SOLVING ENERGY CHALLENGES THROUGH SCIENCE

VTR Fuel Cycle & Waste Management

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National Academy of Sciences

Note: Information regarding site location is **preliminary**. The decision for site location is determined via DOE acquisition processes that have not yet been completed.

Idaho National Laboratory

Content

- VTR Fuel Performance Design Basis
- VTR Fuel Production Process
- VTR Fuel Cycle Backend

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VTR Fuel Performance Design Basis

VTR Driver Fuel Performance Basis

	Experience	Significance
U-10Zr, U-5Fs **Driver Fuel Operation**	• ~13,000 U-Zr rods in 316SS 10 at.% bu • ≥ 120,000 U-Fs rods in 304LSS/316SS 1-8 at.% bu	• Established reliability of the fuel design and fabrication process for nominal reactor operating conditions • Sufficient numbers to capture manufacturing variation
U-10Zr **Through Qualification**	• U-Zr in 316SS, D9, HT9 ≥ 10 at.% bu in EBR-II & FFTF	• Established that specific designs operate as designed under design-basis conditions • Established that the fuel fabrication processes produce fuel that meets the specification • (Did not capture manufacturing variation)
U-Pu-Zr **Burnup Capability & Experiments** **Safety & Operability testing**	• 600 U-Pu-Zr rods; D9 & HT9 to > 10 - 19 at.% in EBR-II & FFTF • 6 RBCB tests U-Fs & U-Pu-Zr/U-Zr(5) • 6 TREAT tests U-Fs in 316SS (9rods) & U-Zr/U-Pu-Zr in D9/HT9 (6 rods)	• Extended knowledge of metallic fuel phenomena • Established capability for specific design features • Design limits identified • Did not capture manufacturing variation, and did not qualify the design for operation under design basis conditions

- We know:
  - Fuel degradation and failure mechanisms
  - Reliability demonstrated for relevant designs with variation of multiple batches and lots
  - Pu impacts on degradation and failure mechanisms
  - Key design and operating limits, or how to otherwise ensure margin
  - Fuel design parameters that provide ample safety and reliability margin

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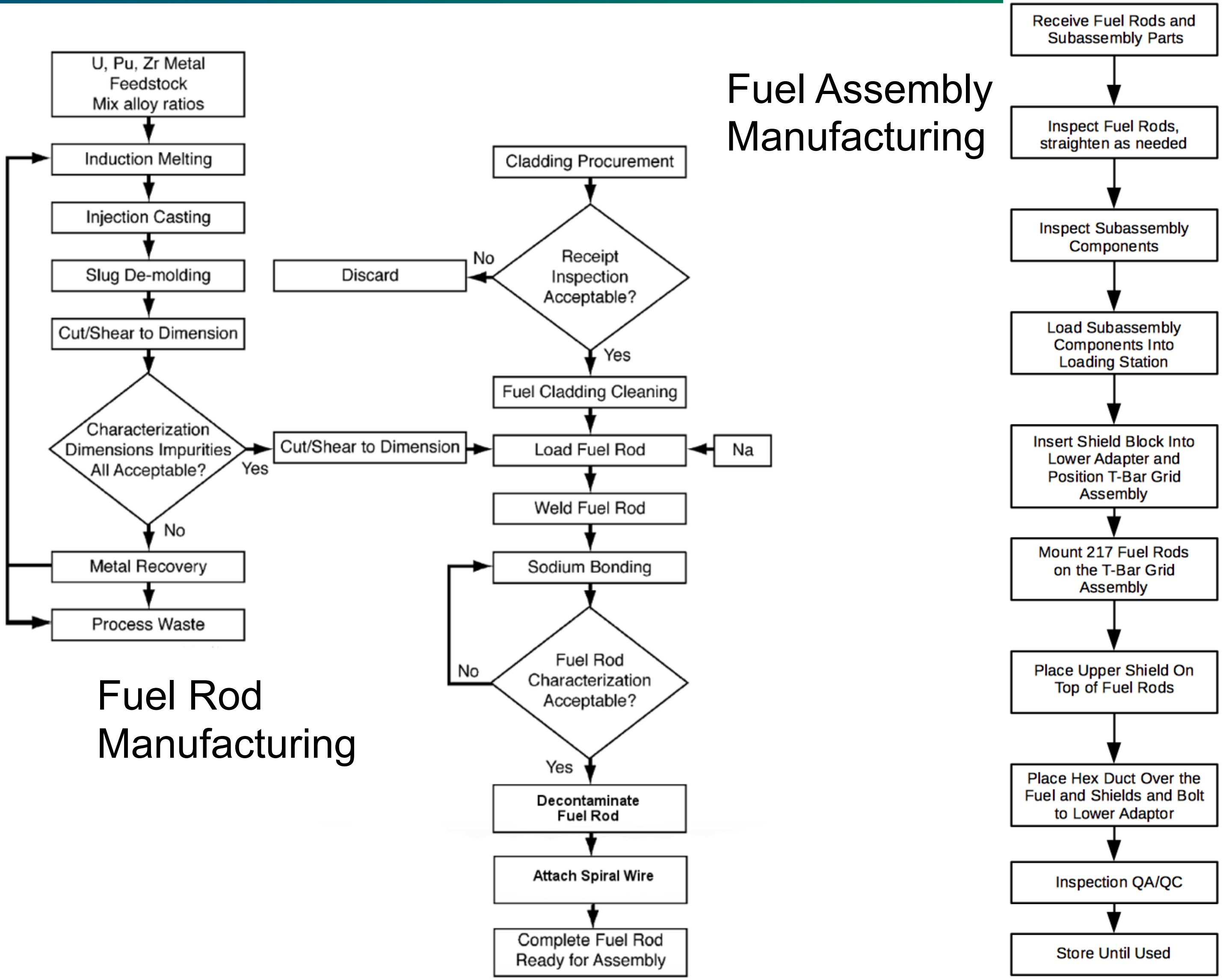
VTR Fuel Production Process and Basis

Reference VTR Fuel Production Process

- Adaptation of EBR-II metal fuel process to larger scale and Pu alloy
  - EBR-II U-10Zr:
    - ~20-25 assemblies/yr
    - ~4.5 kgHM/assembly
  - VTR U-Pu-Zr:
    - 45 assemblies/yr
    - 40 kgHM/ assembly
  - For U-Pu-Zr, requires glovebox line and CAT-I facility
- "Hands-on", except as modified to reduce personnel radiation exposure
- Vacuum injection casting
  - Previously deployed in 6 different locations



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VTR Fuel Cycle Backend

VTR Fuel Cycle Backend Overview

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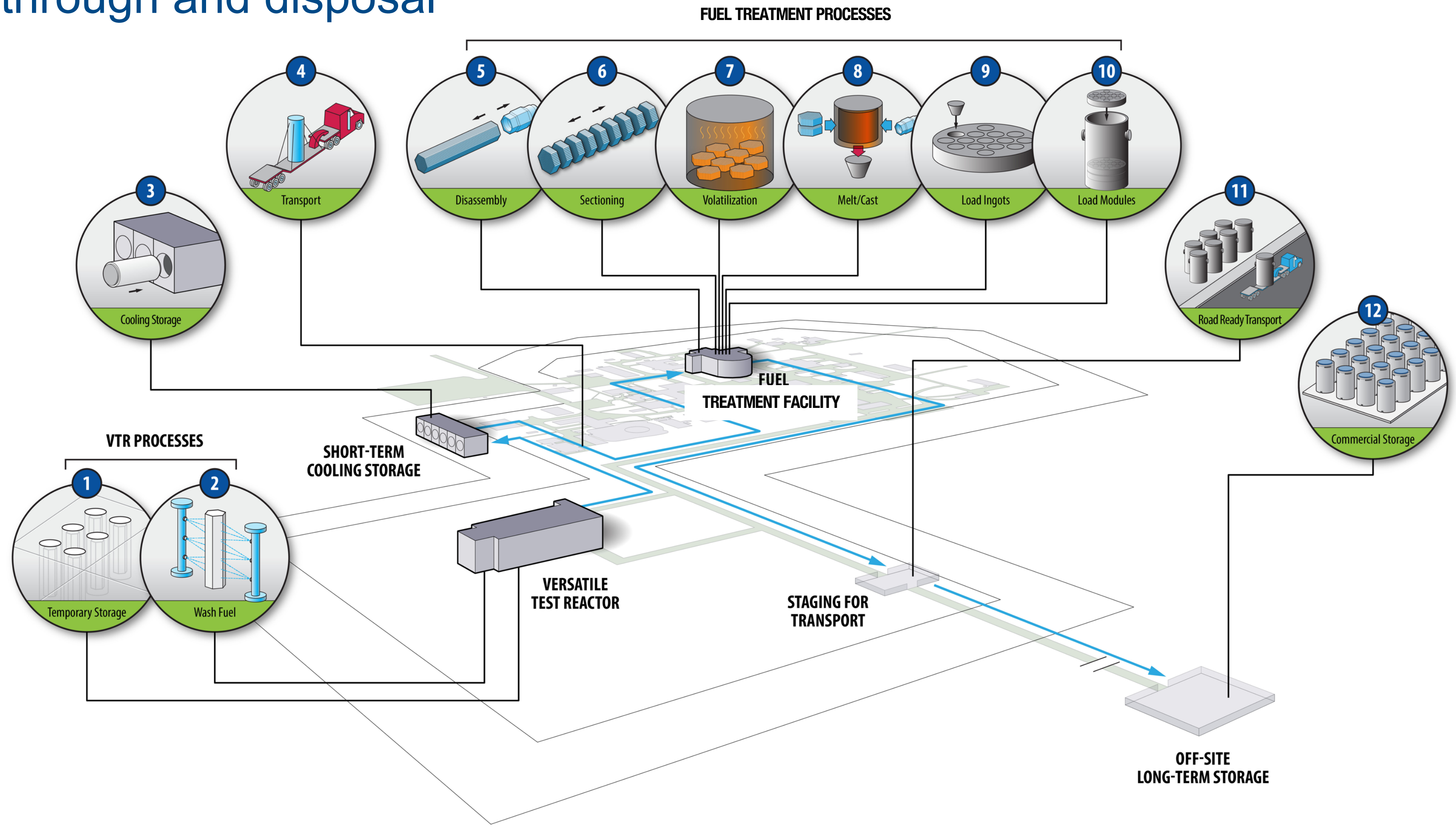
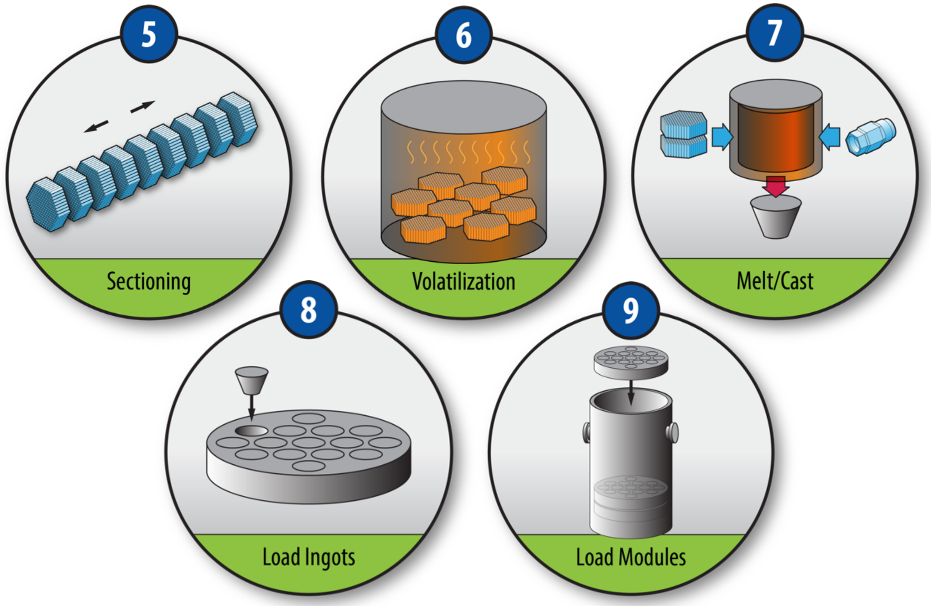
Objective: once-through and disposal



VTR Fuel Treatment – INL Treatment Alternative

- VTR fuel treatment objective is simple: prepare spent fuel for disposal
- Assumes Na-bonded VTR spent fuel will not be suitable for direct disposal
  - Fuel design w/o Na bond is proposed as the advanced VTR fuel design (VTR)
- FCF electrometallurgical treatment equipment has insufficient capacity for the VTR throughput rate and plutonium content
- Proposed: simple melt-distill-dilute process
  - Different flowsheet approach than used in FCF
  - Section and melt the fuel assembly (incl. hardware)
  - Distill Na from the melt
  - Some developmental testing needed
  - Would require a new distillation furnace scaled-up from those used in fuel cycle development program



VTR Fuel Treatment Process Basis

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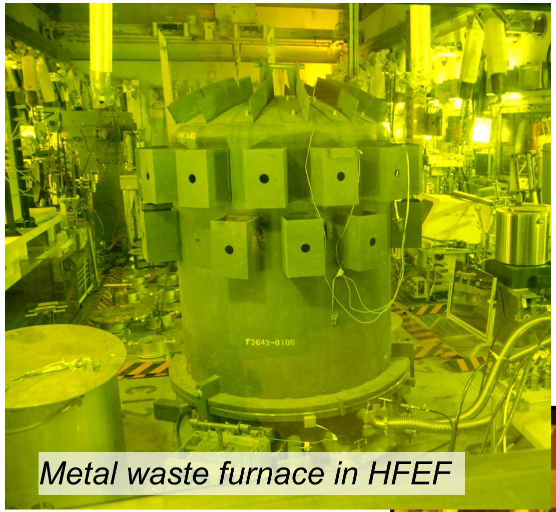
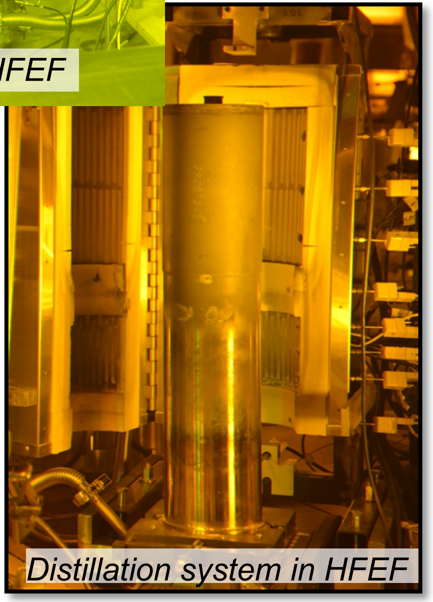
- Recent and increasing experience with distillation
  - Salt distillation from EBR-II spent fuel cladding hulls
  - Preparing non-radioactive demonstration of Na distillation from low-burnup Fermi 1 fuel
- Dilution of Pu in waste ingot to < 10 wt.% renders that material as Safeguards and Security Attractiveness Level D (per DOE-STD-1194-2011)
  - Mass of fuel assembly hardware should be sufficient to dilute Pu content to < 10 wt.%



Metal waste furnace in HFEF

Estimated Plutonium Isotopic Composition at Discharge		
	Charge	Discharge
Pu-236	0.00%	0.00%
Pu-238	0.00%	0.01%
Pu-239	94.00%	89.33%
Pu-240	5.00%	9.50%
Pu-241	1.00%	1.10%
Pu-242	0.00%	0.05%

- Fuel characteristics upon discharge (indicative)
  - Average discharge burnup ~ 53 GWd/t
  - $^{240}Pu/^{239}Pu \geq 10.6\%$



Distillation system in HFEF

Anticipated VTR Fuel Waste

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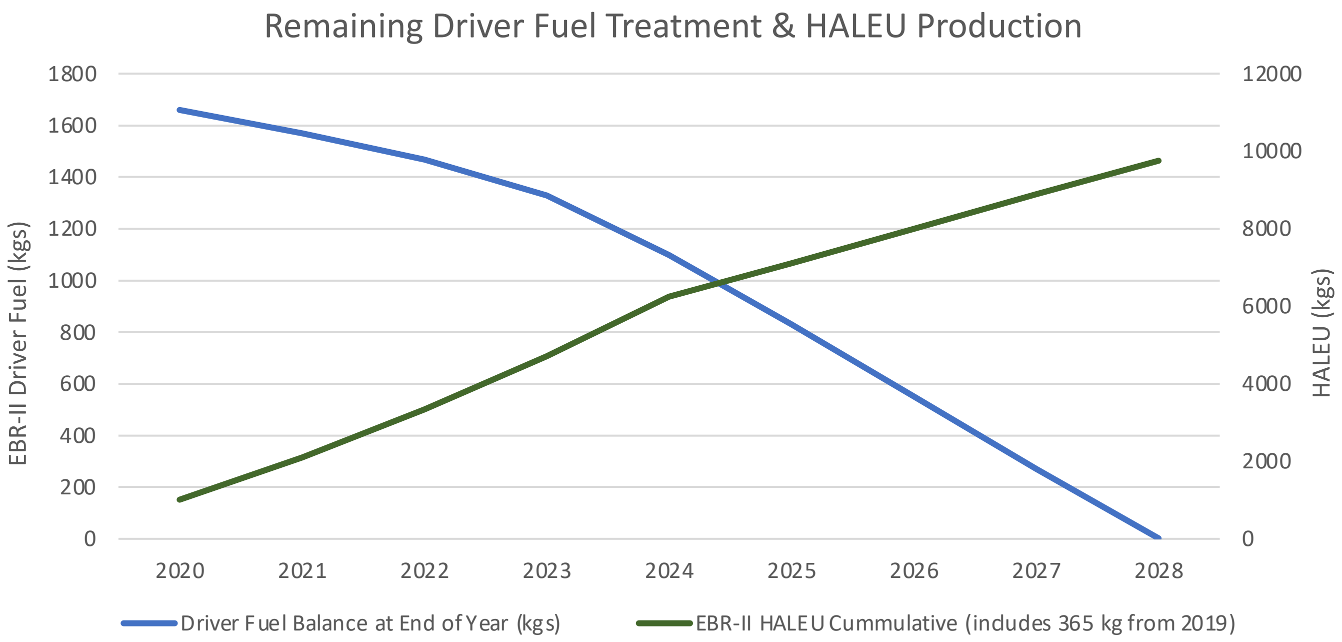
- Consolidated fuel assembly ingots (fuel constituents, non-volatile fission products, hardware)
- Contaminated sodium hydroxide and condensable volatile fission products
- Non-recyclable fuel production scrap (slag and unusable U-Pu-Zr; quartz mold waste; crucible waste)

VTR Reactor and Fuel Waste Streams (draft estimate from input to VTR EIS)

Waste Type	Quantity	Disposition Path (for INL alternative)
LLW	540 m^3/yr	RWMC, RHLLW
MLLW	38 m^3/yr	Nev. Nat. Sect. Site or off-site commercial
TRU	0.89 m^3/yr	WIPP (if defense-related source; ~27 TRUPACT-II/yr)
RCRA & TSCA	7.2 m^3/yr	Off-site commercial
Spent Fuel	1.8 MTHM/yr	HLW interim storage and repository TBD

EBR-II Driver Fuel Treatment Status

Fuel Type	Beginning Inventory	Amount Treated as of 2019	Remaining Inventory
	kgHM	kgHM	kgHM
U-10Zr (incl FFTF)	1,303	1053 (81%)	250
U-5Fs	1,849	467 (25%)	1,382
Experimental Fuels	141	0	141
Totals	3293	1,520 (46%)	1,773



November 2019 Supplemental Agreement to the 1995 Settlement Agreement – requirements:

- Commence treatment w/in 30 days of Supplemental Agreement
- Treat at least 165 pounds (75 kg) heavy metal of Sodium Bonded EBR II Driver Fuel pins per year on a three-year rolling average basis
- Complete treatment of all Sodium Bonded EBR II Driver Fuel Pins by Dec 31, 2028
- Non-HLW waste materials, including fuel pin cladding material, disposed outside of the State of Idaho by Jan 1, 2035
- Any HLW generated during treatment shall be put into a form suitable for transport to a permanent repository or interim storage facility outside of the State of Idaho by a target date of Dec 31, 2035
- Any treated product material not put to beneficial use to be removed from the State of Idaho by Jan 1, 2035. (e.g., HALEU)

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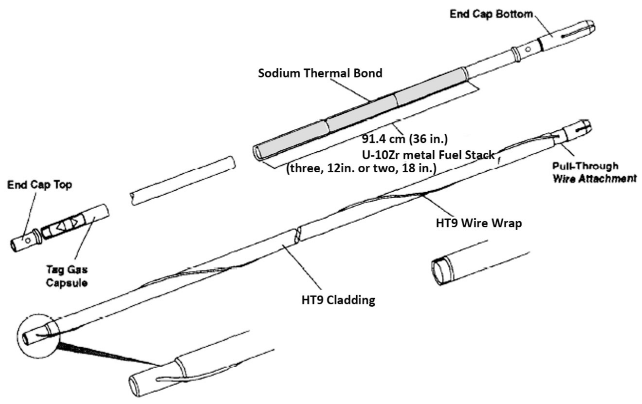
Backup

VTR Driver Fuel Design Basis

- Based upon design parameters with demonstrated fuel performance and reliability in EBR-II and FFTF
- VTR design adjustments place fuel in comparable parameter space
  - Fuel smeared density
  - Cladding r/t ratio
  - Plenum-to-fuel vol. ratio



A. L. Pitner and R. B. Baker, "Metal Fuel Test Program in FFTF," *Journal of Nuclear Materials*, 204 (1993) 124-130.

Table 9
Suggested reference design parameters for mixed oxide (MOX) fuel, U–Pu–Zr fuel, and mixed carbide (MC) fuel based on US experience

Parameter	Mixed oxide (MOX)[a]	U–Pu–Zr[b]	Mixed carbide (MC)[c]
Nominal composition	$(U,Pu)O_2$	U–20Pu–10Zr	(U,Pu)C
Pu/(U + Pu) range	22–30%	17–28%	21–23%
Oxygen-to-metal ratio	1.95	n/a	
Fuel theoretical density	92%	100%[d]	80–82%
Fuel smeared density (% TD)	80–85%[e]	75%	78–79%
Plenum-to-fuel volume ratio	1.0	1.4	1.0
Fuel height	91 cm[f]	91 cm[f]	91 cm
Fuel outer diameter, as-fabricated	0.56 cm[g]	0.5 cm[g]	TBD[g]
Fuel inner diameter, as-fabricated	0.15 cm[g]	n/a	n/a
Fuel-cladding bond	He	Na	He
Cladding material	HT9[h] or 20% cw 316SS	HT9[h] or 20% cw 316SS	HT9[h] or 20% cw 316SS
Cladding outer diameter[f]	0.69 cm	0.69 cm	0.94 cm
Cladding inner diameter[f]	0.57 cm	0.57 cm	TBD[g]
Peak linear heat generation rate	44–46 kW/m	49–52 kW/m	66–80 kW/m
Peak inner-wall cladding temperature, nominal	620 °C	620 °C[i]	620 °C
Duct material	HT9[h] or 20% cw 316SS	HT9[h] or 20% cw 316SS	HT9[h] or 20% cw 316SS

D. C. Crawford, D. L. Porter, S. L. Hayes, "Fuels for Sodium-cooled Fast Reactors: U.S. Perspective," *Journal of Nuclear Materials*, 371(2007) 202-231.

VTR: Fuel Manufacturing Experience

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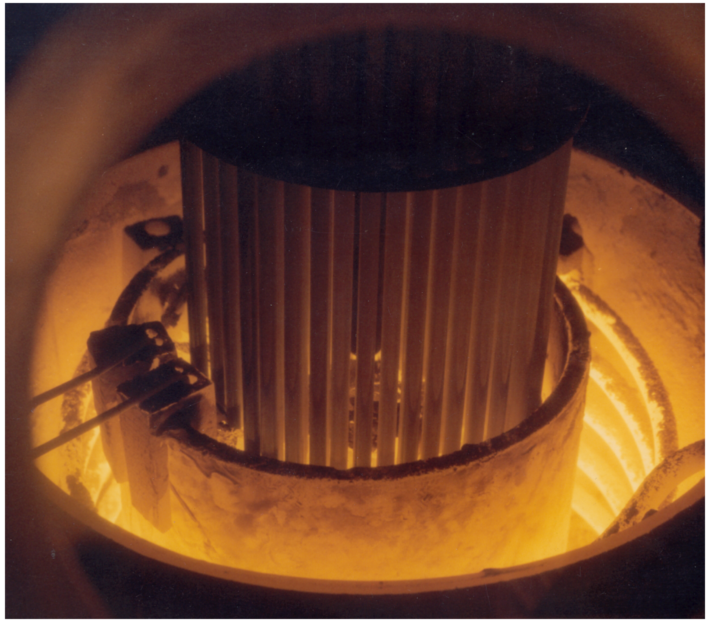

- Metal fuel fabrication compared to oxide fuel:
  - Allows use of simpler injection-casting fabrication technique
  - Smaller footprint lowers cost and allows use of existing facilities
  - Handling 2 to 3 metal slugs per fuel rod, rather than 100s of pellets
  - Allows greater chemistry and dimensional tolerance
- INL/ANL manufactured:
  - Over 133,000 U-bearing metal fuel rods for EBR II
  - Over 600 U-Pu-Zr metal fuel rods for EBR II and FFTF
  - 90% casting yield when fabrication terminated
- EBR-II fuel fabrication process replicated in 6 locations
  - ANL-East coldline, first core, 1960-1961
  - ANL-West hotline in FCF, 1964-1969
  - ANL-West coldline in FCF-20 & FASB, 1967-1986; ANL-West coldline in FMF, 1986-1994
  - Aerojet Nuclear, 1967-1971
  - Atomics International, 1973-1976, 1985
- Sufficient batches and cladding and duct lots to reflect manufacturing variation

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