation will be handled in a manner consistent with operating reactors. Equipment and floor drains, liquid and gaseous radwaste systems and tanks, main condenser, etc. will be used to collect and process normal plant process waste and maintain releases to the environment less than regulatory limits. Monitoring of these releases is expected to be similar to that of large PWRs.

   Postulated releases during an accident are expected to be significantly lower due to the reduction in radionuclide inventory of the core, reduction in pathways available for release (e.g., less systems and connections), enhanced cooling capability through passive features that reduce the likelihood of core uncovery, and/or submergence of the containment. Additionally, the integration of the steam generator and pressurizer inside the reactor vessel eliminates the large diameter piping that would otherwise connect them to the reactor vessel and thus eliminates the large break loss of coolant accident (LOCA) as a design basis accident. The iPWR designs have no reactor vessel penetrations below the top of the core and also have a large water inventory relative to core thermal power. In the case of a postulated small break LOCA, these features would delay the onset of fuel uncovery and concomitant fuel damage even with multiple component failures.

4. Operator access to required systems during events and associated recovery period

Loss of AC power events have a reduced impact on the iPWR designs, since they rely largely on passive features and Class 1E batteries to reach and maintain shutdown conditions for an extended period of time following an event. As a result, the need for operators to access any plant areas outside the control room during the early stages of an event is minimized or eliminated. Following this initial period (expected to be a minimum of 72 hours), external connections will be available to provide additional sources of power and cooling water if needed to maintain the plant in a safe and stable condition.

5. Classification of radionuclide releases and leakage

   Requirements for continuous or periodic sampling and monitoring of gaseous and liquid effluents are expected to be no different from those in effect for current plants and will be met in a similar manner.

6. Management of system contamination through interfaces with radioactive systems

   Design features will be provided to prevent or minimize the potential for radioactivity containing systems to contaminate systems that are not expected to contain radioactivity. These provisions incorporate lessons learned from current operating plants and recently certified licensed ALWRs.

7. Radiological impacts on systems and components

   As indicated in Item 2, equipment not typically located inside the reactor vessel and equipment within or in close proximity to the containment vessel may be subject to higher normal radiation levels than experienced in current plants. These conditions will need to be accounted for in the design and testing of these components to demonstrate that they will reliably perform their functions as assumed in the accident analyses.

8. Leakage detection capability to detect component or system leakage, including intersystem leakage

   Provisions for detection of component or system leakage will be similar to those in use in the current plants.

9. Accident monitoring

   The design of the iPWRs is such that core damage events will progress more slowly than for large LWRs. Radionuclide releases as a result of any core damage will occur later in the event. This will result in a reduced accident source term inventories due to decay of short-lived isotopes. Radioactivity monitoring provisions similar to those of large LWRs will still be needed to assess the amount of activity released.

iPWR Design and Operational Differences

Table 3-1 provides a summary of design and operation-related features that influence the source term or doses associated with a release and compares large PWRs with the iPWRs. **Table 3-2** provides a similar comparison of other-related features. The iPWR (Holtec, mPower, NuScale and Westinghouse) designs differ from each other and from the current operating fleet, AP1000, ESBWR, and APWR designs in ways that impact source terms under normal and accident conditions. Highlighted areas of these tables present characteristics of the iPWRs that differ from the current large PWRs and/or from each other.

Table 3-1. Comparison of Design and Operational Features between iPWRs and Large PWRs

Design Feature	Large PWRs	mPower	NuScale	Westinghouse	Holtec
Core Design					
Fuel Design < 5 weight per cent ^{235}U, as UO_2, Zirconium Alloy Cladding	Yes	Yes	Yes	Yes	Yes
Reduced Fuel Height Assemblies	No	Yes	Yes	Yes	No
Core Power (MWt per Unit)	~3400	530	160	800	525
Peak Fuel Burnup (MWD/MTU)	62,000	Comparable	Comparable	Comparable	Comparable
Coolant Outlet Temperature (°F)	~600	Comparable	Comparable	Comparable	Comparable
Operating Pressure (psia)	2250	Less than	Less than	Comparable	Comparable
Control Rod Drive Mechanisms	External	Internal	External	Internal	External
Burnable Poison Rods	Yes	Yes	Yes	Yes	Yes
Chemical Reactivity Control	Yes	No	Yes	Yes	No
Fuel Cycle Length (months)	18-24	48	24	24	48
Nuclear Steam Supply System (NSSS) Design					
Reactor Vessel, Steam Generator, Pressurizer Integrated into Single Vessel	No	Yes	Yes	Yes	Yes
Large Diameter Primary Reactor Coolant System Piping	Yes	No	No	No	No
Forced Core Cooling	Yes	Yes	No	Yes	No
Lowest Reactor Vessel Penetration Above Top of Fuel	No	Yes	Yes	Yes	Yes
Active Emergency Core Cooling System	Yes	No	No	No	No
Containment and Overall Plant Design					
Containment Location Underground	No	Yes	Yes	Yes	Partially
Submerged Under a Pool of Water	No	No	Yes	Yes	Partially
Containment Internal Volume	Large	Small	Small	Small	Small
Containment Design Pressure (psia)	60-70	Comparable	Higher	Higher	Higher
Containment Sprays	Yes	No	No	No	No
Containment Fan Coolers	Yes	No	No	No	No
Sub-atmospheric Pressure Containment	Varies	No	Yes	Yes	No
Secondary Containment or Confinement	Varies	No	No	No	No
Accident Source Term Release Elevation	Ground Level	Ground Level	Ground Level	Ground Level	Ground level
Control Room Proximity to Release	Varies	Close	Close	Close	Close

Note: iPWR differences from large PWRs are in grey.
PWR=pressurized water reactor; ^{235}U=uranium-235; UO_2=uranium dioxide; MWt=thermal megawatts; MWD/MTU=megawatt days per metric ton; °F=degrees Fahrenheit; psia=pounds per square inch absolute

Table 3-2. Comparison of Other Related Features between iPWRs and Large PWRs

Operations Feature	Large PWRs	mPower	NuScale	Westinghouse	Holtec
Spent Fuel Storage[a]	Wet/Dry	Wet/Dry	Wet/Dry	Wet/Dry	Wet/Dry
NSSS and Containment Inservice Inspection and Testing Frequency	18-24 months	48 months	24 months	24 months	48 months
Standard Plant - Number of Modules/Reactors	one	two	twelve	one	one
Single Control Room for Multiple Modules/Reactors	Varies	Yes	Yes	No	No
Normal Liquid Radwaste Releases	Yes	Yes	No	Yes	Yes
Normal Gaseous Radwaste Releases	Yes	Yes	Yes	Yes	Yes
Air-cooled Condenser	No	Available	Available	Available	Available

[a] This paper does not address dry spent fuel storage source terms.
NSSS=Nuclear Steam Supply System; PWR=Pressurized Water Reactor
Note: iPWR differences from large PWRs are in grey.

IV. Assessment of SMR Source Term Regulatory Issues

Introduction

While SMR source terms impact many facets of the plant operation, environmental impacts, shielding, safety, emergency planning, etc., the focus of this paper is in regard to the source terms to be used for accident dose analysis. That is, the amount of activity that might be released in the event of a postulated accident addressed in Chapter 15 of the Safety Analysis Report (SAR) or Design Certification Document (DCD), the timing of that release, and the resulting offsite and control room operator doses. It is recognized that a subset of accident dose analysis source terms are used in other licensing applications including equipment qualification, siting, and emergency response

SMR source terms are expected to fall into two principal categories for the range of applications evaluated by the NRC for design certification and individual site licensing approval (10 CFR 52 and 10 CFR 50, respectively). For some events, SMR design features or event progression are similar to those of large light water reactors (LWRs) and will result in a similar source term calculation methodology that is currently used and found acceptable by the NRC. This is denoted as the Category 1 source term methodology. The regulatory guide (RG) 1.183 alternative source term (AST) methodology will be used by SMR vendors for Category 1 source term development.

However, one of the goals in designing the SMRs is to create nuclear power plants that use improved and simplified operational and safety features that, in some cases, differ significantly from the features of the current LWR fleet. For those cases, the source terms will be replaced or revised as appropriate to reflect the design and expected transient behavior of these smaller and inherently safer designs. This is denoted as the Category 2 source term methodology. The source term for the loss of coolant accident (LOCA) is a prime example of the need for a Category 2 source term. Since SMRs have no large diameter piping, a large break LOCA cannot physically be postulated as the basis for the standard review plan (SRP) Section 15.6.5[4] analysis of site dose in comparison to 10 CFR 50.34 limits. Therefore, a large break LOCA also is not appropriate as the basis for calculating source terms for emergency response, post accident equipment qualification (EQ) doses or for post accident monitoring, sampling, and shielding. The large break LOCA also does not provide an appropriate basis for security considerations at an SMR. The SMR designers will develop one or more surrogate accident scenarios, denoted as source term design basis accidents (STDBA), that will meet the regulatory intent to address the maximum hypothetical accident[5] (MHA) as expressed in SRP 15.6.5 and 10 CFR 50.34 for the off-site and control room doses, EQ, post accident monitoring & sampling, and shielding, siting, security, and emergency planning. Although the design and safety aspects of iPWRs may preclude any accident scenario that results in substantial core meltdown and fission product release, it is recognized that the analysis of an appropriately determined MHA is necessary to demonstrate that engineered safety features (ESF) provide an acceptable level of protection to the public and control room operators. This section presents the methodology proposed for meeting both categories of source term applications for SMRs: RG 1.183 and new STDBA.

[4] Within the context of this paper, the use of the SRP is intended to also encompass design specific review standards (DSRS) for SMRs.

[5] The maximum hypothetical accident (MHA) is defined in 10 CFR 50.34, 10 CFR 50.67, and 10 CFR 100.11 in terms of control room habitability, accident source term, and siting, respectively with the following common language in the footnote to all three CFR sections, "Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

Category 1: Accident Analysis Using Current Source Term Methodology (Regulatory Guide 1.183)

For SMR designs, the methodology that will be used to analyze offsite and control room doses for many of the SAR Chapter 15 accidents is expected to be similar to that used for current generation large LWRs. The source terms for these dose accidents would be calculated using guidance in RG 1.183. The accident scenarios and the basis for their selection are delineated **Table 4-1**. The source term for each of these accident scenarios will utilize currently accepted practice and computer codes. This involves: (1) the calculation of fuel radionuclide inventory; (2) the definition of limiting values for concentrations of dose-equivalent ^{131}I activity in the primary and secondary (steam system) coolant and dose-equivalent ^{133}Xe in the primary coolant; (3) modeling of core response to events using core physics and system thermal-hydraulic transient calculations; and (4) component and system performance (e.g., valve flows).

Table 4-1. SMR Radiological Consequence Accidents Using RG 1.183 (Category 1 Source Term)

SRP Section [a]	Accident [b]	SMR Basis and Notes
15.1.5	Main steam line system pipe breaks (MSLB)	All designs have main steam systems, main steam isolation valves, and a secondary side coolant activity associated with leakage from the primary coolant system through the steam generator
15.3.3 and 15.3.4	Reactor coolant pump (RCP) rotor seizure or shaft break	Applicable to designs that use reactor coolant pumps (not applicable to the NuScale or Holtec designs)
15.4.8	Control rod ejection	Applicable to designs with control rod drives that are external to the reactor pressure vessel (not applicable to the mPower and Westinghouse designs)
15.6.2	Small primary coolant line break outside containment	All designs have small lines connected to the primary coolant system that penetrate the containment
15.6.3	Steam generator tube rupture (SGTR)	All designs have steam generators with tubes and an assumed primary coolant activity
15.7.4	Fuel handling accident (FHA)	All designs handle spent nuclear fuel under water in a pool (partially applies to Holtec since it doesn't handle individual fuel assemblies)
15.7.5	Spent fuel cask drop	Applicable to all designs because they will transport spent fuel from the spent fuel pool

[a] It is assumed that any SMR design specific review standard (DSRS) would either reference these SRP sections or provide a similar DSRS section. It is also assumed that SRP/DSRS section numbers would be identical and be located in the same numbered section in a design certification document (DCD).
[b] Although no longer specified in SRP Chapter 15, the source term from accidents involving the radioactive waste system are expected to be calculated using the same methodology as that used for large LWRs

Category 2: Accident Analysis Using Source Terms Specific to SMR Design Characteristics

For large LWRs, the large break LOCA has been used as the basis to calculate activity releases and doses for SRP Section 15.6.5, equipment qualification, and siting evaluation. The source term for the large break LOCA (as set forth in RG 1.183) is based on the presumption that, even with the presence and operation of safety-grade systems that preclude it, significant core damage occurs with specifically defined fractions of fission products in the fuel released at two different time intervals (defined as gap release and in-vessel release which assumes large-scale melting of fuel). The approach of applying the RG 1.183 core damage scenario for the small break LOCA is, however, conservative and ignores the benefits associated with SMR design features.

The SMR designs specifically preclude the presence of large diameter primary coolant piping and therefore any postulated large break LOCA scenario. These designs also utilize passive safety features and a primary coolant system that naturally extends the time before any core damage can occur due to larger water volumes per unit of reactor core power. Each SMR designer has three options in regard to assessing the maximum hypothetical accident (now no longer linked to a large break LOCA). One option is to utilize the existing methodology, as defined in RG 1.183, applying it to a small break LOCA despite the fact that the potential for significant core damage is much lower. Moreover, even if progression of the event to reach severe core damage would occur, the onset of core damage would be delayed by a longer time period than a large break LOCA in large LWRs. The second approach is to develop a source term specific to the SMR design; i.e., a mechanistic source term (MST)[6] that takes into account the differences between the SMR designs and the currently licensed LWRs. The third approach is to adopt a hybrid source term comprised of a combination of aspects of a mechanistic source term and the RG 1.183 source term.

The choice to use the RG 1.183 source term for application to an SMR radiological consequences analysis does not involve any deviation from currently accepted NRC licensing regulatory guidance since the release fractions and timing of the source term are prescribed. The only difference would be the smaller magnitude of the fission product inventory commensurate with the lower core power of SMR designs. This approach essentially replaces the large break LOCA with a small break LOCA and places it within the Category 1 source term approach.

Use of an MST will require that the SMR designer develop a design-specific severe accident model using a computer code such as MELCOR or MAAP. This approach would include an analysis of a bounding event or a range of event scenarios that lead to core damage as determined by the Level 1 PRA to determine an appropriate source term defining the calculated core release fractions and timing. This determination of a MST would follow the current assumption of multiple component failures that potentially cause the event to progress to a core damage scenario. NRC guidance for selecting scenarios to analyze for the MST, described in Appendix B for non-LWR designs (SECY 93-0092) and new plant licensing (SECY-05-0006), can be applied to SMR designs. Furthermore, the NRC has used a similar methodology in the recently completed SOARCA study that evaluated large LWRs using PRA event selection and MELCOR analysis to calculate source terms.

A hybrid approach uses the RG 1.183 source term, but modifies some aspects to account for significant differences in SMR transient response behavior such as timing of the onset of fuel damage. Both the hybrid source term and MST methodologies also require the development of an SMR probabilistic risk assessment (PRA) to identify events which result in core damage. The licensing and technical advantages and disadvantages of each approach are presented in **Table 4-2**.

Table 4-2. Comparison of STDBA Source Term Methodologies

Methodology	Advantages	Disadvantages
R.G. 1.183 Source Term	• Clear regulatory guidance specification • Technically simplest	• No credit for SMR source term ameliorating design features • Highest expected source term
Mechanistic Source Term (MST)	• Takes credit for SMR design features to reduce both magnitude and timing of source term • Lowest source term, but still conservative	• Technically most complex • Lowest regulatory certainty
Hybrid of RG 1.183 and MST Source Terms	• Some SMR design performance incorporated to increase timing • Less complex than MST • Better regulatory certainty than MST	• Does not credit all source term ameliorating design features

[6] As defined in SECY 93-092, "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."

The regulatory issue then is to address the licensing and safety intent of the MHA for SMRs. For SMRs, the large break LOCA with core melt will be replaced by one or more source term design basis accidents (STDBAs) which will provide the source term for calculating offsite and control room doses. This source term could also be used for equipment qualification, post accident monitoring, sampling, shielding, siting, security, and input to emergency planning and response[7].

The implementation of the MST or hybrid source term methodologies will provide a core damage scenario analogous to that in RG 1.183, but within the context of the SMR design features and transient response behavior. For example, if a small break LOCA with multiple ECCS component failures were to be selected, the timing of onset of core damage is likely to be significantly greater than that in R.G. 1.183 Table 4. This table specifies either a 30 second delay or, in the case of leak before break plants, a 10 minute delay before the onset of fuel gap release for pressurized water reactors (PWRs). SMR small break LOCAs are expected to result in longer time periods before gap release can occur. Similarly, the R.G. 1.183 LOCA early in-vessel release for PWRs is stipulated to commence at 0.5 hours after the LOCA, whereas for SMRs the progression to core melt is expected to occur more slowly. The longer time delay before releases begin and the longer time required for core damage to progress will be based on SMR design specific transient thermal-hydraulic analyses for a set of severe accident scenarios.

If an MST or hybrid ST methodology is chosen by the SMR designer for licensing application, selection of the SMR surrogate STDBA will be based on the results of a Level 1 design-specific PRA. It is expected that a range of low probability core damage sequences (i.e., 10^{-5} per year or lower) will be determined from the Level 1 PRA. The core damage sequences in the highest frequency bin will be evaluated to select one (or more) that results in the earliest time to core damage which could then be used as the surrogate STDBA.

If a hybrid source term methodology is chosen, the approach would be to use the delay times from a severe accident thermal-hydraulic transient analysis of the STDBA together with the R.G. 1.183 Table 2 PWR core inventory fractions. If an MST methodology is used, the STDBA severe accident transient analysis could be used to calculate both the timing and magnitude of fission product releases from the core.

In any of the source term approaches (the RG 1.183 source term, the MST, or the hybrid source term), unique SMR design-specific source term features could potentially be credited to reduce the release of activity to the environment. One example is scrubbing removal in the pool of water surrounding the containment vessel (NuScale and Westinghouse designs). Iodine source term reduction is identified in RG 1.183 Appendix B as an acceptable mechanism for the fuel handling accidents depending on water depth. The RADTRAD computer code includes a model for fission product scrubbing in pools of water with different depths (NUREG/CR-6604 Section 2.2.3). Another example is enhanced removal of particulate activity from natural processes inside the containment vessel due to the relatively small containment volume and the resulting large surface area to volume ratio as compared to that for large LWRs. Four natural in-containment aerosol deposition processes were recognized and discussed in NUREG-1465 Section 5.5. In addition, RG 1.183 Appendix A Section 3 allows for crediting containment natural deposition and references the NUREG/CR-6189 model that is used in the RADTRAD computer code as an acceptable model.

[7] It is recognized that the STDBA is only one of many inputs to the emergency response plan and emergency planning.

General Considerations on Source Term Application

All exclusion area boundary (EAB), low population zone (LPZ) and control room PWR accident dose limits delineated in Table 6 of RG 1.183 and GDC 19 will be applicable to SMRs regardless of which source term methodology is used.

SMR fuel is expected to be essentially identical to current large LWRs except for fuel assembly height (the one iPWR design exception to different height is the Holtec design), number of fuel assemblies, and possible grid or nozzle mechanical design and to have comparable levels of burnup and enrichment. As such, the calculation of core fission product inventory at end of life is expected to use methods similar to those currently approved by the NRC. Similarly, the use of the PWR design in the SMRs results in iodine spiking being modeled in the same manner as for large PWRs in calculating the increase in primary and secondary coolant iodine inventories in the event of accidents such as the Main Steam Line Break and the Steam Generator Tube Rupture accidents. Three of the four SMR designers are planning to use shorter height standard 17x17 PWR fuel assemblies that will operate at or below the range of thermal-hydraulic parameters (coolant flow, linear heat generation rate) of current large PWRs (exception is the Holtec design which is planning to use full height 17x17 PWR fuel assemblies). SMRs will assume similar levels of fuel failure as those observed in operating plants and will utilize similar limits on primary and secondary coolant concentrations of dose-equivalent ^{131}I and ^{133}Xe which will be delineated by technical specifications. Finally, for both the hybrid source term and MST methods, the chemical form of iodine released from the core will be assumed to be that identified in RG 1.183, given the similarities in fuel design (i.e., materials and operating thermal-mechanical conditions).

Potential for SMR Additional Accident Dose Analyses

The unique design, safety, and operational features of SMRs may result in the development of new accident scenarios that have not been specified in regulatory guidance for requiring dose analysis. For such new accident scenarios, the Level 1 PRA will be used to identify frequency of occurrence and then classify these accidents as either anticipated operational occurrences (AOOs), design basis accidents (DBAs), or severe accidents. Current RG 1.183 Table 6 and GDC 19 dose criteria will be applied, as appropriate to any new event scenarios. If a new scenario is classified as a severe accident, its frequency will be compared to that of the STDBA to determine if the new scenario should become the STDBA based on both frequency and consequences. If not assigned to the STDBA category, new SMR-unique accident scenarios will be incorporated into the severe accident analysis and PRA in SAR Chapter 19.

Potential SMR Design Features to Reduce Activity Releases for the STDBA

The SMR STDBA will incorporate SMR design features that affect radionuclide releases within the context of phenomena and behavior that have been extensively studied and accepted by the NRC such as natural process containment aerosol removal and pool scrubbing.

Since SMR designs include a below ground level containment with a overlying surface building, credit for dilution and deposition within the surface building and filtered release from the surface building could be taken within the context of RG 1.183 guidance for dual containments. The nature and extent of this credit will depend on specific SMR surface building and HVAC features.

By design, SMRs have a smaller containment volume as well as a smaller number and size of containment penetrations as compared to large operating LWRs. Analysis and testing may be used to justify a smaller containment leak rate than that selected for large LWRs which is currently as low as 0.1 weight percent per day. Again, any reduction in post-accident containment leak rate will depend on SMR design specific features.

Consideration of SMR Multiple Reactor Modules

The multiple module/reactor aspect of some SMR designs introduces the issue of whether the accident source term should include more than one module. The Level 1 PRA for each SMR design will evaluate scenarios, including those initiated by external events, in which there is a potential for the source to involve multiple modules. The frequency of events that result in core damage to more than one reactor module will be compared to the frequency of the STDBA developed for a single module that experiences core damage. If the beyond design basis accident frequency analyzed in the PRA is dominated by single module events, they will form the basis for the STDBA presented in SAR Chapter 15. Multi-module events will then be included in the severe accident PRA evaluation as described in SAR Chapter 19.

Consideration of SMR Source Term Uncertainties

The evaluation of SMR source term uncertainties will utilize several complementary, but independent, approaches. First, since SMRs use similar nuclear fuel designs and coolant systems as current large LWRs, insights from recent severe accident research, fission product experiments, and large LWR experience will form a basis for determining source term uncertainties. In addition, sensitivity studies will be performed with the computer codes used to calculate (e.g., MELCOR or MAAP) source terms to quantify the effect of inputs on source term magnitude and timing. Finally, SMR vendor-specific integral scale and component test facility data will be used along with engineering knowledge acquired from extensive system, core, and fuel transient thermal-hydraulic and mechanical analyses that will be performed to complete the design and support the license application. A summary comparison of source term technical aspects of the three STDBA methodologies is presented in **Table 4-3**.

Table 4-3. Technical Aspects of STDBA Category 2 Source Term and Mitigation Methodologies

Technical Aspect	RG 1.183	Mechanistic Source Term	Hybrid Source Term
Iodine chemical form	RG 1.183	RG 1.183	RG 1.183
STDBA gap release magnitude	RG 1.183	Design specific PRA and thermal-hydraulic analysis	RG 1.183
STDBA gap release timing	RG 1.183	Design specific PRA and thermal-hydraulic analysis	Design specific PRA and thermal-hydraulic analysis
STDBA in-vessel release magnitude	RG 1.183	Design specific PRA and thermal-hydraulic analysis	RG 1.183
STDBA in-vessel release timing	RG 1.183	Design specific PRA and thermal-hydraulic analysis	Design specific PRA and thermal-hydraulic analysis
Need for source term uncertainty analysis	No[a]	Yes	Yes
Need to consider new single module STDBA events	Yes[b]	Yes	Yes
Need to consider multiple module STDBA events	Yes[b]	Yes	Yes
Design specific containment leak rate	Yes	Yes	Yes
Design specific containment fission product aerosol deposition	Yes	Yes	Yes
Design specific surface building fission product deposition	Yes	Yes	Yes
Submerged containment fission product removal	Yes	Yes	Yes

[a] If new STDBA events are identified, then source term uncertainty analysis will be required.
[b] Since iPWR SMRs constitute a change in design as compared to large LWRs, justification for use of RG 1.183 will be required.
RG=NRC Regulatory Guide; STDBA=Source Term Design Basis Accident; PRA=Probabilistic Risk Assessment

Identification and Resolution of Potential Regulatory Issues

Potential regulatory issues for SMR source term development were evaluated within the context of their application to SAR Chapter 15 accident dose analyses, siting, EP, EQ, and security. SMR source terms were classified into Category 1 that conforms to currently accepted NRC regulatory practice (i.e., RG 1.183) and Category 2 that differs from RG 1.183. Category 1 source term applications are not expected to present any significant regulatory issues while Category 2 applications may present some regulatory issues. **Table 4-4** lists regulatory issues associated with Category 2 source term methodologies and the approach to resolution of these issues.

Table 4-4. Resolution of Category 2 Source Term Methodology and Mitigation Regulatory Issues

Regulatory Issue	Resolution
Selection of STDBA as LOCA surrogate	Use PRA to identify highest frequency core damage events; detailed analysis results to be used in either hybrid source term method accounting for source term timing differences or in MST method accounting for both source term magnitude and timing
Reduced containment leak rate	Perform containment testing and/or analysis to justify lower leak rates
Increased containment aerosol deposition	Use methodology in RG 1.183 Appendix A Section 3.2[a] for small volume and large surface area to volume ratio to justify increased credit for deposition (e.g., NUREG/CR-6189)
Reactor building dilution and deposition[c]	Use methodology in RG 1.183 Section 4[a] using building design features if SMR can take credit for building
Submerged module water scrubbing	Use methodology in RG 1.183 Appendix B[a] and NUREG/CR-6604 Section 2.2.3 to credit water fission product removal[b]
New unique module accident scenarios	Use PRA to quantify frequency and compare to STDBA in terms of frequency as well as extent and timing of core damage source term
Multi-module accident scenarios	Use PRA to quantify frequency and compare to STDBA in terms of frequency as well as extent and timing of core damage source term
Source term uncertainty analysis	Severe accident research, fission product experimental data, large LWR experience, behavior knowledge for SMR-specific system, core, component, and fuel test data as well as transient thermal-hydraulic analyses (e.g., MELCOR or MAAP computer codes)

[a] As an alternative to RG 1.183, other methodologies that were previously reviewed and approved by the NRC may also be used.
[b] Modeling water scrubbing is complicated by the fact that containment leakage paths into the pool may encompass a range of locations with different submergence depths.
[c] The iPWR reactor building, although not a containment structure as in current large PWRs, will have an extremely robust design to meet aircraft impact, fire safety, and security design and licensing requirements.

Some of the issues identified in Table 4-4 would benefit from additional study and evaluation of existing relevant experiment and test data. These issues are:

- small containment aerosol deposition;
- small piping fission product deposition;
- small containment penetration leak rate testing;
- reactor building fission product dilution and deposition (without safety grade HVAC or technical specifications on leak rate limits);
- submerged containment leakage aerosol removal

These five issues were ranked in terms of their relative importance to iPWR SMR source terms and the type of accidents that they would most likely affect resulting in **Table 4-5**. The most important source term issues that would affect the STDBA and significantly affect calculated doses are: (1) containment aerosol deposition and (2) reactor building fission product dilution and deposition. These two higher importance ranking issues affect both design basis and beyond-design-basis accidents and apply to all four iPWR designs. These issues are also judged to have the greatest impact on potential source term reduction for iPWRs. The remaining issues involving small piping fission product deposition, containment leak rate testing and submerged containment aerosol removal were judged to have a low priority because either: (1) they do not apply to all iPWR designs; (2) their expected effect on reducing the source term is not as large as that of the higher importance issues; (3) the state of knowledge on these subjects is mature; or (4) significant technical, analytical, and testing aspects of design may require unique individual iPWR SMR vendor solutions.

Table 4-5. Ranking of Source Term Issues for Future Research and Evaluations

Issue	Impacts STDBA	Impacts BDBA[a]	Impacts all 4 iPWR Designs	Importance Ranking
containment aerosol deposition	Yes	Yes	Yes	High
small piping fission product deposition	No	Yes	Yes	Low
small containment penetration leak rate testing	Yes	No	Yes	Low
reactor building fission product dilution and deposition	Yes[b]	Yes	Yes	Medium
submerged containment leakage aerosol removal	Yes	No	No	Low

[a] This determination is based on the assumption that beyond design basis accidents would involve containment bypass events.
[b] Although it can affect the source term, it is recognized that the fact that the reactor building will not have leak rate limits or a safety class filtered ventilation system will affect the regulatory acceptability of this issue for the STDBA.
Highest priority and importance ranked issues are in grey.
STDBA=Source Term Design Basis Accident; BDBA=Beyond Design Basis Accident

V. Summary

Source term regulations, their range of applications in nuclear power plant licensing, and regulatory issues unique to SMRs have been presented and evaluated in this paper. The requirements in 10 CFR Part 50 and Part 100, SECYs from 1993 through 2011,and NUREG-0800 standard review plan provide regulatory requirements, guidance, and background for source term development and application that can be used by SMRs. The NRC has recognized that the use of appropriately developed mechanistic source terms for accident analysis can assure regulatory compliance. Moreover, it is acknowledged that unique design features of SMRs may warrant a different approach to source terms than that used for large LWRs. Unique features that differentiate the iPWR SMRs from large LWRs include:

- lower core thermal power;
- shorter height fuel assemblies (except for the Holtec iPWR design);
- elimination of large primary coolant pipes;
- integral NSSS with reactor vessel design and primary coolant water volume that reduces the potential for core uncovery;
- control room location underground and in close proximity to the reactor building;
- fewer and smaller containment penetrations;
- elimination of active containment post-accident systems (i.e., spray and fan coolers);
- greater reliance on passive safety systems; and
- smaller containment volume.

For iPWR SMRs, most Chapter 15 radiological accidents will be analyzed using the identical source term methodology as many large LWRs, denoted as Category 1, with the exception of the large break LOCA. The large break LOCA constitutes a physical impossibility for an iPWR SMR, thus requiring a surrogate, the source term design basis accident (STDBA), which meets the regulatory intent of challenging the Engineered Safety Features with an intact containment and is denoted as Category 2. Three STDBA approaches are proposed: RG 1.183, mechanistic source term (MST), or a hybrid of RG 1.183 and MST.

Potential SMR unique source term release and mitigation regulatory issues and proposed resolutions were identified. Some of these issues may benefit from additional studies and/or evaluation of existing relevant experiment and test data. These issues are: small containment aerosol deposition, small piping fission product deposition; small containment leak rate testing; and fission product dilution and deposition in the reactor building (without safety grade HVAC or leak rate limits. The highest importance and priority issues are: containment aerosol deposition and reactor building fission product dilution and deposition both of which are important to all iPWR SMR designs, affect both STDBAs and beyond design basis accidents, and are expected to have a significant impact on calculated offsite and control room doses.

The planned next step for this paper is to engage the NRC in discussions encompassing the key SMR source term issues and their resolution as discussed herein. These interactions are intended to reach agreement on key SMR source term assumptions and principles thereby providing greater regulatory certainty to SMR designers and licensing applicants using either 10 CFR 50 or 10 CFR 52.

Appendix A
Uses of Source Terms in the Safety Analysis Report

Table A-1. Radionuclide Source Terms Considerations for SMRs[1]

SAR Chapter	Source Term Considerations	SMR Considerations
1. Introduction	N/A	
2. Site Characteristics	• Short-term atmospheric dispersion estimates for accident releases (2.3.4) • Long-term atmospheric dispersion estimates for routine releases (2.3.5)	• Offsite dispersion models are not changed • Reduced site size results in increased offsite dispersion factors with a resulting increase on offsite doses for the same source term as large LWRs • A more compact nuclear island may result in elevated control room intake dispersion factors with a resulting increase in control room doses (also, the potential for challenge to the current models for calculating dispersion factor for control room)
3. Design of SSCs	• Environmental qualification of mechanical and electrical equipment (3.11)	• Same methodology • In-containment SSCs – smaller containment volume results in higher accident pressures and temperatures (NuScale and Westinghouse) and reduced spacing from high radiation sources (less shielding) • Steel containment vessel – higher doses to areas outside the containment vessel • Seismic considerations (due to embedment depth)
4. Reactor	• Fuel system design; fission product inventory (4.2) • Nuclear design; reactivity insertion accidents (4.3) • CRD structural materials (4.5.1)	• Integral PWRs • Fuel designs are similar to those of large LWRs (<5% uranium-235 enrichment oxide fuel, similar operating cycles) • Core fission product inventories much smaller because of lower power level • Possibly some (small) differences in relative levels of specific fission products with extended (4 years for mPower design) operating cycles ○ Total core offload vs. fuel shuffle

SAR Chapter	Source Term Considerations	SMR Considerations
		○ Different burnup patterns • Environmental conditions on internal CRDMs • Smaller core may be more sensitive to fuel misload or inadvertent rod withdrawal
5. RCS and Connected Systems	• RCPB leakage detection; inter-system leak paths (5.2.5) • RV materials (5.3.1) • P-T limits; radiation effects on RCS components (5.3.2)	• Placement of RCS components within integral reactor system may impact radiation effects on components and change pathways for radionuclide release • Radiation effects on NSSS components
6. ESF	• ECCS; performance in radioactive environment (6.3) • Control room habitability (6.4) • Fission product removal and control systems (6.5; 6.5.3) and ESF atmosphere cleanup systems (6.5.1)	• Performance in radioactive environment • Containment response (compact, high pressure, close coupled with RCS) • Event sequence/timing due to containment response • Subatmospheric inside containment (hydrogen control) • Location of control room and air intake relative to release locations
7. I&C	• Accident monitoring instrumentation (7.1-A and 7.5) • Containment isolation (7.1-A)	• Compact location (placement of instrumentation) ○ Higher fluences (less shielding) • More severe containment isolation conditions due to higher pressures and temperatures associated with a smaller containment volume • Difference in parameters to be monitored during refueling?
8. Electrical	• Station Blackout coping duration (8.4)	• No dependence on safety grade AC • Long term cooling differences • Post-Fukushima ○ Multi-module implications ○ FLEX
9. Auxiliary Systems	• Mitigating the radiological consequences of a criticality accident (9.1.1) • Suitable shielding from spent fuel (and new fuel if recycled fuels are used) storage	• Location and storage of lifetime inventory of spent nuclear fuel ○ Pool storage versus dry storage

SAR Chapter	Source Term Considerations	SMR Considerations
	(9.1.2) • Pool building radiation monitoring to protect personnel, to prevent significant offsite radiation doses (9.1.2) • Procedures and engineering controls based on sound radiation protection principles to achieve ALARA occupational doses and doses to the public (9.1.2) • Removal of corrosion products and radioactive materials from SFP water to reduce occupational exposures (9.1.3) • Handling of fuel and spent fuel, which, if dropped, mishandled, or damaged, could cause releases of radioactive materials or unacceptable personnel radiation exposures (9.1.4) • The potential release of radioactive materials from damage to irradiated fuel, a criticality accident, or damage to essential safe-shutdown equipment could cause unacceptable radiation exposures. (9.1.5) • Provisions for detection, collection, and control of system leakage and the means for detecting leakage of radioactivity from one system to another and preclude its release to the environment (9.2.2; 9.2.6; 9.3.1; 9.3.2; 9.3.3) • Process and post-accident sampling (9.3.2) • Control room air filtration system (9.4.1) • Permit personnel access, and control the concentration of airborne radioactive material in plant systems/locations requiring personnel access (9.4.1; 9.4.2; 9.4.3; 9.4.4) • Radiation release to any unrestricted area due to the direct effects of fire suppression activities (9.5.1.2)	○ Available fuel pool water inventory • Unique systems interfaces ○ Radwaste systems interfaces • Unique spent fuel handling ○ RV and NSSS components in fuel handling area ○ Full core offload capability ○ Passive spent fuel pool cooling and instrumentation ○ Height of water above spent fuel storage racks (impact on pool scrubbing for fuel handling accident) ○ Cask drop height • Water chemistry • Fire protection ○ Small space, room for adequate separation of safety divisions/trains ○ Number/availability of personnel ○ Access for fire suppression • Access to areas for maintenance/inspection ○ Shielding, separation
10. Steam and Power Conversion System	• Radiation protection design features, expected radiation levels and degree of access during operation (10.2) • Control the release of radioactive materials to the environment (10.3; 10.4.1; 10.4.2; 10.4.3) • Transfer radioactive gases to the gaseous waste processing system or ventilation exhaust systems (10.4.2)	• No difference in approach anticipated
11. Radioactive Waste Management	• Determination of fraction of fission products released and concentrations of radioactive isotopes in the reactor coolant (11.1) • Monitoring and control of effluents from applicable release points (11.1; 11.5) • Decontamination factors for inplant control measures (11.1)	• Potential difference in on-site management of radwaste inventories ○ Elimination of need for liquid radwaste discharge during normal operations (NuScale design)

SAR Chapter	Source Term Considerations	SMR Considerations
	• Design objectives for doses in unrestricted areas (11.1) • Calculating effluent source terms and releases of radioactive materials in liquid (11.2) and gaseous effluents (11.3) • Impacts of releases of radioactive liquids in ground or surface water (11.2) • Review of the consequences of a liquid tank failure having the potential to release radioactive materials to a potable water supply (11.2; BTP 11-6) • Facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste (11.3) • Design considerations for the use of shielding around portions of sampling equipment (11.5) • Digital computer software used in radiation monitoring and sampling equipment, including software used to terminate or divert process and effluent streams (11.5) • Waste gas system failure (BTP 11-5) • Liquid tank failure (BTP 11-6)	• Use of advanced shielding materials • Use of materials, coatings and operating procedures to minimize decommissioning wastes • Use of automation to minimize personnel exposure time and location • Use of rad waste tank cubicle steel liners to eliminate the liquid tank failure accident as provided for in DC/COL-ISG-013; item 2, Mitigating Design Features.
12. Radiation Protection	• Ensure radiation exposure for personnel is maintained as low as is reasonably achievable (12.1; 12.3-12.4) • Radiation sources affecting inplant radiation protection (12.2) • Coolant and corrosion activation products source terms should be based on applicable reactor operating experience (12.2) • Shielding for each of the radiation sources (12.3-12..4) • Personnel protection features incorporated in the ventilation system designs (12.3-12.4) • Fixed area radiation and continuous airborne radioactivity monitoring instrumentation for normal operation, anticipated operational occurrences, and accident conditions, including the criteria for placement (12.3-12.4) • Dose assessment and dose-reducing measures (12.3-12.4) • Radiation protection facilities and equipment (12.5)	• Use of advanced shielding materials • Smaller gamma sources due to reduction in large pipes and valve sizes • Use of automation to minimize personnel exposure time and location • Small nuclear island and containment size may complicate shielding design and accessibility
13. Operational Programs	• Protective actions for severe reactor accidents (13.3) • Capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials (13.3)	• No difference in approach anticipated • Location of TSC and air intake relative to release locations

SAR Chapter	Source Term Considerations	SMR Considerations
	• Instrumentation to measure effluents at all potential, accident release points (13.3) • Technical support center habitability (13.3)	
14. Initial Test Program and ITAAC	• Radioactive waste systems review, identification of all expected releases of radioactive effluents, methods of treatment, methods used in calculating effluent source terms and releases of radioactive materials in the environment, and operational programs in controlling and monitoring effluent releases and for assessing associated doses to members of the public (14.3.7) • Verification of radiation shielding, confinement or containment of radioactivity, ventilation of airborne contamination, or radiation (or radioactivity concentration) monitoring for normal operations and during accidents (14.3.8)	• No difference in approach anticipated

15. Safety Analysis	• Radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1) and 10 CFR Part 100.21 (15.0.3)
	• Offsite radiological consequences of postulated DBAs (15.0.3)
	• Control room and technical support center habitability (15.0.3)
		• Smaller core fission product inventories
		• Different transport pathways
		• Sensitivity of smaller core to misload of fuel assembly or inadvertent rod withdrawal
		• Standard methods for analyzing radiological consequences of most accidents
		  ○ SGTR
		  ○ MSLB
		  ○ FHA
		  ○ Rod ejection (if applicable)
		  ○ Locked rotor (if applicable)
		  ○ Radwaste system failure
		  ○ Small line break outside containment
		• AST/hybrid approach for radiological consequences of maximum hypothetical accident
		  ○ No large break LOCA
		  ○ LOCA (e.g., delayed onset of core damage)
		• Potential for new events

16. Tech Specs	• Administrative controls and safety limits related to radiological controls and accident radiological consequence analysis (16.0)	• Administrative controls for refueling • Component surveillance, ISI/IST • Technical Specification Limiting Conditions for

SAR Chapter	Source Term Considerations	SMR Considerations
		Operation for I-131 concentration in primary and secondary systems.
17. QA	N/A	
18. HFE	N/A	

19. PRA	• Recognize that there is a point of diminishing returns in risk reduction and that some residual risk will be associated with plant operation (19.2)
	• Do the selected source term categories adequately represent the revised containment event tree (CET) endpoints? (19.2)
	• Does the application affect the timing of release of radionuclides into the environment relative to the initiation of core melt and relative to the time for vessel rupture? (19.2)
		• Component reliability data for FOAK components
		• Common cause failures
		• Multi-module issues
		• Passive mechanisms
		  ○ Higher pressure condensation within containment
		  ○ In-containment behavior of fission products
		• Containment leak rate
		• Lower CDF vis-à-vis ROP and LARs
		• Delayed onset of core damage

[1] Based on proposed integral PWR designs

Appendix B
Mechanistic Source Term Definition

There is not a clear definition of what constitutes a "mechanistic" source term. 10 CFR § 50.67 uses the terminology *alternative source term* due to its comparison with the terminology as described in TID-14844 and a series of early Regulatory Guides. For new reactors, the phrase *mechanistic source term* was introduced during evolutionary and advanced reactor design reviews in the 1980's and 1990's. SECY-93-092, *Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and Canadian Deuterium Uranium Reactor (CANDU) 3 Designs and Their Relationship to Current Regulatory Requirements,* provides an example in which the following phraseology was specifically used:

> "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."

Three conditions were described in SECY-93-0092 for a *mechanistic source term* approach to be found acceptable.

> "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 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."

A parallel policy paper, SECY-93-087, *Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,* also discussed the use of a modern source term approach but used the phraseology *physically-based source term*. In similar fashion, SECY-94-302, *Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs,* and SECY-97-020, *Results of Evaluation of Emergency Planning for Evolutionary and Advanced Reactors,* also described the concept in wording that used *physically-based source term* phraseology. Other SECYs in this period described various aspects of the approach but, as with SECY-93-087, described the concept without using the mechanistic wording (i.e., SECY-90-0016 *Evolutionary Light Water Reactor (LWR) Certification Issues And Their Relationship To Current Regulatory Requirements* used "updated source term" while SECY-94-302 used "new" and "revised" wording when describing the accident source term. In discussing severe accident risk, both SECY-97-132 and SECY-97-171 *Consideration of Severe Accident Risk in NRC Regulatory Decisions* used "revised"). Thus, a consistent and precise definition of what constitutes a *mechanistic source term* does not appear in the regulatory history.

SECY-93-0092 also discusses source terms in conjunction with review areas other than accidents (in relation to existing LWR requirements).

> "Appendix I to 10 CFR Part 50 (ALARA), 10 CFR Part 100 (Reactor Site Criteria, which references the Technical Information Document (TID) 14844 source term), and 10 CFR Part 20 (Standards for Protection Against Radiation) all have limitations on releases related to power plant source terms.
>
> General Design Criterion (GDC) 60 requires that the design includes means to control suitably the release of radioactive materials in liquid and gaseous effluents and to handle waste produced during normal operations and anticipated operational occurrences."

The Commission approved the approach described by the NRC staff in SECY-93-0092 as well in the parallel SECY papers on evolutionary and advanced reactor designs. Of note is a cautionary insight provided by the Commission in their approval for use of the concept. In its Staff Requirements Memorandum (SRM) on SECY-93-0092, the Commission noted:

> "The Commission approves the staff's recommendations including its agreement with the ACRS. Commissioner Rogers questions whether there is sufficient information on each specific reactor design and fuel design extant to enable the staff's three conditions for a mechanistic analysis to be met. He believes that a mechanistic "scenario specific" source term for each reactor concept warrants further consideration before evaluating the acceptability of the design."

After SECY-93-0092, the next appearance of the phrase *mechanistic source term* in SECY papers does not occur until recently in SECY-10-0034, "Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs."

Current Policy on the Use of a Mechanistic Source Term

Following the introduction of 10 CFR § 50.67 for operating reactors, the discussion on source terms continued but with focus on non-LWR use. SECY-02-0139, *Plan for Resolving Policy Issues Related to Licensing Non-Light Water Reactor Designs,* discussed the use of *scenario-specific accident source terms* in conjunction with review of containment and site suitability for the pebble-bed modular reactor (PBMR) design. A contrast in approach between that of operating LWRs and future plants was described as:

> "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 and DBEs 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. Also, the LWR source terms represent a composite of a

number of LWR core melt scenarios and bound a number of accident scenarios. Therefore, the dependence of the analysis on precisely understanding the fission product release characteristics of individual accident scenarios is reduced. However, it should also be mentioned that a limited number of scenario-specific source terms are used in LWR licensing (e.g., reactivity insertion accidents)."

SECY-03-0047, *Policy Issues Related to Licensing Non-Light-Water Reactor Designs,* asked the question: "Under what conditions, if any, should scenario-specific accident source terms be used for licensing decisions regarding containment and site suitability?" stating:

"The staff recommends that the Commission take the following action:

- Retain the Commission's guidance contained in the July 30, 1993, SRM[8] that allows the use of scenario-specific source terms, provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties.

This recommendation will allow credit to be given for the unique aspects of plant design (i.e., performance-based) and builds upon the recommendation under Issue 4[9]. 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."

The Commission approved the staff's recommendation for use of *scenario-specific source terms* in licensing decisions. 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,* summarizes the approach.

"Scenario specific source terms may be used for licensing purposes (e.g., siting) providing the following are met:

- the scenarios to be used for the source term evaluation should be selected from a design specific probabilistic risk assessment, with due consideration of uncertainties.
- the source term calculation, using the selected scenarios, should be based upon analytical tools that have been verified with sufficient experimental data to cover the range of conditions expected and to determine uncertainties.
- the source terms used for licensing decisions should reflect the scenario specific timing, form and magnitude of radioactive material released from the fuel and coolant. Credit may be taken for natural and/or engineered attenuation mechanisms in estimating the release to the environment, provided there is adequate technical basis to support their use.
- The source terms used for assessing compliance with dose related siting requirements should be 95% confidence level values based upon best estimate calculations with quantified uncertainties. Where uncertainties cannot be quantified, engineering judgment shall be used.
- the source terms used in assessing emergency preparedness should be mean values based upon best estimate calculations with quantified uncertainties.

[8] SRM-93-092
[9] Issue 4: To what extent can a probabilistic approach be used to establish the licensing basis?

> The above guidance is intended to provide a flexible, performance-based approach for establishing scenario specific licensing source terms. However, it puts the burden on the applicant to develop the technical bases to support their proposed source terms. Applicants could, however, propose to use a conservative source term for licensing purposes (in order to reduce research and development costs and schedule), provided the use of such a source term does not result in design features or operational limits that could detract from safety. Finally, it should be noted that the use of scenario specific source terms may result in smaller source terms being used for siting purposes then traditionally used for LWR siting.
>
> In developing technology-specific regulatory guides, the staff may propose acceptable conservative source terms(s), if it is feasible to do so.

Issue 3.3 in SECY-10-0034, *Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs.* further describes the topic, noting for integral PWRs:

> "Accident source terms are used for the assessment of the effectiveness of the containment and plant mitigation features, site suitability, and emergency planning. Other radiological source terms are used to show compliance with regulations on dose to workers and the public. The Commission has previously deliberated on the use of design-specific and event-specific source terms, provided there was sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgment was used to bound uncertainties. The source terms for the integral PWRs may be based partly on source term information from current generation LWRs and insights gained from extensive state-of-the-art fission product experiments conducted to understand accident phenomena including fission product transport and release."

The SECY then goes on to describe potential regulatory issues associated with this approach. NRC's recent memorandum, *Status of Staff Activities To Address Mechanistic Source Term Methodology and Its Application to Small Modular Reactors,* (ML113410366) dated December 29, 2011, summarizes:

"A mechanistic source term could contribute to the staff's evaluation in a number of areas (e.g., siting, control room habitability, emergency preparedness, and security considerations). Pertaining to emergency preparedness, the staff recently described the development of an emergency preparedness framework in SECY-11-0152, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," dated October 28, 2011 (ADAMS Accession No. ML112570439). A key factor in developing that framework is the determination of offsite dose considerations and the staff-described elements that would be involved in the development of an "appropriate method" for use in the framework. As noted in SECY-11-0152, the staff anticipates that industry will develop a proposed detailed calculation method to support the framework and the staff, as warranted, will identify and budget work to confirm the acceptability of the industry approach."