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TRISO Fuel Part II: Accident Performance and NRC Engagement

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Changing the World's Energy Future

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Idaho National Laboratory

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TRISO Fuel II:

Accident Performance and NRC Engagement

CNSC Seminar

Idaho National Laboratory

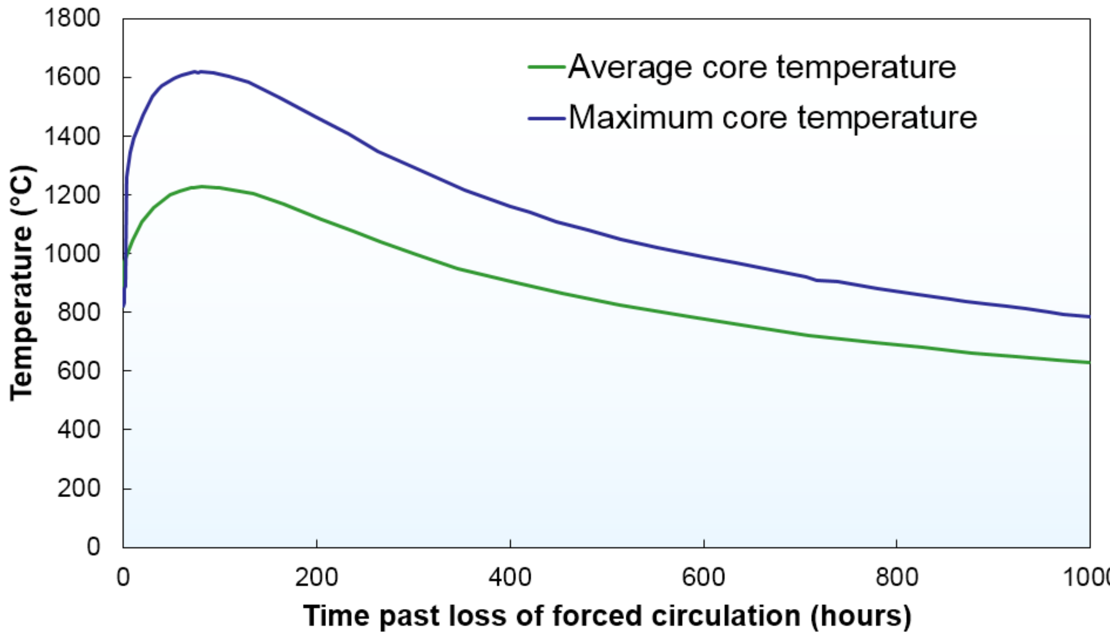
Outline

Part II

- Fuel Accident Performance
- US NRC Engagement on Fuel Qualification

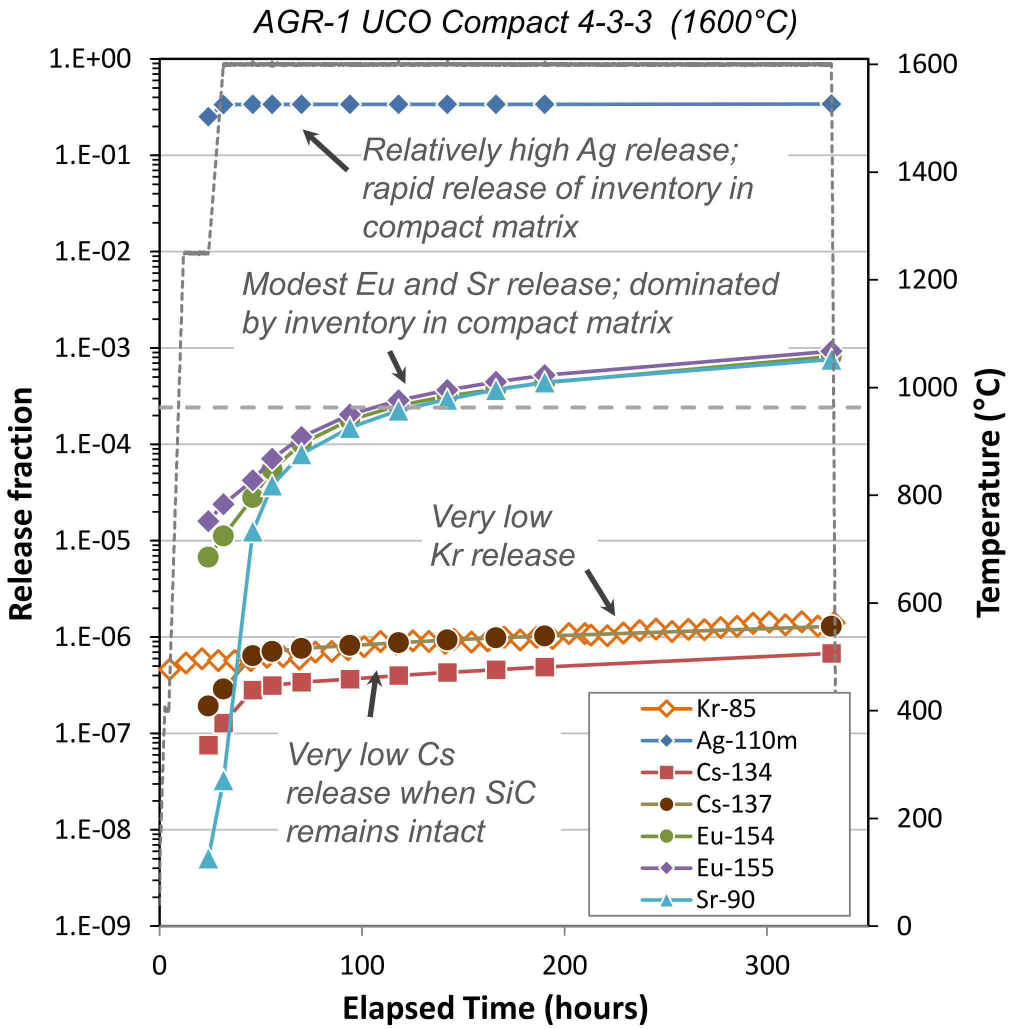
HTGR Accident Safety Testing of TRISO Fuel

- Temperature transients are relatively slow (days)
- Peak fuel temperatures are limited to ~1600°C in modular HTGR designs
- Fuel particles are designed to withstand accident conditions while still retaining key safety-significant fission products
- Total duration at peak temperatures is tens of hours, with a small fraction of the fuel in the core experiencing temperatures near the peak
- Extremely rapid activity insertion accidents (RIAs) are precluded by HTGR core design
- Assess fuel performance by post-irradiation heating tests while measuring fission product release at 1600 – 1800°C.



AGR-1 and AGR-2 Safety Test Performance

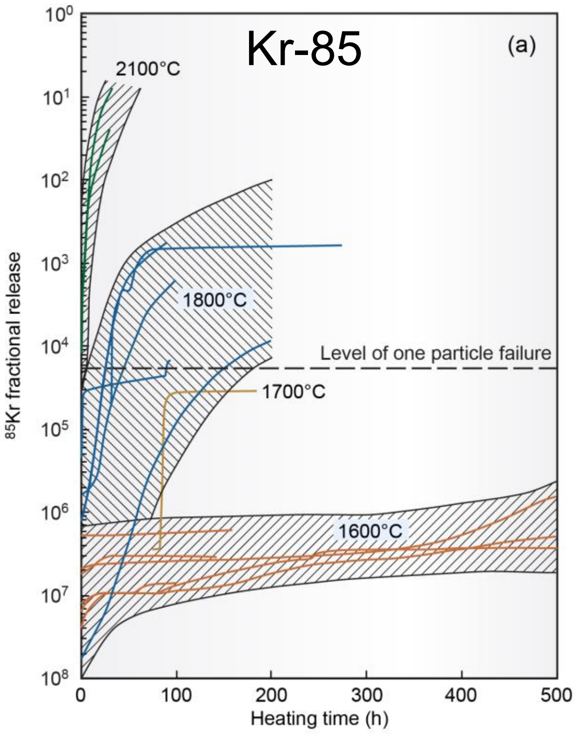
- **Low Cs release** (dependent on intact SiC)
- **Low Kr release**
- **Modest Sr and Eu release** (influenced by irradiation temperature)
- **High Ag release** (dominated by in-pile release from particles)
- **Excellent UCO performance up to 1800°C**
- **Low coating failure fractions** (UCO)
- **UO_2 demonstrates much higher incidence of SiC failure due to CO attack**



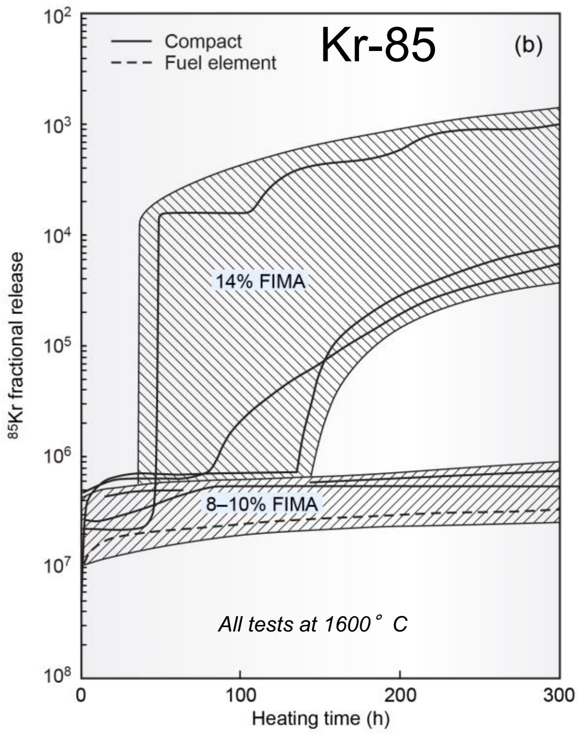
Considerations for Interpreting Isothermal Post-Irradiation Heating Tests

- Meant to help understand fundamental fuel and FP behavior, *not simulate a depressurized loss of forced cooling accident*
- Durations at peak temperature usually greatly exceed those in a real DLOFC accident
- A very limited fraction of the fuel in the reactor core reaches peak temperature during an accident
- Higher temperature tests are performed as margin tests to accelerate the rate of thermally driven processes
- Tests measure fission product release from the compact/pebble and do not include core graphite as a fission product sink

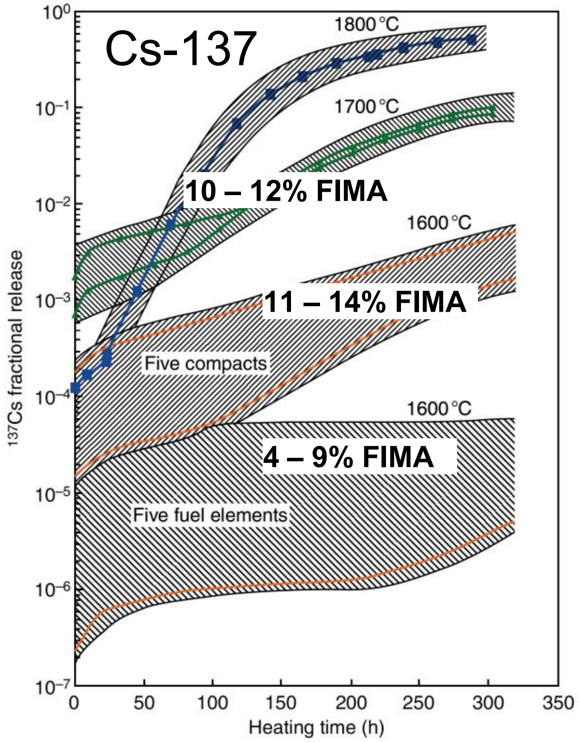
Kr and Cs Release: German UO_2 Results



- No TRISO failures at 1600°C
- TRISO failures occur after short periods at 1800°C



- No TRISO failures at 1600°C with burnup ≤10%
- TRISO failures occur at 1600°C with burnups ~14%



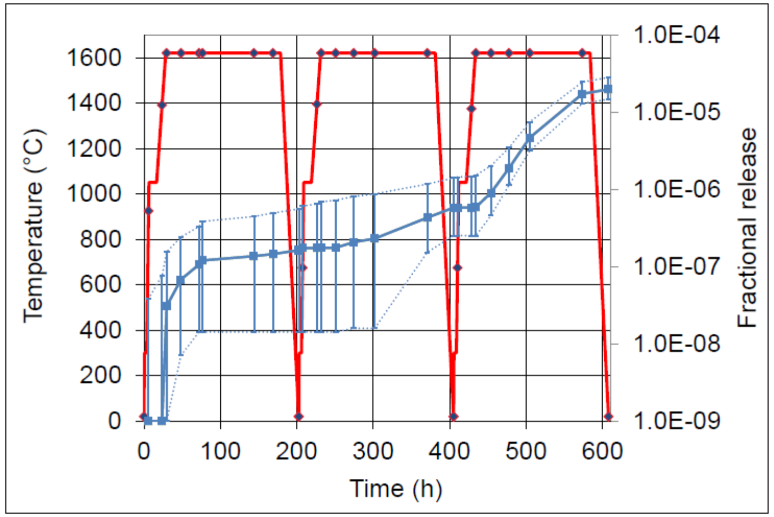
- At 1600°C and burnup <10% FIMA, Cs release remains relatively low
- Increasing burnup and temperature increases SiC layer degradation and Cs release

P.A. Demkowiczet al., TRISO-Coated Particle Fuel Fabrication and Performance, in Konings R.J.M.,(ed.) Comprehensive Nuclear Materials (2012), vol. 3, pp. 151-213 Amsterdam: Elsevier.

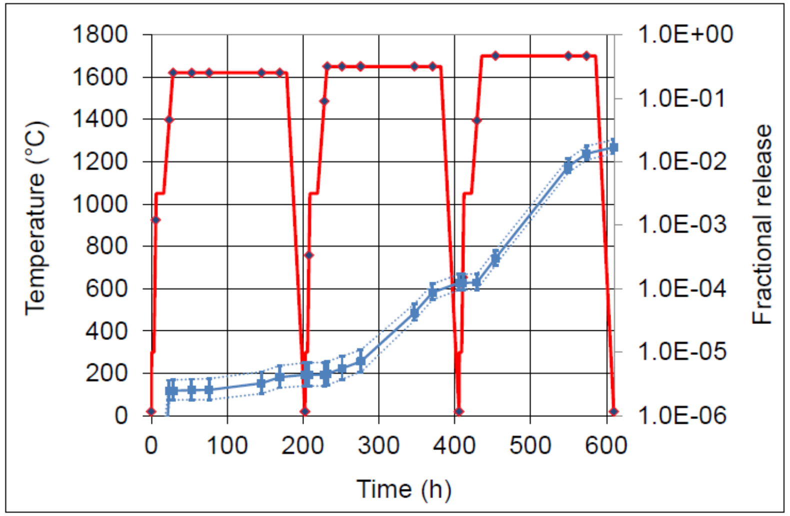
Safety Test Results for HTR-PM UO_2 Fuel (China)

^{137}Cs

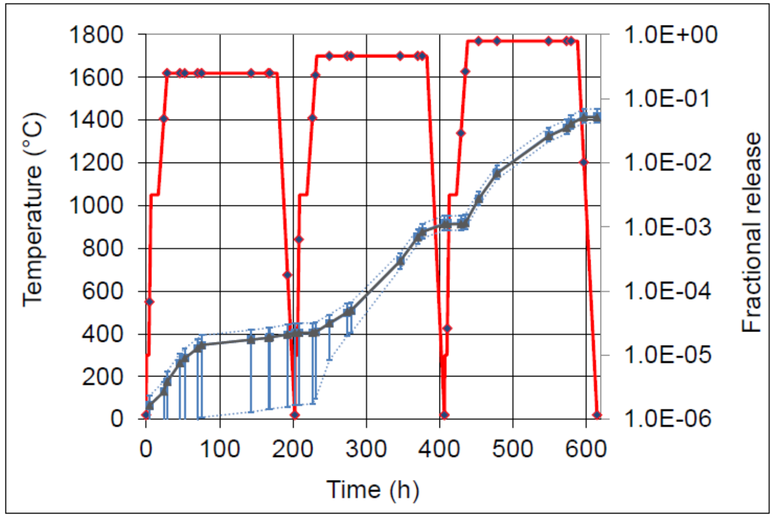
HTR-PM1 (11.6% FIMA; 1023°C)



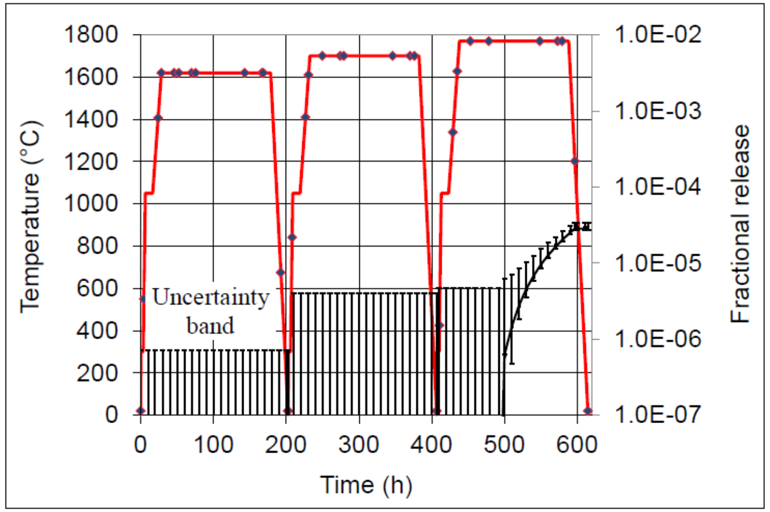
HTR-PM4 (11.9% FIMA; 1017°C)



HTR-PM2 (12.5% FIMA; 1040°C)



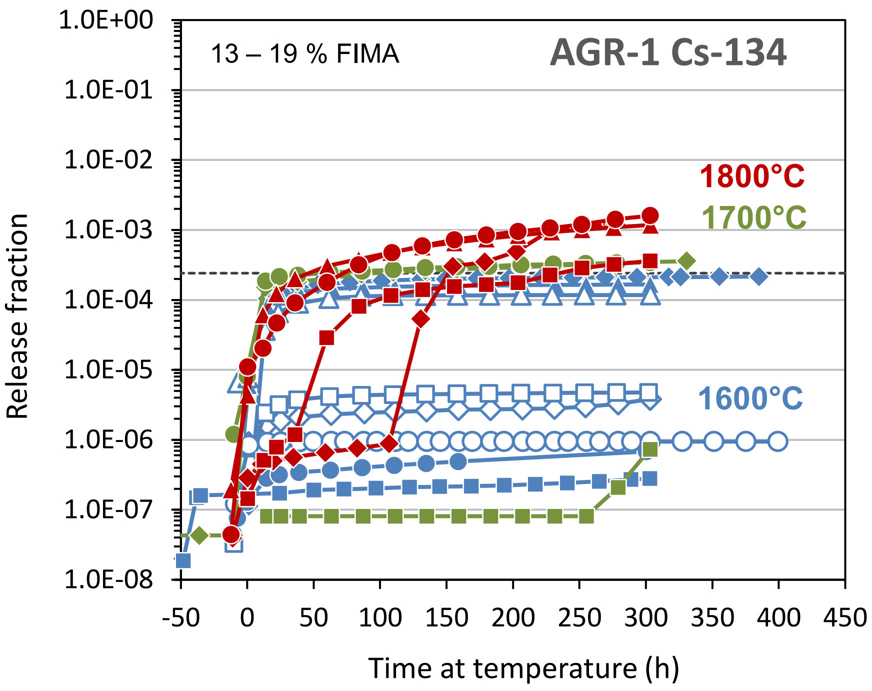
^{85}Kr

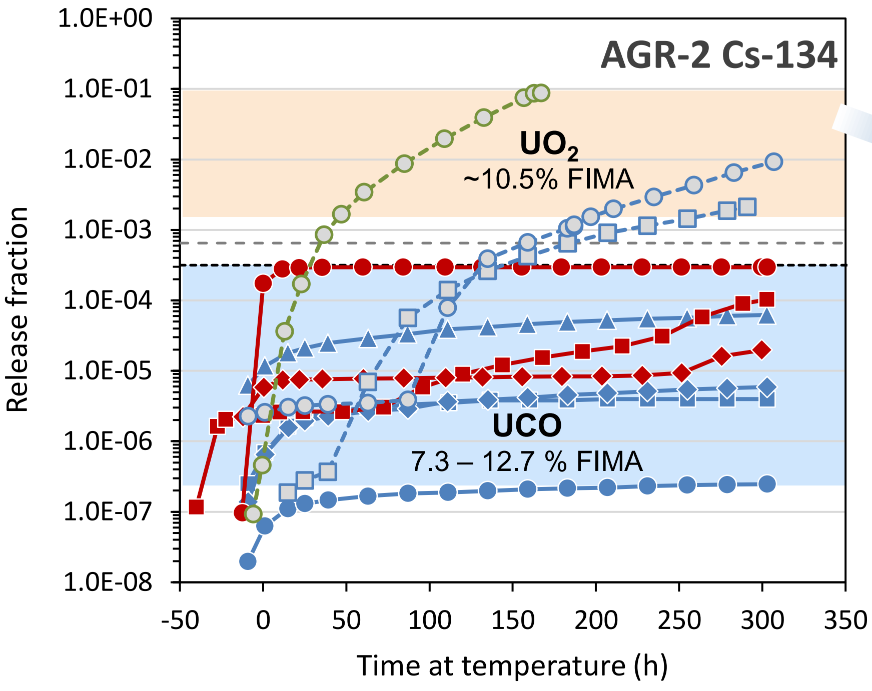


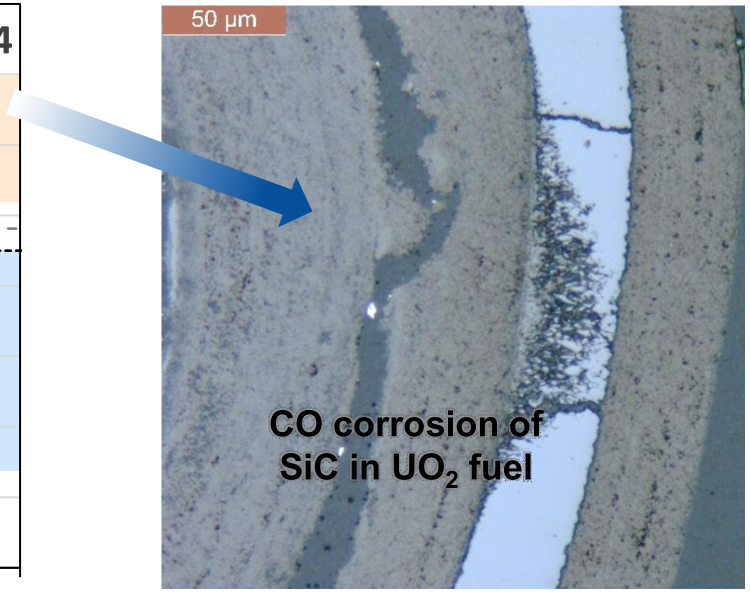
- Spheres fabricated at INET, irradiated in HFR-Petten, and heated in KüFA facility (Karlsruhe, Germany)
- Three 150 h segments at temperatures between 1620 and 1770°C
- No TRISO particle failures
- Increased Cs release as test duration and temperature increased

D. Freis et al., "Burn-up Determination and Accident Testing of HTR-PM Fuel Elements Irradiated in the HFR Petten," Paper HTR 2018-0073, Proceedings of HTR 2018, Warsaw, Poland, October 8-10, 2018

Cesium Release Results: AGR Program Safety Testing

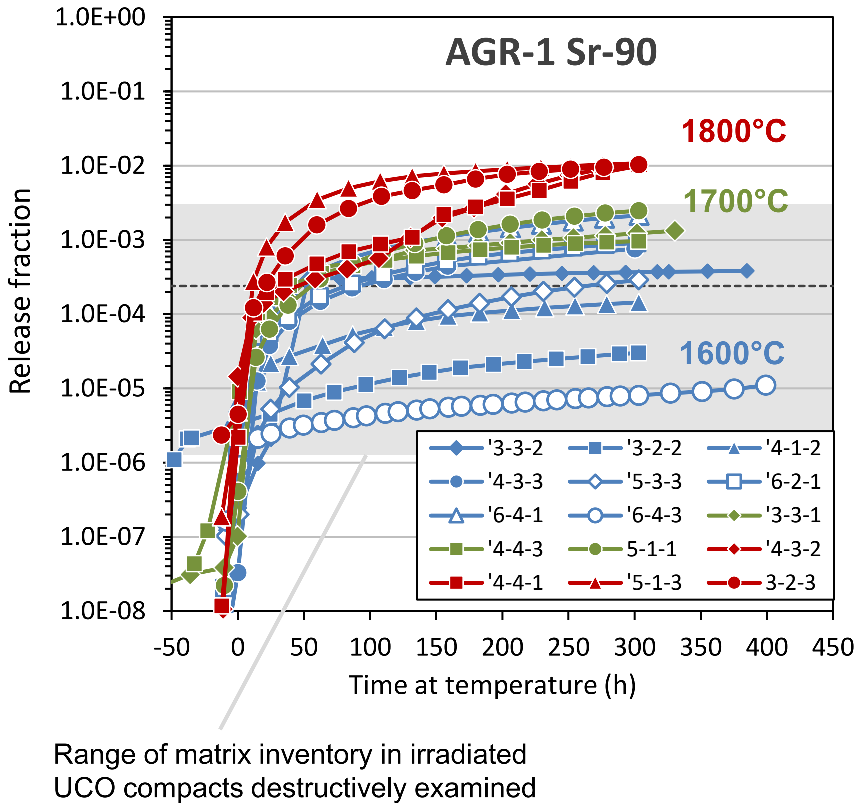


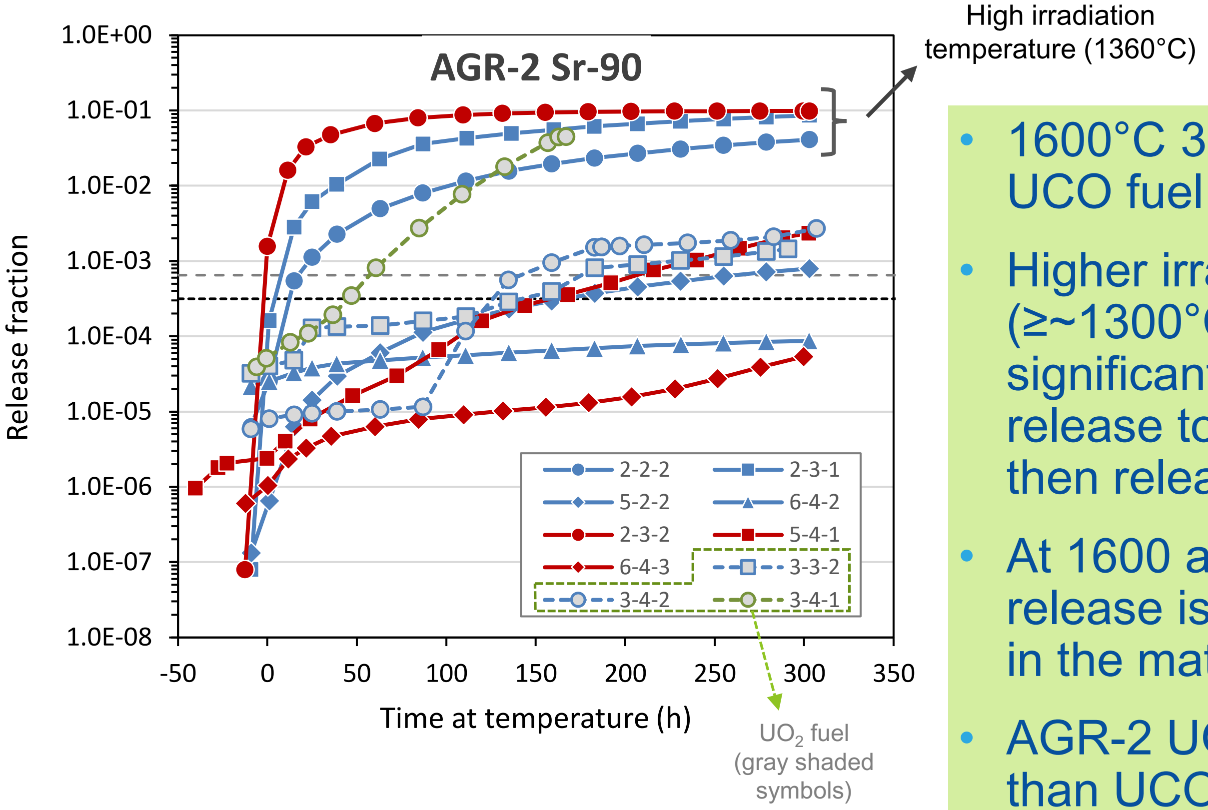




- **UCO fuel**: relatively low Cs release; release $>10^{-4}$ results from discrete SiC layer failure in 1 or more particles
- **UO_2 fuel**: higher Cs release compared to UCO; driven by CO attack on the SiC layer causing more widespread SiC failure

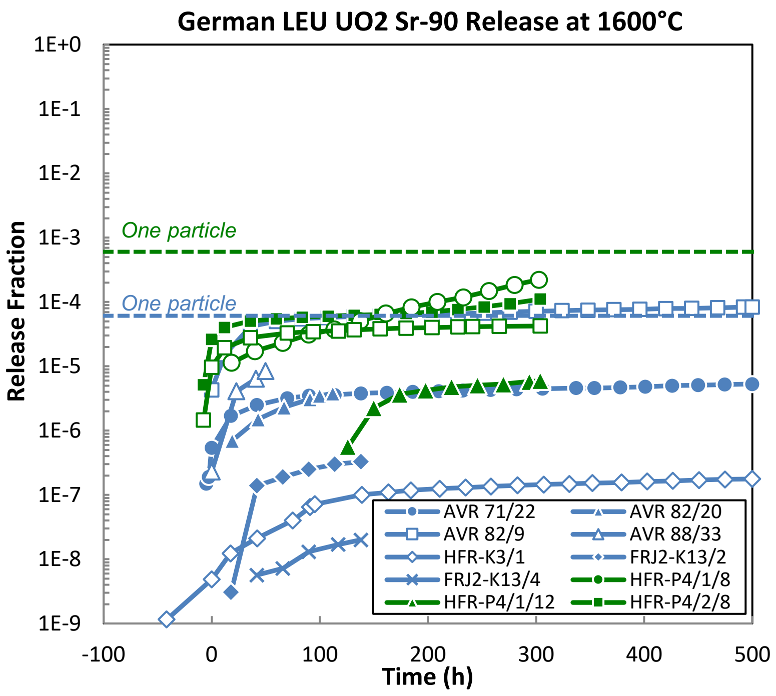
Sr Release: AGR-1 and AGR-2 Fuel

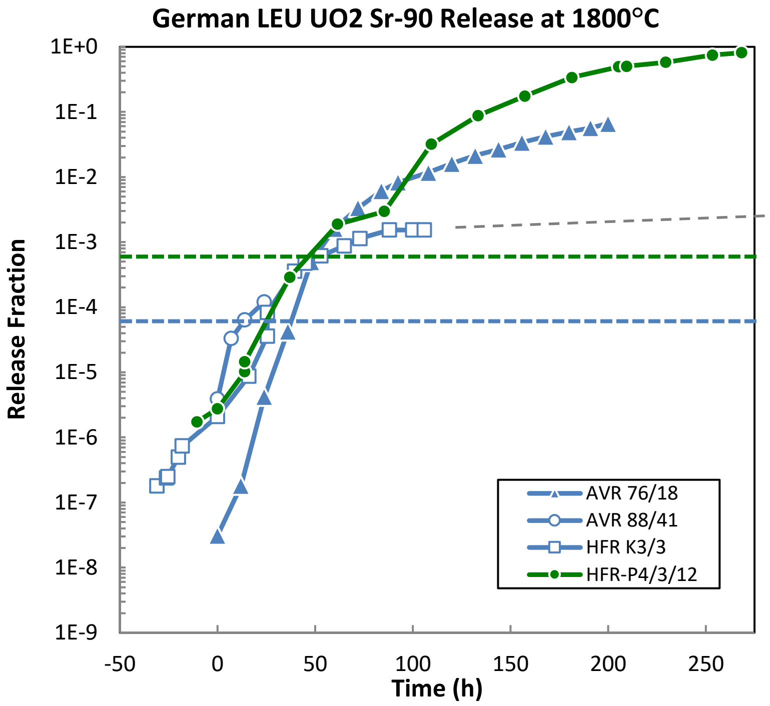


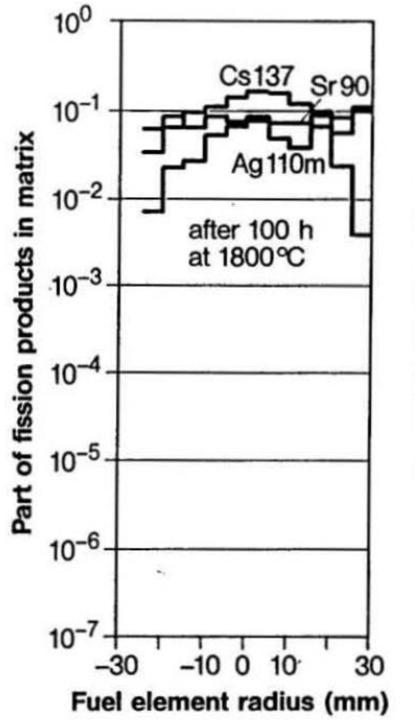


- 1600°C 300 h release from UCO fuel is 8×10^{-6} to 2×10^{-3}
- Higher irradiation temperature (≥~1300°C) results in significantly greater in-pile release to the matrix, which is then released upon heating
- At 1600 and 1700°C the 300 h release is limited to inventory in the matrix
- AGR-2 UO_2 has higher release than UCO at similar temperatures presumably from large number of particles with compromised SiC

Sr Release: German UO_2



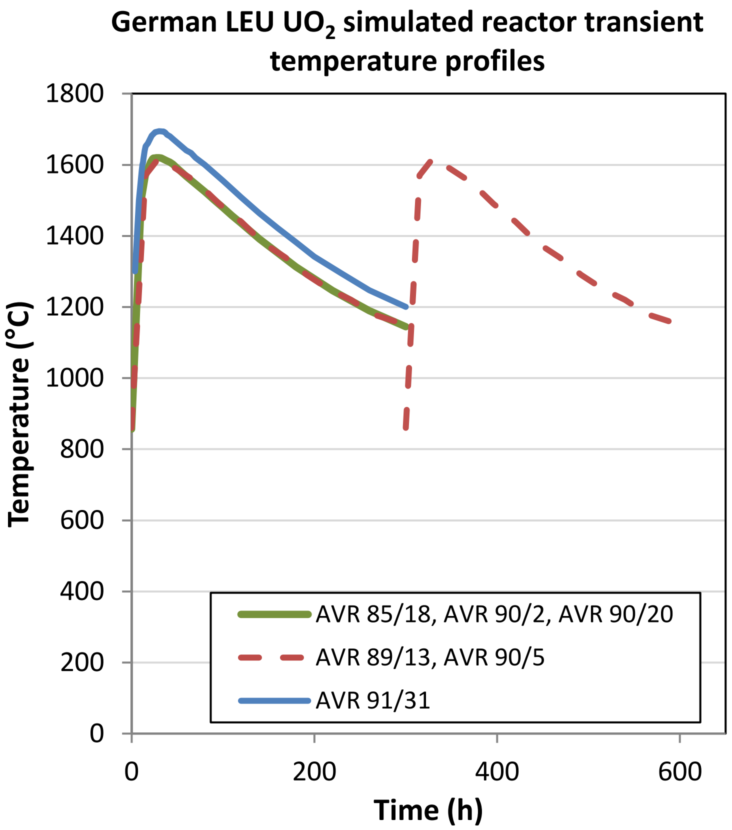


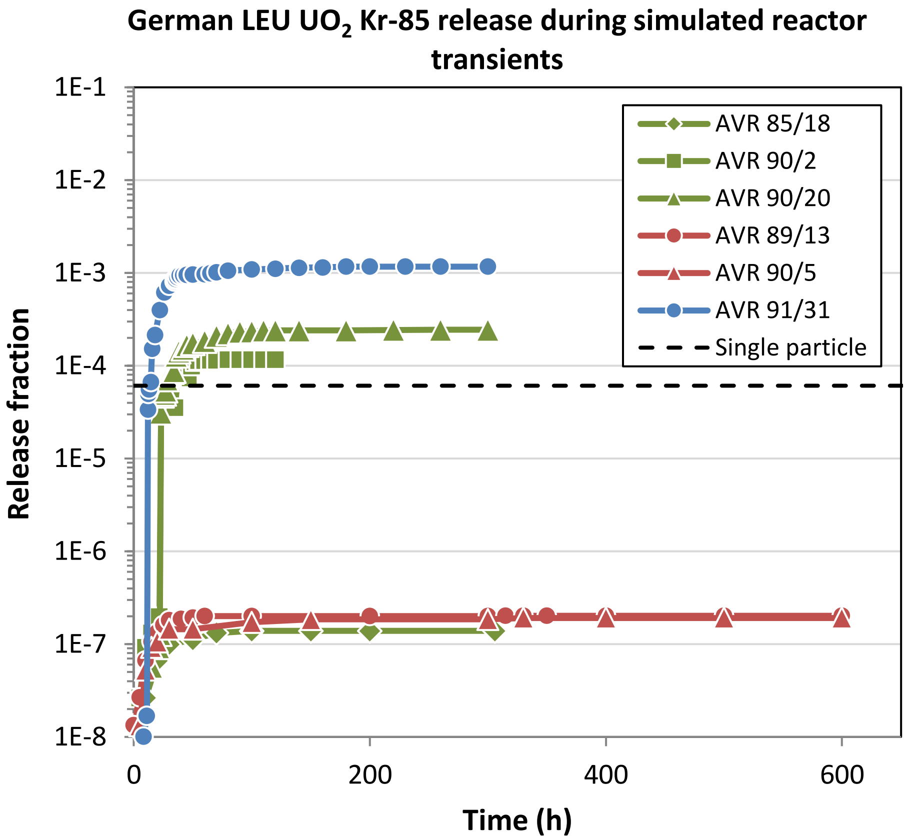


Fission product distribution in HFR-K3/3 sphere matrix after 1800°C for 100 h (from R. Gontard and H. Nabielek, *Performance evaluation in modern HTR TRISO fuels*, HTA-IB-05/90, 1990)

- Sr release is relatively low at 1600°C but significantly increases at 1800°C
- Significant fission product retention in the pebble matrix even at 1800°C

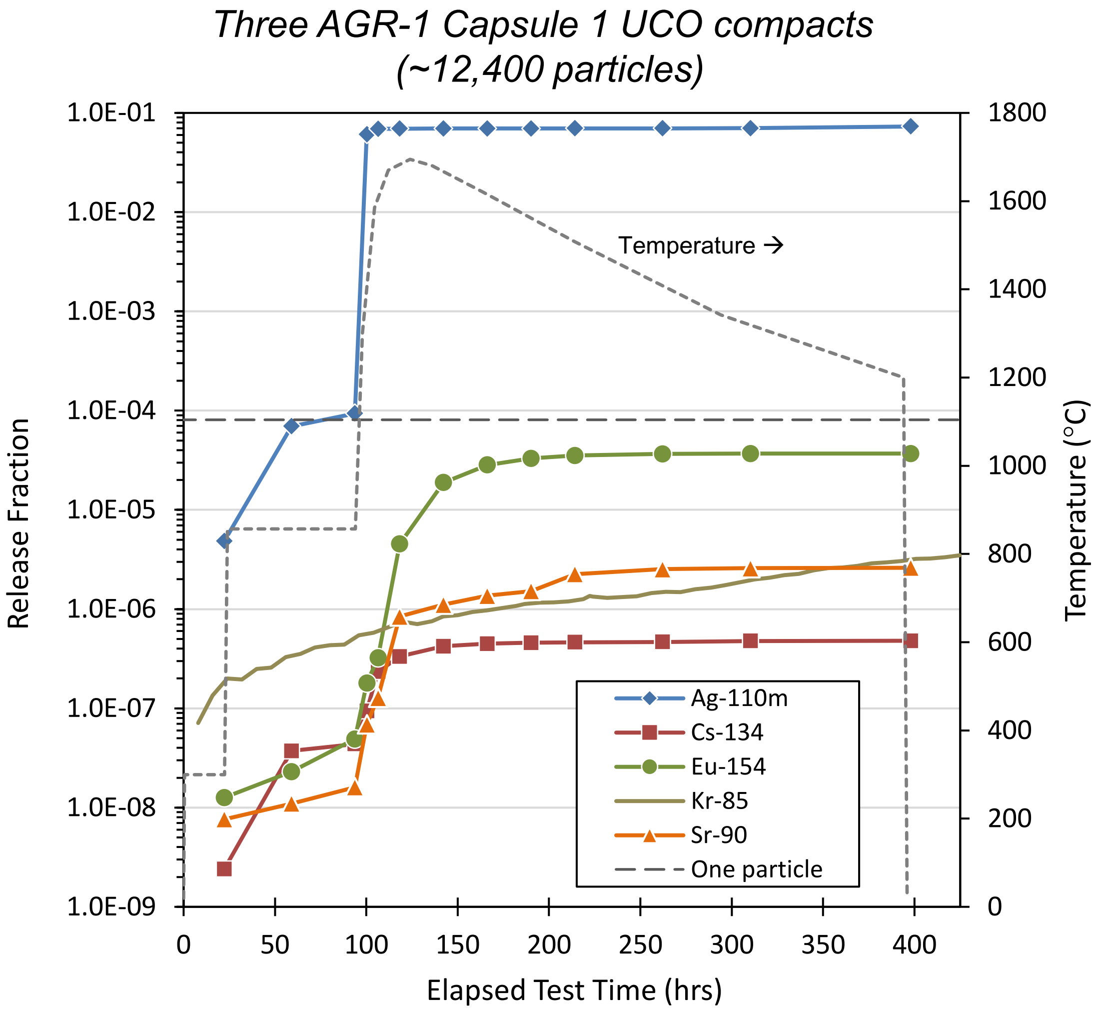
Transient Temperature Testing of German TRISO Fuel





- AVR spheres tested with transient temperature profiles and peak T of 1600 or 1700°C
- Several spheres exhibited ^{85}Kr release higher than single particle level, indicating failures
- Raised questions about the severity of the temperature profile in these tests
- AVR spheres had unknown temperature history, and these spheres from late in core life were suspected of experiencing very high temperatures at high burnup

AGR Program UCO Transient Temperature Safety Tests



- Three AGR-1 UCO compacts (~15% FIMA) heated with a temperature profile similar to AVR 91/31 (peak T ~1700°C)
- Kr and Cs release remained low indicating no particles with SiC or full TRISO failure
- Test was repeated with three AGR-2 UCO compacts (~12.6% FIMA) with similar results (Kr release $<2\times10^{-6}$)
- Results indicate no adverse effects on UCO TRISO particles from rapid* transient to 1700°C

**Temperature ramp rate during final heating to peak temperature is ~117°C/h compared with ~50C/h ramp to peak temperature during isothermal tests*

Fuel Design Safety Approach

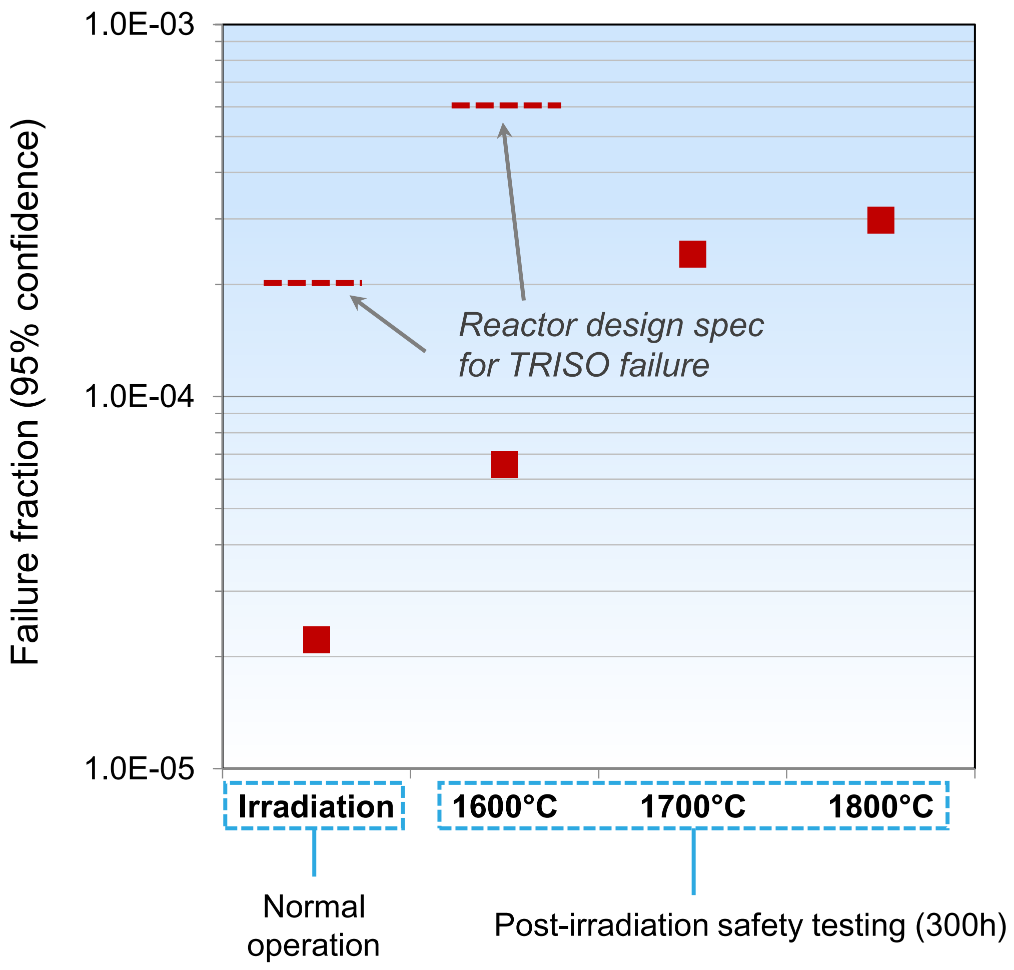
Specifications for particle defects and failure fractions

Parameter	NGNP – 750°C Core Outlet Temperature	
	"Maximum Expected"	"Design"
As-Manufactured Fuel Quality		
HM contamination	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Defective SiC	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
In-Service TRISO Failure		
Normal operation	$\leq 5.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-4}$
Accidents	$\leq 1.5 \times 10^{-4}$	$\leq 6.0 \times 10^{-4}$

- Start with imposed requirement for radionuclide release (e.g., radiological dose at site boundary)
- Establish specifications for as-manufactured contamination levels, particle defects, and in-pile particle failure fractions that can lead to fission product release
- Verify fuel quality with QC measurements
- Demonstrate failure fraction specifications are met during fuel irradiation and safety testing

Measured TRISO Failure Fractions

Experimental TRISO failure fractions for AGR-1 + AGR-2
(upper limit at 95% confidence)



- AGR-1 and -2 TRISO failure fractions meet historic design specifications with ~10X margin

Core Oxidation

- Accident scenarios in steam-cycle gas-cooled reactors can include air or steam ingress into the core (design basis events or beyond design basis events)
- Core behavior under these conditions should be evaluated
    - Graphite and matrix oxidation
    - Fission product volatilization from matrix/graphite and exposed kernels
    - Oxidation of coated particles could impact particle integrity
- Previous work has included characterization of oxidation behavior of core materials:
    - Graphite oxidation data is available in literature
    - Limited data on matrix oxidation
    - Limited data on kernel and coated particle response to core oxidants

Fission gas release from irradiated UCO fuel kernel in response to water vapor injection (HRB-17 experiment)

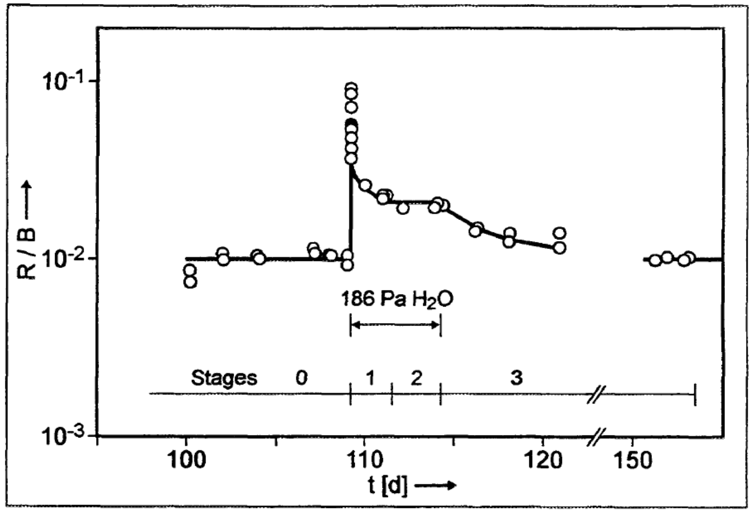


Fig. 5-1: R/B-time profile for Kr-85m before, during, and after a water vapor injection test with 186 Pa of water vapor at 755 °C

IAEA-TECDOC-978 (1997) is a good reference for earlier studies of fuel response to oxidants

Response of Fuel Particles to Oxidizing Conditions

- Modeling fuel response to core oxidation events is highly complex
- Additional experimental data are needed to model fuel response under a range of conditions
- Core conditions (temperature, gas flows, oxidant partial pressure) are very dynamic
- Graphite and matrix consumes oxidants as event progresses
- Specific conditions should be defined to the extent possible through models (temperatures, durations, oxidant partial pressures) to guide experimental efforts

German results on pebble and loose particle heating in 100% air (IAEA-TECDOC-978)

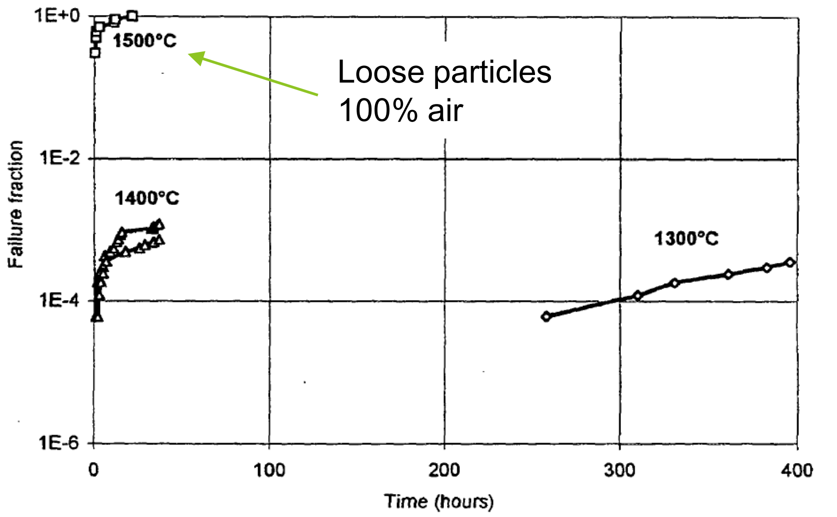
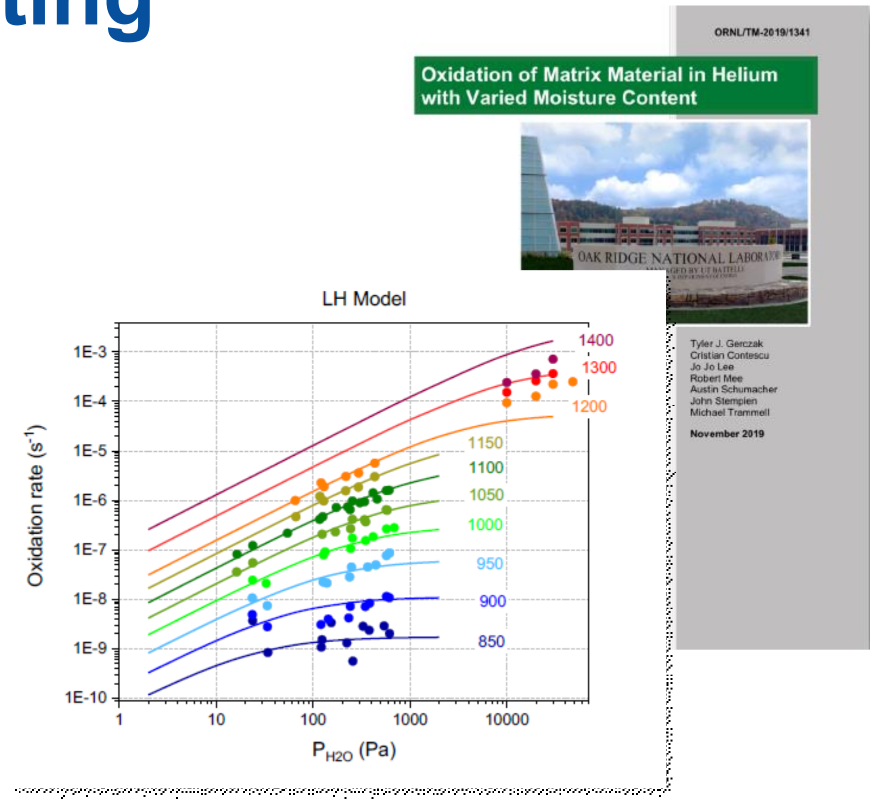


Fig. 5-21: Failure fraction of fuel particles within fuel spheres during heating in air at 1300 °C and of 10 intact, unbonded particles at 1500 °C

Current Status of Fuel Oxidation Testing

- US AGR program is performing dedicated testing to obtain key data:
  - Matrix oxidation tests
  - Irradiated fuel heating tests in air and steam environments to ~1600°C with fission product release measurement (starting ~2022)
- Recent published results on SiC oxidation
  - Generally indicate limited oxidation at times and temperatures relevant to HTGR accidents, but test conditions can vary significantly in the tests
- US DOE is funding studies of SiC and fuel matrix oxidation under HTGR air/steam ingress conditions under the Nuclear Energy University Program (NEUP)



Oxidation behavior of fuel matrix in steam

Fuel Accident Performance Summary

- Extensive accident testing database for fuel heating in helium to 1800°C
- Fuel withstands 300 h at temperatures of 1600°C and above with low failure rates
- Observed particle failure fractions are well below historic reactor design specs
- TRISO fuel fission product release behavior is well-characterized
- Additional data needed under core oxidation conditions

Outline

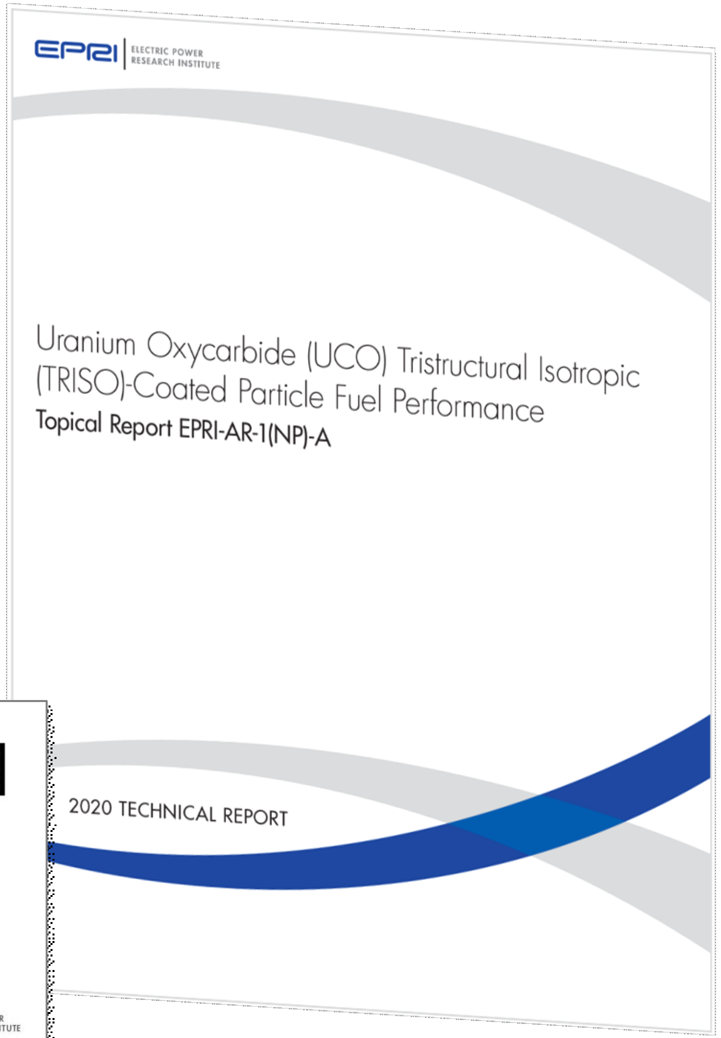
Part II

- Fuel Accident Performance
- US NRC Engagement on Fuel Qualification

UCO TRISO Fuel Performance Topical Report to US NRC

- Overall goal was to generate a pre-approved resource that can be incorporated into subsequent applications and licensing decisions
- Capture portions of the AGR program that are completed while the information is current
- Enable early understanding of key technical areas relative to qualifying TRISO UCO fuel
- Minimizes later efforts by applicant and NRC staff

- INL supported the effort to compile and describe AGR program technical results
- Key stakeholders participated in document review and preparation
- EPRI submitted report to NRC in 2019 https://www.epri.com/research/products/3002019978


EPRI ELECTRIC POWER RESEARCH INSTITUTE

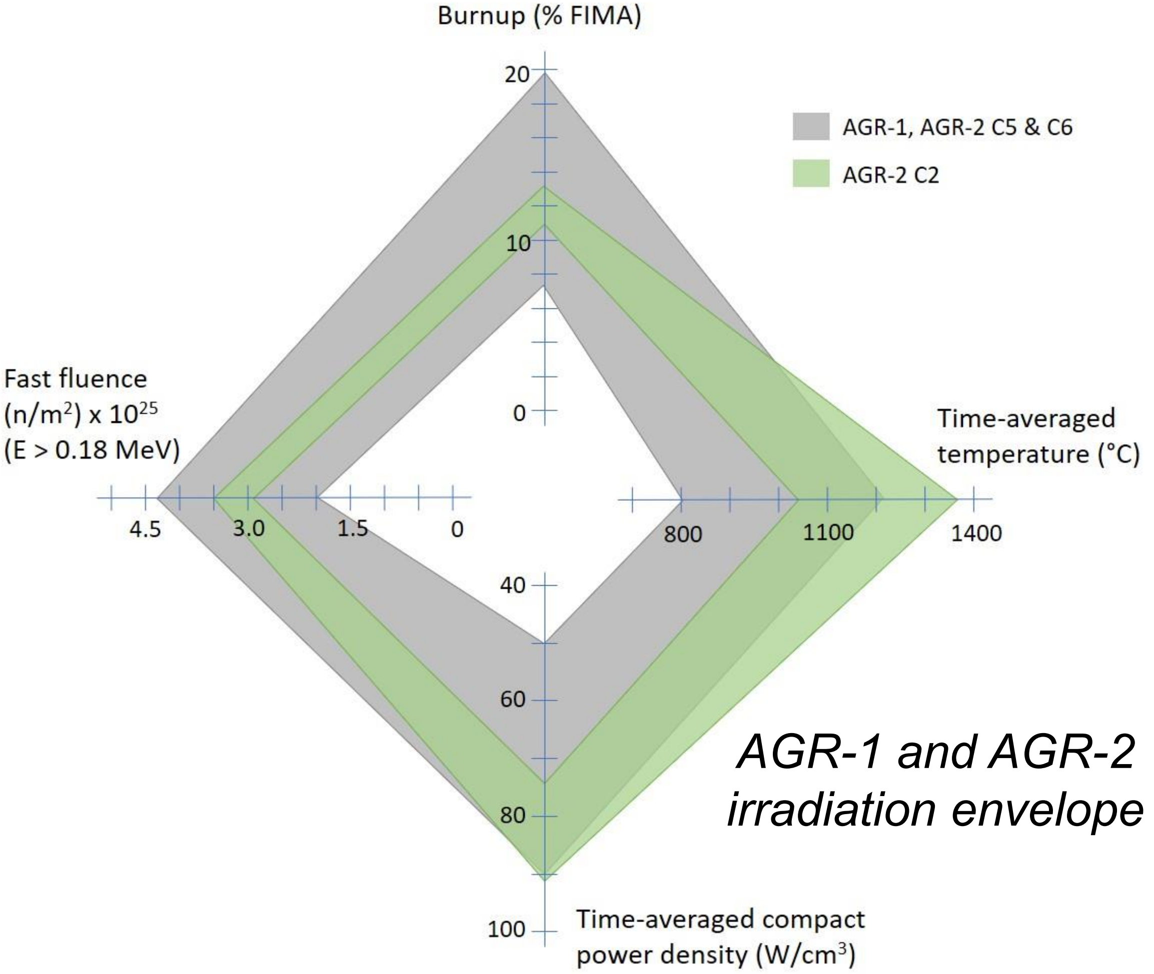
Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance

Topical Report EPRI-AR-1(NP)-A

2020 TECHNICAL REPORT

Topical Report Scope

- Background includes regulatory bases, TRISO fuel historical experience, and current particle design bases
- Focus is on **AGR-1** and **AGR-2** particle fabrication data and particle performance during irradiation and safety testing
- Seeks NRC agreement with the report conclusions, which state that fuel particles fabricated with similar kernel and coating characteristics will produce similar performance within the same irradiation envelope
- "In effect, information presented in the TR would allow applicants or licensees who reference the TR to use the AGR data and associated conclusions for TRISO particle performance"
  –NRC staff, July 8, 2020



AGR-1 and AGR-2 irradiation envelope

NRC Safety Evaluation

https://www.nrc.gov/docs/ML2021/ML20216A453.pdf

- Staff generally finds that the conclusions of the report are “applicable and acceptable”, subject to the limitations and conditions in the Safety Evaluation
- “*This TR forms the basis for establishing the design limits for TRISO fuel.*” –NRC SE
- “*The [Topical Report] represents a modified approach for qualifying a novel fuel design –rather than deterministic limit values, fuel performance is characterized statistically across a population of particles based on test conditions*” –ACRS Aug 4, 2020
- Data does not cover all accident scenarios, but provides data that can be used by applicants to evaluate fuel behavior under their accident conditions
- Limitations:
  1. Scope applies to UCO TRISO fuel only; impact of fuel form (sphere, compact) on particle behavior is the responsibility of the applicant
  2. Applicant’s responsibility to demonstrate that their fuel falls within the ranges of properties specified
- Conditions:
  1. Applicant must evaluate differences between their fuel particles and AGR particles, with specific mention of impact of particle design (kernel diameter, coating thicknesses, peak burnup) on SiC stress
  2. Applicant must demonstrate how their fuel operating conditions are within the AGR-1 and AGR-2 envelope
  3. Short-lived, condensable fission product release data is not supplied in this report, so applicants will have to address their impact separately

Suggested Reading

General TRISO Fuel

- P.A. Demkowicz et al., Coated particle fuel: Historical perspectives and current progress, J. Nucl. Mater. 515 (2019) 434-450
- M.J. Kania, H. Nabielek, H. Nickel, Coated Particle Fuels for High-Temperature Reactors, in Materials Science and Technology, Wiley 2015.
- D.A. Petti et al., TRISO-Coated Particle Fuel Performance, in Konings R.J.M.,(ed.) Comprehensive Nuclear Materials (2012), vol. 3, pp. 151-213 Amsterdam: Elsevier.
- High Temperature Gas Cooled Reactor Fuels and Materials, IAEA, TECDOC-1645 (2010).
- K. Verfondern, H. Nabielek, J.M. Kendall, Coated particle fuel for high temperature gas cooled reactors, Nucl. Eng. Tech. 39 (2007) 603-616.
- D.A. Petti et al., Key differences in the fabrication, irradiation and high temperature accident testing of US and German TRISO-coated particle fuel, and their implications on fuel performance, Nucl. Eng. Des. 222 (2003) 281-297.
- Fuel performance and fission product behavior in gas cooled reactors, IAEA, TECDOC-978 (1997).

Suggested Reading (cont.)

AGR Program Results

- P.A. Demkowicz et al., "Key results from irradiation and post-irradiation examination of AGR-1 UCO TRISO fuel," Nucl. Eng. and Des. 329 (2018) 102–109.
- P.A. Demkowicz et al., AGR-1 Post Irradiation Examination Final Report, INL/EXT-15-36407, Idaho National Laboratory, 2015.
- J.D. Hunn et al., "Post-Irradiation Examination and Safety Testing of US AGR-2 Irradiation Test Compacts," Paper 10 in Proceedings of the 9th International Topical Meeting on High Temperature Reactor Technology (HTR-2018), Warsaw, Poland, October 8–10, 2018. Available at https://www.osti.gov/biblio/1489588
- J.D. Hunn et al., "Initial Examination of Fuel Compacts and TRISO Particles from the US AGR-2 Irradiation Test," Nucl. Eng. and Des., 329 (2018) 89–101.

HTR-PM Fuel

- C. Tang et al., Comparison of two irradiation testing results of HTR-10 fuel spheres, Nucl. Eng. Des. 251 (2012) 453-458.
- S. Knol et al., HTR-PM fuel pebble irradiation qualification in the high flux reactor in Petten, Nucl. Eng. Des. 329 (2018) 82-88.
- D. Freis et al., Burn-up Determination and Accident Testing of HTR-PM Fuel Elements Irradiated in the HFR Petten, Proceedings of the 9th International Topical Meeting on High Temperature Reactor Technology (HTR-2018), 8-10 Oct. 2018, Warsaw, Poland.

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