Establishing Interface Requirements for "Major Portions" Standard Design Approvals

NUCLEAR INNOVATION ALLIANCE

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Executive Summary

This document provides guidance regarding a staged licensing process under 10 CFR Part 52, with a focus on the establishment of interface requirements.[1] Interface requirements can be thought of as boundary conditions for the portion of the design for which a standard design approval (SDA) is being sought. The interfaces stem from the dependency of the systems, structures, and components (SSCs) that are within the scope of the application for an SDA on functional and operational characteristics of SSCs that are not within scope. The proposed method can be used by vendors to specify the interface requirements for major portions of a design submitted under the SDA process (Subpart E to 10 CFR Part 52). Because interfaces depend on the particular portion of the design under consideration, a clear definition of its scope is critical at the outset, as discussed in the companion document to this work, entitled, *Clarifying "Major Portions" of a Reactor Design in Support of a Standard Design Approval*, published by the Nuclear Innovation Alliance in April 2017.

The discussions contained herein can be used as a valuable reference to specify interface requirements in an application for an SDA of a major portion of a design. Once the scope is established, it is suggested that the process include an assessment of the advanced reactor design criteria, found in Regulatory Guide 1.232, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*, for non-light water reactors (LWRs), and Appendix A to Part 50, *General Design Criteria for Nuclear Power Plants*, for LWRs, published by the U.S. Nuclear Regulatory Commission. In addition, the regulations and any technical information necessary to develop the design basis of the major portion within the scope of the SDA must be addressed, as described in Section 4.0. While simple in its general methodology, the proposed process will be iterative. Four illustrative examples of establishing interface requirements for some representative advanced reactors, including a core design, the design of a reactor vessel auxiliary cooling system, a piping system, and the structural design of a reactor building are presented as a way to facilitate the application of the process to any interested vendor's design. Although some of the examples pertain to non-LWRs, the process may be applied to LWRs as well.

[1] "Interface" and "boundary condition" are generally used interchangeably in this document to describe limitations, constraints, assumptions, etc. on the "major portion" that is the subject of an SDA. An interface could include a programmatic requirement or assumption about system performance of a "major portion"; a boundary condition could be a physical constraint or an explicit limit on an interfacing system associated with the safety evaluation of the "major portion."

1.0 Introduction

The successful application of horizontal hydraulic fracturing (fracking) to shale rock formations has been a major contributor to the low energy prices observed in the U.S. over the last decade. As a result, nuclear innovators have had to focus their efforts on designs that guarantee not only enhanced safety, but also economic viability. Several innovative features are being incorporated into design concepts to address these desired characteristics, including elements such as modular construction, integral system design, and passive safety. Modularity involves off-site construction of individual modules in a factory with a controlled environment and shipping to sites via barge, rail and/or truck. Integral system design precludes the need for coolant loops outside the vessel, which eliminates certain postulated accidents, including loss-of-coolant accidents. Some designs incorporate passive safety systems and remove the need for alternating current (ac) and direct current (dc) power supplies during postulated accidents. These features all contribute to simpler designs that may be less expensive and can be built faster than large power plants. They have been applied to LWRs in the past and are now being used by advanced reactor vendors to design reactors that rely on alternative coolants. Such designs, including high-temperature reactors (HTRs), liquid metal fast reactors such as sodium fast reactors (SFRs), and molten salt reactors (MSRs), benefit from these approaches, while also taking advantage of the inherent safety features of the designs.

Many companies currently developing advanced reactors in the U.S. rely primarily on private funding such as venture capital, along with some funding from the Department of Energy (DOE). The Nuclear Innovation Alliance (NIA) report, *Enabling Nuclear Innovation: Strategies for Advanced Reactor Licensing*, published in April 2016, discussed the impact of this funding model on the licensing process. Because many companies are seeking and obtaining investor funding in a phased manner, a staged licensing approach can be more appropriate for some companies. As companies seek larger investments to support more costly later-stage development efforts, they must demonstrate that their likelihood of success is increasing. The use of standard design approval (SDA) in licensing would allow an applicant to submit major portions of a design for approval from the U.S. Nuclear Regulatory Commission (NRC) in an incremental fashion, towards final design certification (DC), which would encourage additional investment by reducing regulatory risk and delays.

In the NRC's regulations, the DC process, specified in Subpart B, "Standard design certifications," to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," requires a complete design to be submitted to the NRC for review. However, the NRC can also conduct staged licensing using the provisions of Subpart E, "Standard design approvals," to 10 CFR Part 52, which indicates in paragraph (a) to § 52.135, "Filing of applications," that:

> "any person may submit a proposed standard design for a nuclear power reactor of the type described in 10 CFR 50.22 to the NRC staff for its review. The submittal may consist of either the final design for the entire facility or the final design of major portions thereof."

In December 2017, the NRC published, *A Regulatory Review Roadmap for Non-Light Water Reactors*. The roadmap indicates that a staged licensing process using the SDA in Subpart E can be advanced to the construction phase in several ways: 1. With an application for a DC that contains the required information for the remainder of the design; 2. In conjunction with an application for either a construction permit (CP) or operating license (OL) under the two-step licensing process in 10 CFR Part 50; or 3. In conjunction with an application for a combined license (COL) that does not reference a certified design. The NRC roadmap references the NIA report, *Clarifying 'Major Portions' of a Reactor Design in Support of a Standard Design Approval*, issued in April 2017. The NIA report explains the term, "major portion," describes the SDA process, and discusses the potential benefits of an SDA to a vendor.

In particular, the 2017 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 DCA [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 April 2017 document 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 CP, or an OL. This document 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.

2.0 Purpose and Scope

The purpose of this report is to provide guidance to vendors of advanced reactors using the 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 Part 52 or 10 CFR Part 50. Although interface requirements are used throughout the regulatory framework, little guidance exists to assist an applicant in the process of establishing these interfaces and their requirements. Because the SDA as part of a staged licensing approach is expected to be used by some vendors, the guidance contained herein should facilitate the design, licensing, and deployment of advanced reactors. The process can be applied to any reactor type.

Interface requirements are common in the existing licensing process for LWRs and are referenced in a number of NRC regulatory documents. For instance, they are utilized to specify a variety of operating requirements including valve closure or opening times and emergency core coolant system pump startup times, as detailed in Chapter 15, “Transient and Accident Analysis,” of NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*, (SRP). The term “interface requirements” is also used in most Regulatory Guides, to highlight dependencies among the structures, systems and components (SSCs) and their associated regulatory requirements.

An important consideration for vendors developing interface requirements in support of an SDA is the variability of their levels of detail. Interface requirements will depend on an applicant’s planned use of the granted SDA, as well as the SSCs treated in the SDA and those in subsequent submissions. The precision of any analytical tools used will also impact the level of detail required in the interfaces. For example, less geometric detail would be needed as the inputs to an analytical tool that models the heat transfer and coolant flow of a core on a coarse mesh (i.e., on the order of 100 nodes), than what would be needed as the inputs to an analytical tool using a fine grid (i.e., on the order of millions of nodes). Likewise, small variations in particular parameters may lead to significant changes in global variables, necessitating more detail in the inputs to achieve adequate margin to the design requirements.

Another consideration is that SSCs are not all reviewed by the NRC in the same manner. For example, the first step in reviewing a structure is to select representative sections of the structural design – not every structure in the design is reviewed. The NRC refers to these sections as critical sections. Piping and digital instrumentation and control have been typically specified in DC applications through high level, objective attributes that must be satisfied by the final design in a subsequent licensing process using objective measures or methods preapproved by the NRC. The level of detail of the interface requirements will vary by engineering discipline. The experience gained with the licensing and construction of the AP1000 design in the U.S. has demonstrated that specifying small tolerances in allowable values can hinder the flexibility necessary for efficient construction. In contrast, specifying large margins in allowable values for reactor systems, such as emergency heat removal systems, may be more challenging to justify to a regulator and may also reduce economic viability. As a result, determining the appropriate margin in interface requirements should consider the type of SSC, as well as any downstream impacts on construction and economic viability of the design. Because of this variability, characteristics of interface requirements will depend on the particular SSCs of interest and will be case-specific.

Applicants can use this report to inform development of interface requirements in an application for an SDA of a major portion of a design. Interface requirements will need to be satisfied in subsequent licensing submissions, which could be in the form of another SDA, the remainder of the design for a DC or a COL under 10 CFR Part 52, or for a CP or an OL under 10 CFR Part 50.

3.0 Standard Design Approval

A standard design approval is a licensing mechanism specified in Subpart E of 10 CFR Part 52 that permits an applicant to submit a design for an entire nuclear power plant or for major portions of a design to the NRC for review. If granted, an SDA becomes part of a design's formal record in support of its final license application, and can be referenced in another SDA, COL, DC, CP, or OL. The SDA can also support a staged licensing approach, as it allows a vendor to apply for an SDA for one major portion of the design at a time. The incremental nature of this approach may benefit applicants by enabling regulatory review and approval of a portion of the plant before the entire design is developed:

1. This could avoid a possible situation in which an NRC licensing decision requiring a change in the subject portion would have consequent impacts on the rest of the design, requiring re-design and re-review. If the vendor can identify portions of the design with high regulatory uncertainty likely to incur such effects (novel design features not previously reviewed by NRC in any similar form), then the vendor can seek an SDA for that portion prior to finalizing contingent portions of the design and prior to submitting them for NRC review.
2. This could allow vendors to focus on attaining approval for the portions of the design that are most critical to their safety or business case before requiring any further investment beyond a general description of the SSCs not covered by the SDA. An SDA for the subject portion would reduce the risk of regulatory delays or barriers and thus enable investment from funding sources with lower risk tolerance.

It is likely that the NRC licensing costs incurred by a vendor using staged licensing to ultimately achieve a DC, COL, CP, or OL would exceed those incurred when including the entire design in one application, if that single application were processed without major revisions. This is in part due to the need for the vendor to establish interface requirements and for the NRC to review them, as well as the need for an additional submission that demonstrates all interface requirements specified in the multiple SDAs have been satisfied. In addition, staged licensing will likely lengthen the time needed to receive regulatory approval of the entire design which increases the likelihood that the original staff reviewers who started the design review will not be available for subsequent stages; the NRC's core team approach may mitigate this issue. However, substantial re-design and re-review on a complete application could also be costly.

Further, incremental approval may enable increased regulatory certainty earlier in the development process that could be important for certain designs. The development of advanced reactor technology requires large investments which may be best made in a graduated fashion. Given the nature of such investments, it is critical for companies to demonstrate incremental licensing progress alongside design development so that they can prove economic and overall project viability in order to secure continued funding. Successful NRC review of a key design segment can reduce overall regulatory risk by providing assurance to investors, technology partners, and prospective utilities that the new technology is both viable and worth the continued investment.

In the end, it will be at the discretion of each vendor to assess which approach strikes the right balance for their design and their business case.

An application for an SDA is reviewed both by the NRC staff and the Advisory Committee on Reactor Safeguards (ACRS). An SDA documents the NRC staff's findings of the design approval but does not preclude issues resolved during the review from being revisited during rulemaking for a DC, or during hearings associated with an application for a CP or COL. An SDA does have a level of issue finality with respect to the NRC staff and the ACRS: the requirement in 10 CFR 52.145, "Finality of standard design approval; information requests," indicates that the staff can raise an issue that has already been approved in the SDA process only if its burden on the vendor is justified by its safety significance, as specified in 10 CFR 50.54(f), and if it is approved by the Executive Director for Operations of the NRC or his/her designee. An SDA lasts for 15 years. In contrast to an SDA, an application for a DC must include the essentially complete design of a plant. A DC involves reviews by the staff, the ACRS and the Commission. Once approved, rulemaking occurs, and public comments are resolved. The certified design is specified in a rule as an Appendix to 10 CFR Part 52. Issue finality for a DC is more robust than for an SDA, and is specified in 10 CFR 52.63, "Finality of standard design certifications." Like an SDA, a DC lasts for 15 years.

Another use of SDA is to provide an optional feature of a design. For example, a vendor can submit their base design for DC, and then pursue SDA for an optional feature of the base design, such as a power uprate, an added energy storage feature for market or load following, or a non-electric application such as hydrogen production or desalination. In the case of a power uprate, the application for SDA would cover the SSCs affected by the power uprate with interface requirements that link to the SSCs of the standard design not affected by the uprate. A utility could then choose to reference the certified design, or reference it while deviating from the design by referencing the SDA.

4.0 Methods to Develop Interface Requirements

The rule language of 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 rule only requires that information be submitted to the extent that the requirements are applicable to the major portion of the design for which the SDA is being sought. Therefore, as discussed in the April 2017 NIA report, *Clarifying "Major Portions" of a Reactor Design in Support of a Standard Design Approval*, the application for an SDA must clearly define the scope of the SDA – i.e. the SSCs that are included in the application and the engineering disciplines that will be addressed in the application - and the technical basis that demonstrates how the principal design criteria (PDC) or other applicable design requirements are fulfilled.

The highest level of design requirements includes the design, fabrication, construction, testing, and performance requirements for SSCs that have been categorized as safety significant. The PDC are one example of a set of such requirements, but others could, in principle, be developed, and the report does not intend to highlight the PDC as the only option. To assist a vendor in developing design requirements for their design, the NRC issued Regulatory Guide 1.232, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*, which delineates examples of acceptable advanced reactor design criteria (ARDC) for non-LWRs. Since the ARDC (or other selected criteria), and in turn the PDC, are specified by the vendor at a high level, the vendor will need to propose its own detailed safety limits and requirements to demonstrate the fulfillment of the design criteria. Depending on the novelty of its design, a vendor may also have to specify additional design criteria beyond those stipulated in the ARDC. Instead of relying on deterministic principles to define the design specific safety limits and requirements necessary to satisfy the design criteria, a risk-informed, performance-based licensing approach is being developed with support from DOE called the Licensing Modernization Project (LMP; NEI 18-04), which can be used to introduce risk-informed concepts to define these specific safety limits and perhaps reduce the number of design criteria necessary to ensure safety.

The requirements of 10 CFR 52.137(a)(24) that pertain to SDA in Subpart E to Part 52 require the submission of interface requirements between the submitted portion of the design and the remainder of the design. The process for developing such requirements should be generic (given the range of advanced reactor designs) and should consider relevant NRC regulations and applicable guidance, such as RG 1.232.

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 sufficient detail to provide the NRC staff with an adequate basis for a safety determination. A subsequent application for an SDA of the remaining portions of the design will need to demonstrate that the interface requirements are satisfied.

The concept of interface requirements in an application for a DC is outlined in 10 CFR 52.47(a)(24), which states that an application for a DC must contain:

> “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.”

Furthermore, 10 CFR 52.47(a)(25) indicates that an applicant for a DC must address:

> “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.”

Although this definition applies to a DC, the same definition can be used for the SDA process by substituting the terms “SDA” and “approval” for the terms “DC” and “certification,” respectively.

Figure 1 depicts a process to establish interface requirements in support of an SDA for a major portion of a design. In the context of a conceptual plant design, the portion of the design for which an SDA is being sought is defined, as discussed in the aforementioned April 2017 NIA report. In the next step, the designer reviews applicable regulations and guidance to develop design criteria. Applicable regulations and guidance could include the advanced reactor design criteria (ARDC), specified in RG 1.232, as well as the regulatory requirements contained in 10 CFR Parts 50 and 52. Some of the current guidance and regulations will be directly applicable, some will be applicable to LWRs alone and will need to be excluded or reformulated. For regulations that do not explicitly exclude the design in consideration but are not applicable to that design, the vendor should develop design specific criteria and will need to request an exemption to the rule under 10 CFR 52.7, "Specific exemptions." In a final category, the underlying purpose of the rule does not apply and an exemption to the rule must be included in the application for an SDA. When developing a basis for exemptions, the LMP can be used to introduce risk-informed concepts or a vendor can rely solely on deterministic principles. This process will result in the development of design criteria for the major portion. Likewise, regulations and guidance are used to define boundary conditions and interface requirements in conjunction with the design process.[2] The interfaces stem from the dependency of the SSCs that are within the scope of the application for an SDA on functional and operational characteristics of SSCs that are not within this scope. Then, an application for SDA is submitted and reviewed by the NRC. Interface requirements also inform complementary SDA applications for other major portions of the design, or subsequent applications for a design certification or construction permit.

It is important to recognize that when the excluded portions of a design are subsequently submitted for approval, previously defined interface requirements must be satisfied or the design for the major portion originally approved will need to be revisited. To help prevent this, some margin should be included in interface requirements. This margin would not be included to ensure safety of the design but would be included to allow flexibility in the design of the excluded portions, which would help minimize the need for modifications to the portions of the design that have already been granted an SDA. It is recommended that appropriate records of the interface requirements be maintained to facilitate the design process of excluded portions of the design. The amount of margin for each interface requirement will be design-specific and will need to consider the cost effectiveness of the design as well as design flexibility.

As depicted by dashed lines in Figure 1, calculations generated as part of an application for an SDA for a major portion may be referenced by a future application for an SDA for excluded portions. These interfaces should be recorded and tracked to ensure that subsequent portions of the design use appropriate values. Unlike formal interface requirements for the major portion, the process described herein does not recommend that these interfaces be submitted to the NRC in support of an SDA of the original major portion. Tracking these interfaces is

[2] Guidance is a broad category, and includes documents issued by several organizations such as the NRC, DOE, the Nuclear Energy Institute, NIA and other organizations interested in advanced reactors. Some examples include the Licensing Modernization Project, NIA documents discussed herein, and reports issued by the NRC, such as the aforementioned NRC Roadmap. There are also more detailed documents, such as NUREG/CR-6844, "TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents," and previously issued preliminary safety evaluation reports of earlier non-LWR designs. Codes and standards should also be considered as designs are developed. They are developed by consensus standard organizations, such as the American Society of Mechanical Engineers (ASME) and the American Nuclear Society. When endorsed by the NRC, codes and standards provide the detailed design, special treatment and fabrication requirements for SSCs, such as mechanical, electrical, and piping systems. Whereas regulations generally tend to be more performance-based and establish at a high-level what safety function must be achieved, codes and standards provide details specifying ways in which the design can achieve the particular safety function.

recommended only as a means of enhancing the efficiency of the design process for the excluded portions.

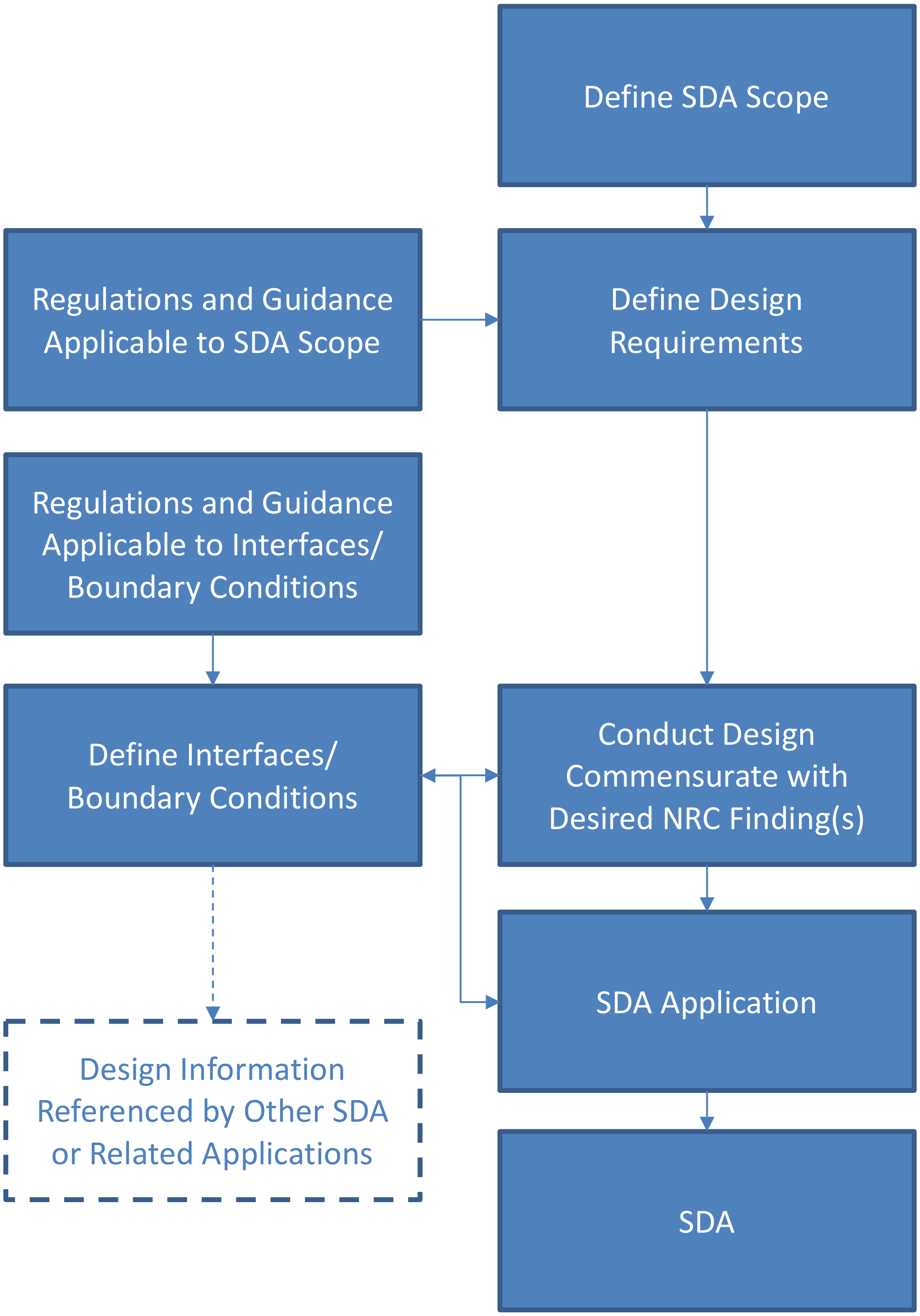

Figure 1: Process for Developing Interface Requirements in Support of an SDA

5.0 Example Cases

In order to demonstrate the process, four hypothetical example cases are evaluated for interface requirements using the approach shown in Figure 1. The cases provide examples of interface requirements for an advanced reactor core design, a reactor vessel auxiliary cooling system (RVAC), an advanced reactor Class I piping design, and the structural design of a reactor building. To make the examples below as widely applicable as possible, the general ARDC specified in RG 1.232 are used as substitutes for design specific PDC, rather than the technology-specific SFR-DC and HTGR-DC for sodium fast reactors and high-temperature gas reactors, respectively. As discussed in Section 5, "Content," of the aforementioned 2017 NIA report, there are three general categories of scope of an application for an SDA:

> "The SDA can be thought of as similar to a design certification (DC) but covering fewer SSCs. In this situation the level of design detail for the selected "major portions" would be comparable to a DC application, and the interface requirements would be described in a manner analogous to a DC application with a significant amount of conceptual design information (CDI)." The core design is an example of this category.
>
> "An SDA could be developed that sets forth major portions associated with concepts or design aspects that are novel relative to existing US technology. In this example, the SSCs might not be as fully developed as for a DC application; in this case, the NRC staff approval (i.e., safety evaluation report, or SER) also would be less substantive, but still could be useful (e.g., an NRC SER that approves fundamental operational and safety strategies would enable the design to proceed with reduced licensing risk)." The RVAC design for an SFR or MSR, and the high-temperature piping design for [an advanced reactor] design not using an integral reactor coolant system (RCS) are examples of this category.
>
> "The 'mixed' category could reflect aspects of the two categories above. For example, an SDA could cover preliminary design information for an essentially complete design, equivalent to the level of design information and detail required for a construction permit, but without site specific considerations." The structural design of a reactor building is an example of this category.

This section contains tables that delineate the interface requirements of the SDA under consideration and are organized by each ARDC. Here, ARDC are used as a substitute for design specific PDC to make the examples as generally applicable as possible. The process shown in Figure 1 involves the consideration of regulatory requirements, guidance documents, and the ARDC, but the examples refer only to the ARDC so as to limit the scope of the examples. The interfaces that are described in these examples stem from the dependency of the SSCs that are within the scope of the application for an SDA on functional and operational characteristics as well as requirements of SSCs that are not.

For any given application for an SDA, not all of the ARDC will apply. In contrast, some ARDC will apply to most applications for an SDA. For example, ARDC 1, “Quality standards and records,” will likely apply to any major portion that includes SSCs important to safety. The applicant’s Quality Assurance (QA) program for SSCs important to safety would serve as a programmatic interface requirement for ARDC 1.

5.1 Case 1 - Core Design

Developing an application for an SDA of a reactor core will generally involve a large portion of a design’s PDC, as many of the primary safety functions of any reactor design are performed in the fuel and the core (e.g. ensuring adequate reactivity control, core cooling, and the prevention of fission product release). The scope of the core SDA example includes the fuel design and specified acceptable fuel design limits developed for the advanced reactor fuel, design, and geometry of the vessel internals, and analyses of heat transfer, coolant flow, and reactor physics to demonstrate the safety of the plant’s response to postulated equipment failures or malfunctions.

In this example, only those SSCs that are part of the core will be addressed in the application for an SDA. The design information in the FSAR will be specified at the same level of detail as an FSAR that would be submitted as part of an application for a DC and is similar to the first category of SDAs discussed above and in the 2017 NIA report. The example assumes the following conditions: 1) topical reports pertaining to the QA program and methods developed to analyze the fuel behavior, system heat transfer and coolant flow conditions and reactor physics of the core during normal operation, anticipated operational occurrences (AOOs) and postulated accidents (including beyond design basis accidents included in the licensing basis of the design) have been submitted to the NRC; 2) AOOs and postulated accidents have been selected; 3) a general design of the RCS excluding the core has been developed; 4) the general design of the reactivity control measures and characteristics have been developed; and 5) a general design of a structure performing the containment function has been developed. The SDA considers conditions under normal and postulated accidents and does not address conditions during fuel storage. While the LMP process has not been assumed, it could, among other things, be used to introduce risk-informed approaches to selecting postulated accidents.

Using the general ARDC as a substitute for design specific PDC, the interface requirements were developed, as delineated in Table 1. This example, as well as subsequent examples, are intended to illustrate the process rather than yield a complete set of interface requirements. Furthermore, the example considers only the regulatory requirements specified in the general advanced reactor ARDC and does not consider any rules contained in 10 CFR Part 50 or 52.[3] In this example, interface requirements were identified for several ARDC because information from numerous systems is needed to perform the analyses necessary to design the core. This

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[3] In addition to the ARDC considered in this illustrative example, an applicant should consider rules in 10 CFR Part 50 or 52, as appropriate, as well as relevant guidance documents when preparing an application for an SDA.

information includes, but is not limited to: logic and actuation times of the instrumentation and control system and the reactor protection system; heat transfer and stored energy in the structures; functional and operational characteristics of the reactivity control system; nominal and accident conditions and associated heat removal rates of heat removal systems external to the vessel; and nominal and accident conditions and associated heat removal rates of the structure(s) performing the containment function. Although not included in Table 1, considerations for the core major portion would include numerous secondary interface requirements as the impact of AOOs and postulated accidents must be considered when establishing the limiting conditions for operation, limiting safety system settings, and design specifications for safety related SSCs.

For the core example, interface requirements were identified in seventeen key areas of the ARDC:

- Quality standards and records
- Design basis for protection against natural phenomena
- Sharing of structures, systems, and components
- Instrumentation and control
- Reactor coolant boundary
- Reactor coolant system design
- Containment design
- Electric power systems
- Protection system functions
- Protection system requirements for reactivity control malfunctions
- Reactivity control systems
- Reactor coolant inventory maintenance
- Residual heat removal
- Emergency core cooling
- Containment heat removal
- Containment atmosphere cleanup
- Structural and equipment cooling

Table 1: Interface Requirements for the Core SDA

ARDC Number	Title	
1	Quality standards and records	Interface Requirement
		The QA program will have been submitted to the NRC as a topical report. 1. The FSAR will include justification that the approved QA program was followed during the development of the SDA. 2. The FSAR will include justification for the safety classification of the SSCs included in the SDA. 3. The FSAR will include justification that the SSCs included in the SDA have been designed commensurate with their safety significance and the QA program.
2	Design basis for protection against natural phenomena	Interface Requirement
		The ability of the SSCs of the core unit to withstand the design basis seismic event will be addressed in the FSAR. The comparison of the FSAR design assumptions to those relating to an actual site will be addressed in a future submission. Adequate margin should be included in the assumed seismic event to provide flexibility in siting the design. 1. The FSAR will specify the seismic design parameters (e.g., earthquake design response spectra). This parameter will be compared to that evaluated for a future site.
3	Fire protection	No Dependence
4	Environmental and dynamic effects design bases	No Dependence
5	Sharing of structures, systems, and components	Interface Requirement
		Any sharing of SSCs will be addressed in a future submission. 1. The FSAR will include a commitment that any SSC important to safety credited in the calculations performed under ARDC 10, 11, 12, 33, 34, and 35 shall not be shared among

		nuclear power units unless it can be shown that such sharing will not significantly impair its ability to perform its safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.
10	Reactor design	Included in FSAR as part of major portion
11	Reactor inherent protection	Included in FSAR as part of major portion
12	Suppression of reactor power oscillations	Included in FSAR as part of major portion
13	Instrumentation and control	Interface Requirement
		The instrumentation and control (I&C) system will be addressed in a future submission. 1. The FSAR will include a summary of the controls that shall be provided, including the time needed for the system to perform the function, to ensure that variables and systems that can affect the fission process, the integrity of the reactor core, and the reactor coolant boundary are maintained within appropriate ranges. This information is used in analyses required under ARDC 10, 11, 12, 33, 34, and 35.
14	Reactor coolant boundary	Interface Requirement
		The material selection and structural design of the reactor coolant boundary will be addressed in a future submission. 1. The FSAR will include information pertaining to the reactor coolant boundary that is necessary to perform analyses required under ARDC 10, 11, 12, 33, 34, and 35, such as break sizes possible in the reactor coolant boundary.
15	Reactor coolant system design	Interface Requirement
		The FSAR will include information pertaining to auxiliary, control, and protection systems that is necessary to perform analyses required under ARDC 10, 11, 12, 33, and 35, such as volumes, frictional and form loss characteristics, parameters that affect stored heat in the systems, etc. 1. The FSAR will include information pertaining to the RCS design that is necessary to

		perform analyses required under ARDC 10, 11, 12, 33, 34, and 35, such as volumes, frictional and form loss characteristics, parameters that affect stored heat in the systems, etc.
16	Containment design	Interface Requirement
		In this example, the containment function is assumed to be carried out by a structure and will be addressed in a future submission.[4] 1. The FSAR will include any information relevant to the containment design that is necessary to conduct analyses required under ARDC 10, 11, 12, 33, 34, and 35, such as nominal temperatures and pressures, heat removal rates, etc.
17	Electric power systems	Interface Requirement
		The electrical power systems will be addressed in a future submission. 1. The FSAR will include information pertaining to the electrical system that affects the calculations performed under ARDC 10, 11, 12, 33, 34, and 35, such as availability of ac and dc power during AOOs and postulated accidents.
18	Inspection and testing of electric power systems	No Dependence
19	Control room	No Dependence
20	Protection system functions	Interface Requirement
		The protection system functions will be addressed in a future submission. 1. The FSAR will include a summary of the protection system functions that shall be provided, including the time needed for the system to perform the function, to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded as a result of AOOs, and accident conditions are detected

[4] For a design which relies on the fission product retention function of its fuel, such as in the case of TRISO fuel, information such as, but not limited to the following would need to be included in the FSAR to satisfy this PDC: the conditions of the fuel developed during normal operation, AOOs, postulated accidents, and beyond design basis accidents calculated using NRC approved analysis tools; the estimated fission product release rates under these conditions calculated using NRC approved analysis tools; and sufficient test data pertaining to fuel behavior and fission product release rates.

		and systems and components important to safety are initiated to operate. This information, such as trip setpoints, logic, and time necessary for actuation is used in analyses required under ARDC 10, 11, 12, 33, 34, and 35.
21	Protection system reliability and testability	No Dependence
22	Protection system independence	No Dependence
23	Protection system failure modes	No Dependence
24	Separation of protection and control systems	No Dependence
25	Protection system requirements for reactivity control malfunctions	Interface Requirement The protection system requirements for reactivity control malfunctions will be addressed in a future submission. 1. The FSAR will include information relevant to the protection system that are necessary to conduct analyses required under ARDC 10, 11, 12, 33, 34, and 35, such as trip setpoints, logic and time necessary for actuation.
26	Reactivity control systems	Interface Requirement The reactivity control systems will be addressed in a future submission. 1. The FSAR will describe the reactivity control systems, including the time it takes to increase or decrease reactivity and the effect of malfunctions, in order to perform calculations required under ARDC 10, 11, 12, 33, 34, and 35.
28	Reactivity limits	Included in FSAR as part of major portion
29	Protection against AOOs	Included in FSAR as part of major portion
30	Quality of reactor coolant boundary	No Dependence
31	Fracture prevention of reactor coolant boundary	No Dependence
32	Inspection of reactor coolant boundary	No Dependence
33		Interface Requirement

	Reactor coolant inventory maintenance	The reactor coolant inventory maintenance program will be addressed in a future submission. 1. The FSAR will include information about the largest break in the RCS that can be accommodated by the reactor coolant makeup system, if included in the design.
34	Residual heat removal	Interface Requirement
		The residual heat removal system will be addressed in a future submission. 1. The FSAR will include information pertaining to the residual heat removal system that affects the calculations required under ARDC 10, 11, 12, 33, and 34, such as heat removal rates by the system as a function of RCS parameters, such as pressure, temperature, coolant inventory, etc.
35	Emergency core cooling	Interface Requirement
		The emergency core cooling system will be addressed in a future submission. 1. The FSAR will include information pertaining to the emergency core cooling system that affects calculations required under ARDC 35, such as heat removal rates, inventory addition rates, if included in the design, natural circulation rates necessary for adequate cooling, etc.
36	Inspection of emergency core cooling system	No Dependence
37	Testing of emergency core cooling system	No Dependence
38	Containment heat removal	Interface Requirement
		In this example, the containment function is assumed to be achieved by a structure. The containment heat removal system will be addressed in a future submission. 1. The FSAR will include information pertaining to the containment heat removal system that affects calculations performed under ARDC 35, such as heat removal rates, etc...
39	Inspection of containment heat removal system	No Dependence.

40	Testing of containment heat removal system	No Dependence.
41	Containment atmosphere cleanup	Interface Requirement In this example, the containment function is assumed to be achieved by a structure and will be addressed in a future submission. The containment atmospheric cleanup system will be addressed in a future submission. 1. The FSAR will include information pertaining to the containment atmospheric cleanup system that affects the calculations performed under ARDC 35, such as the removal rate of containment gaseous species.
42	Inspection of containment atmosphere cleanup systems	No Dependence
43	Testing of containment atmosphere cleanup systems	No Dependence
44	Structural and equipment cooling	Interface Requirement The structural and equipment cooling system will be addressed in a future submission. 1. The FSAR will include information pertaining to the structural and equipment cooling system that affects the calculations performed under ARDC 10, 11, 12, 33, 34 and 35, such as heat removal rates from the structures as a function of the temperature of the structures.
45	Inspection of structural and equipment cooling systems	No Dependence
46	Testing of structural and equipment cooling systems	No Dependence
50	Containment design basis	No Dependence
51	Fracture prevention of containment pressure boundary	No Dependence
52	Capability for containment leakage rate testing	No Dependence

53	Provisions for containment testing and inspection	No Dependence
54	Piping systems penetrating containment	No Dependence
55	Reactor coolant boundary penetrating containment	No Dependence
56	Containment isolation	No Dependence
57	Closed system isolation valves	No Dependence
60	Control of releases of radioactive materials to the environment	No Dependence
61	Fuel storage and handling and radioactivity control	No Dependence
62	Prevention of criticality in fuel storage and handling	No Dependence
63	Monitoring fuel and waste storage	No Dependence
64	Monitoring radioactivity releases	No Dependence

5.2 Case 2 - Reactor Vessel Auxiliary Cooling System Design

An RVAC system is a passive decay heat removal system that cools the reactor vessel in an SFR or MSR by natural convection to the environment. In one example of this system, decay heat from the reactor fuel in an SFR will be transferred to the reactor vessel by natural circulation of the coolant to a graphite moderator. The graphite heats and conducts heat to the reactor vessel. As the temperature of the vessel increases, heat is radiated through a gas such as argon, to the containment vessel or in some designs, a guard vessel. Natural convection of air is then established with down flow of cold air in the outer channel and up flow of hot air through the inner channel adjacent to the containment vessel. The hot air is then released to the environment. The RVAC is similar in principle to a reactor cavity cooling system in a HTGR.

This example will cover only a preliminary RVAC design. Interface requirements will be established for the excluded portions of the design which have been developed at a level sufficient to provide boundary conditions to the RVAC. This example case is similar to the second category of SDAs discussed in the 2017 NIA report. Because the heat removal is driven by passive means, the RVAC system is essentially a structure and the level of detail of the design in the FSAR should be sufficient to describe key analysis tools and key design assumptions. The design of the RVAC will consider the effects of natural phenomena, such as seismic events, as well as the heat transfer rates required during AOOs and postulated accidents to fulfill the safety functions of RVAC. Key design assumptions, such as those relating to seismic design spectra should include some margin, such that design can be constructed at a range of sites. As a result, developing an application for an SDA of the RVAC will involve PDC associated with structural design and decay heat removal, including analyses demonstrating the ability of the RVAC system to remove decay heat removal.

Using the general ARDC as a substitute for design specific PDC, the interface requirements were developed, as shown in Table 2. This example is intended to illustrate the process rather than yield a complete set of interface requirements. Furthermore, the example considers only the regulatory requirements specified in the general ARDC and does not consider any rules contained in 10 CFR Part 50 or 52.[5] In this example, interface requirements were identified for thirteen key areas of the ARDC because information from numerous systems is needed to perform the analyses necessary to design the RVAC:

- Quality standards and records
- Design basis for protection against natural phenomena
- Fire protection
- Environmental and dynamic effects design bases
- Instrumentation and control
- Containment design
- Protection system functions
- Residual heat removal
- Emergency core cooling
- Containment heat removal
- Inspection of containment heat removal system
- Testing of containment heat removal system
- Containment design basis

[5] In addition to the ARDC considered in this illustrative example, an applicant should consider rules in 10 CFR Part 50 or 52, as appropriate, as well as relevant guidance documents when preparing an application for an SDA.

Table 2: Interface Requirements for the RVAC SDA

<table>
<tr><td>ARDC Number</td><td>Title</td><td></td></tr>
<tr><td rowspan="2">1</td><td rowspan="2">Quality standards and records</td><td>Interface Requirement</td></tr>
<tr><td>The design of the RVAC will be required to be performed under a QA program. The QA program will have been submitted to the NRC as a topical report.

1. The FSAR will include justification that the approved QA program was followed during the development of the RVAC.

2. The FSAR will include justification for the safety classification of the SSCs of the RVAC.

3. Justification that the SSCs of the RVAC have been designed in accordance with their safety significance and the QA program.</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Design basis for protection against natural phenomena</td><td>Interface Requirement</td></tr>
<tr><td>The ability of the SSCs of the RVAC to withstand the design basis natural phenomena will be addressed in the FSAR. The comparison of the FSAR design assumptions to those relating to an actual site will be addressed in a future submission. Adequate margin should be included in the assumed values for the natural phenomena to provide flexibility in siting the design.

1. The FSAR will specify seismic, hurricane, and tornado design parameters (e.g., earthquake design response spectra, soil conditions, tornado and hurricane wind speeds, etc.). These parameters will be compared to those evaluated for a future site.</td></tr>
<tr><td rowspan="2">3</td><td rowspan="2">Fire protection</td><td>Interface Requirement</td></tr>
<tr><td>The RVAC is required to have a fire protection program. The fire protection program will be addressed in a future submission.

1. The FSAR will include a commitment that the materials used in the RVAC structure will use noncombustible and fire-resistant materials</td></tr>
</table>

		wherever practical, particularly in locations with SSCs important to safety.
4	Environmental and dynamic effects design bases	Interface Requirement
		The RVAC design is required to consider environmental and dynamic effects from pipe breaks and internal missiles. The design basis for these environmental and dynamic effects will be addressed in the FSAR. The analysis, performed in accordance with SRP methods, will be based on assumptions related to accident temperatures and pressures, pipe whipping, if applicable, and discharging gas or fluid. Adequate margin should be included in the assumed conditions to provide flexibility in the rest of the plant design. A future submission will compare assumed parameters to those determined from more accurate RCS design calculations and review of missile sources. 1. The FSAR will identify dynamic effects parameters (e.g., accident temperatures and pressures, pipe impact energy, discharging gas or fluid conditions).
5	Sharing of structures, systems, and components	No Dependence
10	Reactor design	No Dependence
11	Reactor inherent protection	No Dependence
12	Suppression of reactor power oscillations	No Dependence
13	Instrumentation and control	Interface Requirements
		The instrumentation and control (I&C) system will be addressed in a future submission. 1. The FSAR will include a summary of the instrumentation and controls that shall be provided, including the time needed for the system to perform the function, to ensure that the RVAC system can perform adequately. This information is used in analyses required under ARDC 34 and 35.
14	Reactor coolant boundary	No Dependence

15	Reactor coolant system design	No Dependence
16	Containment design	Interface Requirement The structure performing the containment function will be addressed in a future submission. 1. The FSAR will include information pertaining to the containment design that affects the design of the RVAC system, such as geometry, material selection, and temperature ranges during AOOs and postulated accidents.
17	Electric power systems	No Dependence
18	Inspection and testing of electric power systems	No Dependence
19	Control room	No Dependence
20	Protection system functions	Interface Requirement The protection system functions will be addressed in a future submission. 1. The FSAR will include information pertaining to the protection system functions that affects the design of the RVAC, its ability to perform its safety functions and information necessary for calculations performed under ARDC 34 and 35.
21	Protection system reliability and testability	No Dependence
22	Protection system independence	No Dependence
23	Protection system failure modes	No Dependence
24	Separation of protection and control systems	No Dependence
25	Protection system requirements for reactivity control malfunctions	No Dependence
26	Reactivity control systems	No Dependence
28	Reactivity limits	No Dependence
29	Protection against AOOs	No Dependence

30	Quality of reactor coolant boundary	No Dependence
31	Fracture prevention of reactor coolant boundary	No Dependence
32	Inspection of reactor coolant boundary	No Dependence
33	Reactor coolant inventory maintenance	No Dependence
34	Residual heat removal	Interface Requirement
		The design of the other SSCs of the residual heat removal system will be addressed in a future submission. 1. The FSAR will describe the role of the RVAC in residual heat removal and include calculations using an NRC approved method to demonstrate the RVAC can remove adequate decay heat to ensure SAFDLs and the design conditions of the reactor coolant pressure are not exceeded. 2. The FSAR will describe information pertaining the RCS behavior that affects the ability of the RVAC to perform its functions during shutdown and AOOs, such as vessel temperatures and required heat removal rates.
35	Emergency core cooling	Interface Requirement
		The design of the emergency core cooling system will be addressed in a future submission. 1. The FSAR will describe the role of the RVAC during postulated accidents and will use an NRC approved method to demonstrate the RVAC can remove adequate heat to ensure that fuel and clad damage that could interfere with continued effective core cooling is prevented. 2. The FSAR will describe information pertaining to the RCS behavior that affects the ability of the RVAC to perform its functions during postulated accidents, such as vessel temperatures and required heat removal rates.

36	Inspection of emergency core cooling system	No Dependence
37	Testing of emergency core cooling system	No Dependence
38	Containment heat removal	Interface Requirement The remaining features of the design of the containment heat removal system will be addressed in a future submission. 1. The FSAR will describe the role that the RVAC plays in removing heat from the containment during accident conditions. 2. The FSAR will describe information establishing the containment behavior that affects the ability of the RVAC to perform its functions during postulated accidents, such as vessel temperatures and required heat removal rates.
39	Inspection of containment heat removal system	Interface Requirement The inspection of the containment heat removal system other than the RVAC will be addressed in a future submission. 1. The FSAR will describe the ability of the RVAC system to be inspected.
40	Testing of containment heat removal system	Interface Requirement The testing of the containment heat removal system other than the RVAC will be addressed in a future submission. 1. The FSAR will describe the ability of the RVAC system to be tested.
41	Containment atmosphere cleanup	No Dependence
42	Inspection of containment atmosphere cleanup systems	No Dependence
43	Testing of containment atmosphere cleanup systems	No Dependence

44	Structural and equipment cooling	No Dependence
45	Inspection of structural and equipment cooling systems	No Dependence
46	Testing of structural and equipment cooling systems	No Dependence
50	Containment design basis	The remainder of the containment design basis will be addressed in a future submission. 1. The FSAR will describe the role the RVAC plays in ensuring the containment does not exceed its pressure and temperature limits during postulated accidents. 2. The FSAR will include calculations performed with NRC approved methods to demonstrate that the RVAC can remove the required heat from the containment during postulated accidents. 3. The FSAR will describe information pertaining to containment behavior that affects the ability of the RVAC to perform its functions during postulated accidents, such as containment vessel temperature and required heat removal rates.
51	Fracture prevention of containment pressure boundary	No Dependence
52	Capability for containment leakage rate testing	No Dependence
53	Provisions for containment testing and inspection	No Dependence
54	Piping systems penetrating containment	No Dependence
55	Reactor coolant boundary penetrating containment	No Dependence
56	Containment isolation	No Dependence
57	Closed system isolation valves	No Dependence

60	Control of releases of radioactive materials to the environment	No Dependence
61	Fuel storage and handling and radioactivity control	No Dependence
62	Prevention of criticality in fuel storage and handling	No Dependence
63	Monitoring fuel and waste storage	No Dependence
64	Monitoring radioactivity releases	No Dependence

5.3 Case 3 - Reactor Coolant System Piping Design

The scope of the RCS high-temperature piping design SDA involves the design, analysis, and testing of the piping system in accordance with Division 5, “Construction rules for high temperature reactors,” of Section III, “Rules for Construction of Nuclear Facility Components,” of the ASME Boiler and Pressure Vessel Code. This standard applies to nuclear facilities with service temperatures in excess of 700°F for ferritic materials and 800°F for austenitic stainless steels or high nickel alloys. (It should be noted that although work is underway, it has not yet been approved by the NRC.) The scope of the example also includes piping support evaluations and piping component fatigue analysis. While this generic example assumes a conventional external piping system, it is noted that there will be some advanced reactor designs that will use an integral design for the RCS which would preclude the need for a piping system that performs a safety function. For those designs that do not use an integral RCS and rely on a coolant loop that is part of the pressure boundary, carrying high-temperature gas or fluid from the core exit to a heat exchanger, the example may apply.

In this example, a less detailed design of the piping system is needed because the application for an SDA will focus on methods of design and analysis, rather than the detailed piping design and is similar to the second category of SDAs discussed in the 2017 NIA report. The example assumes the function of the RCS piping is to provide coolant to the reactor core and to ensure that that the design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation, including AOOs. In addition, the reactor coolant boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The provisions of the Section III, Division 5 standard, discussed above, should address this requirement.

The detailed piping design, which requires detailed design information relating to building design and location of key RCS components (e.g., reactor vessel, heat exchangers, and coolant pumps, if used), is assumed to not be available until later in the design process. However, the analysis of design pressures, thermal expansion, seismic demands, pipe breaks, creep rupture, and cyclic fatigue can be performed using assumed piping design parameters (e.g., component masses, piping material properties, diameter, and wall thickness). Interface requirements, identified in the application for an SDA, will ensure that key piping design parameters are satisfied once the reactor building design is finalized.

The design description of the RCS piping system in the FSAR is primarily focused on analysis methods, codes and standards relating to detailed design, and in-service inspection procedures. The level of detail in these descriptions should be sufficient to describe key analysis tools, design codes and standards, and key assumptions. Further, these FSAR descriptions should clearly identify how the design specific PDC are satisfied.

The analysis of the RCS will consider the effects of natural hazards (seismic) as well as normal operating and accident loads (pipe whip, if applicable, accident pressures and temperatures, etc.). Key design assumptions, such as those relating to seismic demands, operating temperatures, design pressures, pipe breaks (or leak before break if applicable), thermal stratification, and cyclic fatigue, should include adequate margin to ensure the piping design is enveloped by future site-specific parameters, such as seismic loads.

The example is intended to illustrate the process rather than yield a complete set of interface requirements. Furthermore, the example considers only the regulatory requirements specified in the general advanced reactor ARDC and does not consider any rules contained in 10 CFR Part 50 or 52.[6] Interface requirements are necessary so that the advanced reactor cooling piping system design parameters can reference or be referenced by other SDAs (e.g., reactor building and core design SDAs, etc.). The key design and analysis assumptions are also incorporated into the interface requirements, as appropriate. In the case of the reactor coolant piping system SDA, example interface requirements are described in Table 3, below. Requirements were identified in ten key areas of the ARDC:

- Quality standards and records
- Design bases for protection against natural phenomena
- Environmental and dynamic effects design bases
- Reactor coolant boundary
- Reactor coolant system design
- Quality of reactor coolant boundary
- Fracture prevention of reactor coolant boundary
- Inspection of reactor coolant inventory maintenance
- Piping systems penetrating containment
- Reactor coolant boundary penetrating containment

The FSAR will describe how the reactor cooling system piping design satisfies the design specific PDC. In addition, the FSAR will describe the methods of design and analysis. For the

[6] In addition to the ARDC considered in this illustrative example, an applicant should consider rules in 10 CFR Part 50 or 52, as appropriate, as well as relevant guidance documents when preparing an application for an SDA.

piping system example, existing regulatory guidance such as the SRP may be referenced to the extent practicable, until advanced reactor specific guidance is issued.

Table 3: Interface Requirements for the Reactor Coolant System Piping Design SDA

ARDC Number	Title	
1	Quality standards and records	Interface Requirement
		The design of the RCS piping will be required to be performed under a QA program. The QA program will have been submitted to the NRC as a topical report. 1. The FSAR will include justification that the approved QA program was followed during the development of the piping system. 2. The FSAR will include justification for the safety classification of the SSCs included in the piping system. 3. The FSAR will include justification that the SSCs included in the piping system have been designed in accordance with their safety significance and the QA program.
2	Design basis for protection against natural phenomena	Interface Requirement
		The RCS piping design will consider the effects of natural phenomena, such as earthquake events. The analysis will be based on assumptions relating to natural phenomena and external events. Adequate margin should be included in the assumed values for the natural phenomena to provide flexibility in the design. The comparison of the SDA design assumptions to those relating to an actual site will be addressed in a future submission. 1. The FSAR will describe seismic, hurricane, and tornado design parameters (e.g., earthquake design response spectra, soil conditions, tornado and hurricane wind speeds, etc.) that affect the piping system design.
3	Fire protection	No Dependence

4	Environmental and dynamic effects design bases	Interface Requirement The RCS piping design is required to consider environmental and dynamic effects from pipe breaks and internal missiles. The design bases for these environmental and dynamic effects will be addressed in the SDA. The analysis will be based on assumptions related to accident temperatures and pressures, pipe whipping, if applicable, and discharging gas or fluid. Adequate margin should be included in the assumed values to provide flexibility in the design of other portions of the plant. A future submission will compare assumed parameters to those determined from more accurate RCS design calculations and review of missile sources. 1. The FSAR will describe dynamic effects parameters (e.g., accident temperatures and pressures, pipe impact energy, discharging gas or fluid conditions) that affect the piping system design.
5	Sharing of structures, systems, and components	No Dependence
10	Reactor design	No Dependence
11	Reactor inherent protection	No Dependence
12	Suppression of reactor power oscillations	No Dependence
13	Instrumentation and control	No Dependence
14	Reactor coolant boundary	Interface Requirement The RCS piping will be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The design bases for the piping will be addressed in the SDA. The analysis will be based on assumptions related to required design temperature and pressure demands. A future submission will include the design of the remainder of the reactor coolant boundary and will compare the assumed RCS parameters used in the piping design to those

		determined from more accurate RCS design calculations. 1. The FSAR will describe RCS parameters that affect the piping design, such as temperatures, pressures, flow rates, etc.
15	Reactor coolant system design	Interface Requirement The RCS piping will be designed to have sufficient margin to ensure the design conditions of the reactor coolant boundary are not exceeded during any condition of normal operation, including AOOs. The design of the piping will be included in the FSAR. A future submission will include the design of the remainder of the reactor coolant system design. 1. The FSAR will describe conditions developed during normal operation and AOOs that will affect the piping system design, such as pressurization or depressurization rates, temperature ranges, etc.
16	Containment design	No Dependence
17	Electric power systems	No Dependence
18	Inspection and testing of electric power systems	No Dependence
19	Control room	No Dependence
20	Protection system functions	No Dependence
21	Protection system reliability and testability	No Dependence
22	Protection system independence	No Dependence
23	Protection system failure modes	No Dependence
24	Separation of protection and control systems	No Dependence
25	Protection system requirements for reactivity control malfunctions	No Dependence
26	Reactivity control systems	No Dependence
28	Reactivity limits	No Dependence

29	Protection against AOOs	No Dependence
30	Quality of reactor coolant boundary	Interface Requirement The quality of the reactor coolant boundary will be addressed in a future submission. 1. The FSAR will include justification that the piping system has been designed and will be fabricated, erected, and tested to the highest quality standards. 2. The FSAR will include a commitment that a means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage coming from the piping system.
31	Fracture prevention of reactor coolant boundary	Interface Requirement The fracture prevention of the remaining portion of the reactor coolant boundary functions will be addressed in a future submission. 1. The FSAR will include justification that the piping system shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the piping system behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. 2. The FSAR shall include justification that the piping design reflects consideration of service temperatures, service degradation of material properties, creep, fatigue, stress rupture, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws. 3. The FSAR will describe the conditions developed during operating, maintenance, testing and postulated accident conditions that

		affect the fracture prevention of the piping system.
32	Inspection of reactor coolant boundary	Interface Requirement
		Inspection of the reactor coolant boundary will be addressed in a future submission. 1. A commitment that components that are part of the reactor coolant boundary shall be designed to permit (1) periodic inspection and functional testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor vessel.
33	Reactor coolant inventory maintenance	No Dependence
34	Residual heat removal	No Dependence
35	Emergency core cooling	No Dependence
36	Inspection of emergency core cooling system	No Dependence
37	Testing of emergency core cooling system	No Dependence
38	Containment heat removal	No Dependence
39	Inspection of containment heat removal system	No Dependence
40	Testing of containment heat removal system	No Dependence
41	Containment atmosphere cleanup	No Dependence
42	Inspection of containment atmosphere cleanup systems	No Dependence
43	Testing of containment atmosphere cleanup systems	No Dependence
44	Structural and equipment cooling	No Dependence
45	Inspection of structural and equipment cooling systems	No Dependence

46	Testing of structural and equipment cooling systems	No Dependence
50	Containment design basis	No Dependence
51	Fracture prevention of containment pressure boundary	No Dependence
52	Capability for containment leakage rate testing	No Dependence
53	Provisions for containment testing and inspection	No Dependence
54	Piping systems penetrating containment	Interface Requirement
		Since the design of the structure performing the containment function is not assumed to be complete, the design of piping systems penetrating the containment structure will be addressed in a future submission. 1. The FSAR will include a commitment that piping systems penetrating the containment structure shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems. 2. The FSAR will include a commitment that such piping systems shall be designed with the capability to verify, by testing, the operational readiness of any isolation valves and associated apparatus periodically and to confirm that valve leakage is within acceptable limits.
55	Reactor coolant boundary penetrating containment	Interface Requirement
		Since the design of the structure performing the containment function is not assumed to be complete, the design of containment structure penetrations and isolation valves will be addressed in a future submission. 1. The FSAR will include a commitment that each line of the piping system that is part of the reactor coolant boundary and that

		penetrates the containment structure shall be provided with containment isolation valves, as follows, unless it can be demonstrated that the containment isolation provisions for the piping system are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. 2. The FSAR will include a commitment that other appropriate requirements to minimize the probability or consequences of an accidental rupture of the piping system or of lines connected to them shall be provided as necessary to ensure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing; additional provisions for in-service inspection; protection against more severe natural phenomena; and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.
56	Containment isolation	No Dependence
57	Closed system isolation valves	No Dependence
60	Control of releases of radioactive materials to the environment	No Dependence
61	Fuel storage and handling and radioactivity control	No Dependence

62	Prevention of criticality in fuel storage and handling	No Dependence
63	Monitoring fuel and waste storage	No Dependence
64	Monitoring radioactivity releases	No Dependence

5.4 Case 4 - Reactor Building Structural Design

The interface requirements for an application for an SDA of the structural design of a reactor building are provided in this section. This example reactor building is similar to that used in the NuScale SMR plant design to enclose the collection of multiple reactor modules in the coolant pool. In this design, the reactor building is not a leak tight, pressure-retaining structure but provides robust protection against natural hazards. It is assumed in this example that the location (or siting) of the advanced reactor has not yet been determined. The application for an SDA of the building structure would therefore include some details pertaining to the structural design of the reactor building and higher-level assumptions pertaining to the site characteristics that affect the structural design. This case is therefore an example of the third category of SDAs discussed in the 2017 NIA report. There are potential schedule and economic advantages in developing the building design in advance of (or simultaneously with) the site selection process.

The scope of the reactor building SDA example involves the design and analysis of a hypothetical Seismic Category I reactor building. The reactor building is assumed to house the reactor, the structure that performs the containment function, if used, and supporting systems and components. The function of the reactor building is assumed to provide protection from natural hazards and support for internal systems and components.

The design description of the reactor building in the FSAR is primarily focused on analysis methods and codes and standards relating to detailed design and descriptions of representative sections of the structural design. The NRC refers to these sections as critical sections. The level of detail in these descriptions should be sufficient to describe key analysis tools, design codes and standards, and key design assumptions. The design of the reactor building will consider the effects of natural hazards (e.g., seismic and wind) as well as normal operating and accident loads (pipe whip, accident pressures and temperatures, etc.). Key design assumptions, such as those relating to the seismic design spectra, maximum tornado and hurricane wind speeds, range of meteorological conditions, and soil properties, should include some margin, such that design can be constructed at a range of sites. It is noted that HTRs using TRISO fuel may credit the fission product retention feature of the fuel to serve as functional containment.

Many advanced designs operate at pressures well below LWR pressures, so that energetic blowdowns are precluded in the event of a pipe break. As a result, the containment function in such designs will likely be of lighter weight than a traditional LWR containment. Nonetheless, in some cases, these structures will be of sufficient size and mass that they will require explicit consideration in the reactor building design and dynamic analysis models. For all designs, the RCS will need to be considered in the design and analysis models as well.

The NRC reviews LWR structural design and analysis methods in accordance with NUREG--0800 Chapter 3.5, “Barrier Design,” Chapter 3.7, “Seismic Design,” and Chapter 3.8, “Structural Design.” As the criteria in these SRP sections are independent of any particular reactor design, they will likely remain applicable to the design and analysis of an advanced reactor building.

For this case, the reactor building is assumed to be constructed with either reinforced concrete or steel plate composite (SC) modules. Reinforced concrete structures for nuclear facilities are designed in accordance with American Concrete Institute (ACI) 349; structural steel is designed to American Institute of Steel Construction (AISC) N690. The use of SC modules is increasingly common in new reactor designs and offers the potential advantage of reducing construction costs and schedule. Appendix N9 to AISC N690 and AISC Design Guide 32, “Design of Modular Steel-Plate Composite Walls for Safety-Related Nuclear Facilities,” provide guidance for the design of SC structures.

The description of the reactor building design in the FSAR will include descriptions of critical sections. Critical sections, which represent typical reactor building walls, floors, and basemat, are provided to specify the essential design features of the building, such as overall wall and floor thicknesses and location of steel reinforcement. The description of critical sections should contain a basic level of information, such as nominal wall thickness, minimum concrete strength, and minimum required area of steel reinforcement. However, these descriptions should not be overly detailed (e.g., specifying rebar size and spacing), so that flexibility during the detailed design phase and during fabrication and construction is maximized.

The process for developing interface requirements for the application for an SDA of the structural design of the reactor building involves the comparison of the SDA scope to the PDC. In this example, the ARDC listed in RG 1.232 are a substitute for the PDC. The example is intended to illustrate the process rather than yield a complete set of interface requirements. Furthermore, the example considers only the regulatory requirements specified in the general advanced reactor ARDC and does not consider any rules contained in 10 CFR Parts 50 or 52.[7] For the reactor building example, interface requirements were identified in six key areas of the ARDC:

- Quality standards and records
- Design bases for protection against natural phenomena
- Fire protection
- Environmental and dynamic effects design bases

[7] In addition to the ARDC considered in this illustrative example, an applicant should consider rules in 10 CFR Part 50 or 52, as appropriate, as well as relevant guidance documents when preparing an application for an SDA.

- Reactor coolant system design
- Containment design

The FSAR will describe how the reactor building design satisfies the PDC. In addition, it will describe the methods of design and analysis. For the structural design of the reactor building, existing regulatory guidance such as the SRP may be referenced to the extent that it is applicable to the structural design of the advanced reactor. Alternatively, a vendor may opt to propose new criteria tailored to their design in place of the SRP.

Table 4: Interface Requirements for the Reactor Building Structural Design SDA

ARDC Number	Title	
1	Quality standards and records	Interface Requirement
		The design of the reactor building will be required to be performed under a QA program. The QA program will have been submitted as a topical report to the NRC. 1. The FSAR will include justification that the approved QA program was followed during the development of the reactor building structural design. 2. The FSAR will include justification for the safety classification of the structures in the reactor building design. 3. The FSAR will include justification that the structures included in the reactor building have been designed in accordance with their safety significance and the QA program.
2	Design basis for protection against natural phenomena	Interface Requirement
		The design basis natural phenomena and the ability of the SSCs to withstand the design basis natural phenomena will be addressed in the SDA. The comparison of the SDA design assumptions to those relating to an actual site will be addressed in a future site-specific submission. Adequate margin should be included in the assumed values for the natural phenomena to provide flexibility in the design. 1. The FSAR will specify seismic, hurricane, and tornado design parameters (e.g., earthquake

		design response spectra, soil conditions, tornado and hurricane wind speeds, etc.) that affect the structural design of the reactor building. These parameters will be compared to those evaluated for a future site.
3	Fire protection	Interface Requirement
		The reactor building is required to have a fire protection program. The fire protection program will be addressed in a future submission. 1. The FSAR will indicate that the materials used in the reactor building structure will use noncombustible and fire-resistant materials wherever practical, particularly in locations with SSCs important to safety.
4	Environmental and dynamic effects design bases	Interface Requirement
		The reactor building structural design is required to consider environmental and dynamic effects from pipe breaks and internal missiles. The design bases for these environmental and dynamic effects will be addressed in the SDA. The analysis will be based on assumptions related to accident temperatures and pressures, pipe whipping, if applicable, and discharging gas or fluid. Adequate margin should be included in the assumed to provide flexibility in the design of other portions of the plant. A future submission will compare assumed parameters to those determined from more accurate RCS design calculations and review of missile sources. 1. The FSAR will describe dynamic effects parameters (e.g., accident temperatures and pressures, pipe impact energy, discharging gas or fluid conditions) that affect the structural design of the reactor building.
5	Sharing of structures, systems, and components	No Dependence
10	Reactor design	No Dependence
11	Reactor inherent protection	No Dependence

12	Suppression of reactor power oscillations	No Dependence
13	Instrumentation and control	No Dependence
14	Reactor coolant boundary	No Dependence
15	Reactor coolant system design	Interface Requirement
		Due to its assumed large size, the RCS is represented as a sub-system in the dynamic analysis of the reactor building. Key RCS parameters, such as mass and stiffness, will be represented in the seismic analysis models. 1. The FSAR will describe RCS structural response characteristics such as component mass, stiffness, and location that affect the structural design of the reactor building.
16	Containment design	Interface Requirement
		Due to its assumed large size, the containment is represented as a sub-system in the dynamic analysis of the reactor building. Key containment parameters, such as mass and stiffness, will be represented in the seismic analysis models. 1. The FSAR will identify the structural response characteristics of the structure performing the containment function such as structure mass and stiffness that affect the structural design of the reactor building.
17	Electric power systems	No Dependence
18	Inspection and testing of electric power systems	No Dependence
19	Control room	No Dependence
20	Protection system functions	No Dependence
21	Protection system reliability and testability	No Dependence
22	Protection system independence	No Dependence

23	Protection system failure modes	No Dependence
24	Separation of protection and control systems	No Dependence
25	Protection system requirements for reactivity control malfunctions	No Dependence
26	Reactivity control systems	No Dependence
28	Reactivity limits	No Dependence
29	Protection against AOOs	No Dependence
30	Quality of reactor coolant boundary	No Dependence
31	Fracture prevention of reactor coolant boundary	No Dependence
32	Inspection of reactor coolant boundary	No Dependence
33	Reactor coolant inventory maintenance	No Dependence
34	Residual heat removal	No Dependence
35	Emergency core cooling	No Dependence
36	Inspection of emergency core cooling system	No Dependence
37	Testing of emergency core cooling system	No Dependence
38	Containment heat removal	No Dependence
39	Inspection of containment heat removal system	No Dependence
40	Testing of containment heat removal system	No Dependence
41	Containment atmosphere cleanup	No Dependence
42	Inspection of containment atmosphere cleanup systems	No Dependence
43	Testing of containment atmosphere cleanup systems	No Dependence
44	Structural and equipment cooling	No Dependence

45	Inspection of structural and equipment cooling systems	No Dependence
46	Testing of structural and equipment cooling systems	No Dependence
50	Containment design basis	No Dependence
51	Fracture prevention of containment pressure boundary	No Dependence
52	Capability for containment leakage rate testing	No Dependence
53	Provisions for containment testing and inspection	No Dependence
54	Piping systems penetrating containment	No Dependence
55	Reactor coolant boundary penetrating containment	No Dependence
56	Containment isolation	No Dependence
57	Closed system isolation valves	No Dependence
60	Control of releases of radioactive materials to the environment	No Dependence
61	Fuel storage and handling and radioactivity control	No Dependence
62	Prevention of criticality in fuel storage and handling	No Dependence
63	Monitoring fuel and waste storage	No Dependence
64	Monitoring radioactivity releases	No Dependence

6.0 Conclusions

A method has been developed to specify the interface requirements for major portions of a design submitted under the SDA process (Subpart E to 10 CFR Part 52). The method is compatible with existing licensing guidance for advanced reactors, such as NRC RG 1.232, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*, and the April 2017 NIA report, which provides guidance on defining major portions of an advanced reactor design. It is also expected to be compatible with the results of DG-1353, *Guidance for A Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors*.

The process for developing interface requirements is relatively straightforward. Once a vendor clearly defines the scope of the SDA and develops design requirements, the detailed design will be performed in accordance with ARDC, rules, guidance, and industry codes and standards applicable to those SSCs included in the major portion. Key analysis assumptions and design parameters, as well as an assessment of relevant NRC regulations and ARDC, will be used to develop interface requirements. Interface requirements should be described in the FSAR submitted as part of the application for an SDA to enhance the efficiency of the review. Although not depicted in Figure 1, the design process is likely to be iterative as design choices are made.

Four illustrative examples of establishing these interfaces, including a core design, an RVAC design, a high-temperature piping system, and the structural design of a reactor building are presented to aid a vendor in applying this process to a specific design.

While every effort has been made to develop a robust and generically applicable process for identifying interface requirements, stakeholder input and experience will be used to make revisions and add clarification to the process described herein.

7.0 Table of Acronyms

ac	Alternating Current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AISC	American Institute of Steel Construction
AOO	Anticipated Operational Occurrences
ARDC	Advanced Reactor Design Criteria
ASME	American Society of Mechanical Engineers
CDI	Conceptual Design Information
COL	Combined License
CP	Construction Permit
DC	Design Certification
dc	Direct current
DOE	Department of Energy
HTGR	High-temperature Gas-cooled Reactor
HTR	High-temperature Reactor
I&C	Instrumental and control system
LMP	Licensing Modernization Project
LWR	Light Water Reactor
MSR	Molten Salt Reactor
NIA	Nuclear Innovation Alliance
NRC	Nuclear Regulatory Committee
OP	Operating Permit
PDC	Principal Design Criteria
QA	Quality Assurance
RCS	Reactor Coolant System
RVAC	Reactor Vessel Auxiliary Cooling
SAFDL	Specified Acceptable Fuel Design Limit
SC	Steel Plate Composite
SDA	Standard Design Approval
SER	Safety Evaluation Report
SFR	Sodium Fast Reactor
SRP	Standard Review Plan
SSC	Systems, structures, and components
TRISO	Tristructural-Isotropic

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