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NUREG-2246

Fuel Qualification for Advanced Reactors

Draft Report for Comment

Office of Nuclear Reactor Regulation

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ABSTRACT

Proposed advanced reactor designs use fuel designs and operating environments (e.g., neutron energy spectra, fuel temperatures, neighboring materials) that differ from the large experience base available for traditional light-water reactor fuel. The purpose of this report is to identify criteria that will be useful for advanced reactor designs through an assessment framework that would support regulatory findings associated with nuclear fuel qualification. The report begins by examining the regulatory basis and related guidance applicable to fuel qualification, noting that the role of nuclear fuel in the protection against the release of radioactivity for a nuclear facility depends heavily on the reactor design. The report considers the use of accelerated fuel qualification techniques and lead test specimen programs that may shorten the timeline for qualifying fuel for use in a nuclear reactor at the desired parameters (e.g., burnup). The assessment framework particularly emphasizes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of the experimental data used to develop and validate evaluation models and empirical safety criteria.

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ABBREVIATIONS AND ACRONYMS

AFQ	accelerated fuel qualification
AOO	anticipated operational occurrence
ARDC	advanced reactor design criterion
ED	experimental data
EM	evaluation model
FAST	fission accelerated steady-state test
FQAF	fuel qualification assessment framework
G	goal
GDC	general design criterion/criteria
GESTAR	General Electric standard application for reactor fuel
LWR	light-water reactor
OBE	operating basis earthquake
PCMI	pellet-clad mechanical interaction
PCMM	predictive capability maturity model
PIRT	phenomena identification and ranking table
SAFDL	specified acceptable fuel design limit
SARRDL	specified acceptable radionuclide release design limit
SSC	structure, system, and component
SSE	safe shutdown earthquake
TRISO	tristructural-isotropic
U-10Zr	uranium alloy with 10 weight percent zirconium
U-Pu-10Zr	uranium-plutonium alloy with 10 weight percent zirconium
UO_2	uranium dioxide

1 INTRODUCTION

1.1 Purpose

The objective of nuclear fuel qualification is to "demonstrat[e] that a fuel product fabricated in accordance with a specification behaves as assumed or described in the applicable licensing safety case, and with the reliability necessary for economic operation of the reactor plant" (Crawford, et al., 2007). Proposed advanced reactor designs have fuel designs and operating environments (e.g., neutron energy spectra, fuel temperatures, neighboring materials) that differ from the large experience base available for traditional light-water reactor (LWR) fuel. Nuclear fuel affects many aspects of the overall design of a nuclear power plant, and qualification of nuclear fuel has traditionally involved long development times. The purpose of this report is to provide a fuel qualification assessment framework for use with advanced reactor designs that satisfies regulatory requirements. Specifically, the framework provides criteria derived from regulatory requirements that, when satisfied, would support regulatory findings necessary for licensing. The framework follows a top-down approach in which a set of base goals[1] support high-level regulatory requirements.[2] This report provides the bases for the identified "base goals" and clarifying examples for the types of information that an applicant would need to provide in order for the NRC to determine that these goals are satisfied and regulatory requirements are met. Appendix A lists all goals within the framework.

This framework relies on regulatory requirements that are applicable to applications for design certifications, combined licenses, manufacturing licenses, or standard design approvals. While the requirements of Title 10 of the Code of Federal Regulations (10 CFR) 50.43(e) are not applicable to applications for a construction permit, the remaining requirements, identified in Section 2.1, are generically applicable to power reactor applications. Accordingly, the framework provides applicants with criteria for satisfying regulatory requirements for applications for a design certification, combined license, manufacturing license, standard approval, and for the development of a fuel qualification plan to support a construction permit application.

1.2 Safety Case

The role of nuclear fuel in the protection against the release of radioactivity can vary depending on the reactor design[3]. For example, facilities that use traditional oxide fuels with metal cladding are designed with robust barriers (e.g., containment buildings) to prevent the release of radioactive material under postulated accident conditions, whereas a facility that uses tristructural-isotropic (TRISO) fuel may credit a series of barriers (including barriers within the fuel itself) to prevent the release of radioactive material (i.e., a functional containment (NRC, 2018a)). Thus, in the nuclear fuel qualification process, it is essential to specify the fission product retention functions of the nuclear fuel (this is addressed under Goal (G) 2, "Safety Criteria," in Section 3.2 of this report).

[1] A base goal is a goal that is not decomposed any further but is supported by evidence.
[2] "High-level" in this context refers to its position in the framework. Regulatory requirements are located near the top of the framework and lower-level goals are provided that, if satisfied, provide bases for satisfying the regulatory requirements.
[3] Fuel qualification literature often use the term "safety case". This term is undefined but generally refers to the safety functions that the fuel is relied upon to perform. Principally among these safety-functions is the protection against the release of radionuclides.

1.3 Scope

Nuclear fuel affects many aspects of nuclear safety, including neutronic performance (e.g., reactivity feedback), thermal-fluid performance (e.g., margin to critical heat flux limits), fuel mechanical performance, reactor core seismic behavior, fuel transportation, and storage. The scope of this report focuses on the identification and understanding of fuel life-limiting failure and degradation mechanisms due to irradiation during reactor operation. The assessment criteria in Section 3 of this report draw on regulatory experience gained from licensing solid fuel reactor designs (particularly LWR designs), results from advanced reactor fuel testing performed to-date, and accelerated fuel qualification (AFQ) considerations. An attempt has been made to develop generically applicable criteria. However, some criteria may not apply to liquid fuel forms (e.g., molten salt reactor fuel), and these fuel forms may require additional or alternate criteria (see Section 2.2.4 for guidance on molten salt reactor fuel).

2 BACKGROUND

2.1 Regulatory Basis

Nuclear fuel qualification to support reactor licensing involves the development of a basis to support findings associated with regulatory requirements that apply to the nuclear facility. This section discusses these requirements and their relationship to this report. Note that satisfying the fuel qualification framework criteria only “partially addresses” the requirements associated with the nuclear facility. This is because the fuel qualification framework provides a means to identify the safety criteria for the fuel and it is the safety criteria for the fuel that establish the performance criteria for some structures, systems, and components (SSCs) of the facility. Therefore, addressing the criteria in the fuel qualification framework provides the information necessary to meet the regulations, but does not in and of itself satisfy regulatory requirements. The requirements are fully addressed through the description and analysis of these SSCs in an application.

The relevant regulatory requirements are as follows:

- 10 CFR 50.43(e)(1)(i) requires demonstration of the performance of each safety feature of the design through either analysis, appropriate test programs, experience, or a combination thereof. The assessment framework developed in Section 3 of this report (1) provides a means to identify the safety features of the fuel necessary to comply with regulatory requirements (see Goal (G) 2, “Safety Criteria,” in Section 3.2), and (2) clarifies the types of evidence (e.g., analysis, testing, experience) typically expected to demonstrate these safety features. In accordance with the scope of this report, the safety features assessed in Section 3 are associated with the identification and understanding of fuel life-limiting failure and degradation mechanisms that are due to irradiation during reactor operation.

- The regulation in 10 CFR 50.43(e)(1)(iii) requires that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. This range appears in G2.1.1, “Definition of Fuel Performance Envelope,” which is discussed in Section 3.2.1.1 of this report. Additionally, the evaluation model assessment framework in Section 3.3 provides criteria for assessing analytical tools, and the experimental data assessment framework in Section 3.4 provides criteria for data adequacy.

- General Design Criterion (GDC) 2 and Advanced Reactor Design Criterion[4] (ARDC) 2, “Design bases for protection against natural phenomena,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic licensing of production and utilization facilities,” requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Appendix S to 10 CFR 50, “Earthquake engineering criteria for nuclear power plants,” implements GDC 2 as it pertains to seismic events and defines specific

[4] Regulatory Guide 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” (NRC, 2018b) provides guidance on how the GDC in Appendix A to 10 CFR Part 50 may be adapted for non-LWR designs.

earthquake criteria for nuclear power plants. This appendix established definitions for safe shutdown earthquake (SSE), operating basis earthquake (OBE), and safety requirements for relevant SSCs. These SSCs are necessary to assure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe-shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures. The safety functions generally associated with nuclear fuel include control of reactivity, cooling of radioactive material, and confinement of radioactive material[5]. The requirements related to natural phenomena can be partially addressed by satisfying G2.3, “Safe Shutdown,” which is discussed in Section 3.2.3.

- GDC 10 and ARDC 10, “Reactor Design,” require that specified acceptable fuel design limits (SAFDLs) or specified acceptable radionuclide release design limits (SARRDLs) not be exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). This requirement can be partially addressed by satisfying G2.1, “Design Limits during Normal Operation and AOOs,” which is discussed in Section 3.2.1.

- GDC 27 and ARDC 26, “Combined Reactivity Control Systems Capability,” require, in part, the ability to achieve and maintain safe shutdown under postulated accident conditions and assurance that the capability to cool the core is maintained. This requirement can be partially addressed by satisfying G2.3, “Safe Shutdown,” which is discussed in Section 3.2.3.

- GDC 35 and ARDC 35, “Emergency Core Cooling,” require an emergency core cooling system that provides sufficient cooling under postulated accident conditions; they also require that fuel and clad damage that could interfere with continued effective core cooling is prevented. This requirement can be partially addressed by satisfying G2.3, “Safe Shutdown,” which is discussed in Section 3.2.3.

- The regulations in 10 CFR 50.34(a)(1)((ii)(D), 10 CFR 52.47(a)(2)(iv), and 10 CFR 52.79(a)(1)(vi) require an evaluation of a postulated fission product release. This requirement can be partially addressed by satisfying G2.2, “Radionuclide Release Limits,” which is discussed in Section 3.2.2.

The fuel qualification assessment framework in Section 3 of this report provides guidance to facilitate an efficient and transparent licensing review in the area of fuel qualification. The guidance provided in this report is not a substitute for the Commission’s regulations, and compliance with the guidance is not required.

2.2 Related Guidance

Several guidance documents are available or are in development that address nuclear fuel qualification. This section discusses these guidance documents and their relationship to this report.

[5] These “fundamental safety functions” are identified in the IAEA safety glossary (IAEA, 2018) and are also incorporated into NRC regulations. Reactivity control is specified by GDC 27 and ARDC 26; heat removal is specified by GDC/ARDC 10, GDC 27, ARDC 26, and GDC/ARDC 35; radionuclide retention is specified by GDC/ARDC 10 and is associated with the requirements under 10 CFR 50.34(a)(1)((ii)(D), 10 CFR 52.47(a)(2)(iv), and 10 CFR 52.79(a)(1)(vi).

2.2.1 NUREG-0800, Section 4.2

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 4.2, Revision 3, "Fuel System Design," issued March 2007 (NRC, 2007), lists acceptance criteria that staff considers in a licensing review for a LWR fuel system. Section 3.2 of this report captures the objectives of the fuel system safety review as follows:

- Assurance that the fuel system is not damaged as a result of normal operation and AOOs can be demonstrated, in part, by meeting G2.1, "Design Limits during Normal Operation and AOOs," which is discussed in Section 3.2.1.

- Assurance that fuel system damage is never so severe as to prevent control element insertion when required can be demonstrated, in part, by meeting G2.3, "Safe Shutdown," which is discussed in Section 3.2.3. Section 3.2.3.2 discusses the specific item of control element insertion.

- Assurance that the number of fuel rod failures is not underestimated for postulated accidents can be demonstrated, in part, by meeting G2.2, "Radionuclide Release Limits," which is discussed in Section 3.2.2.

- Assurance that coolability is always maintained can be demonstrated, in part, by meeting G2.3, "Safe Shutdown," which is discussed in Section 3.2.3. Section 3.2.3.1 discusses the specific item of maintaining a coolable geometry.

NUREG-0800, Section 4.2, provides guidance regarding traditional LWR fuel and the licensing bases for traditional LWR power plants. Specifically, NUREG-0800, Section 4.2, evaluates fuel system designs for known fuel failure mechanisms from traditional LWR fuel (i.e., uranium dioxide (UO_2) fuel with zirconium-alloy cladding), identifies specific testing for addressing key LWR fuel phenomena, and includes empirical acceptance criteria based on testing of LWR fuel samples. As such, the specific acceptance criteria provided in NUREG-0800, Section 4.2, may not apply or may not suffice to address advanced reactor technologies that use different fuel forms, or address situations in which the fuel plays different roles in the protection against the release of radionuclides. However, this report incorporates lessons learned from the development of the acceptance criteria in NUREG-0800, Section 4.2, as follows:

- The significant effect of fuel manufacturing parameters on fuel performance is addressed through G1, "Fuel Manufacturing Specification," which is discussed in Section 3.1.

- Limitations on test facilities and the risks of obtaining irradiated fuel data are discussed in the experimental data assessment framework in Section 3.4 and are also mentioned in Section 3.2.2.3.1.

2.2.2 ATF-ISG-2020-01

ATF-ISG-2020-01, "Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept," issued January 2020 (NRC, 2020a), provides supplementary guidance to NUREG-0800, Section 4.2. The guidance was developed using a phenomena identification and ranking table (PIRT) process[6] and is specific to applications involving fuel products with chromium-coated zirconium alloy cladding. Like the guidance in NUREG-0800, Section 4.2, the specific phenomena identified in ATF-ISG-2020-01 may not apply to advanced reactor technologies. However, the PIRT process may be used to identify failure mechanisms and necessary features of an evaluation model, as discussed in the evaluation model assessment framework in Section 3.3 of this report.

2.2.3 Regulatory Guide 1.233

Regulatory Guide 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," issued June 2020 (NRC, 2020b), provides guidance for a modern, risk-informed approach to licensing reviews. This approach emphasizes assessing facility risk by quantifying event frequencies and the associated radiological consequences. The consequence evaluation aspect of the risk assessment is addressed, in part, by G2.2, "Radionuclide Release Limits," which is discussed in Section 3.2.2.

Additionally, Regulatory Guide 1.233 discusses fundamental safety functions. Fuel qualification partially addresses the fundamental safety functions of control of reactivity, cooling of radioactive material, and confinement of radioactive material by incorporating the role of the fuel in these safety functions in G2, "Safety Criteria," which is discussed in Section 3.2 of this report, as follows:

- Confinement of radioactive material is partially addressed by G2.1, "Design Limits during Normal Operation and AOOs," and G2.2, "Radionuclide Release Limits."

- Control of reactivity and cooling or radioactive material are partially addressed by G2.3, "Safe Shutdown."

2.2.4 Guidance in Development

The U.S. Nuclear Regulatory Commission (NRC) staff is currently developing guidance in additional areas related to fuel qualification. As discussed in Section 1.3, the safety case for reactors that use nonsolid fuel forms may require additional or alternative criteria to those in this report. To that end, the NRC is supporting the development of a proposed methodology for molten salt reactor fuel salt qualification (ORNL, 2018) (ORNL, 2020).

Additionally, G2 addresses the role of the fuel in the protection against the release of radioactivity, as discussed in Section 1.2. G2 is supported by source term considerations, as detailed in G2.2.1, "Radionuclide Retention Requirements," and G2.2.3, "Conservative Modeling of Radionuclide Retention and Release." Furthermore, G2.1, "Design Limits during Normal Operation and AOOs," discusses SARRDLs, which involve the use of a source term. The NRC is supporting the development of source term guidance for non-LWRs which may affect this aspect of fuel qualification (SAND, 2020) (INL, 2020). [7]

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[6] See Regulatory Guide 1.203, "Transient and Accident Analysis Methodologies," for more information on the PIRT process (NRC, 2005).

[7] The guidance developed on source term does not alter the fuel qualification framework. Both the guidance on source term and the fuel qualification framework accommodate a graded approach to source term where simplified, conservative models can be used to reduce the data requirements.

2.3 Accelerated Fuel Qualification

AFQ involves, in part, the use of advanced modeling and simulation to inform constituent and system selection and to enable integral fuel performance analyses (Terrani, et al., 2020). The AFQ process, shown in Figure 2-1, supports the identification of important parameters and phenomena for targeted characterization through separate-effects tests.

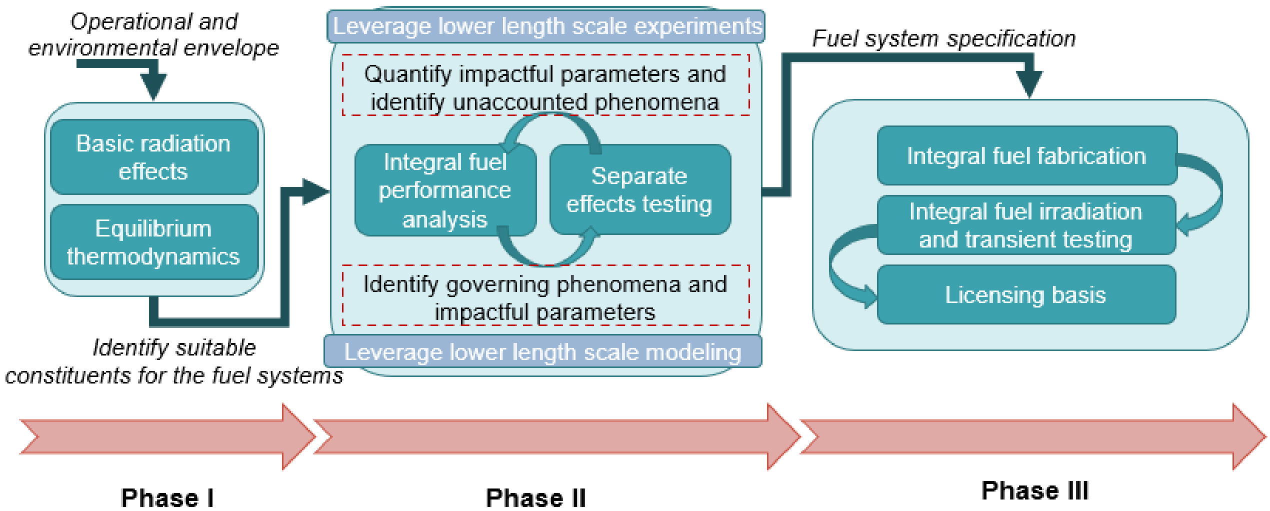

Figure 2-1 AFQ Process Workflow (Terrani, et al., 2020)

Advanced separate-effects testing techniques, such as fission accelerated steady-state testing (FAST) (Beausoleil II, Povirk, & Curnutt, 2020) and MiniFuel (Petrie, Burns, Raftery, Nelson, & Terrani, 2019), can reduce the time needed to achieve a given burnup and provide basic data on material behavior and property evolution under irradiation conditions. The information obtained through these analyses and separate-effects tests could help justify the adequacy of the evaluation model as part of Evaluation Model (EM) G1, "Evaluation Model Capabilities," which is discussed in Section 3.3.1. Additionally, validated physics-based models may support some extrapolation of evaluation models beyond the limits of available integral test data, as noted under EM G.2.2.4, "Restricted Domain," in Section 3.3.2.2.4. Ultimately, the AFQ process relies on integral irradiation test data to validate engineering scale fuel performance codes and to confirm the performance and safety of the fuel system under prototypic conditions. Accordingly, the integral test data produced as part of the AFQ process appear to be consistent with the considerations in the experimental data assessment framework discussed in Section 3.4.

2.4 Lead Test Specimens

Much of the data necessary to qualify fuel for use come from irradiated test specimens. Lead test specimens have been successfully used in operating reactors to obtain data at the needed exposures and are discussed in NUREG-0800, Section 4.2, as well as in Section 3.4.2 of this report. Section 3.4.2 of this report further discusses the potential for use of lead test specimens beyond what has been traditionally used for LWRs that can be useful for advanced reactor technologies.

2.5 Assessment Framework

The top-down development of an assessment framework is not a novel approach in the regulatory process. Similar assessment frameworks have been developed in the code scaling, applicability, and uncertainty evaluation methodology (NRC, 1989), the evaluation model development and assessment process (NRC, 2005), and the “objectives hierarchy” discussed in NUREG/BR-0303, “Guidance for Performance-Based Regulation,” issued December 2002 (NRC, 2002). Another top-down assessment framework, developed for thermal margin evaluations for LWRs, was based on many years of safety reviews (NRC, 2019). Assessment frameworks have facilitated safety reviews and have been shown to increase transparency about information needs, to promote efficiency by focusing attention on areas of recognized importance, and to clarify the logic behind decisions.

3 FUEL QUALIFICATION ASSESSMENT FRAMEWORK

This section on the fuel qualification assessment framework (FQAF) systematically identifies fuel safety criteria. The comprehensive list of safety criteria, called a fuel assessment framework, is informed by existing regulatory requirements, regulatory guidance, and staff experience with safety reviews for nuclear fuel in both LWRs and non-LWRs. The fuel assessment framework is developed using a top-down approach that starts with the high-level goal (G) that the fuel be qualified for use and then decomposes this goal into subgoals. Meeting the subgoals indicates that the higher-level goal is met. Each subgoal can either be further decomposed into other subgoals, or if no further decomposition is deemed necessary, the subgoal may be considered a base goal and evidence must be provided to demonstrate that the base goal is satisfied. In this report, base goals are identified by the use of grey boxes.

Consistent with the purpose of fuel qualification (see Section 1.1) and with a regulatory focus on safety, this report uses the following definition for fuel qualification:

> Fuel is qualified for use if reasonable assurance exists that the fuel, fabricated in accordance with its specification, will perform as described in the safety analysis.

This statement is captured figuratively in Figure 3-1, which decomposes fuel qualification into two supporting goals. These goals are further decomposed into lower level supporting goals, until criteria are obtained which can be directly verified by evidence. The subsections that follow describe the process, criteria, and associated evidence necessary to demonstrate fuel qualification.

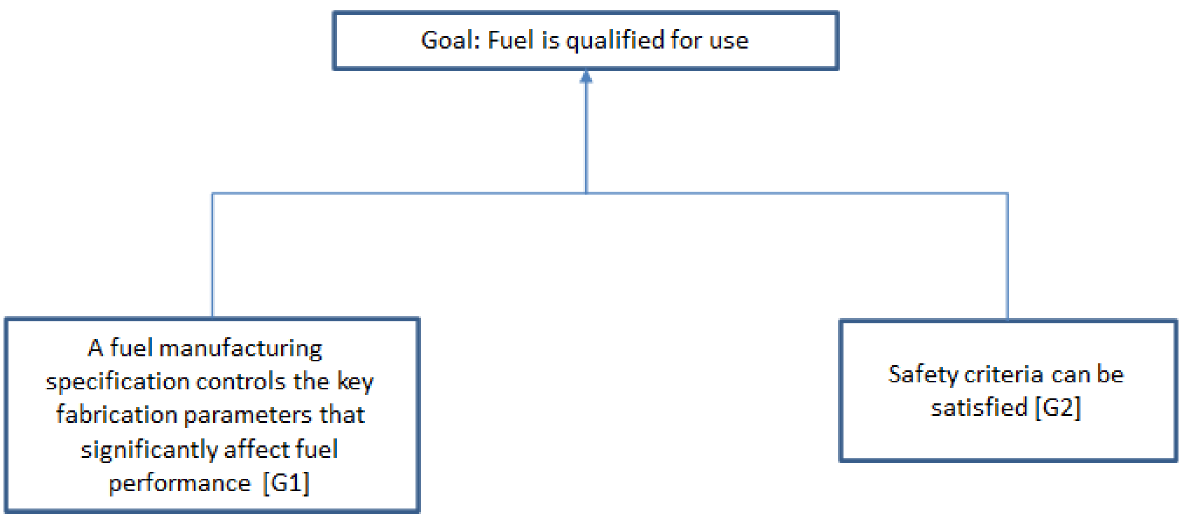

Figure 3-1 Decomposition of the Main Goal

3.1 G1—Fuel Manufacturing Specification

Fuel performance during normal operation and accident conditions can be highly sensitive to the fuel fabrication process. For example, failure criteria during reactivity-initiated accidents for LWRs with zirconium-based cladding depend upon the heat treatment of the cladding because of its impact on microstructure (NRC, 2020c). Similarly, key manufacturing parameters have been identified for TRISO fuel that must be controlled to ensure satisfactory performance (EPRI, 2020). Staff recognizes that manufacturing processes for a nuclear fuel product may evolve over the product life cycle; therefore, a complete manufacturing specification is not expected as part of the licensing documentation. However, the licensing documentation should include sufficient information to ensure the control of key parameters affecting fuel performance during the manufacturing process. The goal G1 is decomposed as shown in Figure 3-2 to identify the specific types of information to be included in licensing documentation.

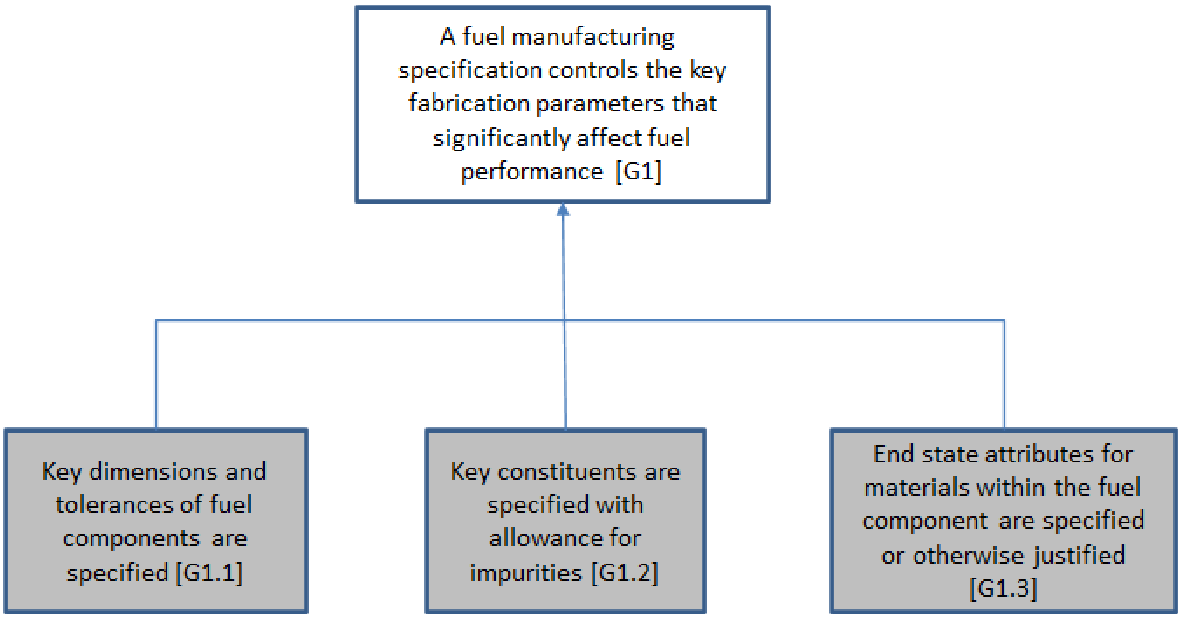

Figure 3-2 Decomposition of G1, "F uel Manufacturing Specification"

3.1.1 G1.1–Dimensions

Key dimensions and tolerances for fuel components that affect performance should be specified. Consistent with the scope of this report, as discussed in Section 1.3, these dimensions and tolerances should be specific to components that affect fuel life-limiting failure and degradation mechanisms that are due to irradiation during reactor operation (e.g., fuel pellet and cladding dimensions, key assembly dimensions). It is recognized that some of dimensions can be controlled by an approved change process (e.g., General Electric Standard Application for Reactor Fuel (GESTAR)).

3.1.2 G1.2—Constituents

Key constituents of fuel components (e.g., uranium dioxide (UO_2) fuel, uranium-plutonium-zirconium fuel alloys with specified concentrations (U-Pu-10Zr), cladding material) should be specified, along with allowances for impurities.

3.1.3 G1.3—End State Attributes

End state attributes for the materials within fuel components (e.g., microstructure) should be specified or otherwise justified. The information necessary to capture the desired end state of the material may take several forms. For example, specific manufacturing processes (e.g., cold-working, heat treatments, acid pickling, deposition techniques) that are essential to create the microstructure may be indicated in lieu of end state attributes. In some cases, it may be preferable to use performance-based end state attributes that can be supported through periodic testing and reporting (NRC, 2016). Additionally, it may be possible to demonstrate insensitivity to manufacturing processes so that end state attributes need not be specified in licensing documentation. Licensing documentation should provide sufficient justification for cases where a specific material is insensitive to manufacturing processes.

3.2 G2—Safety Criteria

An evaluation of the safety case involves an assessment against safety criteria, which are generally associated with protection against the release of radioactive material but also address the fundamental safety functions of heat removal and reactivity control. In general, many safety criteria for nuclear fuel depend on the events to which the fuel is subjected. Specifically, nuclear fuel is expected to retain its integrity under conditions of normal operation, including the effects of AOOs, but some degree of fuel failure can be accommodated for low-frequency design-basis accident conditions (i.e., those not expected to occur during the life of the plant). The goal G2 is decomposed as shown in Figure 3-3 to address the varying types of safety criteria for the range of events for which nuclear fuel should be qualified.

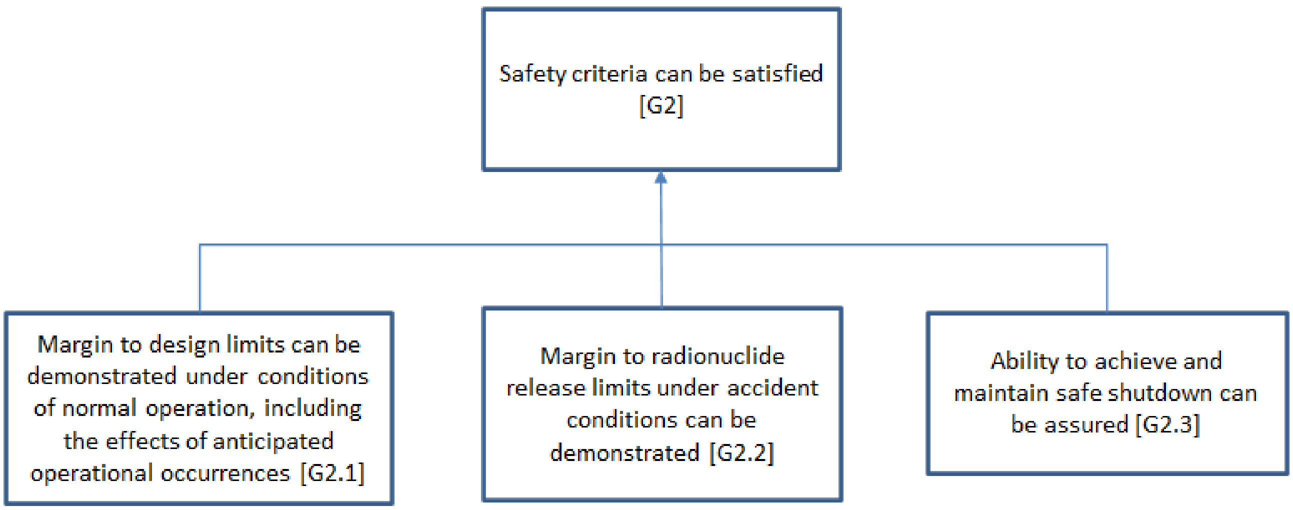

Figure 3-3 Decomposition of G2, "Safety Criteria "

3.2.1 G2.1—Design Limits during Normal Operation and Anticipated Operational Occurrences

Fuel integrity is expected to remain intact under conditions of normal operation, including the effects of AOOs. Alternatively, some designs may use SARRDLs, which allow a small degree of radionuclide release from the fuel (NRC, 2018b). Multiple degradation mechanisms and failure modes may exist; limits need to be established to protect against all of them. At the highest level, the assessment of a fuel against design limits for normal operation and AOOs requires knowledge of the conditions that the fuel is exposed to (i.e., the performance envelope) and a method to assess the fuel performance under those conditions (i.e., an evaluation model). These supporting goals, shown in Figure 3-4, are discussed below.

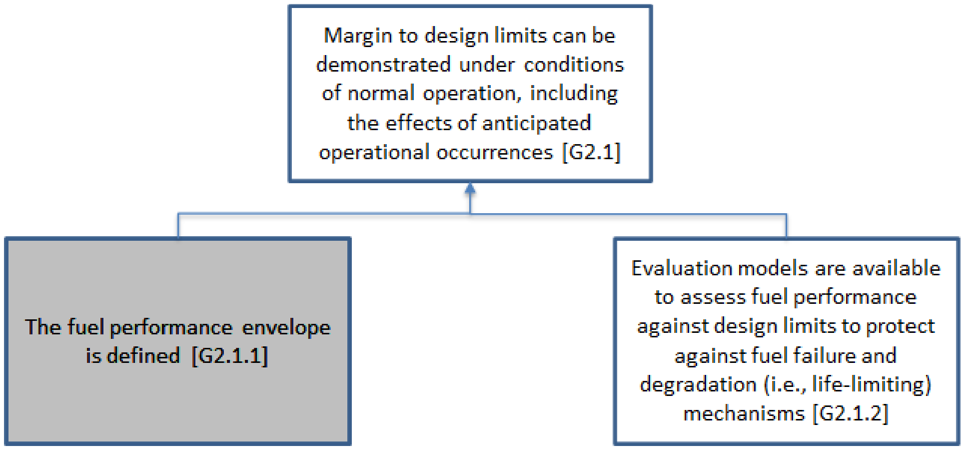

Figure 3-4 Decomposition of G2.1, "Design Limits During Normal Operation and AOOs"

3.2.1.1 G2.1.1—Definition of Fuel Performance Envelope

The fuel performance envelope specifies the environmental conditions and radiation exposure under which the fuel is required to perform. This performance envelope informs the safety analysis and technical specifications for the design (i.e., limiting conditions for operation). It is noted that irradiation-induced growth and fission product swelling of fuel components are often life-limiting phenomena for the fuel design. The envelope may be specified by fuel designers and may constrain the design of the reactor and associated systems. Alternatively, a reactor design may be proposed that imposes constraints on fuel performance. In support of G2.1, the goal G2.1.1 can be met by specifying the conditions (e.g., temperatures, pressures, power), exposure, and transient conditions that the fuel is expected to encounter during normal operation, including AOOs. Additionally, G2.1.1 supports G2.2, which addresses the fuel contribution to the source term during design-basis accidents, as discussed in Section 3.2.2.1. Accordingly, this goal can be fully satisfied by specifying the conditions the fuel is expected to encounter during normal operation, AOOs, and design-basis accidents.

3.2.1.2 G2.1.2—Evaluation Model

This goal—that evaluation models are available to assess fuel performance against design limits to protect against fuel failure and degradation mechanisms—requires the specification of means of evaluating fuel for performance, failure, and degradation. The assessment of evaluation models supports several goals and is further decomposed. Therefore, Section 3.3 provides a separate assessment framework for evaluation models. G2.1.2 is satisfied by meeting the supporting goals in that framework for fuel performance during normal operation and AOOs.

3.2.2 G2.2—Radionuclide Release Limits

Radiological consequences under postulated accident conditions are an essential consideration in nuclear power plant licensing. Under postulated accident conditions, some fuel failure is possible, which contributes to the accident source term. As radionuclide inventory originates from the nuclear fuel, fuel qualification should include characterizing the behavior of the fuel under accident conditions, so that its contribution to the accident source term can be determined in a suitably conservative manner. Accordingly, the goal G2.2—the ability to demonstrate margin to radionuclide release limits under accident conditions, in relation to fuel qualification—is supported by three goals related to the fuel contribution to the accident source term. These goals, shown in Figure 3-5 (along with G2.1.1, which also supports G2.2), are discussed further below.

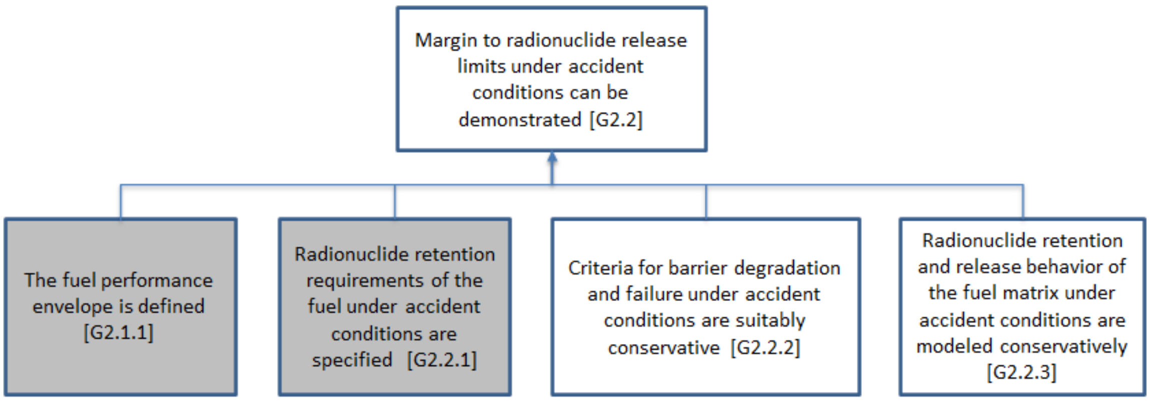

Figure 3-5 Decomposition of G2.2, "Radionuclide Release Limits"

3.2.2.1 G2.1.1—Definition of Fuel Performance Envelope

Section 3.2.1.1 already discussed G2.1.1. In support of G2.2, this goal can be satisfied by specifying the design-basis accident conditions to which the fuel is subjected. Design-basis accident conditions depend on reactor design; however, as discussed in Section 3.2.1.1, conditions to which the fuel is subjected during design-basis accidents may be specified independent of the reactor design, leading to constraints on the design of the reactor and associated systems. The types of design-basis accident conditions that should be considered include transient overpower events (e.g., reactivity-initiated accidents), transient undercooling events (e.g., loss-of-coolant accidents), and externally applied loads (e.g., fuel handling, transportation, seismic activity, and major piping failures).

3.2.2.2 G2.2.1—Radionuclide Retention Requirements

The role of nuclear fuel in the safety case can vary between reactor designs and fuel types. For example, traditional LWR fuel that uses UO_2 pellets with zircalloy cladding is not credited to retain cladding integrity under large-break loss-of-coolant accidents[8], while advanced reactor designs may credit retention of radionuclides within the fuel under accident conditions. Additionally, plant site characteristics such as proximity to population and weather patterns may further influence radionuclide retention requirements (even for the same reactor and fuel

[8] NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," (NRC, 1995) states that, "Assuming that the coolant loss cannot be accommodated by the reactor coolant makeup systems or the emergency core cooling systems, fuel cladding failure would occur with the release of radioactivity located in the gap between the fuel pellet and the fuel cladding."

design). To satisfy G2.2.1, the degree of radionuclide retention within the fuel system should be specified.

3.2.2.3 G2.2.2—Criteria for Barrier Degradation

Radionuclide barrier (e.g. fuel cladding) failure and degradation mechanisms under accident conditions (e.g., pellet-clad mechanical interaction (PCMI) and high enthalpy failure, temperature-induced reactions and phase transformations) must be understood when the design credits retention of barrier integrity (e.g., during reactivity-initiated accidents in LWRs, or considering the potential for fission product attack of the silicon carbide layer in TRISO fuel at high temperatures). As such, the goal G2.2.2 is decomposed into two supporting goals, shown in Figure 3-6.

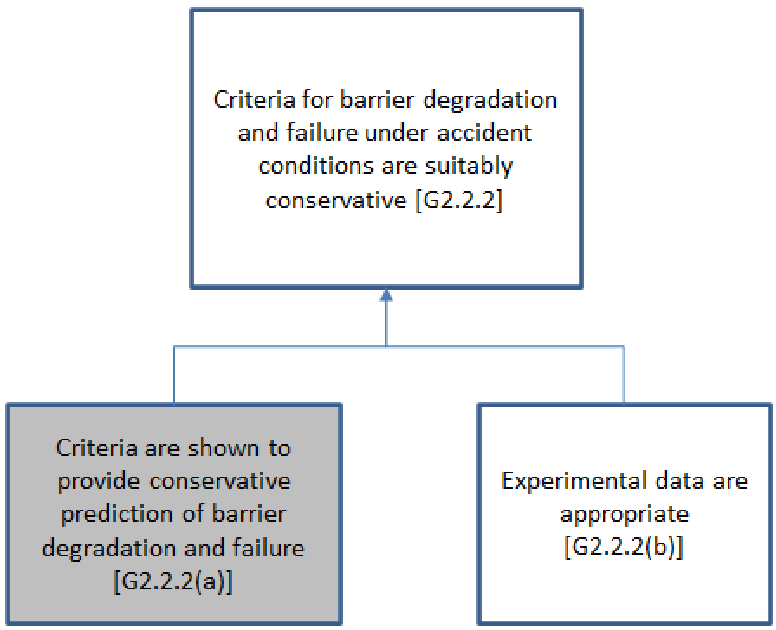

Figure 3-6 Decomposition of G2.2.2, "Criteria for Barrier Degradation"

3.2.2.3.1 G2.2.2(a)—Conservative Criteria

Criteria used to determine barrier degradation should be suitably conservative. These criteria are expected to be established based on transient testing and irradiated fuel samples, as discussed under G2.2.2(b). Ideally, to establish a statistical confidence level (e.g., 95/95), criteria would be established through a regression analysis using experimental data, then validated by assessment against a separate and independent set of data (see Section 3.4.1 (Experimental Data (ED) G1) for a discussion on data independence). However, this ideal scenario may not be realized due to challenges associated with obtaining irradiated fuel samples and conducting transient testing for design-basis accident conditions. The amount of experimental data supporting the criteria should be proportional to the degree of understanding of key degradation and performance phenomena (NRC, 2020c). If the data collected are not sufficient to support statistical modeling, a conservative or bounding approach may be required.

3.2.2.3.2 G2.2.2(b)—Experimental Data

This goal is satisfied through an evaluation against the experimental data assessment framework in Section 3.4.

3.2.2.4 G2.2.3—Conservative Modeling of Radionuclide Retention and Release

Consistent with the requirements specified as part of G2.2.1 and discussed in Section 3.2.2.2, radionuclide retention and release behavior of the fuel under accident conditions should be modeled conservatively. This goal is related to the barrier degradation criteria specified in G2.2.2 and discussed in Section 3.2.2.3, but it differs in its focus on radionuclide retention within the fuel matrix (e.g., UO_2 pellet or uranium alloy with 10 percent zirconium (U-10Zr) fuel ingot) or fuel particle (e.g., fuel compact for a TRISO-based fuel). This goal is decomposed into two supporting goals, as shown in Figure 3-7.

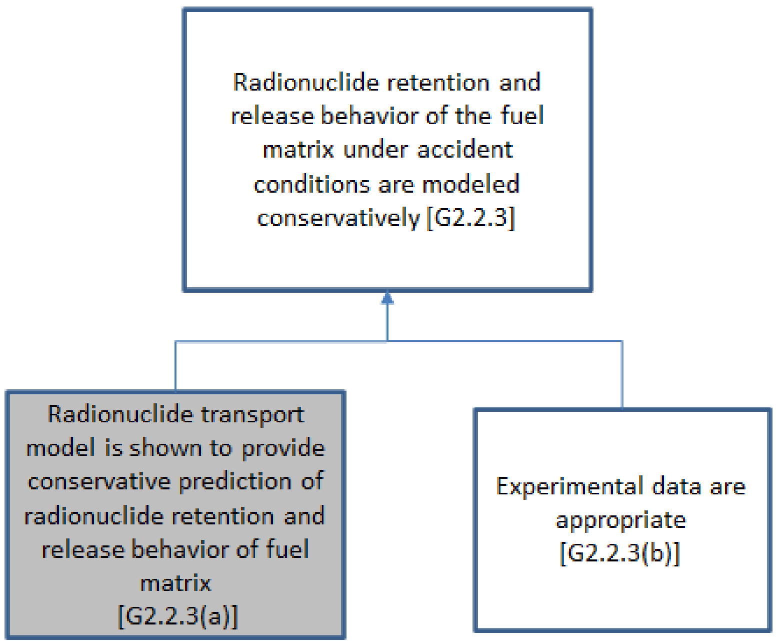

Figure 3-7 Decomposition of G2.2.3, "Conservative Modeling of Radionuclide Retention and Release"

3.2.2.4.1 G2.2.3(a)—Conservative Transport Model

The model of radionuclide transport within the fuel matrix should be conservative. As in the case of barrier degradation criteria, discussed in Section 3.2.2.3.1, challenges associated with obtaining and testing irradiated fuel samples may make it difficult to obtain sufficient data to characterize the transport model in a statistical manner; therefore, conservative or bounding estimates may be required. Additionally, previous source term models for LWRs have generally included some degree of expert judgment. A clarifying example of how to develop a suitably conservative radionuclide transport model is available in regulatory guidance on accident source terms (NRC, 2000).

3.2.2.4.2 G2.2.3(b)—Experimental Data

This goal is satisfied through an evaluation against the experimental data assessment framework in Section 3.4.

3.2.3 G2.3—Safe Shutdown

Safe shutdown of a nuclear plant refers to a state in which the reactor is subcritical, decay heat is being removed, and radionuclide inventory is contained. The international atomic energy agency (IAEA) refers to this as a *safe state* (IAEA, 2018). The ability to achieve safe shutdown in any scenario needs to be assured. Therefore, criteria need to be established to ensure that a coolable geometry is maintained in all scenarios and that fuel system damage is never so severe as to prevent control element (e.g., control rod) insertion when required. These supporting goals, captured in Figure 3-8, are discussed below.

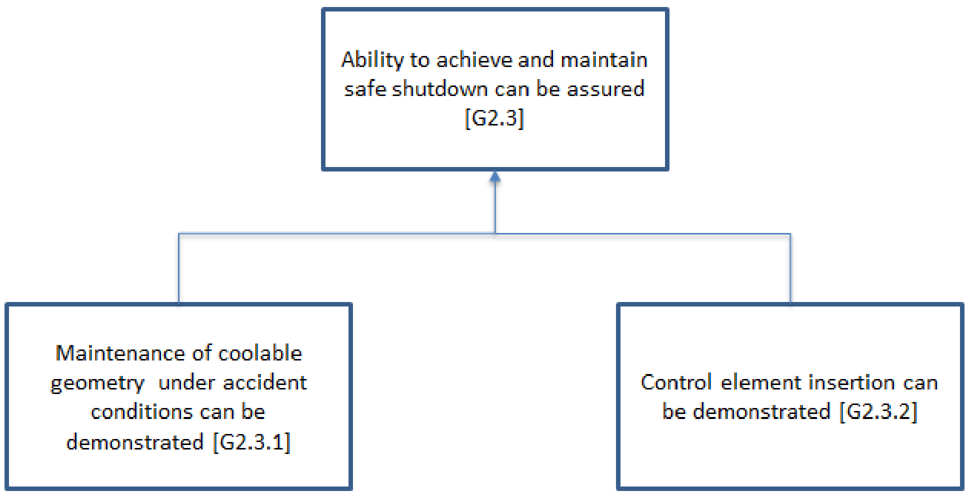

Figure 3-8 Decomposition of G2.3, "Safe Shutdown"

3.2.3.1 G2.3.1—Maintaining Coolable Geometry

The maintenance of a coolable geometry is identified as a supporting goal in achieving and maintaining safe shutdown. It is further decomposed into the supporting goals shown in Figure 3-9, which are discussed below.

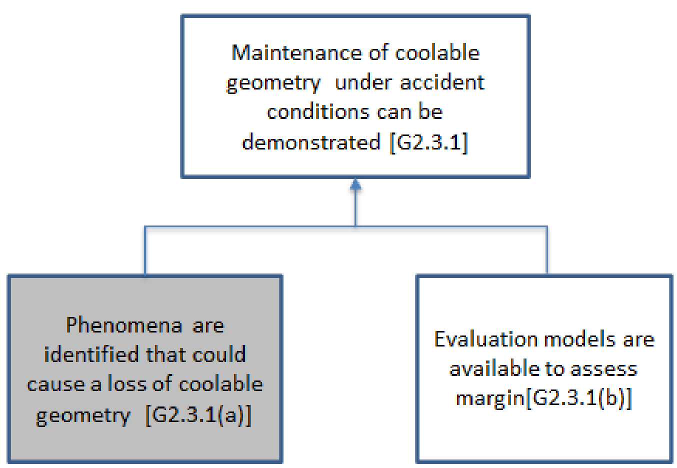

Figure 3-9 Decomposition of G2.3.1, "Maintaining Coolable Geometry"

3.2.3.1.1 G2.3.1(a)—Identification of Phenomena

Phenomena that could cause the loss of coolable geometry should be specified. Existing NRC regulations and guidance applicable to design basis accidents specify some acceptance criteria for these events that are intended to prevent such phenomena from significantly altering core geometry under postulated accident conditions. Examples of phenomena that could cause the loss of coolable geometry include: (1) fuel melt, (2) fuel swelling and fuel pellet and cladding fragmentation and dispersal during transient overpower events, and (3) loss of cladding ductility or long-term cladding phase stability during loss-of-coolant accidents.

3.2.3.1.2 G2.3.1(b)—Evaluation Models

Several evaluation models may be needed to demonstrate that coolable geometry is maintained. These models typically involve the use of conservative criteria and the evidence needed to meet this goal depends on the associated phenomena. For example, a conservatively chosen criterion such as the onset of fuel melting should not require a detailed evaluation model supported by integral testing, but an empirically based criterion such as energy deposition for fuel dispersal or peak cladding temperature for cladding embrittlement requires the demonstration of an appropriate margin against experimental data. Historical examples of acceptable empirical criteria include those developed for transient overpower (NRC, 2020c) and loss-of-coolant accidents (Hache & Chung, 2000). In addition to these empirical models for demonstrating a coolable geometry, analytical models have been used to demonstrate that coolable geometry is maintained for internal and external events (Framatome, 2018).

The evaluations performed to demonstrate coolable geometry vary in terms of complexity, form simple conservative criteria to detailed dynamic response models. The most general case that applies to all these situations is the generic evaluation model assessment discussed in Section 3.3. Accordingly, this goal is satisfied through a comparative assessment against the evaluation model assessment framework in Section 3.3. The application of the evaluation model assessment framework should follow a graded approach in accordance with the level of understanding of the physical phenomena and conservatism in the criteria.

3.2.3.2 G2.3.2—Control Element Insertion

Control element insertion is identified as a supporting goal in achieving and maintaining safe shutdown. It is further decomposed into the supporting goals shown in Figure 3-10, which are discussed below.

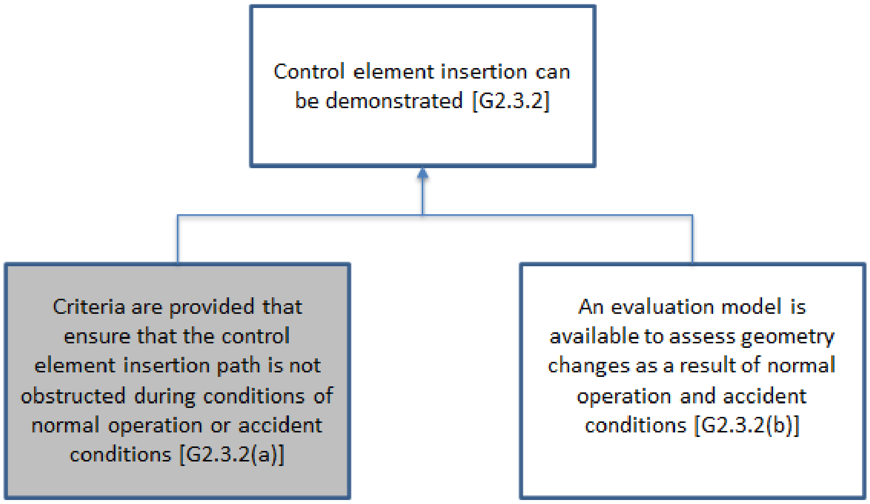

Figure 3-10 Decomposition of G2.3.2, "Control Element Insertion"

3.2.3.2.1 G2.3.2(a)—Identification of Criteria

Criteria should be specified to ensure that the control element insertion path is not obstructed during normal operation or accident conditions. These criteria should consider loads from both internal and external (e.g., seismic) events. An example of such a criterion for traditional LWRs is the stress limit imposed on the control rod guide tubes to inhibit distortion of the insertion path.

3.2.3.2.2 G2.3.2(b)—Evaluation Model

The evaluation performed to demonstrate that control element insertion can be assured has typically involved a stress analysis to ensure that the control element insertion path is not deformed as a result of internal and external events. This is typically done using a separate evaluation model. Accordingly, this goal is satisfied through a comparative assessment against the evaluation model assessment framework in Section 3.3.

3.3 Assessment Framework for Evaluation Models

The term "evaluation model" here is used in the generic sense. Typically, an evaluation model is an analytical tool, a computer code, or a combination of such tools. However, the use of a sophisticated tool such as a computer code may not be necessary to evaluate fuel performance. For example, a simple mathematical expression or set of data can serve as an evaluation model, if sufficient evidence exists to support its use.

The evaluation model assessment framework developed here is designed to be generically applicable. In particular, it supports G2.1.2, which addresses the evaluation of design limits under conditions of normal operation and AOOs, G2.3.1(b), which addresses maintaining coolable geometry, and G2.3.2(b), which addresses control element insertion. The evaluation model assessment framework presented here overlaps conceptually with the goals previously established for criteria for barrier degradation (Section 3.2.2.3) and radionuclide retention and release (Section 3.2.2.4). The latter two goals, however, have historically involved empirical evaluation models based on destructive testing using irradiated nuclear fuel under accident conditions. Accordingly, goals for barrier degradation and radionuclide retention and release are provided separately from the evaluation model assessment framework of this section.

The top-level goal of an acceptable evaluation model is supported by the goals of (1) adequate modeling capabilities and (2) assessment against experimental data. These supporting goals are shown in Figure 3-11 and discussed below.

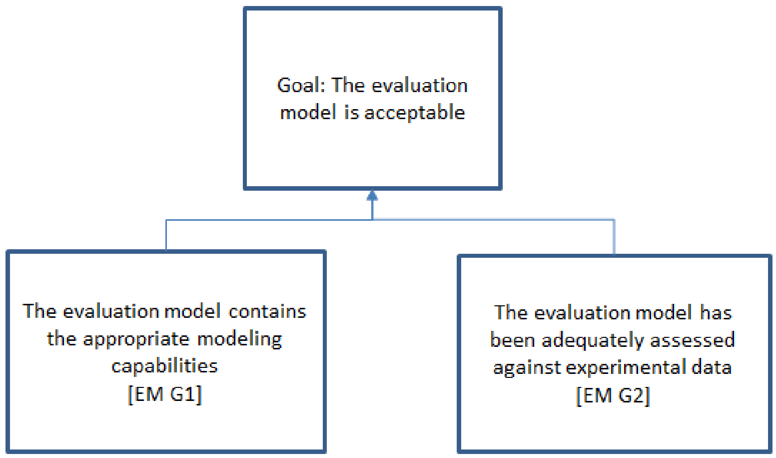

Figure 3-11 Decomposition of the Main Goal for Evaluation Model Assessment

3.3.1 EM G1—Evaluation Model Capabilities

The evaluation model capabilities goal is decomposed into three supporting goals as shown in Figure 3-12. This decomposition is informed by the predictive capability maturity model (PCMM) framework, which identifies “representation and geometric fidelity” and “physics and material model fidelity” as assessment elements (SAND, 2007). The evaluation model assessment framework also considers other elements of the PCMM framework. Specifically, EM G2 addresses “model validation” and “uncertainty quantification and sensitivity analysis”; see Section 3.3.2. The remaining elements of the PCMM framework, “code verification” and “solution verification,” are expected to be addressed as part of a quality assurance program for the design, analysis, and fabrication of a nuclear power facility. The goals supporting EM G1, shown in Figure 3-12, are discussed below.

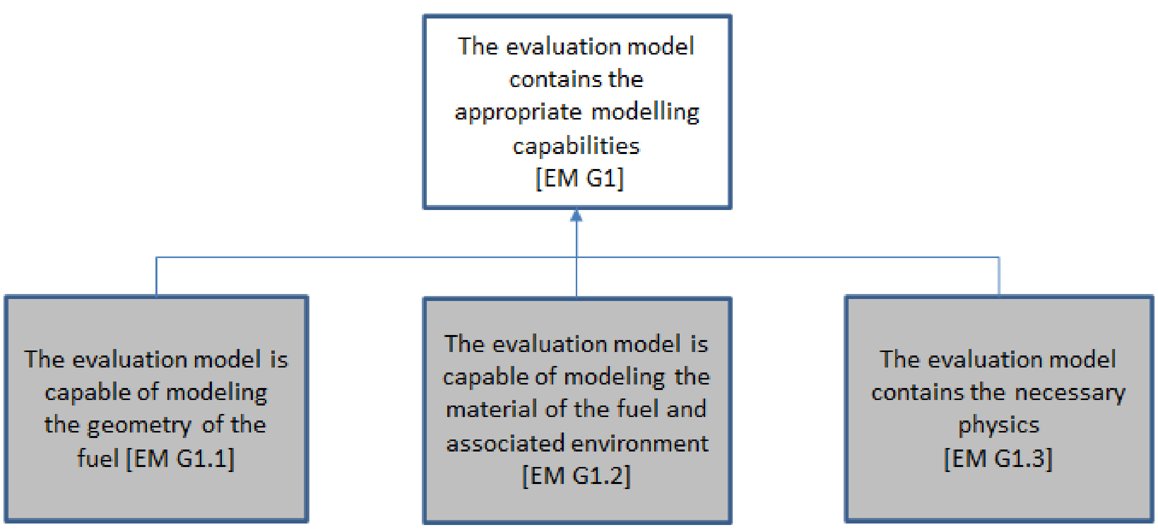

Figure 3-12 Decomposition of EM G1, “Evaluation Model Capabilities ”

3.3.1.1 EM G1.1—Geometry Modeling

The evaluation model should be capable of modeling the geometry of the fuel system. Table 3 of the PCMM provides guidance on the levels of maturity needed to assess the geometry, including consideration of peer review (SAND, 2007). It is recognized that some fuel designs may require simplifying assumptions to address difficulties in geometric modeling. For example, TRISO-based particulate fuel involves coupled phenomena occurring at different geometric scales (e.g., micro-scale within the TRISO particle, meso-scale within the fuel compact, and macro-scale within the reactor core). Geometric modeling for such particulate fuel could involve simplifications and assumptions that a less heterogeneous fuel design may not require. Additionally, the evaluation model should be able to capture geometric changes due to irradiation and exposure to the in-reactor environment (e.g., fuel swelling, cladding creep, oxide layer growth). Irrespective of imposed simplifications, the geometric modeling scheme should be appropriately justified, and the integrated evaluation model should be validated through the assessment process under EM G2.

3.3.1.2 EM G1.2—Material Modeling

The evaluation model should be capable of modeling material properties of the fuel system and its surrounding environment. This includes changes in material properties due to irradiation and exposure to the in-reactor environment (e.g., thermal conductivity degradation in nuclear fuel, changes to melting temperature, eutectic formation, changes to Young's modulus). Table 3 of the PCMM provides guidance on the levels of maturity needed to assess the material modeling, including considerations for model calibration against test data and peer review (SAND, 2007). The material modeling scheme should be justified, and the integrated evaluation model should be validated through the assessment process under EM G2.

3.3.1.3 EM G1.3—Physics Modeling

The evaluation model should be capable of modeling the physical processes that affect fuel performance. This goal requires knowledge of failure mechanisms, including changes due to irradiation and exposure to the in-reactor environment for the specified fuel, as well as fuel contribution to the SARRDL, if applicable. The evaluation model is expected to include sufficient physics modeling to address known degradation mechanisms (e.g., cladding oxidation and hydrogen pickup, fuel rod internal pressure, cladding strain, fuel assembly growth and wear, stress and fatigue for fuel components). Table 3 of the PCMM provides guidance on the levels of maturity needed to assess the physics modeling, including considerations for model calibration against test data and peer review (SAND, 2007). The physics models incorporated into the evaluation model should be justified, and the integrated evaluation model should be validated through the assessment process under EM G2. Means of justification include the use of an expert panel to develop a PIRT (PNNL, 2019) and internal review based on past experience, legacy data (ANL, 2018), or separate-effects testing (Beausoleil II, Povirk, & Curnutt, 2020) (Petrie, Burns, Raftery, Nelson, & Terrani, 2019).

3.3.2 EM G2—Evaluation Model Assessment

Evaluation model assessment is an essential process that provides confidence in the application of the evaluation model. To ensure that evaluation model predictions are suitably conservative, they should be assessed against appropriate experimental data. For statistically based modeling approaches, any bias or uncertainty in the evaluation model prediction should be adequately quantified, so that design and safety analyses can account for such bias or uncertainty. For conservative modeling approaches, the evaluation model should suitably bound the experimental data. The assessment process comprises two supporting goals, shown in Figure 3-13, which are discussed below.

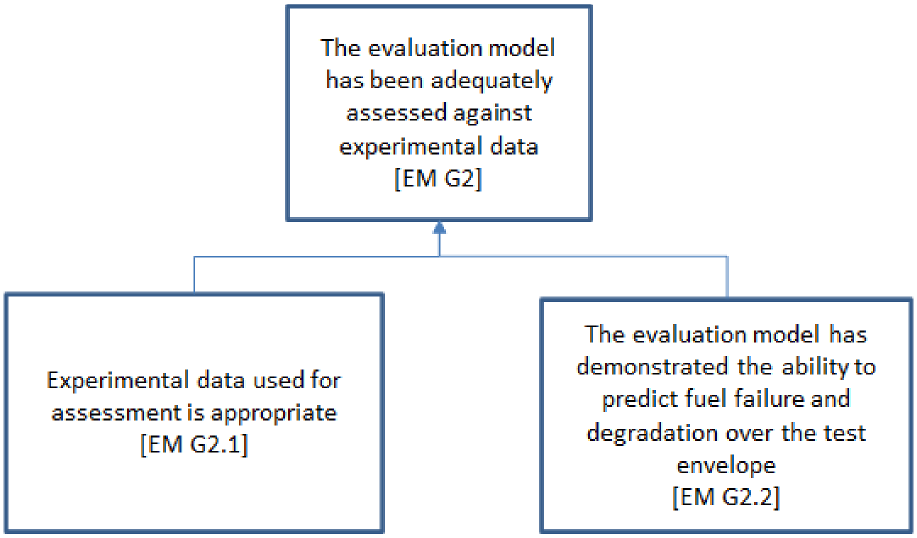

Figure 3-13 Decomposition of EM G2, "Evaluation Model Assessment"

3.3.2.1 EM G2.1—Experimental Data

This goal is satisfied through an evaluation against the experimental data assessment framework in Section 3.4.

3.3.2.2 EM G2.2—Demonstrated Prediction Ability over Test Envelope

EM G2.2 involves the comparison of evaluation model predictions against experimental data, which should establish uncertainties and biases and identify limitations in the applicability of the evaluation model. EM G2.2 is satisfied by meeting the four supporting goals shown in Figure 3-14, which are discussed below.

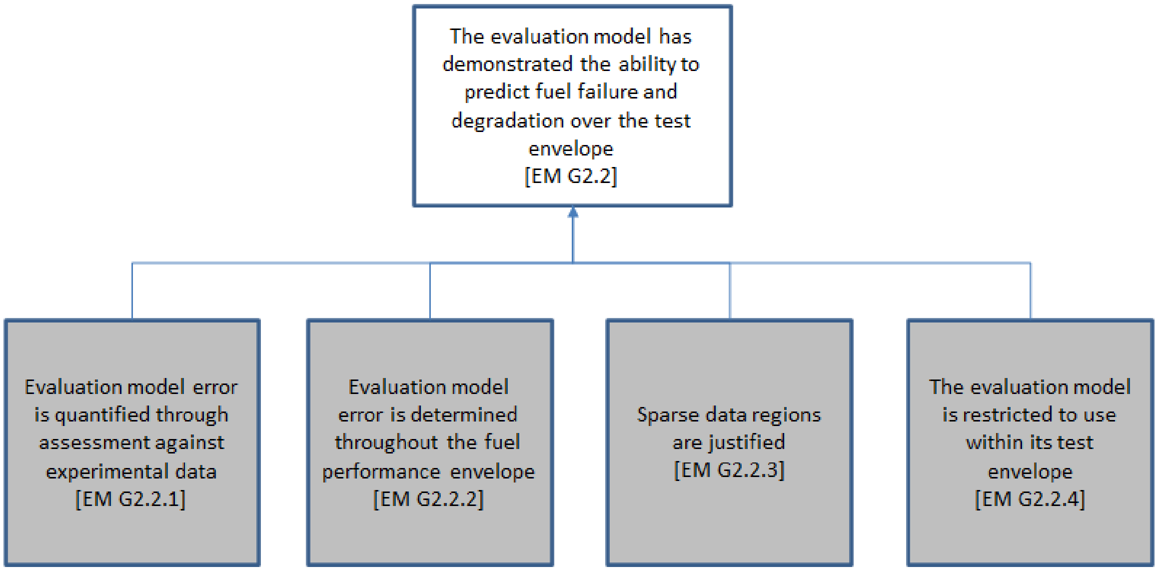

Figure 3-14 Decomposition of EM G2.2 , "Demonstrated Prediction Ability Over Test Envelope"

3.3.2.2.1 EM G2.2.1—Quantification of Error

Uncertainties and biases for figures of merit need to be sufficiently understood to establish confidence in the evaluation model. It is expected that, to determine uncertainties and biases, the predictions of the evaluation model for assessment cases will be compared against assessment data, and the differences between measured and predicted values will be quantified. If sufficient data exist, then statistical confidence levels can be placed on the uncertainties of the evaluation model predictions. However, a more bounding or conservative approach can also be taken (e.g., applying a bias or penalty to the model predictions, showing that the model is inherently conservative). EM G2.2.1 can be satisfied by a statement on the evaluation model biases and uncertainties, along with justification through a quantification of the ratio of predicted to measured values for assessment cases.

3.3.2.2.2 EM G2.2.2—Span of Validation Data

Assessment data should be distributed throughout the fuel performance envelope. The fuel performance envelope, discussed in Sections 3.2.1.1 and 3.2.2.1, is used to specify the test envelope; accordingly, assessment data should be available to assess the evaluation model over the entire span of the performance envelope. However, it is recognized that certain regions of the fuel performance envelope may not require data. For example, post-irradiation examination of an integral test specimen may not be necessary for low-burnup fuel. In such cases, it may suffice to provide justification that those regions do not require data (e.g., that limiting phenomena are known not to be present below a specified burnup). EM G2.2.2 can be satisfied by demonstrating that assessment data are available over the entire performance envelope, and by justifying any gaps in assessment data.

3.3.2.2.3 EM G2.2.3—Data Density

Assessment data should be appropriately distributed throughout the fuel performance envelope. As discussed in Section 3.3.2.2.2, it may be acceptable to have regions in the performance envelope where the evaluation model is not directly supported by assessment data from integral experiments. However, in regions that do require assessment data, a sufficient number of data points should be available for assessment of the evaluation model. It is reasonable to expect data density to be greater near conditions of normal operation, as fuel designers may require additional data to satisfy fuel reliability targets. However, any sparse data regions (i.e., regions of low data density) in the fuel performance envelope need adequate justification. EM G.2.2.3 can be satisfied by justifying the data density throughout the fuel performance window.

3.3.2.2.4 EM G2.2.4—Restricted Domain

Use of the evaluation model should be restricted to application domains for which the model has been assessed. Application of an evaluation model outside of the supporting test envelope (see Section 3.4.2) may be justified based on physical arguments (e.g., that the evaluation model provides a simplified or bounding treatment of physical phenomena). Justification for extrapolation of a model outside of the test envelope is strengthened by the use of physics-based models, such as those discussed in Section 2.3, which are informed by fundamental information about fuel evolution and behavior, as opposed to empirically derived models (Terrani, et al., 2020). EM G2.2.4 can be satisfied by specifying the application domain of the evaluation model as supported by the test envelope and by additional physical arguments as necessary.

3.4 Assessment Framework for Experimental Data

The assessment of experimental data is the largest area of review for fuel qualification. The assessment framework developed here supports all goals requiring evaluations against assessment data. Because a fuel qualification program involves several types of experiments (e.g., steady-state irradiation of integral test specimens, transient ramp testing, design-basis accident testing), and because of transient test facility limitations and challenges associated with irradiated fuel testing, it is recognized that the level of evidence expected to support a goal can vary depending on the type of data collected. The assessment framework presented in this section discusses this variance in the level of evidence as applicable. The main goal for assessment data is decomposed, as shown in Figure 3-15, into four supporting goals, which are discussed below.

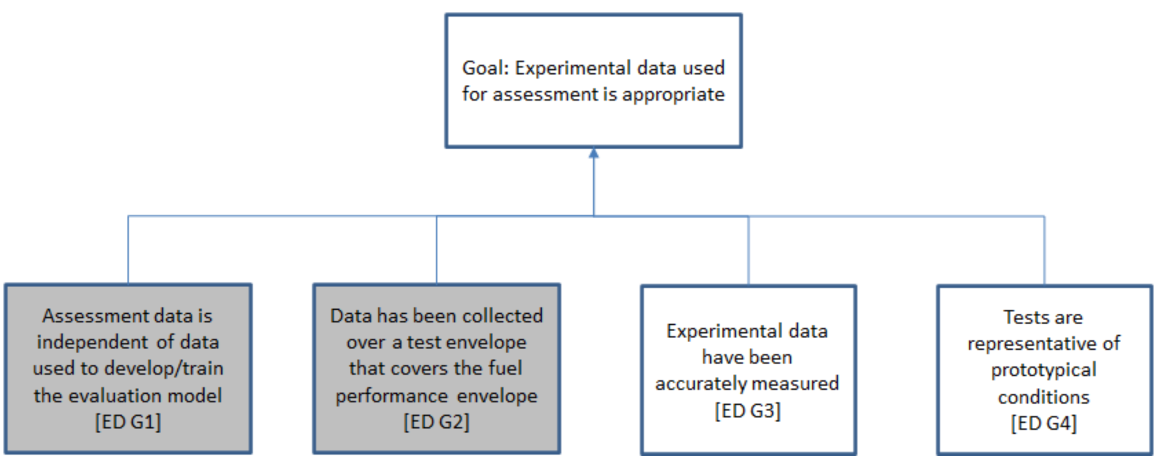

Figure 3-15 Decomposition of the Main Goal for Data Assessment

3.4.1 ED G1—Independence of Validation Data

Assessment data consist of experimentally measured values that are used to quantify the error in the evaluation model. Ideally, assessment data should be independent from any data used in the development (i.e., training) of the evaluation model. Although it may seem appropriate to use training data, training data cannot provide an accurate assessment because the evaluation model has already been “tuned” to those data. That is, quantifying the error of the training data would only show how well the model can predict the data used to generate it, not how well the model can predict data not used to generate it. Substantially more data points appear in the application domain (an infinite number) than were used to generate the model, and these are the points of most interest in future uses of the model; therefore, the focus should be on estimating the error over those points, not on the points used to generate the model. Thus, experimental data that were not used to train the model should be held in reserve and used to validate the model. Maintaining validation data separate from the model development process helps avoid a potential source of bias that could provide a distorted indication of the model’s accuracy for future uses.

In some instances, however, the validation data and the training data are one and the same. Methods exist in machine learning for determining whether the selection of the training data affects the resulting uncertainty; such methods include random subsampling and k-fold cross-validation. In each of these methods, the available data are randomly separated into subsets of training and validation data. The training data are used to develop the coefficients of the model, and the validation data are used to determine the overall uncertainty of the model. The process is then repeated with different randomly selected training and validation data sets. These methods can provide reasonable estimates of the impact of using the same data for training and validation.

The discussion of data independence has so far considered scenarios where a sufficient number of data points exist to train and validate a model using statistical approaches (i.e., model regression and the calculation of confidence intervals). It is recognized, however, that only limited data may be available because of the challenges associated with obtaining irradiated fuel samples. Experience from transient overpower testing has shown that it may be acceptable to develop criteria without separating the data into training and validation sets (NRC, 2020c). Similarly, fission gas release and swelling models have been proposed based on a limited amount of test data (Lee, Kim, & Jung, 2001). ED G1 can be satisfied by demonstrating that the data used in the evaluation model assessment are sufficiently independent.

3.4.2 ED G2—Test Envelope

Data should be collected over a test envelope that spans the performance envelope (see Section 3.2.1.1). The performance envelope should address normal operation, AOOs, and postulated accident conditions. The development of the test envelope should consider (1) steady-state integral testing of the fuel system in a prototypical environment, (2) high-power and undercooling tests to address AOO conditions and to assess design margins, (3) power ramp testing to assess fuel performance during anticipated power changes, and (4) design-basis accident tests to establish margin to fuel breach and contribution to the source term under accident conditions. Typical design-basis accident scenarios of interest include overpower events (e.g., reactivity-initiated accidents) and undercooling events (e.g., loss-of-coolant accidents).

Many of the data necessary for fuel qualification come from irradiated test specimens. However, test specimens at the desired conditions may sometimes be unavailable. In such situations, it may be possible to use lead test specimens to extend the burnup limits of a fuel type. In some cases, direct examination of lead test specimens may provide a basis to support extending applicability of an evaluation model to a new burnup range. In other cases, irradiated lead test specimens may become the subject of subsequent tests under transient or accident conditions to assess evaluation models applicable under such conditions.

Lead test specimen programs have traditionally allowed for the placement of a limited number of test specimens in nonlimiting regions of the reactor core to maximize the safety margin. However, an extended use of lead test specimens (e.g., relaxation of the number and/or location of the test specimens) may be allowable if justified by a safety analysis that includes margin to account for the uncertainty in the performance of fuel above its burnup limit. The use of fuel above its qualified limit should be supported by sufficient monitoring to detect potential failures. Methods are available, such as gas tagging (McCormick & Schenter, 1974) (Pollack, Lewis, & Kelly, 2013), that can be used to identify the precise source of potential fuel failures. Additionally, if lead test specimens are subjected to conditions beyond existing data ranges, a licensing review may be necessary to ensure the appropriate level of safety before the extended limits are applied to the fuel design. ED G2 can be satisfied by demonstrating that the test envelope addresses the necessary performance envelope for the fuel design.

3.4.3 ED G3—Data Measurement

An understanding of measurement accuracy is essential to establish confidence in the data used to develop and assess evaluation models. This goal is decomposed, as shown in Figure 3-16, into three supporting goals, which are discussed below.

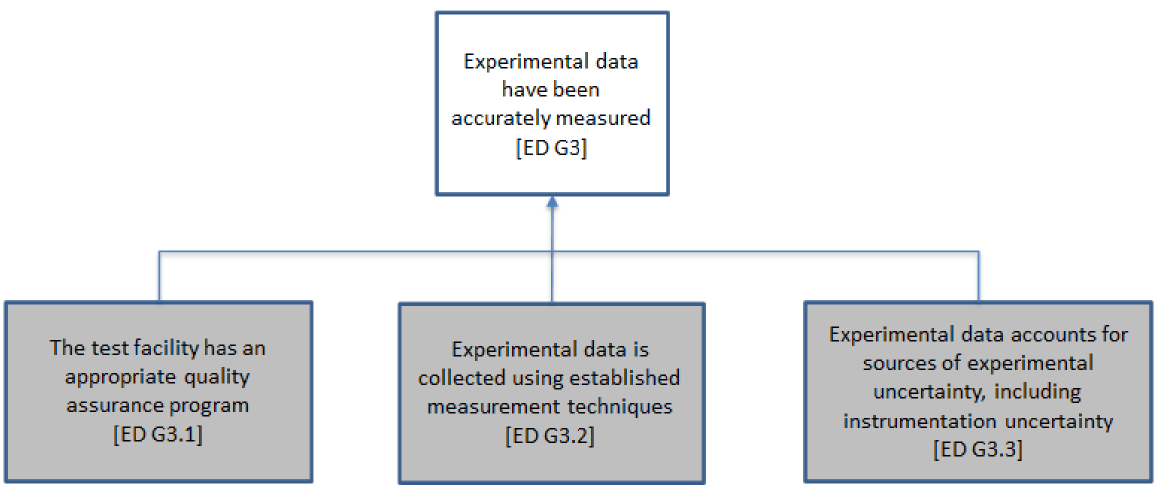

Figure 3-16 Decomposition of ED G3, "Data Measurement"

3.4.3.1 ED G3.1—Test Facility Quality Assurance

Experimental data should be collected under an appropriate quality assurance program. Standards such as the American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA)-1 are available for test facility quality assurance. Provisions may also be applied to existing data to make them compliant with quality assurance requirements (ANL, 2020). ED G3.1 can be satisfied by demonstrating that data were collected under an appropriate quality assurance program or by otherwise justifying the use of existing data.

3.4.3.2 ED G3.2—Measurement Techniques

Data should be collected using established or otherwise proven measurement techniques. The use of novel or first-of-a-kind measurement techniques should be adequately justified. ED G3.2 can be satisfied by specifying the measurement techniques and justifying the use of any novel or first-of-a-kind techniques.

3.4.3.3 ED G3.3—Experimental Uncertainties

An error analysis should be performed to assess sources of bias and uncertainty in each experiment. Measurement uncertainty should be quantified when possible, and its overall impact on assessment data should be discussed. ED G3.3 can be satisfied by providing an experimental error analysis.

3.4.4 ED G4—Test Conditions

The test conditions should be representative of prototypical conditions. Test specimens used in experiments should be representative of the proposed fuel design (i.e., the fuel design submitted for safety review). This goal is decomposed, as shown in Figure 3-17, into two supporting goals, which are discussed below.

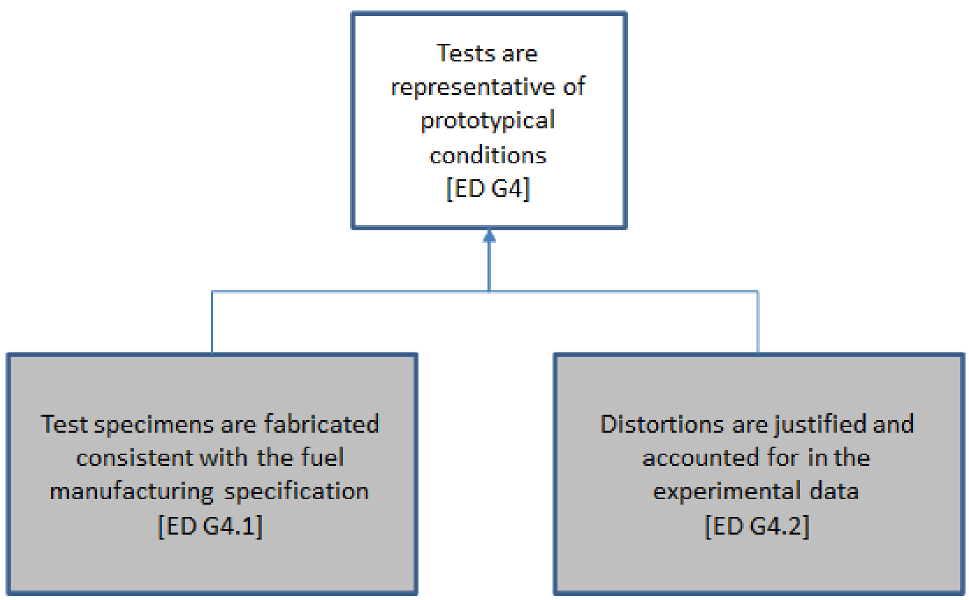

Figure 3-17 Decomposition of ED G4, "Test Specimens"

3.4.4.1 ED G4.1—Manufacturing of Test Specimens

Test specimens should be fabricated consistently with the manufacturing specification. (This goal is associated closely with G1, "Fuel Manufacturing Specification" (Section 3.1), which emphasized that fuel performance during normal operation and accident conditions can be highly sensitive to the fuel fabrication process.) It may be possible to provide justification for any acceptable differences in fabrication between the fuel and test specimens. Such justifications will be addressed case by case. ED G4.1 can be satisfied by demonstrating that test specimens are fabricated consistently with the fuel manufacturing specification.

3.4.4.2 ED G4.2—Evaluation of Test Distortions

Test distortions should be evaluated. Test distortions arise from differences between the test and the actual conditions under which the fuel is expected to perform (e.g., differences in test dimensions, initial and boundary conditions). An example of an expected test distortion is the geometry distortion typical of transient testing in a test reactor, as test reactors are generally too small to accommodate full-size fuel designs. ED G4.2 can be satisfied by an analysis of test distortions and justification for any identified distortions.

4 SUMMARY AND CONCLUSIONS

Section 3 of this report presents a systematic evaluation and justification of the requirements for qualifying nuclear fuel, and the table in Appendix A includes a concise list of the criteria identified to support a determination that nuclear fuel is qualified for use. These criteria provide a basis to support regulatory findings in the area of fuel qualification, as follows:

- The regulation in 10 CFR 50.43(e)(1)(i), requiring that the performance of each safety feature of the design has been demonstrated, is satisfied for the fuel by demonstrating that the safety criteria (G2 of the FQAF, discussed in Section 3.2) can be satisfied, which requires information to provide assurance that the fuel will perform as described in the safety analysis.

- The regulation in 10 CFR 50.43(e)(1)(iii) requires that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. This requirement can be satisfied by (1) specifying the fuel performance envelope, which covers a sufficient range of conditions (G2.1.1 of the FQAF), and (2) by demonstrating that assessed evaluation models and empirical criteria are capable of evaluating the fuel performance over the performance envelope (G2.1.2, G2.2.2, G2.2.3, G2.3.1, and G2.3.2(b) of the FQAF). Sections 3.2.1.1, 3.2.2.1, 3.2.2.3, 3.2.2.4, 3.2.3.1, and 3.2.3.2.2 discuss these topics further.

- GDC 2 and ARDC 2 require that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. G2.3 of the FQAF (discussed in Section 3.2.3) partially addresses this requirement through assurance of the ability to achieve and maintain safe shutdown.

- GDC 10 and ARDC 10 require that SAFDLS or SARRDLs not be exceeded under any conditions of normal operation, including the effects of AOOs. This requirement is satisfied, in part, by demonstrating margin to design limits under conditions of normal operation, including the effects of AOOs (G2.1 of the FQAF, discussed in Section 3.2.1).

- GDC 27 and ARDC 26 require, in part, the ability to achieve and maintain safe shutdown under postulated accident conditions. G2.3 of the FQAF (discussed in Section 3.2.3) partially addresses this requirement through assurance of the ability to achieve and maintain safe shutdown.

- GDC 35 and ARDC 35 require an emergency core cooling system that provides sufficient cooling under postulated accident conditions. They also require that fuel and clad damage that could interfere with continued effective core cooling is prevented. G2.3 of the FQAF (discussed in Section 3.2.3) partially addresses these requirements through assurance of the ability to achieve and maintain safe shutdown.

- The regulations in 10 CFR 50.34(a)(1)((ii)(D), 10 CFR 52.47(a)(2)(iv), and 10 CFR 52.79(a)(1)(vi) require an evaluation of a postulated fission product release. This requirement is partially addressed by demonstrating margin to radionuclide release limits under accident conditions (G2.2 of the FQAF, discussed in Section 3.2.2).

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APPENDIX A
LIST OF ALL GOALS

Table A-1 List of Goals in Fuel Qualification Assessment Framework

GOAL				Fuel is qualified for use
G1				Fuel is manufactured in accordance with a specification
	G1.1			Key dimensions and tolerances of fuel components are specified
	G1.2			Key constituents are specified with allowance for impurities
	G1.3			End state attributes for materials within fuel components are specified or otherwise justified
G2				Margin to safety limits can be demonstrated
	G2.1			Margin to design limits can be demonstrated under conditions of normal operation and AOOs
		G2.1.1		Fuel performance envelope is defined
		G2.1.2		Evaluation model is available (see EM Assessment Framework)
	G2.2			Margin to radionuclide release limits under accident conditions can be demonstrated
		G2.1.1		Fuel performance envelope is defined
		G2.2.1		Radionuclide retention requirements are specified
		G2.2.2		Criteria for barrier degradation and failure are suitably conservative
			(a)	Criteria are conservative
			(b)	Experimental data are appropriate (see ED Assessment Framework)
		G2.2.3		Radionuclide retention and release from fuel matrix are modeled conservatively
			(a)	Model is conservative
			(b)	Experimental data are appropriate (see ED Assessment Framework)
	G2.3			Ability to achieve and maintain safe shutdown is assured
		G2.3.1		Coolable geometry is ensured
			(a)	Criteria to ensure coolable geometry are specified
			(b)	Evaluation models are available (see EM Assessment Framework)
		G2.3.2		Control element insertion can be demonstrated
			(a)	Criteria are provided to ensure that control element insertion path is not obstructed
			(b)	Evaluation model is available (see EM Assessment Framework)

Table A-2 List of Goals in Evaluation Model Assessment Framework

GOAL	Evaluation model is acceptable for use		
EM G1	Evaluation model contains the appropriate modeling capabilities		
	EM G1.1	Evaluation model is capable of modeling the geometry of the fuel system	
	EM G1.2	Evaluation model is capable of modeling the material properties of the fuel system	
	EM G1.3	Evaluation model is capable of modeling the physics relevant to fuel performance	
EM G2	Evaluation model has been adequately assessed against experimental data		
	EM G2.1	Data used for assessment are appropriate (see ED Assessment Framework)	
	EM G2.2	Evaluation model is demonstrably able to predict fuel failure and degradation mechanisms over the test envelope	
		EM G2.2.1	Evaluation model error is quantified through assessment against experimental data
		EM G2.2.2	Evaluation model error is determined throughout the fuel performance envelope
		EM G2.2.3	Sparse data regions are justified
		EM G2.2.4	Evaluation model is restricted to use within its test envelope

Table A-3 List of Goals in Experimental Data Assessment Framework

GOAL	Experimental data used for assessment are appropriate	
ED G1	Assessment data are independent of data used to develop/train the evaluation model	
ED G2	Data has been collected over a test envelope that covers the fuel performance envelope	
ED G3	Experimental data have been accurately measured	
	ED G3.1	The test facility has an appropriate quality assurance program
	ED G3.2	Experimental data are collected using established measurement techniques
	ED G3.3	Experimental data account for sources of experimental uncertainty
ED G4	Test specimens are representative of the fuel design	
	ED G4.1	Test specimens are fabricated consistent with the fuel manufacturing specification
	ED G4.2	Distortions are justified and accounted for in the experimental data

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11. ABSTRACT (200 words or less)

Proposed advanced reactor designs use fuel designs and operating environments (e.g., neutron energy spectra, fuel temperatures, neighboring materials) that differ from the large experience base available for traditional light-water reactor fuel. The purpose of this report is to identify criteria that will be useful for advanced reactor designs through an assessment framework that would support regulatory findings associated with nuclear fuel qualification. The report begins by examining the regulatory basis and related guidance applicable to fuel qualification, noting that the role of nuclear fuel in the protection against the release of radioactivity for a nuclear facility depends heavily on the reactor design. The report considers the use of accelerated fuel qualification techniques and lead test specimen programs that may shorten the timeline for qualifying fuel for use in a nuclear reactor at the desired parameters (e.g., burnup). The assessment framework particularly emphasizes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of the experimental data used to develop and validate evaluation models and empirical safety criteria.

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