US-NRC – CNSC Memorandum of Cooperation

INTERIM SUMMARY JOINT REPORT
concerning
Tristructural Isotropic (TRISO) Fuel Qualification

February, 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.

1. Introduction

This summary interim report provides a status update on the CNSC/NRC Joint tristructural isotropic (TRISO) fuel assessment project [1]. The final report will assess TRISO fuel (shown in Figure 1-1) against the goals provided in the fuel qualification framework from the Nuclear Energy Agency (NEA) report, "Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors," (RPFQ) [2] and NUREG-2246, "Fuel Qualification for Advanced Reactors," [3]. These goals are reproduced in Tables 1 through 3 below with boxes outlined in red indicating that the goals are partially addressed by the contents of this interim report. This interim report specifically addresses (1) reactor technologies considered in the study, (2) the regulatory basis for advanced nuclear reactor fuel qualification in Canada and the United States[1], (3) known degradation mechanism and failure modes for TRISO fuel, and (4) transient behavior of TRISO fuel. This report was supported by technical input provided by Pacific Northwest National Laboratory (PNNL).

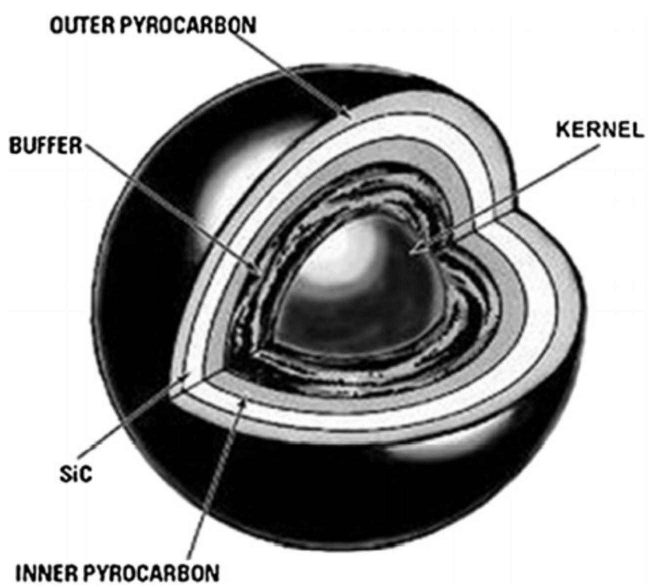

Figure 1-1 TRISO fuel particle [4]

[1] NEA-RPFQ and NUREG-2246 incorporate the regulatory basis into the fuel qualification framework. So explicit goals for the regulatory basis are not provided in Tables 1 through 3.

Table 1 List of Goals in Fuel Qualification Assessment Framework

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

Table 2 List of Goals in Evaluation Model Assessment Framework

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

Table 3 List of Goals in Experimental Data Assessment Framework

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

2. Summary

2.1. Regulatory Technologies Considered in the Study

This project considers reactor technologies that use TRISO fuel with the following design parameters:

Coolant	Structural Form	Time Averaged Volume-Averaged Temperature*	Peak Burnup*	Peak Fast Fluence*
Helium Air or dry-nitrogen FLiBe (molten salt)	Pebble Bed Prismatic	955 – 1296 ℃	19.6 percent fissions per initial metal atom (FIMA)	4.3 x 10^{25}n/m (E > 0.18 MeV)

*This range spans the Advanced Gas Reactor (AGR)-1 and AGR-2 data

The parameters identified above partially establish the performance envelope and provides input to addressing G2.1.1, “Fuel performance envelope is defined,” from NUREG-2246 [3] and NEA-RPFQ [2]. In addition to these parameters, other values are being considered (e.g., power density). Additionally, representative conditions to address fuel performance under transient and accident conditions (e.g., undercooling and overpower events) are currently being investigated. One area of emphasis for transient and accident consideration is the rate and limit for overpower transients. Specifically, rapid reactivity insertions of sufficient magnitude to make the reactor prompt-critical may not be credible for reactor designs using TRISO fuel. Specification of a generic performance envelope for TRISO fuel is an ongoing task that will be a focus of effort over the next quarter.

Additionally, the performance envelope discussed above is specific to UCO-TRISO fuel. Fuel manufacturing requirements are addressed in the fuel qualification framework by G1, “A fuel manufacturing specification controls the key fabrication parameters that significantly affect fuel performance,” from NUREG-2246 and NEA-RPFQ. The identification of key fabrication parameters will be addressed in a future report. These key parameters will consider, in part, kernel composition (e.g., carbon-to-oxygen ratio), kernel size and porosity, fuel compact packing fraction, and the parameters identified in Table 5-5 of the EPRI-AR-1 [5]. TRISO particle overcoating has been identified as an important parameter as is prevents direct particle-to-particle contact [6]. Overcoating thickness (generally ~200 μm) can be affected at higher fractions and may limit the applicability of the existing data.

2.2. Regulatory Basis for Advanced Reactor Fuel Qualification

2.2.1. Regulatory Basis for Fuel Qualification at US NRC

The relevant regulatory requirements associated with fuel qualification are identified in NUREG-2246 [3]. The relevant requirements for TRISO fuel qualification are as follows:

- The regulation in 10 CFR 50.43(e)(1)(i) requires demonstration of the performance of each safety feature[2] 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,"

[2] 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.

implements GDC 2 as it pertains to seismic events and defines specific earthquake criteria for nuclear power plants. Appendix S to 10 CFR 50 establishes 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.

- 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 specify a SARRDL.

- 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.

2.2.2. **Regulatory Basis for Fuel Qualification at CNSC**

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

- 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 licenced, 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.

2.3. Degradation Mechanisms and Failure Modes for TRISO Fuel

Addressing EM G1.3, “Physics Modeling” in NUREG-2246 [3] and NEA-RPFQ [2] 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 [6], [10], [5] [11]. 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 [5], [6].
- **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 [5].
- **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 [6].
- **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 ℃ [6].
- **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 phenomena may be a bigger factor at higher burnup values [6], [10].
- **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 [6].
- **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 [6], [11].

Additionally, differences in TRISO fuel forms that are associated with the fuel compact will be addressed separately. Specifically, NUREG-2246 and NEA-RPFQ include assessment frameworks to address applicability of experimental data. ED G4, "Test Conditions" of NUREG-2246 and ED G4, "Test Specimens" of NEA-RPFQ require that tests supporting fuel qualification be representative of the prototype.

2.4. Transient Behavior of TRISO Fuel

As discussed in Section 2.1 of this report, the specification of a generic performance envelope for UCO-TRISO fuel is an ongoing task. This includes the identification of representative transient conditions under which the fuel is expected to perform. Rapid reactivity insertions of sufficient magnitude to make the reactor prompt-critical may not be considered credible for reactor designs using UCO-TRISO fuel. However, previous TRISO fuel designs have been exposed to conditions resulting from large and rapid reactivity insertions [12]. Fuel tests conducted at the Nuclear Safety Research Reactor (NSRR) in Japan introduced reactivity insertions ranging from $2.16 to $4.14 and showed that TRISO particles with fresh UO_2 kernels failed above energy depositions of 1.40 kJ/g-UO_2, where the peak fuel temperature exceeded the melting point of the UO_2 kernel [13]. UCO has a lower melting temperature than UO_2, so a lower failure threshold for UCO-TRISO may be expected than what was observed for UO_2-TRISO. NRC and CNSC are questioning (1) the applicability of these transient tests to current UCO-TRISO, and (2) the need for such transient testing to support UCO-TRISO fuel qualification. Assessing the transient behavior of UCO-TRISO is the focus of near term efforts.

3. References

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[10] 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.

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