U.S. NRC—CNSC Memorandum of Cooperation

FINAL REPORT
concerning
Tristructural Isotropic (TRISO) Fuel Qualification

June 2023

Approved by

Mohamed K. Shams Digitally signed by Mohamed K. Shams Date: 2023.06.11 20:44:48 -04'00'

Dr. Mohammed K. Shams, Director
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
United States Nuclear Regulatory Commission

Approved by

Recoverable Signature

X C. Ducros

Signed by: Ducros, Caroline

Dr. Caroline Ducros, Director General
Directorate of Advanced Reactor Technologies
Canadian Nuclear Safety Commission

Recoverable Signature

X Melanie Rickard

Signed by: rickard, melanie

Melanie Rickard, Director General
Directorate of Assessment and Analysis
Canadian Nuclear Safety Commission

DISCLAIMER: The NRC and the CNSC have prepared this final report to inform stakeholders of the current project status for performing a generic assessment of TRISO fuel. The information contained in this document has not been subject to NRC and CNSC management and legal review, and its contents are subject to change and should not be interpreted as official agency positions.

Preface

On August 15, 2019, the Canadian Nuclear Safety Commission (CNSC) and the United States Nuclear Regulatory Commission (U.S. NRC), two of the world's leading nuclear regulators, signed a joint memorandum of cooperation (MOC) aimed at enhancing technical reviews of advanced reactor and small modular reactor technologies. This MOC is intended to supplement and strengthen the existing memorandum of understanding between the two parties, signed in August 2017.

CNSC–U.S. NRC cooperation provides opportunities for both agencies to share scientific information about technical matters that could support more efficient reviews of small modular reactors and advanced reactor technologies. Cooperative activities can be conducted with acknowledgment of differences in Canadian and U.S. regulatory frameworks and licensing processes, while leveraging fundamental scientific and engineering findings from each others reviews to the extent practicable.

Activities under the MOC are coordinated by a subcommittee of the U.S. NRC-CNSC Steering Committee, called the Advanced Reactor Technologies and Small Modular Reactors (ART-SMR) Sub Committee, which approves and prioritizes work plans to accomplish specific cooperative activities under the MOC.

Cooperative activities between both organizations are established and governed under Terms of Reference and are intended to do the following:

- Contribute to better use of regulator resources by leveraging the technical knowledge and resources between the U.S. NRC and the CNSC.
- Enhance the depth and breadth of understanding of the respective staff of the CNSC and the U.S. NRC about the counterpart nation's regulatory review activities and requirements.
- Enhance the joint opportunities for learning and understanding the advanced reactor and small modular reactor technologies being reviewed.

The decision of the CNSC and the U.S. NRC to cooperate in activities that concern specific reactors, and their associated vendors depends on the design and is based on the following four factors that a vendor must address in a proposed work plan that both regulators accept:

(1) To what extent is the vendor engaging in meaningful prelicensing activity with each regulator?

(2) How are the vendor's engagement activities in each country similar, such that the outcome of cooperation will be useful? For example, the objectives of the CNSC's vendor design review process are different than those of the U.S. NRC's certification and prelicensing engagement processes, yet opportunities exist for leveraging information between the two regulators.

(3) What are the timelines for engaging with each regulator?

(4) How is the vendor sharing information about its design with both regulators to enable cooperation to occur?

Table of Contents

1. Introduction

This section documents the history underpinning the decision by the U.S. Nuclear Regulatory Commission (U.S. NRC) and the Canadian Nuclear Safety Commission (CNSC) to establish this cooperative activity.

1.1. Relevant Vendor Engagement with the U.S. NRC

In 2020, the U.S. NRC reviewed the Electric Power Research Institute (EPRI) topical report EPRI-AR-1(NP)-A, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance," dated November 20, 2020 [1]. Additionally, the U.S. NRC has reviewed topical reports from two vendors of advanced reactors that are proposing the use of TRISO fuel in their reactor designs: Kairos Power (KP) and X-energy. In 2021, the U.S. NRC reviewed topical report KP-TR-010-NP, Revision 2, "KP-FHR Fuel Performance Methodology," issued November 2020 [2], which describes a plan to validate the use of the BISON fuel performance code to the KP-FHR (Fluoride-salt-cooled High-temperature Reactor). Additionaly, KP submitted an application for a nonpower construction permit in September 2021 [3]. In 2020, the U.S. Department of Energy (U.S. DOE) selected X-energy to deliver a commercial TRISO fuel fabrication facility and a four-module version of its Xe-100 high-temperature gas-cooled reactor (HTGR) by 2027 as part of the DOE Advanced Reactor Demonstration Program (ARDP). Preapplication interaction with X-energy has increased over the recent years, including in response to the submittal of "Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology," Revision 2, dated August 16, 2021 [4]. The U.S. NRC anticipates that licensing activities with X-energy (e.g., topical reports) will continue to increase.

1.2. Relevant Vendor Engagement with the CNSC

Through the CNSC Vendor Design Review (VDR) process, the CNSC has worked with two vendors of advanced reactors that are proposing to use TRISO fuel in their reactor designs: Ultra-Safe Nuclear Corporation (USNC) and X-energy. A VDR is a feedback mechanism that enables the CNSC staff to provide feedback early in the design process based on a vendor's reactor technology. The CNSC completes the assessment at the vendor's request. The VDR process is described in detail in CNSC REGDOC-3.5.4, "Pre-Licensing Review of a Vendor's Reactor Design," issued 2018 [5].

The word "prelicensing" signifies that a design review is undertaken before the submission to the CNSC of a license application by an applicant seeking to build and operate a new nuclear power plant. An application by a vendor for a design review is not an application for a license to prepare a site or to construct or operate a nuclear power facility, and it is not an indication of intent to proceed with a project. The objective of a design review is to verify, at a high level, the acceptability of a nuclear power plant design with respect to Canadian nuclear regulatory requirements and expectations, as well as Canadian codes and standards. These reviews also identify fundamental barriers to licensing a new design in Canada and ensure that a resolution path exists for any design issues identified in the review. The CNSC prelicensing activity is similar to pre-application activities such as white-paper reviews performed by the US NRC [6].

Global First Power (GFP) is seeking CNSC approval for a license to prepare the site for a micro modular reactor at the Chalk River Laboratories site in Renfrew County, Ontario, approximately 200 kilometers northwest of Ottawa. A CNSC license is required under subsection 24(2) of the Nuclear Safety and Control Act in order for the project to proceed. In March and April 2021, GFP submitted management system documentation in support of its application for a license to prepare a site for a micro modular reactor on Atomic Energy of Canada Limited property at the Chalk River Laboratories site. On May 6, 2021, the CNSC determined that this documentation and GFP's plan for additional submissions were sufficient to begin the technical review as part of the licensing application process.

The proposed project includes a nuclear plant that contains an HTGR to provide approximately 15 megawatts (thermal) of process heat to an adjacent plant through molten salt, which will generate electrical power, heat, or both over an operating lifespan of 20 years. The reactor technology vendor for this project, USNC, has engaged with the CNSC to conduct a VDR of the reactor technology proposed for deployment at the Chalk River Laboratories site.

1.3. Considerations in Agreeing on the Scope and Objectives of Cooperative Activities between the CNSC and the U.S. NRC

Advanced reactor vendors with designs that propose the use of TRISO fuel have engaged with the CNSC and the U.S. NRC. Vendors are submitting plans to the U.S. NRC for fuel qualification through topical reports and to CNSC through the VDR process. Guidance in the area of fuel qualification is available in the Nuclear Energy Agency (NEA) document NEA/CNRA/R(2020)1, "Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors," issued December 2020 (NEA-RPFQ) [7], and NUREG-2246, "Fuel Qualification for Advanced Reactors," issued March 2022 [8]. NUREG-2246 was subject to a public comment period during which industry stakeholders, through the Nuclear Energy Institute, emphasized the importance of having advanced-reactor-specific examples (including TRISO fuel) to address fuel qualification review criteria. The development of NUREG-2246 also addressed a requirement established by the U.S. Nuclear Energy Innovation and Moderation Act [9] to develop guidance to support advanced nuclear fuel licensing.

2. Statement of Scope and Objectives for the Cooperative Activities

The CNSC and U.S. NRC staff will work together to establish a common regulatory position on TRISO fuel qualification based on existing knowledge and to identify any potential analytical or testing gaps that would need to be addressed to enable TRISO use in advanced reactor licensing applications. This project aims to do the following:

- Provide the evidentiary basis to support regulatory findings for items associated with fuel qualification that are generically applicable to TRISO fuel based on currently available information.
- Identify areas of TRISO fuel qualification that are design dependent.
- Highlight areas where additional information, testing, or both is still needed to support regulatory approval.

3. Regulatory Basis

3.1. Regulatory Basis for Fuel Qualification at the U.S. NRC

The relevant regulatory requirements associated with fuel qualification for TRISO fuel are as follows:

- The regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.43(e)(1)(i) requires demonstration of the performance of each safety feature[1] of the design through either analysis, appropriate test programs, experience, or a combination thereof.

- 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.

- 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.

- The regulations in 10 CFR 50.34(a)(3)(i), 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a) require that the principal design criteria (PDC) be provided for a construction permit, design certification, combined license, standard design approval, or manufacturing license. Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," establishes the minimum requirements for PDC for water-cooled nuclear power plants. Appendix A to 10 CFR Part 50 also established that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units.

  Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," issued April 2018 [10], provides guidance on how the GDC in Appendix A to 10 CFR Part 50 may be adapted for non-light-water-reactor (non-LWR) designs and contains technology inclusive advanced reactor design criteria (ARDC) as well as specific sodium fast reactor and modular high temperature gas reactor (MHTGR) design criteria (DC). While the GDC and ARDC are not requirements for non-LWR designs, the GDC, ARDC, and MHTGR-DC identified below address safety functions generally associated with nuclear fuel that are not otherwise captured by NRC regulations (e.g., reactivity control, heat removal, confinement of radionuclides). Accordingly, the NRC staff expects that information be provided that addresses the design aspects described in the following as part of fuel qualification:

  – GDC 2 and ARDC 2, "Design bases for protection against natural phenomena," require that structures, systems, and components (SSCs) important to safety be

---

[1] Nuclear fuel contributes to the reactivity balance and is a source of heat generation and fission products. Therefore, nuclear fuel is generally recognized as impacting the safety functions of reactivity control, heat removal, and confinement of radioactive material.

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, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50 implements GDC 2 as it pertains to seismic events and defines specific earthquake criteria for nuclear power plants. This appendix establishes definitions for safe-shutdown earthquake and operating-basis earthquake, and safety requirements for relevant SSCs. These SSCs are necessary to ensure 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. Though the implementation approach to address ARDC 2 may vary from GDC 2, the intention of ARDC 2, like GDC 2, is to ensure SSCs important to safety maintain capability to perform safety functions. The safety functions generally associated with nuclear fuel include control of reactivity, cooling of radioactive material, and confinement of radioactive material.[2]

- GDC 10 and ARDC 10, "Reactor design," require that specified acceptable fuel design limits 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). Reactor designs that use TRISO fuel are generally expected to use SARRDLs.

- MHTGR-DC 16, "Containment Design," requires provision of a functional containment, consisting of multiple barriers internal or external to the reactor and its cooling system, or both, to ensure that the function containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- 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 provide assurance that the capability to cool the core is maintained.

NUREG-2246 also identifies GDC 35 and ARDC 35, "Emergency core cooling," as applicable to fuel qualification. However, reactor designs that use TRISO fuel are generally not expected to contain an emergency core cooling system. Additionally, NUREG-2246 does not list MHTGR-DC 16, which addresses the use of fuel as part of a functional containment. Advanced reactor designs that use TRISO fuel are generally expected to credit the fuel as part of a functional containment.

---

[2] "Fundamental safety functions" are discussed in the International Atomic Energy Agency (IAEA) Safety Glossary [29]. The following pertain to fundamental safety functions: reactivity control in GDC 27 and ARDC 26; heat removal in GDC and ARDC 10, GDC 27, ARDC 26, and GDC and ARDC 35; functional containment in MHTGR-DC 16; and radionuclide retention in GDC and ARDC 10 and associated 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).

3.2. Regulatory Basis for Fuel Qualification at the CSNC

REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants," issued 2014, provides the following criteria for water-cooled reactor facilities [11]:

- Fuel assemblies and the associated components shall be designed to withstand the anticipated irradiation and environmental conditions in the reactor core, and all processes of deterioration that can occur in operational states. The fuel shall remain suitable for continued use after AOOs. At the design stage, consideration shall be given to long-term storage of irradiated fuel assemblies after discharge from the reactor.

- Fuel design limits shall be established to include, as a minimum, limits on fuel power or temperature, limits on fuel burnup, and limits on the leakage of fission products in the reactor cooling system. The design limits shall reflect the importance of preserving the fuel matrix and cladding, as these are first and second barriers to fission product release, respectively.

- The design shall account for all known degradation mechanisms, with allowance being made for uncertainties in data, calculations, and fuel fabrication.

- In design-basis accidents, the fuel assembly and its component parts shall remain in position with no distortion that would prevent effective postaccident core cooling or interfere with the actions of reactivity control devices or mechanisms. The design shall specify the acceptance criteria necessary to meet these requirements during design-basis accidents. The requirements for reactor and fuel assembly design shall apply in the event of changes in fuel management strategy, or in operating conditions, over the lifetime of the plant.

- Fuel design and design limits shall reflect a verified and auditable knowledge base. The fuel shall be qualified for operation, either through experience with the same type of fuel in other reactors, or through a program of experimental testing and analysis, to ensure that fuel assembly requirements are met.

- Acceptance criteria should be established for fuel damage, fuel failure, and fuel coolability. These criteria should be derived from experiments that identify the limitations of the material properties of the fuel and fuel assembly, and related analyses. The fuel design criteria and other design considerations are discussed below.

The CNSC is developing REGDOC 2.4.5, "Nuclear Fuel Safety" [12], to clarify the requirements and provide guidance for the design, operation, monitoring, and safety assessments of fuel for operating reactor facilities. This document has implicit concentration on operating Canada Deuterium Uranium (CANDU) reactors, but it remains as technology neutral as practicable. It applies, primarily, to fuel programs and designs that are already licensed, and to modified or new fuel designs envisioned for operating plants at the time of the publication of this document.

The high-level concepts and technology-neutral information also apply to proposed new reactor facilities, including technologies other than water-cooled reactors. If a design other than a CANDU reactor is being considered for licensing in Canada, the associated fuel design, qualification, and oversight will be subject to the safety objectives, high-level safety concepts, and safety management requirements associated with REGDOC 2.4.5, where applicable.

4. TRISO Fuel Assessment

This section contains a generic assessment of TRISO fuel based on the framework provided in NEA-RPFQ and NUREG-2246. It incorporates positions documented in the NRC safety evaluation for EPRI-AR-1(NP)-A but also highlights areas that the topical report did not address such as fuel performance evaluation model requirements, potential testing needs to address potential accident conditions, and accident source term considerations.

4.1. G1—Fuel Manufacturing Specification

Most key manufacturing parameters in the following sections were obtained from EPRI-AR-1(NP)-A. These key manufacturing parameters correspond to the fuel performance testing from the Advanced Gas Reactor Fuel Development and Qualification (AGR)-1 and AGR-2 programs. The AGR-1 and AGR-2 fuel specifications appear in INL/MIS-05-00238, Revision 1, "AGR-1 Fuel Product Specification and Characterization Guidance," issued April 2006 [13], and SPC-923, Revision 3, "AGR-2 Fuel Specification," issued January 2009 [14].

4.1.1. G1.1—Dimensions

4.1.1.1 TRISO Particle

The parameters for the TRISO fuel particle in table 1 are expected to be applicable to all technologies that use TRISO fuel.

Table 1 TRISO Particle Parameters

Particle Dimension	95% Confidence Interval Extrema	95%/98% Tolerance Limit Extrema	Basis
Buffer thickness (μm)	96.5–105.2	75.2–124.7	c
IPyC thickness (μm)	38.6–41.1	32.4–47.6	c
SiC thickness (μm)	34.8–36.2	30.6–41.2	c
OPyC thickness (μm)	39.1–44.3	33.6–51.6	c
OPyC aspect ratio	1.057[a]	1.102[b]	c
SiC aspect ratio	1.040[a]	1.068[b]	c

[a] Upper bound of 95% confidence interval
[b] 99% coverage tolerance interval
[c] EPRI-AR-1(NP)-A, Table 5-5

In addition to the TRISO coating parameters provided in Table 1, kernel size is also an important factor. Section 5.3.6 of EPRI-AR-1(NP)-A states that, "because the kernel is thermomechanically decoupled from the coating layers, there is not a *unique* set of kernel specifications that are critical to successful TRISO fuel as long as the scaling discussed in Section 4.2 [of EPRI-AR-1(NP)-A] is considered." The scaling presented in section 4.2.6 of EPRI-AP-1(NP)-A, and discussed in section 3.5 of the NRC staff safety evaluation, used a simplified stress calculation, given by equation 1 to obtain a simplified tensile stress metric (STSM) given by equation 2.

$$\sigma \propto \frac{BV_k}{V_b}\frac{r_{SiC}}{t_{SiC}} \tag{1}$$

$$STSM = \frac{BV_k}{V_b}\frac{r_{SiC}}{t_{SiC}} \tag{2}$$

where

σ	=	tensile stress
B	=	maximum burnup in fissions per initial metal atom (FIMA)
V_k	=	volume of the fuel kernel
V_b	=	volume of the buffer
r_{SiC}	=	inner radius of silicon carbide (SiC) layer
t_{SiC}	=	thickness of SiC layer

Based on the information provided in section 5.3.6 of EPRI-AR-1(NP)-A the AGR-1 and AGR-2 test data cover an STSM up to 0.810 at the 99th percentile. Accordingly, kernel sizes and burnup limits that maintain the STSM below a value of 0.810 at the 99th percentile are acceptable.[3]

Fuel designs that satisfy the bounds discussed in this section would satisfy Condition 1[4] of the NRC safety evaluation for EPRI-AR-1(NP)-A regarding key dimensions for TRISO particles. However, as discussed in the NRC safety evaluation for EPRI-AR-1(NP)-A (specifically, Condition 1), particle dimensions that are outside the bounds discussed in this section may also be acceptable but would require additional justification.[5]

4.1.1.2 Fuel Compact/Pebble

Dimensions for the fuel compact/pebble are expected to vary among the different reactor vendors. Accordingly, this area of review will be addressed on a design-dependent basis.

---

3 Sections 2.4 and 3.4.2 of NUREG-2246 discuss the use of lead test specimens to obtain data at the needed exposures.

4 Condition 1 from the NRC staff's safety evaluation for EPRI-AR-1(NP)-A states that, "An applicant or licensee referencing [EPRI-AR-1(NP)-A] must evaluate any discrepancies between their fuel particles and the TRISO particles used in the AGR program—specifically, reviewing the ranges specified in Table 5-6 for stress values to capture any effects from different kernel sizes to ensure the data in the [topical report] remain applicable."

5 Section 5.3.6 of EPRI-AR-1(NP)-A states that, "Ultimately it will be up to an applicant to provide justification for applying AGR-1 and AGR-2 particle performance results to a TRISO fuel population that deviates from AGR-1 and AGR-2 fuel properties."

4.1.2. G1.2—Constituents

4.1.2.1 TRISO Particle

Constituents of the fuel kernel should be within the limits provided in Table 2.

Table 2 TRISO Fuel Kernel Constituents

Parameter	Limit	Basis
Enrichment	< 20% U-235	EPRI-AR-1(NP)-A, section 5.3.1[6,7] [1]
UC_x molar fraction	~29 to ~32% with a mean of ~30%[a,b,c]	EPRI-AR-1(NP)-A, table 5-2 [1]
Individual impurities	≤ 100 ppm-weight (wt)% for each impurity of Li, Na, Ca, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, Al, and Cl	AGR-1 Fuel Specification [13]
Process impurities	≤ 1500 ppm-wt% for each impurity of P, S	AGR-1 Fuel Specification [13]

[a] This represents a mean value for the population. Consistent with AGR-2, critical limits are not specified (see table 4 from the EPRI letter dated February 26, 2020 [15]).
[b] Calculated molar fraction with the remaining material being uranium dioxide (UO_2).
[c] Assumes that no other compounds besides UC_x and UO_2 are present.

Controlling the amount of UC_x is important because of the following:

- Too little UC_x in the fuel kernel can increase the production of carbon monoxide (CO) during irradiation, which increases the potential for fuel failure due to (1) pressure vessel failure of the SiC, (2) kernel migration failure, and (3) nonretentive SiC failure (see section 4.3.1.3 for a description of degradation mechanisms and failure modes).

- Too much UC_x can result in insufficient oxygen in the kernel to oxidize rare-earth fission products, leading to fission product attack of the SiC layer. Fuel containing greater than 75 percent uranium carbide (UC_2) was observed to experience considerable fission product attack of the SiC coating by the rare-earth fission products lanthanum, cerium, praseodymium, and neodymium [16].

Section 5.3.6 of EPRI-AR-1(NP)-A clarifies that, "The AGR program chose to target about 30% uranium carbide in their kernel fabrication to provide ample carbide phase to meet a burnup of ~20% FIMA while experiencing negligible CO gas formation." Limitation 2 of the NRC safety evaluation for EPRI-AR-1(NP)-A states that additional justification will be needed if the UO_2/UC_2 ratios differ meaningfully from those used in the AGR program. Kernel composition outside these limits would require additional justification.

---

6 Section 5.3.1.1 of EPRI-AR-1(NP)-A clarifies that AGR-1 UCO kernels had a nominal enrichment of 19.7 percent uranium (U)-235.

7 Section 5.3.1.2 of EPRI-AR-1(NP)-A clarifies that AGR-2 kernels had a nominal enrichment of 14 percent U-235.

Research on the fuel kernel composition concluded the following [16]:

> Irradiation experiments conducted to date suggest that a conversion level of 35% [UC_2] is optimum with ±20% latitude. Experiments are currently being conducted under accelerated irradiation conditions to verify this tentative speculation.

EPRI-AR-1(NP)-A referenced a thermochemical study that did not use validated analyses to suggest that UC_x content as low as 5.5 percent may be sufficient to achieve burnups up to 16 percent FIMA in UCO TRISO [17]. Based on the discussion above, there is speculation that UC_x concentrations beyond those provided in Table 2 of this report would be acceptable. The range provided in Table 2 of this report is based upon the information available from AGR-1 and AGR-2. Experimental evidence or additional justification would be needed to support values for UC_x molar fractions beyond those provided in Table 2.

The buffer, inner pyrolytic carbon (IPyC), and outer pyrolytic carbon (OPyC) layers of the TRISO particle are made of pyrolytic carbon with the end-state attributes provided in section 4.1.3.1 of this report. Impurity limits are not specified for pyrolytic carbon or SiC.

4.1.2.2 Fuel Compact/Pebble

Constituents for the fuel compact/pebble may vary among the different reactor vendors. Accordingly, this area of review will be addressed on a design-dependent basis. SPC-1352, Revision 8, "AGR-5/6/7 Fuel Specification," dated March 9, 2017, gives an example AGR fuel compact material and component specification [18].

4.1.3. G1.3—End-State Attributes

4.1.3.1 TRISO Particle

End state attributes should be within the limits provided in Table 3.

Table 3 TRISO End State Attributes

Particle Property	95% Confidence Interval Extrema	95%/98% Tolerance Limit Extrema	Basis
Buffer density (g/cm^3)	1.04–1.11[a,b]	N/A	EPRI-AR-1(NP)-A, table 5-5 [1]
IPyC density (g/cm^3)	1.84–1.92	1.808–1.958	EPRI-AR-1(NP)-A, table 5-5 [1]
SiC density (g/cm^3)	3.196–3.209	3.191–3.217	EPRI-AR-1(NP)-A, table 5-5 [1]
OPyC density (g/cm^3)	1.878–1.924	1.850–1.949	EPRI-AR-1(NP)-A, table 5-5 [1]
IPyC anisotropy (BAF_{True})	1.024[b]	1.036[c]	EPRI-AR-1(NP)-A, table 5-5 [1]

Particle Property	95% Confidence Interval Extrema	95%/98% Tolerance Limit Extrema	Basis
OPyC anisotropy (BAF_{True})	1.018[b]	1.030[c]	EPRI-AR-1(NP)-A, table 5-5 [1]
SiC microstructure	N/A (see discussion below)	N/A (see discussion below)	AGR-1 Fuel Specification [13]

[a] Range of measured means only. No confidence intervals available.
[b] Upper bound of 95% confidence interval.
[c] Upper bound of 99% confidence interval.

4.1.3.1.1 Silicon Carbide Microstructure

AGR-1 and AGR-2 fuel specification did not include quantitative limits on SiC microstructure but used a visual standard to represent an upper bound on acceptable grain size, with no specified lower bound.[8] The NRC staff's safety evaluation for EPR-AR-1(NP) states that, "the expectation is that an applicant referencing [EPRI-AR-1(NP)-A] would institute a similar control [to the visual standard used for AGR-1 and AGR-2] on manufactured TRISO particles." The purpose of the visual standard is to establish a qualitative upper bound on the acceptable grain size to prevent the formation of large, columnar grains that could allow for enhanced fission product transport.

The AGR-1 program used three variations in the SiC coating manufacturing process, which produced a range of as-built grain sizes [1]. Studies on the as-built SiC microstructure using AGR-1 and AGR-2 data did not establish a significant correlation between SiC microstructure and TRISO particle performance [19]. Work is planned to characterize the AGR program as-built SiC microstructure to provide quantitative data to better understand the range of grain sizes that yielded AGR program-like fission product retention and potentially replace the qualitative visual standard on the upper grain size, if necessary, and inform the development of any future criteria on the upper range of grain size, if necessary.

4.1.3.1.2 Manufacturing Process

In a letter dated February 26, 2020, EPRI stated the following regarding EPRI-AR-1(NP):

> Because uninterrupted coating is the de facto standard in modern TRISO fabrication, it is considered a process requirement when applying the results of this topical report [EPRI-AR-1(NP)].

Accordingly, fuel designs that rely on AGR-1 and AGR-2 data must be manufactured using an uninterrupted coating process.

---

8 This is provided as figure 5-2 from EPR-AR-1(NP)-A [1] and figures 1a and 1b from the AGR-1 fuel specification [13].

4.1.3.2 Fuel Compact/Pebble

Section 5.3.4 of EPRI-AR-1(NP)-A describes the fuel compact process for AGR-1 and AGR-2 and clarifies that fuel particles were overcoated with resonated graphite. This overcoat serves to prevent particle-to-particle contact and help achieve the desired volumetric packing fraction of fuel particles within compacts/pebbles. UCO TRISO fuel compacts used in AGR-1 and AGR-2 irradiations had a packing fraction of 37 percent[9]. Based on the packing fraction used for the UCO TRISO compacts in AGR-1 and AGR-2, fuel designs that rely on AGR-1 and AGR-2 data should be fabricated with a fuel compact/pebble packing fraction below 40 percent. Licensees or applicants proposing packing fractions above 40 percent would need to justify that sufficient protection is provided to prevent particle-to-particle contact.

Additional end-state attributes for the fuel compact/pebble may vary among the different reactor vendors. Accordingly, complete specification of the end-state attributes for the fuel compact/pebble should be addressed on a design-dependent basis.

4.2. G2—Safety Criteria

4.2.1. G2.1—Design Limits during Normal Operation, Anticipated Operational Occurrences, and Design-Basis Events

4.2.1.1 G2.1.1—Definition of Fuel Performance Envelope

The fuel performance envelope specifies the operating conditions and irradiation exposure under which the fuel is required to perform. The fuel performance envelope can be determined by (1) using an existing fuel test envelope and designing the reactor operating conditions to stay within those limits, (2) estimating the reactor operating conditions and creating a test envelope that encompasses the expected operation, or (3) using additional test data to cover the operating envelop not covered by existing data. The AGR program test envelope was created to cover the expected operation of an HTGR as described in Section 6.6 and figure 6-29 of EPRI-AR-1(NP)-A, which compares the AGR-1 and AGR-2 time-average temperature to an HTGR design with an outlet temperature of 750 degrees Celsius (°C). Table 4 in Section 4.4.2 of this document gives other important test conditions that define the irradiation envelope. The AGR program TRISO particle data are also applicable to reactor designs other than HTGRs (e.g., molten salt cooled reactors), as the TRISO particles are assumed to be protected by the presence of the matrix overcoat.

A similar consideration was used to construct the AGR-1 and AGR-2 safety tests, which start at the upper end of the irradiation temperatures and expand the temperature range up to 1,800°C, bounding expected accident conditions. Additional data may be needed to expand the test envelope for designs that are expected to have higher steady-state irradiation temperatures or accident conditions not bounded by the AGR program safety tests.

4.2.1.2 G2.1.1—Evaluation Model

---

[9] The maximum packing fraction for random close-packed spheres is approximately 64 percent.

An evaluation model is used to assess fuel performance against the design limits to preclude unacceptable degradation or failures. Evaluation models allow for evaluating normal, AOO, and accident conditions that may not have been explicitly part of the fuel test program but have been approved to be capable of predicting those conditions (e.g., a range of heat-up or reactivity events). Additional information on establishing an appropriate evaluation model is described in section 4.3.1.

4.2.2. G2.2—Radionuclide Release Limits

The role fuel is assumed to play in retaining fission products can vary significantly among fuel types, other barriers credited to retain fission products, and siting considerations. Generally, TRISO fuel is assumed to be a major fission product barrier, while other SSCs take on less of a fission product retention role as in current LWR designs. For example, in most TRISO-fueled designs, the TRISO particles are part of a functional containment in which a barrier or set of barriers taken together effectively limits fission product transport. The fission product release from TRISO particles is a function of the manufacturing specification (e.g., allowed manufacturing defects, dispersed uranium) and in-service failures, which are related to the fuel performance envelope. The releases from the particle can be based on experimental data or the use of an approved fuel performance code if the underlying data cover the performance envelope or limited extrapolation can be justified. In both cases, uncertainty must be quantified, or conservative assumptions employed. The allowed particle radionuclide release limit is accordingly a function of any other barriers credited in the functional containment concept and meeting the site regulatory dose criteria. Typically, a mechanistic source term model is used, assuming a large enough source term, to predict releases to the environment. As such, the allowed fuel particle release is a function of the specific design and desired siting that allow the regulatory requirements to be met.

4.2.3. G2.3—Safe State

4.2.3.1 G2.3.1—Maintaining Coolable Geometry

4.2.3.1.1 TRISO Particle

TRISO fuel is generally expected to act as part of a functional containment. Accordingly, TRISO particles are expected to maintain their integrity under accident conditions.[10] Preventing SiC thermal decomposition, discussed in section 4.3.1.3 of this report, provides assurance that the integrity and coolability of the TRISO particle are maintained.

4.2.3.1.2 Fuel Compact/Pebble

The fuel compact/pebble functions, in part, to provide structural integrity and thermal conductivity for the fuel. Therefore, the fuel compact/pebble needs to maintain its structural integrity to ensure a coolable geometry. A carbonaceous material is a common host matrix material for TRISO particles with temperature capabilities well above the SiC thermal

---

[10] This contrasts with traditional LWR fuel, for which the cladding is not credited to retain fission products under some design-basis accidents (e.g., loss of coolant accident [28]).

decomposition temperature. Accordingly, maintaining the TRISO particle temperature below the SiC thermal decomposition limit should also ensure coolable geometry of the fuel compact/pebble. Since fuel compacts/pebbles may vary among the different reactor vendors, specifying criteria to ensure coolable geometry of the fuel compact/pebble is addressed on a design-dependent basis.

4.2.3.2 G2.3.2—Negative Reactivity Insertion

4.2.3.2.1 G.2.3.2(a)—Identification of Criteria

The means of negative reactivity insertion are design dependent. Accordingly, criteria to ensure that the means of negative reactivity insertion are not obstructed during normal operation or accident conditions cannot be provided on a generic basis.

Fuel qualification is important because fuel assemblies, fuel structures, or both may form part of the negative reactivity insertion path. Reactor designs that use TRISO fuel may or may not have fuel assemblies or fuel structures that form part of the negative reactivity insertion path (e.g., fuel may be placed in a graphite block where the graphite block—not the fuel itself—forms part of the negative reactivity insertion path).

4.2.3.2.2 G.2.3.2(b)—Evaluation Model

An evaluation model to assess the means of ensuring negative reactivity insertion is expected to be done on a design-specific basis.

4.3. Evaluation Model

Section 3.3 of NEA-RPFQ and section 3.3 of NUREG-2246 describe evaluation models generically such that the models may be sophisticated analytical tools like computer codes, simplified mathematical expressions, or comparisons against data. While the CNSC and the NRC do not have sufficient information to assess specific computational codes on a generic basis the information in this section addresses the needs of an evaluation model to adequately assess UCO TRISO fuel.

4.3.1. EM G1—Evaluation Model Capabilities

4.3.1.1 EM G1.1—Geometry Modeling

The evaluation model should include the geometry associated with the fuel form, which is spherical for the particle and typically spherical or cylindrical for the fuel form (pebble or compact). TRISO particles may contain a specified amount of asphericity, defined by the ratio of major to minor axis lengths. This ratio is referred to as the aspect ratio. Table 5-5 of EPRI-AR-1(NP)-A provides the measured AGR-1 and AGR-2 aspect ratios. An aspherical particle has a higher localized SiC stress, thereby increasing the probability of SiC overpressurization failure. Therefore, the evaluation model must account for the aspect ratio defined by the fuel specification. The higher SiC stress due to particle asphericity is evaluated using either a two-dimensional or three-dimensional model [20].

Pebble or compact fuel forms are usually modeled assuming symmetry consistent with the fuel form. Manufacturing tolerances associated with the fuel form should be evaluated but are expected to have a negligible effect on TRISO particle performance.

4.3.1.2 EM G1.2—Material Modeling

The evaluation model should be capable of modeling fuel material properties and environmental conditions, including material property changes with irradiation. For a TRISO-based fuel system, the properties needed include the following:

- melting temperature
- thermal conductivity
- specific heat capacity
- thermal expansion
- swelling
- density
- fission product diffusivity
- elastic modulus and Poisson's ratio
- strength and Weibull modulus

4.3.1.3 EM G1.3—Physics Modeling

Addressing EM G1.3, "Evaluation model is capable of modeling the physics relevant to fuel performance," in NUREG-2246 and NEA-RPFQ requires knowledge of failure mechanisms, including changes due to irradiation and exposure to the in-reactor environment. Several degradation mechanism and failure modes have been identified for TRISO fuel based on past experience, legacy data, and the use of expert panels [1], [21], [22], [23]. Some of the degradation mechanisms and failure modes identified through past experience or expert panels have been addressed by the development of UCO-TRISO fuel or have not been observed in testing. The NRC and the CNSC expect that some failure modes and degradation mechanisms will be addressed through evaluation models for fuel performance, and some may be addressed by G1, "A fuel manufacturing specification controls the key fabrication parameters that significantly affect fuel performance," from NUREG-2246 and NEA-RPFQ. The treatment for each of the identified degradation mechanisms and failure modes (i.e., analyze with an evaluation model, control through manufacturing, or other treatment) is an ongoing effort that will be discussed in future reports. Degradation mechanisms and failure modes are identified below:

- **Pressure vessel failure of standard ("intact") particles**—Tensile stress in the SiC layer exceeds the strength of the SiC layer [1], [21].

- **Pressure vessel failure of particles with defective or missing coatings**—Pressure vessel fails due to manufacturing defect. Some number of defective particles is expected, in part because of the large number of TRISO particles present in the reactor [1].

- **Irradiation induced IPyC cracking failure**—Cracking of the IPyC layer may occur during irradiation-induced shrinkage due to the buildup of internal stresses when the internal stresses become greater than the fracture strength [21].

- **SiC thermal decomposition failure**—Exposure to high temperature causes decomposition of the SiC layer. Radionuclide release from TRISO fuel due to SiC layer thermal decomposition is generally not observed at temperatures below 1,600 ºC [21].

- **Debonding between IPyC and SiC layers failure**—Debonding occurs when the radial stress that develops between the IPyC and SiC layers, due to shrinkage of the IPyC layer, exceeds the bond strength between layers [1].

- **Kernel migration failure**—Failure occurs when movement of the fuel kernel penetrates the TRISO coating. Kernel migration occurs when a thermal gradient exists across the particle and the chemical equilibrium C/CO is different on each side of the particle. Mass transport of CO is moved down the temperature gradient, and the kernel is moved up the temperature gradient [1], [21].

- **Fission product attack failure**—Degradation of the SiC layer can occur due to interaction with fission products, specifically palladium [1].

- **Nonretentive SiC failure**—The SiC layer can be degraded through corrosion by CO and interaction with cesium. Corrosion by CO is assumed to happen at elevated temperatures if the IPyC layer is porous or cracked. The exact mechanism of degradation by cesium is not well known. This phenomenon may be a bigger factor at higher burnup values [21], [22].

- **Creep failure of pyrocarbon**—Thinned and failed pyrocarbon has been observed in some post-irradiation heating tests. These results were determined for test with temperatures greater than 2,000℃ for long durations. The observed failures did not lead to failure of the SiC layer [21].

- **Kernel-coating mechanical interaction failure**—Mechanical interaction can occur between TRISO layers due to kernel swelling, closing the gaps between the kernel and coatings. This failure has not been reported experimentally, but this failure mechanism may be a bigger factor at higher burnup values [21], [23].

4.3.2. EM G2—Evaluation Model Assessment

As stated in NUREG-2246, Section 3.3.2, the purpose of the evaluation model assessment is to ensure the model conservatively predicts degradation or failure for normal operation, AOOs, and design-basis accidents when compared to appropriate experimental data. The evaluation model should consider any basis or uncertainties to conservatively bound the degradation or failure fraction cohorts (i.e., the combinations of coating layer failures). For UCO TRISO particles, some failure modes described in section 4.3.1 may not have to be considered in the evaluation model, based on experimental data observations and the expected performance envelope of the design. For example, UCO kernels with UCx content consistent with that described in section 4.1.2.1 might not consider kernel migration as a failure mode. Since the degradation and failure modes are a function of both manufacturing and the design-specific performance envelope, justification should be provided for modes not considered in the evaluation model.

Not all failure models can be assessed as reflected in Section 7.4, "SiC Failure Mechanisms," of EPRI-AR-1(NP)-A, which notes that the dominant SiC failure mechanisms described differ significantly from those currently embedded in the fuel performance models. Incorporation of this failure mode into the models is likely to be challenging due to the complex nature of coating layer interactions (buffer-IPyC interface interactions and a focused chemical attack of the SiC layer) and a lack of some key data (e.g., buffer-IPyC bond strength). If the evaluation model is unable to predict an observed failure mode, the failure mode should be addressed by incorporating experimental data or providing justification that the overall failure fraction is sufficiently conservatively to account for failure modes not modeled. The ability of the evaluation model to demonstrate its ability to predict degradation or failures over the test envelope is supported by the four supporting goals discussed below.

4.3.2.1 EM G2.1—Experimental Data

Section 4.4.2 describes the experimental data used as the assessment data.

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

The ability of the evaluation model to predict the experimental data should include establishing biases and uncertainties and identifying any limitations in the evaluation model. The following support demonstration of the predictive capability of the model over the test envelope.

4.3.2.2.1 EM G2.2.1—Evaluation Model Error is Quantified

To ensure a conservative evaluation model, quantification of the evaluation model error is necessary. As stated in NUREG-2246, either a statistical confidence level, if enough data exist, or a bounding conservative approach may be used to determine model error. For TRISO particles, the evaluation model should conservatively predict releases from intact coating layers, in addition to the number of failed fraction cohorts. Evaluation model biases and uncertainties will be specific to the computation model and assessment data.

4.3.2.2.2 EM G2.2.2—Validation Data Cover Performance Envelope

The validation data used should be well distributed within and bound the design performance envelope. If the validation data do not bound the performance envelope, additional justification should be provided, demonstrating the figures of merit (e.g., in-service failure fraction) are conservatively predicted. Section 4.4.2 gives the AGR program steady-state irradiation and safety case data test envelope. Examples demonstrating the AGR program distribution of fuel temperature data versus duration are given in EPRI-AR-1(NP)-A, figures 6-26, 6-27, and 6-28. These figures indicate the data collected were well distributed within the temperature ranges examined.

4.3.2.2.3 EM G2.2.3—Use of Spare Data Is Justified

As discussed in NUREG-2246, fewer data may be available for some areas within the performance envelope. Generally, the data density for steady-state irradiation, consistent with normal operations, is higher than near the edges of the performance envelope associated with accident conditions. Though this data density assumption is true for the AGR program, there is still adequate data across the entire temperature range tested: 412,336 particles irradiated at steady-state time-averaged conditions (1360 ºC) and 45,804 and 26,028 particles at 1,600°C and 1,800°C, respectively. Thus, additional experimental data is not needed within the tested temperature envelope. For operating conditions beyond these temperatures additional experimental data would be required.

4.3.2.2.4 EM G2.2.4—Application Domain Is Consistent with Model Assessment

The evaluation model should be restricted to application domains for which the model has been assessed. The end of irradiation time-average temperatures for AGR-1 and AGR-2 are summarized in table 6-4 in EPRI-ARI-1(NP)-A. The time-average temperature in table 6-4 ranges from 800 to 1,360°C. While the time-average maximum temperature should be limited to 1,360°C, short-term operation (~75 days) up to 1,500°C is supported by the AGR-2 Capsule 2 data, as shown by figure 6-28 in EPRI-ARI-1(NP)-A, without identifying any cliff-edge effects (i.e., no large increase in failed coating layers associated with the 1,500°C population).

4.4. Assessment of Experimental Data

The assessment of experimental data used to qualify a fuel design is a key step to ensuring predictable fuel performance and evaluating model performance. The experimental data should be collected over the test envelope, which covers the expected fuel performance envelope; accurately measured; and represent prototypical conditions. Also, the assessment data should be sufficient such that an evaluation model can be developed and still provide independent data to assess the evaluation model results (i.e., data not used in training the model).

4.4.1. ED G1—Independence of Validation Data

In an optimal situation, sufficient experimental data exist to both develop the model and assess the evaluation model experimental error. Often, sufficient steady-state irradiation data exist to develop the evaluation model and determine independent, statistically derived upper and lower tolerance limits. The AGR program irradiated a significant number of particles at steady-state, as discussed in section 4.3.2.2.3, and hence a sufficient body of data likely exists for both evaluation model development and an independent error determination. For the safety tests, fewer particles were tested but may still be adequate to develop a statistical error.

While specific particle performance parameters and potential failure modes are difficult to predict, the main parameter of interest is the fractional fission product release by isotope of interest, which is a function of steady-state and transient in-service intact and failed coating cohort releases. Manufactured coating defect cohorts and the amount of dispersed uranium are usually set by a manufacturing specification and may or may not be part of the fuel performance evaluation model. The AGR irradiation data provides measured fractional fission product releases by isotope of interest, which can be compared to code predictions. However, for normal operations and AOOs, the expected small release of these radionuclides by diffusional mechanisms is likely not a safety concern. For accident conditions, when a larger quantity of fission products can be released due to increased diffusion and particle failure fractions, it is necessary to conservatively predict the release of these radionuclides for dose assessment calculations. In this case, AGR data can be used to validate code failure fractions and the overall fission product releases. Ultimately, the applicant must determine whether sufficient data exist for a statistical error determination based on an assessment of the AGR experimental data and the performance envelope of the design. If a statistical error method is not implemented, a bounding evaluation model error should be estimated and will be evaluated on a case-by-case basis.

4.4.2. ED G2—Test Envelope

During normal operation and AOOs, fission products can be released from TRISO particles due to coating manufacturing defects, in-service failures, and diffusion through intact particles. The fuel performance envelope directly affects the in-service failures and release through intact particles. For TRISO particles manufactured consistent with AGR program specifications, the steady-state irradiated values appear in table 4, which is derived from EPRI-AR-1(NP)-A, table 6-6. The steady-state irradiated values are analogous to normal plant operation and provide the initial fuel conditions for transient and accident analyses. Figure 6-29 of EPRI-AR-1(NP)-A gives an example comparing the AGR time-average fuel temperature distribution irradiation test data to an expected HTGR normal operating condition.

Table 4 Maximum Steady-State Irradiated Values for Key Parameters for AGR-1 and AGR-2

Property	AGR-1 and AGR-2 Capsules 5 and 6	AGR-2 Capsule 2
Burnup (% FIMA)	19.6[a]/13.2[b]	13.2[b]
Fast fluence (n/m^2 x 10^{-25}; E > 0.18 MeV)	4.30	3.47
Peak time-average temperature (°C)	1,210	1,360
Time-average compact power density (W/cm^3)	90	92
Time-average particle power (mW/particle)	66[c]/86[d]	88

a. Burnup limit corresponds to AGR-1 with a 350 µm kernel diameter.
b. Burnup limit corresponds to AGR-2 with a 427 µm kernel diameter as described in Section 4.1.1.1
c. AGR-1 values
d. AGR-2 Capsule 5 and 6 values

The AGR-2 fuel kernel diameter is larger than that of the AGR-1, and hence the burnup is limited to 13.2 percent FIMA or to the maximum value that maintains the STSM below a value of 0.810 at the 99th percentile to remain consistent with the normalized AGR stress performance. For peak particle burnups beyond these values, additional justification is needed and will be evaluated on a case-by-case basis.

For transient conditions consistent with an AOO, a peak particle temperature of 1,600°C is a reasonable estimate of an upper bound value. This temperature is based on EPRI-AR-1(NP)-A figure 7-6, which demonstrates the typical AGR program intact releases over approximately 325 hours, and figure 7-15, which depicts the SiC layer and full TRISO failure fractions (upper limit at 95 percent confidence) for AGR-1 and AGR-2. Design-specific considerations may impact the temperature value used as an acceptance criterion; 1,600°C is used here as a representative upper bound value based on the AGR program referenced safety tests (slow heat-ups), but another value may be appropriate. In addition, design-specific transients that introduce very rapid heat-ups (e.g., large rapid positive reactivity insertions) may require other criteria, such as preventing kernel melt, beyond the chosen peak particle temperature.

Figure 7-6 reveals there are relatively low fractional fission product releases except for radiosilver (Ag-110m). Generally, high AOO particle temperatures occur for durations significantly shorter than 325 hours (i.e., typical time scale of minutes), and the earlier fractional releases (i.e., under approximately 50 hours) shown in figure 7-6 are more representative of the expected behavior. As shown in figure 7-15, only a small number of SiC failures occur at 1,600°C (at the 95 percent confidence level) such that continued operation following such a transient may be possible.

Ultimately, whatever peak particle temperature is used, the applicant is still required to demonstrate the SARDDL is not violated, ensure the appropriate dose criteria or limits are met, and show that the plant is able to restart while accommodating any continuing normal operation in-service failures (i.e., primary side activity remains within technical specifications and a subsequent AOA will remain below the SARDDL). Higher AOO peak TRISO particle temperatures may be acceptable based on future data, the assumed retention capabilities, or the retention capabilities of other credited fission product barriers needed to demonstrate compliance with the appropriate dose criteria or limits (e.g., mechanistic source term analysis results).

Generally, design-basis accidents (or licensing basis events of high consequence but low probability of occurrence) have peak TRISO temperatures or a higher population of particles at temperatures higher than normal operations and AOOs (or both). Both intact coating releases and in-service coating failures are expected to increase with the higher design-basis-accident temperatures as demonstrated by the releases shown in figures 7-7, 7-8, and 7-15 of EPRI-AR-1(NP)-A. In general, UCO plots in figures 7-7 and 7-8 show increasing releases with temperature. Likewise, the number of expected transient failures also increases above 1,600°C, as illustrated in EPRI-AR-1(NP)-A figure 7-15 and summarized in table 7-2; there is a relatively large increase in the number of SiC failures between 1,700°C and 1,800°C (7 at 1,700°C versus 23 at 1,800°C), and the 95 percent confidence level increases by a factor of approximately 2. In addition, the data indicate a small number of TRISO failures begins to occur at temperatures above 1,700°C, but it is important to note the 95 percent confidence level demonstrates a low probability of TRISO failures at 1,800°C. Ultimately, the value of the design-basis accident peak particle temperature is a function of the role the particle plays in fission product retention, the assumed retention capabilities, or the retention capabilities of other credited fission product barriers needed to demonstrate compliance with the appropriate dose criteria or limits (e.g., mechanistic source term analysis results). As noted above in the discussion regarding AOOs, acceptance criteria in addition to a peak particle temperature may be needed to ensure the appropriate dose criteria or limits are met for a specific design.

The AGR program safety testing, which was performed at temperatures of 1,600°C, 1,700°C, and 1,800°C, as shown in table 7-2 of EPRI-AR-1(NP)-A, focused on heat-up accidents associated with HTGR designs, particularly the depressurized loss-of-forced cooling (DLOFC) and pressurized loss-of-forced cooling (PLOF). The DLOFC is generally considered the most limiting design-basis accident for HTGRs [24], as rod ejections are normally precluded by design. Rod ejections are also normally precluded by design for FHR designs due to the low differential pressures across the primary-side coolant boundary. The out-of-pile AGR safety tests heat fuel compacts in a furnace at rates that represent those expected of DLOFC and PLOF events (0.01°C per second (s) to 0.10°C/s). Even excluding the rod ejection event, reactivity-initiated events, also referred to as overpower transients, such as control or element withdrawal, can generate heat-up rates significantly faster than those tested in the AGR program (1°C/s to 1,000°C/s) [24].

Since the AGR program heat-up rates are relative slow, additional justification is needed to address events with faster heat-up rates, especially for events that potentially introduce a higher temperature gradient across the particle. Two failure modes, kernel melt and kernel-coating mechanical interactions, could potentially cause SiC or TRISO failures during an overpower event. A series of integral fuel safety tests were conducted on fresh UO_2 fuel for reactivity insertions consistent with a rod ejection event [25]. Though these events are typically precluded by design in many advanced reactor designs, the rod ejection safety tests provide insight to the energy deposition needed to cause failure of a TRISO particle. Table 5 shows the energy deposited and the estimated energy to cause failure.

Table 5 TRISO Rod Ejection Reactivity-Initiated Accident Tests

Reactor	Kernel	Type	Energy Deposition	Pulse Width	Fuel Failure
NSRR	UO_2	Element and loose particles	500–2,300 J/g UO_2	5 ms	>1,400 J/g
NSRR	UO_2	Loose particles	500–1,700 J/g UO_2	5 ms	>1,400 J/g
HYDRA	UO_2	Element and loose particles	100–1,700 J/g UO_2	1–2 ms	>1,300 J/g
IGR	UO_2	Loose particles	>10,000 J/g UO_2	700–30,000 ms	Matrix failure

In addition to determining the energy deposition from failure, the work of Umeda et al. (2010) [26] provides insights into the energy deposition needed to develop cracks in the kernel, which could lead to stresses in the buffer and coating layers. As shown in figure 1 of Umeda et al. (2010) and reproduced in figure 1 below, energy depositions greater than approximately 580 joules per gram (J/g) (.58 kilo (k)J/g) UO_2 are necessary to begin kernel crack formation.

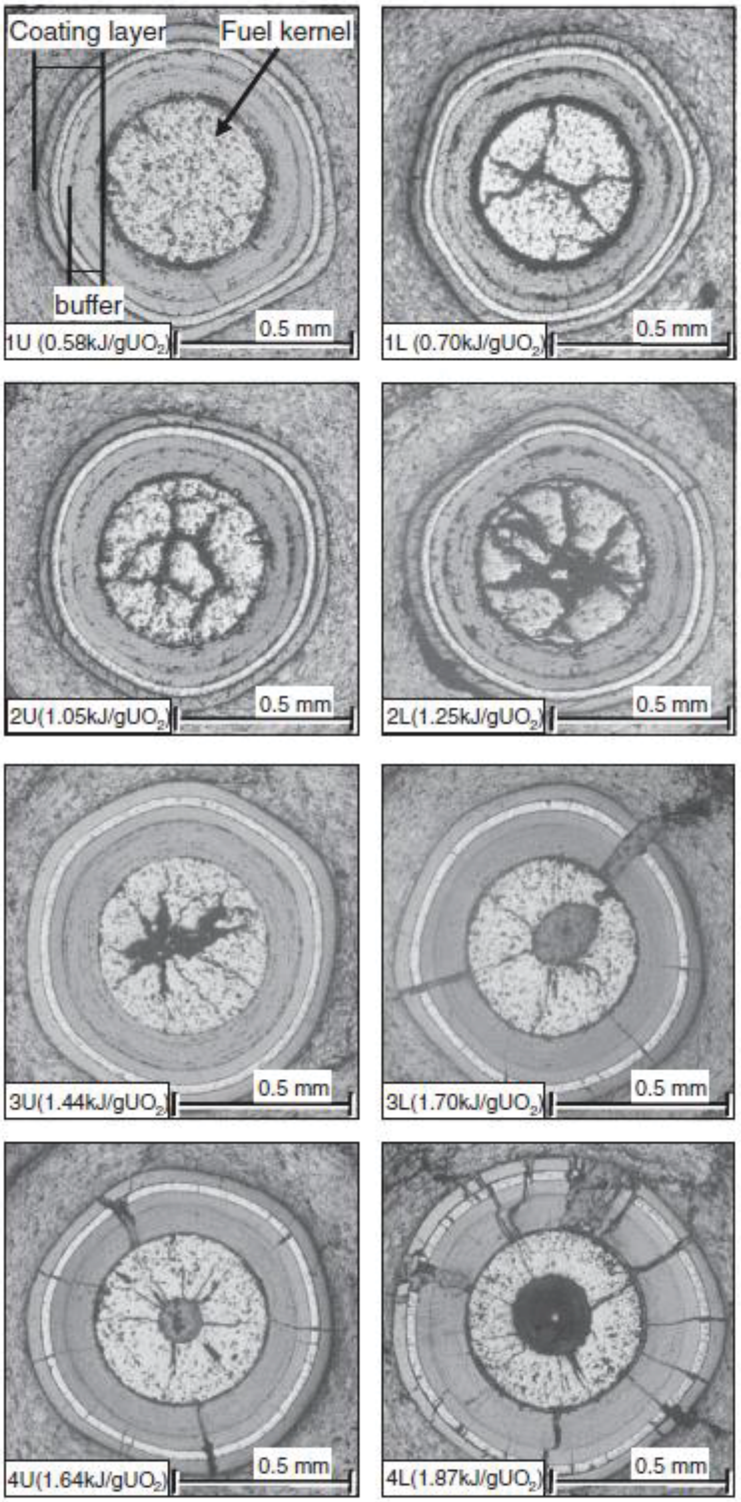

Figure 1 Post test visual examination from reactivity-initiated accident testing [26]

If rod ejection is considered as part of the design basis, it should be noted that the expected failure threshold energy for irradiated UCO fuel will be less than presented in table 5 based on the lower melting temperature of UCx and the addition of irradiation-generated solid fission fragments increasing the kernel size and reducing the buffer porosity.

The work described by table 5 and presented in figure 1 is representative of a rod ejection event in which the heat generated does not have adequate time to dissipate through the particle and into the surrounding environment. The time over which a power excursion caused by a rod or element withdrawal occurs is a function of the specific reactor kinetics and mechanical limits associated with the rod or element drive system. Overpower transients caused by events such as control rod withdrawal usually occur on the order of seconds verses the millisecond rate of an ejected rod. For overpower transients on the order of seconds, the heat generated in the kernel has time to dissipate to the surrounding environment. An analysis to demonstrate this was performed in response to an NRC request for additional information during the Next Generation Nuclear Plant (NGNP) licensing process [27]. As shown in figure 2, for energy deposition durations over 1 second, the change in the maximum temperature across the particle is small and largely independent of the initial energy deposited. This indicates the heat generated has time to transfer to the environment, and large temperature-induced stresses (e.g., caused by fission gas release, kernel expansion) are not expected.

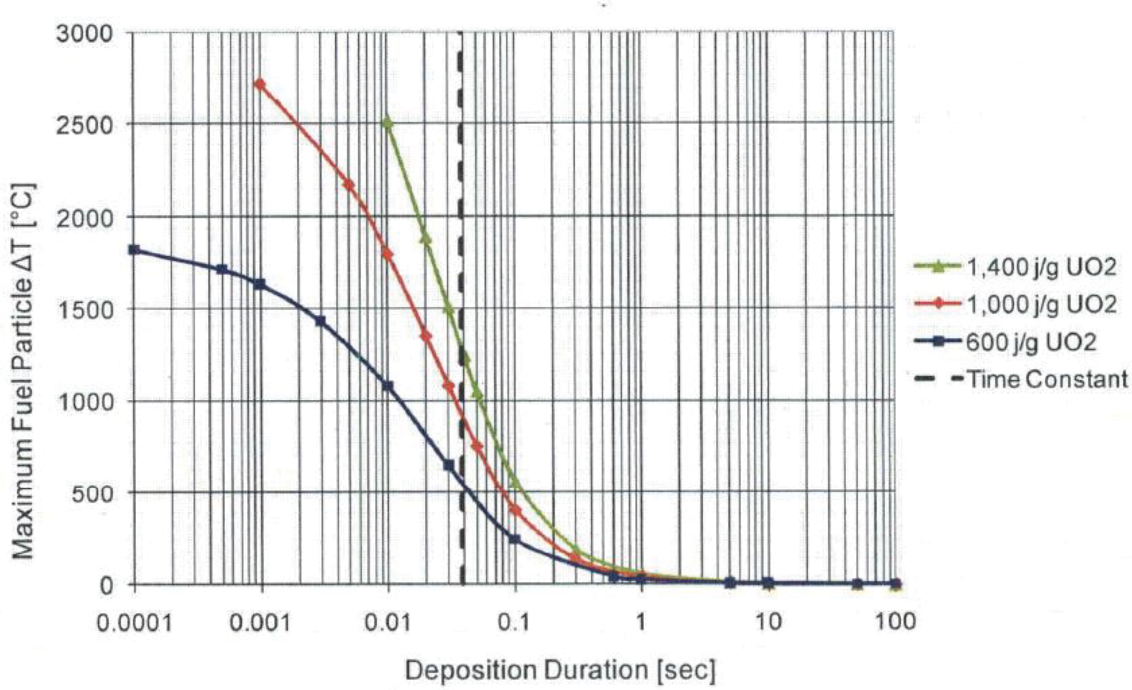

Figure 2 Overpower maximum fuel particle change in temperature (ΔT) versus deposition duration.

Based on the short TRISO particle thermal time constant consistent with established designs, failures due to melt and mechanical kernel-coatings interaction for overpower events that have durations greater than 1 second are not expected or will be relatively low (e.g., failures may occur at limits of the manufacturing specification ranges). A design-specific evaluation or justification should be performed that evaluates a range of overpower event reactivity insertions to ensure margin to kernel melt, maximum SiC temperature, and other failure modes evaluated in the fuel performance evaluation model (e.g., SiC overpressurization, palladium attack) of other events either remain bounding or any increases in intact and failed fuel fractions are conservatively estimated.

The test envelope should consist of irradiation tests which cover the expected normal operation and transient conditions appliable to the design. Transient conditions typically considered to challenge fuel releases are heat-up and reactivity induced overpower events. As discussed in Section 4.3.2.2.3, the AGR program consists of significant number of steady-state irradiations and safety case tests which adequately address the heat-up type events. The AGR program did not perform tests associated with a rapid positive reactivity insertion. Based on the short TRISO thermal time constant a design specific justification can be made that overpower testing is not necessary if rod ejection events are precluded.

4.4.3. ED G3—Data Measurement

An understanding of measurement accuracy is important to establish confidence in the data used to develop and assess evaluation models or to establish acceptance limits. The goal of establishing confidence in the experimental data is supported by three areas: test facility quality assurance, measurement techniques, and experimental uncertainty quantification. Experimental data should be collected under an appropriate quality assurance program that meets applicable regulatory requirements.

4.4.3.1 ED 3.1—Test Facility Quality Assurance and G3.2—Measurement Techniques

The assessment of the QAPD included the manufacturing, irradiation, safety testing, and post irradiation examination (PIE) associated with the AGR program. Hence, the QAPD covers both G3.1, "Test Facility Quality Assurance" (i.e., Advance Test Reactor irradiation), and measurement techniques, which would apply to both fabrication and post irradiation examinations. As discussed in the staff's safety evaluation of EPRI-AR-1(NP)-A, the NRC assessed the NGNP Quality Assurance Program Description (QAPD) as part of the NGNP licensing effort. The staff found that the quality assurance program met the criteria of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 and was acceptable for use during the technology development and high-level design phases of the NGNP project. The staff's safety evaluation of EPRI-AR-1(NP)-A agreed with the previous staff's assessment that the AGR program quality assurance plan was acceptable for technology development. Therefore, the AGR program satisfies the goal of providing quality data necessary for licensing applications.

4.4.3.2 ED 3.3—Experimental Uncertainties

Section 6.5 of EPRI-AR-1(NP)-A contains a detailed assessment of AGR-1 experimental uncertainties. For key variables expert judgment was used to select uncertainty ranges. These key variable uncertainties were used to statistically determine the time-average volume average and time-average maximum fuel temperatures. Table 6-5 of the EPRI report provides AGR-1 capsule temperature uncertainties for the time-average, volume-average fuel (TAVA) and time-average maximum fuel temperatures. The overall uncertainty in the calculated temperatures for AGR-1 ranged from 2.0 to 6.5 percent (-40 to 60°C at one sigma and 100 to 120°C at two sigma), depending on irradiation time (thermal conditions), the capsule, and the temperature parameter being predicted. Similar values were reported for the AGR-2 capsule analyses, with uncertainties at one sigma from 30 to 40°C for the TAVA and 35 to 45°C for the maximum time-average temperature. Based on this information, the AGR program satisfies the goal of establishing measurement uncertainties for use in licensing applications.

4.4.4. ED G4—Test Conditions

The test conditions should be representative of the prototypical conditions. Satisfying this goal is achieved by satisfying two subgoals: (1) the test specimens are fabricated consistent with the fuel manufacturing specification, and (2) distortions are justified and accounted for in the experimental data.

4.4.4.1 ED G4.1—Manufacturing of Test Specimens

The AGR program fuel was manufactured within the fuel specification as noted by INL/MIS-055-00238 [13] and SPC-923 [14] for AGR-1 and AGR-2, respectively. Therefore, the AGR-1 and AGR-2 programs meet the manufacturing test specimen goal.

4.4.4.2 ED G4.2—Evaluation of Test Distortions

Test distortions arise from differences between the test and actual conditions under which the fuel is expected to perform. The AGR program was constructed to support the expected needs of HTGRs as shown by figure 6-26 of EPRI-AR-1(NP)-A. As discussed in this section, the AGR program test envelope contains a wide array of operating conditions that satisfy both the steady-state and transient conditions expected for most HTGR and FHR designs. If anticipated plant operating conditions are outside the range of the AGR test envelope, then distortions due to operating conditions should be evaluated.

The TRISO particles created by the AGR program are full scale and hence do not introduce any scale distortion. Likewise, burnup was accumulated in a continuous and approximately linear fashion representative of actual in-service burnup accumulation (figures 6-9 and 6-10 of EPRI-AR-1(NP)-A). Compacts or pebbles may be physically different in size than those used in the AGR program, but those distortions can likely be accommodated analytically if the matrix material properties are well known. While a final determination of distortions is dependent on the final proposed design, the AGR program provides multiple kernel sizes, a range of coating layer attributes, and an extensive test envelope dataset such that distortions are minimized.

5. Conclusions

The CNSC and U.S. NRC staff have established a common regulatory position on TRISO fuel qualification based on existing knowledge (e.g., AGR program) and identified design-dependent analytical or testing gaps that should be addressed to enable TRISO use in advanced reactor licensing applications. Specifically, this paper summarizes the data, criteria, and approaches for supporting a regulatory finding that fission product retention is sufficient for anticipated TRISO-fueled reactor designs. Though final particle releases are specific to the applicant's design (e.g., retention by other fission product barriers) and siting needs, the existing data and analyses summarized here can be generically applied when the particle attributes and testing parameters provided in tables 1, 2, and 4 are used.

The AGR program provides useful insight to manufacturing specifications that control the number and type of manufactured defects, the amount of dispersed uranium, and attributes of the SiC coating layer. The SiC coating is the primary fission product barrier in TRISO fuel and, as noted in section 4.1.3.1.1, grain size is a consideration in the SiC coating layer's ability to retain fission products. While the visual standard provides an adequate qualitative means to evaluate SiC grain size, a quantitative measure would enhance the understanding of the as-built grain size distribution which yielded AGR program fission product releases.

The extent and quality of the AGR-1 and 2 TRISO particle data are sufficient for steady-state evaluation model development within the bounds described in table 4. The AGR-1 and AGR-2 safety test data are also sufficient within the range of conditions tested (slow heat-up events) and for overpower event durations of 1 second or greater. The AGR program data cover a wide range of conditions consistent with the range of many proposed TRISO-fueled reactor designs (e.g., HTGR, FHR, and heat-pipe), but the final determination will be based on the applicant's design-specific submittal(s).

The role of the fuel compact is to contain and protect the TRISO particles. The fuel compact historically consists of a carbonaceous matrix material that envelops the particles. The fuel compact design ensures maintenance of a coolable geometry and, in some designs, control rod or element insertion under all design-basis conditions. Traditionally, fuel compact designs take the form of pebbles or cylinders, but they are not limited to these forms. The AGR program included cylindrical fuel compacts typically associated with noble-gas-cooled prismatic designs. The AGR program provides useful information for noble-gas-cooled reactors but offers limited information for designs that have non-noble-gas coolants such as FLiBe. Accordingly, qualification for the fuel compact/pebble should be addressed on a design-dependent basis.

The AGR program provides the largest and most modern experimental dataset for UCO particle fuel. The particle data are of sufficient quality and quantity to qualify the TRISO particle within the tested performance envelope. For particle specifications (e.g., kernel size greater than AGR-2) and operating conditions that lie outside those tested by the AGR program, additional testing or justification is needed to qualify the TRISO particle. The AGR program compact data may be sufficient for licensing specific designs, but the applicant will be responsible for qualifying compact designs that meet its design needs. That demonstration could involve testing, analysis, or comparison to historical data if the data are of sufficient quality.

6. Works Cited

[1] EPRI, *Transmittal of Published Topical Report EPRI-AR-1(NP)-A, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance",* 2020 (ADAMS Accession No. ML20336A052).

[2] Kairos Power, *KP-TR-010-NP, "KP-FHR Fuel Performance Methodology",* 2020 (ADAMS Accession No. ML20318A226).

[3] Kairos Power, *Submittal of the Preliminary Safety Anlaysis for the Kairos Power Fluride Salt Cooled, High Temperature Non-Power Reactor (Hermes),* 2021 (ADAMS Accession No. ML21272A376).

[4] X-energy, *Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology,* 2021 (ADAMS Accession No. ML21246A289).

[5] CNSC, *REGDOC-3.5.4, "Pre-Licensing Review of a Vendor's Reactor Design",* 2018.

[6] NRC, *White Paper: Draft Pre-application Engagement to Optimize Advanced Reactors,* 2021 (ADAMS Accession No. ML21145A106).

[7] NEA, *Regulatory Perspectives on Nuclear Fuel Qualifcation for Advanced Reactors,* 2020 (ADAMS Accession No. ML22018A099).

[8] NRC, *NUREG-2246, "Fuel Qualification for Advanced Reactors",* 2022 (ADAMS Accession No. ML22063A131).

[9] *Nuclear Energy Innovation and Modernization Act,* 2019 (https://www.congress.gov/115/bills/s512/BILLS-115s512enr.pdf).

[10] NRC, *Regulatory Guide 1.232, "Guidance for developing principal design criteria for non-light-water reactors",* 2018 (ADAMS Accession No. ML17325A611).

[11] CNSC, *REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants",* 2014.

[12] CNSC, *REGDOC-2.4.5, "Nuclear Fuel Safety" (Draft),* 2022.

[13] INL, *INL/MIS-05-00238-Revision-1, "AGR-1 Fuel Product Specification and Characterization Guidance",* 2006.

[14] INL, *SPC-923, Revision 3, "AGR-2 Fuel Specification",* 2009.

[15] EPRI, *Responses to Requests for Additional Information (RAIs) on Topical Report EPRI-AR-1(NP), "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance",* February 26, 2020 (ADAMS Accession No. ML20058A040).

[16] F. J. Homan, T. B. Lindemer, E. L. Long, T. N. Tiegs and R. L. Beatty, "Stoichiometric Effect on Performance of High-Temperature Gas-Cooled Reactor Fuels from U-C-O System," *Nuclear Technology,* vol. 35, pp. 428-441, 1977.

[17] J. W. McMurray, T. B. Lindemer, N. R. Brown, N. R. Reif, R. N. Morris and J. D. Hunn, "Determining the minimum required uranium carbide content for HTGR UCO fuel kernels," *Annal of Nuclear Energy,* vol. 104, pp. 237-242, 2017.

[18] INL, *SPC-1352, Revision 8, "AGR-5/6/7 Fuel Specification",* March 9, 2017.

[19] T. J. Gerczak, J. D. Hunn, R. A. Lowden and T. R. Allen, "SiC layer microstructure in AGR-1 and AGR-2 TRISO fuel particles and the influence of its variation on the effective diffusion of key fission products," *Journal of Nuclear Materials,* vol. 480, pp. 257 270, 2016.

[20] W. Jiange, J. D. Hales, B. W. Spencer, B. P. Collin, A. E. Slaughter, S. R. Novascone, A. Toptan, K. A. Gamble and R. Gardner, "TRISO particle fuel performance and failure analysis with BISON," *Journal of Nuclear Materials,* vol. 548, 2021.

[21] PNNL, *PNNL-31427, TRISO Fuel: Properties and Failure Modes,* 2021 (ADAMS Accession No. ML21175A152).

[22] NRC, *NUREG/CR-6844 Main Report, Vol. 1, "TRISO-Coated Particle Fuel Phenomena Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents,* 2004.

[23] IAEA, *TECDOC-1645, "High Temperature Gas Cooled Reactor Fuels and Materials",* 2010.

[24] N. R. Brown, "A review of in-pile fuel safety test of TRISO fuel forms and future testing opportunties in non-HTGR applications," *Journal of Nuclaer Materials,* vol. 543, June 2020.

[25] K. Fukeda, K. Hayashi and K. Shiba, "Fuel Behavior and Fission Product Release under HTGR Accident Conditions-Fission Product Transport Processes in Reactor Accidents,," Hemisphere Publishing, New York, 1990.

[26] M. Umeda, T. Sugiyama, F. Nagas, T. Fuketa, S. Ueta and K. Sawa, "Behavior of Coated Fuel Particle of High-Temperature Gas-Cooled Reactor under Reactivity-Initiated Accident Conditions," *Journal of Nuclear Science and Technology,* pp. Vol 47:11, pp. 991 997, 2010.

[27] INL, "CNN 224915, Contract No. DE-AC07-05ID14517—Next Generation Nuclear Project Submittal—Response to Nuclear Regulatory Commission Request for Additional Information".

[28] NRC, "NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" 1995 (ADAMS Accession No. ML041040063).

[29] IAEA, *IAEA Safety Glossary - Terminology Used in Nuclear Safety, Nuclear Security, and Radiation Protection 2022 (Interim Edition,* 2022.