X energy

Xe-100 SMR & XENITH Microreactor: Codes & Standards Insights

April 4, 2024

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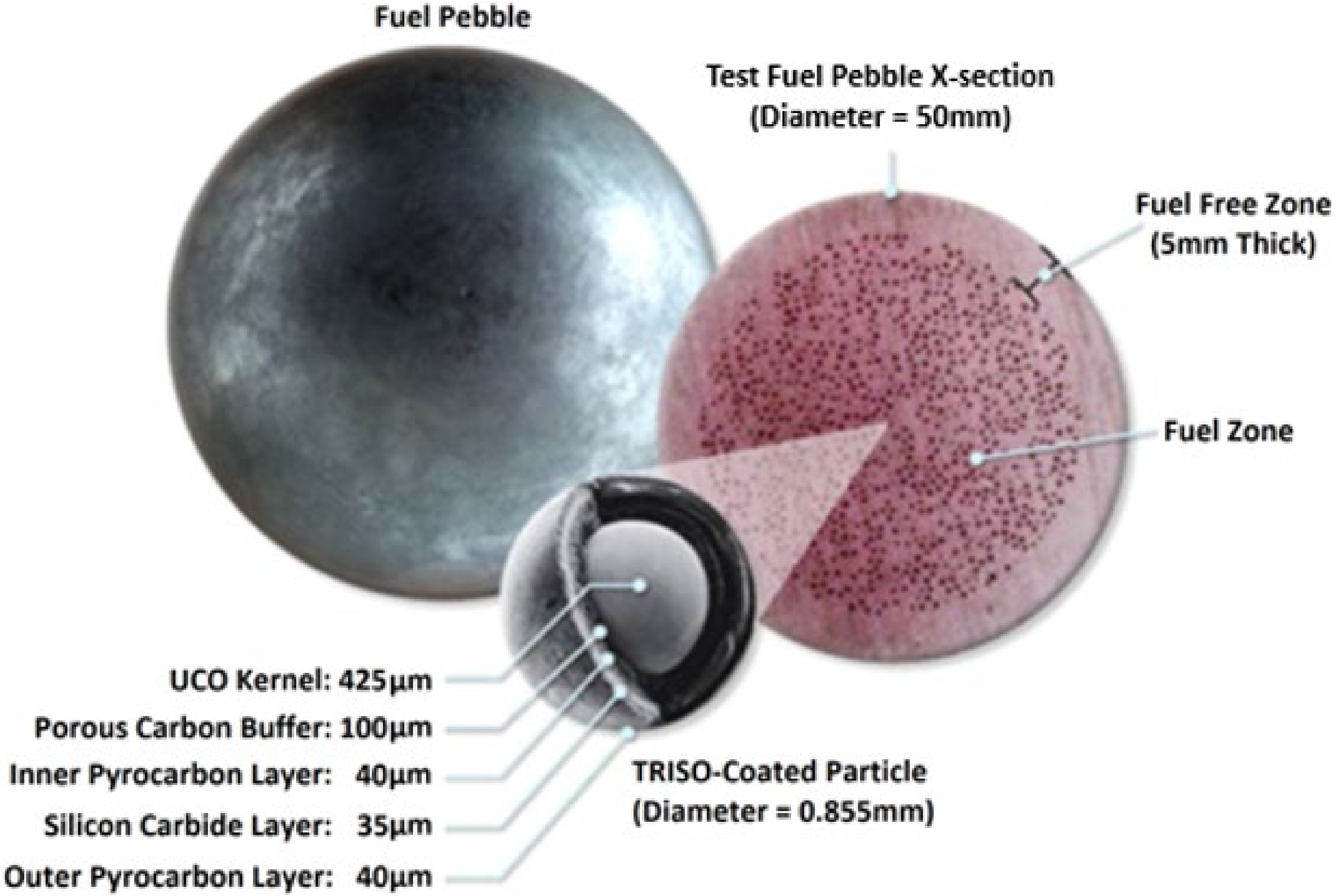
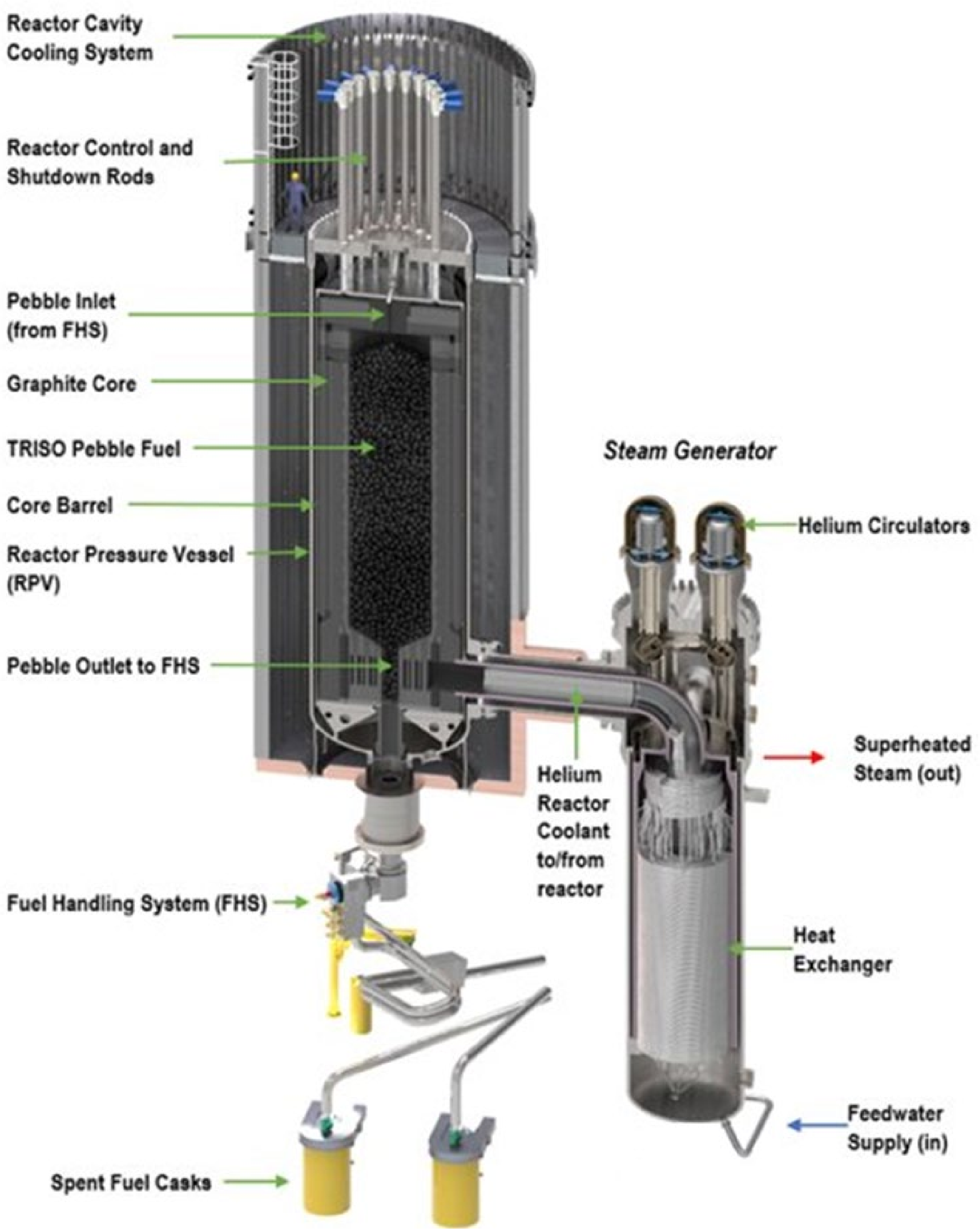
Xe-100 SMR: Overview

- 200MWt/80MWe helium-cooled High Temperature Gas Reactor (HTGR) with TRISO particle fuel pebbles

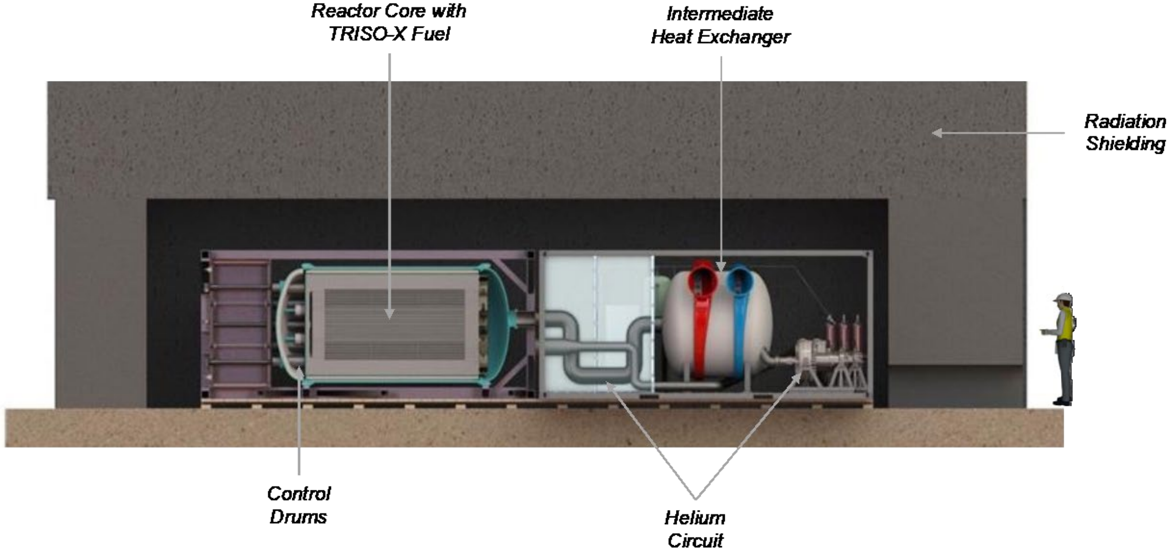


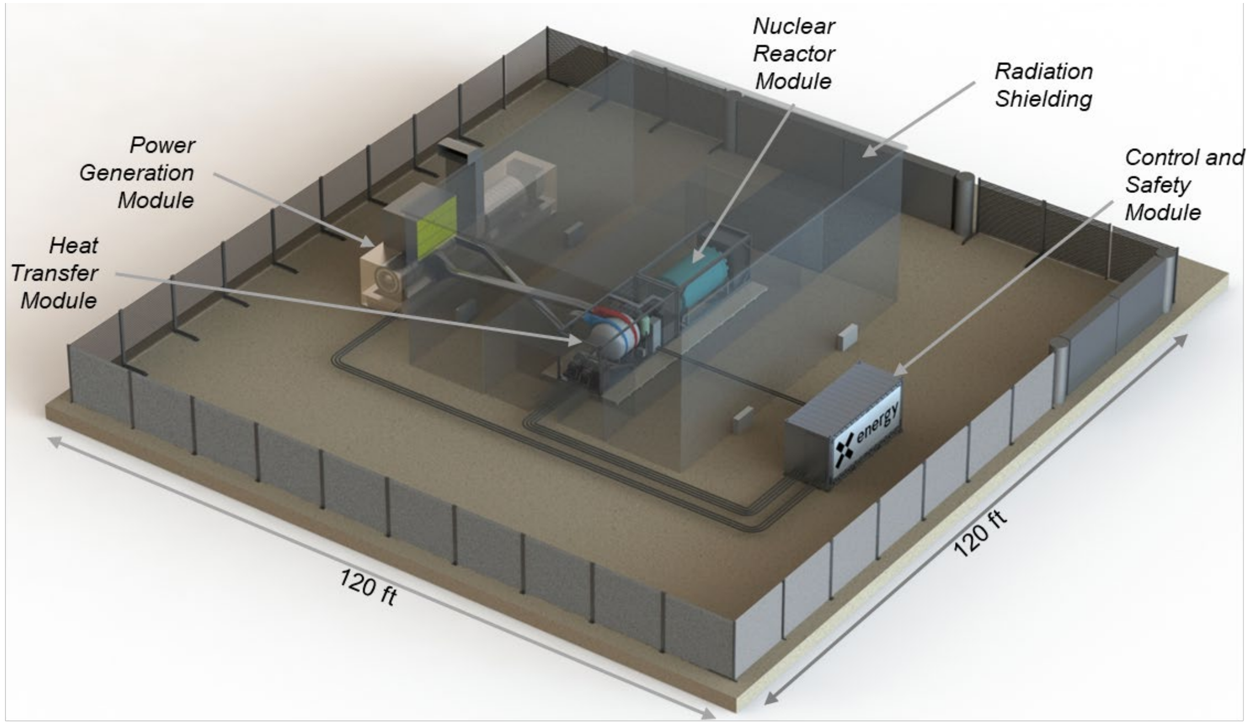


X-Energy Next-generation Integrated Transportable High-temperature (XENITH) Microreactor Plant – Overview

Reactor

- Next generation HTGR with novel features that enable high performance in a small, factory assembled, and factory fueled road transportable package
- Proven and tested UCO TRISO fuel particles (19.75% LEU)
- High temperature tolerant graphite core structure
- ASME compliant core barrel and helium pressure boundary





Attributes:

✓ Targeting 5 MW net electricity output with optional combined heat and power cycles from high-temperature clean air exhaust

✓ Modular design supports rapid deployment, operational flexibility, maintenance and inspection

✓ Inherent safety with no operator actions required

✓ Rapid power ramp rates and step load changes for stand-alone operation or integration with renewables in a microgrid

✓ Configurable radiation shielding enables the siting of public structures close to the site with no radioactive materials left onsite

Codes & Standards Insights – Summary

- Reactor Core Components: navigating emerging guidance for High Temperature Metallic & Nonmetallic Materials
  – The Code is new and has unintended constraints on the users
- Quality Assurance: Cost implications
  – "Over-application" of the most stringent QA standards may not actually result in improved reliability and yet has major cost implications
- Civil/Structural Codes & Standards: Quantifying the performance basis
- Risk-Informed Performance-Based Design
  – X-energy intends to leverage risk-informed performance-based methods in NEI 18-04/RG 1.233 to add special treatments to commercial C&S to right-size design requirements to meet required safety functions (RSFs) and non-safety related with special treatment (NSRST) probabilistic risk assessment safety functions (PSFs) and associated reliability and capability targets
  – Example: Start with ASCE 7 and add design requirements to meet design loads needed to meet RSFs and NSRST PSFs
  – Example: Use a suppliers ISO 9001 program and add requirements to meet a graded-approach to QA as one of several special treatments to meet the reliability and capability targets for the particular system

Reactor Core Components

- ASME Boiler & Pressure Vessel Code (BPVC), Section III, Division 5 “High Temperature Reactors” is very new in the time scale of C&S development
  - 2017 Edition endorsed by NRC staff with limitations in RG 1.87 R2 in January 2023
  - ASME Code Committees are now working to address the limitations by 2025 Edition
  - Time scale of endorsement of next Edition uncertain
- As the new Code is exercised by designers, “real-world” gaps and limitations (some unintentional) become increasingly apparent and are being discussed in the various ASME Code Committees
- Example: barriers to qualification of graphite (see next slide)
- Example: limited data for high temperature metallic components for long life power reactor operation (e.g. 60 years)
- Example: lack of guidance in Sec. III, Div. 5 on dissimilar metal welding for high temperature applications

Graphite Qualification Limitations

- Requirements for Generation of Design Data for Graphite Grades (Mandatory Appendix HHA-III) are prohibitively over-constraining
  - 288 grade-specific specimens are required, for each material property for each grain direction
  - Substantial amount of irradiated and oxidized data is expected
  - Material properties available in the Baseline Program (Unirradiated Graphite Testing) from INL are not sufficient to meet the requirements
  - Hard to obtain all data points across the wide range of operating envelope temperatures and irradiation fluences
- Data is grade-specific – this is a barrier to entry for any new graphite suppliers which don't already have the substantial volume of irradiated data expected
  - Hundreds of irradiation capsules would be needed to complete a full irradiation campaign, 10-15 years (a non-starter)
  - Very limited availability in test reactors for irradiating more specimens anyway
  - Industry needs success pathways to support scaling up graphite supply significantly
- Data available comes from a wide range of sources, including historical sources for which QA programs are difficult to assess, and from national lab testing which is difficult to qualify to NQA-1
- Additionally, there are issues with the steps for calculating the Probability Of Failure (POF) (HHA-3217). Ongoing ASME code activities involving reactor designers are trying to resolve to these issues.

Quality Assurance

- There is a need for risk-informing QA standard requirements so they can be specified commensurate with the inherently different safety basis for advanced reactors (vs LWRs)
  - "Over-application" of the most stringent QA standards may not actually result in improved reliability from a performance-based perspective and yet have major cost implications for advanced reactors
  - The issue is at the nexus between QA codes (NQA-1 and ISO 9001) and other codes (ASME, ASCE, IEEE)
- NEI 18-04 allows for a risk-informed performance-based approach to QA, but designers are on their own in deciding how to achieve this
- ASME NQA-1 is logical for ensuring quality of some components (e.g. those which tie back to its roots in the nuclear Navy submarine program), but has been extrapolated and applied across other technical disciplines where it does not logically contribute additional reliability, but adds cost
  - Electrical engineering community recognized this and developed a body of their own C&S for QA of electrical/I&C equipment (IEEE Stds. 279, 344, 603, 2012, 1028, etc.)
  - Civil/structural engineering community ensures a high degree of reliability & quality in commercial and industrial building construction by other means (substantial "operating experience" in modern non-nuclear buildings surviving hazards like earthquakes)
  - Graphite cracking is probabilistic and irradiated data availability is limited – not all historic data can be readily qualified to ASME NQA-1
- Guidance for using a graded approach to quality commensurate with contribution to safety or risk is needed (e.g. ISO-9001 to meet Appendix B requirements)

Civil/Structural Codes & Standards

- Civil/structural C&S must be prioritized as these structures are a major cost driver for small reactors
  – There is an urgent need to develop cost-effective nuclear quality C&S for civil/structural components
  – This is a cross-cutting issue across all advanced reactor technologies
  – Design Loads: ASCE 7 (commercial code) versus ASCE 43 (nuclear code)
  – Steel specs: AISC 360 (commercial code) versus AISC N690 (nuclear code)
  – Concrete specs: ACI 318 (commercial code) versus ACI 349 (nuclear code)
  – Materials standards such as ACI 318 and AISC SCM/341 could be usefully deployed for new build nuclear. These standards already are used for mission-critical infrastructure, with imperative that they can be safe for the public. While ACI 349 and AISC N690 are derivatives developed on what was important for LWRs – the safety basis for advanced reactors is different.
- Quantifying the performance basis for existing C&Ss is needed for application of NEI 18-04
  – ASCE 4, 43, and 41 have good, clear performance bases.
  – ASCE 7 has some (e.g., performance goals) but also legacy provisions that obscure performance implications.
  – ACI, ASME codes are relatively opaque when it comes to performance basis
  – Pivoting to the USGS tools for seismic probabilistic hazard assessment, vs dated/costly SSHAC process
  – Could merge ASCE 4 and ASCE 43, eliminate legacy provisions, and point to the "commercial standards" ASCE 7, ASCE 41, ACI 318, and AISC SCM/341 - could be accomplished in the short term

Summary

- Timeline is key. Identify what needs to / can be done in 2yrs, not 10yrs.
  - This is recognized across industry, e.g. Advanced Reactor Codes and Standards Committee (ARCSC), ASME Conference for Advanced Reactor Deployment (CARD)
- A limiting factor is that consensus code committees are supported by volunteer time.
- More & dedicated resources are needed to move faster on these activities.

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