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Carbide and Nitride Fuels for Advanced Burner Reactor

Taek K. Kim, C. Grandy, and R. N. Hill
Argonne National Laboratory

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Content

- **Background**
  - Background of Advanced Burner Reactor (ABR)
  - Fast fuel fuel options
  - Fast fuel experiences in U.S.
- **Reference ABR-1000 Reactor Concepts**
  - Design objective and constraints
  - Comparison of equilibrium core performance characteristics
  - Comparison of passive safety features
- **Conclusions**

Background

- **Advanced Burner Reactor (ABR) has been developed under GNEP program**
  - Design constraints and fuel forms (metal and oxide) were determined from fast reactor experiences of U.S.
- **Design Activities**
  - Reference concept with1000 MWt power rating using metallic and oxide fuels
  - Various conversion ratio (0.25 – 1.0) in same core layout
  - TRU (MA) target in heterogeneous core concept
  - Design innovations (FP vented fuel, advanced shielding materials, nanofluid for advanced coolant, etc)
  - Alternative core concepts studies
  - Alternative fuel form assessment
    - *Carbide and nitride fuels with expectation that high density, high melting temperature, and excellent thermal conductivity can mitigate disadvantages of both metallic and oxide fuels*
    - *Potential problems of carbide and nitride fuels in reprocessing and irradiation were not considered in this study*

GNEP Fuel Cycle Strategy

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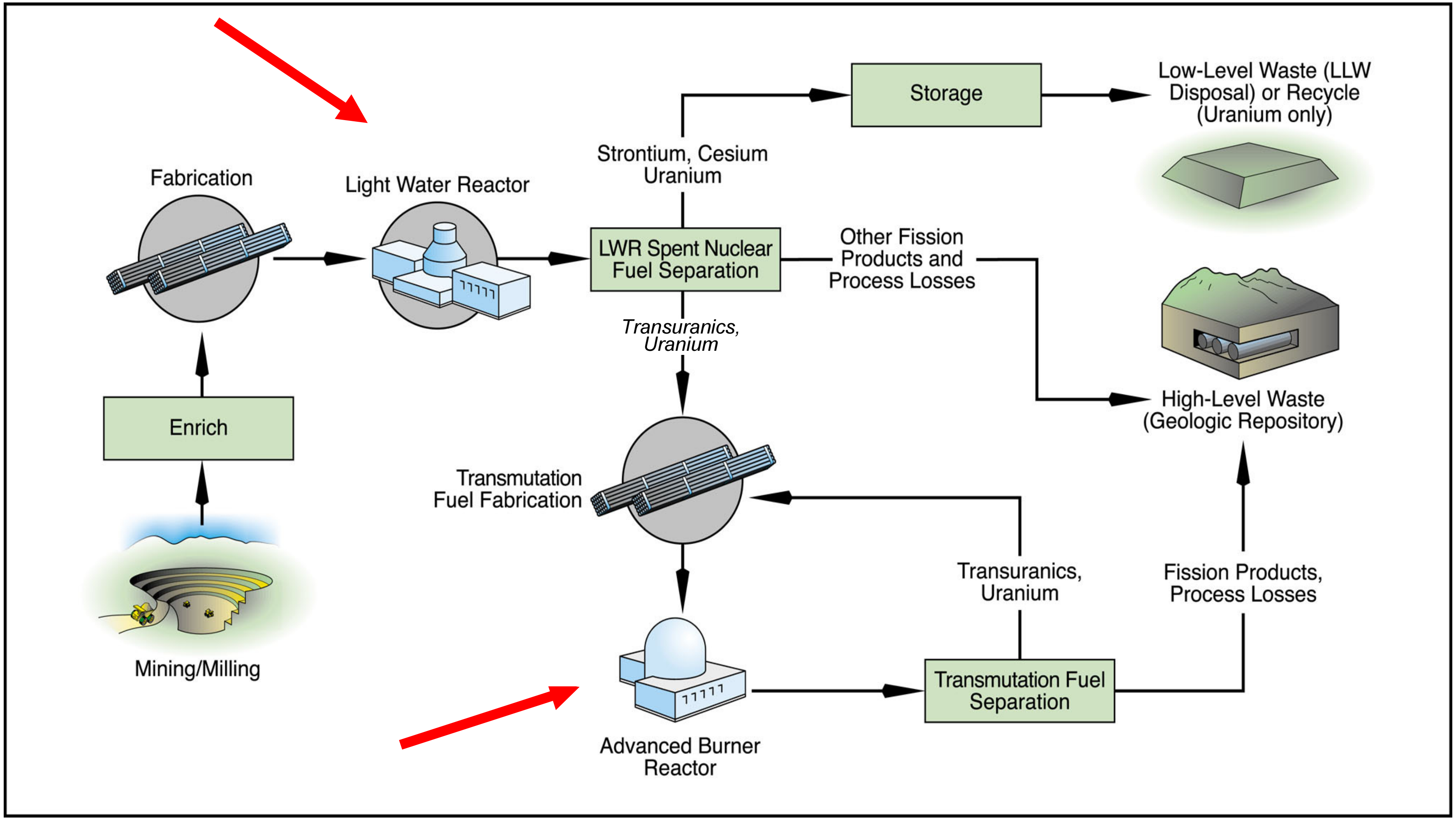

Fast Reactor Fuels and Design Issues

	Metal U-20Pu-10Zr	Oxide UO_2-20PuO_2	Nitride UN-20PuN	Carbide UC-20PuC
Heavy Metal Density, g/cm^3	14.1	**9.3**	13.1	12.4
Melting Temperature, ºK	**1350**	3000	3035*	2575
Thermal Conductivity, W/cm-ºK	0.16	**0.023**	0.26	0.20
Operating Centerline Temperature at 40 kW/m, ºK, and (T/T_{melt})	1060 (0.8)	2360 (0.8)	1000 **(0.3)**	1030 **(0.4)**
Fuel-Cladding Solidus, ºK	**650 (TRU)**	1675	1400	1390
Thermal Expansion, 1/ºK	17E-6	12E-6	10E-6	12E-6

- **Design Issues**
  - Fuel/cladding chemical interaction
  - Fuel/cladding mechanical interaction
  - Fuel/coolant compatibility
  - Fuel swelling (current metal and oxide fuel pin designs accommodate fuel swelling)

Fast Reactor Fuel Experience in U.S – Metal Fuel

- **Metal Fuel was originally selected by early fast reactors (EBR-I, EBR-II, Fermi)**
  - Easy fabrication, high thermal conductivity, high density (allows higher breeding ratio)
  - Severe burnup limitation observed, but resolved by allowing sufficient space for fuel swelling (low smeared density of 75%)
- **Zr based binary (U-Zr) and ternary (U-Pu-Zr) fuel development**
  - Alloy elements such as Mo, Al, Zr, Cr, Fs added to U or U-Pu metals to improve corrosion resistance, increase solidus temperature, and enhance dimensional stability
  - Zr was selected due to favorable irradiation testing results
- **More than 130,000 rods irradiated in EBR-II and FFTF**
  - Qualified to 10 at.%, demonstrated to 20 at.% with HT9 (or D9) cladding up to 31% Pu
  - Fuel constituent redistribution and Zr-depletion observed, which depends on fuel composition, burnup (lanthanide fission products) and temperature
    - *Ternary fuel-cladding inter-diffusion zone would melt as low as 675 °C*
- **RBCB (Run Beyond Cladding Breach)**
  - Metal fuel is compatible with sodium and does not form fuel-coolant reaction (no further break or wash out), which allows sustained operation with breached metal fuel

Fast Reactor Fuel Experience in U.S – Oxide Fuel

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- **U.S. fast reactor programs adopted oxide fuel in late 1960s and early 1970s**
  - Higher burnup than metal fuel, and synergy with experience in commercial side
  - High melting temperature, but high operating temperature due to low-thermal conductivity
- **More than 50,000 rods irradiated in EBR-II and FFTF**
  - Fast Flux Test Facility (400 MWt) operated with oxide from 1980 to 1992
  - Qualified to 15 - 20 at.% with HT9 cladding up to 33% Pu
  - Fuel restructuring, FCMI, FCCI increase with temperature and burnup
    - *FCCI could be reduced by allowing porosity at fabrication (low smeared density of 80 – 85% TD)*
    - *Due to different swelling speed, lifetime of cladding materials (HT9 or D9) limited by FCMI*
    - *Irradiated till peak pellet burnup of 238 GWd/t and peak fast neutron fluence of 3.9×10^{23} n/cm^2*
- **RBCB (Run Beyond Cladding Breach)**
  - Oxide fuel chemically reacts with sodium, which further opens fuel breaches, raises fuel temperature, and causes secondary breaches
  - Stricter limits on fuel pin failures to prevent potential flow blockages

Fast Reactor Fuel Experience in U.S – Carbide/Nitride Fuels

- **Feasibility studies were performed under Advanced Fuel Program of USDOE in 1970s**
  - Compared to oxide fuel, carbide/nitride fuels have high density, high melting temperature, and excellent thermal conductivity
  - Focused on high breeding and short doubling time
    - *A. SHETH, et al, "Equation-of-State for Advanced Fuels," ANL-AFP-2, Argonne National Laboratory (1974)*
    - *A. SHETH, et al, "Thermal Conductivity Values for Advanced Fuels," ANL-AFP-3, Argonne National Laboratory (1974)*
    - *M. TETENBAUM, et al, "A Review of the Thermodynamics of the U-C, Pu-C, and U-Pu-C Systems," ANL-AFP-8, Argonne National Laboratory (1975).*
    - *W. P. BARTHOLD, "The Breeding Performance of Carbide and Nitride Fuels in 2000 MWe LMFBRs," ANL-AFP-30, Argonne National Laboratory (1976)*
    - *J. F. de PAZ, et al, "Engineering Analysis of Mixed Carbide Fuels for Large Breeder Reactors," ANL-AFP-46, Argonne National Laboratory (1978)*
    - *Etc*
  - Limited irradiation experiences

ABR-1000 Core Concept

- **Capability to demonstrate transmutation, cost reduction, safety characteristics, and qualify fuels and materials (AFCI goals)**
  - Moderately low TRU conversion ratio (~0.7)
  - 1000 MWth compact core and higher power density
  - Sufficient reactivity control mechanism, appropriate reactivity feedback coefficients, and passive safety features
- **Capable of reactor licensing**
  - Use conventional or proven materials/technologies; otherwise use conservative data
- **Flexibility**
  - Allow to interchange between different fuel forms in same core layout
  - Allow variable conversion ratios from low (~0.25) to break-even (~1.0)
  - Allow minor actinide target bed for heterogeneous recycling

Design Constraints

- **Fuel**
  - Possible to use either ternary metallic, oxide (carbide, or nitride)
  - Fuel smeared density is determined to accommodate fuel swelling
  - Maximum TRU enrichment is generally less than 30%
  - Maximize discharge burnup within current irradiation experiences
- **Core structural and cladding material: HT9**
  - Peak fast flux fluence of structural material < 4.0 x 10^{23} n/cm^2
  - Peak 2-sigma cladding inner surface temperature is less than eutectic temperature
- **Others**
  - Peak 2-sigma fuel center line temperature < melting temperature
  - Reduce peak control assembly worth less than 0.7$ or further for passive safety features
  - Coolant outlet temperature: 510° C; mean coolant delta-T: 155 ° C

Computation Methods

- **Equilibrium neutronics calculations**
  - ETOE-2/MC^2-2 for broad-group cross sections based on ENDF/B-V (or VII)
  - DIF3D/REBUS3 for fuel cycle analysis using 3D hexagonal-Z core model (diffusion or transport)
  - VARI3D (first/exact perturbation) code for kinetics parameters and reactivity feedback coefficients
- **Steady-state thermal hydraulic calculations**
  - Single-channel analysis for scoping calculations
  - SE2-ANL (sub-channel analysis) code was employed to check imposed thermal design criteria
- **Passive Safety Feature were investigated using integral parameters for Quasi-State Reactivity Balance**
  - D. C. Wade and E. Fujita, "Trends Versus Reactor Size of Passive Reactivity Shutdown and Control Performances," NSE 103, 182 (1989)
  - Detailed transient analyses performed separately (reference concept only)

ABR-1000 Core Layout

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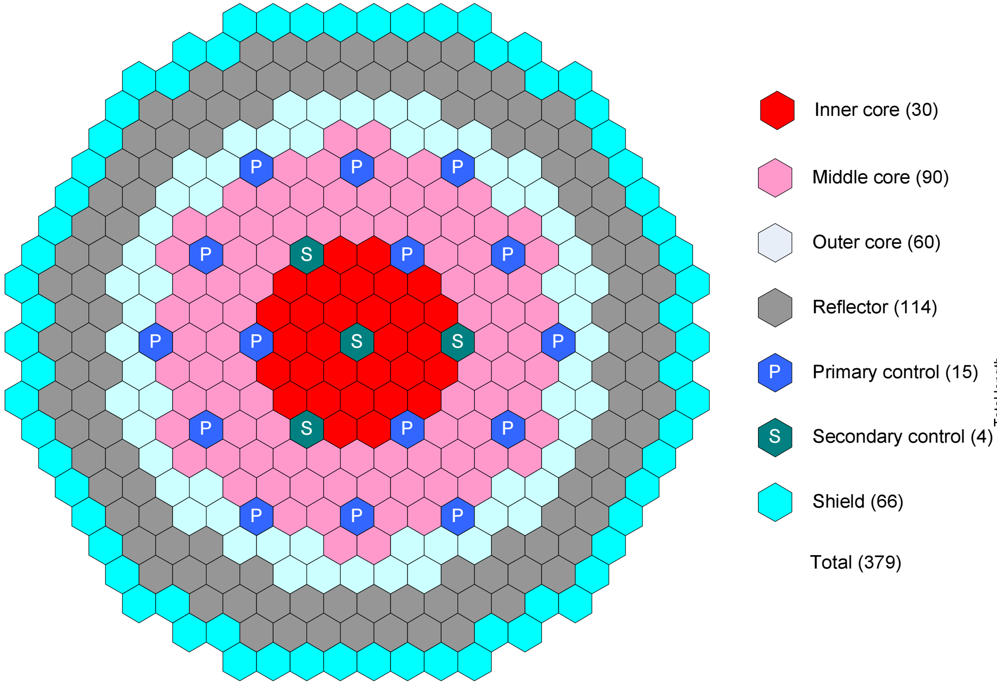
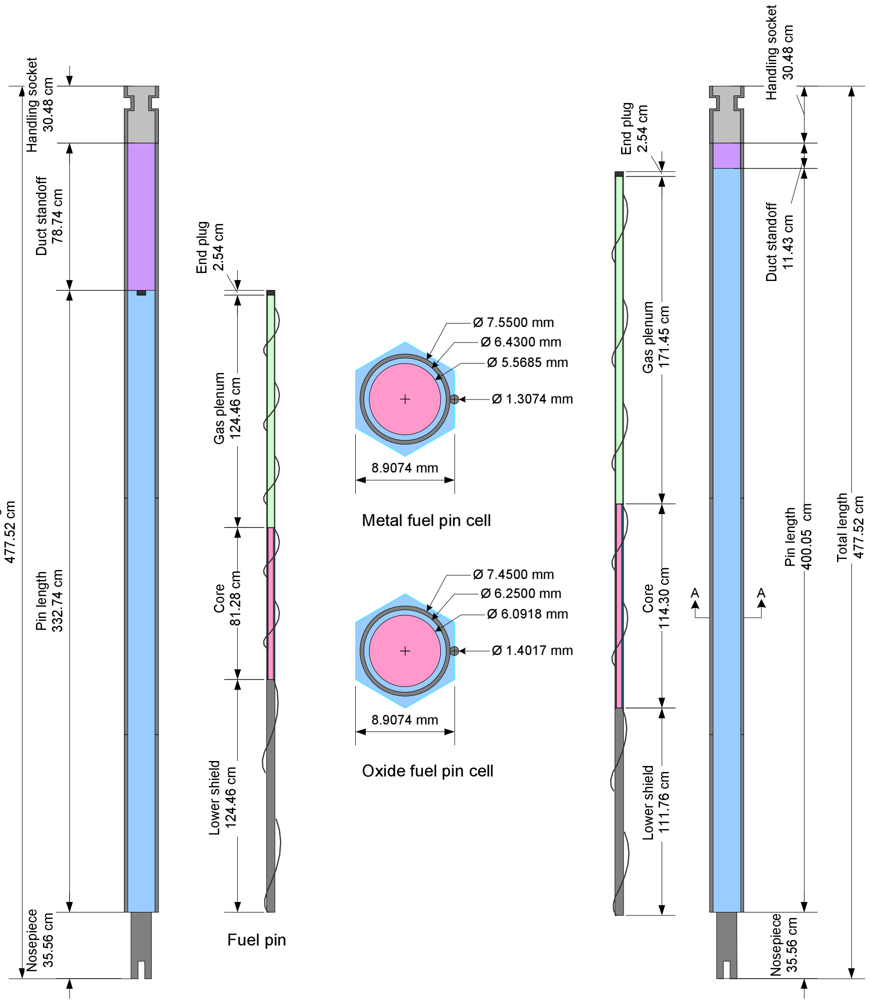


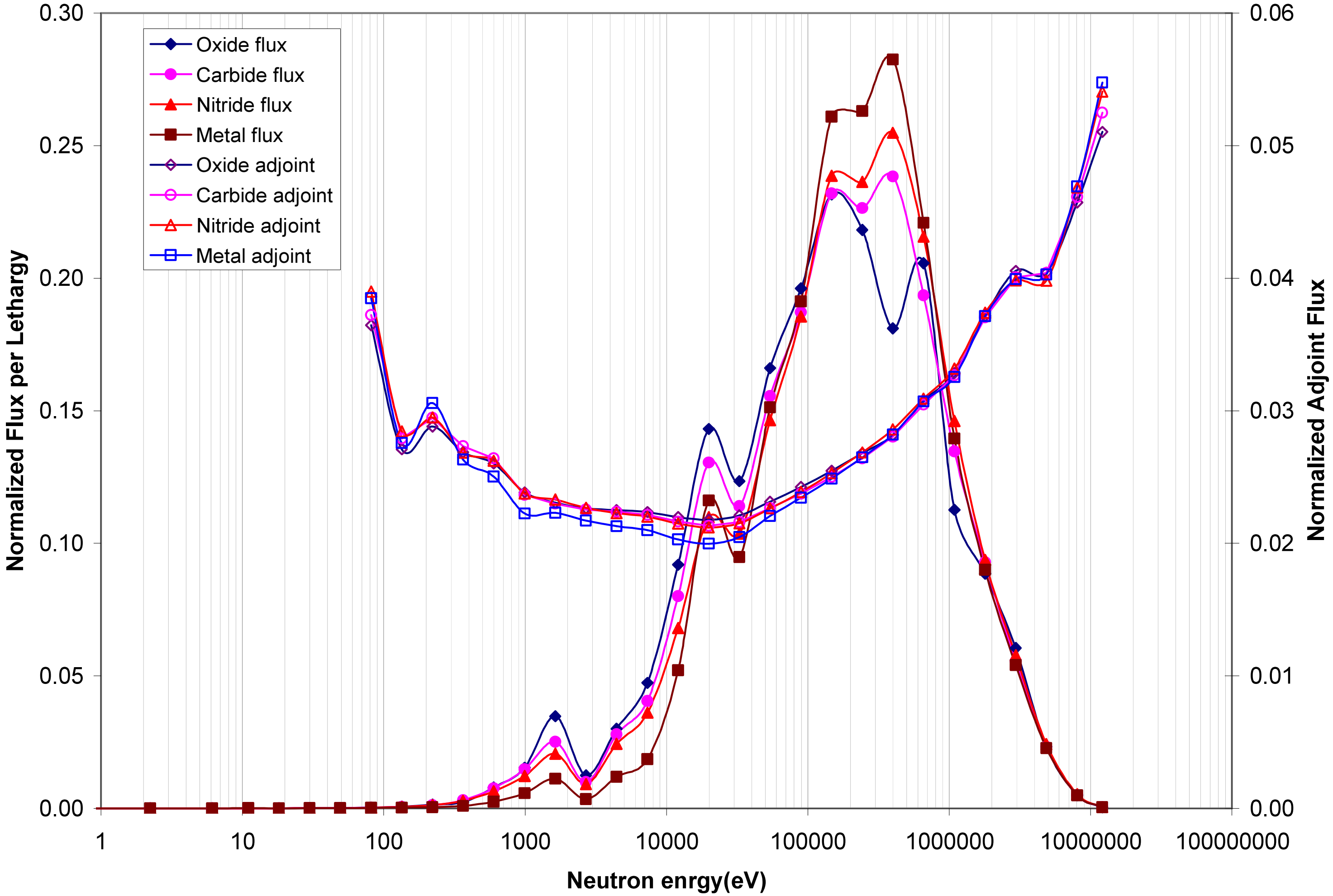


Design Parameters

	Oxide	Carbide	Nitride	Metallic
Number of fuel pins per assembly	271	271	271	271
Assembly pitch, cm	16.14	16.14	16.14	16.14
Assembly height, cm	478	478	478	478
Active core height, cm	114.3	111.8	94.0	81.3
Fuel smeared density, % TD	85.0	75.0	85.0	75.0
Fuel slug/pellet diameter, cm	0.643	0.609	0.609	0.557
Cladding outer diameter, cm	0.772	0.745	0.745	0.755
Pin pitch-to-diameter ratio	1.155	1.195	1.195	1.180
Cladding thickness, cm	0.056	0.060	0.060	0.056
Bond material	He	Na or He	Na or He	Na
Fuel Volume fraction, %	39.0	35.0	35.0	29.2

Number of fuel pins per assembly and outer assembly dimensions are identical to reference ABR for possible use of any fuels within same core layout, but fuel pin design parameters were iteratively determined for maximizing discharge burnup within fast flux fluence limit of HT9 cladding and produce target TRU conversion ratio of ~0.7

Comparison of Neutron Spectrum



Equilibrium Core Performance Characteristics

	Oxide	Carbide	Nitride	Metallic
Cycle length, month	14	12	12	12
Number of batches	5	5	5	4
Average TRU enrichment, %	25.4	24.8	23.0	22.1
Ave. power density at cold state, kW/l	215	231	262	303
Fissile/TRU conversion ratio	0.86 / 0.74	0.86 / 0.74	0.86 / 0.75	0.84 / 0.73
HM/TRU inventory at BOEC, MT	15.1 / 3.9	14.8 / 3.7	15.5 / 3.6	13.2 / 2.9
Average discharge burnup, MWd/kg	117	103	98	93
Specific power density, MW/t	63.0	64.6	61.8	73.2
Peak discharge fast fluence, $10^{23}/cm^2$	4.00	3.86	3.98	4.09
Burnup reactivity loss, %delta-k	2.0	1.7	1.7	2.2
Average linear power, kW/m	17.6	18.8	21.3	23.3
Peak linear power density, kW/m	34.2	33.3	36.2	37.2
Core average flux, $10^{15}/cm^2$-sec	2.47	2.63	2.64	3.23
Fast flux (>0.1 MeV) fraction	0.58	0.62	0.66	0.68
Margin to fuel melt, K	~ 230	> 1000	> 1000	~270
TRU consumption rate, kg/year	-77	-79	-75	-82

Reactivity Feedback Coefficients

Fuel form	Oxide	Carbide		Nitride		Metallic
Bond material	He	Na	He	Na	He	Na
Delayed neutron fraction	0.0031	0.0032	0.0032	0.0033	0.033	0.0033
Prompt neutron lifetime, μ s	0.40	0.39	0.39	0.33	0.33	0.33
Radial expansion coef., ¢/° C	-0.29	-0.32	-0.32	-0.31	-0.31	-0.38
Axial expansion coef., ¢/° C	-0.05	-0.06	-0.06	-0.05	-0.05	-0.06
Fuel density coef., ¢/° C	-0.42	-0.39	-0.39	-0.33	-0.33	-0.69
Structural density coef., ¢/° C	0.09	0.08	0.08	0.07	0.07	0.07
Sodium void worth, $	6.30	5.85	5.94	4.96	5.03	6.64
Sodium density coef., ¢/° C	0.15	0.14	0.15	0.13	0.13	0.16
Doppler constant, $	-1.53	-1.29	-1.26	-1.13	-1.10	-0.82
Doppler coefficient, ¢/° C	-0.13	-0.17	-0.13	-0.14	-0.11	-0.10

Passive Safety Measurement

Fuel form		Oxide	Carbide		Nitride		Metal
Bond material		He	Na	He	Na	He	Na
A, power coefficient, ¢		-66.0	-13.0	-36.0	-14.2	-35.2	-8.1
B, power/flow coefficient, ¢		-42.4	-48.0	-45.0	-45.9	-43.2	-55.6
C, inlet temperature coefficient, ¢/° C		-0.4	-0.5	-0.4	-0.4	-0.4	-0.5
del-rho, transient over power initiative, ¢		38	27	38	33	33	39
Sufficient conditions	A/B < 1	1.55	0.27	0.80	0.31	0.81	0.15
	1 < C-deltaT/B < 2	1.41	1.46	1.43	1.45	1.41	1.33
	D-rho/|B| at BOEC < 1	0.67	0.62	0.65	0.56	0.59	0.55

D. C. Wade and E. K. Fujita, Trends Versus Reactor Size of Passive Reactivity Shutdown and Control Performance, Nuclear Science and Engineering, Vol. 103, pp. 182-195, 1989.

The fast reactor reactivity balance can be written as follows:

$$\delta\rho = [P(t)-1]\,\mathbf{A} + [P(t)/F(t)-1]\,\mathbf{B} + [\delta T_{in}(t)]\,\mathbf{C} + \delta\rho_{external}$$

where $P(t)$ = normalized reactor power

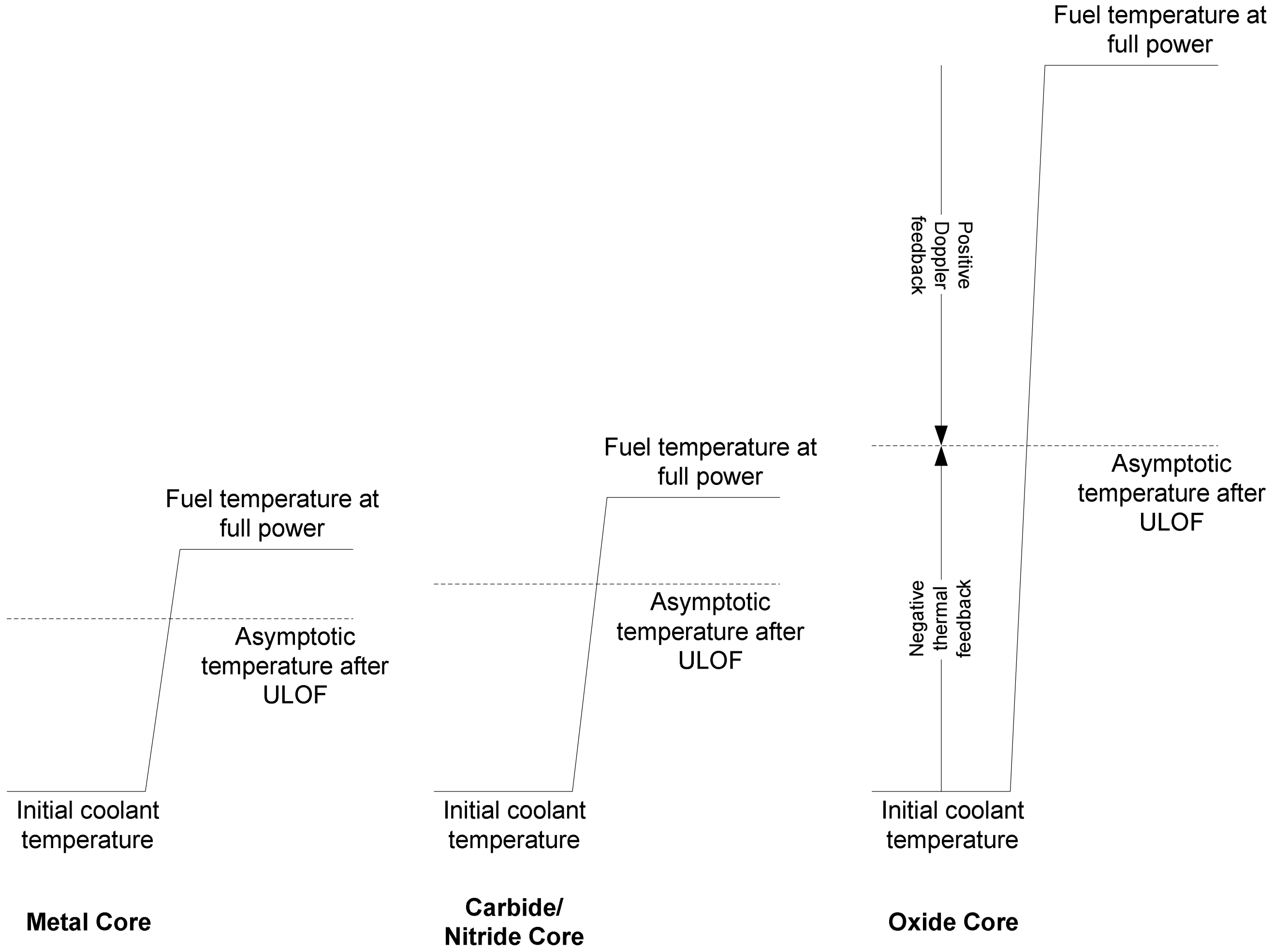
$F(t)$ = normalized core coolant flow

$\delta T_{in}(t)$ = change in coolant temperature at the core inlet

$\delta\rho_{external}$ = externally applied change in reactivity (control rods, etc.)

the relative importance of each of these terms is determined by the grouped reactivity feedback parameters, A, B, and C

Asymptotic Temperature in ULOF



Conclusions

- Impacts of carbide and nitride fuels on ABR core performance and safety were evaluated
  - For consistency, potential design goals and constraints used in reference ABR core were employed to develop ABR core with carbide and nitride fuels
  - Outer assembly dimension of reference ABR core was kept, but intra-assembly design parameters were adjusted
    - *Active core height and fuel volume fraction are between metal and oxide fuels*
  - Core performances parameters and reactivity feedback coefficients were generally between metal and oxide cores
    - *Neutron spectrum is softer than metal, but harder than oxide core*
    - *Discharge burnup (~100 GWd/t) is higher than metal, but smaller than oxide*
    - *TRU enrichment (~24%) is higher than metal, but smaller than oxide*
    - *Marginal changes of kinetics parameters and reactivity feedback coefficients*
  - Good thermal conductivity and high melting temperature lead to a decrease in average fuel temperature significantly, and hence provide huge margin to fuel melt and favorable passive safety features without additional design fixes that were required in oxide core concepts.