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AGR-3/4 Fuel Compact Fission Product Concentration Profiles

July 2024

Changing the World's Energy Future

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Idaho National Laboratory

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GAS-COOLED REACTOR
ADVANCED REACTOR TECHNOLOGIES PROGRAM

AGR-3/4 Fuel Compact Fission Product Concentration Profiles

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Outline

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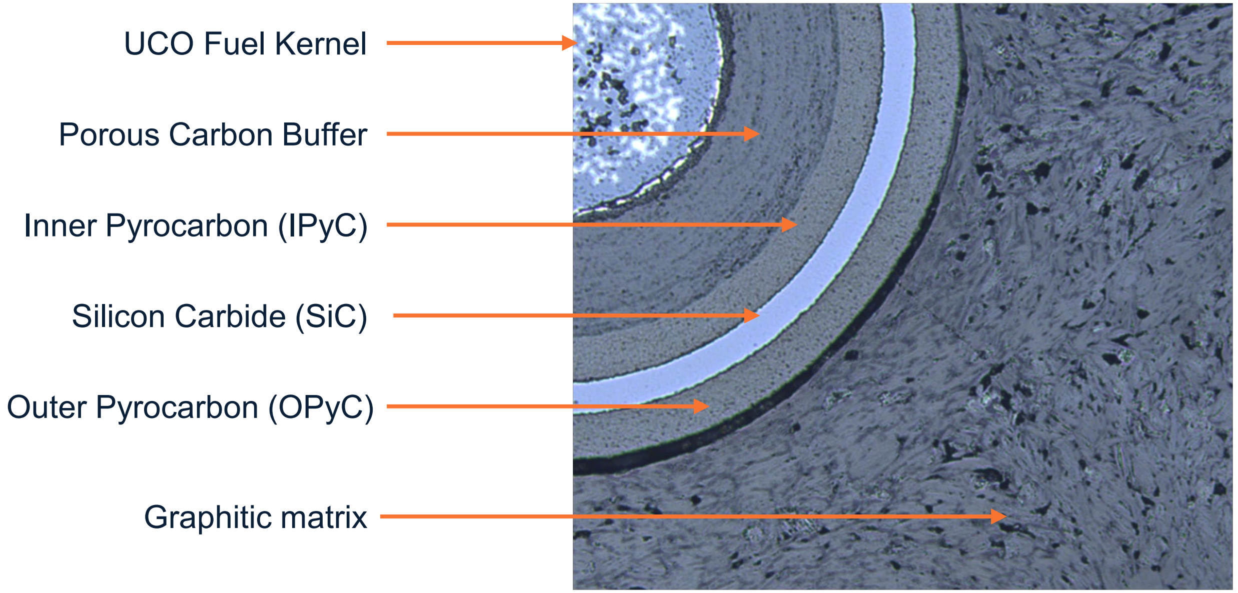
  - As-irradiated vs. FACS-tested
- Summary and Conclusion

Introduction

AGR-3/4

TRISO Fuels Fission Product Source Term

- Each material in TRISO fuel retains or attenuates fission products

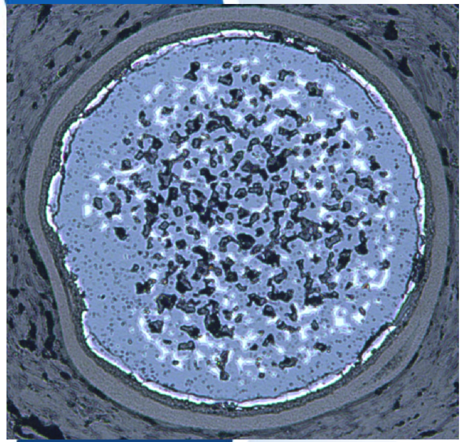


AGR-3/4 Goals

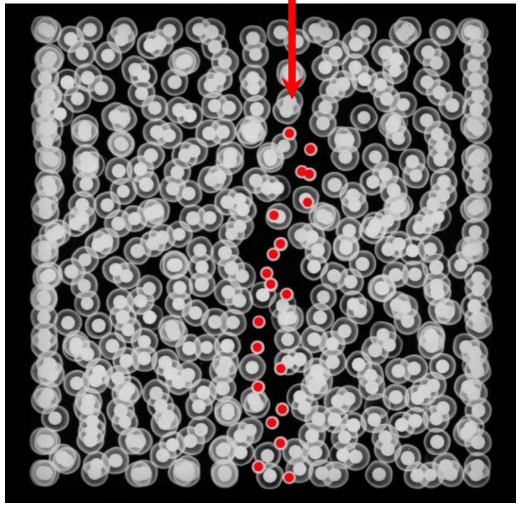
- Improve reactor source term predictions
- Provide some data for validation of source term calculations

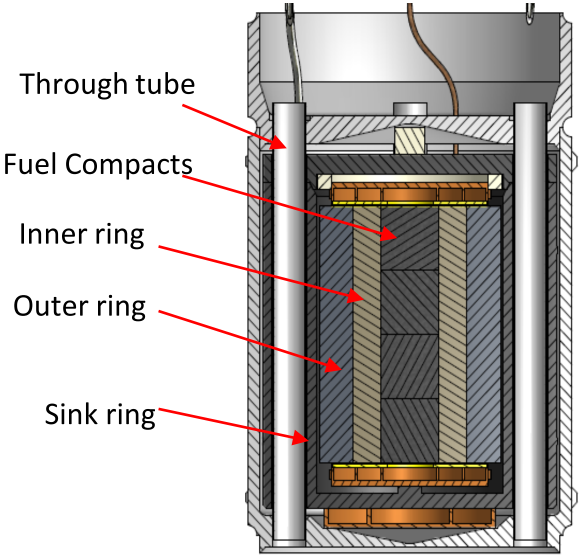
AGR-3/4 Designed to Observe Fission Product Transport from Fuel through Graphite

Designed-to-fail (DTF) Particle

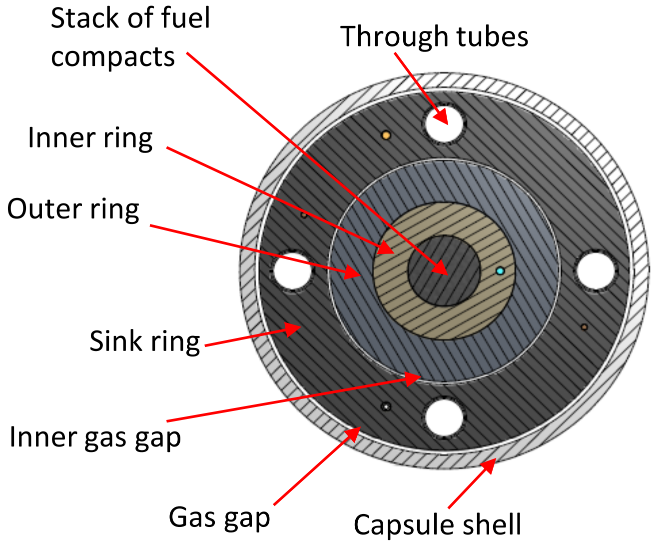


X-ray showing 20 DTF particles in center of compact





Axial cutaway of an AGR-3/4 capsule



AGR-3/4 capsule cross section

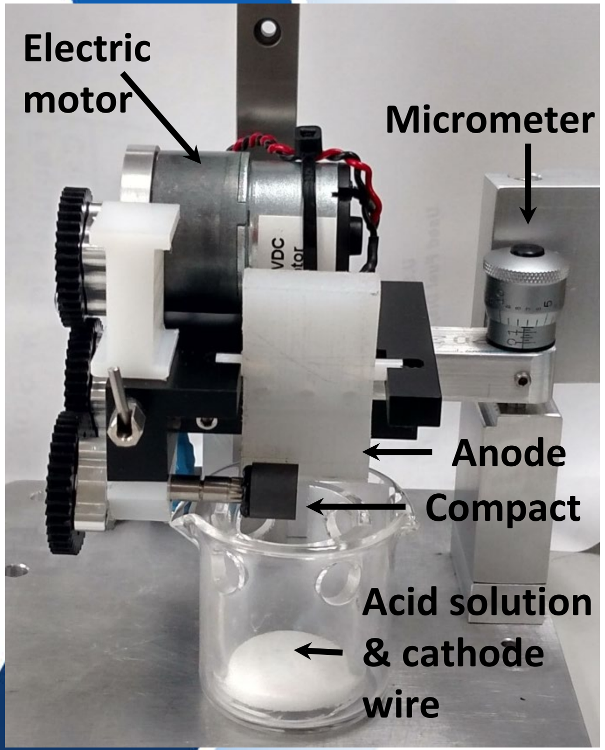
- Observing metallic fission product (e.g., Ag, Cs, Eu, and Sr) transport within graphitic matrix and nuclear grade graphites (IG-110 and PCEA)
- Measuring fission product inventories and spatial distributions within fuel compacts and graphite
- Determinizing diffusion coefficients of metallic fission products within graphitic materials

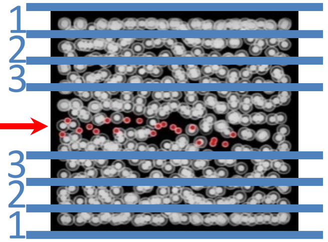
Experimental

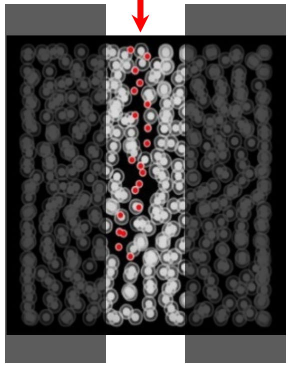
Methods, Sample Selection and Challenges

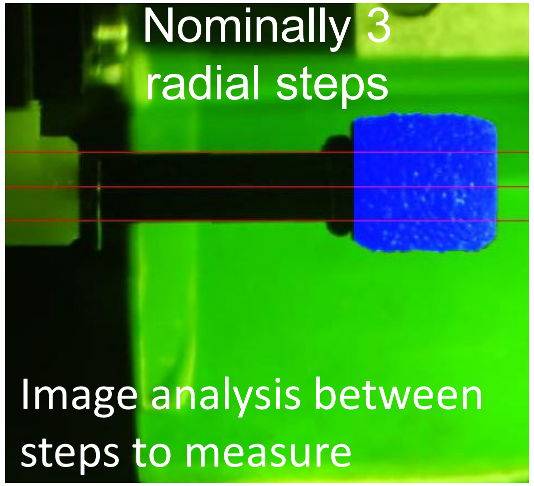
Radial Deconsolidation Method

Measure fission product radial concentration profiles in the compacts









Sample Selections

		Burnup (% FIMA)	Neutron Fluence (10^{25} n/m^2, E>0.18 MeV)	TAVA Temp (°C)	TA Min Temp (°C)	TA Peak Temp (°C)	FACS Temperature (°C)	References
As-irradiated	3-3	12.73	4.28	1205	1170	1242	N/A	Stempien and Cai 2024
	5-3	14.92	5.22	1050	1001	1102		
	5-4	14.98	5.23	989	858	1084		
	7-3	15.00	5.27	1376	1335	1418		
	8-3	14.54	5.07	1213	1171	1257		
	10-3	11.75	3.89	1210	1174	1248		
	12-1	5.87	1.8	849	802	883		
	12-3	5.17	1.41	864	844	884		
	1-4	6.85	2.10	929	866	972		Hunn et al. 2020
	7-4	14.90	5.24	1319	1206	1397		Helmreich et al. 2021
	8-4	14.43	5.02	1169	1068	1242		
	1-3	6.37	1.87	959	942	978		Helmreich et al. 2022
FACS Tested	3-2	12.49	4.17	1196	1154	1240	1600 1700[a]	Report in preparation
	8-2	14.58	5.11	1213	1171	1257	1400	
	10-2	11.96	4.01	1213	1179	1249	1200	Helmreich et al. 2022
	10-4	11.43	3.75	1168	1079	1231	1400	Helmreich et al. 2022
NRAD Reirradiated and FACS-Tested	1-2	5.91	1.66	941	910	971	1400	Report in preparation
	3-1	12.16	4.04	1138	1041	1214	1600	
	8-1	14.51	5.13	1165	1063	1242	1200[b]	
	4-3	14.29	4.89	1035	992	1084	1000	
	10-1	12.08	4.12	1172	1080	1238	1400	Helmreich et al. 2022

a. After the initial isothermal hold at 1600°C for 300 h, the temperature was raised to 1700°C for 48 h.
b. Temperature held at 1200°C for about 300 h. Then three cycles between temperatures <200°C and 1200°C.

R-DLBL - Challenges

The measured actinides and fission products include those that

(a) migrated out of the DTF particles into the surrounding compact matrix,

(b) were retained in the DTF kernels that were leached during DLBL,

(c) migrated through the intact SiC layer into the compact matrix,

(d) were related to uranium contamination present in the compact matrix and/or OPyC at the time the compact was fabricated, **negligible**

(e) were externally introduced by contamination from sources present in the hot cell, **Monitored, negligible**

(f) were from TRISO-coated particles with damaged SiC (or TRISO layers)

- in-pile failure
- as-fabrication defects
- accidentally damaged by the RDLBL process

Needs correction

Complicate the interpretation of the fission product concentration profile

R-DLBL - Correction

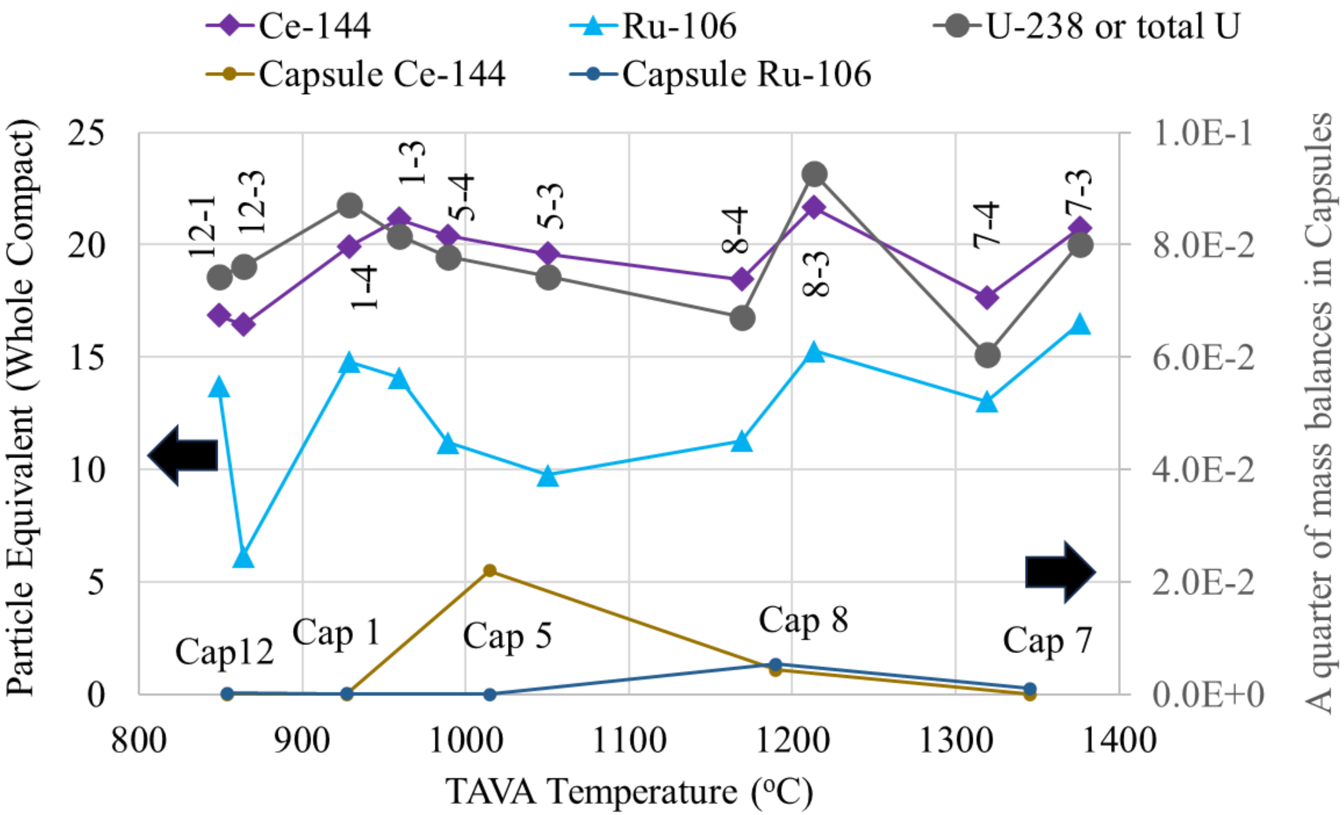
	Compact	Damage
As-irradiated	3-3	Up to ~14 particles at various stages of Segment 1 and Core
	5-3	Segment 1, post-burn leach 1: 1 particle
	5-4	None
	7-3	Segment 1, post-burn leach 2: 1 particle
	8-3	Segment 2, pre-burn leach 2: 1 particle
	10-3	Segments 1, 2, 3 and core: 10-20 particles
	12-1	Segment 1 decon: 2 particles. Segments 2 and 3 decons, Segment 2 post-burn leach 1: 1 particle each
	12-3	None
	1-4	Segment 1, post-burn leach 1: 1 particle
	7-4	Segment 1, deconsolidation: 1 particle
	8-4	None
	1-3	None
FACS Tested	3-2	Segment 3, post-burn leach 1: 1 particle
	8-2	Segment 3, post-burn leach 1: 2 particles
Reirradiated FACS-Tested	3-1	Segment 1, decon: 1 particle. Segment 1 post-burn leach 1: ~6 particles
	8-1	None

Compacts 3-3 and 10-3 had numerous damaged particles, which could not be reasonably corrected.
The Segment 1 of Comact 3-1 was discarded from further discussion.

Results

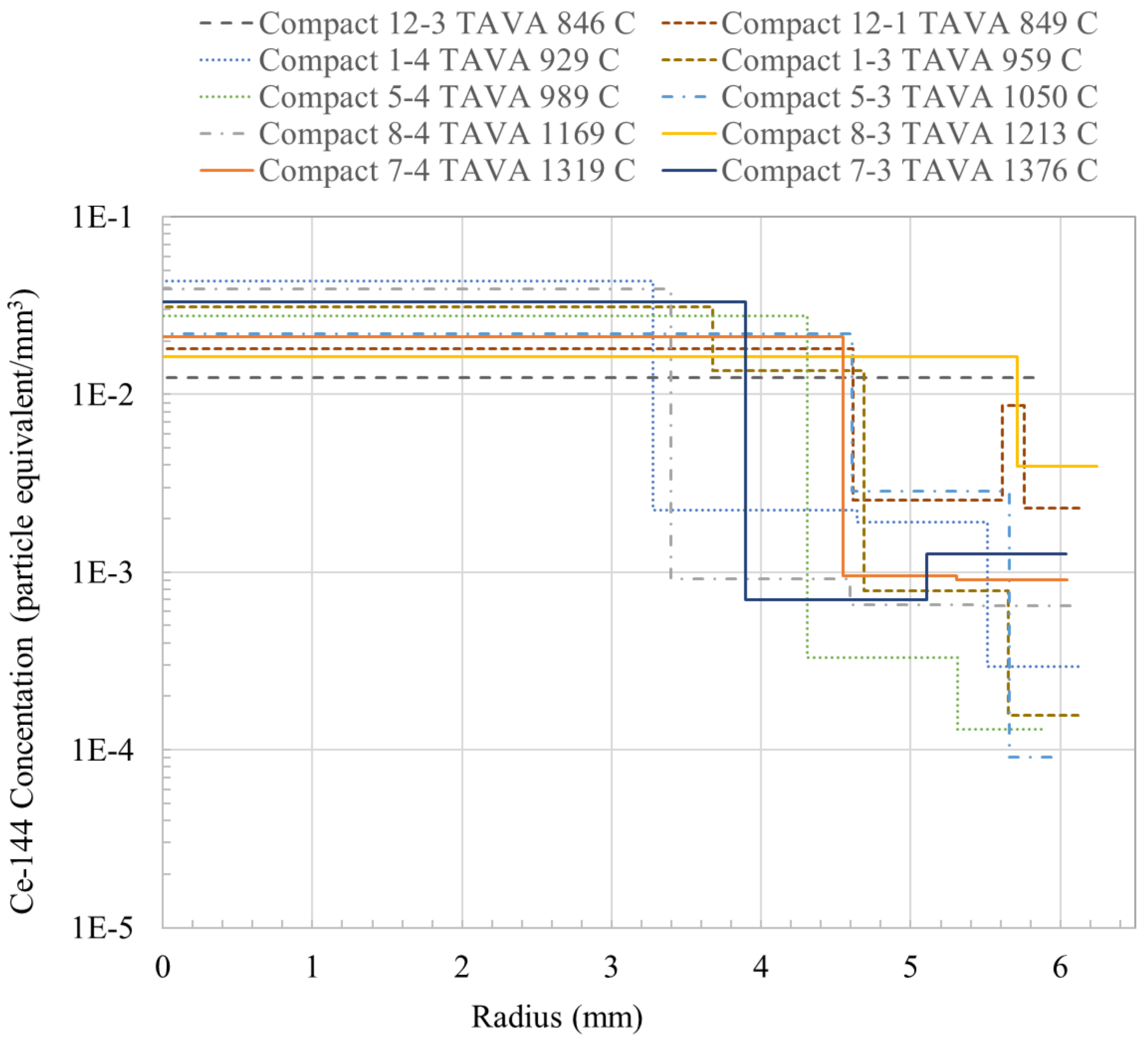
As-irradiated RDLBL

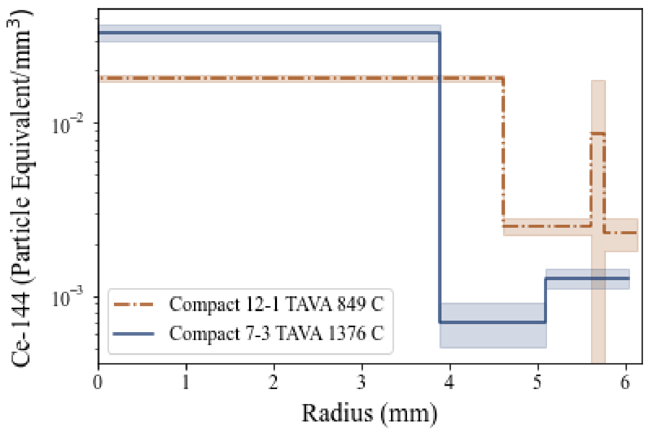
As-irradiated R-DLBL – U, Ce, Ru



- The Ce-144 and U inventories match well with each other and tend to cluster around 20 particle equivalents, the number of DTF particles per compact.
- There is no strong trend in the RDLBL inventories of Ce, Ru, or U vs. TAVA temperature.
- These nuclides were retained in the fuel compact.

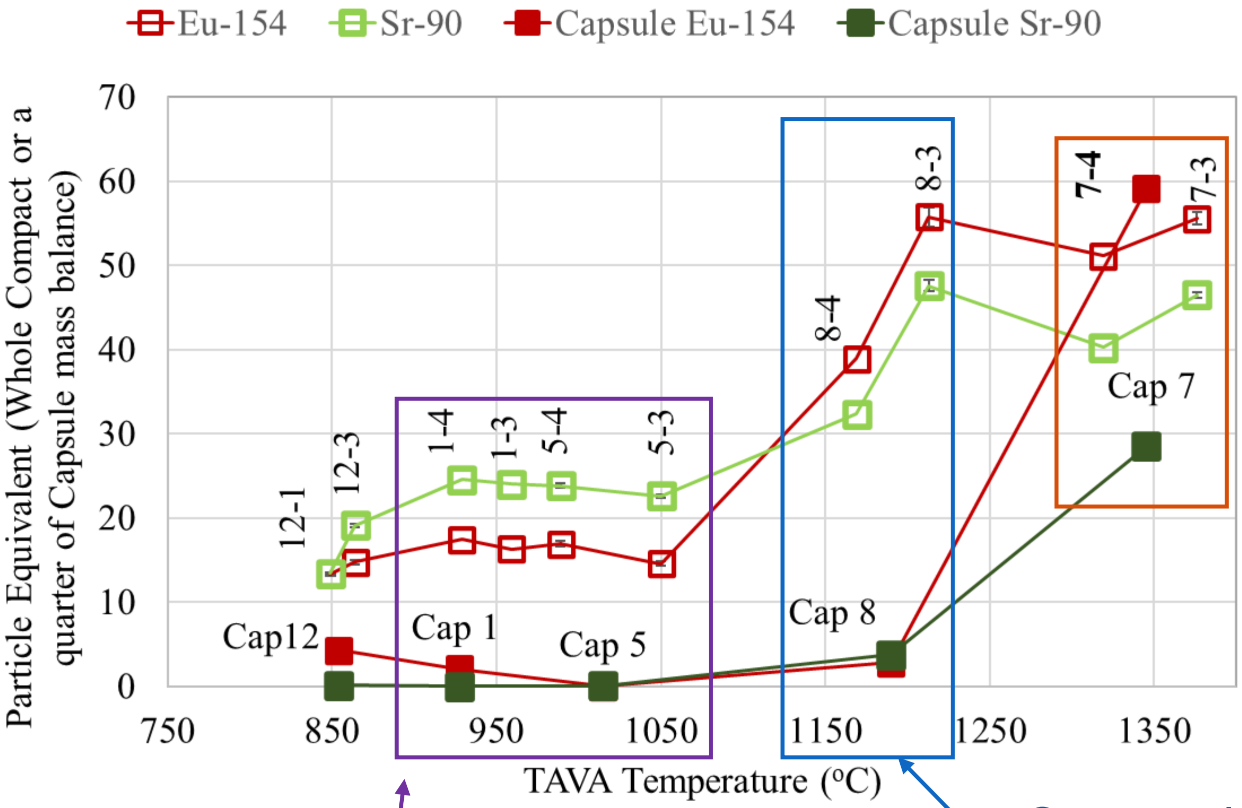
As-irradiated R-DLBL – Ce-144 profile





- The concentrations of Ce 144 generally decreased with increasing radius.

As-irradiated R-DLBL – Eu-154, Sr-90

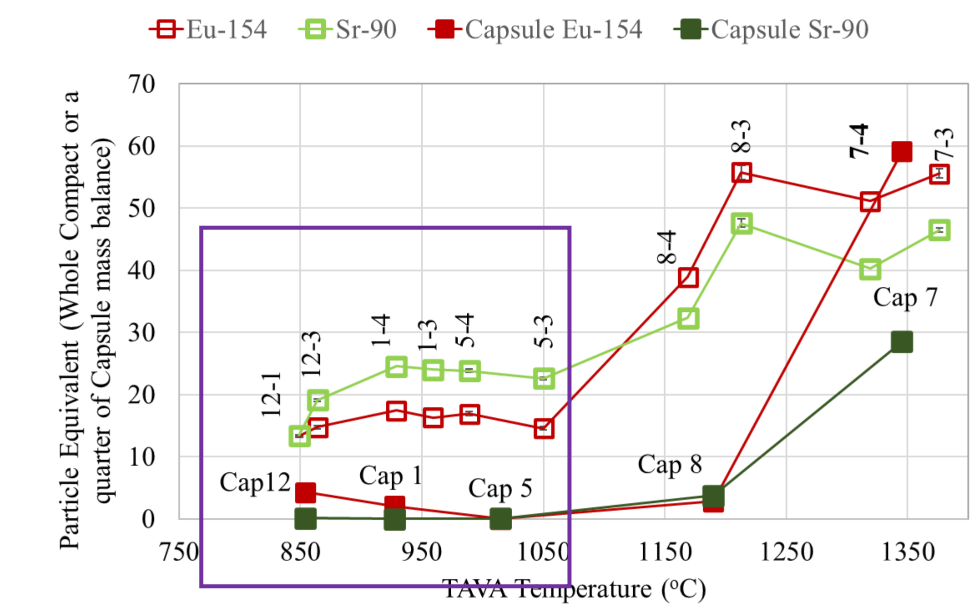


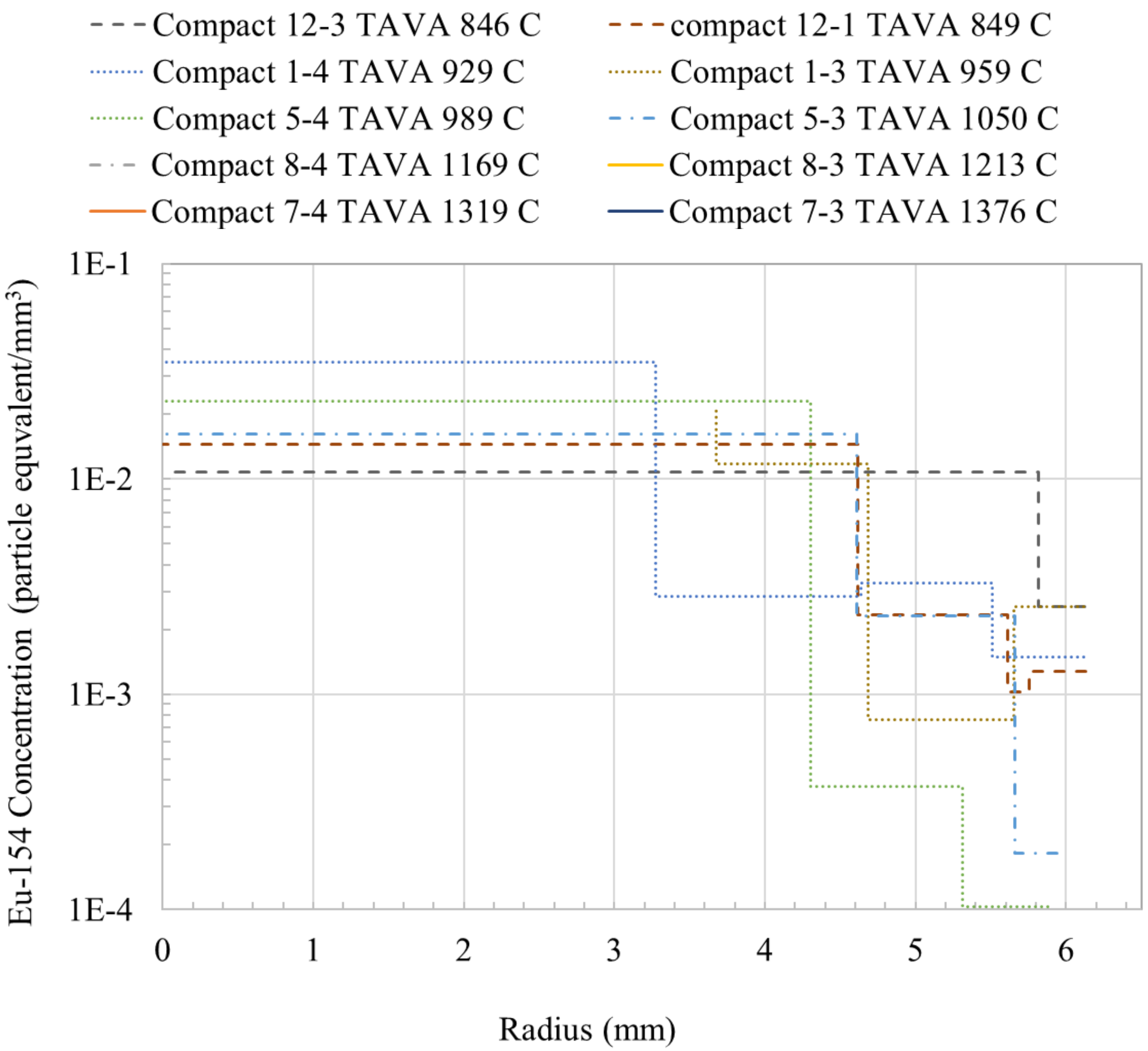
release outside compact

noticeable Sr release through intact SiC

Strong release of Eu and Sr through intact SiC but still kept in the compact matrix

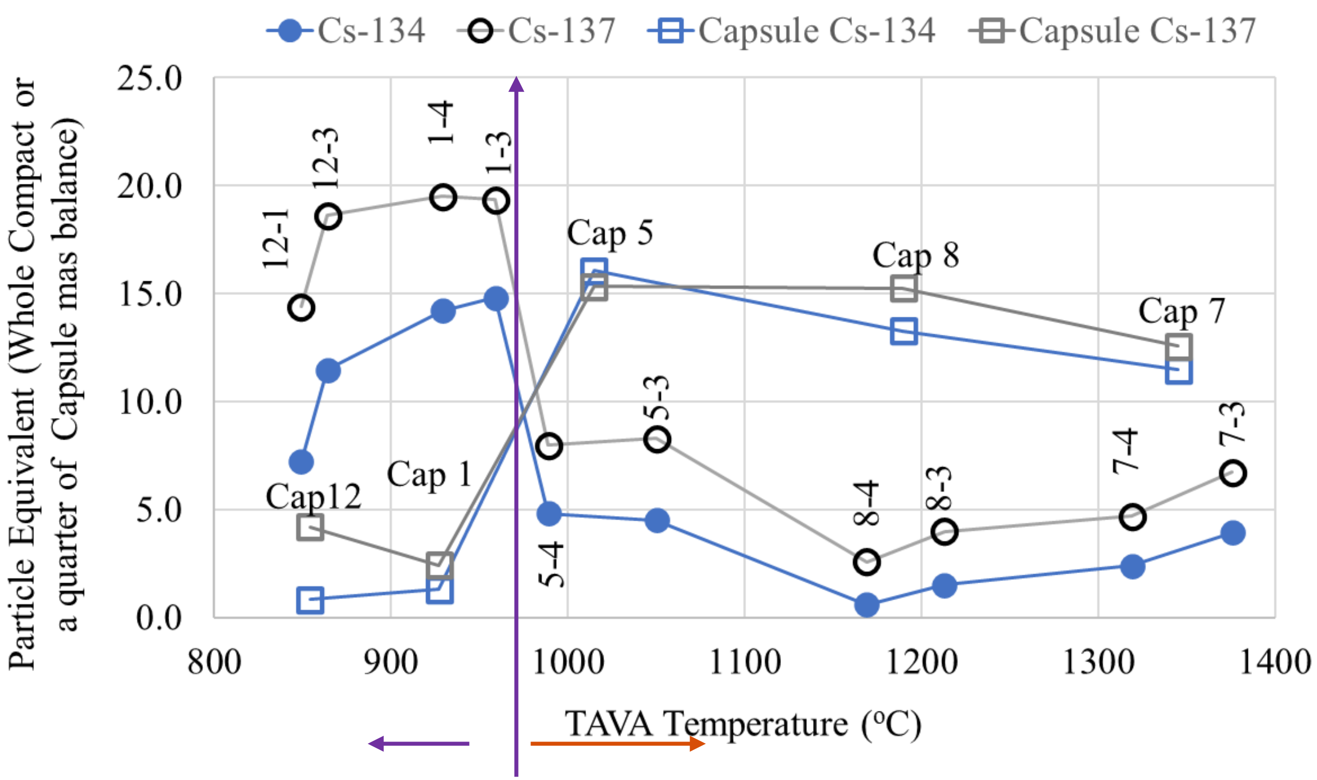
As-irradiated R-DLBL – Eu-154





- The Eu-154 concentration of compacts with TAVA below or at 1050°C decrease with increasing radius.
- The compacts with higher TAVA have flatter Eu-154 concentration.

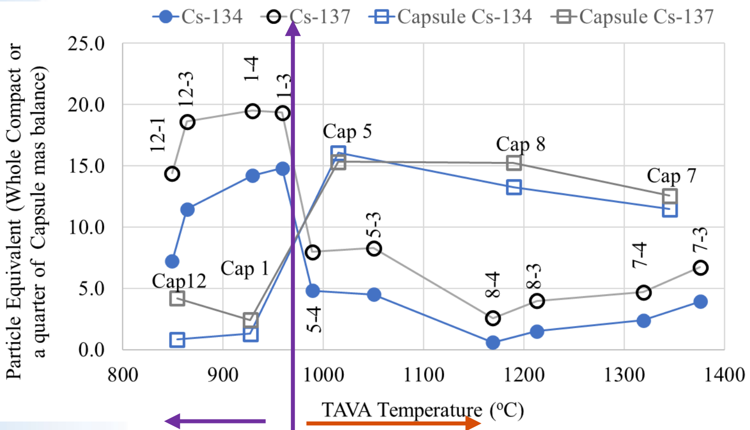
As-irradiated R-DLBL – Cs



Cs from DTF particles kept inside the compact matrix

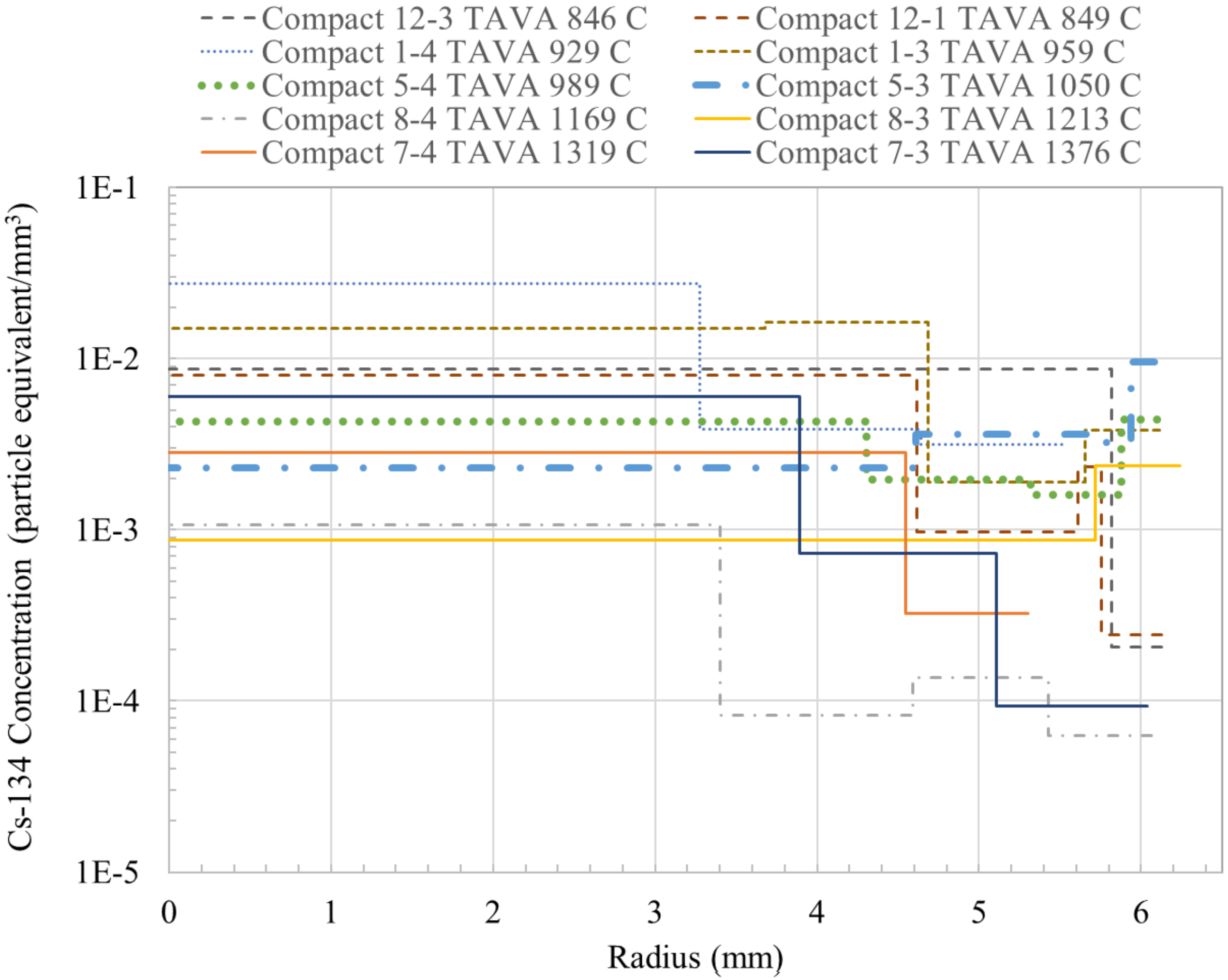
Cs release outside of compacts

As-irradiated R-DLBL – Cs-134



Cs kept inside the compact matrix

Cs release outside of compacts



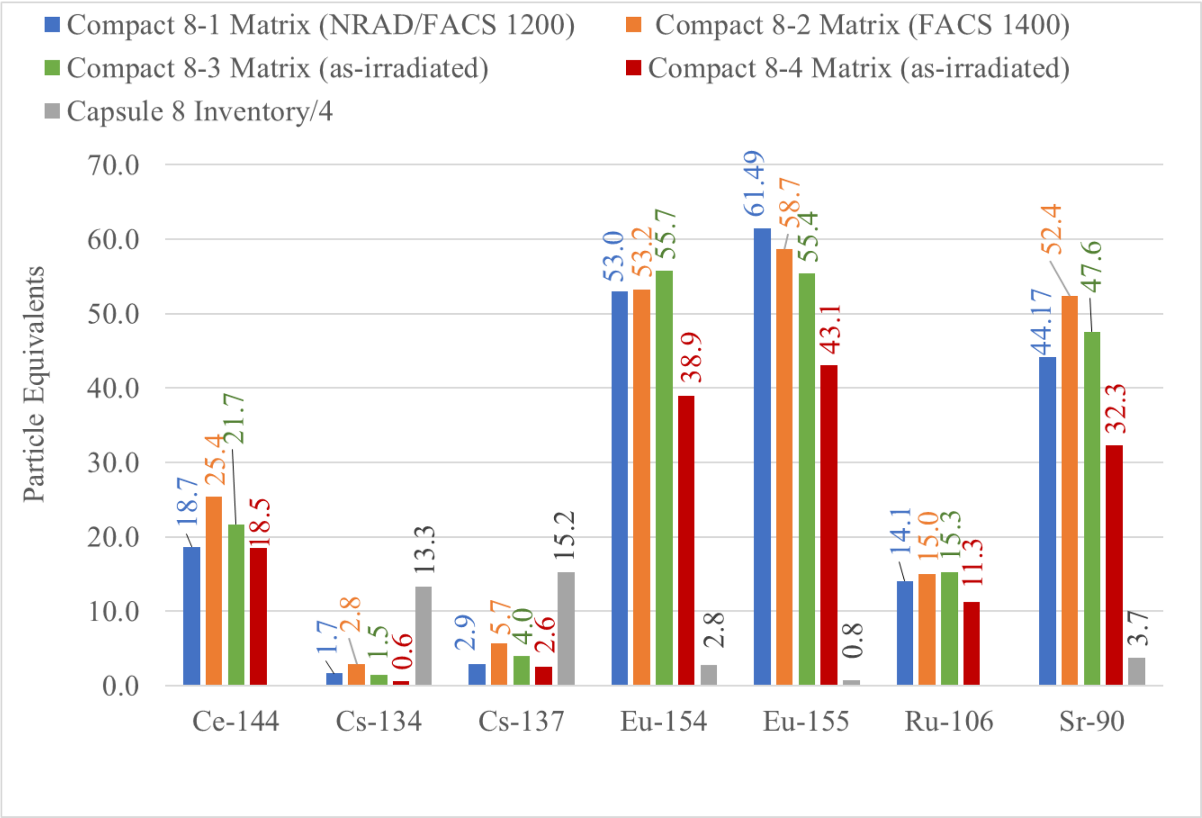
- For compacts with low TAVA, the Cs-134 concentration generally decreased with increasing radius.
- Compacts 5-3 and 5-4 have lower core concentration and flatter profile.
- Compact 8-3 and 8-4 have even lower core concentration.

Results

As-irradiated RDLBL vs. FACS tested

Capsule 8: As-irradiated vs. FACS-tested

- Eu, Sr, Cs released ≤ 0.3 particle equivalent during safety testing for Compacts 8-1 and 8-2
- Compact 8-4 matrix has lowest Eu and Sr than the rest of compacts.
- It is unclear if Cs was under recovered from RDLBL or the mass balance (the capsule shell was not measured)

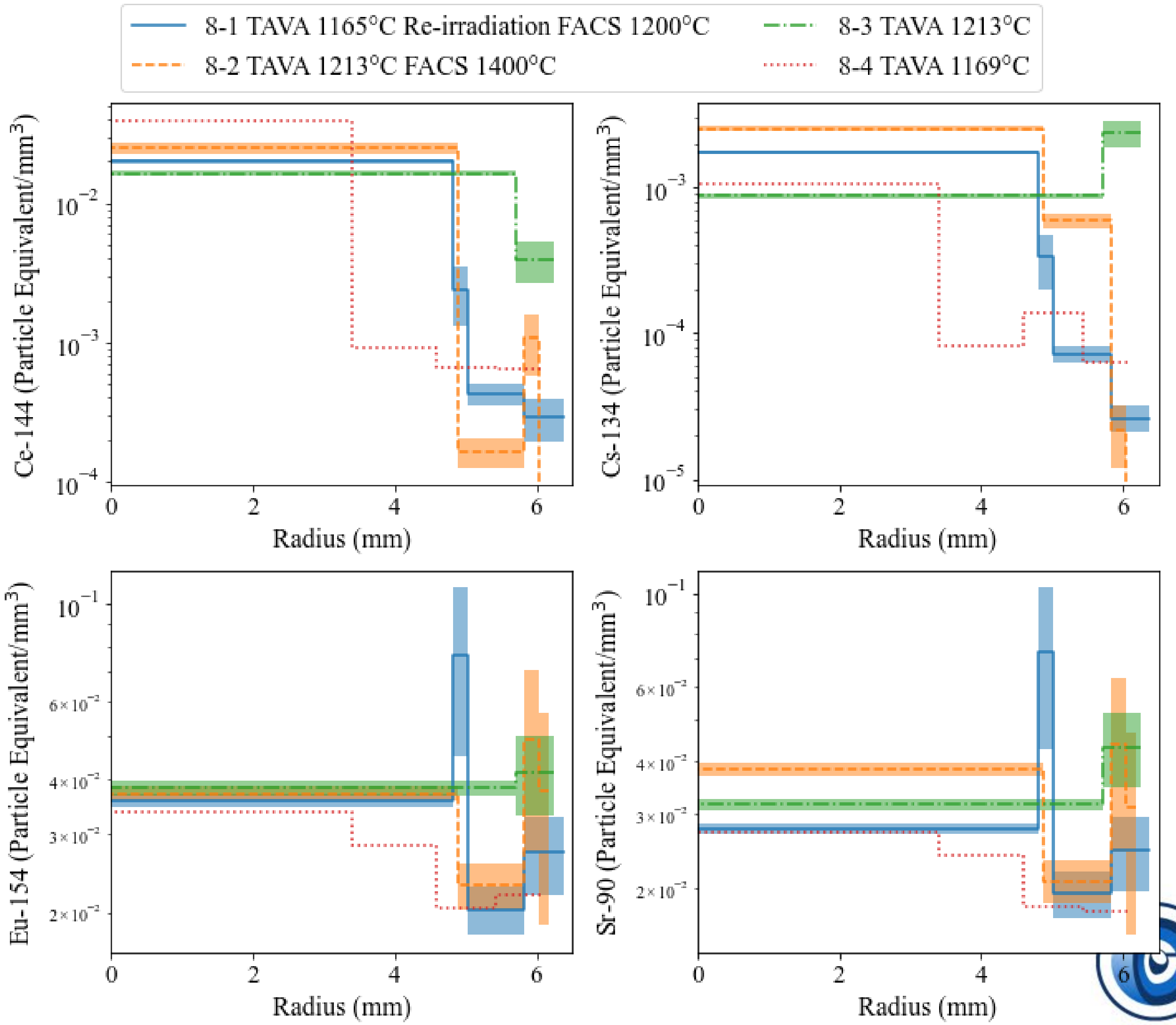


Compact	Burnup (% FIMA)	Neutron Fluence (10^{25} n/m^2, E>0.18 MeV)	TAVA Temperature (°C)	FACS Temp (°C)
8-1*	14.5	5.13	1165	1200**
8-2	14.6	5.11	1213	1400
8-3	14.5	5.07	1213	N/A
8-4	14.4	5.02	1169	N/A

* - NRAD reirradiation
** - 300 h at 1200°C, then three rounds of temperature cycling

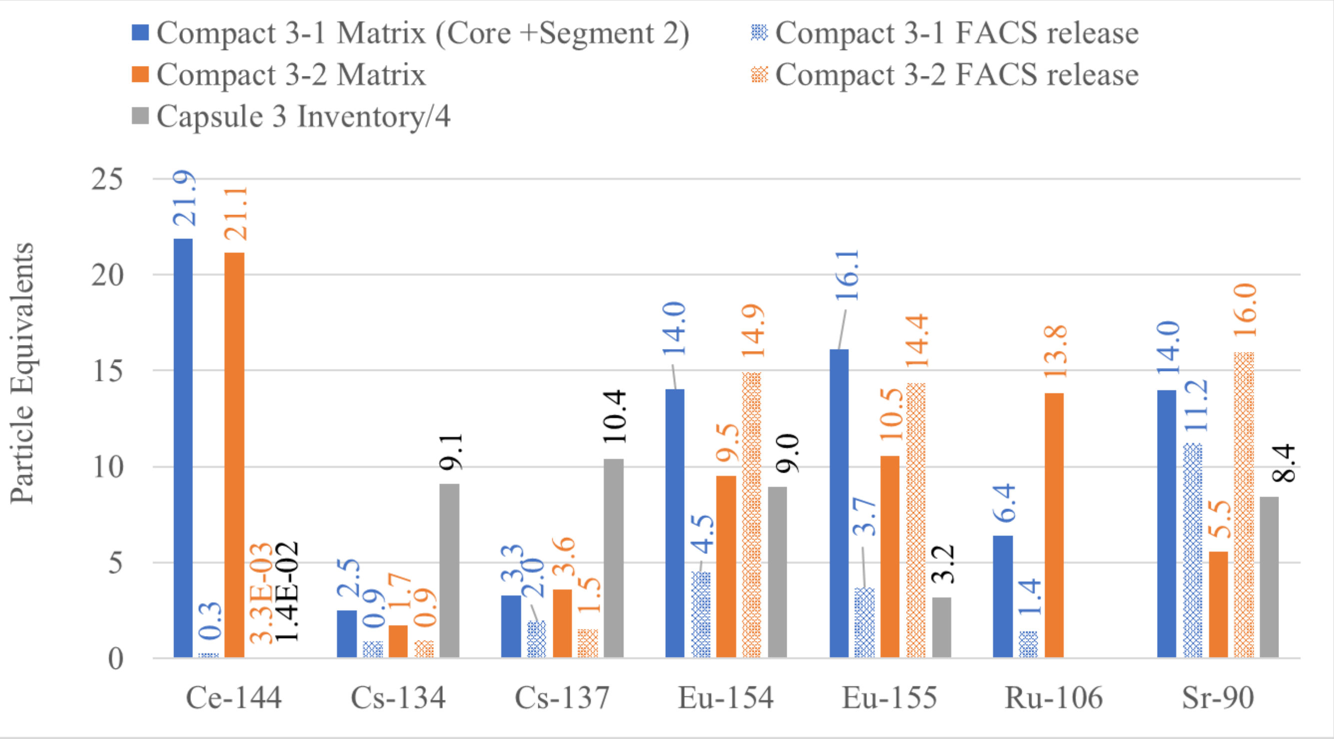
Capsule 8: As-irradiated vs. FACS-tested

- The Ce-144 concentrations outside of the core are lowest for the FACS-tested compacts, suggesting some small FACS release occurred.
- The concentration of Cs-134 was similar except at the outermost segments, which are the lowest for the FACS-tested compacts.
- The core concentration of Eu-154 was similar among the four compacts. Compact 8-1 has higher concentration than 8-4, indicating some releases through intact SiC.
- The general trend of Sr-90 profile followed closely as that of Eu-154.



Capsule 3: As-irradiated vs. FACS-tested

- Cs release is similar between two FACS tests. Cs release is from matrix/OpyC inventory that remained from the DTF particles are irradiation.
- It is unclear if Cs was under recovered from RDLBL or the mass balance (the capsule shell was not measured)
- More Eu and Sr released for 1600/1700°C safety test.



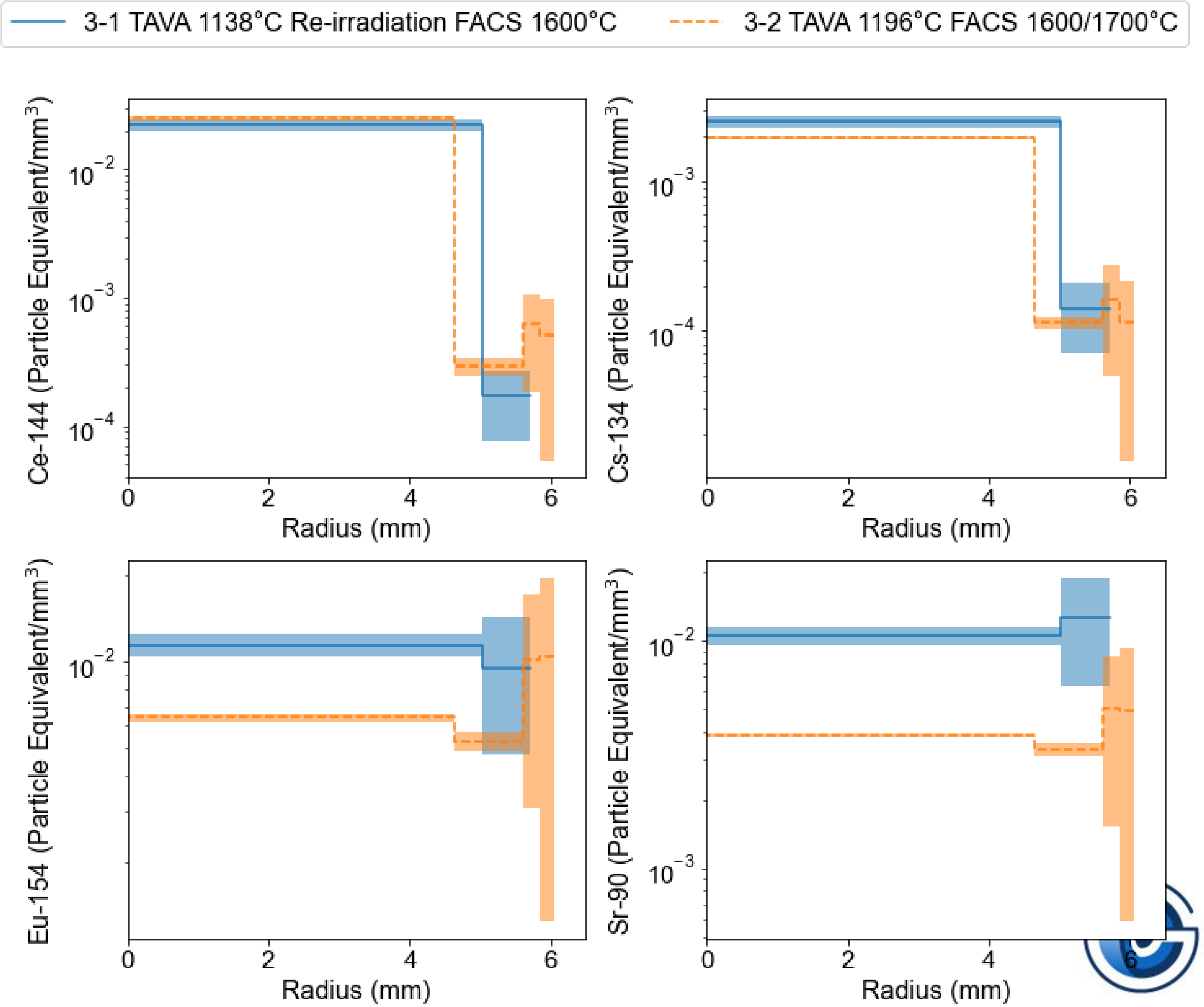
Compact	Burnup (% FIMA)	Neutron Fluence (10^{25} n/m^2, E>0.18 MeV)	TAVA Temperature (°C)	FACS Temp (°C)
3-1*	12.16	4.04	1138	1600
3-2	12.49	4.17	1196	1600/1700**

* - NRAD reirradiation

** - After the initial isothermal hold at 1600°C for 300 h, the temperature was raised to 1700°C for 48 h.

Capsule 3: As-irradiated vs. FACS-tested

- Segment 1 of Compact 3-1 were discarded due to multiple particles damaged from RDLBL.
- The concentration profiles of Ce-144 and Cs-134 were similar.
- The lower core concentration of Eu-154 and Sr-90 of Compact 3-1 compared to 3-2 indicated that Eu and Sr diffused from the core to the outer segment to release outside the compact at higher safety testing temperature.



Summary and Conclusion

Summary and Conclusion

- The RDLBL technique proved challenging to implement and employ in a hot-cell environment.
- Irradiation temperature was found to affect the inventories and radial distributions of fission products in the AGR-3/4 compacts.
  - Ce-144 and Ru-106:
    - Not significantly impact the total inventories but may affect radial concentration profiles.
  - Eu and Sr:
    - TAVA 846 - 1050°C, no discernable temperature dependence and very limited radial transport;
    - TAVA 1050 - 1169°C, significant release through intact driver particles with retention within the compact. Radial concentration decreasing with increasing radius;
    - TAVA 1213 – 1376°C, increased diffusive release through intact coatings and the release from the compact. Radial concentration flat or even increasing with increasing radius.
  - Cs:
    - TAVA 959-989°C and above, Cs being driven out compacts.
    - TAVA < 959°C, significant Cs (up to ~70%) from the DTF particles may be retained in the compact
- The data collected here will be used in comparisons to a detailed fission-product transport model of the AGR-3/4 experiment

Summary and Conclusion

Tasks to Completion of AGR-3/4

- Wrap up the safety test results for as-irradiated compacts
- Understanding reirradiation results, especially the I and Xe results (Re-irradiated safety tests)
- Wrap up RDLBL results for safety-tested and re-irradiated compacts
- Refine fission-product transport model (as-irradiated and safety-test)

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Thank you

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U.S. DEPARTMENT OF ENERGY

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The DOE-ART Graphite R&D program is the primary nuclear graphite research program for the USA. This program focuses on research and development activities necessary to qualify and license graphite components for use within nuclear applications, specifically within advanced reactor designs such as High Temperature Reactor designs. The data generated within the ART Graphite program is intended to be used in conjunction with other publicly available nuclear graphite data such as is contained within the IAEA Nuclear Graphite Knowledge Base. The ART Graphite program is divided into 5 primary research areas providing a combination of data, analysis reports, and pertinent references to describe and explain the trends within the data.