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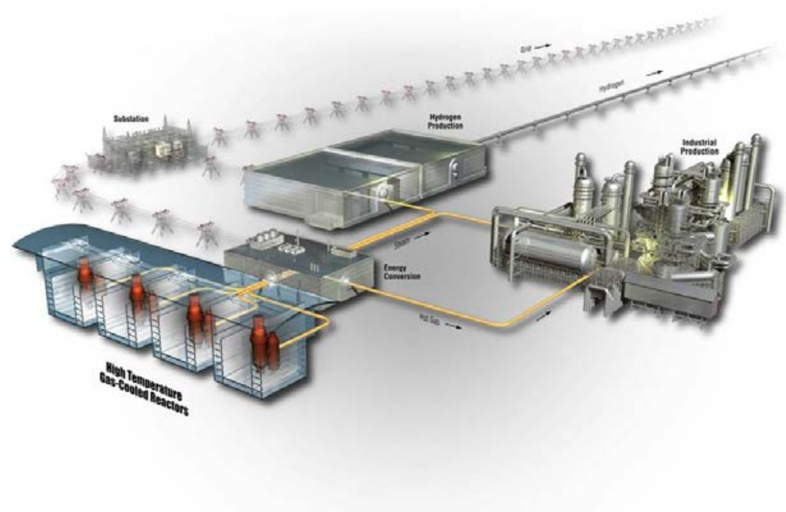
Revision 0

Establishing Jurisdictional Boundaries at Collocated Advanced-Reactor Facilities

August 2020

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



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

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Idaho National Laboratory
Advanced Reactor Technologies
Idaho Falls, Idaho 83415

http://www.art.inl.gov

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

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

Approved by:

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

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SUMMARY

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

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

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

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

CONTENTS

FIGURES

ACRONYMS

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

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

Establishing Jurisdictional Boundaries at Collocated Advanced-Reactor Facilities

1. INTRODUCTION

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

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

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

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

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

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

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

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

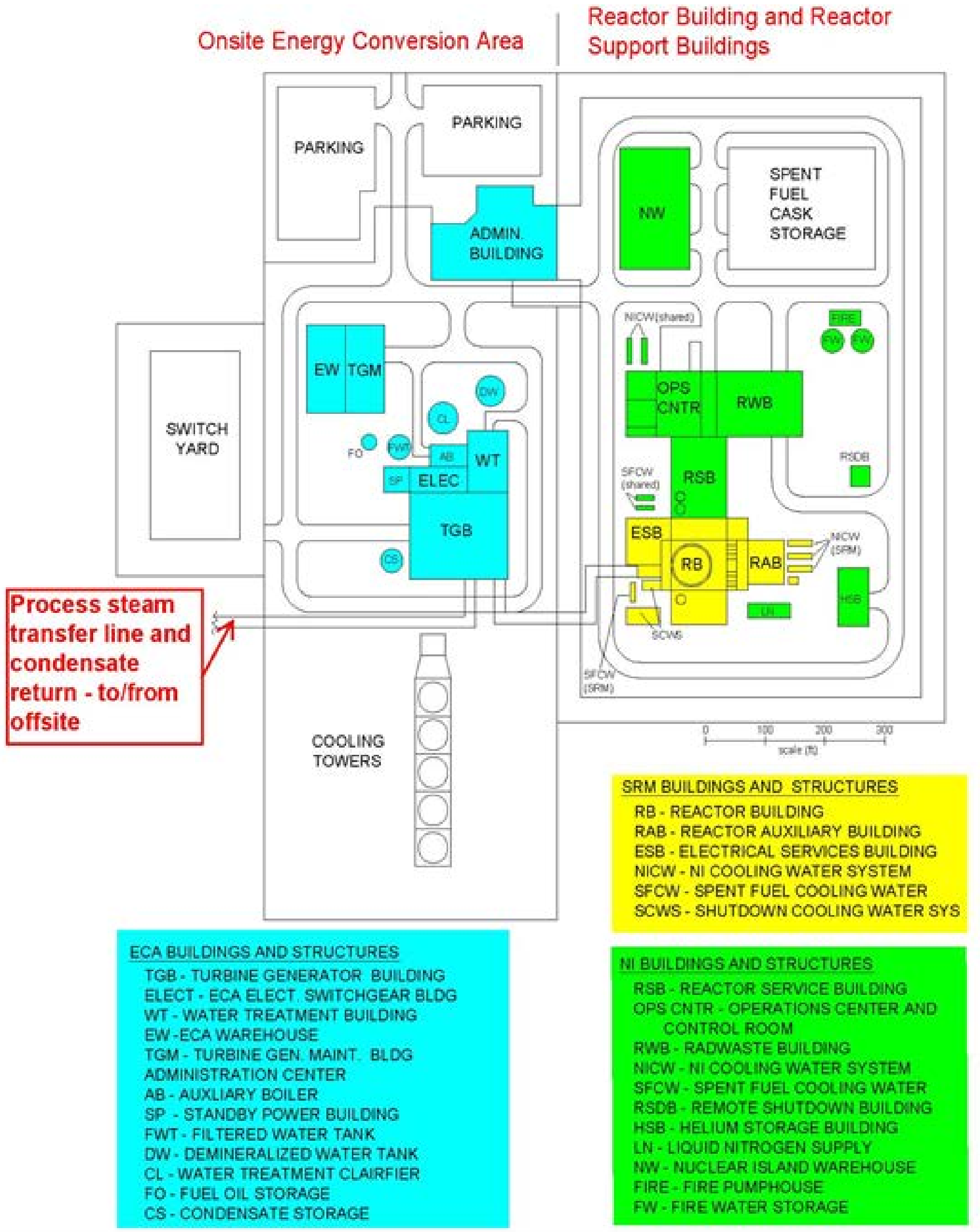

Figure 1. Typical HTGR plant general arrangement.[b]

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

1.1 Approach

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

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

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

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

1.2 Review Objectives

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

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

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

1.3 Scope

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

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

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

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

1.4 Relationship to Other Advanced-Reactor Topics/Papers

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

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

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

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

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

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

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

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

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

10 CFR 52.137 indicates that an application for an SDA must contain a final safety-analysis report (FSAR) that:

> . . . *describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility, or major portion thereof* . . .

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

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

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

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

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

2. REGULATORY FOUNDATION

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

2.1.1 NRC Requirements

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

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

10 CFR 52, Subpart B, “Standard Design Certifications”

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

10 CFR 52, Subpart C, Combined Licenses

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

(a) The application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components of the facility as a whole. The final safety analysis report shall include the following information, at a level of information sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved by the Commission before issuance of a combined license:

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

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

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[i] See 54 FR 15375

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

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

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

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

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

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

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

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

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

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

2.1.2 NRC Policy Statements

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

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

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

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

The staff also notes in that paper:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

2.1.3 NRC Guidance

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

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

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

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

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

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

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

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

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

1. *The review should verify that portions of the MSSS that are essential for safe shutdown of the reactor or for preventing or mitigating the consequences of accidents are evaluated to determine the following:*

    a. *A single malfunction or failure of an active component would not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.*

    b. *Appropriate quality group and seismic design classifications are met for safety related portions of the system.*

    c. *The system is capable of performing multiple functions, such as transporting steam to the power conversion system, providing heat sink capacity or pressure relief capability, or supplying steam to drive safety system pumps (e.g., turbine driven AFW pumps), as may be specified for a particular design.*

    d. *The MSSS design includes the capability to operate the atmospheric dump valves remotely from the control room following a safe-shutdown earthquake (SSE) coincident with the loss of offsite power so that a cold shutdown can be achieved by depending only on safety-grade components.*

2. *The MSSS review should include measures that limit blowdown of the system if a steam line were to break.*

3. *The review includes the design of the MSSS with respect to the following:*

    a. *Functional capability of the system to transport steam from the nuclear steam supply system as required during all operating conditions.*

    b. *Capability to detect and control system leakage and to isolate portions of the system in case of excessive leakage or component malfunctions.*

    c. *Capability to preclude accidental releases to the environment.*

    d. *Provisions for functional testing of safety-related portions of the system.*

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

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

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

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

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

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

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

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

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

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

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

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

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

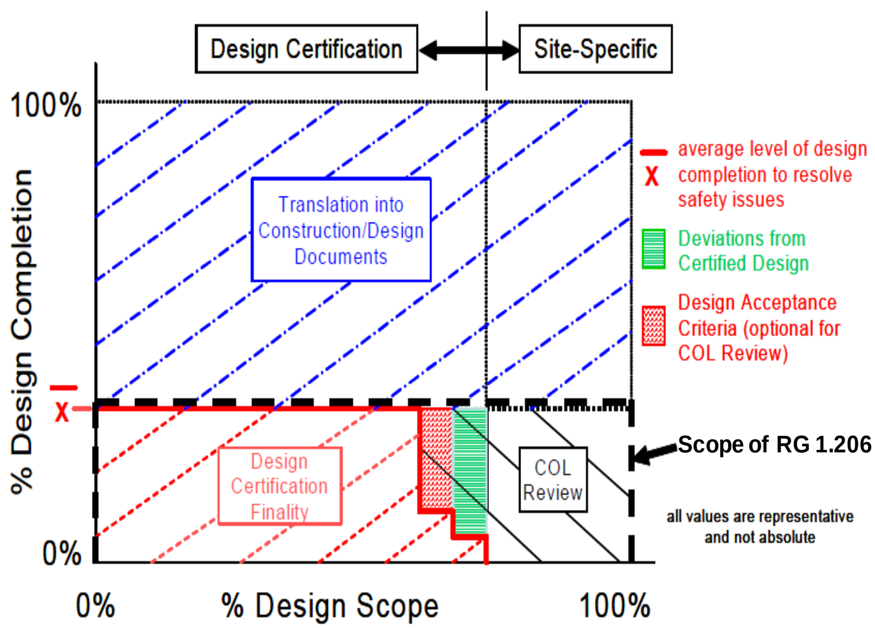

Figure 2. COLA referencing a certified design.

Section C.2.6, Conceptual Design Information—Design Certification

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

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

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

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

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

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

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

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

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

2.2 NRC Historical Precedents

2.2.1 Midland Nuclear Plant

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

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

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

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

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

2.3 Regulatory Foundation for Establishing Top-Level Regulatory Criteria

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

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

2.4 Regulatory Foundation Summary

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

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

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

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

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

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

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

3. DEFINING NUCLEAR-INDUSTRIAL FACILITY AND DESIGN CERTIFICATION BOUNDARIES

3.1 Proposed Approach

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

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

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

3.2 The Nuclear Facility-Industrial Facility Boundary

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

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

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

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

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

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

3.2.1 Security-Related Considerations

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

3.2.2 Nuclear Plant Design and Interface Considerations

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

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

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

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

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

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

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

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

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

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

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

3.2.3 Design Certification Boundary

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

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

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

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

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

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

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

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

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

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

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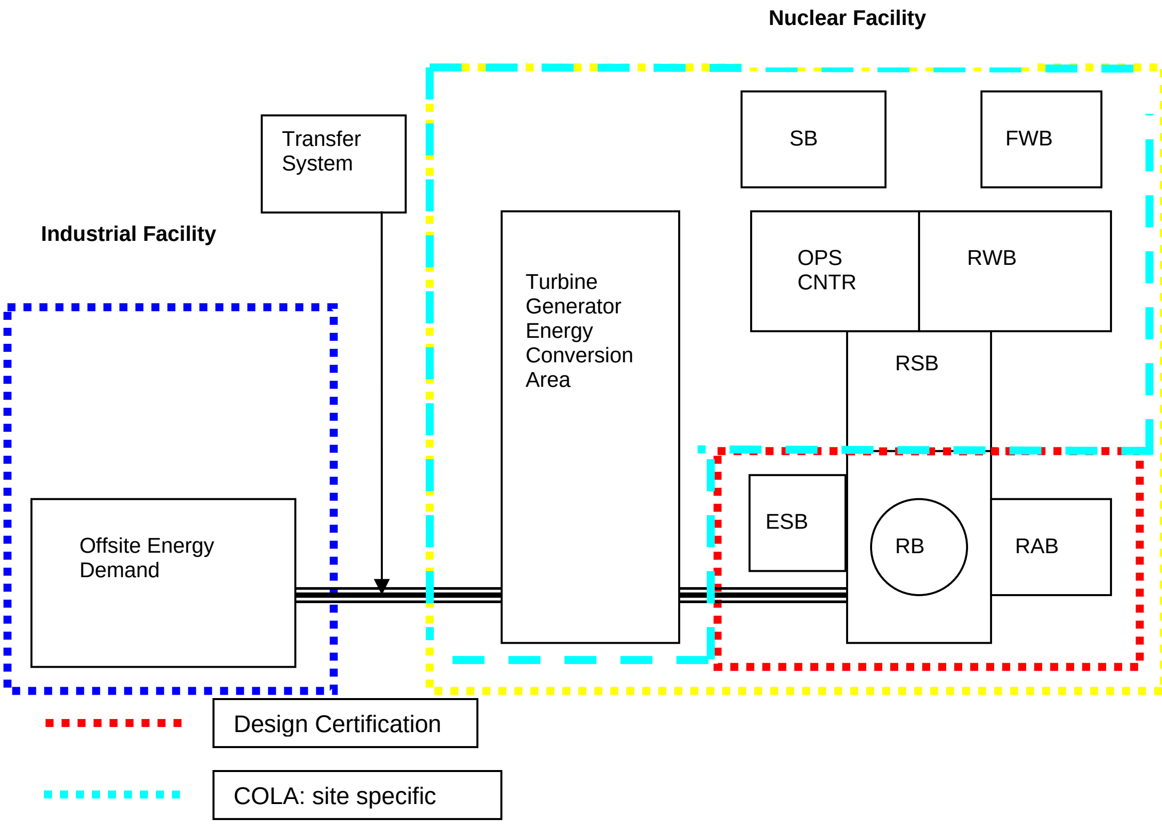
[n] See 10 CFR 50.47(c)(3).

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

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

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



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

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

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

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

[q] While the ECA with the turbine generator was considered physically separate it was still within the HTGR site area and therefore still considered within the *nuclear Island* boundary from a regulatory oversight perspective

design package. The scope of this SRM proposal, and its associated DCA, provides an example that includes:

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

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

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

3.3 Defining COLA Scope

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

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

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

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

3.4 Scope Outside of COLA

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

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

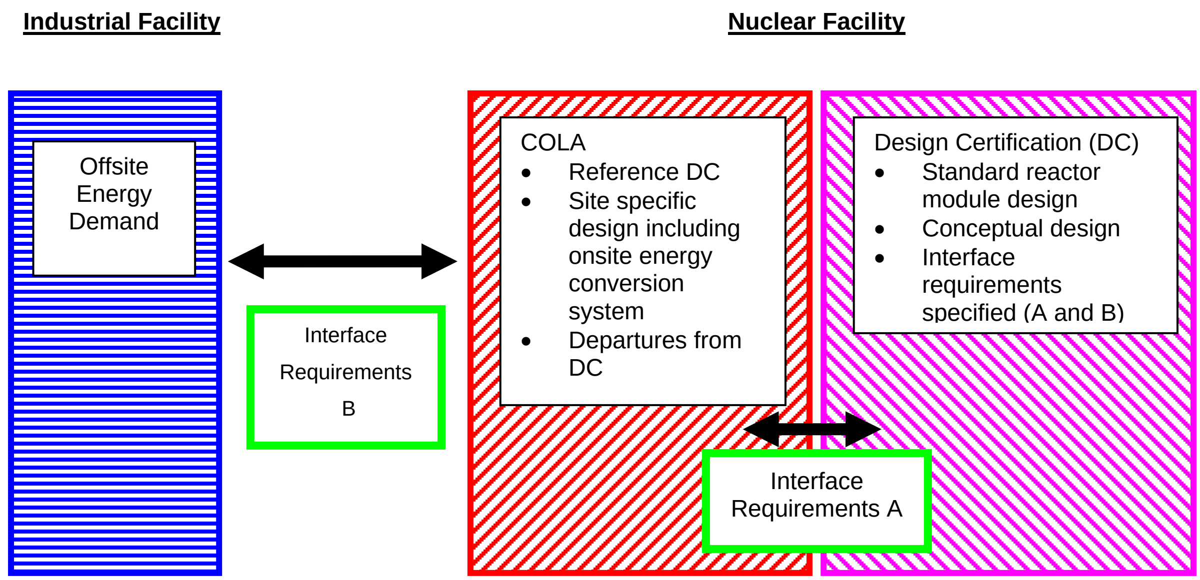

Figure 4. Illustration of approach to nuclear-industrial facility and DC/COL boundaries.

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

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

4. KEY APPROACH ELEMENTS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

   d. Require special treatment for DID adequacy.

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

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

5. REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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