US-NRC – CNSC Memorandum of Cooperation

INTERIM JOINT REPORT #2
concerning
Tristructural Isotropic (TRISO) Fuel Qualification

May, 2022

DISCLAIMER: The NRC and CNSC have prepared this interim 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 SMR 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 other reviews to the extent practicable.

Activities under the MOC are coordinated by a subcommittee of the USNRC-CNSC Steering Committee, called the Advanced Reactor Technologies and Small Modular Reactors (ART-SMR) Sub Committee (the 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 a Terms of Reference and are intended to:

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

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

1. To what extent is the vendor engaging in meaningful pre-licensing 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 pre-licensing 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 their design with both regulators to enable cooperation to occur?

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1. Introduction

This section documents the history underpinning the decision by the US-NRC and CNSC to establish this cooperative activity.

1.1. Relevant Vendor Engagement with the US-NRC

The US-NRC reviewed the Electric Power Research Institute (EPRI) topical report, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance" in 2020 [1]. Additionally, NRC has been invovled with two advanced reactors vendors that are proposing the use of TRISO fuel in their reactor designs, Kairos Power (KP) and X-energy. NRC reviewed topical report KP-TR-010-NP, "KP-FHR Fuel Performance Methodology," [2] in 2021 which describes a plan to validate the use of the BISON fuel performance code to the KP-FHR (Fluoride-salt-cooled High-temperature Reator). Additionaly, KP submitted an application for a non-power construction permit in October 2021 (add reference). In 2020, the United States Department of Energy (US-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 by 2027 as part of their Advanced Reactor Demonstration Program (ARDP). Pre-application interaction with X-energy has been increasing over the recent years, including the submittal of, "Xe-100 Topical Report: TRISO-X Pebble Fuel Qualification Methodology," in 2021 [3]. Licensing activities with X-energy (e.g., topical reports) are anticipated to incresae in accordance with the ARDP award.

1.2. Relevant Vendor Engagement with the CNSC

The CNSC has been involved with two advanced reactors vendors that are proposing the use of TRISO fuel in their reactor designs, X-energy and Ultra-Safe Nuclear Corporation (USNC) via the CNSC Vendor Design Review (VDR) process. A VDR is a feedback mechanism that enables CNSC staff to provide feedback early in the design process based on a vendor's reactor technology. Nuclear power plant designs can include small modular reactor (SMR) concepts, advanced reactor concepts or more traditional designs. The assessment is completed by the CNSC at the request of the vendor. The VDR process is described in detail in CNSC REGDOC-3.5.4 [4].

The word "pre-licensing" signifies that a design review is undertaken prior to the submission of a license application to the CNSC by an applicant seeking to build and operate a new nuclear power plant. An application by a vendor for a review is *not* an application for a license to prepare a site or to construct or operate a nuclear power facility, and is not an indication of intent to proceed with a project. The objective of a 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 assures that a resolution path exists for any design issues identified in the review.

Global First Power (GFP) is seeking CNSC approval for a license to prepare the site for a micro modular reactor (MMR) at the Chalk River Laboratories (CRL) 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 (NSCA) 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 small 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 a high-temperature gas-cooled reactor to provide approximately 15 megawatts (thermal) of process heat to an adjacent plant via molten salt, which will generate electrical power and/or heat 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 in Chalk River.

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

Advanced reactor vendors with designs that propose of the use of TRISO fuel have engaged with CNSC and the US-NRC. At the US-NRC vendors are submitting plans for fuel qualification through topical reports. Additionally, guidance in the area of fuel qualification is available in the Nuclear Energy Agency (NEA), "Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors,"(RPFQ) [5] and NUREG-2246, "Fuel Qualification for Advanced Reactors," [6]. NUREG-2246 was subject to a public comment period where industry stakeholders, through the Nuclear Energy Institute (NEI), emphasized the importance of having advanced reactor specific examples (including TRISO fuel) to address fuel qualification review criteria.

2. Statement of Scope and Objectives for the Cooperative Activities

CNSC and US-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 which would need to be addressed to enable TRISO use in advanced reactor licensing applications. This project aims to:

1. 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
2. identify areas of TRISO fuel qualification that are design dependent, and
3. highlight areas where additional information and/or testing is still needed to support regulatory approval.

3. Regulatory Basis

3.1. Regulatory Basis for Fuel Qualification at US-NRC

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

- The regulation in 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 to 10 CFR Part 50, "General Design Criteria [GDC] for Nuclear Power Plants," establish 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," [7] provides guidance on how the GDC in Appendix A to 10 CFR Part 50 may be adapted for non-LWR designs and contains advanced reactor design criteria (ARDC). While the GDC and ARDC are not requirements for non-LWR designs, the GDC and ARDC 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, NRC staff expects that information be provided that address the design aspects described in the following GDC and ARDC as part of fuel qualification:

    - GDC 2 and ARDC 2, "Design bases for protection against natural phenomena," 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 earthquake

[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.

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[2].

- 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). Reactor designs that use TRISO fuel are generally expected to use SARRDLs.

- MHTGR-DC 16, "Containment Design," requires a functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, be provided 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.

3.2. Regulatory Basis for Fuel Qualification at CSNC

REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants," provides the following criteria for water cooled reactor facilities [8]:

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[2] These "fundamental safety functions" are identified in the IAEA safety glossary [20] 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).

- 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 (DBAs), the fuel assembly and its component parts shall remain in position with no distortion that would prevent effective post-accident 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 in DBAs. 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.

REGDOC 2.4.5, "Nuclear Fuel Safety" [9] is being developed by the CNSC to clarify the requirements and provides guidance for the design, operation, monitoring and safety assessments of fuel for operating reactor facilities. This document focuses on fuel design, operation, monitoring and safety assessments for operating reactors, with implicit concentration on operating CANDU reactors, but 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 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. While this document focuses on CANDU fuel, high-level concepts within it may apply to other technologies. 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, "Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors," [5] and NUREG-2246 [6] . It incorporates positions documented in the NRC safety evaluation for EPRI-AR-1 [1], but also highlights areas that were not addressed by the topical report. Some of the considerations addressed by this report, that were not specifically addressed by the NRC staff review of EPRI-AR-1 include fuel performance evaluation model requirement, potential testing needs to address potential accident conditions, and accident source term considerations.

4.1. G1 – Fuel Manufacturing Specification

Key manufacturing parameters are provided in the following sections and are generally obtained from information provided in EPRI-AR-1 [1]. These key manufacturing parameters correspond to the fuel performance testing from AGR-1 and AGR-2. Additionally, manufacturing parameters are provided for (1) the TRISO particle, and (2) the fuel compact/pebble to the extent possible based on available data.

4.1.1. **G1.1-Dimensions**

4.1.1.1 TRISO Particle

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

Table 1 TRISO Particle

Particle Dimension	95% Confidence Interval Extrema	95%/98% Tolerance Limit Extrema	Basis
Buffer thickness (μm)	96.5-105.2	75.2-124.7	EPRI-AR-1(NP)-A, Table 5-5 [1]
IPyC thickness (μm)	38.6-41.1	32.4-47.6	EPRI-AR-1(NP)-A, Table 5-5 [1]
SiC thickness (μm)	34.8-36.2	30.6-41.2	EPRI-AR-1(NP)-A, Table 5-5 [1]
OPyC thickness (μm)	39.1-44.3	33.6-51.6	EPRI-AR-1(NP)-A, Table 5-5 [1]
OPyC Aspect Ratio	1.057[a]	1.102[b]	EPRI-AR-1(NP)-A, Table 5-5 [1]
SiC Aspect Ratio	1.040[a]	1.068[b]	EPRI-AR-1(NP)-A, Table 5-5 [1]

[a] Upper bound of 95% confidence interval
[b] 99% coverage tolerance interval

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 [1] 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 accepted by NRC staff in Section 3.5 of the 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 SiC layer
t_{SiC}	=	Thickness of SiC layer

Based on the information provided in Section 5.3.6 of EPRI-AR-1(NP)-A [1], the AGR-1 and AGR-2 test data cover a 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.

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[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) 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 % ^{235}U	EPRI-AR-1(NP)-A, Section 5.3.1[6,7] [1]
UC_x Molar Fraction [a,b,c]	30% +/- 5%	EPRI-AR-1(NP)-A, Table 5-2 [1] (see discussion below)
Individual Impurities	≤ 100 ppm-wt% for each impurity of Li, Na, Ca, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, Al, and Cl	AGR-1 Fuel Specification [10]
Process Impurities	≤ 1500 ppm-wt% for each impurity of P, S	AGR-1 Fuel Specification [10]

[a] This represents a mean value for the population. Consistent with AGR-2, critical limits are not specified (see Table 4 from EPRI letter dated February 26, 2020 [11]).
[b] Calculated molar fraction with the remaining material being 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:

1. Too little UC_x in the fuel kernel can increase the production of 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) non-retentive SiC failure (see Section 4.3.1.3 for a description of degradation mechanisms and failure modes).
2. Too much UC_x can result in insufficient oxygen in the kerel to oxidize rare-earth fission products, leading to fission product attack of the SiC layer. Fuel containing greater than 75% 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 [12].

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. The range for UC_x provided in Table 2 of this report addresses Limitation 2 and extends slightly beyond the mean values provided in Table 5-2 of EPRI-AR-1(NP)-A [1] and the values provided in Table 4 from

[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% ^{235}U.

[7] Section 5.3.1.2 of EPRI-AR-1(NP)-A clarifies that AGR-2 kernels had a nominal enrichment of 14% ^{235}U.

EPRI letter dated February 26, 2020 [11]. Kernel composition outside these limits would require additional justification.

Research on the fuel kernel composition concluded that [12]

> *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.*[8]

EPRI-AR-1(NP)-A referenced a thermochemical study that used unvalidated analyses to suggest that UC_x content as low as 5.5% may be sufficient to achieve burnups up to 16% FIMA in UCO TRISO [13]. 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 provide in Table 2 of this report is based upon the information available from AGR-1 and AGR-2. Experimental evidence would be expected to support values for UC_x molar fractions beyond that provided in Table 2.

The buffer, IPyC, and 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.

[8] This paper was published April 15, 1977.

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]
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 [10]

[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 SiC 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. [9] 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." However, the actual implementation of the visual standard is not described in EPRI-AR-1-A [1] or the AGR-1 fuel specification [10]. Accordingly, CNSC and NRC and pursuing additional information regarding the implementation of this visual standard.

[9] This is provided as Figure 5-2 from EPR-AR-1(NP)-A [1] and Figure 1a and 1b from the AGR-1 fuel specification [10].

Studies on SiC microstructure using AGR-1 and AGR-2 data did not establish correlation between SiC microstructure and TRISO particle performance [14]. However, all the samples used in the study were manufactured in accordance with specifications that used a visual standard to place an upper bound on grain size. Accordingly, fuel designs that rely on AGR-1 and AGR-2 data should be manufactured using a suitable reference standard via materials that are traceable to AGR-1 and AGR-2.

4.1.3.1.2 Manufacturing Process

In a letter dated February 26, 2021 EPRI stated the following regarding EPRI-AR-1(NP):

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

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

4.1.3.2 Fuel Compact/Pebble

Section 5.3.4 of EPRI-AR-1(NP)-A described 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% [1].[10] Based on the packing fraction used for the UCO TRISO compacts used 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%. Packing fractions above 40% 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 addressed on a design dependent basis.

4.2. G2-Safety Criteria

4.2.1. G2.1-Design Limits during Normal Operation and Anticipated Operational Occurrences

4.2.1.1 G2.1.1 Definition of Fuel Performance Envelope

(to be completed)

[10] The maximum packing fraction for random close packed spheres is approximately 64%.

4.2.2. **G2.2-Radionuclide Release Limits**

(to be completed)

4.2.3. **G2.3-Safe State**

4.2.3.1 G2.3.1-Maintaing Coolable Geometry

4.2.3.1.1 TRISO Particle

TRISO fuel is generally expected to function as part of a functional containment. Accordingly, TRISO particles are expected to maintain their integrity under accident conditions.[11] 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 is 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. Accordingly, the fuel compact/pebble needs to maintain its structural integrity to ensure a coolable geometry. Graphite is a common host matrix material for TRISO particles and graphite is known to increase in strength at temperatures well above the SiC thermal decomposition temperature. Accordingly, maintaining the TRISO particle temperature below the SiC thermal decomposition limit should also ensure coolable geometry of the fuel compact/pebble. The fuel compact/pebble may vary among the different reactor vendors. Accordingly, 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.

This goal is associated with fuel qualification because fuel assemblies and/or fuel structures may form part of the negative reactivity insertion path. Reactor designs that use TRISO fuel may not have fuel assemblies and/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). Accordingly, this goal may not be applicable to fuel qualification for some reactor designs.

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[11] This is in contrast to traditional light water reactor fuel where the cladding is not credited to retain fission products under some design basis accidents (e.g., loss of coolant accident) [21].

4.2.3.2.2 G.2.3.2(b)-Evaluation Model

The 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 [5] and Section 3.3 of NUREG-2246 [15] describe evaluation models generically such that evaluation models may be sophisticated analytical tools like computer codes, simplified mathematical expressions, or comparisons against data. CNSC and NRC do not have sufficient information to assess any specific computation code. The information provided in this section addresses the needs of an evaluation model adequately evaluation UCO TRISO fuel.

4.3.1. EM G1-Evauation Model Capabilities

4.3.1.1 EM G1.1 – Geometry Modeling

(to be completed)

4.3.1.2 EM G1.2 – Material Modeling

(to be completed)

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 [15] and NEA-RPFQ [5] 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], [16], [17], [18]. Some of the degradation mechanisms and failure modes encountered in past experience or identified by expert panels have been addressed by the development of UCO-TRISO fuel or have not been observed in testing. Accordingly, NRC and CNSC expect that evaluation models for fuel performance be cable of addressing some of the failure modes and degradation mechanisms, but 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 (e.g., 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 [16], [19].
- **Pressure vessel failure of particles with defective or missing coatings** – Pressure vessel failure due to manufacturing defect. Some amount of defective particles are expected, due, in part, to the large number of TRISO particles present in the reactor [19].

- **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 [16].
- **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 1600 ℃ [16].
- **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.
- **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.
- **Fission product attack failure** – Degradation of the SiC layer can occur due to interaction with fission products, specifically palladium.
- **Non-retentive 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 [16], [17].
- **Creep failure of PyC** - Thinned and failed PyC has been observed in some post irradiation heating tests. These results were determined for test with temperatures greater than 2000 ℃ for long durations. The observed failures did not lead to failure of the SiC layer [16].
- **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 [16], [18].

4.4. Experimental Data - Advanced Gas Reactor

(to be completed)

4.4.1. ED G1-Indpendence of Validation Data

(to be completed)

4.4.2. ED G2-Test Envelope

(to be completed)

4.4.3. ED G3-Data Measurement

(to be completed)

4.4.4. **ED G4-Test Conditions**

(to be completed)

5. Conclusions

(to be completed)

6. References

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