From: Richards, Andrew
To: RulemakingComments Resource
Cc: Vietti-Cook, Annette; Chazell, Russell; Beall, Bob; Schrader, Eric; Baranwal, Rita; Tilden, Jay; Tullis, Cathy; Boyd, Dallas
Subject: [External_Sender] DOE Comment to Docket ID NRC-2015-0225
Date: Friday, September 25, 2020 3:33:34 PM
Attachments: 2020.09.25 - Letter - DOE NNSA Response to NRC SMR EPZ.pdf

Good afternoon,

Attached is a comment from Dr. Rita Baranwal, Assistant Secretary of Nuclear Energy at the U.S. Department of Energy; and, from Mr. Jay Tilden, Associate Administrator & Deputy Under Secretary at the National Nuclear Security Administration, to the U.S. Nuclear Regulatory Commission's Proposed Rule for Emergency Preparedness for Small Modular Reactors and Other New Technologies (Docket ID NRC-2015-0225). Thank you.

Andrew Richards
Chief of Staff
Office of Nuclear Energy
U.S. Department of Energy

Department of Energy
Washington, DC 20585

September 25, 2020

The Honorable Kristine L. Svinicki
Chairman
Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, Maryland 20852

Dear Chairman Svinicki:

This letter is sent as a follow-up to the Department of Energy's (DOE) May 22, 2020, letter from the Office of Nuclear Energy (NE) to your office in support of the Nuclear Regulatory Commission (NRC) staff's rulemaking efforts for a technology-neutral, dose-based, consequence-oriented emergency preparedness (EP) framework for small modular reactors (SMR) and other new technologies (ONT). NE and the National Nuclear Security Administration (NNSA), seek to reinforce the official DOE position as expressed in the May 22, 2020 letter from NE.

A key component of NE's mission is to assure the availability of safe and economic nuclear energy generation options for the United States, such as SMRs and other advanced reactor designs. NE's research and development investments are focused on assuring that these technologies can provide the nation's energy needs while assuring a level of health and safety consistent with the requirements established by the NRC. We acknowledge the role of NRC in reviewing and evaluating the advanced reactor license applications to assure they can be constructed and operate safely. Both NE and NNSA agree reduced emergency planning zones (EPZs) acknowledge the advancement in safety and the reduced source terms of SMRs and ONTs. In addition, DOE notes that the NRC is required by the Nuclear Energy Innovation and Modernization Act, Public Law No. 115-439 section 103, to develop new processes for licensing nuclear reactors. NE and NNSA are fully supportive of NRC's efforts in this regard. The discussion below provides clarification regarding the DOE position on issues raised in several topical areas.

Regarding EPZ size, it is DOE's position that the new NRC approach to emergency preparedness for SMRs and ONTs will assure public safety because it would establish EPZs based on the calculated doses from the design-specific attributes of the particular nuclear facility to be licensed. The one rem total effective dose equivalent (TEDE) criterion proposed in the new rule is as strict as the dose thresholds that were used to establish the ten-mile exposure pathway EPZ for large water-cooled nuclear power plants in NUREG-0396.

In regards to defense-in-depth (DID) against nuclear accident scenarios, it is DOE's position that, under the approach outlined in the proposed rule, emergency planning will continue to provide the last layer of DID for low-probability accidents and the size of the EPZ will be commensurate with the magnitude and probability of potential consequences from the SMRs and ONTs, as it is now for large light water reactors (LWR) in NUREG-0396. Only by meeting the strict standards of the proposed rule (1 rem TEDE for a spectrum of credible accidents) would a licensee be able to reduce the size of their EPZ below the default 10- and 50-mile EPZs. To accomplish this, reactor designers will be required to increase the level of protection afforded by all layers of DID, specifically accident prevention and mitigation measures, thereby preventing significant radiological releases from occurring. The proposed approach is expected to provide equivalent or better protection over a similar, wide spectrum of very unlikely nuclear accident scenarios through fundamental design of technologies and facilities in addition to mitigation strategies. As an example, the recently-approved Clinch River early site permit EPZ methodology encompasses event sequences with a probability of once in 10 million years ($1x10^{-7}$); compared to more probable accident sequences of once in 100,000 years (on the order of $1x10^{-5}$) that were the principal basis of the 10 mile EPZ established by NUREG-0396. This is a much more conservative approach than that provided under NUREG-0396 and a simple 10-mile EPZ designation.

Regarding the lack of operational history for SMR technologies, it is the position of DOE that it is NRC's mandate and charter to assess the potential source term, potential accident scenarios, and ultimate safety of any new nuclear technologies. NRC's exercise of this mandate thereby protects the public and ensures the safety of the nuclear facility. NRC has the capability, and the responsibility, to do this for existing technologies as well as for new technologies with little or no operational history. As with all reactors and new nuclear technologies, source terms for SMRs are determined analytically and conservatively, based on physical processes and characteristics of the technology design. The likelihood or not of accident scenarios is based on probabilistic risk assessment (PRA) using operational data from similar systems and components at other facilities. Further, the state of PRA technology and the quality of results have improved since the generation of PRA data that was used to develop NUREG-0396. In this sense, it is DOE's position that NRC's proposed approach is more conservative than the generic 10-mile EPZ currently required for a facility of any size. In addition, Draft Regulatory Guide 1350 requires identification and characterization of the specific hazards posed by multi-modular units, evaluation of the impacts of these hazards, and requires description of the planning or emergency response functions that will mitigate the impacts of the identified multi-module hazards.

NRC's proposed rule is robust and provides guidance for using the integrated decision making process, and that it should consider the defense-in-depth philosophy, maintain sufficient safety margins, and include treatment of uncertainties. In addition, an applicant should justify that the PRA performed is acceptable for this use, and that it considers internal and external hazards, all modes of operation, and all significant radionuclide sources. The PRA should also include event sequences involving single or multiple modules/units, if applicable, to provide useful risk insights into the source term selection process.

The Department believes that if SMRs and ONTs can be shown to be lower consequence designs and can provide the same levels of protection with smaller EPZs, they should be treated differently; this will have a significant positive impact on the economic viability and eventual commercialization and deployment of the new advanced nuclear plants.

Regarding Ingestion Pathway EPZ (IPZ) requirements, the Department does not believe that the proposed rule would weaken or eliminate the intent of the IPZ requirements. The proposed approach to accomplishing the intended level of safety performance is different, but the intended safety outcome is not. The approach allows applicants the flexibility to propose strategies to assure that the intended safety outcome is accomplished, and to make the case to NRC that their approach can and will succeed. It is DOE's position that under this approach, the NRC will maintain its ability to review an applicant's technology and ensure that the public is protected and safety maintained.

Further, the Department believes that the NRC's proposed rule and guidance on federal, state, local, and tribal capabilities for contamination interdiction requires the licensee to demonstrate external capabilities that support specific required functions necessary for interdiction [DG-1350, p. 9, 3.a. and 3.b]. The licensee must have and demonstrate the ability to recommend protective actions to offsite authorities, make notifications to the external organizations with the necessary interdiction capabilities, and monitor and assess radiological conditions to support taking those protective actions, and maintain the staffing necessary to implement these functions. It is the Department's position that together, these requirements ensure that the capabilities for ingestion pathway mitigation are provided and can be effectively utilized.

It has been suggested that the proposed rule should incorporate requirements that consequence analysis be performed for low-probability events, security considerations, combined emergency scenarios, and other beyond-design-basis events to inform the ultimate EPZ determination. The existing NRC framework does not require these types of "consequence" analyses to demonstrate the effectiveness of the 10- and 50- mile EPZs for a new design. The proposed rule would require SMRs or ONTs to develop a documented evaluation of the consequences of low-probability and other beyond-design-basis events. Security considerations are addressed under different rules for both the existing and proposed EP approaches. Under the new rule, these evaluations and considerations would inform any changes in the EPZ size.

Regarding the need to address hazard analysis and emergency planning for mixed-mode or multi-module advanced reactor facilities, as with the existing EP framework, multi-module events are not explicitly addressed in the proposed EP rule because it is considered in other regulatory guidance. Draft Regulatory Guide 1350 (*Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities*, May 2020*)* requires identification and characterization of the specific hazards posed by multi-modular units, evaluation of the impacts of these hazards, and description of the planning or emergency response functions that will mitigate the impacts of the identified hazards. Other aspects of the NRC regulatory framework ensure multiple modules and mixed facilities will not result in undue risk of any new nuclear technologies.

The Department agrees that a clearer description of the criteria for the EPZ size determination could be provided, especially concerning the scope of "credible" accidents. The Department views the approach proposed by NRC as a less-prescriptive, goal-setting approach in which the regulator articulates its high-level intent (in this case, limiting dose to the public), and requires applicants to make their cases that the regulatory intent will be fulfilled. This approach is technology-neutral in a way that the existing prescriptive LWR-based approach cannot be. That same technology-neutrality makes it difficult to describe in detail the implementation of the staff approach, since it will apply to a range of technologies, including some that are not yet developed. It is the Department's position that Draft Regulatory Guide 1350 makes it clear that NRC staff review of applicant analyses will be required.

The applicant will document and submit the analysis underlying the EPZ determination, which will be available for public review. NRC's review of that analysis will be conducted using their transparent public processes and the final EPZ determination, as part of a licensing decision, will be subject to intervention by affected parties.

Regarding the rule's lack of guidance on the conduct of drills and exercises related to off-site radiological EP planning, the Department feels that EP drills and exercises are important to maintaining the ability of plant operating staff to respond appropriately to actual emergency conditions but that the specificity should not appear in "Regulatory Rulemaking." Draft Regulatory Guide 1350 states that "Program elements that may be implemented and evaluated according to a graded approach include periodicity between inspections, drills, exercises, number of performance objectives, and staffing." Consistent with the performance-based character and technology-neutrality of the NRC approach, drills and exercises are not explicitly prescribed in either the rule or the Draft Regulatory Guide. Applicants will be responsible for proposing periodicity of inspections, drills, exercises, and staffing, subject to NRC review and concurrence. It is the Department's position that the NRC approach allows the appropriate flexibility of the applicant to determine drill and exercise requirements subject to approval by NRC.

Finally, technical experts at the Department's Idaho National Laboratory have prepared the following technical documents relevant to the proposed rulemaking:

- "*Determining the Appropriate Emergency Planning Attributes for Microreactors*," (INL/EXT-20-58467, September 2020) – Provides a framework, using a graded approach, for appropriately structuring emergency planning requirements to reduce emergency planning zones for microreactors while still meeting applicable safety requirements.

- "*Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities*," (INL/EXT-20-58717, August 2020) – Provides a risk-informed approach, using either conservative or best-estimate methods, to determine radiological source terms for various types of advanced reactors.

- "*Establishing Jurisdictional Boundaries at Collocated Advanced-Reactor Facilities*," (INL/EXT-20-57762, August 2020) – Examines how jurisdictional boundaries might be established at an advanced nuclear reactor that is collocated with a non-nuclear industrial facility.

The Department respectfully submits these reports for the record. The conclusions of these reports provide additional technical input supporting the Department's position as described in this letter.

If you or your staff have any questions, or wish to further discuss the DOE position on this issue, please contact Ms. Alice Caponiti, NE Deputy Assistant Secretary for Reactor Fleet and Advanced Reactor Deployment, at alice.caponiti@nuclear.energy.gov or (301) 903-6062.

Sincerely,

Dr. Rita Baranwal
Assistant Secretary for Nuclear Energy
U.S. Department of Energy

Jay A. Tilden
Associate Administrator
& Deputy Under Secretary
National Nuclear Security Administration

INL/EXT-20-58467
Revision 0

Determining the Appropriate Emergency Planning Attributes for Microreactors

Jason Christensen

Peter Jordan

Wayne Moe

INL/EXT-20-58467
Revision 0

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Jason Christensen
Peter Jordan
Wayne Moe

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Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

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EXECUTIVE SUMMARY

Microreactor designs incorporate design features that provide very low reactor decay heat power at 24 hours after a shutdown that is manageable, in comparison with larger reactors. This attribute translates into a low probability of core damage and negligible offsite dose. This paper provides the conceptual framework for appropriately structuring emergency planning requirements for microreactors, while ensuring that the U.S. Nuclear Regulatory Commission's (NRC) commitment to safety is met. Due to the diversity in reactor designs throughout the microreactor community, this paper identifies the necessary concepts that should be considered to ensure that emergency plans for microreactors are appropriate for the risk. This paper does not mandate specific design features for microreactors but provides reactor designers a conceptual framework for a scalable, graded approach to emergency planning standards that should be considered in their individual designs in order to support simplified emergency plans.

Given that accident source terms associated with microreactors are essentially negligible when compared with those for large light-water reactors, revisions to emergency planning standards are justified. A graded approach to implementing emergency planning guidance can be used to appropriately structure microreactor emergency plans and reduce the size of the emergency planning zone and plume exposure pathway. Appropriately structuring emergency planning requirements will better optimize licensee and offsite agencies' emergency planning resources and reduce the resources associated with emergency planning.

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FIGURES

TABLES

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ACRONYMS

COLA	combined operating license application
CFR	Code of Federal Regulations
EAB	exclusion area boundary
EAL	emergency action level
EOF	Emergency Operations Facility
EP	emergency plan
EPA	Environmental Protection Agency
EPZ	emergency planning zone
ERDS	Emergency Response Data System
FEMA	Federal Emergency Management Agency
FR	Federal Register
ISG	interim staff guidance
LPZ	low population zone
LWR	light-water reactor
NOUE	Notice of Unusual Event
NRC	U.S. Nuclear Regulatory Commission
ONT	other nuclear technologies
PAG	Protective Action Guide
REM	roentgen equivalent man
RG	Regulatory Guide
RTR	research or test reactor
SMR	small modular reactor
SRP	Standard Review Plan
TEDE	total effective dose equivalent
TSC	Technical Support Center
UE	unusual event

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Determining Appropriate Emergency Planning Standards for Microreactors

1. Introduction

Microreactor technologies are currently being designed and developed to offer the nuclear industry a new, modern approach to providing electricity and industrial process heat. Multiple types of microreactors are being proposed, sharing similar size characteristics, limited quantities of radioactive material (i.e., a very small potential radiological source term), and low post-shutdown decay heat. Inherent and passive design features, along with highly autonomous operational characteristics, are being incorporated to further enhance reliability and public safety. Because microreactor designs are largely expected to preclude the possibility of significant offsite radiological consequences in the event of accident, it is proposed that changes be implemented to key current emergency preparedness (EP) requirements to reflect these different characteristics.

The purpose of this report is to examine current emergency planning regulations and associated guidance and propose alternative emergency planning standards concerning the installation and operation of commercial microreactors.

1.1. Objectives

Objectives of this paper include:

- Identify and summarize existing regulatory policy, guidance, and standards pertaining to EP as it applies to microreactor technology
- Identify and summarize key regulatory, technical, and policy issues relative to resizing an emergency planning zone (EPZ) scaled appropriately to microreactors
- Discuss key differences in microreactors EP needs when compared to existing light-water reactors
- Review important considerations for determining microreactor onsite and offsite emergency planning requirements [i.e. the 16 emergency planning standards of 10 CFR 50.47(b)]
- Propose alternative emergency planning standards for microreactors that may be considered for use by industry and regulatory stakeholders.

1.2. Scope

Regulatory requirements pertaining to plant siting, areas of owner control, and onsite and offsite planning zones have evolved over recent decades with a focus on large light-water reactor (LWR) power plants. Today's regulatory framework reflects decisions appropriate for large LWRs but does include allowances for small LWRs and non-LWRs. Multiple microreactor suppliers are pursuing submissions to the U.S. Nuclear Regulatory Commission (NRC) for standard design certification. One such microreactor design, the Oklo "Aurora" system, was submitted in early 2020. This design, as well as all subsequent microreactor concepts, will likely display characteristics of small reactor core sizes, passive accident mitigation features, lower power densities, very low decay heat, low or essentially no probability of severe accidents, slower accident progression, and small or negligible dose consequences both offsite and onsite. The microreactors included in this scope are generally considered to be $\leq$20 MW_{th}.

1.3. Statement of Issues

Current EP and Federal Emergency Management Agency (FEMA) regulations and guidance have not been updated to sufficiently reflect recent advances in advanced reactor (i.e., non-LWR) design safety. To address this concern, in 2015 the Commission approved an NRC staff's recommendation to initiate rulemaking to revise EP regulations as presented in SECY-18-0103 and guidance for small modular reactors (SMRs) and other nuclear technologies (ONT). In 2019, the Commission approved the NRC staff's proposed rulemaking discussed in SECY-18-0103 (known as 10 CFR 50.160) to create a new EP regulation for SMRs and ONTs. Microreactors are to be treated as a subset of this reactor population. This rulemaking is currently undergoing evaluation by affected stakeholders to ensure that microreactors are adequately addressed in the new regulations; this report will not focus on this evaluation but rather will examine issues associated with applications that may choose to use the 16 planning standards of 10 CFR 50.47(b) in their site EP program.

1.4. Summary of Outcome Objectives

The goal of this examination is to identify important emergency planning considerations for commercial microreactor (≤20 MW_{th}) deployment and operation; the focus of this review is on the 16 planning standards of 10 CFR 50.47(b) in the context of the proposed 10 CFR 50.160 rulemaking and to ensure that specific attributes of microreactor design, operation, and accident analysis and mitigation have been adequately addressed.

Current NRC requirements are structured to support large LWRs (e.g., ≥ 1,000 MW_{th} in power rating); these units can present significant consequences to the health and safety of the public and environment in the event of an accident. Microreactor design and operation, on the other hand, are display smaller reactor core sizes, passive accident mitigation features, lower power densities, lower probability of severe accidents, slower accident progression, and smaller dose consequences both offsite and onsite. This can therefore lead to substantially reduced EPZ size, reduced onsite and offsite emergency planning response requirements, and reduced numbers of response staff.

Regulatory requirements that may warrant modification and update for the purposes of microreactor emergency planning are addressed in Section 3. A description of the proposed rule 10 CFR Part 50.160 is provided, but this report is intended to compliment this activity by instead focusing on the existing EP standards found in 10 CFR Part 50.47(b) for use on the microreactor technology class.

2. Regulatory Foundation

2.1. NRC Requirements

2.1.1. Glossary of Planning Zones Around a Nuclear Power Plant

Under current regulation, multiple areas or zones of planning are expected around a nuclear power plant. Reactor siting regulations in 10 CFR Part 100 specify two zones, defining these as:

> ***Exclusion area*** means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

> ***Low population zone*** means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

Emergency planning regulations in 10 CFR Part 50 also identify two zones. Unlike siting zones, the definition of the EPZs is not specified exactly in the regulations but can be summarized from guidance documents, notably NUREG-0396. Accordingly, these zones are shown in Figure 1 and are as follows:

> ***Plume exposure pathway EPZ*** for nuclear power reactors shall consist of an area about 10 miles (16 km) in radius. The principal exposure sources from this pathway are (a) whole body external exposure to gamma radiation from the plume and from deposited material and (b) inhalation exposure from the passing radioactive plume. The time of potential exposure could range from hours to days.

> ***Ingestion pathway EPZ*** shall consist of an area about 50 miles (80 km) in radius. The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway. The principal exposure from this pathway would be from ingestion of contaminated water or foods such as milk or fresh vegetables. The time of potential exposure could range in length from hours to months.

Used, but not defined in security regulations, is the term “owner-controlled area”. This term is generally interpreted to be equivalent to the exclusion area required by 10 CFR Part 100.

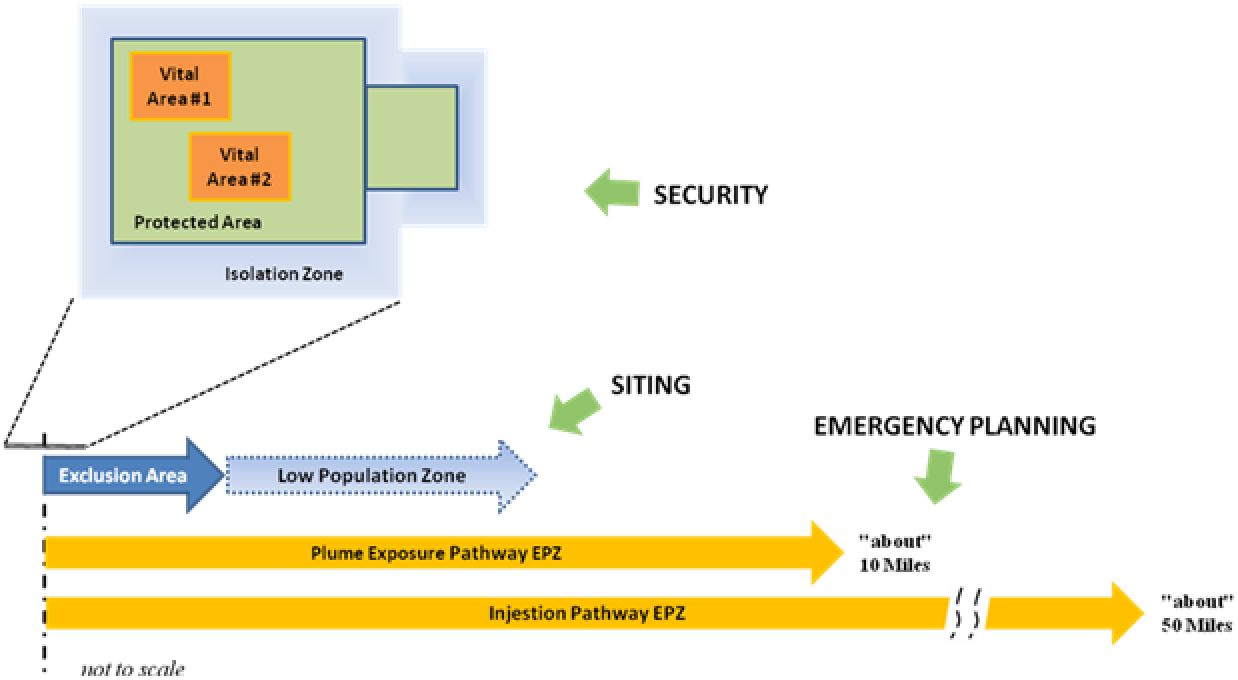

Figure 1. The zones around a Nuclear Power Plant.

2.1.2. Siting Regulations

Considerations for the acceptable implementation of site suitability requirements for nuclear power stations are described in Regulatory Guide (RG) 4.7, R3, "General Site Suitability Criteria for Nuclear Power Stations," (March 2014). The guide discusses the major site characteristics related to public health and safety and environmental issues for determining the suitability of sites for LWR nuclear power stations. It does not provide separate regulatory requirements for ONTs, SMRs, or microreactors. Presumably though, these same site considerations would be applicable to the evaluation of any planned nuclear facility.

A reactor licensee is required by 10 CFR 100.21(a) to designate an exclusion area and to have the authority to determine all activities within that area. In addition, the licensee is required to designate an area immediately surrounding the exclusion area as a low population zone (LPZ). The LPZ is required to be of such size that an individual located on its outer boundary during the postulated accident would not receive a radiation dose in excess of a 25- roentgen equivalent man (REM) total effective dose equivalent (TEDE). The size of the LPZ depends, in part, on aspects of the plant design.

Because of potential differences in the SMR and ONT designs, the proposed rule 10 CFR Part 50.160 does not contain an evaluation of a generic type of plant. Instead, SMR and ONT applicants will develop their EPZ sizes based on the accident source terms, fission product releases, and accident dose characteristics for the specific plant design. The recommended analyses, as documented in "Required Analyses for Informing Emergency Planning Zone Size Determinations" (dated June 2018), will be performed in conjunction with the criterion that the EPZ should encompass the area where the public would receive a post-accident dose of 1 REM or more over 96 hours.

2.1.3. Emergency Action Levels

10 CFR 50, Appendix E, Section IV, "Content of Emergency Plans", paragraph B, mandates that emergency plans must contain "emergency action levels (EALs)." These may also be termed "emergency classes." They are used for the grouping of off-normal events or conditions according to (1) potential or

actual effects or consequences and (2) resulting onsite and offsite response actions. There is no prescriptive guidance for the development of EALs, but rather they are developed by the technology designer based on the anticipated radiological consequence of progressive off-normal plant events; NRC agreement is required for this approach to be accepted in licensing actions. EALs are used for (1) determining the need for notification to and participation of various agencies and (2) determining when and what type of protective measures should be considered. The four current EALs, in ascending order of severity, are:

- Notification of Unusual Event (NOUE, sometimes abbreviated as UE)
- Alert
- Site Area Emergency
- General Emergency

These EALs apply to both nuclear power plants and research and test reactors of any power level. Declarations by the licensee of any EAL requires notifications made to the NRC and offsite organizations, as applicable.

2.1.4. Emergency Planning Regulations

Appendix E to Part 50 requires that each reactor license applicant provide plans for coping with emergencies in order to comply with 50.34 or 52.79. NRC Regulatory Guide 4.7, revision 2, "General Site Suitability Criteria for Nuclear Power Stations, March 2014," states that adequate plans must be developed for two areas (or EPZs): the plume exposure pathway and the ingestion pathway. As stated in 10 CFR 50.47, these EPZs for nuclear power plants are generally established at radii of 10 miles and 50 miles, respectively. This requirement exists as a result of context applicability to large LWR power facilities. Comparable requirements sized for very small reactor facilities are not addressed. However, both 10 CFR 50.33(g) and 50.47(c)(2) allow for the size of the EPZ to be determined on a case-by-case basis for reactors with an authorized power level of less than 250 MW_{th}.

As described in NUREG-0654 R2, NUREG-0396 established the technical basis for the 10 mile-radius plume exposure pathway and the 50 mile-radius ingestion exposure pathway applicable to a conventional large LWR. Over the years, however, there have been licensing actions for smaller commercial reactors, research reactors, and fuel storage facilities that allowed for smaller EPZs or removed the need for an EPZ beyond the site boundary, based upon the Environmental Protection Agency (EPA) Protective Action Guides (PAGs).

The operations and risk associated with microreactors and their low potential accident hazards may be more closely related to small research and test reactors and their low thermal output. RG 2.6 R2 presents guidance for developing emergency plans for research and test reactors in accordance with 10 CFR 50, Appendix E requirements. Appendix E notes that potential radiological hazards to the public associated with research and test reactors involve different considerations than those associated with larger nuclear power reactors. The RG applies to research and test reactors and other nonpower facilities under 50.21 for Class 104 licenses. RG 2.6 also applies to commercial and industrial facilities under 50.22 for Class 103 licenses. However, as additionally provided in Part 50.22,

> *a facility is deemed to be for industrial or commercial purposes if the facility is to be used so that more than 50% of the annual cost of owning and operating the facility is devoted to the production of materials, products, or* ***energy for sale or commercial distribution [emphasis added]****, or to the sale of services, other than research and development or education or training.*

Microreactors are expected to fall into this category and, presumably, would therefore be licensed in accordance with Part 50.22.

2.2. Environmental Protection Agency Protective Action Guides

The EPA has developed PAG Manual (EPA-400R-17/001, January 2017) to assist public officials in planning emergency responses to radiological incidents, which could release radioactive materials into the environment in quantities that warrant protective action. A PAG is defined as the projected radiological dose to an individual at which a specific protective action to reduce or avoid that dose is recommended. NUREG-0396 and EPA-400 identified the PAG dose guidelines of 1 rem to the whole body and 5 rem to the thyroid as doses at which public protective actions should be undertaken. Specifically, NUREG-0396 states:

> *The concept of Protective Action Guides was introduced to radiological emergency response planning to assist public health and other governmental authorities in deciding how much of a radiation hazard in the environment constitutes a basis for initiating emergency protective actions. These guides (PAGs) are expressed in units of radiation dose (rem) and represent trigger or initiation levels, which warrant pre-selected protective actions for the public if the projected (future) dose received by an individual in the absence of a protective action exceeds the PAG. PAGs are defined or definable for all pathways of radiation exposure to man and are proposed as guidance to be used as a basis for taking action to minimize the impact on individuals.*
>
> *The nature of PAGs is such that they cannot be used to assure that a given level of exposure to individuals in the population is prevented. In any particular response situation, a range of doses may be experienced, principally depending on the distance from the point of release. Some of these doses may be well in excess of the PAG levels and clearly warrant the initiation of any feasible protective actions. This does not mean, however, that doses above PAG levels can be prevented or that emergency response plans should have as their objective preventing doses above PAG levels. Furthermore, PAGs represent only trigger levels and are not intended to represent acceptable dose levels. PAGs are tools to be used as a decision aid in the actual response situation. Methods for the implementation of Protective Action Guides are an essential element of emergency planning. These include the predetermination of emergency conditions for which planned protective actions such as shelter and/or evacuation would be implemented offsite.*

In the 1970's, the Joint NRC/EPA Task Force on Emergency Planning recommended that the PAGs be updated and used to structure a framework for offsite emergency response actions tied to a spectrum of postulated accidents from minor through severe (Class 9). PAGs are used to define the EPZs. The following criteria were used to determine the generic distance for the plume exposure pathway EPZ:

- The EPZ should encompass those areas in which the projected dose from design-basis accidents could exceed the EPA PAGs.
- The EPZ should encompass those areas in which consequences of less severe Class 9 (core melt) accidents could exceed the EPA PAGs.
- The EPZ should be of sufficient size to provide for the substantial reduction in early severe health effects in the event of the more severe Class 9 accidents.

The PAGs are critical for EP and are based on the dose at certain distances. Based on the intrinsic differences that are expected to be associated with microreactors and their minimal offsite event consequences, the PAGs could potentially demonstrate attainment with a significantly smaller EPZ, thus suggesting that the emergency planning standards may need a significant adaptation and update as a consequence of the downscaled EPZ.

2.3. SECY Papers and NRC Policy

SECY-11-0152, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," dated October 28, 2011, discusses the NRC staff intent to develop a technology-neutral, dose-based, consequence-oriented EP framework for SMR sites that accounts for design differences, modularity, and collocation with industrial facilities, as well as a scalable emergency planning zone size. This SECY paper again noted that the size of the EPZ could also be determined on a case-by-case basis for reactors with an authorized power level of less than 250 MW_{th}.

SECY-11-0152 discusses the implementation of 10 CFR 50.47(b)(11) requirements for compliance with exposure guidelines consistent with EPA PAGs. The current EPA PAG guidance provides that licensed facilities that can demonstrate that accident doses at the site boundary would not exceed the PAGs should not be required to have either defined EPZs or comprehensive offsite emergency planning. Although the guidance in NUREG-0396 and EPA-400 was written for large LWRs, the underlying principle of using dose savings to determine EPZ size should be applicable to small reactors.

As a policy issue, SECY-11-0152 states:

> *EP programs for SMR sites should address implications of a smaller source term and passive design features associated with SMRs. One approach could be to have offsite EP requirements scaled to be commensurate with the SMR accident source term, fission product release, and associated dose characteristics, which are all a function of the licensed reactor power level.*
>
> *If projected accident offsite doses are less than 1 rem at the site boundary, then no EPZ beyond the site boundary would be required and the offsite emergency planning requirements would be limited. Specific EP requirements would be commensurate with the size of the EPZ . . . based on offsite dose.*

The NRC is initiating a rulemaking to address EP requirements and address considerations for reactor types other than large LWRs (Docket ID: NRC-2015-0225). When completed, this rulemaking would establish a new 10 CFR 50.160.

2.4. Proposed Rulemaking of 10 CFR 50.160

In December 2019, the NRC published a proposed rulemaking for emergency planning that can be invoked by allowing an applicant to choose between 10 CFR 50.160 and the emergency plan requirements found in Appendix E to 10 CFR 50 (including the planning standards found in 10 CFR 50.47(b)). This proposed rule is a performance-based, technology-inclusive, risk-informed, and consequence-oriented approach to emergency planning for SMRs and ONTs. As a performance-based approach, this proposed rule will provide a basis for EP through the review of design- and site-specific accident scenarios. This varies significantly from the previous deterministic approach of Appendix E and 10 CFR 50.47(b). The technology-inclusive approach allows for design considerations of each specific design to be considered in the development of an emergency plan. This includes passive safety characteristics, new fuel types, and other processes that enhance safety within the designs. This will create different plans for each design but will allow reactor applicants to fully utilize the specific safety features of their design. As a risk-informed and consequence-oriented approach, this proposed rule will focus on the level and severity of consequences related to a credible accident. Being risk-informed rather than risk-based allows emergency planning to be more independent of accident probability. Guidance for this proposed rule would be found in DG-1350, "Performance-Based Emergency Preparedness for Small

Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities" (ADAMS ML18082A044).

The alternative EP requirements would also adopt a new, scalable plume exposure pathway EPZ. According to the NRC, the new alternate requirements of 10CFR 50.160 will reduce the number of exemption requests from EP requirements, promote regulatory clarity and stability, and provide a reasonable assurance that SMR or ONT licensees will implement adequate protective measures. Also, this proposed rule would credit safety enhancements built into the advanced designs as well as credit the smaller size and benefits of these reactors associated with postulated accidents.

Another major provision of this proposed rule and guidance would be an alternative, performance-based framework that will be detailed in 10 CFR 50.160. This performance-based framework would include (1) the demonstration of emergency response functions through the development and maintenance of performance objectives and regular drills and exercises, (2) on- and offsite planning activities, (3) the consideration of credible hazards associated with collocated NRC-licensed and non-licensed industrial facilities, and (4) a required description of the boundary and physical characteristics of the plume exposure pathway EPZ and ingestion response planning capabilities. This proposed rule places focus on the actual performance of drills and exercises rather than control of emergency plans.

Each applicant/licensee is anticipated to have performance-based requirements that would be specific to the design of the plant. The NRC may need to develop additional guidance to cover the specifics of each design. Performance objectives would be developed and maintained by calendar quarter, and the NRC would review the objectives and metrics as well as use them during routine and periodic inspections to ensure that the licensee is maintaining adequate emergency planning and preparedness.

One major benefit to microreactor applicants of the proposed rule is the scaled EPZ. For instance, facilities with EPZs that do not extend beyond the site boundary would not be required to include tribal, state, and local government organizations in radiological drills and exercises. However, applicants/licensees would still be required to establish an emergency classification system to determine the need for notification of offsite response organizations. The licensee/applicant would be required to demonstrate the assessment, classification, monitoring, and repairs to facility malfunctions, including returning the facility to a safe condition.

Licensees and applicants would also be required to demonstrate protective actions; communications to the emergency response staff, NRC, and offsite response organizations; and ensure a continuity of operations through shift changes and other potential staff issues. Staffing should be sufficient to respond to all emergency conditions and perform necessary tasks until the augmenting staff arrives onsite. The licensee/applicant will also have the ability to assess and monitor radiological conditions, including personnel contamination, radiological releases, and the early indication of loss of adequate core cooling. Finally, the licensee/applicant would need to show the ability to reenter the plant, move people in and out of the plant, and perform operations to secure the plant. Critiques of these drills and exercises (or responses to actual emergencies) should be performed to ensure that the performance of emergency response functions would be evaluated for areas of improvement. Deficiencies would be tracked through a corrective action program.

Applicants and licensees subject to the "Emergency Response Data System" ([ERDS], as identified in Appendix E to 10 CFR 50, Section VI) would be responsible for identifying the data links with NRC and OROs as required. ERDS capabilities would be reviewed for each applicant. No changes are proposed to the ERDS regulations.

This proposed rule for Emergency Planning on SMRs and ONTs will continue to be developed and eventually codified in 10 CFR 50.160, but this report will strictly focus on utilizing and modifying current standards under 10 CFR 50.47(b) for use in microreactors.

2.5. NRC Guidance

The NRC has endorsed multiple guidance documents to assist applicants in developing EPs to address regulatory requirements. While using such guidance is generally not required and applicants can choose other alternatives to address regulatory requirements, utilizing established regulatory guidance does help ensure that all requirements are met in a way the staff will find acceptable, thereby increasing confidence and improving regulatory review efficiency. It is important to remember, however, that existing EP guidance was written largely for large light-water reactors and SMRs and ONTs.

Regulatory requirements associated with the siting and design of nuclear facilities are promulgated in 10 CFR 100, "Reactor Siting Criteria," 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Part 52 applications are required to include general information, as required under 10 CFR 50.33. Each facility must have a defined exclusion area and LPZ as defined by 10 CFR 100.3 and 10 CFR 50.2.

10 CFR 50.34(a)(10) requires facility applications to plan for coping with emergencies; 10 CFR 50, Appendix E sets forth items to be included in these plans.

10 CFR 50.33(g) and 50.47(c)(2) establishes general EPZ size for power reactors as 10 miles for the plume exposure pathway and 50 miles for the ingestion exposure pathway. However, for reactors of power levels less than 250 MW_{th}, the EPZ size may be determined on a case-by-case basis.

The onsite and offsite emergency response plans for nuclear power reactors must meet the 16 emergency plan planning standards set forth in 10 CFR 50.47(b).

ISG-029

On February 26, 2020, the NRC published draft interim staff guidance (ISG 029), "Environmental Considerations Associated with Microreactors," for public comment (85 Federal Register (FR) 11127). This draft guidance (ML20054B832) sought to assist NRC staff in determining the scope and scale of environmental reviews of microreactor applications. While this action is not specifically directed at emergency planning for microreactors, ISG 029 does provide insights into the NRC consideration of microreactor issues that may be associated with emergency planning. For example, it acknowledges that:

- Microreactor applications include a number of deployment purposes, such as power generation or industrial applications, potable water, hydrogen production, etc.
- Very small advanced reactor designs may have limited or zero radiological releases during normal operations
- Risks from accidents may be limited
- Some designs may not have credible severe accidents associated with it.

NUREG-0396

The concept of EPZs and their incorporation into the requirements and guidance for nuclear power plants emergency planning was introduced in NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (NRC 1978). NUREG-0396 discusses generic EPZs "as a basis for the planning of response actions, which would result in dose savings in the environs of nuclear facilities in the event of a serious power reactor accident." The nominal EPZ size was generally selected as 10 miles for the plume exposure pathway and 50 miles for the ingestion exposure pathway; these were chosen in order to assure that EPA PAGs would not be exceeded, based on the characteristics of a design basis and severe accident consequences associated with large LWRs.

As was noted in Section 1.4 of this paper, microreactor design and operations are anticipated to differ markedly from large LWRs by including a much smaller reactor core size, passive accident mitigation features, lower power densities, lower to potentially no probability of severe accidents, slower accident progression, and much smaller dose consequences to both offsite and onsite populations. These factors should justify a substantially reduced EPZ size than was envisioned in NUREG-0396. Along with a reduced EPZ size, fewer onsite and offsite emergency planning response requirements and response staff are needed, making EP needs more consistent with the power levels and risks associated with research and test reactors. Therefore, the EPZ size as described in NUREG-0396 would not be appropriate. As provided by 50.33(g) and 50.47(c)(2), for reactors of power levels less than 250 MW_{th}, a different EPZ size may be determined on a case-by-case basis. Given that EPZs for microreactors are not expected to extend beyond the facility exclusion area, it is appropriate to presume that the need for substantive offsite emergency planning responses will not be required as well as substantially reducing the onsite emergency planning response needs.

Regulatory Guide 1.101

RG 1.101, "Emergency Response Planning and Preparedness for Nuclear Power Reactors, R-5, June 2005," provides guidance for complying with the requirements of 10 CFR 50 and Appendix E to Part 50 with respect to emergency planning and preparedness. It endorses NUREG-0654/ FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, R-2," and includes guidance for collocated facilities. Although not specifically stated in RG 1.101, this guidance is also structured to address emergency planning requirements for large LWRs rather than reactors of substantially lower power levels. It does specifically state, however, that applicants

> *"...are free to propose other means to achieve compliance with applicable regulations."*

NUREG-0654

As noted above, NUREG-0654 provides criteria for compliance with emergency planning and preparedness requirements. It endorses the EPZ concept from NUREG-0396 with the caveat that the 10- and 50-mile EPZs are applicable to LWRs rated at 250 MW_{th} or greater. This acknowledges and introduces the concept that emergency planning requirements may be scaled for smaller reactors but does not provide any quantitative guidance.

Regulatory Guide 4.7

RG 4.7, "General Site Suitability Criteria for Nuclear Power Stations, R-3, March 2014," describes a method that the NRC considers acceptable to implement the site suitability requirements for nuclear power plants. It discusses major site characteristics related to public health and safety and environmental issues that the staff considers in determining site suitability for LWR facilities. It notes that adequate plans must be developed for the 10-mile plume exposure pathway and the 50-mile ingestion exposure pathway. This guidance is focused on the needs of large LWR facilities and does not address facilities of very small power output.

NUREG-0800

NRC's Standard Review Plan (SRP) for nuclear power plant applications, including early site permits and Combined Operating License Applications (COLs), is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition." Section 13.3, "Emergency Planning, R-3, March 2007," describes the areas of review and acceptance criteria for emergency planning as described in the applicant's safety analysis report. In particular, reviews are made against the requirements of 10 CFR 50.47 and 10CFR 50 Appendix E, which establish requirements for emergency preparedness. As noted in SRP acceptance criteria, onsite and offsite emergency response plans must meet the standards established in 10 CFR 50.47(b) and applicable requirements of Appendix E to Part 50. Compliance with these regulations is determined by using the guidance of RG 1.101, R-2, which endorses NUREG-0654 and, through it, NUREG-0396 and NUREG-0696.

2.6. Federal Emergency Management Agency

As described in NUREG-0654/FEMA-REP-1, R2, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, December 2019," both the NRC and FEMA evaluate the adequacy of emergency plans that pertain to offsite organizations such as state, local, and tribal governments within the EPZs surrounding commercial nuclear power plants. The evaluation criteria of this document address those elements and attributes of emergency plans and preparedness programs that are directly tied to meeting the planning standards in 10 CFR 50.47(b) and 44 CFR 350.5(a) and, for the NRC, are also used to assess compliance with 10 CFR 50, Appendix E.

If the NRC determines that the assurance of offsite radiological EP is not required for specific facilities where the EPZs do not extend beyond the site boundary, then FEMA determinations regarding reasonable assurance under 50.54(s)(3) would likely not be needed. The only offsite actions to be performed would be those associated with a community general response capability, which are not unique to radiological emergency response, e.g. fire, medical, law enforcement. Facility designers and license applicants will need to establish appropriate credible accident source terms, fission product release, and associated dose characteristics in order to establish a scaled approach for EP for the design and operation of the plant under consideration.

3. Microreactor Emergency Planning Considerations

3.1. Emergency Planning Zone Requirements

As stated in the regulatory basis for the proposed "Rulemaking for Emergency Preparedness for Small Modular Reactors and Other New Technologies" (docket ID: NRC-2015-0225), the technical basis for establishing scaled EPZ sizes are outlined in current power reactor and nonpower guidance NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants (December 1978)." As already discussed, NUREG-0396 information has been used to establish fixed-radius EPZ requirements for large LWRs at 10 miles (plume exposure pathway EPZ) and 50 miles (ingestion exposure pathway EPZ); these EPZs have been incorporated into Appendix E to Part 50. A footnote also recognized that reduced EPZs may be appropriate for reactors with smaller authorized power levels of less than 250 MW_{th} for which the EPZ may be determined on a case-by-case basis (but only referred to gas-cooled reactors). A similar rationale should be applicable for small microreactors with authorized licensed power levels up to 20 MW_{th}.

With the recent advent of small, non-LWR designs, SECY-11-0152, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," discusses the NRC staff's intent to develop a technology-neutral, dose-based, consequence-oriented EP framework suited to small modular reactor sites that accounts for variation in design approach, modularity, and potential collocation with nonregulated energy users. The SECY also notes that resized EPZs may accompany this requirement update. However, the staff's discussion in SECY-11-0152 does not specifically address the very small size and very low power levels anticipated for microreactor designs or their potentially unique operating characteristics.

As a policy issue presented in SECY-11-0152,

> *"EP programs for SMR sites should address the implications of a smaller source term and passive design features associated with SMRs."*

This paper suggests that this consideration be further developed and applied to microreactor designs as a subset of SMRs. Offsite EP requirements should be scaled to be commensurate and appropriate with the source terms, fission product release, and associated dose characteristics that are characteristic and attributable to microreactor technology. As stated in SECY-11-0152,

> *"The revised EPA PAG guidance (issued in 1992 as EPA-400-R-92-001) provides that licensed facilities that can demonstrate that accident doses at the site boundary would not exceed the PAGs should not be required to have either defined EPZs or comprehensive offsite emergency planning."*

This consideration can support offsite EP requirements scaled to be commensurate with microreactor source term, fission product release, and associated dose characteristics.

3.2. EP Standards for Research and Test Reactors

NRC RG 2.6 and NUREG-0849 provide information on the EP standards for use in research and test reactors (RTR). These standards differ significantly from the EP standards for commercial power reactors under 10 CFR 50.47(b). Specifically, RG 2.6 states

> *"From its review of safety analysis reports for research and test reactors and other non-power production and utilization facilities, and based on the radionuclide inventory and postulated radioactive releases at these facilities, the NRC staff determined that the potential radiological hazards to the public associated with the operation of these facilities are less than those associated with the operation of commercial nuclear power plants."*

Since microreactors are similarly sized to RTRs, one might consider the application of separate planning standards as defined in NUREG-0849. However, RTR operating characteristics and their use in commercial power generation and industrial applications would likely involve different deployment and public risk issues that effectively preclude them from being assigned into this category. Relatedly, commercial power microreactors would be included in Class 103 licenses as defined by 10 CFR 50.22 for commercial and industrial facilities. As defined by 50.22,

> *such a facility is deemed to be for industrial or commercial purposes if the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training.*

As such, microreactors are expected to fall within this regulated facility class. This report will therefore presume alternative requirements to existing EP requirements for large light-water reactors will be necessary rather than seek regulatory treatment as an RTR.

3.3. Analysis of Emergency Planning in the Oklo Aurora COLA

In March 2020, Oklo Power, LLC. submitted a combined operating license application (COLA) to the NRC for a new microreactor concept. This single-digit-megawatt-power commercial fast reactor design was designated Aurora and, at this time, NRC is performing acceptance reviews of the submission in advance of regulatory safety evaluations and determinations of adequate safety. A public version of the COLA is available under NRC Docket No. 99902046 and is on the NRC website under ADAMS Accession No. ML20075A000.

Aurora is the first very small nuclear power design to seek an NRC license to build and operate a commercial microreactor. In its COLA documentation, Oklo describes Aurora as inherently safe with no reliance on secondary systems, electricity, or human action to maintain safety. The safety approach discussed in the COLA is predicated on a maximum credible accident risk analysis derived from extensive examinations of a spectrum of internal and external events. Safety is presumed inherent in large part due to its very small size, small radionuclide inventory, low power density, low fuel burnup, a robust fuel design, and cooling systems that are independent of the presence of water. Aurora developers believe there is no credible radiological release scenario (see Chapter 5 of Part II of the COLA) that is associated with the design. The safety analysis discussed in the COLA submission is reflected in discussions of projected EP needs.

Part VII of the COLA contains key supporting documents and plans for Aurora. Enclosure 3 of Part VII specifically addresses the emergency planning proposed for the design (see NRC ADAMS Accession No. ML18134A086). The following are key highlights of this discussion relative to microreactor-oriented EP standard development.

The stated objective of the Aurora emergency planning is to provide defense-in-depth protection for the reactor unit and onsite personnel. Because design safety was determined by the supplier to preclude the need for substantive offsite emergency planning (as is required of the current LWR fleet), EP is proposed to comply with only the applicable onsite aspects of Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR 50) Section 50.47, "Emergency Plans," and applicable onsite regulations in 10 CFR 50 Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

Important characteristics of the Aurora emergency planning include:

1. *Organization and Responsibilities*: A Plant Manager will oversee various plant monitors assigned to track reactor parameters and assure site security. Since the Manager will often be absent from the site during normal operations, site monitors will be relied upon to track key parameters and initiate reactor trips when warranted. During plant emergencies, these staff will transition into emergency operations roles projected as necessary for plant-level response. Because of the safety attributes associated with the design, the supplier proposes that emergency response pathways emphasize communications with community emergency response organizations (especially those related to fire, medical, and security capabilities), as well as with the NRC.
2. *Emergency Classification*: Emergency types are classified based on credible reactor events and other emergency situations that require appropriate levels of emergency response. Existing regulations outline four classes of emergency conditions event groups, based on their relationship to potential offsite radiological consequences (See Section 2.1.3for further discussions on event classification). Aurora developers believe no credible emergency event exists that can lead to a consequence greater than the least severe existing class of event (i.e., the NOUE). A NOUE could be initiated at an Aurora installation by either manmade events or natural phenomena that creates a hazard that did not previously exist. Because no radioactive material release requiring offsite response are postulated, the remaining three notification levels (i.e., Alert, Site Area Emergency, and General Emergency) are not considered credible and are not applicable.
3. *Emergency Action Levels*: The Aurora supplier believes no credible site emergency can produce exposures beyond the site boundary in exceedance of EPA PAGs for projected site dose. On this basis, an offsite radiological response capability is unnecessary.
4. *Emergency Planning Zone*: The plume exposure and ingestion exposure pathway comprise the same EPZ for Aurora, which is limited to the exterior boundary of the Aurora powerhouse. As there will be no radiological releases associated with the maximum credible accident, the PAGs are met through an EPZ limited to the Aurora powerhouse, thereby eliminating the need to establish an offsite emergency planning zone response capability. Consequently, parts of 10 CFR 50.47 and 10 CFR Part 50, Appendix E related to offsite emergency monitoring and response would no longer serve the underlying the intent of the regulation by ensuring a rapid response to protect the public in the case of an offsite radiological event.
5. *Emergency Facilities and Equipment*: A "monitoring room" is to be designated in the powerhouse where the onsite alarm station and emergency support center will be established. In

the event a site evacuation is needed, a preselected location outside the powerhouse will be designated as an Emergency Operations Facility and be available to coordinate facility assessments, response, and recovery activities.

6. *Maintaining Emergency Preparedness*: Plant personnel assigned EP duties will be trained commensurate with their role and decision-making responsibilities. Drills will be regularly conducted to test emergency response equipment and staff proficiencies. Training drills for radiological releases beyond the site boundary are deemed unnecessary and will not be performed. Emergency plans will be annually updated, and equipment inspections and calibrations regularly performed.

At this time, NRC has not provided opinions or determinations concerning the adequacy of these proposed emergency planning attributes. However, in order to request regulatory acceptance of the generic Aurora Emergency Plan, exemptions were requested by the developer in accordance with 10 CFR 50.12. As is stated in Part V of the COLA, "Non-Applicabilities and Requested Exemptions," (see ADAMS Accession No ML20075A006), exemptions sought in this regard include:

- 10 CFR 50.47(b), in part
- 10 CFR 50.47(b)(4), in part
- 10 CFR 50.47(b)(6), in part
- 10 CFR 50.47(b)(7)
- 10 CFR 50.47(b)(9)
- 10 CFR Part 50, Appendix E, Section IV.E.8.b
- 10 CFR Part 50, Appendix E, Section IV.F.1, in part
- 10 CFR Part 50, Appendix E, Section VI.1, in part
- 10 CFR Part 50, Appendix E, Section VI.2.a, in part
- 10 CFR Part 50, Appendix E, Section VI.3.d, in part

Further information on the technical and regulatory basis for requesting these exemptions can be found in Section 3.6 of Part V of the COLA.

3.4. Analysis of Emergency Planning Standards for Microreactors

As is discussed in Section 2.4, alternative EP requirements are being proposed that would be scaled and applicable to smaller classes of reactors. Microreactors are not explicitly identified or discussed in this rulemaking initiative but, based upon anticipated maximum power levels ($\leq$ 20 MW_{th}), would likely be included in ONTs.

To be maximally effective and efficient, microreactor EP requirements should be scalable (perhaps as a function of their licensed power levels), be performance-based, and strongly oriented towards potential consequences, while taking into account the unique design characteristics, expected safety enhancements, and potential for slower accident progression. Changes to the 16 planning standards of 50.47(b) will be necessary to meet this objective. Issues and concerns related to these changes are discussed below according to each standard as it relates to microreactor technologies:

EP Standard 10 CFR 50.47(b)(1)

> *Primary responsibilities for emergency response by the nuclear facility licensee and State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response*

organization has staff to respond and to augment its initial response on a continuous basis.

Discussion:

The emergency plan should describe the functions, as applicable, to the emergency planning of federal, state, and local government agencies and the assistance that they would provide in the event of an emergency.

The very low power level associated with microreactors, negligible potential source terms, enhanced passive and automated response safety features likely to be demonstrated in association with microreactors should make the need for substantial dedicated radiological response capabilities external to the owner-controlled area largely unneeded. If the potential for radiological release outside of the owner-controlled area can be successfully demonstrated as highly unlikely through accident analysis by the applicant and shown to be confined within the site boundary the need for offsite entities or organizations, including local, state, and federal, could be reduced to a minimal precautionary need. The need for offsite entity response would likely be confined to security incidents and industrial, nonnuclear incident responses; emergency response plans could be dramatically simplified as a result.

EP Standard 10 CFR 50.47(b)(2)

On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility response in key functional areas is maintained at all times, timely augmentation of response capabilities support, and response activities are specified.

Discussion:

Onsite microreactor facility staffing requirements are not known at this time and may vary significantly by plant manufacturer and license. Microreactor operation may be highly autonomous or even remote, which would minimize or eliminate existing fleet staffing requirements and resources except during and after the initial reactor startup. Staff augmentation for emergency response would need to be redefined based on the analysis of credible operational occurrences and their potential consequences.

EP Standard 10 CFR 50.47(b)(3)

Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

Discussion:

Due to the low risk of offsite consequences expected of microreactors, the need for an Emergency Operations Facility (EOF) in the immediate area may not be required or, alternatively, could be a shared space with an existing local agency. Any offsite support that is needed would likely be commensurate with the capabilities required for any industrial facility (e.g., fire, medical, law enforcement).

EP Standard 10 CFR 50.47(b)(4)

A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by the facility licensees for determinations of minimum initial offsite response measures.

Discussion:

The four standard emergency classes currently associated with EALs in order of increasing severity are as follows:

- Notification of Unusual Events - This notification *"may be initiated by either man-made events or natural phenomena that can be recognized as creating a significant hazard potential that was previously nonexistent. No releases of radioactive material requiring offsite responses are expected."*

  Situations that lead to this class include:

  1. Security threats
  2. Natural phenomena
  3. Facility emergencies such as a prolonged fire (longer than 15 minutes)

- Alert - This notification would be initiated for events of radiological significance as to require notification of the emergency organization for the specific emergency. Under this class, it is unlikely that offsite response or monitoring would be necessary.
- Site Area Emergency - A site area emergency may be initiated when events such as the major damage of fuel or cladding and actual or imminent failure of fission product barriers is expected. Monitoring at the site boundary should be conducted to assess the need for protective actions. However, because of their very low power level and small source term, this class of alert is not considered plausible and would not be included in the facility emergency plan.
- General Emergency - A general emergency may be initiated by accidents that result in the uncontrolled release of radioactive material. However, because of their very low power level and small source term, this class of alert is not considered plausible and would not be included in the facility emergency plan.

It would be reasonable to assume that the enhanced safety and low consequence potential could allow changes in the structure of EAL currently addressed in 10 CFR 50, Appendix E, Section II.D and NUREG-0654, Rev. 2, Section II.D. While such a change would likely not negate the need for notification and EAL, the existing structure could be maintained with the assumption that the Site Area Emergency and General Emergency alerts are typically implausible and do not require significant planning.

EP Standard 10 CFR 50.47(b)(5)

> *Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow-up messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.*

Discussion:

It is anticipated that postulated radioactive releases from credible microreactor accidents will show that offsite radiological doses to the general public will not exceed the EPA PAGs of 1 rem whole body or 5 rem thyroid. The EPZ associated with such a demonstration can also be expected to remain within the facility's exclusion area boundary ([EAB] – owner-controlled property). Therefore, such a facility would not be expected include the General Emergency class of accidents requiring federal assistance as part of the emergency plan. State and local response beyond fire, medical support, and/or law enforcement consistent with an industrial facility would equally not be required. A notification system that informs federal, state, and local organizations (consistent with the emergency action level) could be maintained if desired.

Procedures will be established for notification to the NRC of any deviation from the facility's technical specifications. Such notification could be provided consistent with 10 CFR 50.72/50.73 (licensee event report) or alternately consistent with the guidance of NRC Information Notice 2009-31, "Nonpower Reactor Licensee Notifications to the NRC During an Incident."

EP Standard 10 CFR 50.47(b)(6)

> *Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.*

Discussion:

It can be presumed that a commercial microreactor design will successfully demonstrate to NRC that postulated radioactive releases from credible incidents associated with facility operation will likely not result in offsite radiological doses to the general public (i.e., exceed the EPA PAGs of 1 rem whole body or 5 rem thyroid.) This in turn allows the EPZ to be defined as not extending beyond the facility's EAB, therefore leading to these facilities being excluded from the General Emergency class of accidents. State and local responses would not be required other than the fire, medical support, and/or law enforcement consistent with a nonnuclear industrial facility.

EP Standard 10 CFR 50.47(b)(7)

> *Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.*

Discussion:

By presuming that a microreactor applicant can successfully demonstrate that postulated radioactive releases from credible incidents will not result in offsite radiological doses to the general public exceeding the EPA PAGs of 1 rem whole body or 5 rem thyroid, the EPZ can be defined as coinciding with the facility's EAB. In such a situation, there would be no required information to be released to the public or news media in the event of accident, because such an accident that triggers notification would not plausibly exist. The licensee/operator may elect to provide information regarding facility operation or condition for public awareness consistent with its established public information policy.

EP Standard 10 CFR 50.47(b)(8)

> *Adequate emergency facilities and equipment to support the emergency response are provided and maintained.*

Discussion:

The microreactor design and operation reduce the potential consequences of worse-case scenarios that might lead to adverse radiological consequences to the health and safety of the public or the environment beyond the site boundary. Establishing emergency response facilities (such as an offsite emergency response facility comparable to existing large LWR facilities) would not be necessary if such safety can be successfully demonstrated during a licensing safety assessment. Emergency response equipment for radiological monitoring may still be necessary to assure that no dose limit is exceeded (see EP Standard 10 CFR 50.47(b)(9)).

EP Standard 10 CFR 50.47(b)(9)

> *Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.*

Discussion:

If microreactor applicants can successfully demonstrate that the potential consequences of worse-case scenarios would not lead to adverse radiological consequences to the health and safety of the public or the environment beyond the site boundary, emergency response facilities (such as an offsite emergency response facility comparable to existing large LWR facilities) would be unnecessary. Emergency response equipment for radiological monitoring may be deemed necessary to ensure that dose limits are not exceeded.

EP Standard 10 CFR 50.47(b)(10)

> *A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.*

Discussion:

Depending on design safety, a range of protective actions will need to be developed for the emergency workers and, if necessary, the surrounding public. Exposure guidelines for onsite workers would be established by facility procedures during operations and emergency situations.

EP Standard 10 CFR 50.47(b)(11)

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

Discussion:

Although specific control measures may be adapted to reflect the risks associated with specific designs, radiological exposure controls will be needed onsite and for offsite emergency workers commensurate with their potential for exposure. Licensees must plan to meet applicable exposure guidelines. Offsite emergency worker exposure controls would be limited based on the EPZ plume pathway.

EP Standard 10 CFR 50.47(b)(12)

> *Arrangements are made for medical services to contaminated injured individuals.*

Discussion:

Facility procedures should provide for the offsite medical services of facility personnel that may be injured or contaminated consistent with the operation of any nuclear and industrial facility. This includes the extent that is required for microreactor facilities that may be remotely operated and/or otherwise may not have onsite staff. While the amount of radioactive material released would undoubtedly be smaller than would be expected for a large LWR, the material that is released would still pose a radiological threat requiring possible personnel decontamination and methods for handling and transporting contaminated personnel and material.

EP Standard 10 CFR 50.47(b)(13)

General plans for recovery and reentry are developed.

Discussion:

General plans for recovery and reentry following a nuclear facility event should typically be addressed by maintenance and repair procedures. However unlikely such an occurrence may be, procedures will need to be developed concerning the containment of any radioactive material that has been dispersed within the EAB.

EP Standard 10 CFR 50.47(b)(14)

Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

Discussion:

The emergency plan is expected to describe:

- the initial training and periodic retraining program
- annual onsite emergency drills to be conducted
- provisions for critiques of drills
- development of written scenarios for drills
- biennial review and update of the emergency plan and implementing procedures
- provisions to ensure the operational readiness of emergency communications and emergency health physics equipment

Since no offsite radiological release above regulatory limits is expected for a microreactor operational event, periodic emergency drills would be limited to the personnel and public within the EPZ plume pathway. Emergency drills could be conducted with any onsite personnel in accordance with plant procedures. Plant operations that may be conducted remotely may suggest that no operating personnel may normally be present onsite.

EP Standard 10 CFR 50.47(b)(15)

Radiological emergency response training is provided to those who may be called on to assist in an emergency.

Discussion:

Periodic emergency drills would be conducted based on the EPZ size and plume pathway. Radiological Emergency response training would be provided to all staff that have a role in emergency response.

EP Standard 10 CFR 50.47(b)(16)

Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

Discussion:

Because an emergency plan is required for all sites regardless of accident capability and EPZ size, the responsibilities for plan development and review would be established, and all planning staff would be properly trained.

4. Summary of Proposed Microreactor Planning Standards

4.1. Enabling Assumptions

Applicants will satisfactorily demonstrate to NRC during licensing safety assessments:

1. That significant offsite radiological consequences to the public are not a credible event for all normal and off-normal design conditions and, therefore, the reactors would qualify for reduced EP capabilities.
2. That the site will have an EAB that is collocated with or fully contained within the site owner control boundary.
3. That defense-in-depth precautions are in place that effectively ensure that alternative reductions in EP standard requirements will provide an adequate and appropriate reliability of outcomes that preclude public risks.
4. That onsite EP capabilities will adequately cover plausible event contingencies that include fire, medical, and law enforcement responses.

4.2. Changes to Regulatory Guidance and Requirements

As was noted in the proposed rule for emergency planning for SMRs and ONTs, if a license applicant adequately demonstrates that a plume exposure pathway EPZ can be established at the site boundary, the NRC would not necessarily mandate offsite radiological emergency planning activities for that site. NUREG-0396 provides this exemption for reactors with power levels less than 250 MW_{th} to have reduced EPZ sizes. Given that EPZs for microreactors are not expected to extend beyond the facility exclusion area, it is appropriate to presume that the need for substantive offsite emergency planning responses will not be required as well as substantially reducing the onsite emergency planning response needs.

If such an EPZ can be approved, a revised emergency planning standards structure could be developed similar to those that follow in Table 1 below:

Table 1. Suggested 10 CFR 50.47 Emergency Planning Standards for Microreactor Technologies.

Current Planning Standard	Basis for Change	Proposed Microreactor Planning Standard
§50.47(b)(1): Assignment of responsibility (organizational control) *Primary responsibilities for emergency response by the nuclear facility licensee and by state and local organizations within the EPZs have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.*	With minimal source terms, the licensee emergency response organizational structure can be greatly simplified and refocused on risk factors specific to the microreactor technology. Offsite emergency response organizational structures can be reduced, due to smaller impact zones and fewer affected jurisdictions. If radiological risks are demonstrated to be minimal, offsite emergency responses could emphasize non-radiological (industrial) scenarios.	No change in Planning Standard Emergency organizational control will still be required for the capabilities needed for onsite and offsite responses to the hazards associated with the licensed facility. However, if the EPZ is confined to the site boundary, offsite organizational planning may become negligible. Allowances should be provided for removing the offsite applications associated with this EP standard from sites without an EPZ that extends beyond the site boundary.
§50.47(b)(2): Onsite emergency organization *On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.*	On-shift staffing requirements will be reduced and derived from the job task functions needed to support reactor design and operation. Staff augmentation needs will likely be lower as well. Required response times are expected to increase. On-shift emergency response capabilities must still adequately address initial facility responses for the design (even if remotely operated). Timely staff augmentation may be secured from offsite resources if response times allow.	Revise Planning Standard *Onsite emergency response organization* *On-shift facility licensee responsibilities for emergency response are unambiguously defined* ***and enabled with adequate resources. Ensure that adequate staffing is available to address initial facility accident responses in key functional areas that assure safe design conditions are met. Onsite and/or offsite response capability augmentation will be available as needed to ensure public safety under all normal and off-normal design conditions.***
§50.47(b)(3): Emergency response support and resources *Arrangements for requesting and effectively using assistance resources have been made,*	It is expected that offsite fire, law enforcement, and ambulance services may be needed commensurate with other (nonnuclear) industrial facilities and be the primary response capability.	Revise Planning Standard *Emergency response support and resources* *Arrangements for requesting and* ***securing*** *effective assistance*

arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.	A microreactor licensee will seek to use an existing (non-licensee owned) near-site EOF for offsite response control.	*resources have been made, arrangements to accommodate **response staff at a near-site** EOF have been made, and other organizations capable of augmenting planned responses have been identified.*
§50.47(b)(4): Emergency classification system *A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.*	The four levels of emergency classification remain intact, i.e., 1. Notification of Unusual Event 2. Alert 3. Site Area Emergency 4. General Emergency General Emergency (and perhaps Site Area Emergency) conditions are not expected to be met by standard microreactor designs.	No change in Planning Standard The existing classification system can be applied with general recognition that Site Area Emergencies and General Emergency conditions are likely implausible events that do not require emergency planning.
§50.47(b)(5): Notification methods and procedures *Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow-up messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.*	This EP standard will not apply to installations that do not have an EPZ beyond the site boundary. For those sites with an offsite EPZ, the number of participating agencies and jurisdictions will be defined by zone size. Sites with reduced EPZ size should benefit from commensurately reduced notification requirements, but these requirements will not be eliminated.	Revise Planning Standard *Notification methods and procedures for offsite impacts* *Procedures have been established for notification, by the licensee, of state and local response organizations and for notification of emergency personnel by all organizations **(in accordance with the emergency action level)**; the content of initial and follow-up messages to response organizations and the public has been established; and the means to provide early notification and clear instruction to the potentially affected populace within the plume exposure pathway Emergency Planning Zone have been established.*
§50.47(b)(6): Emergency communications *Provisions exist for prompt communications among principal response organizations*	The need for prompt notification and supporting systems is reduced or eliminated because the potential for significant release of radioactive material is likely to be	Eliminate Planning Standard Plans for emergency communications will be developed (as required by safety

to emergency personnel and to the public.	reduced or absent. This requirement can be presumed addressed under *§50.47(b)(5).*	analysis) under *§50.47(b)(5) Notification methods and procedures for offsite impacts.*
§50.47(b)(7): Public education and information *Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.*	Information dissemination needs are dramatically reduced due to the smaller plume exposure EPZ potential. Any information required to be distributed to the public could be initially distributed during licensing and periodically reinforced and updated thereafter in conjunction with existing public emergency response service announcements and programs. Installations without an offsite EPZ would not be required to deliver periodic information to the public.	Eliminate Planning Standard Microreactor public education and information actions could be an added component to existing state and local education and emergency response plans. Public education and information items would operate in conformance with applicable state and local requirements.
§50.47(b)(8): Emergency facilities and equipment *Adequate emergency facilities and equipment to support the emergency response are provided and maintained.*	Equipment must be provided, adequate and appropriate to the risks posed by the installation, but needs would be lessened due to safer designs. Potential to consolidate Technical Support Center (TSC) and EOF into a single facility, due to a lessened and more reasonable timing of emergency response actions; TSC and EOF could be combined with existing collocated facilities.	No change in Planning Standard It should be recognized, however, that radiological response equipment and facilities would be needed at levels commensurate with the risks posed by the installation. Fire, security, and medical response capabilities from state/local entities would be still needed.
§50.47(b)(9): Accident assessment *Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.*	Assessment of accidents will still be required.	No change in Planning Standard Assessment of accidents will still be a required capability of licensees.
§50.47(b)(10): Protective response *A range of protective actions has been developed for the plume*	This standard is required for sites having an offsite EPZ. Limited offsite protective actions are needed, due to a smaller plume exposure EPZ. Installations	Revise Planning Standard *Protective response*

exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide, as appropriate. Guidelines for the choice of protective actions during an emergency, consistent with federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.	without an offsite EPZ need not provide EP for an offsite protective response. This standard should be amended to allow for protective responses commensurate with risks associated with the design. However, the standard can be deleted for designs where a safety assessment shows an EPZ beyond the EAB is unnecessary.	***For installations requiring an offsite EPZ,*** *a range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public.* ***This range of actions should consider the need for evacuation, sheltering, and prophylactic use of potassium iodide.*** *Guidelines for the choice of protective actions during an emergency are developed and in place. Protective actions for the ingestion exposure pathway EPZ are developed and appropriate to the locale.*
§50.47(b)(11): Radiological exposure control *Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.*	Standard still required onsite. Fewer offsite requirements would exist for smaller plume exposure EPZ.	Retain Planning Standard Although specific control measures may be adapted to reflect design risks, radiological exposure controls will be needed onsite and for offsite emergency workers commensurate with their potential for exposure. Licensees must plan to meet applicable exposure guidelines.
§50.47(b)(12): Medical and public health support *Arrangements are made for medical services for contaminated injured individuals.*	Standard still required onsite. Offsite support will be less due to the smaller impact zone and consequently fewer jurisdictions.	No change in Planning Standard Planning still required for medical support to contaminated injured individuals either onsite or offsite.
§50.47(b)(13): Recovery and reentry planning and post-accident operations *General plans for recovery and reentry are developed.*	General plans for recovery and reentry commensurate with design.	No change in Planning Standard Recovery, reentry, and operations plans will be needed commensurate with the design and plant procedures.
§50.47(b)(14): Exercises and drills *Periodic exercises are (will be) conducted to evaluate major*	More limited scope for onsite and participating offsite agencies/jurisdictions due to smaller EPZ.	No change in Planning Standard While exercises will be more limited as a result of lesser emergency response needs, those

portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.		capabilities that are still required must be periodically exercised commensurate with the need for such capabilities.
§50.47(b)(15) Radiological emergency response training *Radiological emergency response training is provided to those who may be called on to assist in an emergency.*	Fewer onsite requirements. Offsite requirements limited to fire/rescue/medical and affected jurisdiction.	No change in Planning Standard While less radiological response training will be required, required radiological emergency response staff must be properly trained.
§50.47(b)(16): Responsibilities for emergency planning *Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.*	Less onsite effort is required to maintain plans and program. Offsite is integrated into all-hazards planning, instead of unique REP plans as discussed in Appendix A.	No change in Planning Standard While less emergency planning resources will be required and may be combined with all-hazards planning, that capability must be identified and capable.

4.3. Next Steps

This report provides a description of current EP standards and details why many elements of the current standards are not appropriate for microreactors. Alternative emergency planning standards are proposed for microreactors for industry and NRC consideration. This report does not provide an explicit evaluation of the NRC's proposed rulemaking on emergency planning for SMRs and ONTs (also known as 10 CFR 50.160). However, licensees and applicants will have the option to choose between existing standards and, once finalized and published, the proposed rulemaking of 10 CFR 50.160. Should an applicant choose to use existing regulations, this report discusses the changes in the EP standards that should be considered.

The commercial success of microreactor designs are assumed to be a function of the incorporation of elements and features that provide a low probability of core damage and, in the event of a core damage accident, a high assurance of containment integrity and low offsite dose. Given that the accident source terms associated with microreactors are projected to be significantly lower than those for large LWRs, revisions to emergency planning requirements (e.g., simplification of requirements) are justified. This justification may require considerable technical analysis associated with source term calculations and EPZ plume exposure pathways. A graded approach to implementing emergency planning guidance should be used to appropriately structure microreactor emergency planning requirements by focusing on the unique attributes and technological advantages associated with microreactor designs.

References

Code of Federal Regulations Part 50 (10 CFR 50) Section 50.47, "Emergency Plans."

Code of Federal Regulations Part 50 (10 CFR 50) Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

EPA PAG guidance (issued in 1992 as EPA-400-R-91-001 Protective Action Guide (PAG) Manual).

EPA, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents (2017 PAG Manual)," EPA-400/R-17/001, January 2017.

NRC, ACRS letter October 19, 2018.

NRC, Draft Interim Staff Guidance (ISG) 029, "Environmental Considerations Associated with Micro-Reactors," 85 FR 11127, Docket ID: NRC-2020-0051 (ML20054B832).

NRC, FRN 7590-01-P, "Emergency Preparedness for Small Modular Reactors and Other New Technologies," Proposed Rule, docket ID: NRC-2015-0225.

NRC, Information Notice (IN) 2009-31, "Nonpower Reactor Licensee Notifications to the NRC During an Incident," December 10, 2009 (ML092680467).

NRC, NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," 1978 (ML12188A053).

NRC, NUREG-0654/FEMA-REP-1, R2, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, December 2019," (ML19347D139).

NRC, NUREG-0696, "Functional Criteria for Emergency Response Facilities, February 1981," (ML051390358).

NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition." Section 13.3, "Emergency Planning, R-3, March 2007," (ML063410307).

NRC, NUREG-0849, "Standard Review Plan for Review and Evaluation of Emergency Plans for Research and Test Reactors." October 31, 1983 (ML062190191).

NRC, RG 1.101, "Emergency Response Planning and Preparedness for Nuclear Power Reactors, R-5, June 2005," (ML050730286).

NRC, RG 2.6, Rev. 2, "Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities," September 2017 (ML17263A472).

NRC, RG 4.7, R3, "General Site Suitability Criteria for Nuclear Power Stations," (March 2014).

NRC, "Rulemaking for Emergency Preparedness for Small Modular Reactors and Other New Technologies (docket ID: NRC-2015-0225)," (ML16309A332).

NRC, SECY-11-0152, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," October 28, 2011.

NRC, SECY-18-0103, “Emergency Preparedness for Small Modular Reactors and Other Nuclear Technologies,” Proposed Rule, docket ID: NRC-2015-0225 (ML18134A076).

Oklo, “Oklo Power Combined Operating License Application for the Aurora at INL,” NRC Docket No. 99902046, ADAMS Accession No. ML20075A000, March 2020.

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Risk-Informed, Performance-Based, Technology-Inclusive Regulatory Infrastructure

June 2020

Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities

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Kurt Vedros
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Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

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EXECUTIVE SUMMARY

This report summarizes a risk-informed, performance-based, and technology-inclusive approach to determine source terms for dose-related assessments at advanced nuclear facilities to support the NRC's Non-LWR Vision and Strategy Near-Term Implementation Action Plans (ADAMS Accession No. ML16334A495) [1] and the NRC's response to the Nuclear Energy Innovation and Modernization Act (NEIMA) Public Law No: 115-439, of January 2019 [2]. This approach uses a graded process that allows both the non-mechanistic source terms calculation methods, which adopt conservative approaches and assumptions based on known physical and chemical principles, and, more importantly, the mechanistic source term calculation methods, which consider design-specific scenarios and use best-estimate models with uncertainty quantification for a range of licensing basis events to be used for the design and licensing of advanced nuclear technologies.

The source terms developed with this graded approach and radionuclide inventories elsewhere in the facility that are determined during source term analysis can be used to address licensing issues to support the application processes of 10 CFR Part 50 for a construction permit and operating license or 10 CFR Part 52 for a Combined Operating License (COL), Standard Design Certification, Early Site Permit, Standard Design Approval or Manufacturing License. They can also be used for other purposes, including equipment environmental qualification, control room habitability analyses, and assessments of severe accident risks in environmental impact statements.

There are many advanced reactor concepts being developed, including the high-temperature gas-cooled reactor, sodium-cooled fast reactor, lead-cooled fast reactor, molten-salt reactor, and microreactor. The graded approach presented in this report for source terms determination is, to the extent possible, generic to any of these reactor designs and to future reactor designs.

This report provides information on the review of the regulatory foundation for the use of conservative bounding source terms as well as event-specific mechanistic source terms for advanced nuclear reactor designs.

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TABLES

ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
AOO	anticipated operational occurrence
AST	alternative source term
BDBE	beyond design basis event
CD	core damage
CDF	core damage frequency
DBA	design basis accident
DBE	design basis event
DBEHL	design basis external hazard level
DID	defense-in-depth
DOE	Department of Energy
EAB	exclusion area boundary
EPA	Environmental Protection Agency
EPZ	emergency planning zone
F-C	frequency - consequence target chart
FMEA	failure modes and effects analysis
HTGR	high-temperature gas-cooled reactor
INL	Idaho National Laboratory
LBE	licensing basis event
LMP	licensing modernization project
LPZ	low population zone
LWR	light-water reactor
MACCS	MELCOR Accident Consequence Code System
MCA	maximum credible accident
MHTGR	modular high-temperature gas-cooled reactor
MST	mechanistic source terms
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NGNP	next generation nuclear power plant
NRC	US Nuclear Regulatory Commission
ORIGEN	Oak Ridge Isotope GENeration
PAG	protective action guide

PDS	plant damage state
PIRT	phenomena identification and ranking table
POS	plant operating state
PRA	probabilistic risk assessment
QHO	quantitative health objective
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
SCALE	Standardized Computer Analyses for Licensing Evaluation
SHA	system hazard analysis
SMR	small modular reactor
SRIR	site radionuclides inventories at risk
SSC	structures, systems, and components
STPA	system-theoretic process analysis
TEDE	total effective dose equivalent
UR	undesirable release
URF	undesirable release frequency

1. OVERVIEW

1.1 Purpose

The primary purpose of this report is to describe a risk-informed, performance-based, technology-inclusive determination of source terms for dose-related assessments for advanced nuclear reactor facilities to support the NRC's Non-LWR Vision and Strategy Near-Term Implementation Action Plans (ADAMS Accession No. ML16334A495) [1] and the NRC's response to the Nuclear Energy Innovation and Modernization Act (NEIMA) Public Law No: 115-439, of January 2019 [2].

The regulations in 10 CFR Part 20 [3] establish standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the U.S. Nuclear Regulatory Commission (NRC), which is associated with the assessment of plant conditions and forecast, and actual or projected radiological assessments.

The radiological accident consequences analysis for reactor siting is described in 10 CFR 50.34(a)(1), which establishes regulatory dose criteria at the reactor's exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ) [4]. Guidance on radiological source terms and consequence analysis is derived from this regulation for satisfying regulatory requirements and Commission Policy, as related to limiting the effects on public health and safety and other societal consequences in the event of accidents. Other current NRC regulations associated with source terms include 10 CFR50.49(e)(4), which applies to environmental qualification of electrical equipment based on the most severe design basis accidents (DBA), and control room habitability requirements in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, which specifies habitability dose criteria in the control room under accident conditions for current light-water reactors (LWRs) and may also be considered for advanced reactors.

The variety of advanced nuclear reactor technologies and designs has led to an increased use of radiological consequences as acceptance criteria for decisions related to design and licensing. Examples include the sizing of emergency planning zones (EPZ) based on estimated offsite consequences and safety classification of structures, systems, and components based on their role in preventing or mitigating offsite consequences. In an October 19, 2018 letter from the Advisory Committee on Reactor Safeguards (ACRS) to the Commission, a comment related to draft regulatory guide DG-1350, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," [5] on performance-based EPZ stated that it was "important for the staff to provide guidance on how source terms should be developed." This is because, without additional source terms development guidance to technologies other than those that are LWR-centered, the staff would need to review design and licensing information on a case-by-case basis, which is contrary to the Commission goal of reducing regulatory uncertainty for other nuclear technologies. The ACRS letter further noted that "Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors" (SECY-16-0012) [6] stated that the staff "have been in pre-application discussions with small modular reactor (SMR) designers, and the methods proposed by potential applicants appear to generally build on currently approved methods." Additionally, in a March 19, 2019 letter addressing a review of draft regulatory guide DG-1353 (finalized as RG 1.233), "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," [7] the ACRS stated that "guidance for developing mechanistic source terms should be expanded."

NEIMA directed the NRC to:

> develop and implement, where appropriate, strategies for the increased use of risk-informed, performance-based licensing evaluation techniques and guidance for commercial advanced nuclear reactors within the existing regulatory framework, including evaluation techniques and guidance for the resolution of source terms policy issues described in SECY–93–092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements," [8] and SECY–15–077, "Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies," [9] and identified during the course of reviews by the Commission of commercial advanced nuclear reactor licensing [pre-applications or] applications.

NEIMA specifically identified mechanistic source terms (MST) as one of the issues for which regulatory guidance should be prepared by January 2021. The scope of this document is focused on developing a risk-informed, performance-based, and technology-inclusive methodology for the determination of the source terms up to release to the environment for advanced reactors. Developing methodologies for dose determination, such as transport in the environment, exposure pathways, dose factors, and human health impacts and shielding, is outside the scope of this document.

1.2 Background

The use of postulated accidental release of radioactive materials and consequent radiological doses has long been deeply embedded in the regulatory policy and practices in the licensing and siting of nuclear reactors and protection of public health. However, large uncertainties exist in 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 an accident. Non-mechanistic methods, using conservative approaches and assumptions based on known physical and chemical principles, have been traditionally used for LWRs to yield conservative dose estimates to demonstrate compliance with regulatory requirements. As stated in "Policy Issues Related to Licensing Non-Light Water Reactor Designs" (SECY-03-0047) [10], "current light-water-reactors (LWRs) use site-specific parameters (e.g., exclusion area boundary) and a deterministic predetermined source term into containment to analyze the effectiveness of the containment and site suitability for licensing purposes." The LWR non-mechanistic source terms were first described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," [11] which was published by the United States Atomic Energy Commission in 1962. TID-14844 specified a non-mechanistic approach in the calculation of the amount of fission product inventory release to the containment atmosphere (i.e., "in-containment accident source term" or "source term") to calculate the radiological doses of the "maximum credible accident (MCA)" resulting from substantial core meltdown as a bounding fission product release in an LWR. The LWRs currently operating in the U.S. were licensed originally based on "in-containment source terms" specified in Regulatory Guide (RG)-1.3 [12] and RG-1.4 [13], with the specifications derived from TID-14844. The MCA is postulated as a nuclear accident that would result in a potential hazard that would not be exceeded by any other accident considered credible during the lifetime of the facility. For example, for the operating light-water reactors, the MCA has been frequently postulated as the complete loss of coolant upon the complete rupture of a major pipe (large-break loss-of-coolant accident). Conservative assumptions are used to compensate for uncertainties in the source term calculations for the purpose of calculating offsite doses in accordance with 10 CFR Part 100, "Reactor Site Criteria" [14]. For example, according to TID-14844, 100% of the core inventory of noble gases and 50% of the iodine (half of which are assumed to deposit on containment interior surfaces very rapidly) are assumed available for release to the atmosphere with a constant leakage rate of 0.1% per day. Using this approach would result in exposure doses probably many times higher than what would actually be expected, even if the postulated MCA should occur.

Since the publication of TID-14844, substantial additional information on fission product releases has been developed, in terms of the timing, nuclide types, quantities, and chemical form, based on significant severe accident research. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," [15] which specifies a revised source term methodology to formulate an alternative to the postulated source terms used in the past. This revised source term was more physically based to provide more realistic estimates of the source terms release into containment, given a severe core-melt accident. NUREG-1465 presents representative accident source terms for LWRs (one for pressurized-water reactors and a similar one for boiling-water reactors) and is applicable to the operating LWRs as well as future LWRs. These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. Information on the gap and in-vessel release phases from NUREG-1465 were adapted into the regulatory practices of NRC in 2000 through RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" [16]. RG 1.183 provides guidance on an acceptable alternative source term (AST) for design basis radiological consequences analyses, such as those addressed in Chapter 15 of typical LWR final safety analysis reports. In addition to providing acceptable inputs and assumptions for an AST based on NUREG-1465 [15], RG 1.183 [16] also described the attributes of an acceptable accident source term for licensees that wished to develop their own alternative. An AST is an accident source term that is different from the accident source term used in the original design and licensing of the facility and that has been approved for use under 10 CFR 50.67, "Accident source term." The alternative source term is not based upon a single accident scenario but instead must represent a spectrum of credible severe accident events.

Although initially used only for siting evaluations, the source term has been used in other design basis applications. As discussed in SECY-94-302, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs," dated December 19, 1994, [17] the staff uses reactor accident source terms such as given in TID-14844 [11] and the later issued RG-1.183 [16] not only for assessing potential doses to the public following an accident but also in areas such as:

(1) equipment qualification under 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,"
(2) control room habitability,
(3) engineered safety features,
(4) atmosphere cleanup systems,
(5) primary containment leak rate,
(6) containment isolation timing,
(7) post-accident sampling, and
(8) shielding and vital area access.

Analogous to the LWRs, quantitative determination of the radioactive materials that could potentially escape from an advanced reactor during normal operation or as a result of an accident and ultimately be released to the environment plays a critical role in the facility's design and NRC's requirements to protect public health against radiation hazards. For advanced reactors, as described in the HTGR Mechanistic Source Terms white paper (INL/EXT-10-17997 [18]), the phrase "source terms" refers to the quantities, timing and other characteristics of radionuclides released from the facility to the environment. It is noted that for LWRs, the phrase "source terms" refers to the magnitude and mix of radionuclides released from the fuel to the containment atmosphere, 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 advanced reactors have significant design differences relative to the existing LWRs, specifically with regard to materials, coolant, reflectors, and potential applications. Examples of coolant-based advanced reactor designs include sodium-cooled fast reactors, lead-cooled fast reactors, high-temperature gas-cooled reactors (HTGR), and molten-salt reactors. These designs propose using different barriers to the release of radionuclides, which resulted in the need to change to a technology-inclusive reference release location (i.e., environment vs. containment) in the definition of source term for non-LWRs as compared to that used for LWRs. For additional information on functional containment in lieu of leak-tight containment structures, see SECY-18-0096 [19], approved by the Commission staff requirements memorandum dated December 4, 2018 (SRM-SECY-18-0096 [20]).

Advanced reactors may be designed with various power output levels and fall into three categories—large reactors, SMRs, and microreactors. Although not explicitly defined in the regulation, large reactors are generally designed to operate at thermal power levels greater than 1,000 MWt, SMRs up to 1,000 MWt, and microreactors up to 50 MWt. Advanced reactors are designed with inherent or passive safety features to remove decay heat in an effort to enhance the safety for the plant workers and the public. Advanced reactors may be modularly constructed, and, specifically, the SMRs' small size allows them to be deployed in areas with smaller energy needs, their small size allows for more site flexibility and additional reactor units can be incorporated into the design as needed and clustered to create a multimodule, large capacity power plant.

Microreactors, on the other hand, are designed to be factory manufactured and transported. These reactors are referred to as special purpose reactors with the ability to provide heat and power to remote communities and industrial users. These reactors are designed to be self-regulating and not rely on physical systems to ensure the safe shutdown and removal of decay heat.

Because most advanced reactors are expected to operate at a lower power level, the amount of radioactive material released to the air during normal operations and under accident conditions may be reduced, compared to large LWRs. For example, a reduction in source terms allows the LPZs, EPZs, and the distances required to meet dose-consequence regulatory criteria to be adjusted to better fit the facility size. As SECY-16-0012 [6] stated:

> 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 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.

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, or mechanistic source terms, to estimate the release and behavior of these fission products, which may be applicable to advanced reactors. However, recent NRC activities related to advanced reactors (e.g., functional containment performance criteria (SECY-18-0096 [19]), scalable EPZ sizes (SECY-18-0103 [21]), possible changes to security requirements (SECY-18-0076 [49]), and the licensing basis considerations of RG 1.233 [7]) recognize the limitations of existing LWR-related guidance, which requires a return to first principles such as fundamental safety functions supporting the retention of radionuclides. Toward that end, NEI 18-04 [22], "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development", presents a process for the licensing of advanced non-LWRs developed by the industry-led Licensing Modernization Project (LMP). In that document, a modern, technology-inclusive, risk-informed, and performance-based process is defined for the selection of licensing basis events (LBEs); safety classification of structures, systems, and components (SSCs) and associated risk-informed special treatments; and determination of defense-in-depth (DID) adequacy for non-LWRs. The LMP process uses a set of frequency-consequence criteria (F-C target), as shown in Figure 1-1, to select LBEs and classify SSCs. As described in NEI 18-04, the risk-informed licensing basis uses a F-C target curve to describe dose criteria as a function of event scenario frequency.

In June of 2020, NRC issued RG 1.233 [7] and endorsed NEI 18-04 as "one acceptable method for non-LWR designers to use when carrying out these activities and preparing their applications." Mechanistic source terms play a critical role in evaluating the consequences of LBEs, which are in turn considered in establishing the safety classification and performance criteria for SSCs, and assessing DID for the design and related programmatic controls. The mechanistic source terms are used to estimate the radiological consequences within the analyses of event sequences as described in NEI 18-04 to compare to the F-C target curve in the selection and evaluation of LBEs. RG 1.233 describes the relationship as follows:

> Although NEI 18-04 does not address the topic in detail, the development of mechanistic source terms for designs and specific event families is another element of an integrated, risk-informed, performance-based approach to designing and licensing non-LWRs. The NRC staff expects applications or related reports to describe the mechanistic source terms, including the retention of radionuclides by barriers and the transport of radionuclides for all barriers and pathways to the environs. Where applicable, a facility may have multiple mechanistic source terms and specific event sequences to address various systems that contain significant inventories of radioactive material.

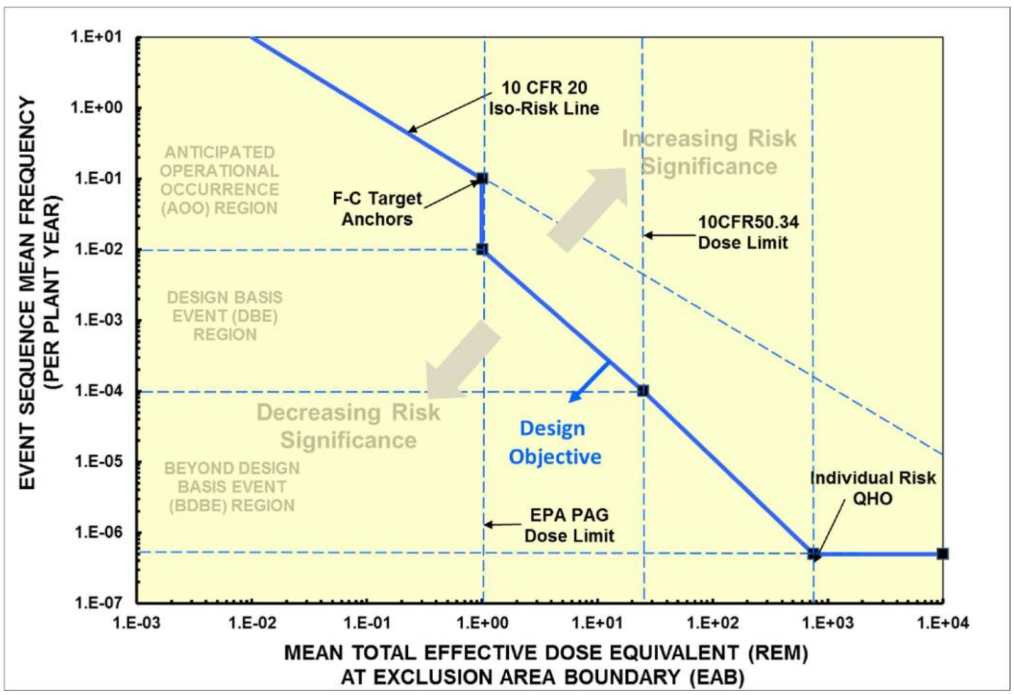


Figure 1-1 F-C target curve (NEI 18-04 [22]).

SECY-03-0047 [10] defines "mechanistic source term is the result of an analysis of fission product release resulting from the design-specific accident scenarios and accident progression being evaluated. It is developed using best-estimate phenomenological models with uncertainty quantification 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 use of a mechanistic analysis includes accounting for fission product retention and removal processes, as illustrated in Figure 1-2 for one non-LWR concept, and can substantially attenuate the magnitude of the release as compared to a more non-mechanistic approach.

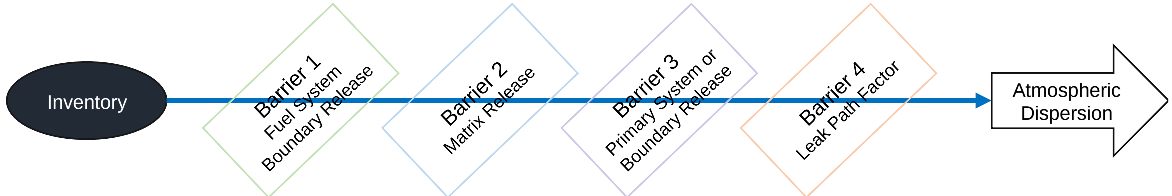


Figure 1-2 Illustration of radionuclides retention and removal process for one non-LWR concept (reproduced from SAND2020-0402 [23]).

The mechanistic source term, for the non-LWR concept illustrated in Figure 1-2, can be correlated using the following multifactor formula:

$$ST\left(S_i, RN_j, t\right) = I\left(RN_j\right) * F(S_i, RN_j, t) * MR(S_i, RN_j, t) * PSR(S_i, RN_j, t) * LPF(S_i, RN_j, t) \quad (1)$$

where:

$ST\left(S_i, RN_j, t\right)$ is the total release to the environment of radionuclide RN_j over the entire release duration time (t)

$I(RN_j)$ is the initial fission product inventory at the time of the reactor accident for radionuclide RN_j

$F(S_i, RN_j, t)$ is the fraction of release of radionuclide RN_j from fuel system boundaries to the fuel matrix

$MR(S_i, RN_j, t)$ is the fraction of release of radionuclide RN_j from fuel matrix to primary system

$PSR(S_i, RN_j, t)$ is the fraction of release of radionuclide RN_j from primary system to leak path

$LPF(S_i, RN_j, t)$ is the fraction of release of radionuclide RN_j from leak path to the environment

Equation (1) shows that all the factors that determine how much of the inventory is released across a given barrier and thus persists to the source term are accounted for in the calculation of source terms. Each factor is, in turn, a function of its initial design characteristics (e.g., materials), operating conditions (e.g., burnup, aging), and transient/accident conditions (e.g., time, temperatures, pressures, chemistry).

SECY-03-0047 [10] states that the mechanistic source terms should be allowed and defines a scenario-specific mechanistic source term that is based upon the characteristics of the fuel and plant to determine the magnitude, timing, and nature of fission product release from the core. “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, Volumes 1 and 2” (NUREG-1860 [24]) further defines the conditions under which design-specific and scenario-specific mechanistic source terms can be used in licensing. These conditions include:

- Having sufficient experimental data to confirm the source term (e.g., quantity and form of radionuclides, timing of release); and

- Accounting for uncertainties in the source term determination (e.g., use 95% confidence level).

Using an MST approach requires the availability of adequate tools and analysis methods with sufficient models and supporting scientific data that simulate the physical and chemical processes that describe the radionuclide inventories and the time-dependent radionuclide transport mechanisms to predict the radiological release for dose calculations. The other important facet in using MST is the development of the scenarios to be analyzed, with which the risk-informed and performance-based approach will be adopted. The risk-informed and performance-based approach integrates probabilistic risk assessment (PRA) methods and MST methodologies into a unified approach aimed at assessing the performance of a particular advanced reactor design to understand likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty.

A "risk-informed" approach considers risk insights together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety. As stated in RG 1.233 [7], "NEI 18-04 describes an expanded role for PRA for non-LWRs beyond current 10 CFR Part 52 requirements or Commission policy for potential applications under 10 CFR Part 50." PRAs are used to estimate risk by predicting what could go wrong, the likelihood of occurrence, and the severity of the consequences. PRAs also ensure that "significant insights are not obscured by artificially biased results derived from the application of uneven conservatisms." The risk-informed approach facilitates the integration of safety, security and preparedness (defense-in-depth) by having risk as a common measure with which to compare and assess the impact of each on the others. As such, the risk-informed approach provides the means to implement a unified concept for protecting public health and safety, the environment and the common defense, and security. It also helps ensure coherence among design, construction, maintenance, operation, security, and inspection.

A "performance-based" approach described in "Strategic Plan, Volume 3" (NUREG-1614, Vol. 3 [25]) focuses on desired, measurable outcomes as the primary basis for regulatory decision-making rather than prescriptive processes, techniques, or procedures. It leads to defined results without specific direction regarding how to attain these results. Performance-based regulatory actions focus on identifying performance measures that ensure an adequate safety margin and offer incentives for licensees to improve safety without formal regulatory intervention by the NRC. The main attributes for a performance-based approach described in NUREG-1614 are: (1) measurable, calculable, or objectively observable parameters that exist or can be developed to monitor performance, (2) objective criteria that exist or can be developed to assess performance, (3) licensees have the flexibility to determine how to meet the established performance criteria in ways that encourage and reward improved outcomes, and (4) a framework that exists or can be developed in which the failure to meet a performance criterion, while undesirable, will not in and of itself constitute or result in an immediate safety concern. Performance-based regulation focuses on effectiveness and efficiency of the decision-making process.

Combining risk-informed and performance-based approaches together yields a comprehensive approach, considering risk insights, engineering analysis and judgment including the principle of DID and the incorporation of safety margins, and performance history. This approach [26] enables the decision-making process to (1) focus attention on the most important activities, (2) establish objective criteria for evaluating performance, (3) develop measurable or calculable parameters for monitoring system and licensee performance, (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes, and (5) focus on the results as the primary basis for regulatory decision-making. Using a risk-informed and performance-based approach allows important scenarios to be identified in the source term evaluation.

2. OBJECTIVE

The objective of this report is to describe a risk-informed, performance-based, technology-inclusive approach to determine source terms for dose-related assessments at non-LWR nuclear facilities. The developed approach uses a graded and iterative process, which allows both the non-mechanistic and more detailed mechanistic methods to be used in performing source term calculations. The non-mechanistic approach uses conservative models and assumptions based on known physical and chemical principles, and mechanistic source term calculation methods consider design-specific scenarios and use best-estimate models with uncertainty quantification for a range of LBEs.

This report supports the NRC staff and the nuclear industry by providing a general description on determining source terms, including mechanistic source terms for facilitating discussions among stakeholders. The approach outlined in Section 3 is applicable to advanced nuclear technologies, such as future non-LWRs, SMRs, microreactors, and may be useful for nonpower production or utilization facilities.

It is noted that advanced reactor applicants are not required to use an MST or the process laid out in a LMP. Applicants may choose to develop a source term for an MCA using mechanistic, deterministic, or a combination of methods. This document is formulated to support these methods.

Although the information in this document is focused on development of an MST for accident assessments to determine offsite dose consequences, the determination of radiological source terms for other licensing assessments has similar features. For example, the determination of the equilibrium coolant radionuclide inventory for assessment of the radiological waste system design would include similar initial steps, such as determination of the core inventory and release to coolant during normal operations. Similarly, the development of non-mechanistic source terms may use some similar steps but with a conservative bias for bounding information.

3. TECHNOLOGY-INCLUSIVE RADIOLOGICAL SOURCE TERMS METHODOLOGY

The end goal of the development of radiological source terms is to use the developed source terms to evaluate the safety and siting of the facility; evaluate radioactive material release mitigation systems, structures, and components; evaluate radiation protection design; or evaluate the environmental qualification of certain equipment to prove that resultant doses are within regulatory criteria.

The focus of this report is on developing mechanistic source term techniques for evaluating offsite radiological consequences, which could be used to make decisions related to matters such as plant design features, siting, and emergency planning zone sizes. Many methodology components are used within the process to determine the source terms. In some cases, a non-mechanistic methodology can be used, and a bounding case can be made for meeting the dose criteria without further use of mechanistic components. Figure 3-1 [18] illustrates a general list of components feeding the pathways to compare to radiological regulatory criteria. The source terms are the key to the bounding calculations and the radiological dose determination; therefore, development of the source terms is not complete until final acceptable radiological doses are determined for the design.

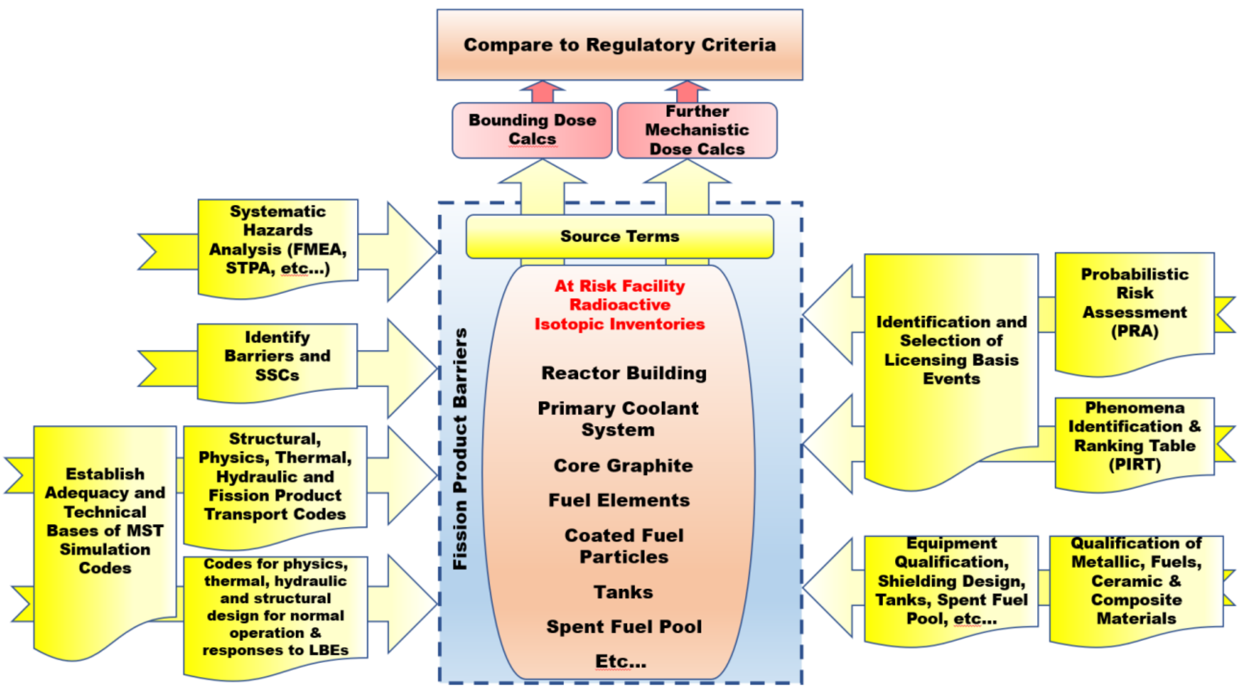


Figure 3-1 Technology-inclusive source terms determination methodology components (modified from Ref. [18]).

Several factors need to be considered in the source term determination for non-LWR technologies. As these are defined and characterized, the influence of each on the calculated dose is established. This influence permits developing a target for each element in the source term calculation to meet the safety goals of the facility design. The development of these targets and the degree to which each element of the source term calculation must be characterized are addressed in the following iterative steps and Figure 3-2 and discussed in the subsequent sections in more details:

Step 1: Identify Regulatory Requirements

Identify the Site & EAB/LPZ radiological consequence regulatory criteria that ensure the health and safety of the public and protect the environment.

Step 2: Identify Reference Facility Design

Select the reference facility design and identify facility system failure modes and safety SSCs of these systems, or needed for these systems, during all foreseeable operating modes. Use a system hazard analysis (SHA) such as Failure Mode and Effects Analysis (FMEA) or System-Theoretic Process Analysis (STPA) as necessary.

Step 3: Define Initial Radionuclide Inventories

Determine equilibrium radionuclide inventories (or appropriate values if equilibrium conditions are not achieved for a particular plant design) in all plant systems (e.g., fuel, barrier 1, barrier 2, etc.) during normal steady-state operation.

Step 4. Perform Bounding Calculations

These bounding calculations are performed to determine the dose consequences of the releasing radionuclide inventories identified by the previous step for the "maximum credible accident." Demonstrate compliance with the established regulatory criteria.

a. If compliance is demonstrated with margins to the F-C targets or other performance measures, prepare the documentation and submit to the NRC for approval, and the process related to assessing offsite consequences may end. If the use of a conservative source term is not able to support the evaluations of design features and offsite consequences, proceed to the next step. Note that margins to F-C targets and assumptions related to SSCs serving to prevent or mitigate events may contribute to other design and licensing decisions such as SSC classification.

Step 5. Conduct SHA and Perform Simplified Calculations

Conduct a SHA (FMEA, STPA, or equivalent) to identify potential SSC failure modes that lead to radioactive releases, as well as to identify a spectrum of postulated LBEs. As described in NEI 18-04, these assessments also contribute to probabilistic risk assessments that are expected to support the design and licensing of advanced reactors.

Develop realistic assessment of the barriers being relied upon for evaluated design basis event (DBE) sequences and resultant inventory release fractions across barriers (Equation 1) based on this analysis. Consider the behavior of the barriers and determine dose consequence by using simplified methods.

If the dose calculations show compliance with established regulatory criteria and the transient and barrier-specific release fractions can be justified to the NRC, the process ends. Otherwise, consider performing more detailed dose calculations using NRC-approved codes and actual site meteorological data. If the calculated dose meets regulatory criteria with margin, prepare the documentation and submit to the NRC for approval, and the process ends. Using siting as an example, if the calculated dose exceeds 10 CFR 50.34(a)(1)(ii)(D) dose criteria, proceed to the next step. Developers may also define performance measures (e.g., lower dose goal than criteria given in regulation) based on design goals such as desiring more flexible siting options or a scalable EPZ.

Step 6. Consider Risk-informed System Design Changes

Consider a system redesign to include additional SSCs as identified by hazard analysis, which will either return to Step 3 or proceed to Step 7.

Step 7. Select Initial List of LBEs and Conduct PIRT

Carry out activities as described in NEI 18-04 to select an initial list of LBEs and to conduct PIRT (Phenomena Identification and Ranking Table) to identify important phenomena for LBEs.

Step 8. Establish Adequacy of MST Simulation Tools

Establish adequacy of MST simulation tools and develop testing programs if needed:

a. Identify and characterize factors and parameters (e.g., temperatures, pressures) affecting radionuclide generation and transport during possible event sequences for the subject reactor technology or nuclear facility.

b. As needed to support meeting the regulatory criteria, identify how well each factor is currently characterized to validate its target in establishing the source term and, where the current characterization is deficient, define the gaps between what is needed and what is known.

c. If needed, develop and complete analytic and testing programs to fill those gaps.

Step 9. Develop and Update PRA Model

Develop and update PRA models for the subject reactor or nuclear facility, which could receive input from Step 12.

Step 10. Identify or Revise the List of LBEs

Use the risk information obtained through the performance of all prior steps to identify or revise the list of LBEs.

Step 11. Select LBEs to Include Design Basis External Hazard Level for Source Term Analysis

Analyze and include external events unique to the site of the facility which can cause LBEs.

Step 12. Perform Source Term Modeling and Simulation for LBEs

Perform source term and dose modeling and simulation for the selected LBEs.

Step 13. Review LBEs List for Adequacy of Regulatory Acceptance

Develop a final list of LBEs. If the final list is not complete, go back to Step 6.

Step 14. Document Completion of Source Term Development

Prepare documentation for source term calculations and submit to the NRC for approval.

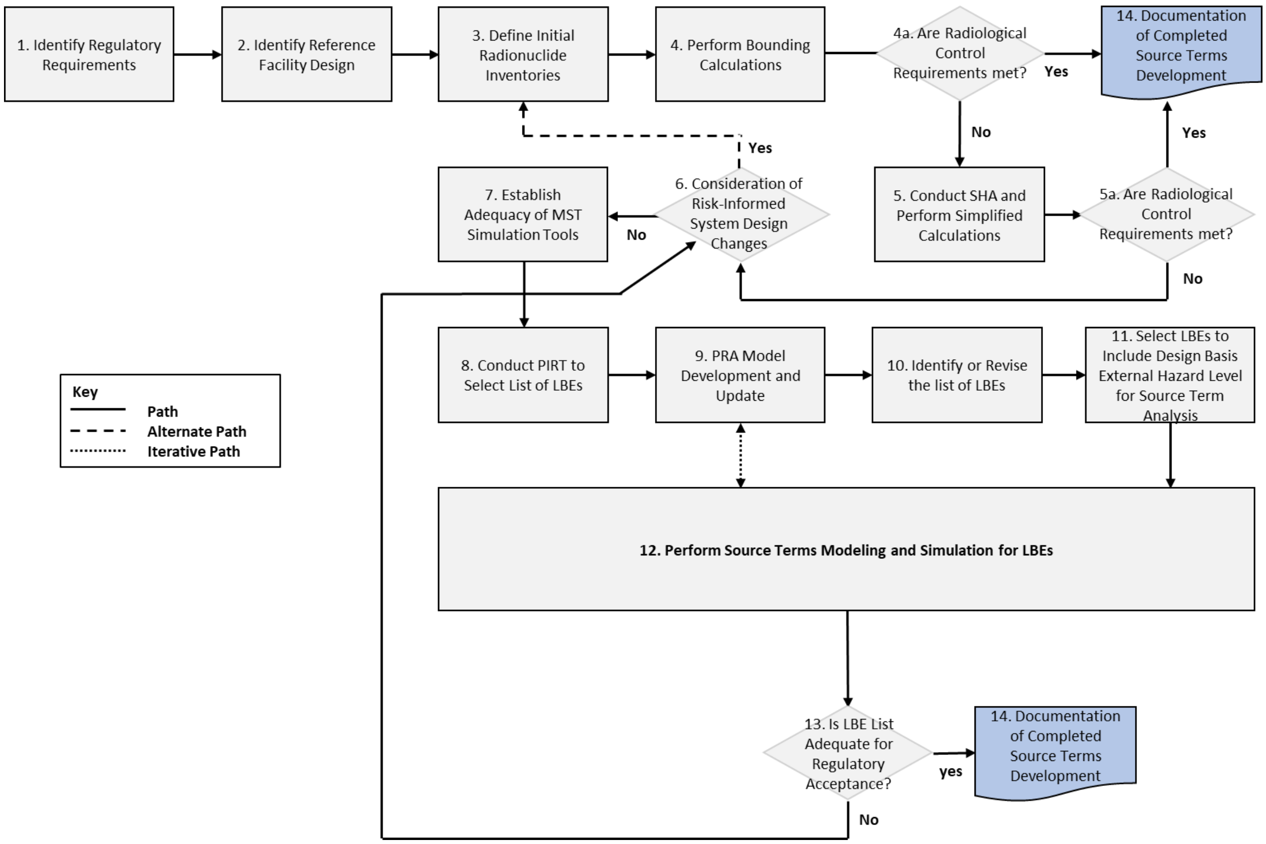


Figure 3-2 Technology-inclusive source terms determination methodology.

When referring to Figure 3-2, there are several pathway loops that can lead to completion of source terms development (Step 14). The first three pathways use a non-mechanistic or simplified mechanistic approach. One is to use initial bounding calculations from Step 4 to meet radiological control requirements. This is intended for facilities that have a small enough initial inventory of source terms to meet radiological control requirements upon a full release of the initial inventory. The second pathway can use the SHA performed in Step 5 to identify barriers and a maximum fractional release to perform a simplified mechanistic bounding analysis that would again meet radiological control requirements. A third pathway, which is still not a full MST approach, is to use the loop of redesign (Step 6) after failing Step 5a and then following through to Step 4a or Step 5a to its conclusion while meeting the radiological control requirements. If these pathways are not sufficient, a complete MST approach is desirable. Steps 6 through 13 are consistent with the MST process defined in NEI 18-04 for selecting and evaluating LBEs. The only exception is the addition of Step 8 to establish the adequacy of MST simulation tools. This step is necessary to ensure the MST simulation tools have acceptable level of pedigree in terms verification, validation, and uncertainty quantification.

3.1 Identify Regulatory Requirements That Require Radiological Source Term Information

Top-level radionuclide control requirements will be established for advanced nuclear facilities using existing regulatory requirements and design goals established by developers. The objective of setting the top-level radionuclide control requirements is to limit the calculated dose under all LBEs so that regulatory requirements for the protection of the health and safety of the plant workers, the public, and the environment are met. Limits on radionuclide release from the reactor building that are consistent with these top-level radionuclide control requirements are needed to establish the target values for all of the barriers to radionuclide release and ultimately to establish allowable in-service fuel failure and as-manufactured fuel quality requirements. The key top-level radionuclide control requirements expected to be imposed for the advanced nuclear reactors or nuclear facilities are listed in Table 3-1. The top-level radionuclide control requirements are based on established regulatory practice, e.g. NRC regulations in 10 CFR 20 [3], 10 CFR 30 [27], 10 CFR 50 [4], 10 CFR 52 [28], 40 CFR 190 [29] and EPA (Environmental Protection Agency) protective action guides (PAGs) [30]. It is noted that 10 CFR 20 limits the radiation doses from licensed operation to individual members of the public. Although not technically applicable to non-LWR designs, 10 CFR 50 Appendix I identifies design objectives for release from LWRs during normal operation to be as low as reasonably achievable. Both of these regulations are concerned with the cumulative dose acquired annually, rather than during a single event. Section 50.34 requires an applicant for a license for a power reactor permit or license to demonstrate that doses at the EAB and the outer boundary of the LPZ from hypothetical accidents (i.e., per event) will meet specified criteria. Part 100 refers to the same dose criteria in 10 CFR 50.34 for determining site suitability. The development of source terms for purposes other than determining an offsite dose may have additional or different regulatory requirements. For example, the environmental qualification of equipment is done per the requirements of 10 CFR 50.49, which does not have specific regulatory dose criteria.

Table 3-1 Top-Level Regulatory Requirements

Top-Level Regulatory Requirements		**Comment**
1	10 CFR 30, Schedule C	Emergency plan
2	10 CFR 50.34(a)(1)(ii)(D) TEDE ≤ 25 rem at EAB over worst two-hour dose period TEDE ≤ 25 rem at outer edge of low population zone (LPZ) for the duration of the passage of the plume	Facility siting Offsite dose criteria
3	10 CFR 50, Appendix I, LWR Design Objectives for Radionuclides in Plant Effluents, dose to individual in unrestricted area: Whole Body Dose ≤ 5 mrem/yr Dose to any organ ≤ 15 mrem/yr	Plant effluents
4	10 CFR 20 Subpart C Occupational Dose Limits: Total effective dose equivalent (TEDE) <5 rem/yr Organ Dose ≤ 50 rem(/yr)	Standards for occupational protection
5	10 CFR 20 Subpart D Public Dose Limits: Annual TEDE ≤ 0.1 rem Hourly External Dose ≤ 0.002 rem	Standards for public protection
6	40 CFR 190 Subpart B Environmental Standards for the Uranium Fuel Cycle, (LWRs), normal operations, annual dose equivalent: Whole Body ≤ 25 mrem Thyroid Dose ≤ 75 mrem Organ Dose ≤ 25 mrem	Standards for fuel cycle
7	10 CFR 52.47 Offsite Dose Criteria for LBEs, standard design certification: TEDE ≤ 25 rem for 2 hours at the EAB TEDE ≤ 25 rem for duration of passage of plume at the LPZ boundary	Offsite dose criteria*

Top-Level Regulatory Requirements		Comment
8	EPA PAGs for Radioactive Release for Public Sheltering & Evacuation (EPA 2017): TEDE over four days ≤ 1 rem Thyroid Dose ≤ 5 rem	Public shelter & evacuation
9	NRC Safety Goal Policy Statement (NRC 1986)	Safety goal

* It is noted that the same offsite dose criteria for LBEs can also be found in 10 CFR52.17 for early site permit, 10 CFR52.79 for combined license, 10 CFR 52.137 for standard design approval, and 10 CFR 52.157 for manufacturing license.

3.2 Identify Reference Facility Design

This step is important because focusing on the specifics of the advanced nuclear design provides the interconnection of all systems with the methodic analysis for the determination of source terms. The subject reference nuclear reactor and facility design is established by the developer when ready for evaluation. The design parameters and features, such as nuclear fuel, reactor core, heat transport systems, and engineered safety features within barrier 1; systems and engineered safety features within barrier 2; etc., are identified (see Figure 1.1). The facility operating modes such as online refueling or shutdown refueling, normal operations, events such as anticipated operational occurrences (AOOs), DBEs and beyond design basis events (BDBEs), and the DBAs are described. The definitions of AOOs, DBEs, BDBEs and DBAs are consistent with those found in NEI 18-04 [22].

3.3 Define Initial Radionuclide Inventories

The initial inventories ($I(RN_J)$ in Equation 1) of the radionuclides important for the calculations of offsite consequences at accident initiation are calculated using NRC accepted computer codes (e.g., the SCALE ORIGEN module for isotope generation and depletion) and methods. The initial inventories calculated are considered as the Site Radionuclide Inventories at Risk (SRIR) for release, and they represent some maximum quantity of radionuclides present or reasonably anticipated for the process or structure being analyzed. Different SRIRs may be assigned for different accidents as it is only necessary to define material in those discrete physical locations that are exposed to a given stress. The initial calculation of radionuclide inventories should include the radionuclides in fuel, and system information and depletion methods are subsequently used to calculate inventories resulting from all radionuclides residing in all systems barriers (i.e., Figure 1-2: barrier 1, barrier 2, etc.) due to an activation and leakage of the initial core inventory. For the generation of fission products in fuel, assumptions on fuel, core design, and management (e.g., operating cycle length, burnup limits, etc.) and the type of inventory (e.g., equilibrium nominal end of life) should be described. The use of conservative modeling assumptions or treatment of uncertainties in the initial inventories should be described. Initial radionuclide inventories are given by isotope either as total activity (for solid fuel) or activity concentration (in fluid).

3.4 Perform Bounding Calculations to Estimate Consequence of Site Radionuclides Inventory at Risk for Release

A bounding analysis employs assumptions that are meant to produce the worst-case consequence resulting from a “maximum credible accident” for a given facility or system of that facility. It is also a starting point analysis for a facility to illustrate the potential, or lack thereof, level of radioactive hazard associated with a facility. A possible resource for such an analysis is 10 CFR 30. Schedule C of 10 CFR 30 contains a list of release fractions and maximum release limits of various isotopes that would avoid the need for public evacuation plan. The release fractions of Schedule C are meant to be the worst-case release for facilities that handle or produce radioactive byproduct material. These release fractions are the result of accident analyses, operation experience, or known physics limitations, for example note the "Nuclear Fuel Cycle Accident Analysis Handbook" (NUREG-1320 [31]) or "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities" (DOE-HDBK-3010-94 [32]). In addition, the dose calculation employs the assumption that annual averaged meteorological weather data is not available and therefore conservative meteorological weather conditions are assumed of Pasquill-Gifford Type "F" plume stability for a wind velocity of 1 m/s, see the "Technical Basis for Regulatory Guide for 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"" (NUREG/CR-2260 [33]).

Proceeding with a bounding analysis for the given facility requires that, after the initial radionuclides inventories at risk are determined, bounding calculations that estimate the consequence for release are performed by calculating the product of the release fraction listed in 10 CFR 30 Schedule C for a particular radioactive isotope times the inventory at risk. If this product is equal to or below the release limit for that isotope as listed in 10 CFR 30 Schedule C, an emergency plan is not needed for responding to a release of radioactive material for facilities applicable to 10 CFR 30. 10 CFR 30 also contains a formula for multiple isotope releases, which is the sum of the ratios of actual release to the release limit. If this sum is less than or equal to one, an emergency plan is not needed for responding to a release of radioactive material. The information in 10 CFR 30 Schedule C is based on showing that the consequences of the release would be less than one rem TEDE offsite. Similar analyses must be performed for comparison to other radiological criteria listed in Table 3-1.

If compliance has been demonstrated, prepare the documentation, including a description of methods, assumptions, and consideration of uncertainty, and submit to the NRC for approval, and the source term determination portion of the design and licensing process ends here, provided that the release fractions used can be justified as applicable to your facility and that the calculated margins to radiological limits have been achieved by the facilities design. Otherwise, proceed to the next step.

3.5 Conduct SHA to Identify Potential Failure Modes and Determine Dose Consequence Using Simplified Methods

In this step, a SHA equivalent to a FMEA [34] or a STPA [35] is conducted to identify potential failure modes that could lead to source terms. The intent is to utilize SHA to identify all release paths described in Figure 1-2 and Equation 1. This information has a two-fold purpose: one is used to take credit for SSCs beyond those credited in the bounding calculations performed in Section 3.4 while providing a simplified source term, and the second purpose is to identify SSCs and barrier penetration pathways for further steps in the deterministic or mechanistic process. The use of a SHA or similar technique is consistent with the discussions in NEI 18-04 [22] on developing a technically sound understanding of the potential failure modes of the reactor concept, how the plant would respond to such failure modes, and how protective strategies can be incorporated into formulating the safety design approach. The incorporation of safety analysis methods appropriate to early stages of design, such as FMEA and process hazard analysis, provide early stage evaluations that are systematic, reproducible, and as complete as the current stage of design permits and support the development of the PRA (see Step 3.9).

A SHA will identify the SSCs and barrier penetration pathways and to some extent the effects of failure in preparation of PRA, PIRT, and modeling analyses.

SHA processes gather system experts and documentation to answer questions about the design pertaining to barriers to radioisotope inventory transport during normal and off-normal operations. Questions answered include:

- What is the failure mode?
- What are the interactions that occur due to the event?
    - What do the interactions cause?
- How likely is a failure to happen?
- What is the effect of the failure on the system?
- What is the outcome in transport of radioisotope inventory release fractions?

The following factors should be considered for the SHA derived release fractions: the SSC damage ratios (fraction of the materials at risk actually impacted by the accident generated conditions), leak path factors (fraction of the radionuclides in the aerosol transported through some confinement barrier that are deposited in a filtration mechanism), airborne release fraction (or airborne release rate for continuous release) (airborne release rate is a coefficient used to estimate the amount of a radioactive material suspended in air as an aerosol and thus available for transport due to a physical stress from a specific accident), and respirable fraction (fraction of airborne radionuclides as particles that can be taken up through air inhaled by the human respiratory system. Particulate releases from LWRs are commonly assumed to include particles with 10-µm Aerodynamic Equivalent Diameter or less).

If SHA can identify and quantify the effectiveness of SSCs and barriers to radioisotope release, it can be used to define release fractions for a spectrum of postulated DBEs. Once the bounding release fractions for the at-risk radionuclides inventory have been determined, a mechanistic source term analysis can be performed using simplified methods. These simplified methods are described in Simplified Approach for Scoping Assessment of Non-LWR Source Terms (SAND2020-0402 [23]). Subsequently, the resulting dose consequence of these source terms can be estimated by using other NRC accepted computer codes and methods.

If the dose consequence analyses demonstrate compliance with radiological criteria listed in Table 3-1, an argument can be made that the source terms do not need to be developed further. The process then moves to the documentation phase, which should include a description of methods, assumptions, and consideration of uncertainty. Otherwise, the process proceeds to using the SHA information attained to complete the subsequent steps.

3.6 Consideration of Risk-Informed System Redesign

As pointed out in NEI 18-04 [22], the design development is performed in phases and often includes a preconceptual, conceptual, preliminary, and final design phase and may include iterations within phases. The subsequent steps may be repeated for each design phase or iteration until the list of LBEs becomes stable and is finalized. If the system as designed is not adequate to meet the radiological safety control requirements of a bounding or mechanistic case, consider a system redesign to include strengthened barriers and/or SSCs as identified by SHA, PIRT, or PRA. During the earlier phases prior to the final design phase, using simplified source term methods (e.g., SAND2020-0402 [23]) to evaluate the release mitigation strategies based on a range of barriers, physical attenuation processes, and system performance can efficiently identify the design features that are most important to mitigate different classes of accident scenarios. The mechanistic source term methodology described in the subsequent steps play a more important role in the evaluation of the mitigation strategies during the final design phase.

System redesign, using a risk-informed approach as shown in "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (RG-1.200 [36]), can direct efforts towards the greatest benefit for meeting radiological regulatory criteria. PRAs used in risk-informed redesign activities may vary in scope and level of detail within each phase. The PRA needs to be maintained and upgraded, where necessary, to ensure it represents the actual state of the design phase.

3.7 Select Initial List of LBEs and Conduct PIRT

As noted in regulatory guide RG 1.233 [7], established methods for addressing radiological source terms for LWRs have limited applicability to non-LWR designs, and mechanistic source term analysis may be used to estimate radiological consequences for such designs. Toward that end, it is necessary to select the initial list of LBEs to develop the basic elements of the safety analysis including mechanistic source term analysis during design development. The initial list of LBEs is to be selected using a deterministic approach based on engineering judgment. This approach has been used for licensing operating LWRs and involves no use of PRA information and insights. NEI 18-04 [22] has a detailed description on how to select the initial list of LBEs.

The MST methodology for the evaluation of the initial list of LBEs will need to meet the three provisions outlined in Section 3.8 from SECY-93-092 [8]. SECY-93-092 further outlines that "The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design." The PIRT process can be used to ensure that these conditions are met. The PIRT process is a systematic way of identifying safety-relevant and safety-significant phenomena and ranking the importance and knowledge level associated with these phenomena for the LBEs. This ranking is ideal for advanced reactors in the conceptual design phase and for assessing through a source terms PIRT whether the transport of fission products can be adequately modeled based on present knowledge levels, as required by the above MST provisions.

The PIRT process consists of nine steps:

1. Identify issues
2. Identify specific objectives
3. Define hardware and scenarios
4. Define evaluation criteria
5. Identify current knowledge base
6. Identify phenomena
7. Develop importance ranking
8. Define knowledge level
9. Develop documentation

During the PIRT process, a comprehensive list of phenomena relevant to safety for potential hardware failure models and accident scenarios is developed by a panel of experts. After that, the importance of the phenomena is ranked either high, medium, or low relative to certain evaluation criteria. The process has previously been applied to understand radionuclide transport in certain advanced reactor systems, for example see “Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)” (NUREG/CR-6944 [37]), and is generalized in a technology-inclusive way in what follows. An example outcome of the process, applicable to mechanistic source term analysis, is given in Table 3-2.

Table 3-2 PIRT - Identify Issues.

#	Phenomenon	Importance	Rationale	Knowledge Level	Rationale	Model Status
1	Transport phenomenon A	High	Primary barrier for radionuclide transport	Low	Lack of, or uncertain, experimental data	Major need
2	Transport phenomenon B	Medium	Minor barrier for radionuclide transport	Medium	Some experimental data available	Minor need
3	Transport phenomenon C	Low	No credit taken for barrier C in source term analysis	High	Well characterized experimentally	Adequate

Table 3-2 also includes a column titled “model status,” which may be used as a part of the process to assess the adequacy of models generally or certain codes in particular to perform mechanistic source term calculations for a given advanced reactor type. Here the status is classified as a “Major need,” “Minor need,” or “Adequate.” A status of “Adequate” would refer to models that are well verified and widely accepted, or that such models have been implemented, verified, and validated in the computer code in question. A status of “Minor need” indicates models that might be improved if informed by some additional experimental data, or such models that need minor modification within a code or are straightforward to implement. A “Major need” indicates models that are speculative in nature, not well informed by experimental data, or highly uncertain, or code implementations that lack such a model entirely in addition to its verification and validation.

To the extent that each phenomenon listed in the table corresponds to transport across a barrier, each is associated with a release fraction across that barrier, as in Equation (1); conservatism in a given transport step (as in the third example in Table 3-2) would correspond to a release fraction of one for that step.

3.8 Establish Adequacy of Mechanistic Source Term (MST) Simulation Tools and Develop Analytic and Testing Programs

The adequacy of the mechanistic source term simulation tools will be assessed in this step to take specific account of the unique features of each reactor type. The use of design-specific and event- or scenario-specific mechanistic source terms can be justified by having sufficient experimental data to confirm the source term (e.g., quantity and form of radionuclides, timing of release) and accounting for uncertainties in the source term determination (e.g., use 95% confidence level). The assessment of the computer codes involves verification, validation, and uncertainty quantification. The factors affecting radionuclide generation and transport for the subject reactor technology or nuclear facility will be identified and characterized. As needed to support meeting the regulatory criteria, identify how well each factor is currently characterized to validate its target in establishing the source term and, where the current characterization is deficient, define the gaps between what is needed and what is known. If needed, develop analytic and testing programs to fill those gaps and determine appropriate programmatic controls (e.g., inspections and surveillances) that may be needed during plant operations. The adequacy of the MST simulation tools can be established according to the provisions specified in "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements" (SECY-93-092 [8]), which states that source terms should be based upon mechanistic analysis provided that:

- 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 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.

Since it may take a long time to complete the testing programs, this step will proceed in parallel with the evolution of the design of an advanced reactor. The completed analytic and testing programs for the source terms would have filled the technical gaps identified between what is needed and what is known. The radionuclide generation and transport phenomena are more fully characterized and understood. The MST computer codes will be updated and validated with the newly acquired data and knowledge.

One important outcome from the completion of the analytic and testing programs is the identification, evaluation, and management of uncertainties. Uncertainties need to be addressed in the calculation of both frequencies and consequences of the event sequences. Since the sequences include rare events and event combinations postulated to occur in complex systems for which there may be limited experience, the consideration of uncertainties is a vital part of understanding and determining the extent of the risk. A range of uncertainties needs to be considered and quantified in the MST calculations, including parameter uncertainty associated with the basic data and model uncertainty associated with analytical physical models and success criteria in the PRA, driven by modeling choices and by the state of knowledge about the new designs and the interactions of human operators and maintenance personnel with these systems. Sensitivity studies should be considered as an important means for examining the impacts of modeling uncertainties. All identified and quantified uncertainties (aleatory and epistemic) should be included in the MST calculations.

NEI 18-04 [22] describes the consideration of uncertainties, including from the MST, in several places, including as follows:

> The PRA's quantification of both frequencies and consequences should address uncertainties, especially those associated with the potential occurrence of rare events. The quantification of frequencies and consequences of event sequences, and the associated quantification of uncertainties, provides an objective means of comparing the likelihood and consequence of different scenarios against the F-C Target....

3.9 PRA Model Development and Update

When using the approach described in NEI 18-04 [22], PRA should be performed to model LBEs in a probabilistic manner. PRA standards, such as ASME/ANS-RA-S, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", ASME/ANS RA-S-1.2, "Severe Accident Progression and Radiological Release (Level 2) PRA Methodology to Support Nuclear Installation Applications", and ASME/ANS RA-S-1.3, "Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications," detail the processes for developing a design-specific PRA. Also, consider the use of the Non-LWR PRA Standard that is currently in development. The ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) issued "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants", ASME/ANS RA-S-1.4-2013, for trial use in 2013. In "Non-Light Water Reactor Implementation Action Plan," SECY 19-0009: Enclosure 1 [38], it is noted that the use of this trial standard by national and international organizations and feedback to the JCNRM will lead to a final draft. The PRA is not just the event tree/fault tree logic model. The PRA consists of a group of analyses which informs the logic model, which in turn informs the consequence modeling.

PRA is iterative with modeling and simulation. PRA both informs the modeling software of the potential LBE sequences and is in turn informed by the outcome of performance tools that validate and/or modify the PRA sequences discussed below. Any design change can affect the PRA, and the PRA should be used to represent the current state of the design in a probabilistic manner for risk-informed decisions.

PRA consists of two over-arching types of analyses, "static" PRA and "dynamic" PRA. Static PRA is solely based on the probability of events occurring in sequences to determine an outcome. Dynamic PRA utilizes the simulation of both probabilistic information and physics-based information.

Static PRA is used for many design and regulatory decisions. Static PRA starts with the probability of basic events occurring based on published or developed performance data. These consist of the frequency of an initiating event, such as loss of offsite power, failures of a component to perform its intended function on demand or over a period of operational time, or failures of operators to perform a specific task within an allotted time. The basic events are placed in logic trees called fault trees for each safety system. Event trees are started by an initiating event and then questioning the safety system fault trees to determine what the likelihood of a specific outcome from an initiating event is. A specific path through the logic trees to an end state provides a probability of the outcome and is called a sequence. For LWRs, all sequences that lead to an end state of core damage (CD) are gathered to calculate the core damage frequency (CDF), which is used in regulatory decisions. A PRA can extend beyond the first level of CD to describe the physical state of the plant and be used to determine the radiological consequences through dose-consequence software programs, such as MACCS. Further information can be gained from static PRAs by utilizing importance measures to determine the most important components in the system to prevent CD and radiological release. Action can be taken to improve the CDF or the state of the plant if a CD were to occur by addressing the highly important components through improvement in design such as increasing system redundancy. The CD and CDF are not descriptive of all technologies, where the "core" can be a very diverse term. By using the definition that CD allows radionuclide inventory to penetrate the first barrier of fuel cladding, a technology-inclusive way of describing the undesirable outcomes of CD and CDF is undesirable release (UR) and undesirable release frequency (URF) of radionuclide inventory from the defined barrier.

Dynamic PRA utilizes physics-based and probability-based modeling to determine the outcome of an initiating event through one sequence. While static PRA is required for regulatory decisions, including licensing, dynamic PRA is a powerful tool in determining the validity of sequence end states. Dynamic

PRA can be performed through the use of physics-based performance tools and simulation. The validation of the outcome of sequences through the event trees is one function of dynamic PRA.

PRA is developed in three levels, as is outlined in the ASME/ANS RA-S series standards. It is recommended to use the most recent edition of "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants" (currently ASME/ANS RA-S-1.4-2013) where there is any conflict between the LWR and non-LWR standard or issues related to the use of offsite consequences in decision-making versus surrogate criteria such as CDF.

The first level of a PRA models the events that cause damage to the inner-most barriers containing the fuel. Traditionally, this has been called core damage; however, the first barriers to containment of the fuel in some designs can differ from what is commonly thought of as a "core." In molten-salt reactors for instance, the fuel is contained in piping, and the "core" might be considered the fuel and piping combination. In other designs, TRISO spheres provide the first barrier within the fuel design itself, but the "core" can be considered the first containment barrier outside of the collection or matrix of TRISO pellets. For consistency, we will refer to the fuel and the first containment barrier as the core and to the first barrier breach as core damage.

The second level of a PRA models the physical state of the facility once a CD event has occurred. This logically turns on and off safety systems based on the event and informs the further capabilities of barriers, leading to consequence modeling.

The third level of a PRA models the consequence, or dose, for evaluation of EAB/LPZ radiological limits and/or the F-C target. This is a level where results can be listed as end states within the PRA ET/FT model, but it is determined by a consequence dose calculation program that utilizes radionuclide transport and dosimetry algorithms, such as MACCS. Level three PRA is informed by the source terms released from the final barrier to the atmosphere. This source term release is determined by performance tools, such as accident progression and source term programs like MELCOR.

The Non-LWR PRA Standard discusses many applications outside of the LWR PRA standard. The Non-LWR PRA Standard's scope also covers many areas outside of those found in other standards and should be used if there are any conflicts between standards. The scope of the Non-LWR PRA Standard (from ASME/ANS RA-S-1.4-2013):

a) Different sources of radioactive material both within and outside the reactor core but within the boundaries of the plant whose risks are to be determined in the PRA scope selected by the user. The technical requirements in this trial-use version of the standard are limited to sources of radioactive material within the reactor coolant system (RCS) pressure boundary (RCPB) (*and just within the RCS for a pool reactor*). Technical requirements for other sources of radioactive material such as the spent fuel system are deferred to future editions (*of the Non-LWR PRA Standard*).

b) Different plant operating states (POSs) including various levels of power operation and shutdown modes.

c) Initiating events caused by internal hazards, such as internal events, internal fires, and internal floods, and external hazards such as seismic events, high winds, and external flooding.

d) Different event sequence end states, including core or plant damage states (PDSs), and release categories that are sufficient to characterize mechanistic source terms, including releases from event sequences involving two or more reactor units or modules for PRAs on multireactor or multiunit plants.

e) Evaluation of different risk metrics including the frequencies of modeled core and PDSs, release categories, risks of off-site radiological exposures and health effects, and the integrated risk of the multiunit plant if that is within the selected PRA scope. The risk metrics supported by this standard are established metrics used in existing light water reactor (LWR) Level 3 PRAs such as frequency of radiological consequences (e.g., dose, health effects) that are inherently technology neutral. Surrogate risk metrics used in LWR PRAs such as core damage frequency and large early release frequency are not used as they may not be applicable to non-LWR PRAs.

f) Quantification of the event sequence frequencies, mechanistic source terms, off-site radiological consequences, risk metrics, and associated uncertainties, and using this information in a manner consistent with the scope and applications PRA.

The use of PRA in the development of a design determines the metrics of the current design (event sequence frequencies, iterative development of mechanistic source terms, offsite radiological consequences, risk metrics, and associated uncertainties) from the source terms that are released and provides a platform for quantifying the effects of modifications on the design for comparison to prior metrics.

3.10 Identify or Revise the List of LBEs

The plant licensing basis is, to a large extent, dependent upon risk information. The risk information obtained from the updated PRA models needs to be fed back into the licensing analysis to ensure that the plant licensing basis remains valid. This would entail updating the list of LBEs initially selected in Step 3.7 with the risk insights obtained from Step 3.9. When the updated risk information indicates that a change in the plant licensing basis is warranted, the appropriate changes will be made to update the list of LBEs.

The selection of accidents to be considered in the identification of source terms plays a lead role in the use of mechanistic source terms, because it defines the specific scenarios and associated release mechanisms used to assess such source terms. In Section 3.7, a methodology for the identification of an initial list of LBEs for non-LWR technology has been presented; those scenarios might include:

- Anticipated Operational Occurrences (AOOs)
- Design Basis Events (DBEs)
- Beyond Design Basis Events (BDBEs)
- Design Basis Accidents (DBAs)

To come up with a robust and inclusive list of LBEs for any advanced reactor technology, a systematic approach is required. LBEs are defined as the events derived from the reactor technology and plant design of interest that are used to derive design-specific performance requirements for structures, systems, and components and are generally inferred from the licensing process. Considering that the selection of such events needs to be performed, potentially, for new technologies, a combination of deterministic and probabilistic methods should be used for both the identification and consequence assessment of such events.

The selection process needs to be considered as an integral part of the overall design process and, consequently, it must be “re-iterated” since its selection (and outcomes) informs the design requirements of safety-related and non-safety-related systems and components. Once an initial set of LBEs is identified, the design can be refined to reduce the likelihood or associated risk of a specific LBE.

The process can be exemplified in multiple stages:

1. A deterministic approach is used to select an initial event set providing a starting point for the assessment of the source terms.
2. The LBEs are updated every time the design and analysis evolve.
3. A review of the LBEs is performed at the end of the design phase to evaluate conservatisms in the selected events.

3.11 Select LBEs to Include Design Basis External Hazard Level for Source Term Analysis

External events are chosen deterministically on a basis consistent with that used for LWRs (SECY-19-0117: Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors) [39]. A set of design basis external hazard levels (DBEHLs) will be selected to form an important part of the design and licensing basis. This will determine the design basis seismic events and other external events that the safety-related SSCs will be required to withstand. When supported by available methods, data, design, site information, and supporting guides and standards, these DBEHLs will be informed by a probabilistic external hazards analysis and will be included in the PRA after the design features that are incorporated to withstand these hazards are defined. Other external hazards not supported by a probabilistic hazard analysis will be covered by DBEHLs that are determined using traditional deterministic methods.

3.12 Perform LBEs Source Term Modeling and Simulation

As previously mentioned, the selection of the LBEs to include in the source term calculations is an iterative process that needs to be repeated in any stage of the design (or when substantial changes to the design are made).

The source term assessment needs to characterize the generation, release, transport, and retention of fission product and activation radionuclides. The modeling of such phenomena requires identification of the "barriers" for the technology of interest. The "barriers" provide mechanisms for the retention of the fission products during normal operation and accident conditions. The process for the development of modeling and simulation tools for non-LWR applications is similar to LWR applications. Once the LBEs are selected and the modeling tools are available, the actual simulation effort can be initiated. These requirements are described in the following subsections.

3.12.1 Requirements for Source Term Modeling and Simulation

Since the publication of "NRC Non-Light Water Reactor (non-LWR) Vision and Strategy – Staff Report: Near-Term Implementation Action Plans," November 2016 (ADAMS Accession No. ML16334A495) [1], there has been dialogue between NRC staff, ACRS, DOE, and industry representatives on computer codes and tools to perform source term modeling and simulation for non-LWRs.

The NRC plan was presented to ACRS on May 1, 2019 (ADAMS Accession No. ML19143A120 [40]) and October 3, 2019 to discuss the NRC staff's ongoing code development to support independent analysis for licensing of non-LWR designs. In its letter of November 4, 2019 (ADAMS Accession No.

ML19302F015 [41]), ACRS emphasized that, ideally, the tools for staff confirmatory analysis should be as independent as practical, validated, understood by the staff, and usable on the staff's computer resources. The ACRS stated that the staff also needed to become sufficiently familiar with applicants' codes to support timely reviews of submitted analyses. The ACRS stated that four principles should underlie the strategy: simplicity, completeness, working the problem backwards from the source term, and scaling down the level of effort of licensing review proportionately as the hazard decreases. The staff likewise advocates the strategies underlying these principles.

The staff's source term evaluation model for non-LWR applications is shown in Figure 3-3. This model is technology-inclusive because it relies on the same codes with the suite of physics models needed for the different non-LWR technologies. A detailed description of these codes and the development process, including identification of technical gaps, is provided in NRC's "Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis" (ADAMS Accession No. ML20030A178 [42]).

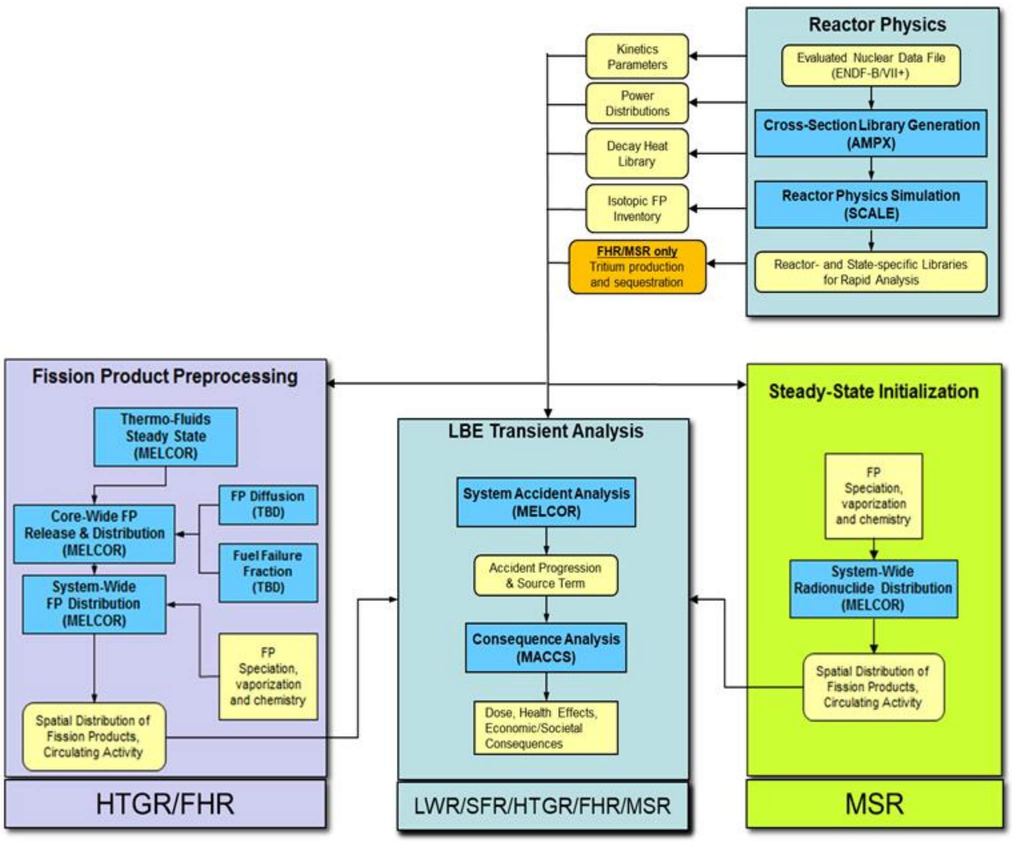


Figure 3-3 NRC evaluation model plan for source term characterization.

In 2020, the NRC began analysis of severe accident progression and source term for three representative advanced reactor designs. This effort is focusing on severe accident phenomenology and source term development and was presented at an advanced reactor stakeholder meeting on February 20, 2020 (ADAMS Accession No. ML20040E155 [43]). The three designs, which have publicly available data, are the following: (1) an HTGR, (2) a liquid-metal-cooled heat pipe reactor plant model (e.g., Los Alamos National Laboratory MegaPower reactor), and (3) a molten-salt-cooled pebble bed reactor plant model (e.g., University of California-Berkeley's Mark I Pebble Bed Fluoride-Salt-Cooled High-Temperature Reactor). In the first phase of this effort, MELCOR is being used to demonstrate how beyond design basis accident progression and source terms can be characterized for the selected three non-LWR design concepts. In the second phase, the MELCOR study results will be used to inform NRC staff, promoting the knowledge and insights needed to:

- Understand beyond DBEs for non-LWR technologies
- Develop guidance to support staff review of non-LWR applications in a timely and efficient manner.

In the final phase of this effort, workshops will be held to inform stakeholders on the staff's approach to perform independent source term analysis for the three representative non-LWR designs to promote dialogue between NRC and stakeholders. The intent of these workshops is to provide sufficient information to reduce uncertainty in the review process for non-LWR vendors developing design-specific source terms.

ACRS was briefed by the U.S. Department of Energy (DOE) concerning the capabilities of DOE computer codes and by industry representatives. The DOE presentation to the ACRS on August 21, 2018 (ADAMS Accession No. ML18254A164 [44]) outlined the DOE strategy for advanced (non-LWR) reactor safety analysis and involved various areas, including neutronics analysis capabilities, fuels modeling capabilities, thermal-hydraulic/system analysis, and source term assessment codes. For the source term analysis, an example involving application of DOE codes for a liquid-metal reactor application is shown in Figure 3-4.

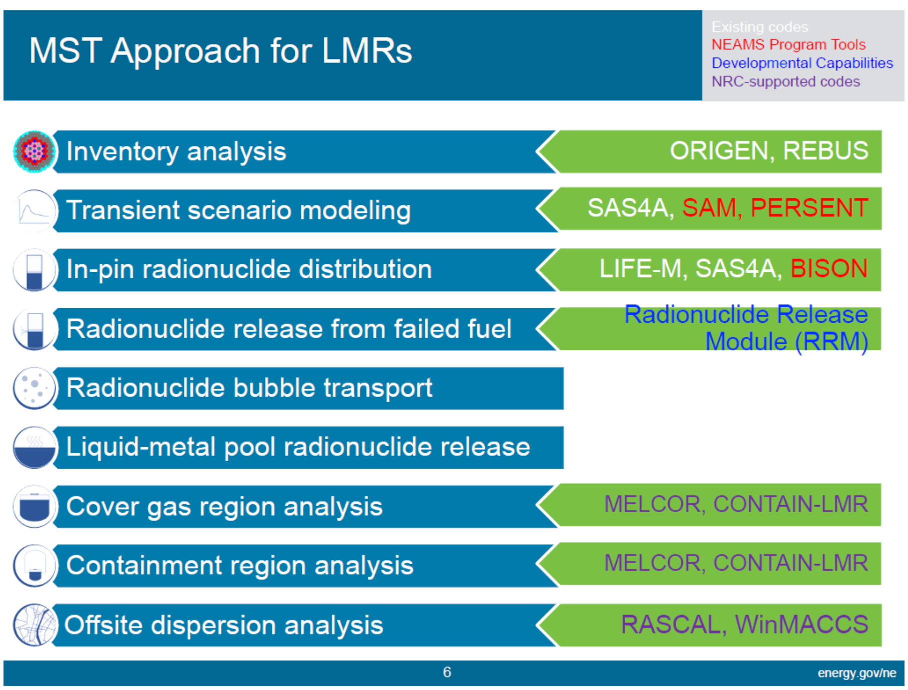
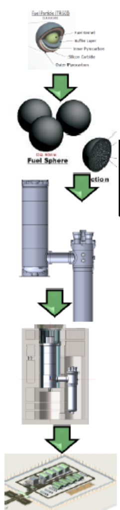


Figure 3-4 DOE code strategy liquid-metal reactor example for source term characterization.

ACRS was briefed on November 16, 2018 (Transcript at ADAMS Accession No. ML18340A016 [45]) by industry representatives working in MSR, SFR, and HTGR source term methodology. The vendors were engaged in efforts to characterize the source term, due to its importance in the safety analysis. The degree of computer code development and technical approach by different vendors varied.

As an example of a vendor's approach to characterize source term, Figure 3-5 below (Page 394 of ACRS Transcript) shows that the vendor X-Energy is applying a combination of in-house developed codes, such as XSTERM, and NRC codes such as SCALE and MELCOR. (The codes labeled in the Figure 3-5 as "US/DOE" are NRC codes that are being developed by the DOE national laboratories and the University of Michigan for NRC staff independent analysis).

Source Term Calculation



Source Term Path	Element / Isotope	Form / State	Mechanism	Physical Phenomena	Methods / Software Codes
Releases from TRISO fuel particles	Iodine Silver Strontium Cesium	Gaseous FPs Metallic FPs	- Release from TRISO particles into matrix graphite - Activation of impurities	Temperature, irradiation time, fast fluence, burnup, particle defects, contamination	VSOP-A, VSOP-99, MGT SCALE, PARCS, OREGIN FLOWNEX, STAR-CCM+ XS-Term
Releases from fuel elements (pebbles)	Iodine, Silver Strontium Cesium	Gaseous FPs Metallic FPs Dust Particles	- Diffusion from pebble into the helium stream - Activation of impurities	Temperature, irradiation time, fast fluence, burnup, contamination	VSOP-A, VSOP-99, MGT SCALE, PARCS, OREGIN FLOWNEX, STAR-CCM+ XS-Term
Releases from Pressure boundary	Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Leakage from HPB into building and structures - Activation of impurities	Instrumentation line failure, small & large pipe breaks, plate-out, liftoff	ORIGEN XS-Term
Releases from building	Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Transport throughout building to the environment	Plate-out, liftoff	XS-Term MELCOR
Max dose at site boundary	Iodine, Silver Strontium, Cesium	Gaseous FPs Metallic FPs Dust Particles	- Atmospheric dispersion - Ingestion	Postulates	XS-Term STAR-CCM+

Color Legend: Legacy codes US/DOE Codes X-energy in house code Commercial NQA-1 Code

 Nuclear Energy. Reimagined. 11

Figure 3-5 X-Energy plan for source term characterization.

In general, as shown in the discussion above, the prediction of source term often involves the use of multiple codes that "answer" to different functional requirements:

- Reactor Physics Computer Models:
  - Calculate radionuclide inventories and power distributions in the design.
- Fuel Performance Computer Models:
  - Calculate thermal and stress histories for fuel and identify fuel failure and radionuclide release.
- System Analysis Computer Models:
  - Calculate the progression of accident and radionuclide transport.
  - Requires boundary conditions from fuel performance analysis.
- Radionuclide Transport Models (linked to system analysis models):
  - Calculate radionuclide release and transport within the reactor and surrounding structures.
  - Calculate radionuclide transport from the reactor to the EAB and transport in the atmosphere (plume dispersion).
- Dosimetry Computer Models (linked to radionuclide transport models):
  - Calculate doses within and outside the site boundaries during normal operation and accident conditions. Used to determine whether the plant design meets offsite dose limits and criteria and risk goals.
- Uncertainty Assessment Computer Models:

- Categorize the uncertainties associated with the events' source terms and select the most impactful ones to be considered.
- These models are used in conjunction with the previously mentioned models to characterize the quantification and propagation of uncertainties and perform sensitivity analysis.

3.12.2 Evaluate LBEs Source Term Calculations Against F-C Target

The risk significance of individual LBEs is evaluated against the F-C target (see Figure 1-1). The uncertainties in mechanistic source term determinations and risk assessments are evaluated quantitatively in conjunction with the analytic and testing programs.

3.12.3 Evaluate Cumulative Risk Against QHOs and 10 CFR 20

The following are definitions of the Quantitative Health Objectives (QHOs) taken directly from the NRC 1986 Safety Goal Policy Statement [46]:

- "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed."
- "The risk to the population in the area of nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes."

The average individual risk of prompt (or early) fatality and latent cancer fatality that is calculated in the PRA to compare with the safety goals and the QHOs is the total plant risk incurred over a reactor year. This means the PRA results need to demonstrate that the total plant risk, i.e., the risk summed over all of the accident sequences in PRA, needs to satisfy both the latent cancer QHO and the early fatality QHO. The safety goals, and consequently, the QHOs are phrased in terms of the risk to an 'average' individual in the vicinity of a nuclear power plant per reactor year. The latent cancer QHO is defined in terms of the risk to an average individual within 10 miles and the early fatality QHO in terms of the risk to an average individual within 1 mile of the plant. Therefore, the PRA results need to show that the total integrated risk from the PRA sequences satisfy both the latent cancer QHO and the early fatality QHO. The following objectives should be met in evaluating cumulative risk:

- The total frequency of exceeding a EAB dose of 100 mrem (annual cumulative exposure limits in 10 CFR 20) from all LBEs should not exceed 1/plant-year.
- The average individual risk of early fatality within 1 mile of the EAB from all LBEs shall not exceed 5×10^{-7}/plant-year to ensure that the NRC safety goal QHO for early fatality risk is met.
- The average individual risk of latent cancer fatality within 10 miles of the EAB from all LBEs shall not exceed 2×10^{-6}/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

3.12.4 Identify Risk Significance of LBEs and Perform MST Calculations Against Regulatory Criteria

LBEs are classified in NEI-18-04, which is endorsed by RG 1.233, as risk-significant if the LBE EAB dose exceeds 2.5 mrem over 30 days and the frequency of the dose is within two orders of magnitude of the F-C target. Each design will establish barriers to the release of radioactive material from the fuel, RCS, or other systems, to maintain doses to below the criteria defined for various anticipated or postulated conditions. The specific conditions for each barrier's leakage, temperature, pressure, and time response will be design and event specific. The success of a barrier or combination of barriers in preventing releases from each source within the SRIR may simplify the assessment and preclude the need to assess offsite consequences.

In lieu of event-specific assessments, the analysis of offsite consequences or the leakage past specific barriers may be based on an MCA. Likewise, the leakage from an individual barrier could be assumed for an individual LBE, based on the worst conditions for that barrier from all LBEs. The definition of the MCA or events, as applicable, should be agreed upon between the applicant and the NRC consistent with the technology and safety characteristics of the design. For an MST, the timing, magnitude, and the form of radionuclides released into the barriers and the resulting temperature, pressure, and other environmental factors (e.g., combustible gas) in the barriers during the event should be analyzed mechanistically, with uncertainty considered. Using conservative assumptions is permitted in the MST and MCA dose calculations. For example, the timing of closure and the allowable leak rate is then established such that the worst two-hour dose at the EAB and the dose at the outer edge of the LPZ for the duration of the event do not exceed 25 rem TEDE.

3.13 Select a Final List of LBEs

Since the regulatory structure for advanced reactor technology licensing makes use of PRA, the selection of LBEs may not be a one-time licensing step, carried out at the time of initial plant licensing and remaining fixed. Instead, it is expected that both the selection of LBEs and the safety classification of SSCs may change as the reactor design is evolved and matured, and over the lifetime of the plant operations as new information and operational experience add to, and reshape, the risk insights from maintaining and updating the PRA.

The LBE evaluation provides feedback on whether additional improvements on design and operation should be considered. Such improvements could be motivated by a desire to increase margins against the F-C target criteria, reduce uncertainties in the LBE frequencies or consequences, limit the need for restrictions on siting or emergency planning, or enhance the performance against DID criteria. If improvements are needed, then go back to 3.6. If no improvements are needed, the final list of LBEs and safety-related structures, systems and components is established.

3.14 Documentation of Source Terms and Dose Rates

A document will be prepared to show the calculations of the source terms and dose rates for use in licensing, such as for the bounding analysis case or for the final list of LBEs. This information will be submitted to the NRC for approval as part of an application for a licensing action. The methodology used and scenarios analyzed for the source term and dose rate calculations should be presented in the document. The results from risk-informed and performance-based mechanistic source term calculations should include uncertainty quantification, as applicable, in both in the PRA models and in the mechanistic source term calculations.

4. SUMMARY

A risk-informed, performance-based, technology-inclusive determination of source terms for dose-related assessments for advanced nuclear reactor facilities is developed in this report to support the NRC's Non-LWR Vision and Strategy Near-Term Implementation Action Plans (ADAMS Accession No. ML16334A495 [1]) and the NRC's response to the NEIMA Public Law No: 115-439, of January 2019 [2]. This approach uses a graded process that allows both non-mechanistic source term calculation methods, which adopt conservative approaches and assumptions based on known physical and chemical principles, and, more importantly, the risk-informed and performance-based mechanistic source term calculation methods, which consider design-specific scenarios and use best-estimate models with uncertainty quantification for a range of LBEs, to be used for the design and licensing of advanced nuclear technologies.

The source terms developed with this graded approach and radionuclide inventories elsewhere in the facility that are determined during source term analysis can be used to address licensing issues to support the 10 CFR 52 Combined License (COL) application process. They can also be used for other purposes, including equipment environmental qualification, control room habitability analyses, and assessments of severe accident risks in environmental impact statements. The graded approach presented in this report for source term determination is, to the extent possible, generic to any of ongoing reactor designs and future reactor designs. It provides information on the review of the regulatory foundation for use of conservative bounding source terms as well as event-specific mechanistic source terms for advanced nuclear reactor designs.

5. REFERENCES

[1] U.S. Nuclear Regulatory Commission, "NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy - Staff Report: Near-Term Implementation Action Plans." (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16334A495).

[2] PUBLIC LAW 115–439—JAN. 14, 2019, "An Act to modernize the regulation of nuclear energy."

[3] U.S. Code of Federal Regulations, "Standards for Protection Against Radiation." Part 20, Title 10, "Energy." (10 CFR Part 20).

[4] U.S. Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy." (10 CFR Part 50).

[5] U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1350, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and NonPower Production or Utilization Facilities." August 2018.

[6] U.S. Nuclear Regulatory Commission, SECY-16-0012, "Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors." February 7, 2016.

[7] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors." June 2020.

[8] U.S. Nuclear Regulatory Commission, SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements." April 8, 1993.

[9] U.S. Nuclear Regulatory Commission, SECY-15-077, "Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies." May 29, 2015.

[10] U.S. Nuclear Regulatory Commission, SECY-03-0047, "Policy Issues Related to Licensing Non-Light Water Reactor Designs." March 28, 2003.

[11] U.S. Atomic Energy Commission, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." March 23, 1962.

[12] U.S. Atomic Energy Commission, Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactor." June 1974.

[13] U.S. Atomic Energy Commission, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactor." June 1974.

[14] U.S. Code of Federal Regulations, "Reactor Site Criteria," Part 100, Title 10, "Energy." (10 CFR Part 100).

[15] U.S. Nuclear Regulatory Commission, NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." February 1995.

[16] U.S. Nuclear Regulatory Commission, RG-1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." July 2000.

[17] U.S. Nuclear Regulatory Commission, SECY-94-302, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs." December 19, 1994.

[18] "Mechanistic Source Terms White Paper." INL/EXT-10-17997, July 2010.

[19] U.S. Nuclear Regulatory Commission, SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors." October 16, 2018. (ADAMS Accession No. ML18114A546).

[20] U.S. Nuclear Regulatory Commission, SRM-SECY-18-0096, "Staff Requirements – SECY-18-0096-Functional Containment Performance Criteria for Non-Light-Water-Reactors." December 4, 2018.

[21] U.S. Nuclear Regulatory Commission, SECY-18-0103, "Proposed Rule: Emergency Preparedness for Small Modular Reactors and Other New Technologies (RIN 3150-AJ68; NRC-2015-0225)." October 12, 2018.

[22] NEI Technical Report, NEI-18-04, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development." September 28, 2018.

[23] David L. Luxat, "Simplified Approach for Scoping Assessment of Non-LWR Source Terms." SAND2020-0402, January 2020.

[24] U.S. Nuclear Regulatory Commission, NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, Volumes 1 and 2." December 2007.

[25] U.S. Nuclear Regulatory Commission, NUREG-1614, Vol. 3, "Strategic Plan: FY 2004 – FY 2009." August 2004.

[26] U.S. Nuclear Regulatory Commission, SECY-98-144, "White Paper on Risk-Informed and Performance-Based Regulation." June 22, 1998.

[27] U.S. Code of Federal Regulations, "Rules of General Applicability to Domestic Licensing of Byproduct Material," Part 30, Title 10, "Energy." (10 CFR Part 30).

[28] U.S. Code of Federal Regulations, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Part 52, Title 10, "Energy." (10 CFR Part 52).

[29] U.S. Environmental Protection Agency, "Environmental Radiation Protection Standards for Nuclear Power Operations." Part 190, Title 40, "Energy." (40 CFR Part 190).

[30] U.S. Environmental Protection Agency, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents." EPA-400/R-17/001, January 2017.

[31] U.S. Nuclear Regulatory Commission, NUREG-1320, "Nuclear Fuel Cycle Accident Analysis Handbook." May 1988.

[32] U.S. Department of Energy, DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities." December 1994.

[33] U.S. Nuclear Regulatory Commission, NUREG/CR-2260, "Technical Basis for Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."" October 1981.

[34] U.S. Department of Defense, MIL-STD-1629A, "Procedures for Performing a Failure Mode, Effects and Criticality Analysis." November 24, 1980.

[35] Leveson, N.G., *Engineering a safer world: systems thinking applied to safety*. Cambridge: The MIT Press, 2012.

[36] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." March 2009.

[37] U.S. Nuclear Regulatory Commission, NUREG/CR-6944, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)." March 2008.

[38] U.S. Nuclear Regulatory Commission, SECY-19-0009, "Advanced Reactor Program Status." February 4, 2019. (ADAMS Accession No. ML18346A075).

[39] U.S. Nuclear Regulatory Commission, SECY-19-0117, "Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors." December 17, 2019. (ADAMS Accession No. ML18311A264).

[40] U.S. Nuclear Regulatory Commission, Official Transcript of Proceedings, "Advisory Committee on Reactor Safeguards Future Plant Designs Subcommittee." May 1, 2019. (ADAMS Accession No. ML19143A120).

[41] U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, "Review of Advanced Reactor Computer Code Evaluations." November 4, 2019. (ADAMS Accession No. ML19302F015).

[42] U.S. Nuclear Regulatory Commission, "Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis." January 31, 2020. (ADAMS Accession No. ML20030A178).

[43] U.S. Nuclear Regulatory Commission, "Advanced Reactor Stakeholder Public Meeting." February 20, 2020. (ADAMS Accession No. ML20050E155).

[44] U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, "ACRS Thermal-Hydraulic Subcommittee." August 21, 2018. (ADAMS Accession No. ML18254A164).

[45] U.S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, "Advisory Committee on Reactor Safeguards Thermal-Hydraulic Phenomena Subcommittee." November 16, 2018. (ADAMS Accession No. ML18340A016).

[46] U.S. Nuclear Regulatory Commission, "Safety Goals for the Operations of Nuclear Power Plants: Policy Statement; Republication." August 1986.

[47] D. A. Petti, R. R. Hobbins, P. Lowry, and H. Gougar, "Representative source terms and the influence of reactor attributes on functional containment in modular high-temperature gas-cooled reactors," *Nuclear Technology* **184** (2013) 181–197.

[48] "Scoping Analysis of Source Term and Functional Containment Attenuation Factors." INL/EXT-11-24034, February 2012.

[49] U.S. Nuclear Regulatory Commission, SECY-18-0076, "Options and Recommendation for Physical Security for Advanced Reactors", November 18, 2018, (ADAMS Accession No. ML18170A051, related staff requirements memo ML18324A478)

APPENDIX A: EXAMPLE OF ANALYSIS

This appendix provides an overview of how the methodology might be applied to an advanced reactor design, using a high-temperature gas-cooled reactor as a representative example. For each step in the methodology, a brief overview of the corresponding action or activity is given, and some representative examples of the kind of analysis and output expected from each step are given. These are not intended to be complete. In many cases, numerical values are used as example inputs or outputs of a calculation or analysis; it is important to note that these are only hypothetical and for the purpose of illustration only. They do not represent the results of actual analysis nor are necessarily representative of any particular reactor design or this reactor type generally.

Step 1: Identify Regulatory Requirements

Applicable regulatory requirements and dose limits have been outlined in Section 3.1 and Table 3-1. These are applicable to any reactor type, including an HTGR. For the purpose of this example, consider a prospective site of location and size that dictates the EAB be at most 300 m from the reactor. In the proceeding analysis, the applicant must demonstrate that that the regulatory requirements outlined in Table 3-1 are met for this particular EAB.

Step 2: Identify Reference Facility Design

The reference facility design is that described in [47], a single module 600 MWt thermal prismatic Modular HTGR (MHTGR), with a 700°C helium coolant outlet temperature. The reactor produces high-temperature steam via a steam generator. Barriers and processes important to the transport of fission products in the reactor are illustrated schematically in Figure A- 1.

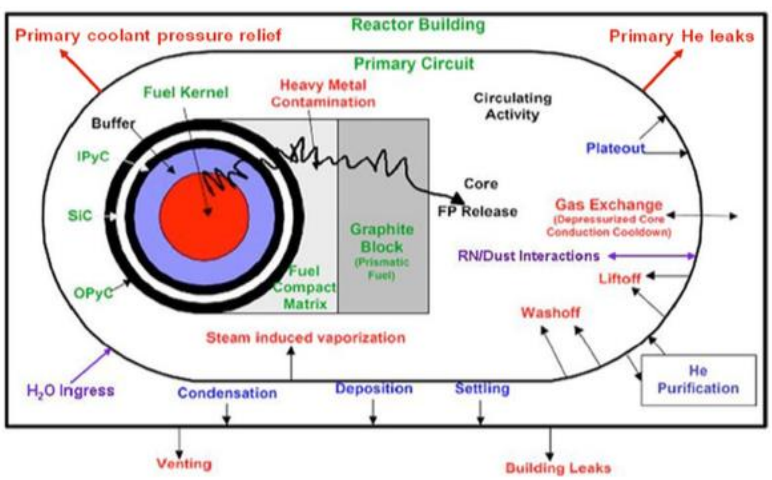


Figure A- 1 Barriers to fission product transport in a HTGR [18].

The reactor uses TRISO fuel, and this constitutes the primary barrier to fission product release. To be released from a fuel particle, radionuclides must be transported through and out of the fuel kernel itself and subsequently through each of the buffer, inner pyrolytic carbon, SiC, and outer pyrolytic carbon layers. Small fractions of the fuel with defects in one or more of the layers may dominate the release for a given radioisotope. Fission products that escape the fuel itself may be retained in the surrounding fuel compact matrix and graphite block; fission products that are transported through these materials are released to the primary coolant. Fission products circulating in the primary coolant may be removed by a coolant purification system; some fraction of these will be deposited on surfaces throughout the primary circuit but are retained by this barrier under normal operating conditions. Accidents involving a breach of the primary circuit may allow circulating activity or re-entrained deposits to be released to the reactor building, where some will be deposited via condensation, settling, or other mechanisms or removed by filters before they can be released through building leaks or vents to the environment.

Step 3: Define Initial Radionuclide Inventories

The applicant defines initial radionuclide inventories via a series of analyses:

A. Neutronics analysis to obtain the spatially varying neutron flux and energy spectrum in the core.

B. Radionuclide generation rates from fission (with input from the neutronics analysis), accounting for activation, decay, etc.

C. The radionuclide generation rates constitute a mass source input to a fission product transport code, which determines how these are distributed throughout the reactor system at the initiation of the accident. Such a code would incorporate models for transport through all the barriers delineated in Figure A- 1, including the kernel and multiple layers of both intact and defective or failed TRISO fuel; models for transport through matrix and graphite materials, and release from these to the primary coolant; and models for transport throughout the primary circuit, incorporating any models necessary to describe the mechanisms involved in deposition or resuspension of radionuclides on/from surfaces. The end results are fission product inventories deposited on different components and portions of the primary circuit, plus inventories remaining in all parts of the TRISO fuel, matrix, and graphite materials in different regions of the core. Table A-1 shows the initial core fission product inventories for the 600 MWt thermal MHTGR reactor design.

Table A- 1 Initial Core Fission Product Inventories for the 600 MWth MHTGR [47, 48].

Fission Product Class	Characteristic Nuclide	Inventory (Curies)
Noble gases	^{133}Xe	3.63E+07
	^{85}Kr	1.90E+05
	^{88}Kr	1.85E+07
I, Br, Te, Se	^{131}I	2.00E+07
	^{133}I	3.60E+07
	^{132}Te	2.71E+07
Cs, Rb	^{137}Cs	1.69E+06
	^{134}Cs	1.90E+06
Sr, Ba, Eu	^{90}Sr	1.69E+06
Ag, Pd	^{110m}Ag	2.81E+04
	^{111}Ag	2.96E+06
Sb	^{125}Sb	2.35E+05
Mo, Ru, Rh, Tc	^{103}Ru	3.61E+07
La, Ce groups	^{144}Ce	2.33E+07
	^{140}La	3.27E+07

Fission Product Class	Characteristic Nuclide	Inventory (Curies)
Pu, actinides	^{239}Pu	4.66E+03

Step 4. Perform Bounding Calculations

Among many other isotopes, the applicant finds, as a result of the analysis in Step 3, that 1.7 MCi of ^{90}Sr are present in the reactor at the initiation of the accident scenario. Schedule C of 10 CFR Part 30 indicates that a release of 1% of the ^{90}Sr inventory must not exceed 90 Ci if no emergency plan is to be considered. In this case, 1% of the ^{90}Sr inventory is 17,000 Ci, far in excess of the 90 Ci limit. Without consideration of any other isotopes, simple bounding analysis is insufficient in this case, and the applicant should proceed with a mechanistic source term analysis.

Step 5. Conduct SHA and Perform Simplified Calculations

Conduct SHA

The applicant performs a SHA on the system to identify the SSCs and barrier penetration pathways and estimate the component and/or interaction likelihood of failures and severity. A team of experts consisting of the designers and subject matter experts are gathered and queried for this task. A partial outcome of such an analysis may include Table A-2, which uses an FMEA. FMEAs use a risk priority number (RPN) to quantify the priority of the hazards, should the design team have the capability to address them. The scale in this FMEA is on a 1–10 for severity, frequency, and detection. The high end of the severity scale for a reactor indicates that a release through the final barrier can be expected. Frequency in the SHA step is relative and estimated. Note that detection is inverse, in that it represents the inability to detect the fault/hazard.

The SHA can inform a redesign, based on recommended actions that, if implemented, will change the likelihood of the hazard, the SSCs in the barrier penetration pathway, or even the barrier penetration pathway itself. The hypothetical examples given in Table A-2 show that two recommended actions are viable for the four failure modes listed. Recommended actions are determined and are generally done by using either an RPN threshold (such as above RPN > 50 in this example) or something that is easy and cost effective to implement that lessens a high severity failure mode, such as adding filters to the building ventilation system in this hypothetical example. The other possible improvement actions in this example, such as possibly increasing the durability of the steam generator tubes or redesigning TRISO fuel, are not economically viable and are left as is.

The SHA is kept current with any design changes. Each new iteration of the SHA risk-informs the design, the simplified source terms quantification or it will inform the PRA. The SHA in this example will be used to inform the PRA.

Table A- 2 Example of FMEA Results.

Process Function	Potential Failure Mode	Potential Causes/ Mechanisms of Failure	Severity	Frequency	Detection	RPN	Recommended Action	Implemented / Date	Severity	Frequency	Detection	RPN
Transport of coolant	Pipe Rupture	Loss of coolant	7	4	2	56	Vibration dampeners	Yes / 29Oct20	7	2	2	28
Heat exchange	Steam generator tube break	Water ingress	7	3	2	42						
Fission Product Retention	FP release from fuel barriers	Particle failure during accident	9	1	2	18						
Environmental Controls	Building overpressure	Venting to environment	9	1	1	9	Add filters	Yes / 10Oct20	6	1	1	6

Perform Simplified Calculations

For the 600 MWth MHTGR, previous safety analyses indicate that breaks in the helium pressure boundary and water ingress events pose the greatest challenges with respect to offsite dose consequences [47, 48]. Step 7 has a more detailed description of these two accidents. The applicant adopts a simplified approach in which attenuation factors (inverse of release fraction) are assigned to a series of barriers to release. The attenuation factors are based on experimental data in conditions intended to bound the range of temperatures experienced in bounding accidents, and the NRC must approve of the specific methodology applied in this case. Some of the determined attenuation factors for non-intact fuel (TRISO failure) are shown in Table A-3. It is the retention in the fuel kernel itself that leads to attenuation in this case.

Table A- 3 Attenuation Factors for non-intact fuel (TRISO failure) for simplified source term calculations [47].

Fission Product Class	**Attenuation Factors: Accident Release from Non-Intact Fuel**	
Confidence Limit	**50%**	**95%**
Noble Gases	10	5
I,Br,Se,Te	10	5
Cs, Rb	1	1
Sr,Ba,Eu	1	1
Ag, Pd	1	1
Sb	1	1
Mo,Ru,Rh,Tc	100	50
La, Ce	100	50
Pu, Actinides	1000	500

If the results from the scoping analysis indicate that the NRC Siting and EPA PAG plume exposures criteria are met, the applicant could skip steps 6–13 and proceed to the last Step. In the example above, no additional retention of certain fission products (including ^{90}Sr from the preceding step) can be assumed for failed fuel during accident conditions. Therefore, the applicant proceeds with the mechanistic approach beginning with the next step.

Step 6. Consider Risk-informed System Design Changes

Based on the recommendations of the FMEA in the preceding step, the applicant decides to add vibration dampeners, in order to decrease the frequency of pipe rupture events that would lead to a loss of coolant. Realizing that such a loss of coolant could result in a building overpressure that necessitates venting directly to the environment, the applicant additionally decides to incorporate filters that would retain fission products and thereby reduce the severity of such an occurrence.

Step 7. Select Initial List of LBEs and Conduct PIRT

Based on the findings of the FMEA in Step 5, the applicant identifies an initial list of LBEs "which may not be complete but are necessary to develop the basic elements of the safety design" [22]. Two events in the initial list of LBEs include those described in [47, 48]:

1. A break in the helium pressure boundary with loss of forced cooling:
    a. Leak or break in the helium pressure boundary piping up to the largest connecting pipe
    b. Reactor trip
    c. Loss of heat transport to the energy conversion system
    d. Loss of shutdown cooling
    e. Immediate depressurization of helium in the helium pressure boundary
    f. Opening of the RB vent to relieve helium pressure.

2. A water ingress event:
    a. Steam generator tube break
    b. Reactor trip
    c. Loss of heat transport to the energy conversion system
    d. Loss of shutdown cooling
    e. Detection of water ingress
    f. Isolation of the steam generator main steam and feedwater lines
    g. Over-pressurization of the helium pressure boundary through the vessel system relief valve
    h. Opening of the reactor building vent to relieve helium and water/steam pressure.

These are thought to encompass all of the relevant transport phenomena that might occur in HTGR accidents, including those resulting from steam interactions and transport in the reactor building plus all of the same transport phenomena occurring inside the helium pressure boundary, core, and fuel that occur during less severe accidents. They are therefore sufficient to develop the basic elements of safety design.

The applicant proceeds to conduct a PIRT to identify the phenomena relevant to the progression of these scenarios and the importance and current knowledge base of these. Such a PIRT has been conducted for the HTGR and is documented in [37]. Some example entries are given in Table A- 4.

Table A- 4 PIRT Sample Results.

Phenomenon	Importance	Rationale	Knowledge Level	Rationale	Model Status
FP transport through fuel block	5 – High	Effective release rate coefficient (empirical constant) as an alternative to first principles (IC and Trans.)	1 – Low 4 – Medium	Depends on specific graphite; expected from material PIRT	Major need
Steam attack on graphite	5 – High	If credible source of water present; design dependent (Trans.)	1 – Low 4 – Medium	Historical data	Major need for severe accidents
Aerosol/dust deposition	5 – High	Gravitational, inertial, thermophoresis, electrostatic, diffusional, turbophoresis (Trans.)	5 – Medium	Reasonably well-developed theory of aerosol deposition by most mechanisms except inertial impact in complex geometries; applicability to NGNP unclear	Minor Mod

Step 8. Establish Adequacy of MST Simulation Tools

At this stage, the applicant has already developed a code intended to model all aspects of fission product transport, in normal and off-normal conditions. As a result of the PIRT findings in the preceding step, the applicant has identified that:

1. Their existing model for fission product transport in graphite is rather uncertain and not adequately informed by relevant experiment data.
2. It does not presently include any model for steam interaction with graphite.
3. The applicant has a model for aerosol transport in the reactor building, but there is a question as to whether some of the deposition mechanisms apply to HTGRs and other mechanisms (i.e. inertial impaction) are thought to be important.

In response to the first two findings, the applicant plans some additional post-irradiation experiments on their fuel and graphite materials. The first involves heating compacts with failed particles in order to observe the resultant distribution of mobile fission products that are transported into the graphite; this data is used to update the models for fission product transport in graphite. The second involves heating fuel compacts and graphite in a furnace in helium atmospheres with varying amounts of steam, at temperatures representative of a severe accident condition. The data collected during these experiments is used to inform a graphite-steam oxidation model that the applicant develops and incorporates into their code.

In response to the third finding, the applicant identifies a large body of data on inertial impaction of dust and aerosols in the existing literature and uses these to develop and implement a model for this into their code. They also review the applicability of the various other deposition mechanisms. Finding some uncertainty as to whether these are applicable or not, the applicant decides to conservatively model only gravitational, inertial, and diffusional deposition.

Step 9. Develop and Update PRA Model

The applicant conducts a PRA to take the hazards and severities identified in the SHA, determine the frequencies of initiators and the probability of failure of mitigating actions, and place them into logic trees. Frequencies were defined within ranges in the SHA. In the PRA, the initiating event frequency (per year) is quantified for each of the events identified. The mitigating systems are modeled as success or failure in fault trees, based on the probability of failure of required components and operator actions. Following the guidance in Section 3.9, the total URF for all sequences from one initiator to UR is the URF for one event. The total sequences from all initiators that lead to UR are summed for a total URF.

The PRA is developed to the current state of knowledge of the design and of the initiating event frequencies and resulting URFs before moving on to the selection of LBEs. The PRA should be kept current to provide a tool to use for risk-informed decision-making and to provide sequences that lead to UR for inclusion in the modeling and simulation step.

In some instances, the results of the modeling will in turn validate or invalidate the sequences developed in the PRA. A sequence tested through modeling that is thought to lead to UR may not, or vice versa. The PRA is then updated to reflect the new information gained through modeling and simulation, and the LBE selection step and modeling is performed again, as necessary, until all are in agreement.

Step 10. Identify or Revise the List of LBEs

Based on the evaluation of the risk information, the applicant expands the list of LBEs to include normal operation and a comprehensive set of AOOs, DBEs, and BDBEs. For example, if the scenarios leading to loss of coolant outlined in Step 7 are determined by the PRA to have an initiating event frequency between 10^{-2} and 10^{-4} per year, these would be classified as DBEs. Examples of LBEs at the far ends of the frequency-consequence spectrum that might be considered include:

- Analysis of tritium transport during normal operations. Tritium generation rates in the HTGR are not high, but tritium is uniquely mobile and may be able to diffuse through parts of the primary helium circuit even during normal operation.
- A severe accident such as a large-break loss-of-coolant accident that results in significant air or steam ingress into the graphite or core. This event is determined to be very unlikely and beyond the design basis.

Step 11. Select LBEs to Include Design Basis External Hazard Level for Source Term Analysis

External hazards are site specific and not design specific. The external hazards considered for the PRA model of the MHTGR use a full-scope PRA treatment of internal and external hazards. However, it is expected that the selection of LBEs performed in Step 10 is based on a PRA that includes internal events but has not yet been expanded to address external hazards. The external events encompass all potential hazards applicable to the site and could include seismic, flooding, high winds, and external fires. It is reasonable to expect that safety function failures will be dominated by events and conditions that exceed the design basis envelope for passive SSCs. Extreme external hazards represent one way this can occur. The DBEHL are defined in this step, which can include the ground motion peak acceleration “g” values for seismic events, maximum wind speed for high winds, maximum flood level for external flooding, and the possible damage from the wildfire events. The safety-related SSCs of the MHTGR are required to be capable of performing their reactor safety functions in response to external events within the DBEHL, and there will be no new LBEs introduced by external hazards.

Step 12. Perform Source Terms Modeling and Simulation for LBEs

As discussed in Section 3.12 above, the applicant uses their fission product transport code (e.g., XSTERM) or the NRC's fission product transport code (MELCOR), supported by neutronic, thermal-hydraulic, and other analysis tools as necessary, to perform an analysis of the LBEs identified in Step 10. At this stage, the code has been revised and informed by the additional experiment data collected as a part of Step 8, verified, and validated. For the two DBAs outlined in Step 7, the following source terms are calculated:

Table A- 5 Example source terms for a break in the He pressure boundary [47].

Fission Product Class	**Nuclide**	**Short Term Release (curies)**			**Long-Term Release (curies)**		
		50%	**Mean**	**95%**	**50%**	**Mean**	**95%**
Noble gases	^{133}Xe	3.99E+01	3.99E+01	3.99E+01	4.92E+01	6.44E+01	1.68E+02
	^{85}Kr	2.10E-01	2.10E-01	2.10E-01	3.39E-01	4.50E-01	1.21E+00
	^{88}Kr	2.06E+01	2.06E+01	2.06E+01	1.38E-04	1.82E-04	4.91E-04
I, Br, Te, Se	^{131}I	5.51E-02	1.61E-01	6.10E-01	2.85E+00	6.11E+00	2.24E+01
	^{133}I	1.00E-01	2.90E-01	1.08E+00	1.17E+00	2.64E+00	9.72E+00
	^{132}Te	7.44E-02	2.17E-01	8.08E-01	3.12E+00	6.51E+00	2.41E+01
Cs, Rb	^{137}Cs	1.17E-01	3.43E-01	1.33E+00	1.15E-01	3.31E-01	1.28E+00
	^{134}Cs	1.93E-02	5.54E-02	2.10E-01	1.32E-01	3.82E-01	1.49E+00
Sr, Ba, Eu	^{90}Sr	1.57E-03	4.56E-03	1.72E-02	1.79E-01	4.71E-01	1.76E+00
Ag, Pd	^{110m}Ag	3.86E-02	1.13E-01	4.22E-01	8.61E-01	2.30E+00	8.51E+00
	^{111}Ag	7.96E-01	2.28E+00	8.93E+00	5.48E+01	1.72E+02	6.49E+02
Sb	^{125}Sb	7.27E-04	2.12E-03	8.15E-03	1.78E-03	5.23E-03	2.05E-02
Mo, Ru, Rh, Tc	^{103}Ru	8.05E-04	2.34E-03	8.68E-03	7.19E-01	1.96E+00	7.54E+00
La, Ce groups	^{144}Ce	9.74E-03	2.75E-02	1.05E-01	4.57E-02	1.30E-01	5.01E-01
	^{140}La	7.46E-04	2.16E-03	8.13E-03	2.88E-02	7.65E-02	2.92E-01
Pu, actinides	^{239}Pu	1.90E-07	5.74E-07	2.10E-06	8.97E-07	2.60E-06	1.01E-05

Table A- 6 Example source terms for a water ingress event [47].

Fission Product Class	**Nuclide**	**Short Term Release (curies)**			**Long-Term Release (curies)**		
		50%	**Mean**	**95%**	**50%**	**Mean**	**95%**
Noble gases	^{133}Xe	3.99E01	3.99E+01	3.99E+01	1.08E02	1.54E02	4.42E02
	^{85}Kr	2.10E-01	2.10E-01	2.10E-01	7.31E-01	1.06E00	3.08E+00
	^{88}Kr	2.03E+01	2.03E+01	2.03E+01	3.05E-04	4.32E-04	1.25E-03
I, Br, Te, Se	^{131}I	1.10E+00	1.90E+00	6.56E+00	3.16E+00	6.90E+00	2.48E+01
	^{133}I	2.01E+00	3.36E+00	1.12E+01	1.31E+00	2.85E+00	1.02E+01
	^{132}Te	1.48E+00	2.54E+00	8.52E+00	3.29E+00	7.00E+00	2.54E+01
Cs, Rb	^{137}Cs	2.37E+00	4.06E+00	1.35E+01	2.59E-01	7.06E-01	2.67E+00
	^{134}Cs	3.77E-01	6.36E-01	2.16E+00	2.92E-01	7.81E-01	3.09E+00
Sr, Ba, Eu	^{90}Sr	3.12E-02	5.28E-02	1.72E-01	3.62E-01	9.59E-01	3.58E+00
Ag, Pd	^{110m}Ag	7.60E-01	1.30E+00	4.31E+00	8.89E-01	2.01E+00	7.33E+00
	^{111}Ag	1.57E+01	2.65E+01	8.81E+01	5.59E+01	1.53E+02	5.60E+02
Sb	^{125}Sb	1.51E-02	2.52E-02	8.35E-02	1.95E-03	5.52E-03	2.17E-02

Fission Product Class	Nuclide	Short Term Release (curies)			Long-Term Release (curies)		
		50%	Mean	95%	50%	Mean	95%
Mo, Ru, Rh, Tc	^{103}Ru	1.57E-02	2.66E-02	8.79E-02	6.91E-01	1.69E+00	6.42E+00
La, Ce groups	^{144}Ce	1.98E-01	3.33E-01	1.08E+00	4.82E-02	1.13E-01	4.08E-01
	^{140}La	1.45E-02	2.46E-02	8.36E-02	2.81E-02	6.66E-02	2.44E-01
Pu, actinides	^{239}Pu	3.70E-06	6.23E-06	2.09E-05	9.07E-07	2.13E-06	7.80E-06

In Tables A-5 and A-6, a "short term" release indicates the prompt release during depressurization of fission products that were present in the primary circuit as a result of normal operations, and a "long-term" release indicates a delayed release associated with heatup of the fuel over the course of the accident. In addition to the radionuclide inventories and timing of the release, mechanistic calculations should address the thermal energy associated with, and physical and chemical forms of, those radionuclides; for example, radionuclides may exist as vapors or may be adsorbed on dust.

Using these mechanistic source terms, the applicant performs atmospheric transport calculations to determine transport to the EAB, followed by dose calculations based on that remaining fraction of radionuclides transported there. The results are summarized in Table A-7:

Table A- 7 Example calculated dose comparison with regulatory criteria [48].

Regulatory Criteria (scenario)	Event Scenario	Exposure	Calculated Dose (rem)	Regulatory Criteria (rem)
EAB at 400m (TEDE)	Break in Helium Pressure Boundary	Worst 2 hours	0.02	25
LPZ at 400m (TEDE)	Break in Helium Pressure Boundary	Cloud Passage	1.32	25
EAB at 400m (TEDE)	Water Ingress Event	Worst 2 hours	0.46	25
LPZ at 400m (TEDE)	Water Ingress Event	Cloud Passage	6.43	25
EPA PAG Plume Exposure Related Dose (TEDE)	Break in Helium Pressure Boundary	4 days	0.04	1
EPA PAG Plume Exposure Related Dose (TEDE)	Water Ingress Event	4 days	0.05	1
EPA PAG Plume Exposure Related Dose (Thyroid)	Break in Helium Pressure Boundary	4 days	0.18	5
EPA PAG Plume Exposure Related Dose (Thyroid)	Water Ingress Event	4 days	0.25	5

The applicant finds that doses resulting from all the LBEs do not exceed the frequency-consequence targets illustrated in Figure 1-1. The cumulative risk is also assessed, and it is found that:

- The total frequency of exceeding a EAB dose of 100 mrem is less than 1/plant-year.
- The average individual risk of early fatality within 1 mile of the EAB from all the LBEs is less than 5×10^{-7}/plant-year.

- The average individual risk of latent cancer fatality within 10 miles of the EAB from all the LBEs is less than 2×10^{-6}/plant-year.

The regulatory limits identified in Section 3.12.3 have therefore been met. Finally, the applicant identifies all events classified as “Risk-significant LBEs,” i.e. those within two orders of magnitude of the frequency-consequence targets in Figure 1-1, as areas for potential future improvement.

Step 13. Review LBEs List for Adequacy of Regulatory Acceptance

The LBE evaluation performed in the previous steps provides feedback on whether additional improvements on design and operation should be considered. Such improvements could be motivated by a desire to increase margins against the F-C target criteria, reduce uncertainties in the LBE frequencies or consequences, limit the need for restrictions on siting or emergency planning, or enhance the performance against DID criteria. The applicant concludes that no improvements are needed for the MHTGR, and the final list of LBEs and safety-related SSCs is established.

Step 14. Document Completion of Source Term Development

Having found, at the completion of the analyses, that all regulatory requirements have been met, the applicant documents these and submits the documentation to the NRC for approval. As the PRA is updated to reflect any changes that occur as part of a continuing design process or modifications to the plant during its operating life, the list of LBEs is revisited and steps 6–14 are repeated as/if necessary based on the updated set of LBEs and PRA results.

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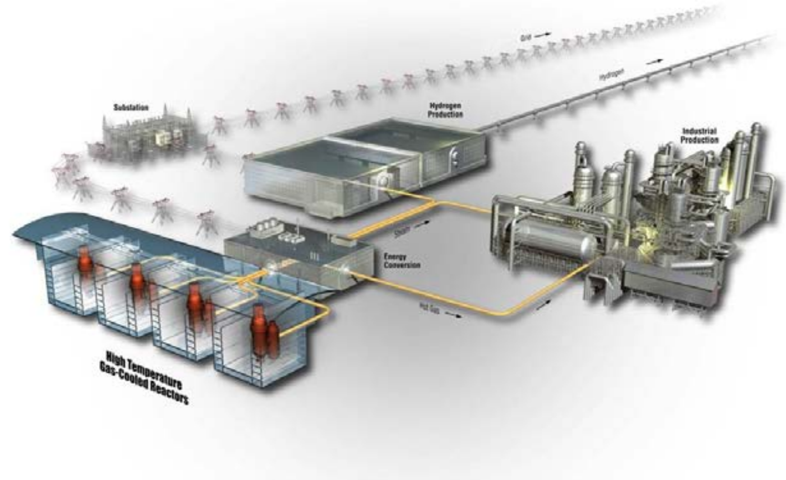
Establishing Jurisdictional Boundaries at Collocated Advanced-Reactor Facilities

August 2020

Changing the World's Energy Future

Wayne L. Moe, Idaho National Laboratory
Thomas E. Hicks



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Wayne L. Moe, Idaho National Laboratory
Thomas E. Hicks

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Idaho National Laboratory
Advanced Reactor Technologies
Idaho Falls, Idaho 83415

http://www.art.inl.gov

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Establishing Jurisdictional Boundaries at Collocated Advanced-Reactor Facilities

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Technical Reviewer: (Confirmation of mathematical accuracy, and correctness of data and appropriateness of assumptions.)

Jason Christensen	08/18/2020
Jason A. Christensen, Licensing Technical Reviewer	Date

Approved by:

Travis Mitchell	8/18/2020
Travis R. Mitchell, Program Manager Advanced Reactor Technologies	Date
Gerhard Strydom	08/19/2020
Gerhard Strydom Co-National Technical Director Advanced Reactor Technologies	Date
Michelle Sharp	8/18/2020
Michelle T. Sharp INL Quality Assurance	Date

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SUMMARY

This paper examines how jurisdictional boundaries might be established at an advanced nuclear reactor facility that is collocated with and physically connected to a non-nuclear industrial facility. This regulatory analysis was done to inform future applicants about opportunities to adapt the nuclear power regulatory framework (administered by the U.S. Nuclear Regulatory Commission (NRC)) to more clearly address how highly regulated nuclear power reactors might share energy transfer systems with non-NRC regulated energy users. End user energy systems are assumed to be conventional, non-nuclear equipment, and stationed in a location that otherwise would not normally be subject to NRC licensing and oversight authority. A 10 CFR Part 52 licensing approach is employed for this analysis.

The review concluded that a regulatory bases already exists for establishing jurisdictional boundaries between a nuclear plant and non-nuclear industrial facilities collocated at the same site. Establishing jurisdictional boundaries between these physically connected facilities would need to address the following considerations:

- NRC would retain full oversight authority over systems, structures, and components (SSC) needing protection under physical-security regulations. These security elements would be part of the nuclear facility.
- All SSCs that perform nuclear safety-related or risk-significant functions would be included within the nuclear facility boundary and under NRC jurisdiction.
- Energy-conversion system(s) located within the nuclear protected-area boundary, are integral to the nuclear facility, and/or are operated by the nuclear facility control room, should be considered part of the nuclear facility. Energy-conversion system(s) located outside the protected-area boundary and separated from the nuclear facility by a transfer system with appropriate interface criteria, could be excluded from nuclear facility scope. Interface criteria must ensure the nuclear facility is not dependent upon or adversely affected by industrial facility events.
- Nuclear safety analysis would be required of all nuclear and industrial systems with respect to potential missiles, security issues, flooding issues, or any other impacts that may influence SSCs that perform a nuclear safety function.
- The regulatory boundary between the nuclear and industrial facilities can be defined by describing the boundary in the nuclear-facility system design, transfer-system(s) design, and interface descriptions with appropriate interface requirements, and pertinent down-stream conceptual-design information. Interface requirements must address industrial facility systems transients and failures. Requirements must assure that no portion of the industrial energy-transfer system performs or adversely affects a nuclear safety function. Appropriate monitoring and detection systems are to be employed. Radioactive material releases from energy transfer system(s) must meet applicable limits.

- To further increase flexibility and streamline the licensing process, another internal nuclear facility boundary could be established for applicants that utilize a standard nuclear plant design. This boundary would be based on information contained in a design certification application (DCA) with remaining site-specific information addressed in a combined license application (COLA). System-specific industrial facility descriptions would not be required in the DCA or COLA, but the COLA would demonstrate how all applicable interface requirements are met.
- Interface requirements would demonstrate a robust ability to maintain safe nuclear operation. Site-related requirements and assumptions associated with the standard design would be shown as met along with all criteria pertinent standard design safety. These requirements are also focused on preserving SSC nuclear safety functions.
- For COLAs that do not reference a design certification, applicants would need to submit design information for the entire nuclear facility. This type of COLA would fully describe nuclear/industrial facility boundary interface requirements and demonstrate how those criteria are satisfied.

CONTENTS

FIGURES

ACRONYMS

ART	Advanced-Reactor Technologies
BDBE	beyond design basis event
BOP	balance of plant
CFR	Code of Federal Regulations
COL	combined license
COLA	combined license application
CP	construction permit
CWS	circulating water system
DBA	design basis accident
DBE	design basis event
DC	design certification
DCA	design certification application
DCD	design certification document
DID	defense in depth
DOE	U.S Department of Energy
ECA	energy conversion area
EPRI	Electric Power Research Institute
F-C	frequency-consequence
FR	Federal Register
FSAR	final safety analysis report
GDC	general design criteria
HTGR	high temperature gas-cooled reactor
ITAAC	inspection, test, analysis, and acceptance criteria
LBE	licensing basis event
LWR	light water reactor
MHTGR	Modular High Temperature Gas-cooled Reactor (GA)
MSSS	main steam supply system
MW(t)	megawatt (thermal)
NEI	Nuclear Energy Institute
NGNP	Next Generation Nuclear Plant
NI	nuclear island
NIA	nuclear innovation alliance
NRC	U.S. Nuclear Regulatory Commission

NSSS	nuclear steam supply system
OL	operating license
PRA	probabilistic risk assessment
PSAR	preliminary safety analysis report
psia	pounds per square inch (absolute)
PWR	pressurized water reactor
R-COLA	reference combined license application
RG	regulatory guide
SAR	safety analysis report
S-COLA	subsequent combined license application
SDA	standard design approval
SER	safety evaluation report
SMR	small modular reactor
SR	safety-related
SRM	standard reactor module
SRP	standard review plan
SSAR	standard safety analysis report
SSC	structures, systems and components
SSE	safe shut-down earthquake

Establishing Jurisdictional Boundaries at Collocated Advanced-Reactor Facilities

1. INTRODUCTION

The Nuclear Regulatory Commission (NRC) has published various strategies enabling a vision that increases the effectiveness and efficiency of NRC advanced-reactor design license application reviews. These activities affect many different attributes of the existing nuclear regulatory framework and necessitate certain changes to attain envisioned objectives. Some of this work is foundational in nature and may be unsupported by a tested regulatory precedent. Establishing a basis for defining and separating jurisdictional authority between conjoined nuclear and non-nuclear (industrial) facilities at a shared (collocated) site is one potential element in modernizing this framework.

Advanced (non-light water reactor [LWR]) nuclear technologies can be used to supply energy to a wide range of commercial use applications. Applications include supplying electricity to a distribution grid, providing electricity directly to facilities not on a grid, steam cogeneration, and high temperature process heat for applications like hydrogen production, hydrocarbon recovery from oil sands/oil shale, or district heating. The varying forms of potential energy demand that could be served by advanced-reactors may require new energy-conversion systems and unique configurations that employ multiple nuclear modules to meet customer requirements for full-power and plant availability.

One sub-class of advanced-reactor design worth noting are the unique deployment opportunities associated with "microreactors." A microreactor is an emerging nuclear-energy supply technology that targets specialized market niches like those in remote locations. Microreactors are very small nuclear reactors with thermal power outputs 100 to 1,000 times smaller than the large LWRs typical of the existing commercial fleet. Such a size could lead to unprecedented levels of unit mobility and transport, employ power-conversion systems integrated into the reactor module itself, and might facilitate a "plug and play" option for quick module installation/change-out at sites situated very close to the end energy user. Microreactors could be deployed with footprints as small as 1,000 ft^2, thereby making them potentially available to entirely new markets currently challenged to access to clean, reliable, and affordable energy. Likely customers include arctic or island communities, remote mining operations, forward military bases, and other installations needing reliable energy to support critical infrastructure.

A key regulatory issue for many such deployments arises when attempting to determine where to draw regulatory boundaries between nuclear facility systems under the jurisdiction of the NRC (i.e., within the scope of a 10 Code of Federal Regulations [CFR] Part 52 design certification [DC], a future Part 53 license, and a combined operating license [COL]), and systems that otherwise normally fall outside of the NRC regulatory scope (i.e., the industrial facility).

This paper examines the current regulatory basis underlying establishment of jurisdictional boundaries at advanced-reactor installations that are proximate to and share systems with non-NRC regulated facilities (i.e., the collocated facility). This review is predicated on having a clear understanding of plant scope as addressed in an advanced-reactor facility Part 52 DC application as well as other parts of plant scope addressed as components of a site-specific combined license application (COLA). Relatedly, it is also important to understand nuclear plant safety issues associated with the collocated facility but not necessarily addressed in typical NRC licensing documentation.

Figure 1, adapted from the General Atomics' Next Generation Nuclear Plant Project (NGNP) Conceptual Design Report, illustrates a typical single-module high-temperature gas reactor (HTGR) plant arrangement. [Ref 1] This arrangement presumes an onsite[a] turbine generator for electric-power

[a] The use of the terms "onsite" and "offsite" refer to inside or outside of the HTGR protected area, which coincides with inside or outside of the nuclear facility boundary.

production and process-heat transfer lines running to an offsite location. Examining configurations such as this provides understanding about the need for workable jurisdictional boundaries between onsite and offsite systems as well within the advanced-reactor nuclear plant configuration itself.

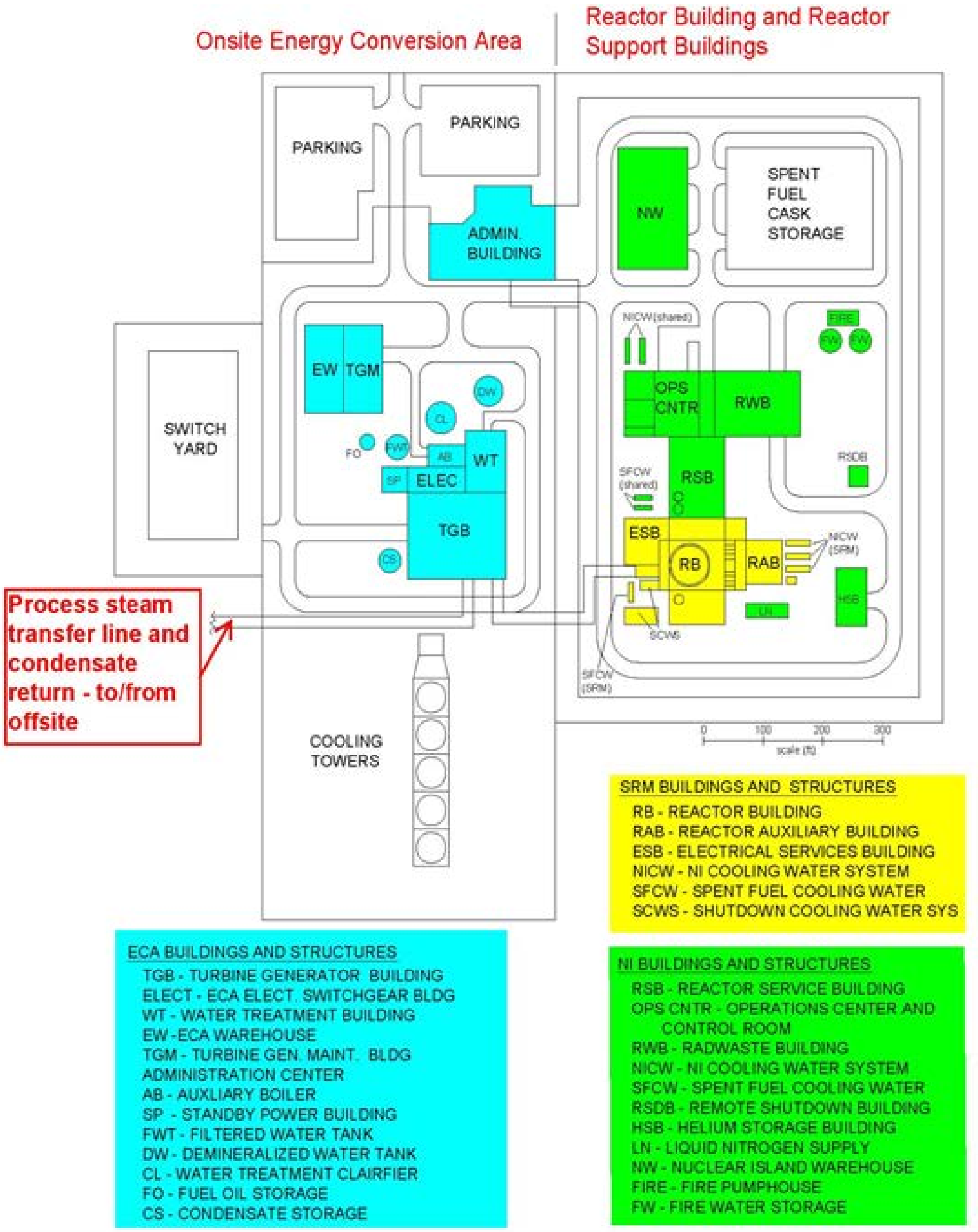

Figure 1. Typical HTGR plant general arrangement.[b]

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[b] "SRM" means "standard reactor module." The SRM is the part of the facility that would be certified under a design certification process. The "NI" is the nuclear island that includes many of the nuclear plant support systems. The "ECA" is the energy conversion area that includes the onsite energy conversion system. Other possible configurations could include multiple modules.

1.1 Approach

Given future applicants will likely maximize the number of energy conversion system configurations that can be deployed at a collocated site, license applications are expected to be structured to encompass as many configuration options as possible within a single DC. Using this assumption, two sets of boundaries can be postulated as an appropriate basis for establishing collocated facility jurisdictions.

One boundary can be derived by understanding the nominal nuclear plant safety scope (an area commonly recognized as appropriate for NRC oversight to assure public safety). The balance of a collocated site (i.e., systems that do not offer a significant potential to adversely affect nuclear safety) would be eligible for consideration as a (non-nuclear) industrial facility and excluded from NRC licensing and oversight authority.

A second boundary could be created by further subdividing the nuclear-facility portion of the plant to provide for standardized systems addressed in the certified portion of a DC application (DCA); the balance of nuclear plant systems not addressed in a DCA would then be addressed in a site-specific COLA that references said DC. This boundary allows use of a DC for a standard part of the design, thereby requiring only one regulatory review for a portion of a plant that may be later deployed at numerous different sites. This means NRC regulatory issues for the standard design would be evaluated and resolved only once, thus streamlining reviews of future site COLAs. It would also facilitate different energy-conversion system that might be used. Furthermore, a standard reactor module DC could be structured to address multimodule configurations should those be required.

To enable jurisdictional separation, very clear understandings are necessary between the applicant and NRC staff regarding boundary interfaces. These interfaces can be considered “points of compliance” and requirements and criteria that operate at those compliance locations are key to successful boundary operation. In fact, such descriptions are essential in establishing the scope and definitions used in both the DC and COL.

1.2 Review Objectives

The objective of this review is to inform stakeholders on key attributes of a proposed approach that can be considered for NRC jurisdictional boundaries at collocated nuclear/industrial sites. It does this by:

1. Reviewing existing regulatory requirements, guidance, and precedents related to the topic
2. Identifying regulatory framework opportunities that allow for the definition of jurisdictional boundaries between an advanced-reactor nuclear facility and a collocated industrial facility
3. Identifying facility design requirements and interface requirements that must be defined to ensure safe operations for nuclear plant interconnection with an industrial facility. The term “interface requirements” is used in most regulatory guides to highlight dependencies among the structures, systems, and components (SSCs) and their associated regulatory requirements
4. Specifying minimum sets of nuclear facility system and interface requirement descriptions that should be established to address the scope of the certified portion of a 10 CFR 52 DC and those that may be appropriately described in a site-specific Part 52 COL.

From this information, a position on the topic can be developed by stakeholders for subsequent review, concurrence, and regulatory/policy action by NRC staff.

1.3 Scope

This paper discusses two sets of likely boundaries typical of future advanced-reactor applications. These are:

- A physical boundary(s) will exist between a nuclear facility (and associated systems) under NRC regulatory jurisdiction, and an industrial facility whose systems would otherwise reside outside of NRC regulatory jurisdiction
- A boundary internal to the nuclear facility can also exist that may be used address a minimum set of plant systems that should be addressed in a 10 CFR 52 DC application; systems that fall outside of the DC scope would fall within the scope of a COLA. Both sides of this boundary would exist within NRC jurisdiction.[c]

Additional observations are provided concerning systems-level interface issues relevant to SSCs that transect these boundaries. Discussions also incorporate risk-informed, performance-based considerations that are now available for use in NRC licensing actions.

It should be noted, however, that because analysis of potential impacts from onsite hazards and nearby industrial hazards (such as chemical toxicity or explosion) is required under existing regulations as part of a comprehensive nuclear facility safety analysis, no changes to these requirements is considered or recommended in the scope of this paper.

1.4 Relationship to Other Advanced-Reactor Topics/Papers

NRC SECY-11-0079, "License Structure for Multi-Module Facilities Related to Small Modular Nuclear Power Reactors," dated June 12, 2011. [Ref [2]]

This report describes NRC positions regarding whether a multimodule reactor plant can be licensed with a single NRC review, hearing, and safety-evaluation report. The paper explains the structure and the duration of a license.

NEI White Paper, "Micro-Reactor Regulatory Issues," dated November 13, 2019 [Ref [3]]

This report outlines proposed changes to current policies for the licensing and regulation of small microreactors. The paper also identifies a need to address several policy and technical issues. It discusses the notion that microreactor designs may be able to demonstrate potential consequences of accidents, even for the worst-case scenarios, would not lead to a significant adverse impact on the health or safety of the public. This may justify alternative approaches to meeting regulations and protecting public health and safety. Included in the report are actions likely necessary to help develop information needed to inform the NRC's consideration of alternative approaches.

NRC, "Staff Requirements, SECY-18-0076, "Options and Recommendations for Physical Security for Advanced-reactors," dated November 19, 2018 [Ref [4]]

This report describes a rulemaking to establish physical security requirements appropriate for advanced-reactors and the use of a performance-based, technology-neutral, and consequence-oriented approach for developing a new physical-security framework. The use of the term "advanced reactor" in the draft regulatory basis appears to be sufficiently broad to encompass microreactors.

Nuclear Innovative Alliance (NIA) report, "Establishing Interface Requirements for 'Major Portions' Standard Design Approvals," dated September 2019 [Ref [5]]

This report provides guidance to advanced-reactors suppliers using the standard design approval (SDA) process regarding the establishment of interface requirements between portions of a design that have been included in the application for an SDA and those that will be submitted at a later date under 10 CFR 52 or 10 CFR 50. Because the SDA, as part of a staged licensing approach, is expected to be used by some suppliers, the guidance contained in this report should facilitate the design, licensing, and deployment of advanced reactors. The process can be applied to any reactor type. The rule language of

[c] This paper does not identify specific boundaries that might be used in a standard design approval (SDA). However, the same concepts that apply to a design certification could be applied to an SDA.

10 CFR 52.137 indicates that an application for an SDA must contain a final safety-analysis report (FSAR) 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, or major portion thereof* . . .

The report states that interface requirements can be thought of as boundary conditions for the portion of the design for which an SDA is being sought. Key safety-significant design attributes and performance characteristics must be addressed in the interface requirements with details sufficient to provide the NRC staff with an adequate basis for a safety determination. An application referencing an SDA will need to demonstrate that the interface requirements are satisfied.

NIA report, "Clarifying 'Major Portions' of a Reactor Design in Support of a Standard Design Approval," dated in April 2017 [Ref [6]]

This report explains, in part, the term "major portion." The NIA document provides examples of a "major portion" as:

> *For example, an SDA could be sought for the structures, systems, and components (SSCs) associated with the "nuclear island," and these SSCs might be completed to a level of detail approximating that for a [design certification application]. Alternatively, if the motivation for an SDA is early staff review of portions of the plant with more programmatic risk (e.g., because of novel design for fuel, security, seismic isolation, etc.), a different set of SSCs might be pursued, with level of detail varying as a function, for example, of the extent of interfacing systems or boundary conditions*.

The NIA report also indicates that NRC approval of a major portion should explicitly list all assumptions regarding its connection to other parts of the design to facilitate NRC's review and the future use of the SDA in subsequent licensing processes. To that end, these interface requirements must also be satisfied by the rest of the design, whether submitted as an application for an additional SDA, a COL, a construction permit (CP), or an operating license (OL). This report provides guidance as discussed in Section 4, "Interfacing Systems and Boundary Conditions," of the April 2017 document regarding the establishment of interface requirements in an application for an SDA of a major portion of an advanced-reactor design. Establishment of interfacing systems and boundary conditions is a critical consideration in defining "major portions." When an SDA is approved by the NRC staff, it will necessarily be associated with various conditions of assumed interfacing boundary conditions, which in turn will have to be satisfactorily demonstrated if the SDA is incorporated into a subsequent CP application, DCA or COLA.

2. REGULATORY FOUNDATION

2.1 U.S. Regulatory Foundation for the Nuclear-Industrial Facility and Design Certification Boundaries

2.1.1 NRC Requirements

In 1989, the NRC published the final rule, 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors" [Ref [7]]. This rule sets forth review procedures and requirements for applications for new licenses and certifications. The rule was modified in 2007 to clarify applicability of various requirements to each licensing process by making necessary conforming amendments throughout NRC's regulations that enhance regulatory effectiveness and efficiency when implementing its processes. The boundary evaluations discussed in this paper are presented in the context of a Part 52 licensing process.

In determining how and where to define the proper boundary between the nuclear and industrial facilities and how to define a boundary between a DCA and a COLA, it is important to identify applicable NRC regulations and guidance that specify expectations for the two applications.

10 CFR 52, Subpart B, "Standard Design Certifications"

Subpart B of 10 CFR 52 defines the regulatory requirements for DCAs. Section 52.47, "Contents of Applications; Technical Information," defines the requirements for technical content of a DCA[d]. Because the contents of DCAs, including inspection, test, analysis, and acceptance criteria (ITAAC)[e], are certified by rulemaking, it is not practical to include optional configurations and equipment as part of the certified portion of the plant. The regulations make provisions for design certifications to include optional configurations (outside of the certified portion of the plant) by allowing these applications to include "conceptual-design" information[f]. Paragraph 52.47(a) states general requirements for the DC FSAR:

> *(a) The application must contain a final safety analysis report (FSAR) 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, and must include the following information:*
>
> *(1) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters*
>
> *(2) A description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. It is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations. Such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed* ***insofar as they are pertinent*** [emphasis added].

Paragraph 52.47(a)(24) states that the design certification may include:

> *A representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the NRC in its review of the FSAR and to permit assessment of the adequacy of the interface requirements in paragraph (a)(25) of this section;*

Paragraph 52.47(a)(25) requires that the DC application contain appropriate interface requirements, and states:

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[d] A standard design certification from the NRC is submitted separately from an application for a COL filed under Subpart C of Part 52 for a nuclear power facility. An applicant for a COL may reference a standard design certification.

[e] ITAAC provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Act, and the Commission's rules and regulations. All ITAAC in the design certification must be verified as complete before fuel load is authorized by the NRC.

[f] NRC's use of the term conceptual has a different context than citing the status of design development as conceptual. In the NRC's context, the energy-system configurations and performance would be conceptualized to support defining interfaces, transients, and accident conditions for which the NRC Office of Nuclear Material Safety and Safeguards is certified. The conceptualized energy conversion systems would not be included in the certification.

> *The interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the FSAR.*

Paragraph (c) of 52.47, defines content requirements for DCAs having certain characteristics. Paragraph (c)(3) addresses modular reactors[g] and requires the following:

> *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.*

In a statement of consideration for the final (1989) rule, NRC stated that Part 52 ". . . provides for certification of advanced designs and permits certification of designs of less than full scope only in highly restricted circumstances." Clearly, NRC intended that DC applications be a complete representation of the plant. The final rule's provisions on scope (see §52.47), reflect a policy that certain designs, especially designs that are evolutions of light-water designs now in operation, should not be certified unless they include all of a plant which can affect safe operation of the plant except its site-specific elements. The NRC provided examples of designs that are evolutions of currently operating light-water designs, including General Electric's Advanced Boiling Water Reactor, Westinghouse's SP/90, and Combustion Engineering's System 80+. NRC further stated that full-scope may also be required of certain advanced designs—namely, the passive light-water designs such as General Electric's Simplified Boiling Water Reactor and Westinghouse's AP600. NRC stated that considerations of safety, not market forces, constitute the basis for the final rule's requirement that these designs be full-scope designs. According to the staff, ". . . long experience with operating light water designs more than adequately demonstrates the adverse safety impact which portions of the balance of plant can have on the nuclear island. Given this experience, certification of these designs must be based on a consideration of the whole plant, or else the certifications of those designs will lack that degree of finality which should be the mark of the certifications" (see 54 FR 15374).

However, the Commission stopped short of stating that no design of incomplete scope could ever be certified.

> *There is no reason to conclude that there could never be a design which protects the nuclear island against adverse effects caused by events in the balance of plant. The final rule therefore provides the opportunity for certification of designs of less than complete scope if they belong to the class of advanced designs. See § 52.47(b) [1987 rule]. Examples of designs in this class include the passive light-water designs mentioned above and non-light-water designs such as General Electric's PRISM, Rockwell's SAFR, and General Atomic's MHTGR. But here too the rule sets a high standard: Certification of an advanced design of incomplete scope will be given only after a showing, using a full-scale prototype, that the balance of plant, cannot significantly affect the safe operation of the plant.*[h]

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[g] Modular designs are defined in § 52.1. Modular plant designs are not just portions of a single nuclear plant, rather they are separate nuclear power reactors with some shared or common systems.

[h] Further discussion regarding prototype requirements for advanced reactors is provided in SOC for the final 2007 Part 52 ru1emaking, 72 FR 49370.

While analyses may be relied upon by the staff to demonstrate the acceptability of a particular safety feature which evolved from previous experience or to justify the acceptability of a scale model test, it is very unlikely that an advanced design would be certified solely on the basis of analyses. Prototype testing is likely to be required for certification of advanced non-light water designs because these revolutionary designs use innovative means to accomplish their safety functions, such as passive decay heat removal and reactivity control, which have not been licensed and operated in the United States.[i]

Section 52.47(c)(2) [2007 rule] [Ref [8]] requires applications for "advanced" nuclear power plants provide an essentially complete scope of design and meet the design-qualification testing requirements in 10 CFR 50.43(e). Advanced designs differ significantly from evolutionary LWR designs or incorporate, to a greater extent than evolutionary designs do, simplified, inherent, passive, or other innovative means to accomplish their safety functions. Examples of advanced nuclear power plant designs listed in the rule include General Atomics' Modular High Temperature Gas-Cooled Reactor (MHTGR), the Simplified Boiling Water Reactor, and Westinghouse's AP600.

10 CFR 52, Subpart C, Combined Licenses

Under 10 CFR 52, Subpart C, Combined Licenses, the NRC specifies its requirements for technical information in the COLA FSAR. Paragraph 52.79, "Contents of applications; technical information in final safety analysis report," states:

(a) The application must contain a final safety analysis report 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 of the facility as a whole. The final safety analysis report shall include the following information, at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before issuance of a combined license:

(2) A description and analysis of the structures, systems, and components of the facility with emphasis upon performance requirements, the bases, with technical justification therefore, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. It is expected that reactors will reflect through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The descriptions shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations. Items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:

(i) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials

i See 54 FR 15375

(ii) The extent to which generally accepted engineering standards are applied to the design of the reactor

(iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials

(iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated.

10 CFR 73, "Physical Protection of Plants and Materials"

10 CFR 73 defines, in part, requirements for the establishment and maintenance of a physical protection system which will have capabilities for the protection of special nuclear material at fixed sites in which special nuclear material is used. Paragraph 73.1 requires, in part, that each licensee establish and maintain a physical protection system which will have capabilities for the protection of special nuclear material. The physical protection system shall be designed to protect against the design basis threats of theft or diversion of special nuclear material and radiological sabotage as stated in § 73.1(a).

10 CFR73.46 requires, in part, that vital equipment must be located only within a vital area, and strategic special nuclear material must be stored or processed only in a material access area. Both vital areas and material access areas must be located within a protected area so that access to vital equipment and to strategic special nuclear material requires passage through at least three physical barriers. Vital area means any area which contains vital equipment. Vital equipment means any equipment, system, device, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems which would be required to function to protect public health and safety following such failure, destruction, or release are also considered to be vital.

10 CFR 73.55 defines requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage. The licensee is required to:

> . . . *establish and maintain an onsite physical protection system and security organization. which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. The physical protection system shall be designed to protect against the design basis threat of radiological sabotage as stated in § 73.1 (a).*

To achieve this general performance objective, the onsite physical-protection system and security organization must include capabilities to meet the specific requirements, such as physical barriers, access restrictions, detection aids, and communications requirements.

These NRC security regulations help define the boundary of the nuclear facility in that any equipment within the security boundary would be governed by these regulations and would thus be required within the nuclear facility.

2.1.2 NRC Policy Statements

SECY-88-202, "Standardization of Advanced-reactor Designs" [Ref [9]]

In SECY-88-202[j], the staff presented a set of criteria that was developed for use in the review of U.S. Department of Energy's (DOE) plans for standardization of three advanced-reactor concepts. Two issues addressed in the paper were: (1) the scope and level of detail of design to be standardized, and (2) plant options (number of reactor modules) to be standardized. The staff's proposed criteria for resolving these issues were developed to be consistent with the intent of the Commission's policies on standardization and advanced reactors. The criteria were consistent with the staff's proposed rulemaking on standard design certifications.

In the SECY, the staff listed four reactor designer concerns for limiting the certified portion of the designs:

1. They stated that all plant safety systems will be contained within the certified envelope (with no system interactions between safety and non-safety portions of the plant capable of affecting performance of the plant's safety functions). This, it was proposed, eliminates the need for NRC to approve anything other than interface requirements for the remainder of the design.
2. They were concerned that if the non-safety portion of the design were certified, NRC would be involved in design and construction verification to a greater extent than necessary.
3. They noted that not certifying the entire plant would allow greater flexibility to incorporate design improvements or improvements in technology without having to go through the process of amending the DC.
4. They stated that to allow utilities the flexibility of procuring the balance of plant in a competitive fashion with design differences to suit their needs, a DC of the entire plant is not desirable.

The staff also notes in that paper:

> *. . . the major contributors to non-standardized plants today are the differences from plant to plant external to the Nuclear Steam Supply System (NSSS). Problems external to the NSSS have been the initiator of many plant shutdowns, the focus of many Generic Safety Issues and have impacted plant safety. However, transients initiated in the non-safety related portions of the advanced designs should have less likelihood of leading to severe accidents. This is because the passive reactor shutdown and decay heat removal systems have the potential for high reliability since they are less vulnerable to failure modes involving active equipment, electric power, or human error. Therefore, even though failures or transients in the balance of plant could challenge safety systems, the overall risk from these challenges should be less than for LWRS. However, since the design and operation of the remainder of the plant is key to ensuring that the interface criteria with safety systems are met, that assumptions regarding accident initiators are maintained, and that operating experience gained on one plant is readily transferable to other plants, submittal of the entire plant for Design Certification is still preferred. This would eliminate the possibility of each plant varying substantially from the others, would make the preparation of a [probabilistic risk assessment (PRA)] and safety analysis more straight-forward and would minimize the time and staff resources required to review individual license applications to assess compliance with interface*

[j] Note that SECY-86-368, "NRC Activities Related to the Commission's Policy on the Regulation of Advanced Nuclear Power Plants," was a predecessor document to SECY-88-202

criteria. In addition, approval of a complete plant design at the Design Certification stage will afford a greater opportunity for wide public participation, as well as reducing the time and resources expended in repeatedly litigating the acceptability of a design at individual hearings.

In short, the benefits to the Commission from standardization are maximized when the entire plant is certified. For these reasons, the staff preference is to standardize and certify the entire plant. However, from the standpoint of performing a technical review, the staff could consider Design Certification of less than the complete plant provided that the certified portion of the plant contains all of the safety systems and the following criteria are met for the non-certified portion:

1. *The interface requirements established for the non-certified portions of the design are sufficiently detailed to allow completion of a final safety analysis and a PRA for the plant.*
2. *Compliance with the interface requirements established for the noncertified portions of the design is verifiable through inspection, testing (separately or in the plant), previous experience or analysis. Compliance with interface requirements dealing with reliability of components or systems shall be verifiable through previous experience or testing.*
3. *A representative design for the non-certified portions of the plant is submitted along with the application for Design Certification as an illustration of how the interface requirements can be met and as an aid in the review of the PRA and safety analysis.*

The above criteria would require certification of all the safety related portions of the plant and sufficient information on the other portions to determine overall safety. The staff would also require that the level of design detail submitted for the certified portion be final design information, equivalent to that provided in order to obtain an FDA. These criteria would ensure that the plant will be built and operated consistent with its safety analysis and PRA. Since the advanced designs are proposing balance of plant systems that are not safety related, the design flexibility desired by the designers would be retained for a large portion of the plant. The acceptability of the three DOE sponsored advanced-reactor concepts with regard to scope and level of detail will be addressed in the respective SERS.

A review of the safety evaluation reports (SER) referenced in SECY-88-0202 did not identify any relevant discussion regarding the topic of this paper.

SECY-10-0034, "Potential Policy, Licensing, And Key Technical Issues for Small Modular Nuclear Reactor Designs," [Ref [10]]

In this report, the staff identified several potential policy and licensing issues that may require resolution during review of design and license applications for some designs. In general, these issues result from key differences between the new designs and current-generation LWRs (such as size, moderator, coolant, fuel design, and projected operational parameters), but also from industry-proposed review approaches and modifications to current policies and practices.

One of the issues discussed, Item 4.4, "Industrial Facilities Using Nuclear-Generated Process Heat," identified potential policy and licensing issues for those facilities used to provide process heat for industrial applications. In this paper the staff stated:

The close coupling of the nuclear and process facilities raises concerns involving interface requirements and regulatory jurisdiction issues. Effects of the reactor on the commercial product of the industrial facility during normal operation must also be considered. For example, tritium could migrate to a hydrogen production facility and become a byproduct component of the hydrogen product.

Resolution of these issues will require interfacing with other government agencies and may require Commission input to determine whether the design and ultimate use of the product is acceptable.

This issue is applicable to license applications for new, first-of-a-kind SMR designs, including the NGNP. However, the staff believes that resolution for this issue need not occur until after a license application is submitted because it concerns site-specific issues associated with the staff's review of an operating license. Once a license application is received, the NRC staff will review how the nuclear facility is connected to the industrial facility, consider the interrelationship between the staffs of both facilities, consider white papers or topical reports concerning this issue that it receives from DOE and potential SMR applicants, discuss design-specific proposals to address this matter, and review similar activities with nuclear and non-nuclear facilities. Should it be necessary, the staff will propose changes to existing regulatory guidance or new guidance concerning the effect of the industrial facility on the nuclear facility in a timeframe consistent with the licensing schedule.

SECY-18-0076, "Options and Recommendations for Physical Security for Advanced Reactors" [Ref [11]]

This paper provides options and a recommendation to the Commission on possible changes to regulations and guidance related to physical security for advanced-reactors, including light-water small modular reactors (SMRs) and non-LWRs. The staff's recommendation is to pursue a limited-scope rulemaking.

The current physical security framework for large LWRs is designed to protect plant features needed to provide fundamental safety functions, such as cooling of the reactor core. The loss of plant features providing these safety functions could lead to damage to a reactor core or spent nuclear fuel, with subsequent release of radioactive materials. The designs and behavior of advanced reactors are expected to be significantly different from large LWRs, however. Advanced-reactor designs are expected to include attributes that result in smaller and slower releases of fission products following a loss of safety function. Accordingly, these designs may warrant different physical security requirements commensurate with risks posed by the technology.

In the paper, the staff recommends a rulemaking to further assess and, if appropriate, revise a limited set of NRC regulations and guidance to provide an alternative to current physical-security requirements for license applicants for advanced reactors. The limited-scope rulemaking effort would evaluate possible performance criteria and alternative security requirements for advanced reactors that have incorporated the reactor attributes defined in the NRC's Policy Statement on the Regulation of Advanced Reactors, specifically designs that incorporate "enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions." [Ref [12]] The alternative physical security requirements and related guidance would support efforts to better address security concerns within the design process, and thereby reduce reliance on armed responders.

The paper identifies four options related to addressing physical security requirements for advanced reactors. Option 3, a limited scope rulemaking, was adopted, and a draft rulemaking was issued in 2019.

The limited-scope rulemaking is intended to provide a clear, alternate, optional set of physical-security requirements in two key areas for advanced reactors and to reduce the need for exemptions to current physical security requirements for applicants that request permits and licenses. Specifically, it would provide a voluntary, performance-based alternative to the prescriptive requirements in 10 CFR 73.55(k)(5)(ii) related to the required minimum number of armed responders and 10 CFR 73.55(i)(4)(iii) related to onsite secondary alarm stations for those advanced reactors that could demonstrate the ability to meet the performance criteria. This limited-scope rulemaking would provide additional benefits for advanced-reactor applicants by establishing greater regulatory stability, predictability, and clarity in the licensing process.

The rulemaking is limited to physical security requirements related to the protection of advanced reactors against radiological sabotage and does not address threats related to theft or diversion. The central theme of the newly proposed rule is to allow flexibility in preventing and mitigating design-basis threats provided that offsite doses are shown to be below the reference values defined in 10 CFR 50.34 and 52.79.

2.1.3 NRC Guidance

Regulatory Guide (RG) 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," [Ref [13]]

This regulatory guide provides information about using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and the content of applications for non-LWRs including, but not limited to, molten-salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors at different thermal capacities. The RG is primarily meant to serve non-LWR applicants applying for permits, licenses, certifications, and approvals under 10 CFR 50, and 10 CFR 52.

Selection of appropriate licensing basis event (LBE), classification and special treatment of SSCs, and assessment of defense in depth (DID) are fundamental to the safe design of non-LWRs. These also support identifying the appropriate scope and depth of information that non-LWR designers and applicants should provide in applications for licenses, certifications, and approvals. The RG endorses Nuclear Energy Institute's (NEI's) 18-04, "Modernization of Technical Requirements for Licensing of Advanced Reactors, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," as one acceptable method for non-LWR designers to carry out assessment activities and prepare applications. The methodology in NEI 18-04 provides a process by which the content of applications will build understanding of system designs and their relationship to safety evaluations for a variety of non-LWR designs. The system design and safety evaluations may also demonstrate compliance with, or justify exemptions from, specific NRC regulations. Although the technology-inclusive methodology provides a common approach to selecting LBEs, classifying SSCs, and assessing DID across a spectrum of designs, the applicability of specific technical requirements in NRC regulations or the need to define additional technical requirements arising from a safety evaluation is made on a case-by-case basis for each non-LWR design.

NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition [Ref [14]]

NUREG-0800 provides guidance to the NRC staff in performing safety reviews of LWRs for various types of license applications, including DC and COLAs under 10 CFR 52. Implementation of the criteria and guidelines contained in the SRP by staff members in their review of applications provides assurance that a given design will comply with NRC regulations and provide adequate protection of public health and safety.

As described in NUREG-0800, designs of SSCs that are to be addressed in a Part 52 DC or COLA (to the extent the SSC is applicable to the specific design being reviewed) include:

- Reactor
- Reactor coolant system and connected systems, including steam generators
- Engineered safety features
- Instrumentation and controls

- Electric power, including offsite and onsite power systems
- Auxiliary systems
- Steam and power-conversion system
- Radioactive-waste management

Because the nuclear/industrial facility boundary may involve a process-heat transfer line or other non-traditional energy-conversion system, it is relevant to review NRC guidance for DC/COLAs dealing with power-conversion systems. In NUREG-0800, SRP, Section 10.3, "Main Steam Supply System," the staff describes the review of the main steam supply system (MSSS) as it extends from the containment up to the turbine stop valve. The specific areas of review are specified as follows:

1. *The review should verify that portions of the MSSS that are essential for safe shutdown of the reactor or for preventing or mitigating the consequences of accidents are evaluated to determine the following:*
   a. *A single malfunction or failure of an active component would not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.*
   b. *Appropriate quality group and seismic design classifications are met for safety related portions of the system.*
   c. *The system is capable of performing multiple functions, such as transporting steam to the power conversion system, providing heat sink capacity or pressure relief capability, or supplying steam to drive safety system pumps (e.g., turbine driven AFW pumps), as may be specified for a particular design.*
   d. *The MSSS design includes the capability to operate the atmospheric dump valves remotely from the control room following a safe-shutdown earthquake (SSE) coincident with the loss of offsite power so that a cold shutdown can be achieved by depending only on safety-grade components.*
2. *The MSSS review should include measures that limit blowdown of the system if a steam line were to break.*
3. *The review includes the design of the MSSS with respect to the following:*
   a. *Functional capability of the system to transport steam from the nuclear steam supply system as required during all operating conditions.*
   b. *Capability to detect and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions.*
   c. *Capability to preclude accidental releases to the environment.*
   d. *Provisions for functional testing of safety-related portions of the system.*

NUREG-0800, Section 10.3, "Acceptance Criteria #3" [Technical Rational], states:

For multiple-unit sites, units may cross-connect the MSSSs for startup, maintenance, or other related purposes. For such shared systems, the licensee must show that each MSSS can perform all of its required safety functions for its respective unit. Meeting GDC 5 will ensure that shared MSSSs at multiple-unit sites will execute their respective safety functions regardless of malfunctions in the other units.

NUREG-0800, Sections 10.4.1, "Main Condensers, Acceptance Criteria #1," states:

Acceptability of the design of the MC [main condensers] and support systems, as described in the applicant's safety analysis report (SAR), is based on meeting the requirements of General Design Criterion 60 (GDC 60) and on the similarity of the design to that of plants previously reviewed and found acceptable. The design of the MC and support systems is acceptable if the integrated design of the system meets the requirements of GDC 60 as related to failures in the design of the system which do not result in excessive releases of radioactivity to the environment.

NUREG-0800, Section 10.4.5, "Circulating Water System" (CWS), Acceptance Criteria #1 [Technical Requirements] states:

GDC 4 requires that structures, systems, and components important to safety shall be designed to accommodate the effects and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Although the circulating water system is not safety related, GDC 4 establishes CWS design limits that will minimize the potential for creating adverse environmental conditions (e.g., flooding of systems and components important to safety). Meeting the requirements of this criterion provides a level of assurance that systems and components important to safety will perform their intended safety functions.

NUREG-0800, Section 10.4.6, "Condensate Cleanup Systems, Acceptance Criteria #2" [Technical Requirements] states:

For indirect cycle (pressurized water reactor (PWR)) plants, SRP Section 5.4.2.1 provides the criteria for acceptable secondary water chemistry. SRP Section 5.4.2.1 refers to the guidelines provided in the latest version in the EPRI report series, "PWR Secondary Water Chemistry Guidelines."

NUREG-0800, Section 10.4.7, "Condensate and Feedwater Systems, Acceptance Criteria #4," regarding heat removal capability, states:

The requirements of GDC 44, as related to the capability to transfer heat from structures, systems and components important to safety to an ultimate heat sink are met by demonstrating that the CFS [condensate and Feedwater system] is capable of providing heat removal under both normal operating and accident conditions. Sufficient redundancy of components is demonstrated so that under accident conditions the safety function can be performed assuming a single active component failure (which may be coincident with the loss of offsite power for certain events.) The system demonstrates capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

Regulatory Guide 1.206, "Applications for Nuclear Power Plants," Rev 1 [Ref [15]])

This RG provides information on the format and content of applications for nuclear power plants submitted to the NRC under 10 CFR 52, which specifies the information to be included in an application. The revised RG is divided into two parts. One section (C.1) supplies guidance for the organization, content, and format of an application under 10 CFR 52, which includes an applicant's transmittal letter and a series of multiple parts developed based on lessons learned from submitted applications to date. Subsections C.1.1–11 address each of the multiple parts of an application under 10 CFR 52, discuss the applicability and parts for different types of applications, and contain guidance for format and content of applications. Section C.2 contains information and guidance on selected regulatory topics related to the preparation, submittal, acceptance, and review of applications under 10 CFR 52. Although Revision 0 of this RG did contain technical application content guidance for describing SSCs in COLAs like NUREG-0800 guidance, the most recent RG revision no longer retains this similarity.

RG 1.206, Section C.1 states that for a COLA referencing a DC, the FSAR is similar in both format and content. However, a key distinction is that the detailed site-specific information should describe all interfaces with the referenced, as well as all departures, supplements, or exemptions from the referenced DC. The NRC staff expects COL applicants who reference a certified design to provide complete designs for the entire facility, including appropriate site-specific design information to replace the conceptual design portions of the Design Certification Document (DCD) for the referenced certified design. Refer to Figure 2, extracted from RG 1.206 (Revision 0), which displays a typical breakdown of design information between DC and COLAs.

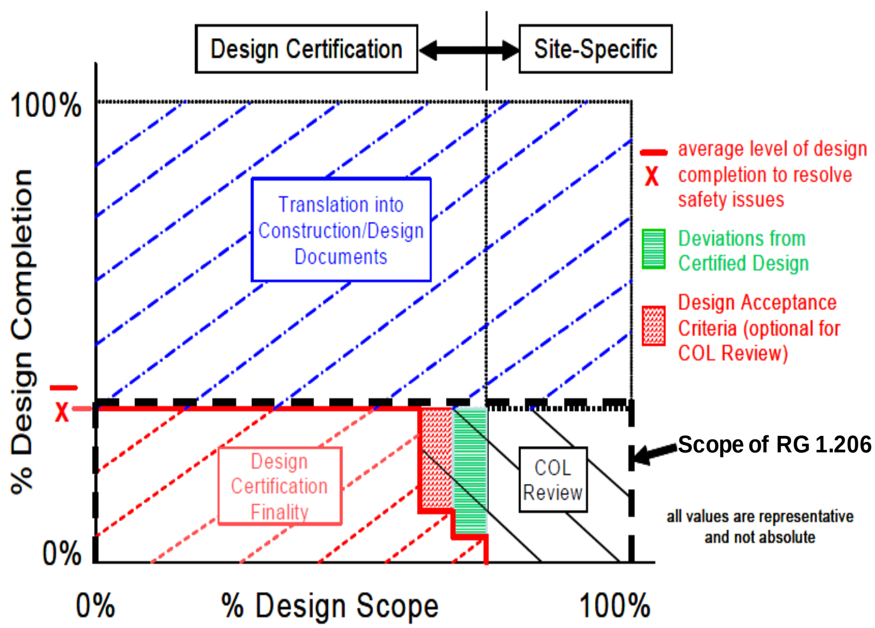

Figure 2. COLA referencing a certified design.

Section C.2.6, Conceptual Design Information—Design Certification

The requirements of 10 CFR 52.47(a)(24), specify that the DC application contain a representative conceptual design for those portions of the nuclear power plant for which the application does not seek certification to aid the staff in its review of the DC FSAR and to permit assessment of the adequacy of the interface requirements in 10 CFR 52.47(a)(25).

COL applicants that reference a DC should provide a complete design for the entire facility, including appropriate site-specific design information to replace any conceptual design portions for the referenced certified design. DC applicants facilitate the NRC staff's review of applications by including in the DCDs conceptual designs that offer a more comprehensive design perspective. These conceptual designs typically include portions of the balance of plant of the nuclear facility. However, because the conceptual portions of the design are not certified, the COL applicant needs to address them. The NRC does not consider replacement of conceptual-design information with actual-design information to be a departure from the DC because the conceptual design was never certified. However, for those instances in which the actual design differs from the conceptual-design information, the COL applicant should explain how these differences will affect the NRC's evaluation of the certified design and the design PRA, as applicable.

The level of detail needed for the site-specific designs that replace conceptual designs should be consistent with the level of detail provided in the DCD for the non-conceptual (or specific) designs and should be sufficient to resolve all safety issues.

RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants—LWR Edition," Appendix A, "Interfaces for Standard Designs" [Ref [16]]

As stated in 10 CFR 52.47, the DC is to describe an essentially complete plant with the option for representative conceptual designs for those portions of the plant for which the application does not seek certification. This may be accepted provided appropriate interface requirements are also identified. The conceptual design is intended to aid the NRC in its review of the DC FSAR and to permit assessment of the adequacy of the interface requirements. RG 1.70, Appendix A provides guidance regarding acceptable approaches for describing standard plant interfaces:

> *Safety-related interfaces must be identified and defined for standard designs submitted under Option 1 (Reference Systems) of the Commission's standardization policy to establish the requirements that must be met and assumptions that must be verified by other unspecified portions of a nuclear plant design to ensure that systems, components, and structures within the standard design will perform their safety functions. Safety-related interfaces also include information that may be useful in the design and staff review of the unspecified portions of the plant design. The safety functions of a standard design are those essential functions that ensure (1) the integrity of the reactor coolant pressure boundary; (2) that the specified acceptable fuel design limits are not exceeded as a result of anticipated transients; (3) the capability to shut down the reactor and maintain it in a safe shutdown condition; and (4)the capability to prevent or mitigate the consequences of an accident that could result in radiation exposures in excess of applicable guidelines. Interfaces are used, therefore, to provide a basis for ensuring that the matching portions of a nuclear plant design, as described in a PSAR for a CP application that references the standard design or in another Standard Safety Analysis Report (SSAR) for a matching portion of the plant, are compatible with the standard design regarding the safety-related aspects of the plant design.*
>
> *This appendix describes safety-related interfaces, for light-water reactors only, that should be presented at the preliminary design stage of review by the reactor vendor in a Nuclear Steam Supply System SSAR (NSSS-SSAR) and by the architect-engineer in a Balance-of-Plant SSAR (BOP-SSAR). The interfaces for a BOP-SSAR, are also directly applicable to an SSAR describing an entire nuclear plant (NSSS plus BOP but excluding utility- and site-specific items). This appendix also describes an acceptable format for presenting interfaces in an SSAR.*
>
> *Criteria for determining the acceptability of interfaces, as necessary for safety, are not included in this appendix. While not identified specifically as interface acceptance criteria, the criteria are part of other guidance already made*

available by the NRC, including that contained in the regulations, regulatory guides, and codes and standards.

RG 1.70, Appendix A, II. "Sources of Interfaces," identified interfaces for standard designs as being derived from the following sources:

1. *Requirements for safe operation of the standard design that must be satisfied by matching portions of the plant design or by the utility (e.g., cooling water and electric power requirements for the NSSS that must be provided by the BOP, an in-service inspection program for the NSSS and BOP that must be provided by the utility).*
2. *Assumptions made for the standard design that must be more precisely defined during the design coordination effort between the reactor vendor and the architect engineer or between the architect-engineer and the utility (e.g., mass and energy release rates during a LOCA specified by the reactor vendor that must be coordinated with the containment design provided by the architect-engineer).*
3. *Site-related design assumptions upon which the standard design is based.*
4. *Criteria pertinent to the standard design described in the SSAR under review that may be useful for the design and staff review of matching systems, components, and structures (i.e., within the standard design, safety criteria for the items including codes and standards, General Design Criteria, and regulatory guides).*

2.2 NRC Historical Precedents

2.2.1 Midland Nuclear Plant

The application for a CP of the Midland Nuclear Plant identified a dual pressurized-water reactor (PWR) with each reactor core proposed at 2,452 MW(t). The application was filed with the Atomic Energy Agency (the predecessor agency to the NRC) on January 13, 1969. The CP application included a preliminary safety-analysis report (PSAR) and 32 amendments [Ref [17]]. Following staff review and a public hearing before the Atomic Safety and Licensing Board, CPs were issued on December 15, 1972. The application for an OL was filed in 1977 but construction of the plant was halted and never completed as a nuclear power plant. However, the Midland plant does identify the single historical precedent for a commercial nuclear power plant providing steam offsite to an industrial facility; a situation not unlike what is envisioned for collocated advanced reactors.

A feature of the Midland design was the provision to furnish process steam as well as electricity to an industrial facility adjacent to the nuclear plant site. The steam in normal plant operation was to be furnished at various pressures and quantities [from 50 to 675 pounds per square inch (absolute) (psia)]. Two headers for each pressure were to transport 191 psia and 50 psia steam to the site boundary. A single additional header was to transport 675 psia steam to the site boundary. The radioactivity content of the steam was required to comply with the limits set forth in 10 CFR 20.

The Midland process-steam control system was designed to control high- and low-pressure process steam to the industrial plant and to control transfers between process-steam operating modes. There were three modes of operation. In Mode 1, Unit 1 supplied steam for both high- and low-pressure evaporators. Extraction steam from the turbine provided heating steam to low-pressure evaporators. Mode 2 was similar to Mode 1, except the heating steam to low-pressure evaporators was provided by means of pressure-reducing valves from the main steam header. In Mode 3, Unit 2 supplied heating steam for both high- and low-pressure evaporators. The control system was designed to provide smooth transfer from one mode of operation to the other.

Approximately 75% of the steam heat energy supplied by the nuclear boiler system was to be used to generate electrical energy. Steam containing the remaining heat energy was to be transported to the site boundary for process use by the industrial plant. Most of the steam was to be condensed and returned to the nuclear boiler system as heated feedwater. The steam not condensed was to be replaced by treated makeup water from the industrial energy user.

Based on its review, the staff concluded that the power-conversion system, including the provision to supply steam to the industrial facility, was in conformance with the regulatory criteria and design bases, could perform its designed functions, and was therefore acceptable.[k] The scope of this review is similar to that discussed in this paper for the energy conversion system.

2.3 Regulatory Foundation for Establishing Top-Level Regulatory Criteria

Top-level regulatory criteria for an energy transfer system can be determined by reviewing example interface requirements in RG 1.206, (Revision 0) Section 10, which provide the NRC guidance regarding FSAR content for the power conversion system and SRP Sections 10.2–4, which also address the power-conversion system. The safety functions of the nuclear facility that must be preserved through the interface using requirements ensure:

1. Integrity of the functional containment, including the fuel particles, the fuel matrix, and fuel-element graphite (if applicable), primary-coolant transport circuit, and reactor building
2. Capability of the fuel to stay within design limits as a result of anticipated transients
3. Capability to shut down the reactor and maintain it in a safe shutdown condition
4. Capability to prevent or mitigate the consequences of an accident that could result in radiation exposures in excess of applicable guidelines.

2.4 Regulatory Foundation Summary

In general, NRC regulations and guidance specify that DC and COLAs together will contain a complete description of the nuclear energy plant, including safety and non-safety portions of plant systems. With respect to the non-safety portions of the plant, the staff expects these SSCs will be evaluated to ensure impacts to the safety basis are acceptable. The regulations and guidance documents do not describe situations such as the Midland arrangement with respect to scope of NRC regulatory jurisdiction. However, the Midland experience does provide an example where NRC approved a configuration in which process steam could be used in a facility not under their nominal jurisdiction. It appears reasonable to conclude that facilities that use process-steam heat and are located offsite could be considered outside NRC regulatory jurisdiction given proper sets of interface requirements are employed.

NRC regulations and guidance require plant descriptions in DC and COLAs to be sufficient to permit understanding of system designs and their relationship to associated safety evaluations. All items pertinent to supporting the safety analyses would need to be described. For advanced-reactor applications, content expectations set in accordance with expectations identified in NEI 18-04 [Ref [18]] would include SSC descriptions that:

1. Mitigate the consequences of design basis events (DBE) to within the licensing basis event (LBE) frequency-consequence (F-C) target and mitigate design basis accidents (DBA) that only rely on the safety-related (SR) SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions.

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[k] Further information can be found in NUREG-0793, “Safety Evaluation Report related to the operation of Midland Plant, Units 1 and 2 Docket Nos. 50-329 and 50-330,” dated May 1982.

2. Prevent the frequency of beyond design basis events (BDBE) with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target.
3. Prevent or mitigate any LBE from exceeding the F-C target or make significant contributions to the cumulative-risk metrics selected for evaluating the total risk from all analyzed LBEs.
4. Require special treatment for DID adequacy.

NRC guidance for DC applications does provide for some systems not to be covered within the scope of that certification. Guidance specifies that conceptual design information and interface requirements be provided in the DC application. In such cases, site-specific COLAs would then address these areas with site specific design.

Regulations for modular reactor plants require that an application for certification 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.

3. DEFINING NUCLEAR-INDUSTRIAL FACILITY AND DESIGN CERTIFICATION BOUNDARIES

3.1 Proposed Approach

Because advanced-reactor modular designs are expected to be capable of supporting many different end use applications, site-specific designs that address energy-conversion systems and specific configurations with multiple modules could vary widely. Given this diversity, it is proposed that two sets of regulatory boundaries be established that effectively support requisite flexibility. These boundaries should be structured to confirm to the over-arching licensing strategy developed by applicant yet maintain an effective regulatory safety assessment pathway for NRC reviewers.

There will need to be clear understanding between the applicant and NRC staff regarding which systems are associated with each boundary and where those systems physically reside within the nuclear facility (and are therefore subject to DCA or COLA review). Systems identified as falling outside of the nuclear facility would be considered part of the industrial facility and beyond nominal NRC jurisdiction. There should be similar clarity regarding what plant scope is going to be addressed in an advanced-reactor DCA; remaining plant scope would be addressed in a site-specific COLA.

The following subsections expand upon key issues associated with the two boundary definitions.

3.2 The Nuclear Facility-Industrial Facility Boundary

Historically, NRC licensed commercial nuclear power plants are built and operated under provisions contained in 10 CFR 50. This generally involved a licensing review of the complete plant that included the nuclear steam-supply system, support systems, and balance-of-plant systems (i.e., energy conversion systems). These systems were typically installed within the nuclear site boundary and most areas were within the security-perimeter fence. As such, there was little question that all systems fell under NRC regulatory oversight.

Under 10 CFR 52, NRC will receive a nuclear power plant license application that includes a complete design for the entire facility. This is because requirements for a COLA contained in 10 CFR 52 necessitate that the FSAR provide sufficient description to permit understanding of systems design and an evaluation of their relationship to safety. Items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems, require discussion by the applicant "*insofar as they are pertinent*." This is a key term in the requirement. The following paragraphs discuss the basis for defining what may be pertinent with respect to an advanced-reactor configuration that sends steam or heat to offsite user(s) or otherwise utilizes non-traditional energy conversion system(s).

The NUREG-0800 (and RG 1.206 for COLAs) specify that complete descriptions of SSCs discussed in Part 52 be provided in the final DCA or COLA. For a site-specific design, the DCA would provide conceptual-design information and leave it to the COLA to address final design. In either case, before a license would be issued under Part 52, a complete description of the plant would need to be submitted to the NRC for review prior to approval. (See Section 2 for additional discussions of these documents.)

The challenge for the advanced-reactor applicant will be to describe enough of the plant and associated plant interfaces so as to exclude (offsite) customer energy-demand systems while still demonstrating to the staff sufficient protections are in place for the nuclear facility to provide a reasonable assurance of safety; this would include system transients that may be initiated in and transmitted from customer operated systems.

While an obvious starting point for establishing jurisdictional control might be the physical demarcation between the nuclear facility and industrial facility (as could be defined by the physical boundary of the nuclear plant site or a protected-area boundary security fence), it is also necessary to define the boundary at a systems-level; this is essential in order to assure nuclear plant safety. Since certain systems will undoubtedly traverse site-based physical boundaries, the advanced-reactor DCA or COLA needs a safety analyses that adequately bounds customer-initiated transients as might be communicated through boundary traversing systems. The safety analyses will therefore need to describe bounding assumptions for a plausible spectrum of customer-initiated transients and utilize appropriate and robust interface requirements that are met by process connections to the energy customer facility. As discussed earlier, there is precedence in the Part 52 DC process for using interface requirements for this purpose. For example, Part 52 DC application process allows those parts of the plant deemed to be site-specific and outside the scope of the DC, to provide interface requirements that must be met by the COL applicant and the design that is used at the site.

Interface requirements can take the form of process limits or equipment-design requirements. For instance, the DC may require a COLA to specify the site-specific ultimate heat sink that provides cooling of emergency service water such that maximum supply water temperature is 95°F under peak-heat-load conditions. Or it may require that the site-specific electrical-system design ensures the probability of losing power during the loss of power generated by the nuclear unit or transmission network, or the loss of the largest load, is minimized [see Ref [19]]. Other interface requirements may include criteria for site-specific firewater supplies. Interface requirements, such as those used in LWR DC and COL licensing, can provide useful insights as to how advanced-reactor licensing might approach creating adequate separation between nuclear and industrial facility systems.

3.2.1 Security-Related Considerations

As discussed in Section 2, 10 CFR 73 defines (in part), requirements for establishing and maintaining a physical protection system with capabilities to protect special nuclear material at fixed sites where special nuclear material is used. Both vital areas and material-access areas must be located within a protected area. Because of these security requirements, any nuclear facility boundary would need to encompass all areas of the plant that must be addressed within the plant's protected area (e.g., vital areas) as would be defined in their security plan.[1]

3.2.2 Nuclear Plant Design and Interface Considerations

Another major consideration in nuclear/industrial boundary definition pertains to SSCs that perform safety-related or risk-significant functions for the advanced reactor. All such systems would need to

[1] 10 CFR 73.2 defines “protected area” as an area encompassed by physical barriers and to which access is controlled.

reside within the nuclear facility jurisdictional boundary. The jurisdictional boundary definition would not apply with respect to other SSCs that are not safety related or risk significant; however, these SSCs could still challenge the plant or create transients that trigger nuclear safety-system mitigations. An approach to addressing this concern for areas outside of the safety-related and risk-significant SSCs in an HTGR example is proposed below.

A standard HTGR plant would include a primary-to-secondary heat-transfer device, such as a steam generator or an intermediate heat exchanger. This system would transfer heat from the helium primary system to a secondary medium - water in the case of the steam generator and helium in the case of an indirect process-heat supply system. This secondary medium would then transfer steam/process heat to an energy conversion system such as an onsite electrical-generator or other transfer system made up of pipes, valves, pumps, instrumentation, etc., that provides secondary steam or gas to an offsite customer. The heat transfer fluid would then be returned to the HTGR primary-system heat exchanger. This transfer system would start at the secondary-side outlet of the primary-system heat exchanger, traverse the HTGR site (nuclear facility), and leave the HTGR site to enter the customer (industrial) facility. A similar transfer line would provide return flow back to the HTGR heat exchanger. The logical interface boundary between the two facilities would be at some point in the transfer system before the feeding part of the system departs the HTGR site and after the return line enters the HTGR site. The energy-transfer function of this pipe is not unlike a transmission cable leaving the site that transfers electric power offsite. Interface requirements and the nuclear-facility-side protection devices must be identified and defined sufficiently so that the safety analysis can bound all possible transients that might be initiated at the industrial facility.

Based on the requirements in Part 52, guidance in NUREG-0800 and RG 1.206, and industry precedents, an energy-conversion system located within the HTGR protected area (such as a turbine generator that produces electric power), is likely integral to the operation of the nuclear side of the plant and under control of the HTGR control room; this would be considered within the nuclear facility rather than a part of the industrial facility. This conclusion is based on 10 CFR 52.47 and 52.79 requirements for DC and COLAs to describe systems “insofar as they are pertinent,” and the integral relationship the onsite electric power system would exhibit with the nuclear facility, including but not limited to electric plant control from the HTGR control room, impact on electric power supplies to the HTGR plant, the potential for turbine-generator missiles, proximity with respect to security issues, water quality of steam-generator feed, cooling-tower plume impacts, and flooding issues with the condenser cooling system. However, it may be justifiable to exclude from the nuclear facility (and Part 52 licensing scope) an energy-demand system such as a process-heat system for a petrochemical process or an offsite turbine generator that is located outside of the protected area, independent from the HTGR site such that the system is not controlled from the HTGR facility. Nor would the HTGR be dependent on, or adversely affected by, any system outputs (provided appropriate interface requirements are established to preclude deleterious transfer system effects).

Regardless of whether the energy-conversion and demand systems are within the nuclear facility, safety analysis would be required with respect to potential hazards due to missiles, security issues, flooding issues, process-steam feedback, or any other plausible impact to HTGR SSCs that perform a safety function. An offsite energy-demand system would require a process-heat transfer system that would serve as the interface between the HTGR and customer sites. Analysis would need be performed of the potential impacts that the transfer system might impose on the HTGR, and both preventative and mitigative measures would be necessary based on the safety analyses.

To understand the scope of this analysis, a review of NUREG-0800 guidance and RG 1.206 (Revision 0) concerning energy-conversion systems offers further insight. These guidance documents describe regulatory requirements and acceptance criteria such systems must meet. If one considers the aforementioned energy-transfer system as akin to a main steam-supply system in a pressurized LWR, it could be expected that this system would have monitoring and, if necessary, isolation capability similar to main steam isolation valves.[m] If the downstream portion of the process-heat transfer system ran to customer property, then appropriate interface requirements would be implemented for sections of pipe leading up to the point where the nuclear-facility isolation or other protection devices exist. Similarly, the condensate return line from the industrial facility to the nuclear facility would also need to be evaluated for impacts such as line breaks, water quality for use in the steam generator, and heat-removal needs.

Example interface requirements can be noted in RG 1.206, (Revision 0), Section 10.2-4, which provide NRC guidance regarding FSAR content for the power-conversion system. In reviewing these cases, a set of high-level design and interface requirements can be proposed for a transfer system. The combination of nuclear-facility transfer-system design and interface requirements imposed on the site-specific portion of the transfer-system design (for the industrial facility) would need to demonstrate all applicable requirements for energy conversion systems would be met.

Review of applicable regulatory guidance yields a list of functional requirements that could be imposed on the combination of nuclear-facility transfer-system design and interface requirements needed to meet applicable regulatory requirements. These are:

1. Failures or transients within the industrial-facility portion of the transfer system would not preclude safety-related portions of the nuclear facility from functioning as required during normal operations, anticipated operational occurrences, and accident conditions.
2. Nuclear-facility plant system transients caused by industrial-facility systems or the electrical-transmission grid would be limited (in frequency and severity) and analyzed in the plant's safety analyses.
3. No portion of the transfer system within the scope of the industrial facility would be required to perform any safety, risk-significant, or safe-shutdown function or be relied upon as a supporting system to a safety-related system.
4. The transfer system would have monitoring capabilities to detect disturbances and, if required by the advanced-reactor safety analysis, facilitate appropriate responses during transients and accidents.
5. Releases of radioactive material from the transfer system would need to meet all required limits as is determined to be applicable to the discharge. Monitoring and/or sampling may needed to ensure applicable limits are met.

Once the above functional requirements for each interface with the industrial facility are met, an appropriate nuclear-facility boundary can be established. Components that need to physically reside within the protected-area boundary to satisfy security requirements would be part of the nuclear facility.

3.2.3 Design Certification Boundary

Having defined the nuclear facility as those systems that fall within the regulatory jurisdiction of the NRC, the next step is to determine the scope of advanced-reactor systems that should fall within a DC and those to be addressed in a COL. This discussion will focus on the boundary between the DC and the site-specific portion of the nuclear plant; both areas are within the nuclear facility boundary and exist under NRC jurisdiction.

Because of the potential for modularity in advanced-reactor designs, future DCAs may only request certification for a portion of what is typically part of a recent LWR DCAs. While areas such as the control room, radiological-waste facility, and reactor service building may be included within the nuclear-facility boundary, they may be excluded from an advanced-reactor DC along with typical secondary-side design

[m] The HTGR safety analysis may determine that such isolation capability is not required in which case this design feature would not be a boundary consideration

elements. This can be successfully done by defining interface requirements for affected systems and structures. The basis for such an approach is discussed below.

Standard plant systems include those expected to be described in DCDs. 10 CFR 52.47 describes the type of information to be included in a DCA. For a modular nuclear reactor design, the DC must describe and analyze the possible operating configurations of the reactor modules with common systems, interface requirements, and system interactions. The DC final safety analysis must also account for differences among the configurations, including any restrictions that will be needed during construction and startup of a given module and ensures the safe operation of any module already operating.[n] Plant systems described in the DC would be reviewed and approved by NRC and contain interface requirements for those portions of the plant outside of the DC (see Item 2 below).

Part 52 provides for the development of a DCA for a standard advanced-reactor module as part of a single or multimodule reactor plant using different site-specific information. This can also apply to different energy-conversion systems (e.g., turbine generators for electric power production or process steam-delivery system equipment) for modules. The certified portion of the plant would include standard parts of the nuclear facility but exclude site-specific design details. The DCA would then utilize conceptual-design information for an energy-conversion system and provide interface requirements that address:

1. Requirements for safe operation of the standard design that must be satisfied by matching portions of the site-specific design
2. Site-related design assumptions upon which the standard design is based
3. Criteria pertinent to the standard design described in the DCA that may be useful for the design and NRC review of matching SSCs within the standard design, safety criteria for the items including codes and standards, principal design criteria, and regulatory guides
4. Requirements need to preserve SSC safety functions identified in NEI 18-04 guidance and discussed in Section 2.4 of this document.

As was already noted in Section 2 and in accordance with 10 CFR 52.47(a)(24) requirements, a representative conceptual design for portions of the plant for which the application does not seek certification will be necessary to aid NRC reviewers in understanding the FSAR and permit assessment of interface requirements adequacy. The interface requirements must be sufficiently detailed to allow a full evaluation of a complete FSAR.

The certified portion of the reactor plant design and safety analysis would need to bound all worst-case operating and accident scenarios for potential site-specific energy-conversion systems. The DCA would also include conceptual-design descriptions of equipment and interface requirements for potential operating configurations. However, conceptual-design information would not be expected to be included in the final certified design. Each COLA that references the DC would describe site-specific design details/operating information and show that the site-specific systems, including the energy-conversion system, does satisfy applicable DCD interface requirements. NRC would then review and document approval of COLA information in an SER. Subsequent COLAs (S-COLA) referencing the same design certification and using the same site-specific systems could replicate the information provided in the initial reference COLA (R-COLA), thereby avoiding redundant NRC review of information; this strategy is allowed under the NRC's “one issue, one review, one position” design-centered review approach.[o]

The DCA would be crafted to provide the degree of flexibility desired by the applicant regarding future deployments and address interfaces, transients and accident conditions for a full range of nuclear

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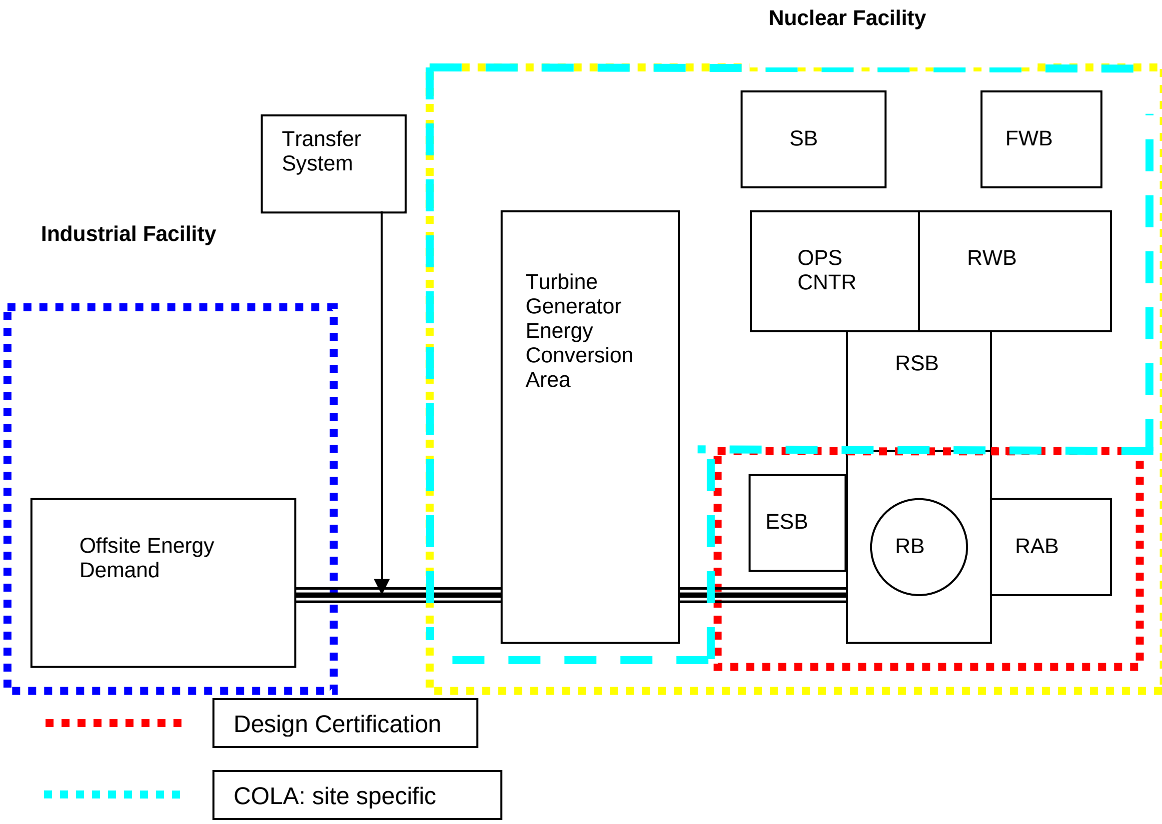
[n] See 10 CFR 50.47(c)(3).

[o] For further information, please refer to Regulatory Information Summary 2006-06, New Reactor Standardization Needed To Support The Design-Centered Licensing Review Approach, and Regulatory Guide 1/206 Revision 1, Section C.2.7

facility and energy-conversion system configurations, operating conditions, process demands, and integrated risk that include total accident source terms. The DC would also address multimodule operations of varying ratings and configurations at candidate installations along with effected operations whenever one or more other modules are being constructed, tested, or while one or more other modules are refueling, in shut-down for maintenance, or undergoing decommissioning.

Figure 3 illustrates typical demarcations for a single module HTGR (used as an example) between the nuclear and industrial facility, and demarcation between the DC and COLA.



RB: Reactor Building including reactor vessel, primary circuit, cross vessel, secondary circuit pressure vessel, piping connecting the primary helium circuit to support systems, (e.g., shutdown cooling system, primary helium service and purification system)
RAB: Reactor Auxiliary Building
ESB: Electrical Service Building
RSB: Reactor Support Building
OPS CNTR: Operations Center and Control Room
RWB: Radwaste Building
SB: Security Building
FWB: Fire Water Building and Fire Pump House

Figure 3. Notional regulatory demarcation boundaries for the example HTGR.

It has been noted that a General Atomics Inc., conceptual design report submitted to the U.S. DOE, proposed a 350-MWt steam-cycle modular helium reactor to operate at high temperature as a gas-cooled, graphite-moderated reactor utilizing a prismatic graphite block fuel form to provide process heat and steam to an offsite industrial facility. The arrangement for this demonstration plant consists of two onsite nuclear islands (NI)[p] and the onsite energy-conversion area (ECA)[q]. The NI contains the reactor building and other SSCs comprising the standard reactor module (SRM) and the adjacent balance-of-NI structures house SSCs related to plant control, fuel handling and storage, and various reactor-service and auxiliary systems. The ECA constitutes the balance of plant, including the turbine generators for electricity production and the process-steam delivery-system equipment. While the General Atomics conceptual design report did not specially address regulatory boundaries, it did seek a DC for the SRM portion of the

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p The term *Nuclear Island* used in the General Atomics report is not synonymous with the term *nuclear facility* used in this report to define the systems within the NRC oversight boundary.

q While the ECA with the turbine generator was considered physically separate it was still within the HTGR site area and therefore still considered within the *nuclear Island* boundary from a regulatory oversight perspective design package. The scope of this SRM proposal, and its associated DCA, provides an example that includes:

- SSCs within the reactor building
- SSCs within the reactor auxiliary building
- SSCs within the electrical services building
- NI cooling water system
- Spent-fuel cooling-water system
- Shutdown cooling-water system

Other SSCs within the proposed NI such as the control room, reactor service building, and radiological-waste building, would not be within the scope of the certified design. The ECA would also not be included within the scope of the SRM. A DC for such an SRM would then need to provide conceptual-design information and interface requirements for the portion of the NI not addressed as part of the DC and the ECA systems and structures.

The process-heat lines that traverse offsite would be part of the nuclear-facility scope up to the point of the nuclear/industrial facility boundary, at which they would enter the industrial facility. This line would need to satisfy the boundary-interface requirements discussed in Section 3.2 of the paper (Ref 1).

3.3 Defining COLA Scope

The plant scope that would be addressed as a function of a specific site (i.e., those portions outside the nominal DC scope) would fall into two subcategories:

1. Plant systems that are not part of the DC but description is expected to be addressed in a COLA. The COLA would address interface requirements identified in the DC for systems not within the DC.
2. Plant systems, the description of which would not be expected in detail in the DC or the COLA, except as necessary to describe how applicable DCD/COLA interface requirements are met by these systems. These systems would be considered part of the industrial facility. Detailed descriptions of these plant systems and programs would not be reviewed or approved by the NRC. However, depending on the specific design, the COLA would contain explicit interface requirements for those portions of the industrial plant that interface with COLA systems.

The COLA referencing a DC would provide site-specific design information for all areas addressed as conceptual design in the applicable DC including the energy-conversion system. The COL application would also provide information demonstrating that the site-specific design satisfied the interface requirements in the DC. For a COLA that does not reference a DC, the applicant would need to submit design information on the entire plant within the nuclear facility and could forego inclusion of conceptual-design information.

The first COLA for a site-specific plant arrangement could serve as the R-COLA with S-COLAs following that reference the same design certification and use the same site-specific systems. This practice of replicating information provided in the R-COLA by using S-COLAs minimizes redundant NRC reviews by taking advantage of the NRC “one issue, one review, one position” design-centered review approach (see Refs 8 and 13).

3.4 Scope Outside of COLA

No specific descriptive system information would be necessary in the COLA concerning the scope of the plant outside the nuclear facility. This part of the plant would be outside the scope of typical NRC review. The COLA would focus on demonstrating how interface requirements specified in either the

COLA or DC would be met by the industrial facility interface. Figure 4 illustrates the overall nuclear-industrial facility boundary approach that would result.

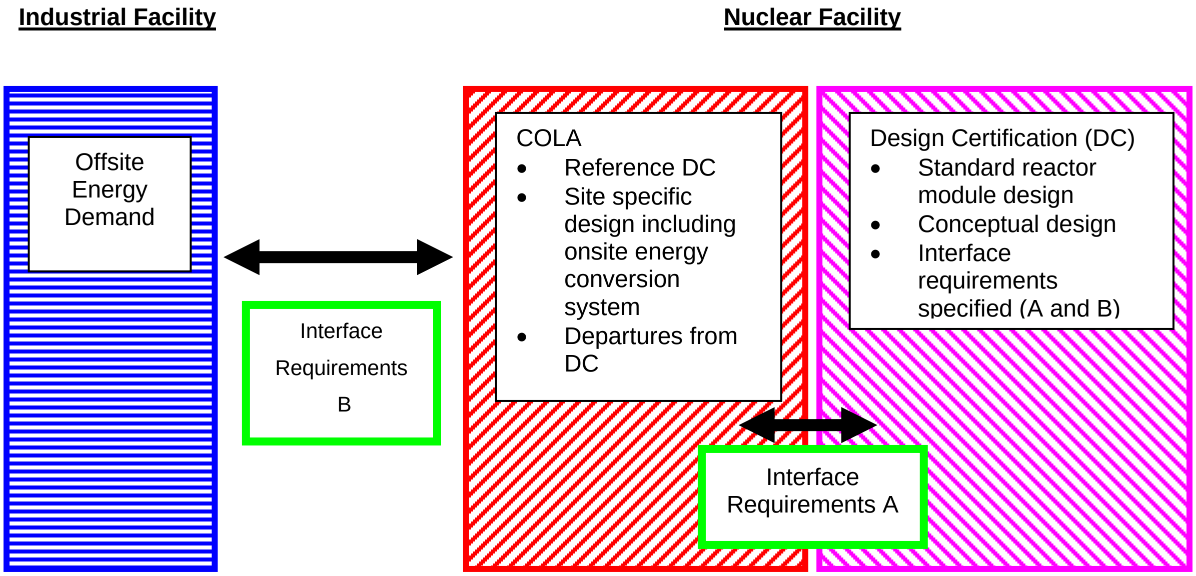

Figure 4. Illustration of approach to nuclear-industrial facility and DC/COL boundaries.

3.5 Protection from Transients and Hazards Generated from Facilities Outside NRC Regulatory Jurisdiction

As already discussed, the DC safety analyses bounds any transients initiated within the industrial facility. In the case of explosion hazard, the DC would need to specify appropriate analyses demonstrating that offsite explosion hazards were bounded by the DC analyses. The COLA would provide analysis demonstrating that the DC interface requirements were met. Specific system descriptions of the industrial facility would not be required in the COLA beyond what is needed to demonstrate interface requirements and hazard types were properly analyzed (e.g., providing a list of hazardous chemicals, their quantities and distance from the site buildings).

4. KEY APPROACH ELEMENTS

An approach has been proposed regarding how regulatory boundaries can be established between an advanced-reactor nuclear facility and energy end-user facility. Interfaces would be relied upon to separate the industrial facility from nuclear facility jurisdiction. To enable this concept, agreements between involved stakeholders are needed regarding the following boundary definition attributes:

1. The NRC has full regulatory jurisdiction over plant facilities that must be protected under physical-security regulations and all SSCs within the plant's security boundary; these components would be part of the nuclear facility.
2. All SSCs that perform safety-related or risk-significant functions for the advanced-reactor would be included within the nuclear facility boundary.
3. An energy-conversion system that is located within the advanced-reactor protected-area boundary, is integral to the facility, and is controlled by the nuclear facility control room, would be considered within the nuclear facility. An energy-conversion system could be excluded from the nuclear-facility jurisdictional scope if it is located outside the protected-area boundary and separated from the nuclear facility by a transfer system with robust interface criteria that operate to ensure the nuclear facility is not dependent on or adversely affected by events occurring in the industrial facility.

4. Regardless of whether the energy-conversion system lies within the nuclear facility, analysis would be required of the system with respect to potential missiles, security issues, flooding issues, or other impacts to SSCs that perform a nuclear safety function.

5. With respect to regulatory jurisdiction, the boundary between the advanced-reactor nuclear facility and the industrial facility can be defined by properly describing these boundaries in the nuclear-facility system design, transfer-system design, and using interfaces with appropriate sets of conceptual-design information and interface requirements. The following elements are suggested as representing an appropriate set of high-level design and interface requirements for this boundary.[r]

    a. Failures or transients within the industrial facility portion of the transfer system would not preclude safety-related portions of the nuclear facility from functioning as required during normal operations, anticipated operational occurrences, and accident conditions.

    b. Nuclear-facility plant system transients caused by industrial-facility systems or the electrical transmission grid would be limited (in frequency and severity) and analyzed in the plant's safety analyses.

    c. No portion of the energy-transfer system residing within the scope of the industrial facility would be required to perform any nuclear safety or safe-shutdown function or be relied upon as a supporting system to a safety-related system.

    d. The transfer system would have monitoring capabilities to detect and, if required by the safety analysis, facilitate appropriate responses during transients and accidents.

    e. Releases of radioactive materials from the transfer system would meet required limits. Monitoring and sampling may be required, as necessary, to ensure such limits are met.

6. Specific-system descriptive information would not be needed for the DCA or COLA for plant scope outside the nuclear facility as this part of the plant would be considered outside the normal scope of NRC review. Instead, the COLA would be obliged to demonstrate how interface requirements contained in either the COLA or DC would be met by industrial facility interfaces.

7. The advanced-reactor nuclear facility can be further subdivided into systems addressed within a 10 CFR 52 DCA and those described in a site-specific Part 52 COLA. The DCA would, as necessary, address the degree of flexibility desired by the applicant regarding the deployment of the advanced-reactor type and describe and analyze the possible operating configurations of associated reactor modules. The analysis would include common systems, interface requirements, system interactions, and account for differences among configurations; it would also include any restrictions necessary during construction and module startup to ensure the safe operation of any already operating module(s). At minimum and using guidance contained in NEI 18-04, SSCs addressed in the scope of a DC should include those SSCs that perform the following functions:

    a. Mitigate the consequences of DBEs to within the LBE F-C target, and mitigate DBAs that only rely on the SR SSCs to meet the dose limits of 10 CFR 50.34 using conservative assumptions

    b. Prevent the frequency of BDBEs with consequences greater than the 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target

    c. Prevent or mitigate any LBE from exceeding the F-C target or make significant contributions to the cumulative-risk metrics selected for evaluating the total risk from all analyzed LBEs

    d. Require special treatment for DID adequacy.

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[r] Any interface with the industrial facility would involve a transfer system that could provide steam or process heat to the customer and return condensate or makeup fluid to the nuclear facility.

8. Conceptual design information and interface requirements are to be provided in the DCA, as appropriate, in order to address SSCs not within the scope of the DC. These interface requirements would address[s]:
   a. Requirements for safe operation of the standard design that must be satisfied and matched to respective portions of the site-specific design
   b. Site-related design assumptions upon which the standard design is based
   c. Criteria pertinent to the standard design described in the DCA that may be useful for the design and review of matching systems, components, and structures (within the standard design, safety criteria for the items including codes and standards, principal design criteria, and regulatory guides)
   d. Requirements to preserve the specific advanced-reactor safety functions.[t]
9. A site-specific COLA referencing a DC would provide site-specific design information for all areas that was addressed as a conceptual design in the applicable DC. This would include the energy-conversion system if such a system is within the nuclear facility boundary. Additionally, the COLA would need to provide information demonstrating that the site-specific design satisfied interface requirements contained in the DC. Verification would be needed to ensure the nuclear-industrial facility boundary interface requirements were satisfied.
10. For COLAs that do not reference a DC, the applicant would need to submit design information on the entire nuclear facility and would not include facility conceptual design information. This type of COLA would describe the nuclear industrial facility boundary interface requirements in its entirety and show they are satisfied by site-specific design.

5. REFERENCES

[1] General Atomics, Next Generation Nuclear Plant (NGNP) Prismatic HTGR Conceptual Design Project Conceptual Design Report - Steam Cycle Modular Helium Reactor (SC-MHR) Demonstration Plant, NGNP-R00016, Revision 0, 12/23/2010.

[2] Nuclear Regulatory Commission, SECY-11-0079, “License Structure For Multi-Module Facilities Related To Small Modular Nuclear Power Reactors,” Adams Accession Number ML110620459, dated June 12, 2011.

[3] NEI White Paper, “Microreactor Regulatory Issues,” dated November 13, 2019.

[4] NRC “Staff Requirements,” SECY-18-0076, “Options and Recommendations for Physical Security for Advanced-Reactors,” dated November 19, 2018.

[5] Nuclear Innovative Alliance (NIA) report, “Establishing Interface Requirements for “Major Portions” Standard Design Approvals,” dated September 2019.

[6] Nuclear Innovative Alliance (NIA) report, “Clarifying ‘Major Portions’ of a Reactor Design in Support of a Standard Design Approval,” dated in April 2017.

[s] These interface requirements for site specific SSCs would be in addition to those specified for the nuclear - industrial facility boundary

[t] It is assumed that these functions would be derived from guidance such as that contained in NEI Technical Report “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” NEI-18-04, Revision 1, August 2019

[7] 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors," Final Rule, 54 FR 15372, 4/18/1989.

[8] 72 FR 49352, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Final Rule, 8/28/2007.

[9] SECY-88-202, "Standardization of Advanced-reactor Designs," July 15, 1988.

[10] SECY-10-0034, "Potential Policy, Licensing, And Key Technical Issues for Small Modular Nuclear Reactor Designs," 3/28/2010.

[11] NRC “Staff Requirements” SECY-18-0076, “Options and Recommendations for Physical Security for Advanced-reactors,” dated November 19, 2018.

[12] NRC “Policy Statement on the Regulation of Advanced Reactors,” Final Policy Statement NRC-2008-0237, ML082750370, October 7, 2008.

[13] Nuclear Regulatory Commission Regulatory Guide RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” June 2020.

[14] NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants”.

[15] Regulatory Guide 1.206, " Applications For Nuclear Power Plants”, Revision 1, October 2018.

[16] RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants – LWR Edition," November 1978.

[17] Midland Plant, Units 1 and 2, Preliminary Safety Analysis Report, October 1968.

[18] Nuclear Energy Institute (NEI), “Modernization of Technical Requirements for Licensing of Advanced-reactors, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” NEI 18-04, Revision 1, Adams Accession Number ML19241A472, August 2019.

[19] Mitsubishi Heavy Industries, Ltd, MUAP- DC0020, Design Control Document for the US-APWR, Tier 1, Revision 2, October 2009.