

DANU-ISG-2026-02

NRC Application Pathway for Reactor Designs Previously Authorized by U.S. Department of Energy or Department of War

Interim Staff Guidance

November 2026

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ABBREVIATIONS

ANS	American Nuclear Society
ARCAP	Advanced Reactor Content of Application Project
ASME	American Society of Mechanical Engineers
CFR	*Code of Federal Regulations*
COL	combined license
CP	construction permit
DBA	design-basis accident
DBHL	design-basis hazard levels
DC	design certification
DOE	U.S. Department of Energy
DOW	U.S. Department of War
EAB	exclusion area boundary
EO	Executive order
EP	emergency preparedness
EPZ	emergency planning zone
ESP	early site permit
FPP	fire protection program
FR	*Federal Register*
FSAR	final safety analysis report
GDC	general design criterion(a)
ISG	interim staff guidance
LBE	licensing-basis event
LE	licensing event
LMP	Licensing Modernization Project
LOCA	loss-of-coolant accident
LPZ	low population zone
LWR	light-water reactor
MACCS	MELCOR accident consequence code system
MD	management directive
MHA	maximum hypothetical accident
MHTGR	modular high-temperature gas-cooled reactor
ML	manufacturing license
MST	mechanistic source term
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NQA	Nuclear Quality Assurance
NSRST	non-safety-related with special treatment
OL	operating license
OMB	Office of Management and Budget
PDC	principal design criterion(a)
PRA	probabilistic risk assessment
QAP	quality assurance program
QAPD	quality assurance program description
RG	regulatory guide
SAR	safety analysis report
SDA	standard design approval
SMR	small modular reactor
SR	safety related

SRM	staff requirements memorandum
SSC	structure, system, and component
STS	Standard Technical Specifications
TEDE	total effective dose equivalent
TMI	Three Mile Island
TS	technical specifications
χ/Q	atmospheric relative concentration

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INTERIM STAFF GUIDANCE

NRC APPLICATION PATHWAY FOR REACTOR DESIGNS PREVIOUSLY AUTHORIZED BY U.S. DEPARTMENT OF ENERGY OR DEPARTMENT OF WAR

DANU-ISG-2026-02

PURPOSE

This interim staff guidance (ISG) is part of the U.S. Nuclear Regulatory Commission's (NRC's) strategy to establish an expedited pathway for NRC review and licensing of reactor facility designs that have been previously authorized and tested and that have demonstrated the ability to function safely by the U.S. Department of Energy (DOE) or the U.S. Department of War (DOW).[1]

The purpose of this guidance is to provide a roadmap of NRC requirements and associated guidance documents in select, foundational areas of reactor facility safety and the protection of public health and safety.[2] As more applicants use this guidance, and the NRC gains experience in licensing reviews for reactor designs that have previously been authorized by the DOE or DOW, this and other guidance will benefit from future revisions. This ISG focuses on key, foundational technical areas that are generally anticipated to share strong overlap between the NRC and DOE or DOW requirements.

The NRC staff encourages developers to consider this guidance before or during development of their DOE or DOW authorization application and related NRC commercial licensing strategy, including consideration of pre-application engagements. This ISG is intended to assist pre-applicants in determining how they can best maximize the overlap between what would be needed to satisfy both sets of requirements. Maximizing the overlap may enable the NRC to leverage more information that has been previously reviewed and accepted by the DOE or DOW when the NRC staff performs its subsequent review of a commercial application.

This ISG is best used in conjunction with other closely related NRC activities, strategies, and stakeholder engagement opportunities as summarized in the "Rationale" section of this ISG. Specifically, this ISG provides guidance to facilitate the preparation of reactor facility applications for construction permits (CPs) or operating licenses (OLs) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 3), or combined licenses (COLs), manufacturing licenses (MLs), standard design approvals (SDAs), and design certifications (DCs) under 10 CFR Part 52,

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[1] Consistent with Executive Order (EO) 14347, "Restoring the United States Department of War," dated September 5, 2025 (Ref. 1), this ISG uses the "Department of War." In accordance with section 2(d) of EO 14347, the regulatory text for the limited scope rulemaking addressing section 5(d) of EO 14300, "Ordering the Reform of the Nuclear Regulatory Commission," dated May 23, 2025 (Ref. 2), continues to use the title "Department of Defense" to avoid creating inconsistencies with or causing confusion in the NRC's existing regulations in Title 10 of the *Code of Federal Regulations*.

[2] As noted in the "Applicability" section, the NRC staff plans to update the guidance in this document to reflect the results of other rulemaking and regulatory guidance updates in progress, as well as updates related to the identification of additional opportunities to leverage DOE or DOW authorizations in areas where those authorizations satisfy associated NRC regulatory requirements.

"Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4), that intend to leverage prior DOE or DOW authorizations.[3] This ISG is also intended to support streamlined NRC review of applications in areas that include pertinent information regarding safety functions related to DOE or DOW authorized reactor facilities by enabling NRC staff to adjust the focus and depth of that review, when appropriate. The portions of an NRC application that are addressed in the appendices of this ISG are expected to overlap with requirements to receive an authorization to construct and operate under DOE or DOW authorities. While this roadmap is primarily intended for developers that may first pursue a DOE or DOW authorization, this roadmap content may also be useful to prospective NRC applicants not pursuing DOE or DOW authorization as a part of their NRC licensing strategy.

This ISG does not affect the NRC's commitment to conducting independent reviews, nor does this ISG alter any underlying regulatory requirements an applicant must meet. Regardless of whether an applicant references a prior DOE or DOW authorization, it remains the applicant's responsibility to demonstrate compliance with all applicable NRC requirements and submit a complete application. The NRC will then conduct the independent assessments necessary to confirm whether such requirements have indeed been satisfied. Where an NRC applicant references information from a prior design authorized and tested by DOE or DOW, the applicant is responsible for justifying the applicability and relevance of the information it seeks to rely upon, identifying any limitations, and providing supplemental information when the referenced analyses or testing do not fully demonstrate compliance with NRC regulations. Where appropriately, the NRC will be able to tailor the focus and depth of its review.

BACKGROUND

Since the 1970s, the NRC has considered and used DOE-generated technical information that is evaluated and incorporated by NRC applicants in portions of their NRC class 103 applications. For example, NRC applicants have used information generated by the DOE and its national laboratory infrastructure to support individual topical areas (e.g., test results supporting the qualification of a specific material or data supporting the validation of an analytical tool) when addressing NRC requirements. This approach is consistent with existing NRC regulations, most notably 10 CFR 50.43(e), when addressing the accomplishment of safety functions in those applications.

Interactions with industry stakeholders and national policies described in recent EOs and as implemented by the associated agencies indicate an emerging and expanding interest in pursuing DOE/DOW authorization for the construction and operation of prototypical reactor facilities for the purpose of demonstrating various reactor designs (or major portions of the designs).[4] Prospective NRC applicants could then incorporate by reference, as appropriate, these DOE- or DOW-authorized demonstrations in support of the applicable portions of follow-on NRC applications for what may be the same or a similar reactor facility design. This overall approach generally aligns with the concepts in 10 CFR 50.43(e) as applied to the NRC licensing or other approval of new reactor facility designs that would use simplified, inherent, passive, or other innovative means to accomplish their safety functions.

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[3] The NRC is issuing this ISG to describe methods that are acceptable to the NRC staff for implementing specific parts of the agency's regulations and to describe information that the NRC staff needs in its review of applications for permits, certifications, approvals, and licenses.

[4] See, for example, EO 14154, "Unleashing American Energy," dated January 20, 2025 (Ref. 5); and EO 14301, "Reforming Nuclear Reactor Testing at the Department of Energy," dated May 23, 2025 (Ref. 6).

NRC applicants pursuing this approach may propose leveraging entire reactor facility designs, or major portions thereof, that have been previously authorized and operated pursuant to a DOE or DOW authorization. Where an applicant references a previously authorized DOE or DOW design, the NRC should adjust the focus and depth of its technical review to the maximum extent appropriate for attributes of the design reviewed in DOE or DOW authorizations. Any adjustments would be commensurate with the scope and depth provided for in that authorization and the applicant's demonstration of its applicability to NRC regulatory requirements. An applicant would still be required to submit a complete application to the NRC such that the agency can determine that all its applicable requirements are met.

RATIONALE

It is anticipated that multiple nuclear reactor facilities of varying designs, outputs, and functional purposes may be authorized and demonstrated through the associated DOE and DOW programs. These could range from only partial designs intended to perform criticality testing, to scaled prototype facilities, all the way to designs intended to be replicated for commercial operations. Therefore, the objective of this guidance is to facilitate the effective leveraging of prior DOE or DOW authorizations of reactor facility designs into applications and the NRC's licensing reviews of those applications for commercial reactor facilities that use those designs. This guidance is part of other activities that form key elements of that leveraging approach. These other activities include the following:

- the initiation of a rulemaking that adds new 10 CFR 50.43(f) to improve NRC licensing review efficiency, where applicable, by explicitly establishing by regulation an additional means for reactor applicants to demonstrate the safety functions of their reactor designs

- an addendum (Addendum No. 9) to the memorandum of understanding between the NRC and DOE, "Coordinating DOE and NRC Technical Expertise and Knowledge on Advanced Nuclear Reactor and Advanced Reactor Fuel Technologies," issued October 2025 (ML25303A288) (Ref. 7) that addresses the following:

  – coordination of DOE and NRC technical readiness and sharing of technical expertise and knowledge on advanced nuclear reactors and advanced reactor fuel technologies

  – enabling documented safety analyses for DOE-authorized nuclear facilities that have been tested and have demonstrated the ability to function safely to be leveraged to streamline and expedite future NRC licensing activities pursuant to EO 14300, section 5(d)

- support for the DOE's development of a crosswalk between DOE and NRC regulations and guidance that reflects similarities and differences to assist stakeholders in identifying opportunities for leveraging DOE authorizations for subsequent NRC licensing activities

The intended outcome of this coordinated approach is to facilitate commercial reactor licensing and industry deployment efficiencies by avoiding the need for both the DOE/DOW and NRC technical staff to review the same industry-developed content when assessing the risks associated with newly proposed commercial reactor facilities.

Realizing the full potential of this approach involves the consideration and implementation of three key components:

(1) Industry stakeholders planning to pursue this path are encouraged to have the "end in mind" when developing the technical and applicable regulatory basis necessary to receive DOE or DOW authorization to construct and operate the reactor facility. Although the DOE or the DOW would have jurisdiction over the facilities with respect to the specific DOE or DOW authorization, there will likely be many opportunities for industry applicants to consider related NRC requirements and implementing guidance when developing their respective DOE or DOW authorization application content. Considering those NRC requirements when developing a DOE or DOW application is expected to result in efficiency gains when a follow-on NRC application is developed and submitted for review.

(2) Industry stakeholders are encouraged to engage in preapplication interactions with the NRC to facilitate the NRC staff's understanding of the application that is either under review by the DOE or the DOW or that was previously reviewed by the DOE or the DOW. Such interactions would aid the NRC staff in understanding whether and to what extent attributes of the application or authorization may overlap with the NRC's regulatory requirements. The memorandum titled "Expectations for NRC Staff Involvement with Prospective NRC License Applicants Planning to Refer to DOE and DOW Authorization Processes," dated December 11, 2025 (Ref. 8), provides more information about the extent of the NRC staff's involvement with prospective applicants planning to implement this proactive preapplication engagement approach.

(3) The NRC's establishment and implementation of the "core team"[5] application review approach has resulted in efficiency gains in the NRC's licensing of new reactor facilities. These efficiencies have included reductions in both application review duration and review resources. This approach is closely connected with the preapplication engagements discussed above; moreover, it would facilitate the NRC's ability to minimize or eliminate the repetition of directly applicable DOE or DOW reviews in areas where the scope and outcome of those reviews align with associated NRC requirements.

APPLICABILITY

Applicants for class 103 licenses, permits, certifications, and approvals must comply with all applicable NRC regulatory requirements. For applicants using the 10 CFR Part 50 process, application requirements include those in 10 CFR 50.34, "Contents of applications; technical information." For applicants using the 10 CFR Part 52 process, application requirements for COLs include those in 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report." Additionally, the application requirements under the 10 CFR Part 52

[5] Core teams support the reviews of applications in a holistic, risk-informed manner. These core teams rely on a small number of staff members who are dedicated full time (or significant part time) to each review. Supplemental subject matter experts support the core team through well-defined, discrete assignments. A project charter defines this project structure and staffing philosophy at the outset of each review. Each charter identifies the core team members and subject matter experts needed to carry out the major activities involved in each review. Roles and responsibilities are also outlined to ensure that staff and management involved in each review clearly understand the expectations for the project.

process for standard DCs, SDAs, and MLs those in 10 CFR sections 52.47, 52.137, and 52.157, respectively.[6]

Under 10 CFR 50.43(e), applications for a DC, COL, ML, OL, or SDA that propose nuclear reactor designs that differ significantly from light-water reactor (LWR) designs licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, will be approved only if the requirements in subparagraphs (1), (2), or (3) are met. This ISG applies to the preparation and review of applications for class 103 permits, approvals, certifications, and licenses that include a direct reference to a prior DOE or DOW authorization per 10 CFR 50.43(f). The content of this ISG may also be useful for prospective NRC applicants to which 10 CFR 50.43(f) does not apply. This ISG does not alter the overall scope of the information to be included in an application for a class 103 license, approval, certification, or permit.

PAPERWORK REDUCTION ACT

This ISG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Part 50 and 10 CFR Part 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), approval numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC 20503.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

GUIDANCE

1. Overall Approach

The development of applications for NRC licenses, certifications, permits, and approvals is a major undertaking in that applicants must provide sufficient information to demonstrate that they meet applicable regulatory requirements. The information and level of detail required vary according to the design and type of application and license or other type of approval sought (e.g., CP, SDA, DC, ML, OL, COL). The content of the application and the NRC staff's review approach are further defined by the reactor technology, the design/licensing approach selected by the applicant, and the extent to which the applicant proposes to reference a prior DOE or

[6] The NRC staff plans to update, as appropriate, the guidance in this document to reflect the culmination of other rulemaking and regulatory guidance activities in progress, as well as updates related to the identification of any additional opportunities to leverage DOE or DOW authorizations in areas where those authorizations satisfy associated NRC regulatory requirements. Several of these recently completed or in-progress rulemakings create new licensing frameworks (e.g., 10 CFR Part 53, "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors" (Ref. 9)). Although separate efforts will address specific guidance associated with other licensing frameworks, the strategies and approaches discussed herein can also be applied. The NRC will seek to leverage to the maximum extent practical in its own independent reviews information from a DOE or DOW authorization, as justified by an applicant in an application, and will adjust the focus and depth of the review, as appropriate.

DOW authorization of the same or similar design. The design/licensing approaches that are available to NRC applicants are associated with a broad range of reactor designs that generally fall into one of the following three categories:

(1) LWR technologies using a “traditional” deterministic approach

(2) non-LWR technologies:

  (a) using a “traditional” deterministic approach

  (b) using the risk-informed and performance-based approach reflected in NRC 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,” issued June 2020 (Ref. 10)

The available and acceptable approaches in each of these three categories are centered around identifying and thoroughly evaluating the potential risks posed by the proposed facility, while establishing compliance with the associated NRC regulatory requirements.[7]

As described below, the NRC’s existing guidance may be used to review applications that reference a DOE or DOW authorization. An applicant who chooses to do so, when demonstrating compliance with applicable regulations, may credit prior analyses, testing, prototype operation, and operating experience, as appropriate, in its application. Such an applicant may also consider including a discussion of the following in the application: (A) the scope of the prior authorization, including the testing, analyses, and operational experience on which it relied, as applicable; (B) the safety functions and design features addressed by such authorization; and (C) any differences from the prior authorization in design, operational, or siting factors relevant to the application that are material to the Commission's licensing review. The extent to which the NRC can tailor the focus and depth of its review will depend on the information provided in the application, including how the referenced authorization supports and is justified as applicable to NRC requirements. For instance, NRC staff may consider tailoring the focus and depth of its review when (1) the relevant design feature(s) and safety function(s) are materially similar to those previously authorized, and (2) the NRC determines that the prior assessment remains applicable and technically valid to the subject application.

The NRC notes that the extent to which prior DOE or DOW information can be leveraged depends on the information an applicant provides and the associated justification as to how it meets NRC requirements. For example, designs authorized by DOE or DOW may differ substantially from those intended for NRC licensing, or the testing or data generated under those prior authorizations may not fully address the information needed to support a commercial license. In all cases, applicants remain responsible for providing any supplemental information necessary to demonstrate that NRC requirements have been satisfied. As discussed in the Rationale section, developers and industry stakeholders are encouraged to pursue DOE or DOW authorizations with the “end in mind,” ensuring their technical and regulatory analyses align as much as practical with those needed for an NRC application. This alignment enables NRC staff to make maximum use of prior authorization information and supports more efficient

[7] The NRC encourages an LWR applicant proposing to use the risk-informed, performance-based process described in RG 1.233 to engage in preapplication discussions with the NRC staff to provide information on its intended implementation of the methodology for its design.

review of subsequent NRC applications. Applicants may, as an option, provide an equivalency map illustrating the NRC regulations satisfied by prior authorization information to aid in the NRC staff's review. Early and sustained preapplication engagement with the NRC further helps applicants understand where DOE or DOW information overlaps with commercial licensing requirements and where additional information may be needed to address gaps.

2. Light-Water-Reactor Technologies Using a "Traditional" Deterministic Approach

The NRC staff has licensed large LWRs using a standard content-of-application methodology. RG 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," issued November 1978 (Ref. 11), and RG 1.206, "Combined License Applications for Nuclear Power Plants," issued June 2007 (Ref. 12), describe the agency's efforts to standardize the format and content of applications for LWRs. Revised in October 2018, RG 1.206, Revision 1 (Ref. 13), has a new title, "Applications for Nuclear Power Plants," and specific applicability to power reactors using LWR technology, with the clarification that the staff also considers it to be generally applicable to other types of reactors (e.g., non-LWRs). Staff review guidance documents, such as NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 14), support the NRC staff in performing consistent and reliable safety reviews of the subject application. Other documents referenced in the appendices to this ISG provide guidance on specific technical areas and the suggested scope and level of detail for applications developed with this approach.

The implementation of this approach results in a 19-chapter format for the safety analysis report (SAR) portion of the NRC application.

3. Non-Light-Water-Reactor Technologies

Depending on the design attributes of the different non-LWR technologies, the NRC regulations and policies may be addressed differently than for traditional LWRs (Ref. 15). It is the applicant's responsibility to consider NRC regulations and fundamental concepts, such as defense in depth, in the design of their specific facility and to identify and satisfy necessary safety requirements.

3.1 Non-Light-Water-Reactor Technologies Using the Risk-Informed and Performance-Based Approach Reflected in NRC Regulatory Guide 1.233

This category includes non-LWR designs that are based on the risk-informed and performance-based design and licensing approach proposed by industry (Ref. 16) and endorsed by the NRC with specific conditions and restrictions (Ref. 10).

The NRC has previously issued guidance to industry and to the NRC staff to standardize the development of content and the subsequent NRC staff review of non-LWR applications that use this approach. A summary of this guidance appears in the Advanced Reactor Content of Application Project ISG (Ref. 17), and RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications

for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," issued March 2024 (Ref. 18), associated with the Technology-Inclusive Content of Application Project.

The implementation of this approach results in a 12-chapter format for the SAR portion of the NRC application.

3.2 Non-Light-Water-Reactor Technologies Using a "Traditional" Deterministic Approach

This category includes non-LWR designs that are based on "traditional" design and licensing approaches that are primarily deterministic. There is a range of regulatory guidance available that may support one or more of the key aspects of this approach. That guidance is generally topic specific and is therefore reflected in the various appendices to this ISG as summarized in section 1.3.

The implementation of this approach generally results in a 19-chapter format for the SAR portion of the NRC application.

4 Topic-Specific Guidance in All Three Categories

This ISG includes a series of appendices, as listed below, that provide applicants and staff with review guidance for selected NRC licensing topical areas. These appendices are not intended to reflect all of the topics to be addressed in an NRC application because of the broad range of reactor technologies, reactor designs, end uses, and licensing approaches that are within the scope of this ISG. However, these topics generally form much of the foundation for the assessment of reactor facility risk and documentation of the facility's associated safety basis, to be reflected in the SAR portion of a class 103 application. These topics are also anticipated to overlap many of the requirements to receive an authorization to construct and operate under DOE or DOW authorities.

A. Quality Assurance Program
B. Licensing Event Selection and Classification of Structures, Systems, and Components
C. Principal Design Criteria
D. Source Term
E. Consequence Analysis
F. Emergency Preparedness
G. Security
H. Fire Protection Requirements
I. Technical Specifications
J. Site Characterization and Design for External Hazards

Prospective NRC applicants are encouraged to establish early preapplication engagements with the NRC to communicate their plan for addressing these and other key topics associated with the development of a complete NRC application (Ref. 8). This early preapplication engagement, preferably performed in parallel with the DOE or DOW authorization effort preceding the planned follow-on NRC licensing action, is key to facilitating the efficient leveraging of those prior authorizations. Additional information on the preapplication and NRC licensing processes can be found on the NRC's website "Guidance for Prospective Applicants" (https://www.nrc.gov/reactors/new-reactors/advanced/new-app/general-guidance) and in NUREG/BR-0468, "Frequently Asked Questions About License Applications for New Nuclear Power Reactors," issued December 2009 (Ref. 19).

Prospective applicants are also encouraged to consider other NRC application process requirements, including those in 10 CFR Part 2, “Agency Rules of Practice and Procedure” (Ref. 20). Associated information on the NRC’s adjudications (hearings) processes, including opportunities for public involvement, can be found on the agency’s website (https://www.nrc.gov/about-nrc/regulatory/adjudicatory).

IMPLEMENTATION

The NRC staff will use the information discussed in this ISG to support review of class 103 applications for CPs, OLs, COLs, MLs, SDAs, and DCs under 10 CFR Part 50 and 10 CFR Part 52. The staff intends to incorporate this guidance in updated form in an RG or NUREG series, as appropriate.

BACKFITTING, FORWARD FITTING, AND ISSUE FINALITY DISCUSSION

This ISG does not constitute backfitting as defined in 10 CFR 50.109, “Backfitting,” and as described in NRC Management Directive (MD) 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests”; does not constitute forward fitting as that term is defined and described in MD 8.4; and does not affect the issue finality of any approval issued under 10 CFR Part 52. The purpose of this ISG is to assist the NRC staff in its reviews of certain applications. Use of this ISG by entities developing such applications would be voluntary by those entities. Thus, this ISG does not meet the definitions of backfitting in 10 CFR 50.109 or forward fitting in MD 8.4 and does not affect the issue finality of any approval issued under 10 CFR Part 52.

CONGRESSIONAL REVIEW ACT

The Office of Management and Budget has determined that DANU-ISG-2026-02, “NRC Application Pathway for Reactor Designs Previously Authorized by U.S. Department of Energy or Department of War,” is a not a major rule as defined in the Congressional Review Act (5 U.S.C. 804(2)).

FINAL RESOLUTION

The staff expects to transition the information and guidance in this ISG into an RG series or NUREG series document, as appropriate. Following the transition of all pertinent information and guidance in this document into an RG or NUREG, or other appropriate guidance, this ISG will be closed.

APPENDIX A—Quality Assurance Program

NRC Regulatory Requirements: Applicants for licenses, permits, or certifications are required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of applications; technical information" (for applications based on 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 3)), or 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report" (for applications based on 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4)), to describe their quality assurance programs (QAPs) as part of their applications. QAPs must meet the requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," unless an exemption can be justified. The Appendix B excerpt below summarizes the required scope of the QAP of a U.S. Nuclear Regulatory Commission (NRC) applicant:

> Nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, manufacture, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

NRC Application Content Guidance: The scope of the QAP will vary depending on whether the application is for a construction permit (CP), operating license (OL), combined license, design certification, standard design approval, or manufacturing license, and the scope of activities covered by the application. For example, a quality assurance program description (QAPD) for a CP application must cover the design, fabrication, construction, postconstruction testing, and preoperational testing activities. A QAPD for an OL must describe the operational activities, and a QAPD for a combined license must cover the scope of both CP and OL plans.

The NRC staff has endorsed the American Society of Mechanical Engineers standard Nuclear Quality Assurance (NQA)-1-2022,[1] "Quality Assurance Requirements for Nuclear Facility Applications" (Ref. 21), as an acceptable method of meeting 10 CFR Part 50, Appendix B.

The NRC recommends that the QAPD be a standalone document that applicants may submit as a topical report and incorporate by reference into the safety analysis report.

Application Review Guidance for NRC Staff: The staff's review of both light-water reactor (LWR) and non-LWR applications using a traditional deterministic approach will be consistent with the guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 17.5, "Quality Assurance Program Description—Design Certification, Early Site Permit and New License Applicants" (Ref. 14), because QAPDs for non-LWR applications are programmatic and not technology specific.

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[1] The latest edition endorsed by the staff should be reviewed in Regulatory Guide (RG) 1.28, Revision 6, "Quality Assurance Program Criteria (Design and Construction)," issued September 2023 (Ref. 22).

The staff's review of non-LWR applications using the Licensing Modernization Project risk-informed approach will be consistent with RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," issued March 2024 (Ref. 18). For quality assurance associated with non-safety-related structures, systems, and components that are risk significant (i.e., non-safety-related with special treatment (NSRST) classification using the Licensing Modernization Project process), additional staff review guidance in NUREG-0800, Section 17.4, "Reliability Assurance Program (RAP)" (Ref. 14), may be appropriate.

NRC Staff Review Considerations: As indicated above, the NRC staff has endorsed an industry standard as one acceptable method for meeting certain regulations. The NRC staff should leverage prior U.S. Department of Energy (DOE) or U.S. Department of War (DOW) authorizations in its review of the applicant's proposed QAP by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the QAP was developed using the NRC guidance reflected above or is based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

Additional References for Applicant and Staff Consideration: The following additional guidance documents may be useful to an applicant, the NRC staff, or both, but are not intended to be a comprehensive listing of relevant guidance documents. These documents are provided as background that an applicant may find useful in preparing the application:

- RG 1.28, Revision 6, "Quality Assurance Program Criteria (Design and Construction)," issued September 2023 (Ref. 22)

- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," Appendix C, "Construction Permit Guidance," issued March 2024 (Ref. 17)

- RG 1.33, Revision 3, "Quality Assurance Program Requirements (Operation)," issued June 2013 (Ref. 23)

- RG 1.164, Revision 1, "Dedication of Commercial-Grade Items for Use in Nuclear Power Plants," issued April 2024 (Ref. 24)

APPENDIX B—Licensing Event Selection and Classification of Structures, Systems, and Components

NRC Regulatory Requirements: Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of applications; technical information" in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 3), and the "Contents of applications; technical information" sections (10 CFR 52.47, 52.79, 52.137, and 52.157) of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4), an applicant must provide a description and analyses of the structures, systems, and components (SSCs) of the proposed facility, including performance requirements and their bases, and the associated evaluations required to show that safety functions will be accomplished. The U.S. Nuclear Regulatory Commission (NRC) determines applicable regulations from within this group, depending on the approval being pursued (e.g., construction permit, combined license, standard design approval).

NRC Application Content Guidance: The SSCs that must be described in the application are generally identified through two closely related processes: namely, the identification of licensing events (LEs) and the classification of the SSCs that are associated with the reactor facility's response to those events. This appendix describes guidance on these two processes for demonstrating compliance with the applicable regulatory requirements.

The term "licensing event" in this guidance refers to the collection of designated event categories in NRC regulations, such as anticipated operational occurrences, design-basis accidents, design-basis events, beyond-design-basis events, and postulated accidents.

The identification of a comprehensive set of LEs is fundamental to the safe design of commercial nuclear plants. Specifically, the safety of a commercial nuclear plant is demonstrated by analyses of the responses of the plant to LEs, which include postulated disturbances in process variables and postulated malfunctions or failures of equipment. The results of such safety analyses are used to (1) demonstrate compliance with NRC regulations or justify requested exemptions from specific NRC regulations, (2) inform the selection of limiting conditions for operation, limiting safety system settings, and design specifications for SSCs to protect public health and safety, and (3) identify the appropriate scope and depth of information that commercial nuclear plant designers and applicants should provide in applications for permits, licenses, certifications, and approvals. Accordingly, it is essential to identify a comprehensive set of LEs that considers all radiological sources at the plant, all internal and external hazards, and all plant operating states.

NRC regulations provide several regulatory frameworks for commercial nuclear plant licensing, thus giving designers and applicants considerable flexibility while also ensuring an acceptable level of safety. Within these frameworks, an acceptable technology-inclusive approach for identifying commercial nuclear plant LEs should address the following overarching principles:

- Identify application-specific factors (e.g., licensing framework, plant-specific design features, and site characteristics).
- Conduct a systematic and comprehensive search for initiating events.
- Use a systematic process to delineate a comprehensive set of event sequences.

- Group initiating events and event sequences into designated LE categories according to the selected licensing framework.

- Demonstrate that the set of LEs is sufficient.

LEs should be identified objectively and without preconceptions or reliance on predefined lists (such as those in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (Ref. 14); previous applications for permits, licenses, certifications, and approvals; and previous probabilistic risk assessments). The use of a “blank sheet of paper” approach helps to avoid pitfalls including, but not limited to, the following:

- the unwitting or unquestioning carryover of assumptions about plant design or behavior

- the use of predefined lists that are out of date and do not reflect contemporary commercial nuclear plant design or operating experience

In short, the identification of LEs, conducted objectively and without preconceptions or reliance on predefined lists, helps to ensure that the final list of LEs is comprehensive and, hence, that the plant design is appropriately analyzed and demonstrated to be safe based on the comprehensive set of LEs.

The classification of SSCs is also a key consideration in the development of the reactor facility design and the content of the associated NRC application. The SSC classifications resulting from the implementation of the available approaches summarized below assist in identifying the applicable regulatory requirements that align with the respective SSC’s importance to safety, including the establishment of design requirements, quality assurance requirements, availability controls, and monitoring programs.

The sections below summarize acceptable options for the identification of LEs and the related classification of SSCs. In many cases, applicants may be proposing to use a combination of approaches to address specific systems, hazards, and related uncertainties. Prospective NRC applicants are encouraged to identify and submit their proposed process for selection of LEs and classification and treatment of SSCs or indicate that they plan to use a previously approved process. Prospective applicants are also encouraged to submit the anticipated list of LEs and the associated list of safety-related and risk-significant SSCs to improve the NRC staff’s understanding of the design, to support discussions of the preliminary SSC classifications, and to prepare for an efficient and effective application review.

Light-Water Reactor-Technologies Using a Predominantly Deterministic Approach: Newer or differing light-water reactor (LWR) designs may consider the draft guidance in Draft Regulatory Guide (DG)-1413, “Technology-Inclusive Identification of Licensing Events for Commercial Nuclear Plants,” issued October 2024 (Ref. 25), when establishing a process for the comprehensive identification of LEs.

Another source of LE identification considerations for LWR technology designs is NUREG-0800 (Ref. 14), although the listed events are associated with specific industry commercial design types and may not completely or adequately reflect event types to be considered in the “blank paper” approach to the comprehensive assessment of new LWR designs described in DG-1413.

NUREG-0800, Section 3.2.2, “System Quality Group Classification” (Ref. 14), is a source of SSC classification guidance for LWRs. It provides an acceptable and primarily deterministic approach to classifying fluid systems determined to be important to safety. As indicated, the deterministic approach reflected in this reference should be complemented, where appropriate, by applying insights from a design-specific probabilistic risk assessment. Note that the “Review Interfaces” portion of section 3.2.2 summarizes SSC classification guidance for other parts of the facility design, including seismic classification and classification of certain electrical and instrumentation systems that are not pressure retaining.

Applicants implementing the alternative regulatory framework reflected in 10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors” (Ref. 3), to establish SSC special treatments can refer to NRC Regulatory Guide (RG) 1.201, Revision 1, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” issued May 2006 (Ref. 26), for SSC categorization guidance. Additional insights and considerations based on experience and lessons learned from implementing this trial use RG are summarized in the NRC’s periodic review of RG 1.201.

Non-Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach: In this category, various regulatory approaches may be taken to establish a comprehensive set of LEs for the proposed reactor facility. At this time, the considerations and guidance in DG-1413 may offer useful information in establishing such LEs; however, DG-1413 is still in a draft state and should not be understood as a final agency position. This guide provides draft technology-inclusive guidance for identifying initiating events, delineating event sequences, and identifying LEs that can be used to inform the design and licensing bases, as well as the content of applications, for commercial nuclear plants.

Applicants pursuing a maximum hypothetical accident approach to LE identification may consider the guidance in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” issued February 1996 (Ref. 27). This NUREG focuses on establishing the technical and licensing bases for proposed research or test reactor facilities to be licensed under NRC class 104. However, it may provide useful insights and considerations for applicants proposing to use a maximum hypothetical accident approach when pursuing NRC class 103 commercial licenses or permits.

Non-Light-Water-Reactor Technologies Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach: The NRC has previously reviewed and endorsed, with clarifications and points of emphasis, a risk-informed and performance-based approach for informing the licensing basis and determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs. That approach, frequently referred to as the Licensing Modernization Project (LMP), addresses licensing-basis event identification and SSC classification. It was developed through a cost-shared initiative led by nuclear utilities and supported by the U.S. Department of Energy (DOE) and is reflected in Nuclear Energy Institute (NEI) 18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” issued August 2019 (Ref. 16). NRC RG 1.233, Revision 0, “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,” issued June 2020 (Ref. 10), presents the results of the NRC’s review of the approach.

RG 1.253, “Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” issued March 2024 (Ref. 18), contains guidance for the development of associated NRC application content for applicants implementing the LMP approach. RG 1.253 guidance is based on the industry’s content development approach for LMP-based applications, reflected in NEI 21-07, Revision 1, “Technology Inclusive Guidance for Non-Light Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology,” issued February 2022 (Ref. 28). The NRC has also issued companion guidance addressing other parts of applications that follow this approach. DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap,” issued March 2024 (Ref. 17), presents that companion guidance.

Application Review Guidance for NRC Staff: Reviewers should assess and confirm that the applicant has applied the key attributes for event selection and SSC classification within its licensing-basis development approach. In particular, reviewers should confirm that the applicant used a systematic process to delineate a comprehensive set of event sequences, that the related initiating events and resulting event sequences have been appropriately designated within the categories associated with the selected licensing framework, and that the SSCs were evaluated and classified consistent with their identified functions using a structured and well-defined approach.

When considering whether the applicant has provided adequate assurance that the set of LEs is sufficient, reviewers should assess how any changes to the previously authorized and operated reactor facility design may have affected previously established LEs and the listing and classification of SSCs required to be described in the application. These considerations may include the commercial facility’s proposed siting, configuration, end uses, and power output, as well as relevant operating experience from the previously proven evolution of the design.

NRC Staff Review Considerations: The NRC staff should leverage prior DOE or U.S. Department of War (DOW) authorizations in its review of this portion of the application by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or the DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether this portion of the application was developed using the NRC guidance reflected above or is based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

APPENDIX C—Principal Design Criteria

NRC Regulatory Requirements: Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of applications; technical information" in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 3), and the "Contents of applications; technical information" sections (10 CFR 52.47, 52.79, 52.137, and 52.157) of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4), an applicant must provide principal design criteria (PDC) in its application.

In 10 CFR Part 50, the U.S. Nuclear Regulatory Commission (NRC) provides regulations for licensing production and utilization facilities, including these excerpts associated with PDC:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," contains the general design criteria (GDC) that establish the minimum requirements for the PDC for water-cooled nuclear power plants. Appendix A also establishes that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units.

- 10 CFR 50.34(a)(3)(i) requires that an application for a construction permit include the PDC for a proposed facility.

The regulations in 10 CFR Part 52 govern the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities:

- 10 CFR 52.47(a)(3)(i) requires that an application for a design certification include the PDC for a proposed facility.

- 10 CFR 52.79(a)(4)(i) requires that an application for a combined license include the PDC for a proposed facility.

- 10 CFR 52.137(a)(3)(i) requires that an application for a standard design approval include the PDC for a proposed facility.

- 10 CFR 52.157(a) requires that an application for a manufacturing license include the PDC for a proposed facility.

The PDC establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) that are important to safety; that is, as stated in Appendix A to 10 CFR Part 50, SSCs "that provide reasonable assurance that the nuclear power plant can be operated without undue risk to the health and safety of the public." The requirement to propose PDC includes a requirement to address the full scope of PDC described in the regulations; this includes "design, fabrication, construction, testing and performance requirements for structures, systems, and components important to safety" (Ref. 17).

Guidance for the development and review of proposed PDC depends on the reactor technology type and on the licensing-basis development approach used by the applicant. Three acceptable approaches to PDC development are summarized below.

Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach: The GDC in Appendix A to 10 CFR Part 50 are an important part of the NRC's regulatory framework. They establish minimum requirements "...for the [PDC] for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission." The GDC serve as the fundamental criteria for the NRC staff when reviewing the SSCs that make up a water-cooled nuclear power plant design, particularly when assessing the performance of their intended safety functions in postulated design-basis events.

Although conformance with the GDC is a regulatory requirement for water-cooled nuclear power plants similar in design and location to currently operating plants, the regulations are also clear that conformance with only the GDC in Appendix A may not provide a complete or sufficient basis for an applicant's proposed PDC. Appendix A states the following:

> There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

NRC Application Content Guidance: To meet the requirements of 10 CFR Part 50 and 10 CFR Part 52 stated above, applicants may consult the following:

- Section 3.1 of Regulatory Guide (RG) 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," issued November 1978 (Ref. 11), provides guidance on the safety analysis report format and content associated with PDC for applications submitted pursuant to 10 CFR Part 50.

- RG 1.206, Revision 1, "Applications for Nuclear Power Plants," issued October 2018 (Ref. 13), presents guidance on the application and safety analysis report format and content for applications submitted pursuant to 10 CFR Part 52.

Application Review Guidance for NRC Staff: Because previous applications were predominantly for water-cooled reactor technology, the NRC staff has well-established review guidance for this technology type and approach. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 14), contains staff guidance for the review of proposed PDC. NUREG-0800 is organized in 19 chapters with appendices. Previous applications for light-water reactors (LWRs) using the "traditional deterministic approach" have been organized in a similar 19-chapter format, with content addressing the review area of each chapter of NUREG-0800. This content includes the basis for the PDC that are related to the review area for the respective chapter. Future LWR applications using this approach would likely be organized in a similar manner. The NRC staff should refer to the corresponding section of NUREG-0800 for review guidance on the related PDC.

Acceptance Criteria: Similarly, the NRC staff should refer to the corresponding section of NUREG-0800 for acceptance criteria. As a minimum, for each area, the application should include a statement describing how the GDC in Appendix A to 10 CFR Part 50 are addressed.

Non-Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach: As indicated above, the GDC in Appendix A to 10 CFR Part 50 are an important part of the NRC's regulatory framework. However, the GDC are not regulatory requirements for non-LWR technologies. The GDC are intended to provide guidance in establishing the PDC for non-LWRs. The resulting PDC serve as the fundamental criteria for the NRC staff when reviewing the SSCs that make up a nuclear plant design, particularly when assessing the performance of their intended safety functions in design-basis events.

NRC Application Content Guidance: As required by 10 CFR Part 50 and 10 CFR Part 52 and stated above, applicants must provide PDC for the proposed facility. The application should include proposed PDC and their justification in the SSC descriptions. Applicants should consider the concepts of the existing GDC in Appendix A to 10 CFR Part 50 as guidance, as noted in the regulation. One acceptable means of implementing this guidance is through the use of RG 1.232, Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," issued April 2018 (Ref. 29).

RG 1.232 is guidance and, as such, represents only one means for satisfying the regulation. If an applicant elects not to consider RG 1.232 in developing its PDC, the application should contain an explanation of how the chosen PDC adequately address the safety concepts described in 10 CFR Part 50, Appendix A, as it applies to the applicant's specific reactor technology. Notably, several of the existing GDC are technology inclusive (such as criteria 1–5).

Application Review Guidance for NRC Staff: The reviewer should consider RG 1.232, including its key assumptions and clarifications, to determine how the applicant addressed the applicable advanced reactor design criteria listed in appendix A to that RG. In addition, the reviewer should check either appendix B to RG 1.232, which is specific to sodium-cooled fast reactors, or appendix C, which is specific to modular high-temperature gas-cooled reactors, if the design reflects one of those technologies.

The reviewer should evaluate how the applicant identified any unique features not addressed by the criteria in RG 1.232 and confirm that the necessary corresponding PDC have been included. The reviewer should evaluate how the proposed PDC establish appropriate design, fabrication, construction, testing, and performance requirements that provide reasonable assurance that the facility can be operated without undue risk to public health and safety.

Acceptance Criteria: The reviewer should ensure that the application includes sufficient discussion of how the PDC were established. If applicable, the discussion should describe how related GDC in Appendix A to 10 CFR Part 50 or the adapted design criteria in RG 1.232 are addressed. The reviewer should ensure that the set of proposed PDC adequately addresses all issues important to safety and provides reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

Non-Light-Water-Reactor Technologies Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach: Nuclear Energy Institute (NEI) 21-07, Revision 1, "Technology Inclusive Guidance for Non-Light Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology," issued February 2022 (Ref. 28), includes guidance for developing proposed PDC for those aspects of the facility design that focus on design functions and features that are safety related (SR) and those that are non-safety-related with special treatment (NSRST). NEI 21-07 describes an approach for developing a two-tiered structure for PDC, including required functional design criteria and complementary design criteria, as defined in NEI 21-07, which also correlates with design

functions classified as SR and NSRST, respectively. The NRC endorses this approach, with clarifications and additions, to developing Licensing Modernization Project (LMP)-based proposed PDC in its Technology-Inclusive Content of Application Project guidance in RG 1.253, "Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," issued March 2024 (Ref. 18). The NRC also considers this approach to be acceptable for developing proposed PDC for those design functions and features of the facility not informed by LMP but determined to be important to the protection of public health and safety (e.g., for normal operations). Accordingly, each applicant is responsible for identifying the need for additional PDC not informed by the LMP process that, because of the technology, design, or site, are necessary to protect public health and safety.

Note that the NRC has determined that for applicants using the LMP process described in NEI 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," issued August 2019 (Ref. 16), and endorsed with clarifications in RG 1.233, Revision 0, "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," issued June 2020 (Ref. 10), SSCs classified as SR and NSRST are both "important to safety."

NRC Application Content Guidance: In accordance with the requirements of 10 CFR Part 50 and 10 CFR Part 52 stated above, applicants must propose PDC for the proposed facility. The application should include proposed PDC and their justification in the SSC descriptions. The application should explain how the LMP process was used to establish the PDC.

Applicants may adopt alternative approaches to proposing PDC for those aspects of the facility design not informed by the LMP process. If these approaches are based on similar risk-informed, performance-based licensing methodologies, the applicant should provide suitable justification for their use and request any necessary exemptions. Exemptions are necessary if applicants do not propose full-scope PDC, as discussed above; that is, if the PDC do not cover all necessary design, fabrication, construction, testing, and performance requirements for all SSCs important to safety.

Application Review Guidance for NRC Staff: The NRC staff should review the application to understand how the applicant used the approach described in NEI 21-07 to develop proposed PDC for those aspects of the facility design that focus on those SR and NSRST design functions and features.

Acceptance Criteria: The reviewer should ensure that the application includes sufficient discussion of how the PDC were established using the LMP process, as well as any alternative approaches that were used to identify PDC for portions of the facility design not informed by the LMP. The reviewer should ensure that the set of proposed PDC adequately addresses all issues important to safety, including those identified in the applicant's probabilistic risk assessment.

NRC Staff Review Considerations: The NRC staff should leverage prior U.S. Department of Energy (DOE) or U.S. Department of War (DOW) authorizations in its review of the applicant's proposed PDC by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or the DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the

application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the PDC were developed using the NRC guidance reflected above or are based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

APPENDIX D—Source Term

NRC Regulatory Requirements: The U.S. Nuclear Regulatory Commission (NRC) requires reactor facility applicants and licensees to demonstrate that the reactor's engineered safety features will prevent or mitigate a release of radioactive material from the reactor facility for a range of accidents. This radioactive material, referred to as the "source term," is defined in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.2, "Definitions" (Ref. 3), as "the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release."

The quantitative determination of radioactive material that could potentially escape from a reactor facility during normal operation or as a result of an accident and ultimately be released off site plays a critical role in the facility's design and the agency's requirements to protect public health against radiation hazards. Pursuant to 10 CFR 50.34, "Contents of applications; technical information," in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and the "Contents of applications; technical information" sections of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (10 CFR sections 52.47, 52.79, 52.137, and 52.157) (Ref. 4), an applicant must perform an evaluation and analysis of this postulated fission product release to demonstrate that the reactor facility meets the dose criteria established for safety system design as documented in the safety analysis report (SAR). The regulations at 10 CFR 52.17, "Contents of applications; technical information," have a similar requirement for the site SAR for an early site permit. Applicable regulations from within this group are determined by the NRC class 103 licensing or permitting outcome being pursued (e.g., construction permit, combined license, standard design approval). The postulated fission product release is described in a footnote to the aforementioned regulations as follows:

> The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

Pursuant to 10 CFR 100.21, "Non-seismic siting criteria" (Ref. 30), the source term is also one element used to determine site suitability for a new stationary power reactor.

Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach:

NRC Application Content Guidance: For the above demonstration, licensees have historically used accident source terms provided in NRC regulatory guidance. The various regulatory source terms for light-water reactors (LWRs), used to assess the maximum hypothetical accident (MHA) and design-basis accidents, establish and confirm the design basis of the nuclear facility, including items important to safety, providing assurance that the plant design meets the safety criteria stated in the above regulatory safety analysis and siting requirements.

NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000 (Ref. 31), presents representative MHA loss-of-coolant accident (LOCA) source terms for a boiling-water reactor and for a pressurized-water reactor derived from physically based evaluations of LWR severe accidents. 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. RG 1.183 provides information on modeling the radionuclide transport and retention within the containment and other safety-related structures, systems, and components (SSCs) and modeling the release to the environment through containment leakage or other credited pathways. RG 1.183 also provides guidance on modeling non-LOCA design-basis accidents, including topics related to source term, such as estimating radionuclide inventory, modeling fuel or system damage, and crediting safety-related SSCs and natural phenomena to provide radionuclide retention and release mitigation.

An applicant may propose an accident source term that differs in source term parameters (timing, release magnitude, and chemical form) from those in the applicable guidance, based on and justified by design-specific features. Regulatory Position 2 of RG 1.183 provides attributes of an acceptable accident source term.

Developers of light-water small modular reactors (SMRs) may use the accident source term (as applicable to the design) in RG 1.183, Revision 1, issued October 2023 (Ref. 32), or propose a design-specific accident source term.

Although RG 1.183 was originally developed as guidance for licensees operating large LWRs to amend their facility license to implement alternative accident source terms pursuant to 10 CFR 50.67, “Accident source term” (i.e., revise their licensing basis), the RG provides source terms that can be used in new reactor licensing for evolutionary large LWRs and LWR SMRs. RG 1.183, Revision 1 (Ref. 32), is the latest revision. However, use of the earlier RG 1.183 (Ref. 31) is acceptable if the plant fits within its applicability, as noted in the background section of RG 1.183, Revision 1. Moving forward, an LWR applicant should be able to use the source term information in any revision of RG 1.183, if it is justified as applicable to the specific design.

Application Review Guidance for NRC Staff: NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (Ref. 14), including Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors,” provides guidance for the review of SARs for LWR nuclear power plants. It contains specific examples of the various regulatory radiological source terms and provides information on the staff’s RGs, which were developed for LWRs.

With respect to potential review precedents, the NRC staff may also refer to these design-specific source terms developed based on RG 1.183 and previously approved for light-water SMR designs, recognizing that they are specific to those designs and the associated postulated core damage event:

- Safety Evaluation Report for NuScale US600, Chapter 15, “Transient and Accident Analyses,” dated July 23, 2020 (Ref. 33)

- Safety Evaluation Report for NuScale US460, Chapter 15, “Transient and Accident Analyses,” dated May 22, 2025 (Ref. 34)

Non-Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach:

NRC Application Content Guidance: Accident source term characterizations in RG 1.183, in particular the MHA LOCA source terms, are focused on LWRs and are therefore not appropriate for direct application to non-LWRs. However, RG 1.183 includes useful information that may be adapted to aid in design-specific accident source term development for non-LWR technologies. Specifically, Regulatory Position 2 of RG 1.183 states attributes of an acceptable accident

source term. Applicants developing deterministic accident source terms may in general use conservative assumptions to bound the expected plant response to credible accidents, such as for the MHA concept discussed in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 1, Chapter 13, “Accident Analysis” (Ref. 27). When developing the accident source terms, applicants may also use more detailed mechanistic analyses of the plant response for a range of accidents, including severe accidents, as a technical basis for the conservatively biased modeling and assumptions such as in the development of the LWR source terms described in RG 1.183.

Application Review Guidance for NRC Staff: The NRC staff review should assess and confirm that the application adequately describes the accident source terms based on and justified by design-specific features and supported by a defensible technical basis. The source term descriptions should include the initial isotopic inventory, release rates from the source inventory (e.g., fuel), chemical forms, radionuclide transport, radionuclide retention and removal within safety-related SSCs or by natural phenomena, and release rates through pathways to the environment. Regulatory Position 2 of RG 1.183 provides attributes of an acceptable accident source term.

If design-specific mechanistic modeling of accidents is used as a technical basis to develop bounding conservative source terms, the NRC staff should assess and confirm that the application adequately describes the mechanistic source term (MST) modeling, 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 MSTs and specific event sequences to address various systems that contain significant inventories of radioactive material.

In Staff Requirements Memorandum (SRM)-SECY-93-092, “SECY-93-092—Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements,” dated July 30, 1993 (Ref. 35), the Commission approved the NRC staff’s recommendation that source terms for non-LWRs be based on a mechanistic analysis. The acceptability of the applicant’s analysis should be based on the staff’s assurance that the following conditions are met:

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

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

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

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

The staff may also consider the relevant portions of review guidance associated with nonpower reactors in NUREG-1537, Part 2, Chapter 13, “Accident Analyses,” issued February 1996 (Ref. 27)

Non-Light-Water-Reactor Technologies Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach

NRC Application Content Guidance: Non-LWR designs may choose to use the risk-informed and performance-based design and licensing approach proposed by industry in Nuclear Energy Institute (NEI) 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,” issued August 2019 (Ref. 16), and endorsed by the NRC with clarifications in RG 1.233, Revision 0, “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,” issued June 2020 (Ref. 10). The approach in NEI 18-04 is also called the Licensing Modernization Project (LMP) approach.

NRC RG 1.253, “Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” issued March 2024 (Ref. 18), presents the NRC staff’s guidance on using a technology-inclusive content-of-application methodology and the LMP to develop specific portions of the SAR included as part of a non-LWR application. Specifically, this RG endorses, with clarifications and additions, the methodology described in NEI 21-07, Revision 1, “Technology Inclusive Guidance for Non-Light Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology,” issued February 2022 (Ref. 28), as one acceptable method for use in developing certain portions of the SAR for an application for a non-LWR construction permit or operating license under 10 CFR Part 50 or a combined license or design certification under 10 CFR Part 52.

RG 1.253 indicates that the applicant should provide information on event sequence source terms specific to its design used in the licensing-basis event consequence analyses. The guidance states that the source term information should cover all radioactive material inventories and include the type, quantity, and timing of the release of radioactive material from the facility during licensing-basis events. The application should include the use of an MST, consistent with the advanced non-LWR probabilistic risk assessment (PRA) standard definition (see appendix A to NEI 21-07).

RG 1.247 (for Trial Use), “Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities,” issued March 2022 (Ref. 36), endorses, with exceptions for trial use, American Society of Mechanical Engineers/American Nuclear Society RA-S-1.4-2021, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants” (Ref. 37), as one acceptable approach to a PRA for use in regulatory decision-making for non-LWRs. In RG 1.247, Regulatory Position 2.C.1.3.16 describes the MST analysis PRA element, and table A-17 indicates that the NRC staff had no objections to the positions in the non-LWR PRA standard. Applicants may find that the PRA standard provides helpful information pertaining to source term phenomena.

Application Review Guidance for NRC Staff: The NRC staff should assess and confirm that the application adequately describes the MSTs, including the retention of radionuclides by barriers and the transport of radionuclides for all barriers and pathways to the environs. This assessment should be based on the staff’s assurance that the conditions associated with

SRM-SECY-93-092 discussed above have been met. The NRC staff should use the principles on MST associated with SRM-SECY-93-092 and RG 1.183, Regulatory Position 2, to aid in its review of the technical aspects of the application's MST.

Considering that the LMP process described in NEI 18-04 is based in part on the use of PRA, the NRC staff should also use the information in the ASME/ANS RA-S-1.4-2021 MST analysis PRA element as an aid in considering the source term phenomena used to develop the scenario-specific MSTs for the design. RG 1.247 also discusses the objectives and characteristics of the non-LWR PRA MST analysis.

NRC Staff Review Considerations: The NRC staff should leverage prior U.S. Department of Energy (DOE) or U.S. Department of War (DOW) authorizations in its review of the applicant's proposed source terms by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or the DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the source terms were developed using the NRC guidance reflected above or are based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

Additional References for Applicant and Staff Considerations: Additional information on source terms, including background information associated with their application to various reactor technology types, can be found on the "Nuclear Power Reactor Source Term" portion of the NRC's website: https://www.nrc.gov/reactors/new-reactors/advanced/references/nuclear-power-reactor-source-term. This content may offer useful information to an applicant, the NRC staff, or both. The additional guidance documents mentioned on this website provide background that an applicant may find useful in preparing the application.

APPENDIX E—Consequence Analysis

NRC Regulatory Requirements: The evaluation of radiological consequences from postulated accidents is a fundamental component of the licensing basis for nuclear power plants. The U.S. Nuclear Regulatory Commission (NRC) requires applicants for construction permits and combined licenses to demonstrate that the design of the facility ensures adequate protection of public health and safety under both normal and accident conditions. These requirements are codified in Title 10 of the *Code of Federal Regulations* (10 CFR) Parts 50, 52, and 100 and are further described through NRC guidance.

As relevant to the scope of this guidance, an applicant for a new stationary power reactor facility can choose to use the two-step licensing process in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 3), comprising a construction permit based on a preliminary design followed by an operating license, or it may apply for a combined license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4), Subpart C, "Combined Licenses." In either framework, the consequence analysis requirements for the safety analysis report are fundamentally the same. In fact, 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," refers directly to some of the 10 CFR Part 50 requirements (e.g., Appendix A, "General Design Criteria for Nuclear Power Plants," with respect to general design criteria (GDC) for light-water reactors (LWRs), and technically relevant positions of the Three Mile Island (TMI) requirements in 10 CFR 50.34(f)). In addition, the requirements of 10 CFR Part 100, "Reactor Site Criteria" (Ref. 30), apply to applications under 10 CFR Part 50 or 10 CFR Part 52.

10 CFR Part 50 Application Requirements: In 10 CFR Part 50, the NRC provides requirements for licensing production and utilization facilities. For the construction permit phase, 10 CFR 50.34(a)(1)(ii) requires that the preliminary safety analysis report include an analysis and evaluation of the design and performance of structures, systems, and components with respect to the prevention of accidents and the mitigation of their consequences. The regulation in 10 CFR 50.34(a)(1)(ii)(D) requires the applicant to provide an evaluation of the consequences of postulated accidents, including those involving the release of radioactive materials, to demonstrate compliance with dose criteria, and to perform a safety assessment of the site and facility. Specifically, the evaluation must determine the following:

> An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[4] total effective dose equivalent (TEDE).
>
> An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE);

As explained in footnote 4 to the regulation, 25 rem TEDE is a reference value for assessment of postulated reactor accidents to evaluate plant design features that are intended to mitigate the radiological consequences of accidents. This use for the safety analysis does not imply that 25 rem TEDE is an acceptable limit for the public.

As discussed in appendix C to this interim staff guidance (ISG), 10 CFR 50.34(a)(3)(i) requires that the preliminary safety analysis report include the principal design criteria (PDC) for the

facility and states that Appendix A to 10 CFR Part 50 establishes the GDC as minimum requirements for LWRs similar in design and location to plants that have previously been issued construction permits. Appendix A to 10 CFR Part 50 states that the GDC are not required for non-LWRs but provide guidance for non-LWR PDC. In Appendix A to 10 CFR Part 50, GDC 19, "Control room," includes a control room radiological habitability design criterion that requires the provision of adequate radiation protection to permit access and occupancy of the control room under accident conditions. Non-LWR designs may or may not propose a similar control room radiological habitability design criterion as a PDC.

For the operating license application, 10 CFR 50.34(b) states requirements for the final safety analysis report (FSAR). Under 10 CFR 50.34(b)(3), the FSAR is required to include the kinds and quantities of radioactive materials expected to be produced in operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits stated in 10 CFR Part 20, "Standards for Protection Against Radiation" (Ref. 38). Additionally, 10 CFR 50.34(b)(11)(ii) requires an applicant for an operating license for a stationary power reactor to include in the FSAR a safety assessment of the site, as in 10 CFR 50.34(a)(1)(ii).

The regulation in 10 CFR 50.34(f) provides additional TMI-related requirements and states that each applicant for a combined license shall demonstrate compliance with the technically relevant portions of the requirements, except for 10 CFR 50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Additional information appears in appendix B to DANU-ISG-2022-01, "Review of Risk-Informed Technology-Inclusive Advanced Reactor Applications—Roadmap," issued March 2024 (Ref. 17). The TMI-related requirements that refer to accident source term releases and may rely on radiological consequence analysis, if technically relevant, and the related topic areas are as follows:

- 10 CFR 50.34(f)(2)(vii)—postaccident access shielding
- 10 CFR 50.34(f)(2)(viii)—postaccident sampling capability
- 10 CFR 50.34(f)(2)(xvii)—accident monitoring instrumentation
- 10 CFR 50.34(f)(2)(xxvi)—leakage control
- 10 CFR 50.34(f)(2)(xxviii)—control room habitability

The regulation in 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," relates to qualification of safety-related equipment with regard to integrated radiation dose during normal and accident conditions.

For new stationary power reactor sites, 10 CFR Part 100 establishes criteria that must be met to ensure that the site is suitable from a radiological safety perspective. In 10 CFR 100.21, "Non-seismic siting criteria," the NRC requires that the site characteristics, including population density, meteorology, and proximity to population centers, be evaluated in conjunction with the potential radiological consequences of postulated accidents. The analysis must demonstrate that the site can support the implementation of emergency plans and that radiological consequences of postulated accidents meet the values given in 10 CFR 50.34(a)(1) at the exclusion area boundary (EAB) and outer boundary of the low-population zone (LPZ).

The regulation in 10 CFR 50.160, "Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities," is an alternative to emergency planning for small modular reactors, non-LWRs, and nonpower production or utilization facilities. To comply with 10 CFR 50.160, 10 CFR 50.33(g)(2) describes sizing criteria for a plume exposure pathway emergency planning zone (EPZ), which requires a consequence

analysis to aid in the EPZ determination. Applicants otherwise comply with 10 CFR 50.47(b) and 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," with consequence analyses as necessary to demonstrate that habitability criteria for emergency response facilities are met as described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," and NUREG-0696, "Functional Criteria for Emergency Response Facilities," issued February 1981 (Ref. 39).

10 CFR Part 52 Application Requirements: The regulations in 10 CFR Part 52 govern the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities. For combined license applications, 10 CFR 52.79(a)(1)(vi) requires a safety assessment of the site, including consideration of major structures, systems, and components of the facility and site meteorology, to evaluate the offsite radiological consequences at the EAB and LPZ.

Other sections of 10 CFR Part 52 address the contents of applications for other approval types, including early site permits (10 CFR 52.17), design certifications (10 CFR 52.47), standard design approvals (10 CFR 52.137), and manufacturing licenses (10 CFR 52.157). All include similar consequence analysis requirements, in particular the evaluation of dose at the EAB and LPZ in the safety analysis report. For all applications under 10 CFR Part 52, other than those for early site permits, the safety analysis report must include an evaluation of the technically relevant positions of the TMI requirements in 10 CFR 50.34(f), with the exception of 10 CFR 50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v). In addition, as stated in 10 CFR 52.79(a)(17), combined license applicants do not need to address 10 CFR 50.34(f)(2)(xxv), which, in part, includes the requirement for an onsite technical support center.

Applicants should also consider 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions" (Ref. 40), as it relates to environmental assessments of radioactive material releases during normal and accident conditions.

NRC Application Content Guidance: The following sections discuss specific guidance for three potential safety analysis approaches for commercial reactor applications and guidance for EPZ sizing for the alternative emergency preparedness framework, regardless of the safety analysis approach taken.

Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach: NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 14), describes the review of accident analyses as historically used in the licensing of LWR power reactors. The discussion in Chapter 15, "Transient and Accident Analysis," includes the design-basis accident (DBA) consequence analyses, with some additional support for evaluation of control room habitability in Section 6.4, "Control Room Habitability System." In particular, Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," describes the review of consequence analyses that use the guidance in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000 (Ref. 31), for LWR source terms for a new LWR application under 10 CFR Part 50 or 10 CFR Part 52.

RG 1.183, Revision 1, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued October 2023 (Ref. 32), provides regulatory assumptions for calculating more realistic source terms and performing a deterministic

radiological consequence analysis of DBAs. Although the source term information is specific to LWR designs similar to those currently operating, this guide could be useful for radiological consequence analysis for advanced and non-LWR designs. The consequence analysis portions of RG 1.183, Revision 1 (Section C.4, "Dose Calculation Methodology"), are not technology specific.

Non-Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach: NUREG-0800, section 15.0.3, describes the review of DBA consequence analyses for new LWR reactor applications. Although the guidance was developed for LWRs and refers to RG 1.183, it includes information that may be useful for non-LWR applications. Although accident source term guidance is applicable to LWR designs, the regulatory requirements, concepts of the deterministic evaluation, and the radiological consequence methods are not technology specific.

NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," and Part 2, "Standard Review Plan and Acceptance Criteria," issued February 1996 (Ref. 27), describes the review of accident analyses and has historically been used in licensing nonpower research and test reactors. NUREG-1537, Chapter 13, "Accident Analyses," describes the postulated maximum hypothetical accident (MHA) approach and provides guidance for research and test reactor accident and consequence analyses that may be useful to applicants seeking class 103 commercial licenses. The MHA refers to a postulated accident scenario in which the potential consequences are shown to exceed and bound all credible accidents. The specific dose locations and dose acceptance criteria discussed in NUREG-1537 are not applicable to power reactors. The values in the regulatory requirement should be used (e.g., 10 CFR 50.34(a)(1)(ii)(D)).

RG 1.183 may provide useful information for satisfying the radiological dose analysis requirements in 10 CFR Part 50 and 10 CFR Part 52 for new non-LWR applicants. Although the source term information is specific to LWR designs similar to those currently operating, this guide could provide useful information on radiological consequence analysis for non-LWR designs. The consequence analysis portions (section C.4) are not technology specific.

With respect to control room habitability design requirements, RG 1.232, Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," issued April 2018 (Ref. 29), may be used by non-LWR reactor designers, applicants, and licensees to develop PDC for any non-LWR designs, as required by the applicable NRC regulations, for nuclear power plants. This guide describes the NRC's proposed guidance for modifying and supplementing the GDC to develop PDC that address two specific non-LWR design concepts: sodium-cooled fast reactors and modular high-temperature gas-cooled reactors, as well as a more general set of advanced reactor design criteria. As noted in RG 1.232, the advanced reactor design criteria and design certifications for these two designs are not considered to be final or binding in terms of what may eventually be required from a non-LWR applicant. An applicant may decide and be able to justify that a control room habitability criterion is not needed for its facility. In that case, no control room consequence analysis would be performed or expected for NRC review.

Non-Light-Water-Reactor Technologies Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach: Non-LWR designs may choose to use the risk-informed and performance-based design and licensing approach proposed by industry in Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,"

issued August 2019 (Ref. 16), referred to as the Licensing Modernization Project (LMP) approach and endorsed by the NRC with clarifications in RG 1.233, Revision 0, “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,” issued June 2020 (Ref. 10).

As part of the Advanced Reactor Content of Application Project (ARCAP), the NRC staff has described the contents and review of applications using the LMP methodology in ISG, including DANU-ISG-2022-01. This ARCAP ISG includes a general overview of the information that should be included in a non-LWR application and a roadmap for staff review of the application.

RG 1.233 endorses, with clarifications, the methodology described in NEI 18-04. RG 1.253, “Guidance for a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” issued March 2024 (Ref. 18), endorses, with clarifications and additions, NEI 21-07, Revision 1, “Technology Inclusive Guidance for Non-Light Water Reactors: Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology,” issued February 2022 (Ref. 28), regarding application content associated with use of that methodology. As discussed in RG 1.253, section C.2 of NEI 21-07 provides acceptable guidance for the discussion of software and analytical tools used to (1) perform the event sequence modeling and quantification, (2) determine the mechanistic source terms, and (3) perform radiological consequence evaluations for the non-DBA licensing-basis events (LBEs) and the DBAs listed in section C.3 of NEI 21-07, and the cumulative dose and risk calculations in section C.4.1 of NEI 21-07. Applicants should discuss the analysis methods and assumptions for the total calculated radiological consequence dose at the EAB, the outer boundary of the LPZ, and the control room (if required—e.g., if operator actions are relied on for safety-significant functions) to demonstrate that the facility meets the requirements of 10 CFR 50.34(a)(1)(ii)(D) or 10 CFR 52.79(a)(1)(vi) and the PDC for the control room (if applicable). An applicant is free to propose different approaches.

The LMP methodology requires that (1) each identified anticipated operational occurrence, design-basis event, and beyond-design-basis event that releases radioactive material to the environment be realistically analyzed to determine its 30-day cumulative TEDE, plus uncertainties, at the EAB to evaluate against a frequency-consequence target curve, (2) DBAs derived from the LBEs be evaluated deterministically with conservative assumptions for comparison with the offsite dose criteria at the EAB and LPZ, (3) the integrated risk from all events considered in the probabilistic risk assessment (PRA) be evaluated against integrated risk targets based on the Commission’s Quantitative Health Objectives (i.e., the annual average individual risk of early and latent fatalities), reported as mean risk without uncertainty, and (4) the cumulative frequency of exceeding 100 millirem per plant-year at the EAB be less than 1.0/plant-year, reported as mean risk without uncertainty. Applicants also need to demonstrate compliance with the dose requirements discussed above in the regulatory requirements section. RG 1.253, section C.3, discusses the LMP approach for consequence analysis. In addition, RG 1.183 may provide generally useful information on radiological consequence analysis for non-LWR designs. RG 1.183, section C.4, is not technology specific and therefore provides guidance for non-LWR dose calculations.

New reactor applications implementing the 10 CFR 50.160 emergency preparedness framework should use the guidance in RG 1.242, Revision 0, “Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities,” issued November 2023 (Ref. 41), which identifies methods

and procedures that the NRC staff considers acceptable for use by applicants and licensees for small modular reactors, non-LWRs), and nonpower production or utilization facilities to demonstrate compliance with performance-based emergency preparedness requirements in 10 CFR 50.160. RG 1.242, appendix A, describes an acceptable approach for EPZ determinations that includes consideration of the use of consequence analyses developed to support other portions of the licensing basis.

Application Review Guidance for NRC Staff: NUREG-0800, chapter 15, section 15.0.3, provides guidance to the NRC staff for reviewing deterministic radiological consequence analyses for LWRs and non-LWRs.

For the methodology and assumptions for calculating the radiological consequences of postulated accidents, the staff will use, where applicable, the regulatory positions stated in RG 1.183 (specifically, section C.4, which is technology inclusive). The reviewer should confirm that the applicant's approach is clearly documented; should review the associated codes and methods that were applied, including input and outputs; and should consider performing scoping calculations.

In the deterministic approach, to model environmental transport of the airborne radiological release, the MHA and DBA consequence analyses use site characteristic or bounding atmospheric dispersion factors (χ/Qs). For design certifications or standard design approvals, site parameter χ/Q values are used in lieu of site information. The DBAs in the LMP approach are treated the same as DBAs in the traditional deterministic approach. The review of the χ/Qs is coordinated with the meteorology review staff.

For non-LWR LMP applications, the staff will also consider the information in RG 1.247 (for Trial Use), "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities," issued March 2022 (Ref. 36), which endorses, with conditions, the non-LWR PRA standard American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants" (Ref. 37). RG 1.247, section C.1.3.17, provides guidance on the radiological consequence analysis PRA element, while table A-18 provides the specific staff position on the ASME/ANS RA-S-1.4-2021 technical requirements for radiological consequence analysis.

In the LMP approach, consequence analyses for the non-DBA LBEs are more realistic and include quantification of uncertainty for the comparison to the frequency-consequence target and evaluation of integrated risk. The applicant may choose to use an accident consequence computational tool developed for level 3 PRA such as the MELCOR Accident Consequence Code System (MACCS) for this purpose. MACCS requires 1 year of meteorological data, which should be representative of the site. The review of the meteorological data and modeling of the atmospheric transport and dispersion should be coordinated with the meteorology review staff and the MACCS subject matter experts in the Office of Nuclear Regulatory Research, as necessary.

The LMP approach also includes quantification of the integrated plant risk. The analyses should calculate (1) the total mean frequency of exceeding a site boundary dose of 100 millirem from all LBEs, (2) the average individual risk of early fatality within 1 mile of the EAB from all LBEs based on mean estimates of frequencies and consequences, and (3) the average individual risk of latent cancer fatalities within 10 miles of the EAB from all LBEs based on mean estimates of frequencies and consequences.

NRC Staff Review Considerations: The NRC staff should leverage prior U.S. Department of Energy (DOE) or U.S. Department of War (DOW) authorizations in its review of the applicant's evaluation of postulated radiological consequences by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the evaluation was developed using the NRC guidance reflected above or is based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

APPENDIX F—Emergency Preparedness

NRC Regulatory Requirements: The U.S. Nuclear Regulatory Commission's (NRC's) emergency preparedness (EP) framework ensures that no initial operating license for a nuclear power reactor or combined license is issued until the NRC finds that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. These requirements are codified in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.47(a)(1) (Ref. 3).

NRC Regulatory Frameworks: There are two EP regulatory frameworks:

(1) 10 CFR 50.47, "Emergency plans," and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities"

(2) 10 CFR 50.160, "Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities"

The requirements in 10 CFR Part 50, Appendix E, and, for power reactors, the planning standards of 10 CFR 50.47 were originally developed for the hazards and emergency planning needs of large light-water reactors (LWRs). Applicants must comply with 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50 or, if justified, may seek exemptions from certain requirements based on differences in design characteristics that significantly reduce radiological risks compared to those of large LWRs. Further, the provisions in 10 CFR 50.33(g)(1) and 10 CFR 50.47(c)(2) provide for scaling of the emergency planning zone (EPZ) on a case-by-case basis for reactors with authorized power levels of less than 250 megawatts thermal.

Applicants for a small modular reactor, as defined in 10 CFR 50.2, may use the EP framework in 10 CFR 50.160. In addition, non-LWR applicants, including microreactor applicants or nonpower reactor applicants, may choose to use 10 CFR 50.160 and the provisions in 10 CFR 50.33(g)(2) for a scalable EPZ. The regulations in 10 CFR 50.47(a) describe the findings that the NRC must make before issuing an operating license under 10 CFR Part 50 or a combined license or early site permit (ESP) under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4). The regulations in 10 CFR 50.47(a) apply to an application submitted under 10 CFR Part 52 for an ESP in which an applicant proposes complete and integrated emergency plans or major portions of emergency plans for review.

NRC Application Content Guidance: Applicants must submit detailed emergency plans as part of their applications. Consistent with the scope of this guidance, these plans must comply with the following:

- the 16 planning standards in 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50; or the requirements in 10 CFR 50.160
- 10 CFR Part 52 – Applies to reactor licensing (i.e., combined licenses and ESPs), which incorporates EP requirements by reference to 10 CFR Part 50, with additional provisions for ESPs with options for submitting "major features" or "complete and integrated" plans

New reactor applications implementing the 10 CFR 50.160 EP framework should use the guidance in Regulatory Guide (RG) 1.242, Revision 0, "Performance-Based Emergency

Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," issued November 2023 (Ref. 41), which identifies methods and procedures that the NRC staff considers acceptable for use by applicants and licensees for small modular reactors, non-LWRs, and nonpower production or utilization facilities to demonstrate compliance with performance-based EP requirements.

Application Review Guidance for NRC Staff: Section 13.3, "Emergency Planning," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 14), contains current staff review guidance associated with EP. This guidance establishes criteria that the NRC staff uses in evaluating whether an applicant or licensee adequately meets NRC regulations to construct and operate nuclear power plants.

NRC Staff Review Considerations: The NRC staff should leverage prior U.S. Department of Energy (DOE) or U.S. Department of War (DOW) authorizations in its review of the applicant's proposed emergency plans by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the emergency plans were developed using the NRC guidance reflected above or are based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

Additional References for Applicant and Staff Consideration: The NRC staff also uses several other key documents to evaluate EP submissions. These include but are not limited to the following:

- RG 1.206, Revision 0, "Combined License Applications for Nuclear Power Plants: LWR Edition," issued June 2007, and Revision 1, "Applications for Nuclear Power Plants," issued October 2018, offer application preparation guidance, including application content guidance for EP (Refs. 9 and 10)
- NUREG-0654/FEMA-REP-1, Revision 2, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued December 2019 (Ref. 42) (joint NRC/Federal Emergency Management Agency document), outlines an acceptable method for meeting EP planning standards
- RG 1.242, Revision 0, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities," issued November 2023 (Ref. 41), provides an acceptable method for meeting the EP requirements in 10 CFR 50.160
- DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap," issued March 2024 (Ref. 17), includes a general overview of the information that should be included in a non-LWR application and a roadmap for staff review of the application

Additional EP guidance documents are available on the NRC public webpage: https://www.nrc.gov/about-nrc/emerg-preparedness/regs-guidance-comm.

APPENDIX G—Security

NRC Regulatory Requirements: All commercial nuclear power plants in the United States are currently licensed under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 3), or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4). Applicants must include the required security plans in their applications consistent with the regulations in 10 CFR 50.34(c)–(d), and 10 CFR 52.79(a)(35)–(36). These regulatory provisions collectively establish the security plan content that must be included as part of a complete application. In 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage" (Ref. 43), the U.S. Nuclear Regulatory Commission (NRC) specifies the requirements for physical protection programs that apply to nuclear power plant applicants and licensees.

A nuclear power plant applicant or licensee may seek an exemption from, or an alternative measure to, one or more of the physical security requirements in 10 CFR 73.55. The processes for doing so are described in 10 CFR 73.5, "Specific exemptions," and 10 CFR 73.55(r). In either case, the applicant or licensee must obtain prior NRC approval.

NRC Application Content Guidance: The following regulatory guides (RGs) describe an approach that is acceptable to the NRC staff to meet the regulatory requirements for security, addressing various requirements in 10 CFR Part 73, "Physical Protection of Plants and Materials":

- RG 5.59, Revision 1, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance," issued February 1983 (Ref. 44)
- RG 5.69, Revision 1, "Guidance for the Application of the Radiological Sabotage Design-Basis Threat in the Design, Development, and Implementation of a Physical Security Program that Meets 10 CFR 73.55 Requirements," issued March 2022 (Ref. 45)
- RG 5.76, Revision 1, "Physical Protection Programs at Nuclear Power Reactors," issued November 2020 (Ref. 46)
- RG 5.71, Revision 1, "Cyber Security Programs for Nuclear Facilities," issued February 2023 (Ref. 47), which describes an approach that is acceptable to the NRC staff to meet the regulatory requirements for cybersecurity in 10 CFR 73.54, "Protection of digital computer and communication systems and networks"

Application Review Guidance for NRC Staff: NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 13.6, "Physical Security" (Ref. 14), provides current staff review guidance associated with physical security. Section 13.6 establishes criteria that the NRC staff uses in evaluating whether an applicant or licensee meets NRC regulations to construct and operate nuclear power plants. The NRC staff should review a design certification application for security elements that are outlined in NUREG-0800, Section 13.6.2, Revision 2, "Physical Security—Review of Physical Security System Designs—Standard Design Certification and Operating Reactor Licensing Applications," issued June 2015 (Ref. 14). The staff may consider an application that does not contain these security elements if one or both of the following criteria apply: (1) physical security systems or functions will not be located within the nuclear island or structures, or (2) physical

security systems will not be integral to the construction of the nuclear island and structures (e.g., independent of building structure, portable). For example, a nuclear island may not exist for some microreactors depending on the reactor design. The physical security program is required before fuel is allowed on site. In addition, the staff should review guidance in NUREG-0800, Section 13.6.6, “Cyber Security Plan,” when evaluating the applicant’s cybersecurity plan.

NRC Staff Review Considerations: The NRC staff should leverage prior U.S. Department of Energy (DOE) or U.S. Department of War (DOW) authorizations in its review of the applicant’s proposed security plans by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or DOW as part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the security plans were developed using the NRC guidance reflected above or are based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

Additional References for Applicant and Staff Consideration: The following additional guidance documents may provide useful information to an applicant, the NRC staff, or both:

- DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications—Roadmap,” issued March 2024 (Ref. 17)
- RG 5.81, Revision 1, “Target Set Identification and Development for Nuclear Power Reactors,” issued July 2024 (Ref. 48)
- RG 5.54, Revision 1, “Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants,” issued June 2009 (Ref. 49)
- RG 5.66, Revision 2, “Access Authorization Program for Nuclear Power Plants,” issued October 2011 (Ref. 50)
- RG 5.75, Revision 1, “Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities,” issued March 2021 (Ref. 51)
- RG 5.77, Revision 1, “Insider Mitigation Program,” issued September 2022 (Ref. 52)
- RG 5.79, “Protection of Safeguards Information,” issued April 2011 (Ref. 53)

APPENDIX H—Fire Protection Requirements

NRC Regulatory Requirements: U.S. Nuclear Regulatory Commission (NRC) requirements for fire protection programs (FPPs) aim to minimize both the probability of occurrence and the consequences of fire. To meet these objectives, the FPPs are designed to provide reasonable assurance, through defense in depth, that a fire will not prevent the necessary safe-shutdown functions from being performed and that radioactive releases to the environment in the event of a fire will be minimized.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.48, "Fire protection" (Ref. 3), holders of a power reactor operating license under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," or a combined operating license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4), must have a fire protection plan that satisfies General Design Criterion 3, "Fire protection," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. The fire protection plan must describe the overall FPP for the facility and outline the programs for fire protection, automatic fire detection and suppression capability, and limitation of fire damage. The fire protection plan must also describe specific features necessary to implement the program, such as administrative controls (e.g., policies and procedures) and personnel requirements for fire prevention and manual fire suppression activities, and the means to limit fire damage to structures, systems, and components important to safety so that the capability to safely shut down the reactor is ensured.

The following sections discuss specific guidance for three categories of commercial reactor applications.

Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach:
Regulatory Guide (RG) 1.189, Revision 5, "Fire Protection for Nuclear Power Plants," issued October 2023 (Ref. 54), describes an approach acceptable to the NRC staff to meet the requirements of 10 CFR 50.48(a). RG 1.189 provides fire protection guidance that identifies the scope and depth of fire protection that the NRC staff would consider as one acceptable way for nuclear power plants to meet fire protection regulations. The RG provides guidance to licensees for preparing adequate content and quality of engineering equivalency evaluations used to support the FPP.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.5.1.1, "Fire Protection Program" (Ref. 14), in conjunction with RG 1.189, provides guidance for assessing FPPs for light-water reactor (LWR) designs using a deterministic licensing approach.

Non-Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach:
Non-LWR applicants that are not using a risk-informed, performance-based approach are subject to the same regulation, 10 CFR 50.48(a), which requires that each operating light-water nuclear power plant has a fire protection plan that satisfies General Design Criterion 3 in Appendix A to 10 CFR Part 50. Advanced Reactor Design Criterion 3 in RG 1.232, Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," issued April 2018 (Ref. 29), provides the corresponding guidance for non-LWR technologies using this approach.

While the scope of DANU-ISG-2022-09, "Risk-Informed, Performance-Based Fire Protection Program (for Operations)," issued March 2024 (Ref. 55), is limited to risk-informed,

performance-based applications, the staff encourages applicants to leverage existing guidance to inform their FPPs and, where applicable, to detail in their applications exceptions to such guidance with associated justifications. As discussed in DANU-ISG-2022-09, relevant guidance and industry standards include, but are not limited to, RG 1.189 (Ref. 54); National Fire Protection Association (NFPA) 804, "Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants" (Ref. 56); NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" (Ref. 57); and NFPA 806, "Performance-Based Standard for Fire Protection for Advanced Nuclear Reactor Electric Generating Plants Change Process" (Ref. 58). Moreover, the application guidance for administrative programs during operations and the associated NRC staff acceptance criteria found in DANU-ISG-2022-09 are generally applicable to non-LWR applications using a deterministic approach. Applicants are encouraged to discuss applicability to their design and licensing approach during preapplication engagements. When the application is not expected to include a probabilistic risk assessment, discussions with staff should include those aspects of the FPP that rely on quantitative results (e.g., use of risk insights to inform FPP change process and surveillance requirements).

Non-Light-Water-Reactor Technologies Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach: DANU-ISG-2022-09 addresses development of an advanced reactor application for an operating license under 10 CFR Part 50 or a combined license under 10 CFR Part 52 associated with a proposed risk-informed, performance-based FPP for operations. Although 10 CFR 50.48(c) is not applicable to non-LWRs, elements and concepts in NFPA 805; the guidance in RG 1.205, Revision 2, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," issued May 2021 (Ref. 59); and the guidance in RG 1.189 can be applied to reactor applications, with justified exceptions or deviations.

NRC Staff Review Considerations: The NRC staff should leverage prior U.S. Department of Energy (DOE) or U.S. Department of War (DOW) authorizations in its review of the applicant's proposed fire protection plan by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or the DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the fire protection plan was developed using the NRC guidance reflected above or is based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

APPENDIX I—Technical Specifications

NRC Regulatory Requirements: All applications for a license to operate a production or utilization facility, including applications for an operating license under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 3), or a combined license (COL) under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4), must include technical specifications (TS). The regulatory requirements for the scope and content of the TS are primarily located in 10 CFR 50.36, "Technical specifications." Specifically, 10 CFR 50.36(c) states that the TS must include the following five sections:

(1) safety limits, limiting safety system settings, and limiting control settings
(2) limiting conditions for operation
(3) surveillance requirements
(4) design features
(5) administrative controls

Also, 10 CFR 50.36 requires, in part, that the application include a summary statement of the bases or reasons for the TS, other than those covering administrative controls. In addition, the TS should be consistent with the applicant's principal design criteria in that all safety-related features specified in the principal design criteria should be addressed in the TS. For applicants pursuing licensing under 10 CFR Part 52, 10 CFR 52.47(a)(11) and 10 CFR 52.79(a)(30) require design certification (DC) and COL applicants, respectively, to propose TS prepared in accordance with 10 CFR 50.36.

NRC Application Content Guidance: The primary requirements for TS are found in 10 CFR 50.36, with additional requirements in 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors." In response to the "Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Plants," dated July 22, 1993 (Ref. 60), the U.S. Nuclear Regulatory Commission (NRC) issued Standard Technical Specifications (STS) as NUREG-series publications for various reactor designs in operation at the time. For applications referencing a reactor design with an existing STS, the plant-specific TS—developed from the proposed safety analysis report (SAR)—should follow the format and align with the content of the most recent applicable STS version. For applications referencing a reactor design with an existing STS, the plant-specific TS developed from the proposed SAR should follow the format and align with the content of the most recent applicable STS version. For applications for which an STS does not exist, applicants must comply with applicable regulations and, where appropriate, are highly encouraged to generally follow the format and structure of existing STS documents. Applicants in this situation are encouraged to engage with the staff during pre-application meetings to discuss their proposed TS.

The Commission, in its policy statement, emphasized that licensees should use plant-specific probabilistic risk assessments, risk surveys, and relevant risk insights from available literature when developing their TS. Likewise, the NRC staff will apply risk insights and probabilistic risk assessment results when reviewing TS-related submittals. Guidance supporting risk-informed decision-making for TS development and modifications includes the following:

- Regulatory Guide (RG) 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued January 2018 (Ref. 61), describes a general approach to developing

risk-informed regulatory applications for licensing-basis changes and discusses specific topics common to risk-informed regulatory applications.

- RG 1.177, Revision 2, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," issued January 2021 (Ref. 62), builds on the concepts of RG 1.174 but focuses on the use of risk analysis to evaluate proposed changes to TS. Its guidance is also useful in evaluating certain aspects of initial TS development.

For 10 CFR Part 50 applications, Chapter 16, "Technical Specifications," in RG 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," issued November 1978 (Ref. 11), describes the level of detail that should be included in a preliminary safety analysis report in support of a construction permit application. Although this RG is specific to light-water reactors (LWRs), its format and content guidance is useful for the development of the SAR for other technology types using a predominantly deterministic approach. Preliminary TS included in the construction permit application should identify and justify the selection of variables, conditions, or other limitations that are determined to be probable subjects of the final TS. TS submitted in support of an operating license application should be the final version of those specifications originally included in the preliminary safety analysis. The numerical values, graphs, tables, and other data should reflect the final refinements in design, results of tests or experiments, and expected method of operation.

For 10 CFR Part 52 applications, Regulatory Position C.1.4 in RG 1.206, Revision 1, "Applications for Nuclear Power Plants," issued October 2018 (Ref. 13), discusses TS for DC and COL applications. While this interim staff guidance (ISG) is intended for applications (i.e., for a construction permit, operating license, manufacturing license, or COL), it is assumed that an application for a COL will either reference a DC or contain all required information in the final SAR. The following guidance applies to these two scenarios, respectively:

(1) A COL application referencing a DC should include proposed plant-specific TS and their bases. The plant-specific TS should be derived from the analyses and evaluations included in the proposed SAR. The plant-specific TS should encompass the generic TS and bases of the certified design, the departures from the DC final SAR, and site-specific information included in the proposed SAR.

(2) A COL application not referencing a DC should include proposed plant-specific TS and their associated bases as required by 10 CFR 50.36 and 10 CFR 50.36a. The plant-specific TS should be derived from the analyses and evaluations included in the proposed SAR and should conform to the format and be consistent with the content of the most recent version of the STS (if applicable)[1] appropriate to the reactor plant design proposed in the COL application.

The following sections discuss specific guidance for three categories of commercial reactor applications.

Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach: LWR applicants may leverage the format and content of STS, as appropriate. While the STS are vendor and design specific, the NRC staff anticipates that prospective LWR designs will closely resemble either passive or active LWR designs currently covered by STS. The staff also

[1] As noted above, STS are not available for all technologies.

acknowledges various considerations that may influence the extent to which STS for previous design generations can be applied (e.g., smaller power output, use of advanced fuels, diverse operational modes, including nonpower generation). These considerations should be thoroughly evaluated and the results reflected in the proposed TS, where applicable, based on the above regulatory requirements. Applicants are encouraged to discuss during preapplication engagements those variables, conditions, or other limitations that are determined to be probable subjects of their proposed TS and the applicability of STS.

Non-Light-Water-Reactor Technologies Using a Predominantly Deterministic Approach: Non-LWR applicants may also leverage the format and content of STS, as appropriate. As mentioned above, RG 1.70, chapter 16, may be useful for applicants in determining the level of detail included in the proposed TS. This content may be scaled to account for features of the design of non-LWRs such as passive safety systems. Additionally, guidance from RG 1.206 may be useful to applicants identifying proposed plant-specific TS and their derivation from analyses and evaluations of the plant-specific design.

Non-Light-Water-Reactor Technologies Using a Risk-Informed and Performance-Based (Licensing Modernization Project) Approach: DANU-ISG-2022-08, "Risk-Informed Technical Specifications," issued March 2024 (Ref. 63), describes the development of the portion of a non-LWR application associated with an applicant's proposed TS, as well as NRC staff guidance on how to review such an application. To provide suitable guidance on risk-informed TS for advanced reactors, that ISG correlates the requirements in 10 CFR 50.36 with the analysis and outputs of the risk-informed approach described in Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," issued August 2019 (Ref. 16).

RG 1.233, Revision 0, "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," issued June 2020 (Ref. 10), endorses, with clarifications, NEI 18-04 and provides a process by which the content of applications will permit understanding of the system designs and their relationship to safety evaluations for a variety of non-LWR designs. NEI 18-04 specifies special treatments, including proposed TS, to address programmatic defense-in-depth attributes.

Application Review Guidance for NRC Staff: NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 16, "Technical Specifications" (Ref. 14), describes the responsibilities for the NRC staff's review of TS. Although this NUREG series document is specific to LWRs, its guidance is useful for reviewing applications for other technology types using a predominantly deterministic approach.

DANU-ISG-2022-08 provides staff review guidance for evaluating an application and reaching a safety finding regarding the proposed TS.

NRC Staff Review Considerations: The NRC staff should leverage prior U.S. Department of Energy (DOE) or U.S. Department of War (DOW) authorizations in its review of the applicant's proposed TS by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory

requirements and whether the TS were developed using the NRC guidance reflected above or are based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.

APPENDIX J—Site Characterization and Design for External Hazards

NRC Regulatory Requirements: The U.S. Nuclear Regulatory Commission (NRC) requirements for site characterization and design for external hazards aim to ensure that the structures, systems, and components (SSCs) relied on to perform functions required for the prevention or mitigation accidents can perform those functions during and following a design-basis external hazard event. External hazard events are different from other events in the accident analysis in that they can cause the failure of many SSCs with a single initiating event. Current NRC regulations and guidance focus on ensuring that the SSCs relied on in the safety analysis are designed to withstand design-basis hazard levels (DBHLs). These DBHLs are established based on characterization of the site in the areas of human-induced external hazards, climatology, meteorology, atmospheric dispersion, hydrology, geology, seismology, and geotechnical engineering. The parameters needed to assess radionuclide transport by air and ground water are generated through the atmospheric dispersion and hydrology site characterization, respectively.

The regulatory requirements related to site characterization for commercial nuclear power plants licensed under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 3), or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4), are codified in 10 CFR Part 100, "Reactor Site Criteria" (Ref. 30).

Regulatory requirements for SSCs to be designed to withstand external hazards are codified in 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design bases for protection against natural phenomena," and 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

Regulatory requirements for including the site characterization information and design bases to address external hazards in both the preliminary and final safety analysis reports are codified in 10 CFR 50.34(a), 10 CFR 50.34(b), and 10 CFR 52.79(a).

NRC Application Content Guidance: The following regulatory guides (RGs) have been issued in each external hazard content area to provide applicants with information on the methods that the NRC has found acceptable for meeting regulatory requirements related to site characterization.

Guidance Related to Meteorology

- RG 1.23, Revision 1, "Meteorological Monitoring Programs for Nuclear Power Plants," issued March 2007 (Ref. 64), presents criteria for an acceptable onsite meteorological monitoring program and the resulting meteorological database that may be used as input to the atmospheric dispersion estimates.

- RG 1.111, Revision 1, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," issued July 1977 (Ref. 65) describes models and assumptions for the estimation of atmospheric dispersion of gaseous effluent releases.

- RG 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," issued February 1983 (Ref. 66), presents criteria for characterizing atmospheric dispersion conditions for evaluating the

consequences of radiological releases to the exclusion area boundary and low-population zone.

- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” issued June 2003 (Ref. 67), describes an acceptable method to meet the regulations concerning the determination of atmospheric relative concentration (χ/Q) values in support of design-basis control room habitability assessments at nuclear power plants.

- RG 1.249, “Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors,” issued August 2023 (Ref. 68), provides guidance for determining χ/Q values, also known as atmospheric dispersion factors, in support of modeling onsite releases to offsite boundaries from a design-basis accident. The guidance includes procedures for using the ARCON code to estimate χ/Qs at the exclusion area boundary and the outer boundary of the low-population zone out to distances of 1,200 meters (3,937 feet) from the nearest edge of a building within the nuclear island.

- RG 1.76, Revision 1, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” issued March 2007 (Ref. 69), describes the method that the NRC staff considers acceptable for use in selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear power plant should be designed to withstand to prevent undue risk to public health and safety within the contiguous United States.

- RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” issued October 2011 (Ref. 70) which provides guidance that the NRC staff considers acceptable for use in selecting the design-basis hurricane windspeed and hurricane generated missiles that a new nuclear power plant should be designed to withstand.

Guidance Related to Hydrology

- RG 1.27, Revision 3, “Ultimate Heat Sink for Nuclear Power Plants,” issued November 2015 (Ref. 71), describes methods and procedures to establish ultimate heat sink features of plant systems required by NRC regulations.

- Draft Regulatory Guide (DG)-1290, Revision 1 (proposed Revision 3 to RG 1.59), “Design-Basis Floods for Nuclear Power Plants,” issued July 2024 (Ref. 72), describes acceptable methods for determination of design-basis floods for nuclear power plants.

- RG 1.102, Revision 1, “Flood Protection for Nuclear Power Plants,” issued September 1976 (Ref. 73), describes the types of flood protection that the NRC staff finds acceptable for the SSCs important to safety identified in RG 1.29, “Seismic Design Classification for Nuclear Power Plants.” This guide also describes acceptable methods for protecting nuclear power plants from flooding effects resulting from probable maximum precipitation.

- DG-1417 (proposed RG 1.256), “Guidance for Assessment of Flooding Hazards Due to Water Control Structure Failures and Incidents,” issued July 2024 (Ref. 74), provides guidance on acceptable methods for evaluating design-basis flooding hazards due to failure or other incidents at manmade water control structures including, but not limited to, dams and levees.

Guidance Related to Geology, Seismology, and Geotechnical Engineering

- RG 4.26, Revision 1, “Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites,” issued August 2023 (Ref. 75), provides guidance for facilitating the NRC staff’s review of volcanic hazards assessments performed by applicants to support the siting of new nuclear power reactors. The RG also provides applicants with methods and approaches that the NRC staff considers acceptable for the assessment of volcanic hazards in applications.

- RG 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” issued March 2007 (Ref. 76), provides guidance on seismic site characterization and contents of applications for new nuclear power plants.

- RG 1.132, Revision 3, “Geologic and Geotechnical Site Characterization Investigations for Nuclear Power Plants,” issued December 2021 (Ref. 77), provides guidance on field investigations for determining the geologic, geotechnical, geophysical, and hydrogeologic characteristics of a prospective site for engineering analysis and design of nuclear power plants.

- RG 1.138, Revision 3, “Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants,” issued December 2014 (Ref. 78), provides guidance on sampling, storage, and laboratory investigations of the properties of soils for engineering analysis and design of nuclear power plants.

- RG 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites,” issued November 2003 (Ref. 79), provides guidance to applicants on acceptable methods for evaluating the potential for earthquake-induced instability of soils resulting from liquefaction and strength degradation, including the conditions under which the potential for such response should be addressed in safety analysis reports and the procedures and criteria currently applied to assess the liquefaction potential of soils ranging from gravel to clays.

Application Review Guidance for NRC Staff: NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)” (Ref. 14), provides guidance to the NRC staff in performing safety reviews of construction permit or operating license applications. More specifically, chapter 2 contains staff guidance for evaluating site characteristics, and chapter 3 contains staff guidance for evaluating the design of SSCs to withstand external hazards. This guidance can be referenced when assessing applications for either light-water reactor (LWR) or non-LWR technologies that are based on a “traditional” deterministic approach. DANU-ISG-2022-02, “Advanced Reactor Content of Application Project,” Chapter 2, “Site Information,” issued March 2024 (Ref. 80), presents application content and review guidance in the area of site information for advanced reactors pursuing a risk-informed and performance-based approach (Ref. 10).

NRC Staff Review Considerations: Many topical areas are associated with site characterization and design for external hazards. The NRC staff notes that the evaluations provided by the applicant are heavily site dependent and may present challenges to optimizing review efficiency. The more the site selected in the NRC application differs from the site associated with the U.S. Department of Energy (DOE) or the U.S. Department of War (DOW) authorization, the less likely it is that information from the authorization will be applicable to the NRC application. However, the NRC staff should leverage prior DOE or DOW authorizations in

its review of the applicant's proposed site characterization and design for external hazards by considering prior DOE or DOW authorization content to the maximum extent practical. The NRC staff should seek to understand the aspects of the application that were reviewed by either the DOE or the DOW as a part of that prior authorization and consider the site- and applicant-specific adoption and applicability of that content as reflected in the application. The extent of any NRC leveraging opportunities, including the focus and depth of the review, would depend on how the applicant demonstrates compliance with applicable NRC regulatory requirements and whether the site characterization and consideration of external hazards were developed using the NRC guidance reflected above or are based on a proposed alternate or hybrid approach. Insights gained during pre-application engagement, including observation of the DOE/DOW authorization process, can support a more efficient review.