Chapt. 10 Fuel Element Performance

10.1) Important Features

Oxide fuel characteristics

- ***Temperature ($T_{max} \approx 2800$ °C)***
    - ***Sintering***
    - ***Grain growth***
    - ***Diffusion***
- $\dfrac{\Delta T}{\Delta x}$
    - ***Stress***
    - ***Pore closing***
    - ***FP redistribution***

10.1.1) Oxide Fuels- (started ≈ 1955) (Fig/Table)

Most important properties of mixed oxides depend on the O/M ratio

$$q = \frac{N_{pu}}{\sum N_{\text{heavy metal}}}$$

$$x = \frac{N_{excess.oxygen}}{\sum N_{heavy.metal}}$$

for example...... $(U_{1-q}Pu_q)O_{2\pm x}$

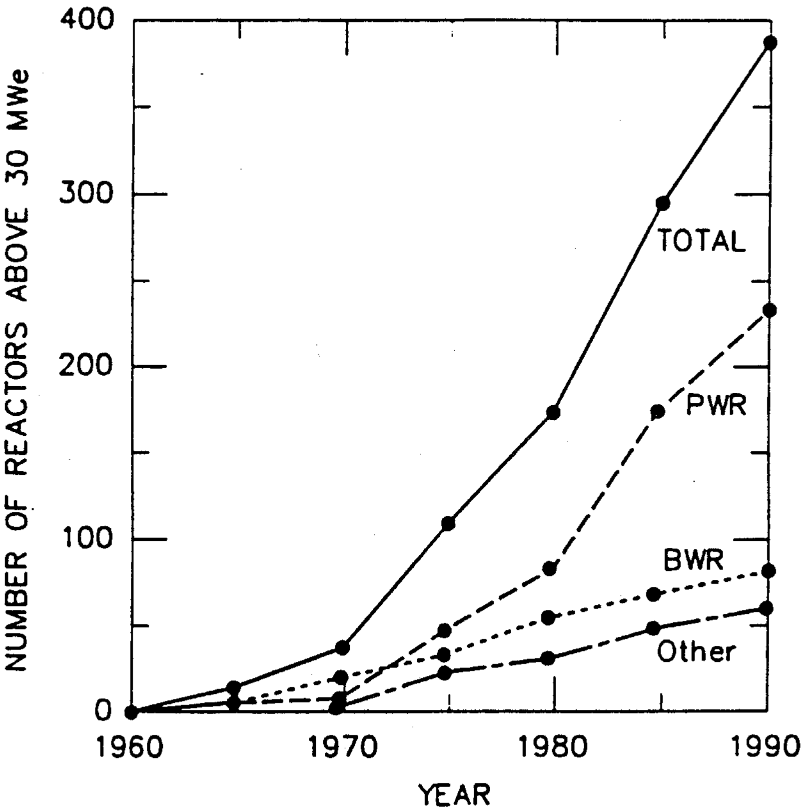


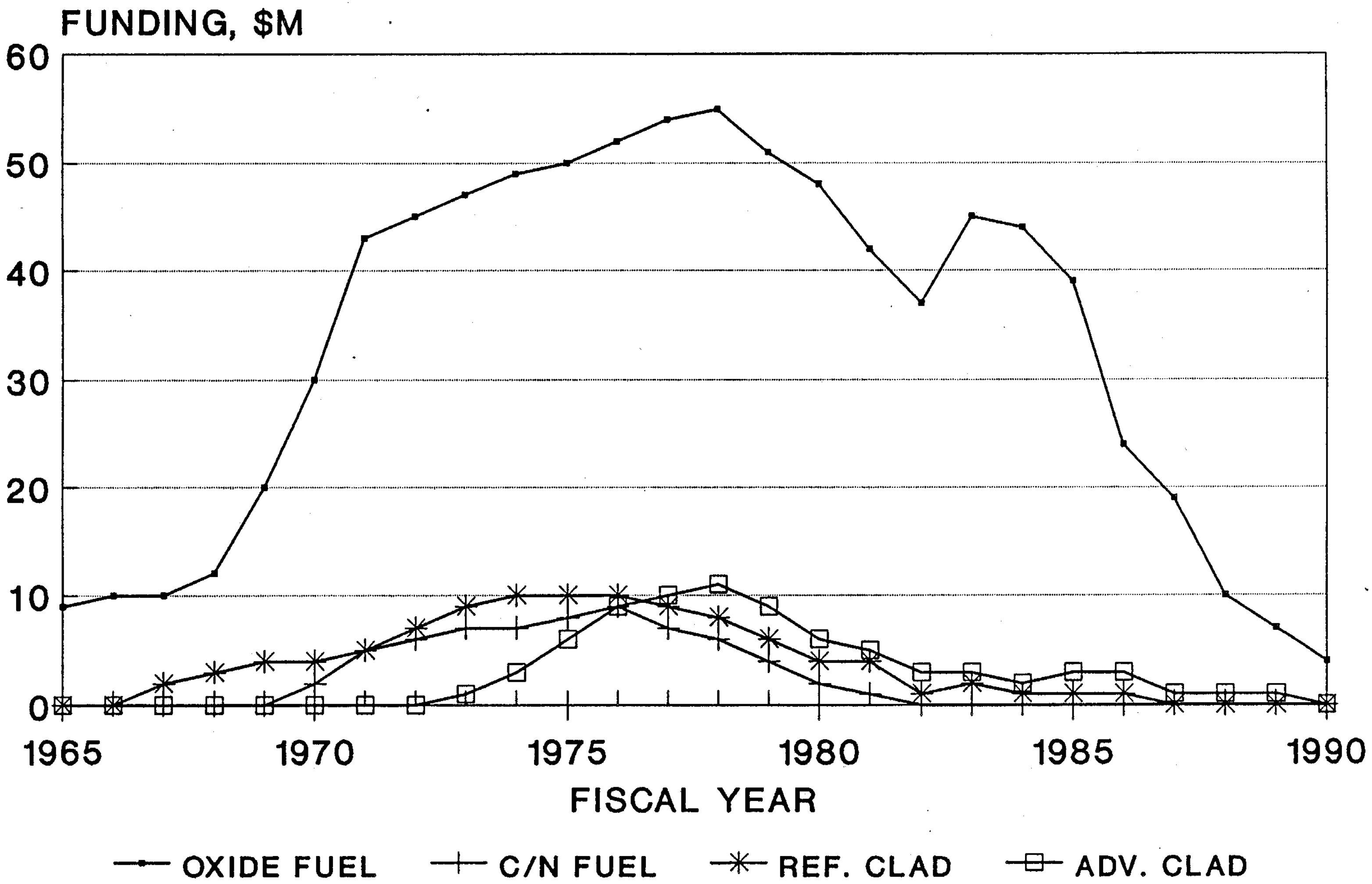
Figure 3-4. Worldwide growth for 1960–1990 in oxide-fuelled power reactors above 30 MW(e). Courtesy: *Nuclear News* (1990).

Table 3-3. Limiting factors in early LWR fuel performance (1965–1975)[a].

Factor	PWR	BWR	Remedies
Hydriding of zirconium	×	×	eliminate fuel moisture; add getters
Scale deposition		×	eliminate copper tubing from feedwater heaters
Enrichment errors	×	×	gamma-scan elements before shipment
Cladding collapse	×		prepressurize elements; use stable pellets
Pellet densification	×	×	stable fuel pellet microstructures
Manufacturing defects (e.g., faulty welds)	×	×	improve QC (now below 5% of in-reactor defects)
Cladding corrosion/fretting		×	rare; improve QA and cleaning; use element spacers
Fuel element growth/bowing	×		control texture, axial clearances, spacer design
Channel bulging		×	thicker channel walls. control residual stress
Pellet–cladding interactions (PCI)	×	×	(1) slow power rise (PWR) (2) power shape control (BWR) (3) fuel "preconditioning" phase (both) (4) pellet design changes (both)

[a] From Levenson and Zebroski (1976).

PROGRAM ANNUAL COSTS



PROGRAM DURATION AND COST

Program Element	*Period*	*Duration*	*Cost*
Oxide Fuel	**1965-1992**	**27 yrs**	**$ 846M**
Carbide/Nitride Fuel	**1969-1982**	**13 yrs**	**64M**
Reference Alloy	**1966-1987**	**21 yrs**	**101M**
Advanced Alloys	**1973-1990**	**17 yrs**	**77M**
		TOTAL	**$ 1,088M**

Hyperstoichiometry O/M > 2.00

Hypostoichiometry O/M < 2.00

Disadvantages of Oxide Fuels

1.) **Low U density (larger core)**

2.) **Low thermal conductivity**
(requires thin rods, produces high T)

Why not use UC, UN, UP, US, etc.?

10.1.2 Fission Properties

Burnup

$$\beta = \frac{\text{Number of Fissions}}{\text{Initial Number of Heavy Metal Atoms}}$$

Rule of Thumb

$$1 \text{ at\% B.U.} \approx 10{,}000 \frac{MWd}{MtU} \quad @\, 200 \frac{MeV}{fission}$$

Fissioning of 1 g of fissile material ≈ 1MWd

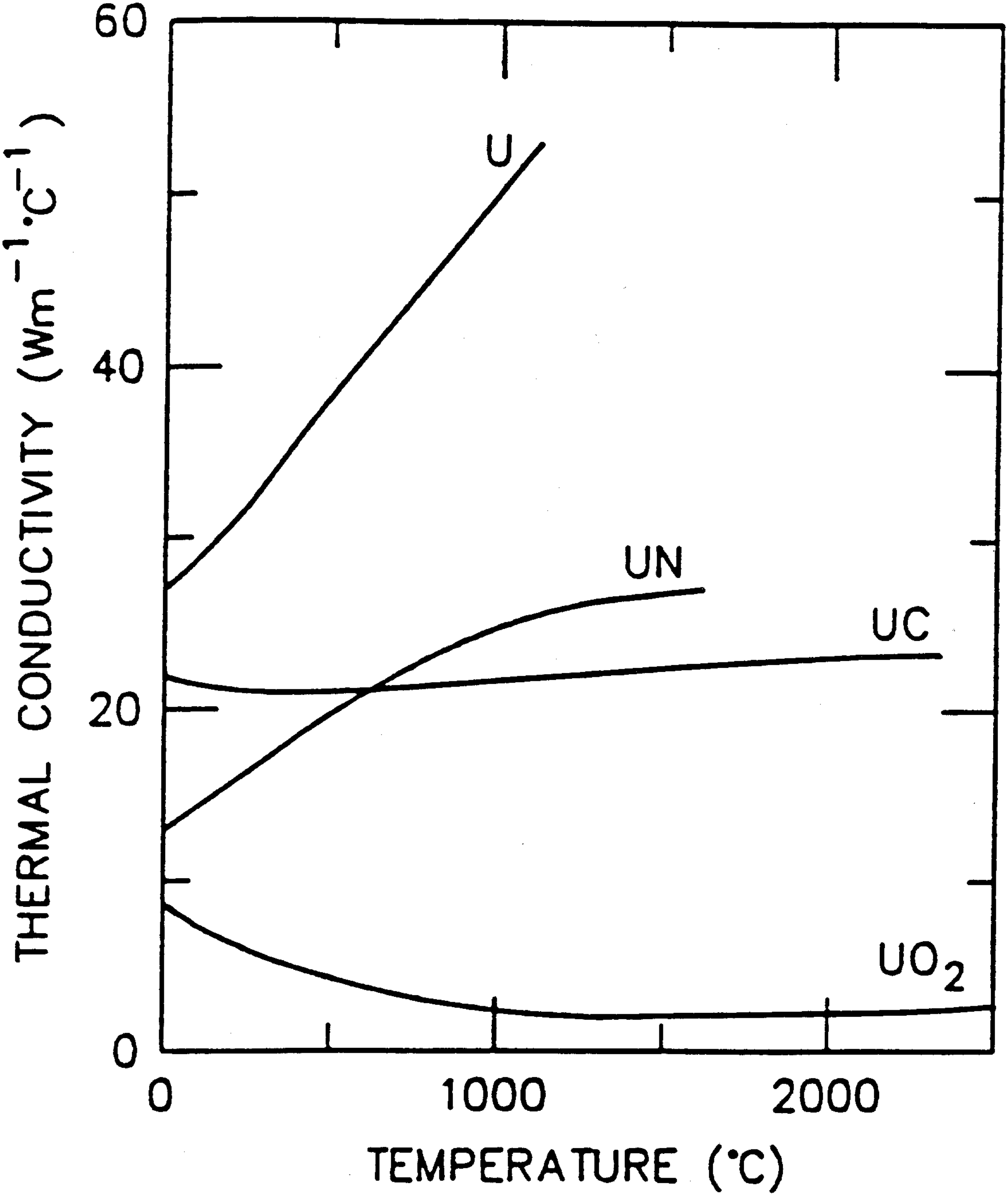


Figure 3-2. Thermal conductivities of major nuclear fuels.

Maximum burn up with no Pu credit;

- ***Natural U*** $6650 \ \frac{MWd}{MtU}$
- ***15% enriched*** $142{,}500 \ \frac{MWd}{MtU}$

Major Differences Between LWR & LMFBR Fuels

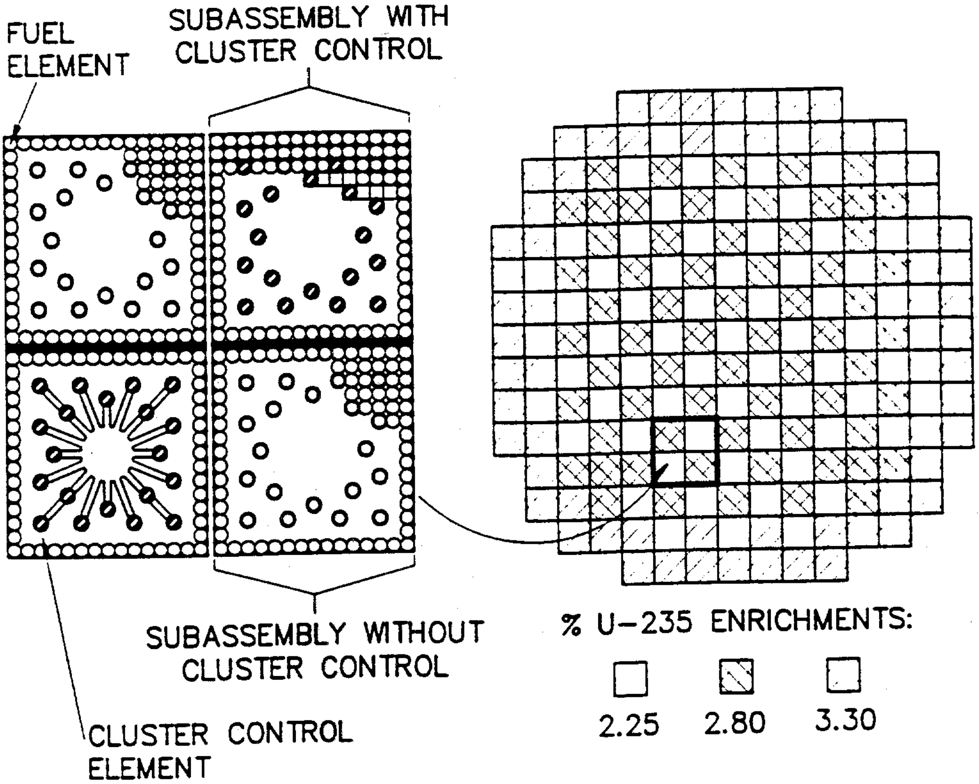
	LWR	LMFBR
Ave neutron energy, eV	0.03	500,000
Fissile isotope	^{235}U	^{239}Pu
σ_f, barns	550	1.8
Neutron flux, (rel)	1	300
Power density, (rel)	1	3 (why?)
Burn up, %	3	10
Enrichment, %	3	15

Differences Between LWR and LMFBR Fuel Assemblies

	LWR	LMFBR
Damage to clad	1	100
Fuel pin diameter, mm	11	6
Core fuel fraction	-	Higher
Cladding	-	Hotter
Size of Core	-	Smaller (no moderator needed)

See Table 10.2 and figures 10.1 thru 10.4

Table 10.2 Comparison of Fuel Element Characteristics		
	Thermal Reactor	**Fast Reactor**
Fuel	UO_2	$(U,Pu)O_{1.96}$
Fuel Pellet Density (% of theoretical)	92	90
Max. fuel centerline temperature (overpower condition)°C	2450	2800
Cladding	Zircaloy-4	316 Stainless Steel
Max. cladding mid-wall temperature °C	380	660
Coolant temperature, °C	H_2O, 280-320	Na, 470-650
Maximum rod linear power, W/cm	620	550
Fuel wrapper assembly	Square, 30x30	Hexagonal, 13 cm across flats
# of pins in assembly	200	220
Fuel-rod outside diameter, mm	10.7	6.3
Cladding thickness, mm	0.6	0.4
Initial fuel-cladding radial gap, mm	0.08	0.07
Length of fueled portion, cm	365	90



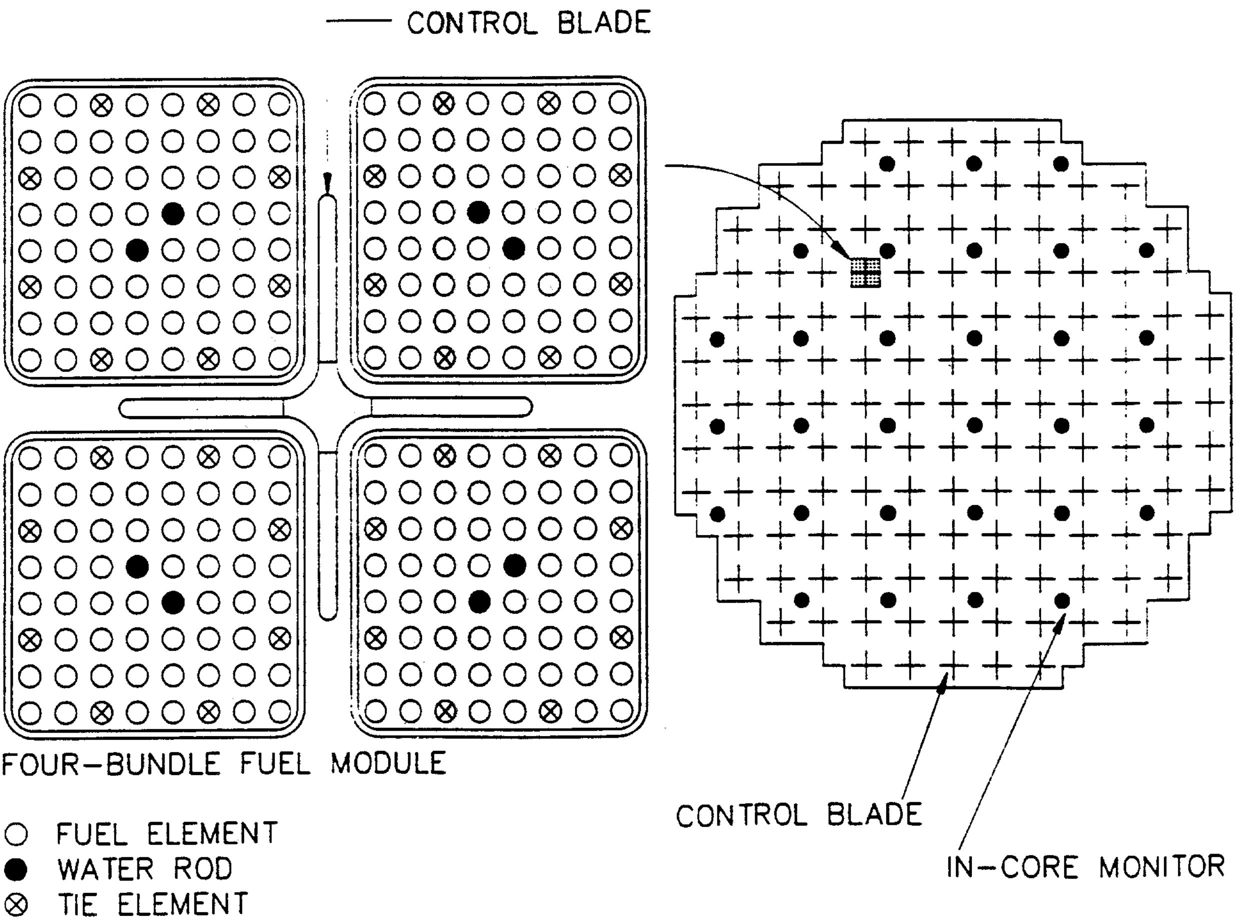


Figure 3-8. Schematic of BWR fuel subassembly and reactor core.

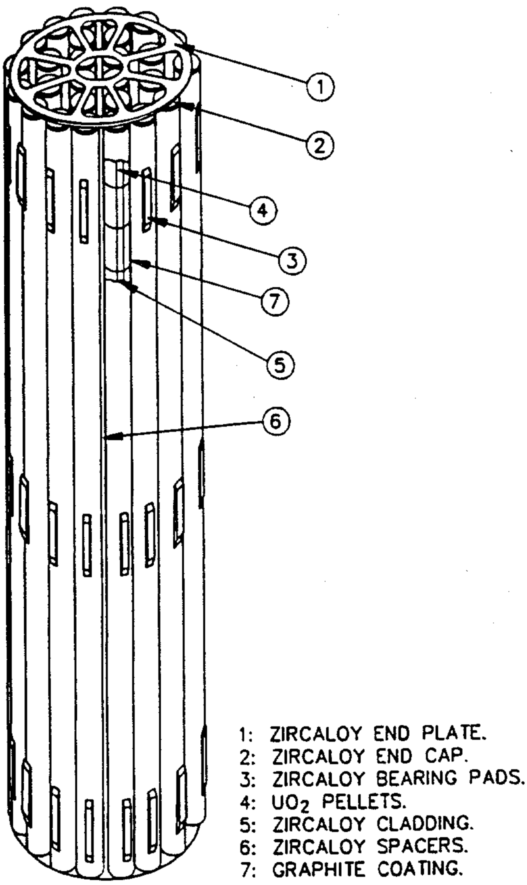


Figure 3-9. Schematic of CANDU fuel bundle. Courtesy: Atomic Energy of Canada.

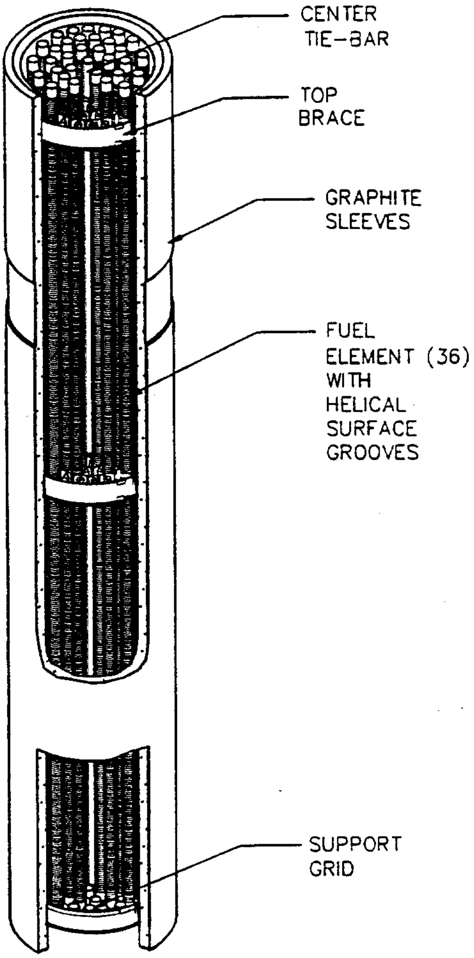


Figure 3-10. Schematic of an AGR subassembly and fuel stringer. Courtesy: United Kingdom Atomic Energy Authority.

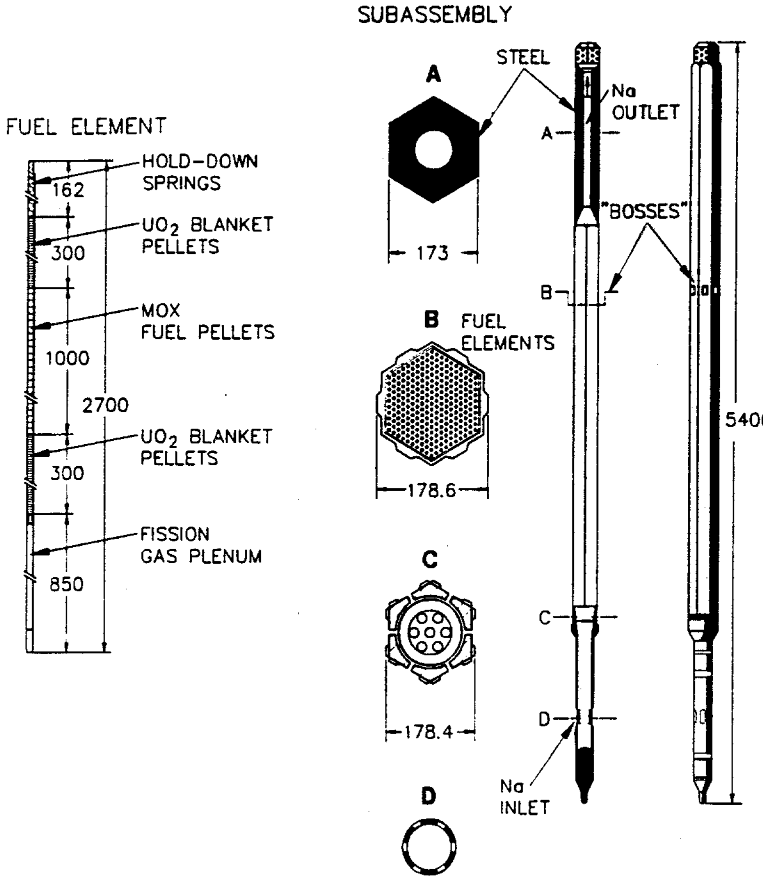


Figure 3-11. Schematic of an FBR fuel element and subassembly for the Superphenix-1 reactor. Courtesy: Commissariat à l'Énergie Atomique.

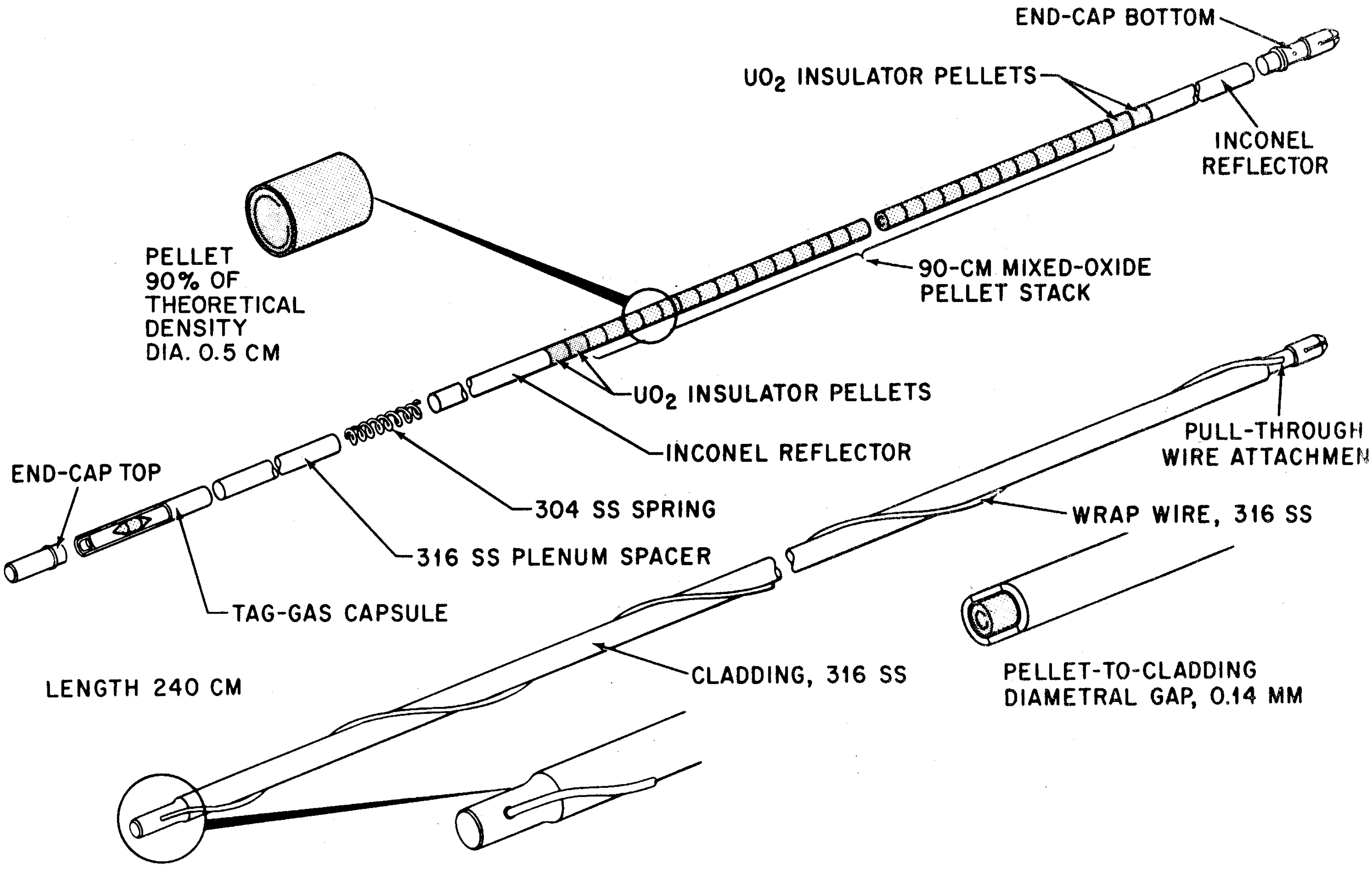


Fig. 10.2 Fuel pin of t' Fast Test Reactor. (Courtesy C. Burgess, Hanford Engi..ering Development Laboratory.)

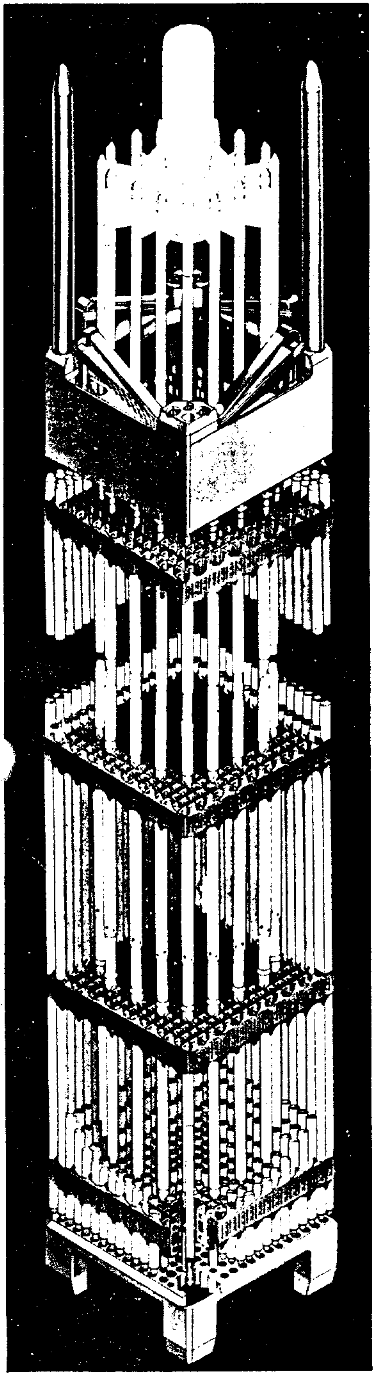
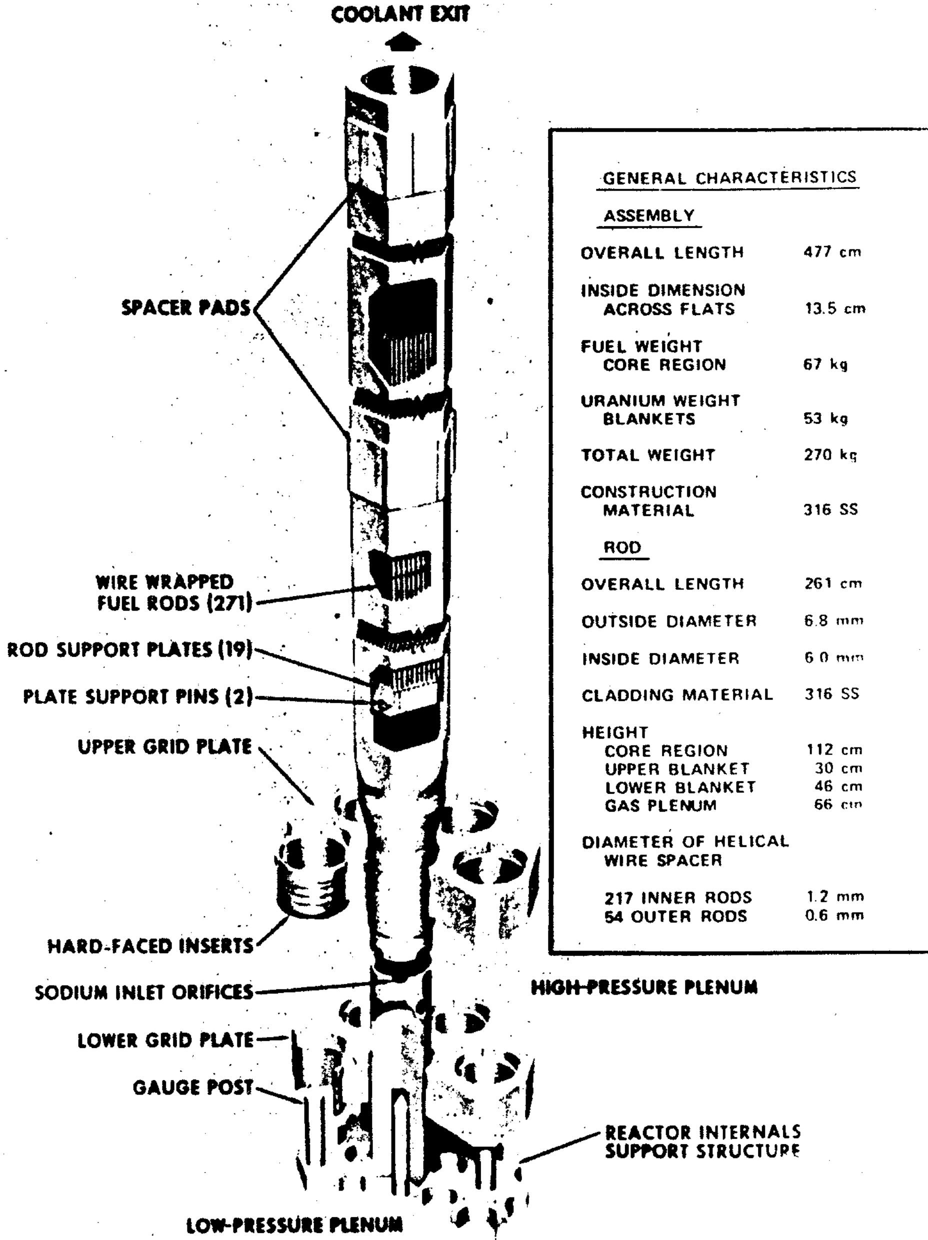


Fig. 10.3 Pressurized-water-reactor fuel assembly. (Courtesy Westinghouse Company.)



GENERAL CHARACTERISTICS

ASSEMBLY	
OVERALL LENGTH	477 cm
INSIDE DIMENSION ACROSS FLATS	13.5 cm
FUEL WEIGHT CORE REGION	67 kg
URANIUM WEIGHT BLANKETS	53 kg
TOTAL WEIGHT	270 kg
CONSTRUCTION MATERIAL	316 SS
ROD	
OVERALL LENGTH	261 cm
OUTSIDE DIAMETER	6.8 mm
INSIDE DIAMETER	6.0 mm
CLADDING MATERIAL	316 SS
HEIGHT	
CORE REGION	112 cm
UPPER BLANKET	30 cm
LOWER BLANKET	46 cm
GAS PLENUM	66 cm
DIAMETER OF HELICAL WIRE SPACER	
217 INNER RODS	1.2 mm
54 OUTER RODS	0.6 mm

Fig. 10.4 LMFBR fuel assembly. (Courtesy L. Bernath, Atomics International.)

10.2 Thermal Properties

10.2.1 Melting Points

T_{MP} of UO_2 = 2865 °C

(2847 °C in some papers)

Several Factors Influence the Melting Point

- *Stoichiometry (phase diagram)*
- *Mixed Oxide ,UO_2 -PuO_2 , (2 figs see correc.)*
- *Burn -Up (figure + table)*

10.2.2 Thermal Expansion

Problems -Cladding stress, Poor heat transfer

Stoichiometry Effect-Figure 10.8

10.2.3 Specific Heat (See Chapter 1)

(Important for dynamic behaviour, $\frac{k}{\rho C_p}$)

$$C_p = \left(\frac{\delta H}{\delta T}\right)_p = C_v + \Big(\underbrace{\frac{\overbrace{\alpha^2}^{\text{thermal expansion}}\ \overbrace{V}^{\text{Molar Volume}}}{\underbrace{\beta}_{\text{compressibility}}}}\Big)T$$

Figure 10.10

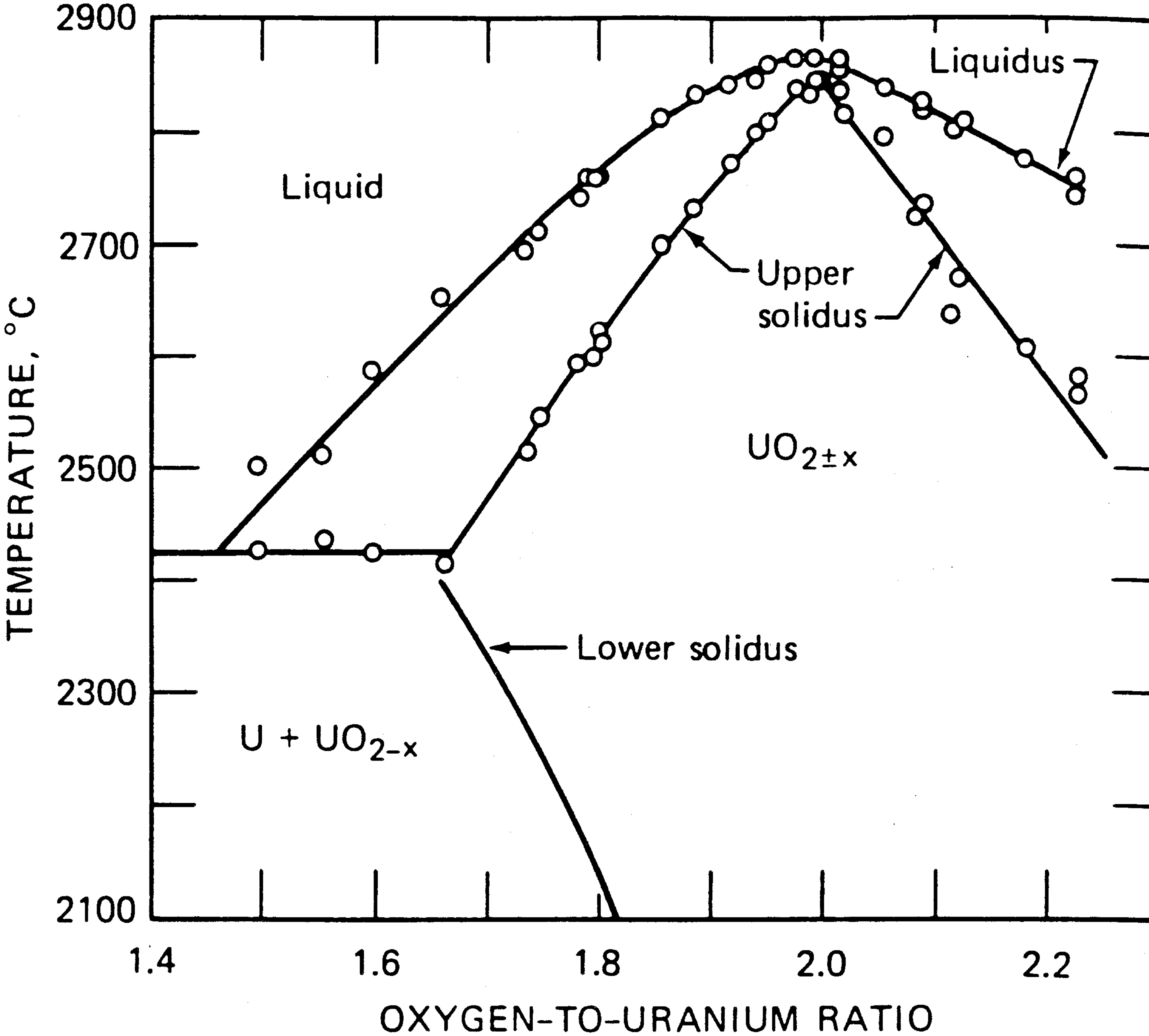


Fig. 10.5 Partial phase diagram for urania from $UO_{1.5}$ to $UO_{2.23}$. The separation of the peaks of the liquidus and solidus curves at O/U = 2.0 is undoubtedly due to measurement errors. The UO_2 melts congruently; thus, the curves should coincide for $UO_{2.0}$. Similarly, the lower solidus curve should intersect the corner of the upper solidus and horizontal lines. [From R. E. Latta and R. E. Fryxell, *J. Nucl. Mater.*, **35**: 195 (1970).]

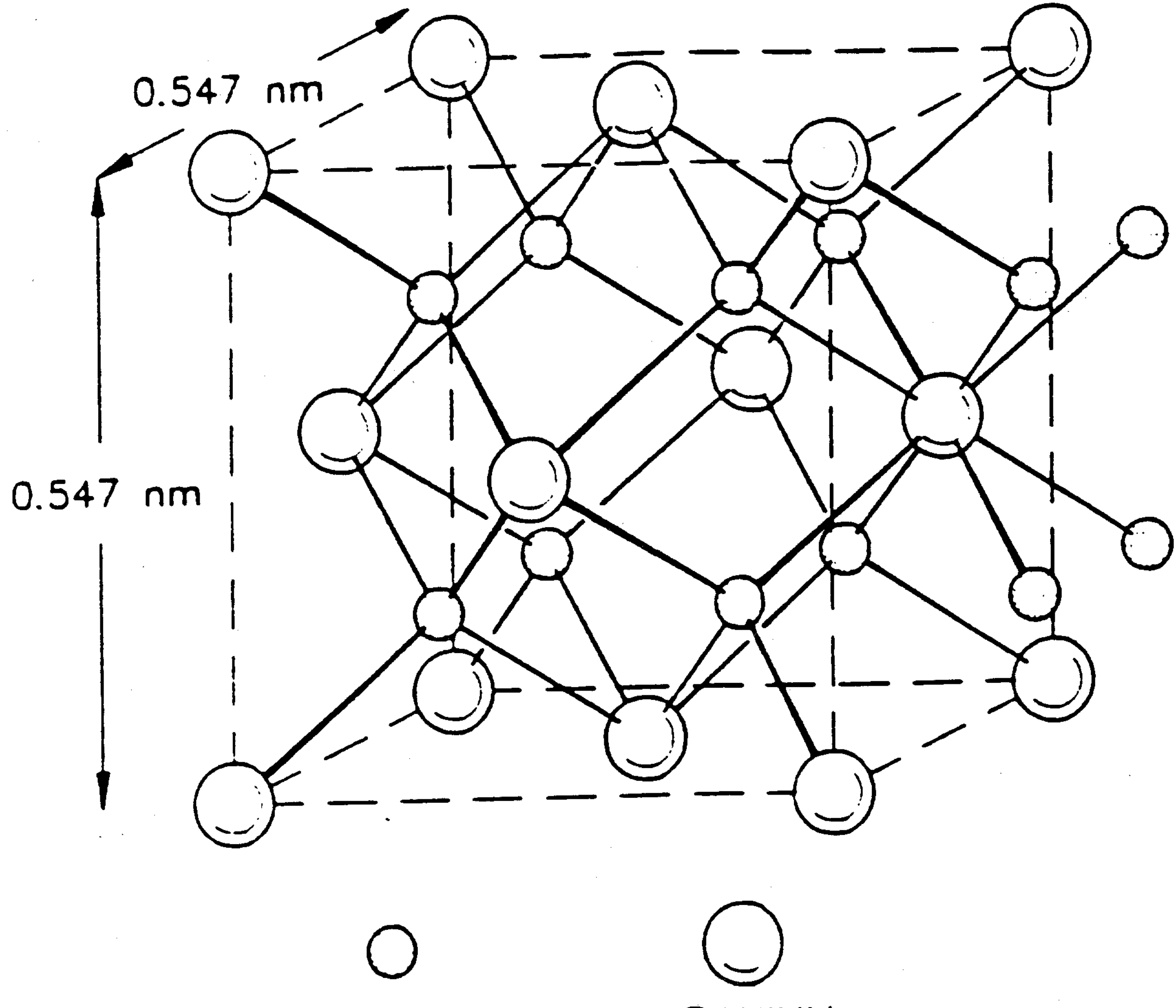


Figure 3-1. Unit cell of stoichiometric UO_2.

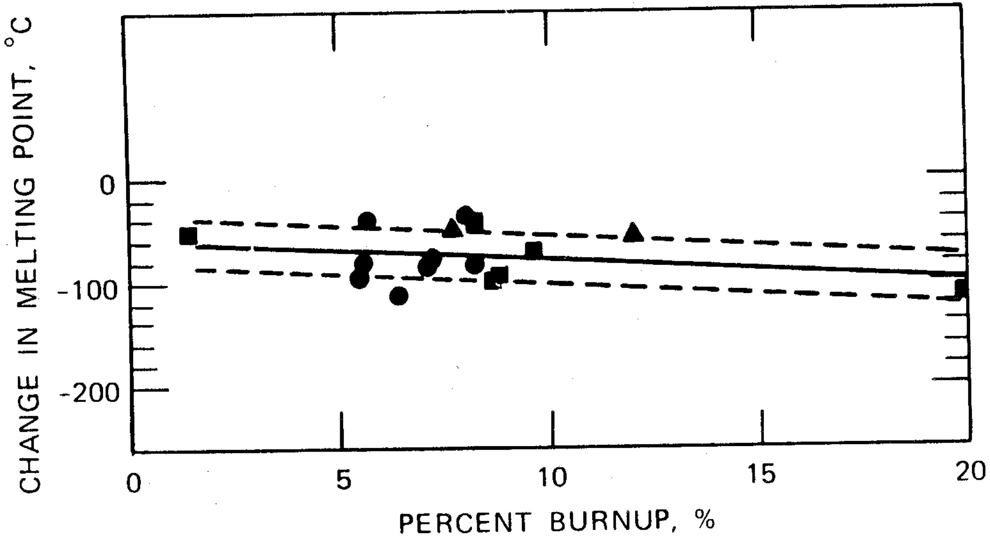


Fig. 10.7 Effect of burnup on the melting point of mixed-oxide fuel material. The dashed lines delineate a band ± one standard deviation wide. ■, 25% PuO_2, O/M = 2.00. ▲, 25% PuO_2, O/M = 1.96. ●, 20% PuO_2, O/M = 2.00. [From A. Biancheria, U. P. Nayak, and M. S. Beck, in *Proceedings of the Conference on Fast Reactor Fuel Element Technology*, R. Farmakes (Ed.), p. 361, American Nuclear Society, Hinsdale, Ill., 1971.]

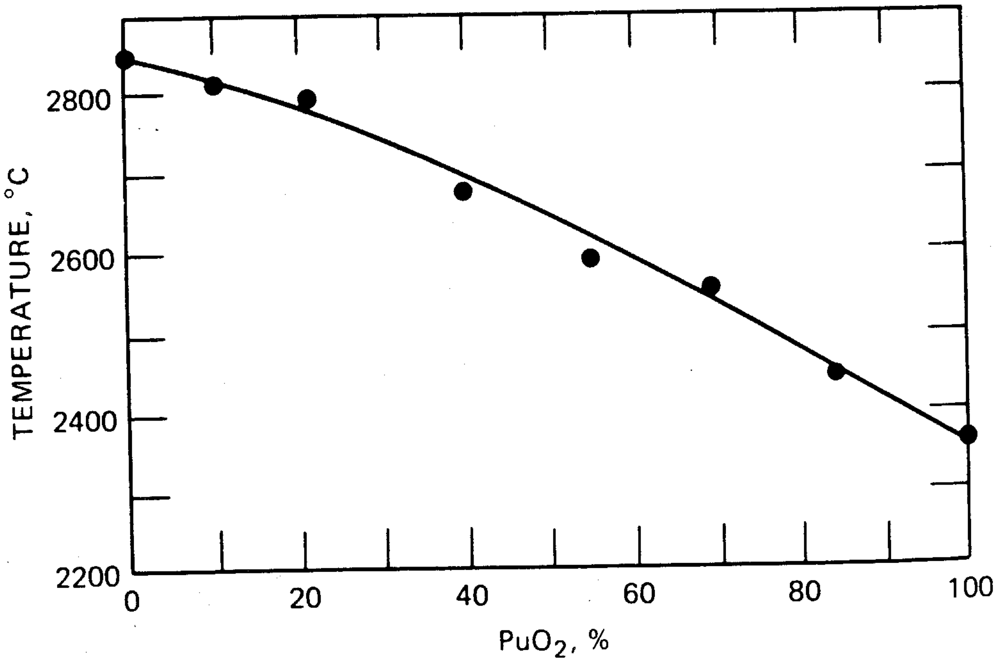


Fig. 10.6 Melting points of mixed uranium—plutonium oxides. (From E. L. Zebroski, W. L. Lyon, and W. E. Bailey, Effect of Stoichiometry on the Properties of Mixed Oxide U—Pu Fuel, in Proceedings of the Conference on Safety, Fuels, and Core Design in Large Fast Power Reactors, Oct. 11-14, 1965, USAEC Report ANL-7120, p. 382, Argonne National Laboratory, 1965.)

Corrections to Olander Book
from Nuclear System Handbook (NSH)

Melting Point, °C

Mole Fraction PuO_2	Solidus	NSH Liquidus	Liquidus Olander
0.0	2847	2847	2865
0.2	2728	2767	
0.4	2632	2685	≈2700
0.6	2553	2600	≈2600
0.8	2487	2530	
1.0	2428	2428	≈2360

Density:

$$\rho = \frac{\text{MW metal atoms}}{\text{MW nat. U}} \left[5875.5+4.97(\%PuO_2)+2540\frac{O}{M}\right] \frac{kg}{m3}$$

Effect of Burnup on MP of Mixed UO_2-PuO_2

For 20-25% PuO_2
from NSH

Burn up MWd/MtU	**Change in Melting Point °C**
15,000	**-62**
25,000	**-64**
50,000	**-69**
75,000	**-74**
100,000	**-80**

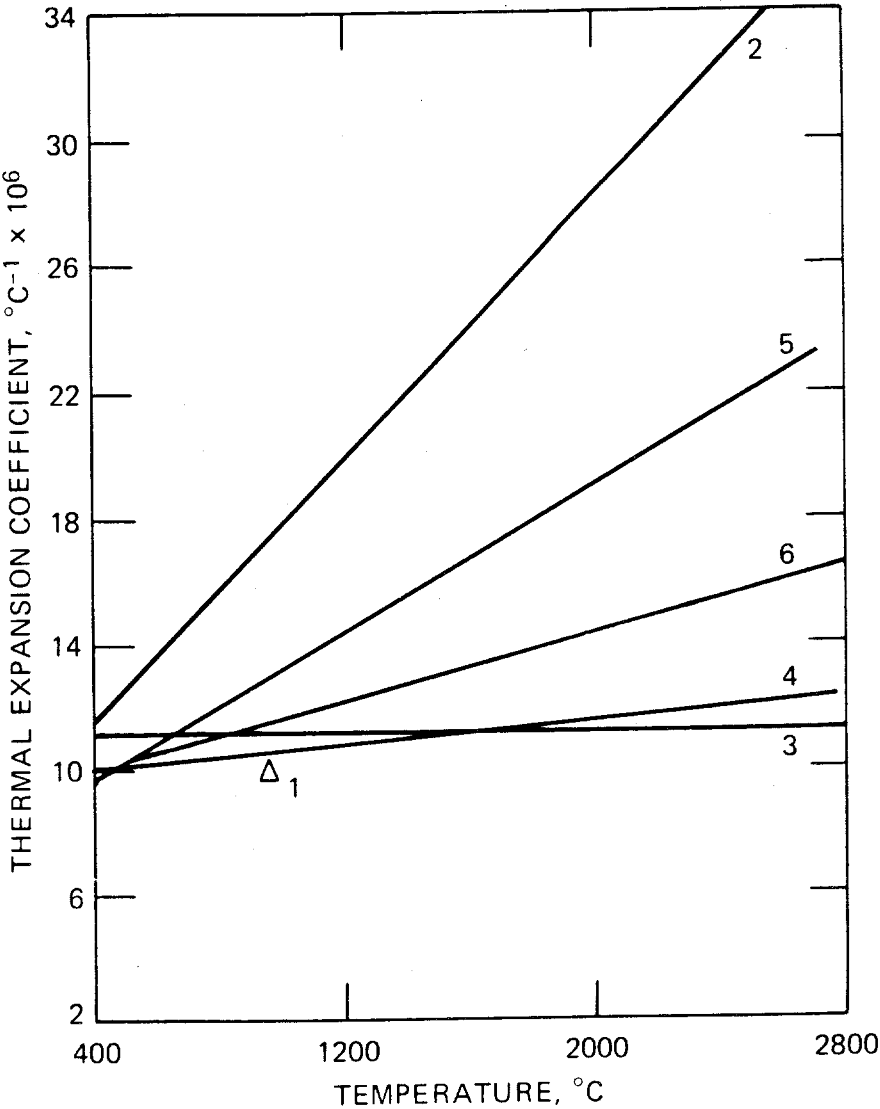


Fig. 10.8 Thermal expansion coefficients of mixed-oxide fuels.

1. $(U_{0.8}Pu_{0.2})O_2$, R. P. Nelson, USAEC Report BNWL-473, 1967.
2. $(U_{0.8}Pu_{0.2})O_{2.10}$, J. Roth and E. K. Halteman, USAEC Report NUMEC-2389-9, 1965.
3. $UO_{2.24}$, ibid.
4. $UO_{2.08}$, ibid.
5. $(U_{0.95}Pu_{0.05})O_{2.11}$, ibid.
6. $(U_{0.85}Pu_{0.15})O_{2.13}$, ibid.

(From F. J. Homan, Parametric Analysis of Fuel-Cladding Mechanical Interactions, USAEC Report ORNL-TM-3508, p. 13, Oak Ridge National Laboratory, August 1971.)

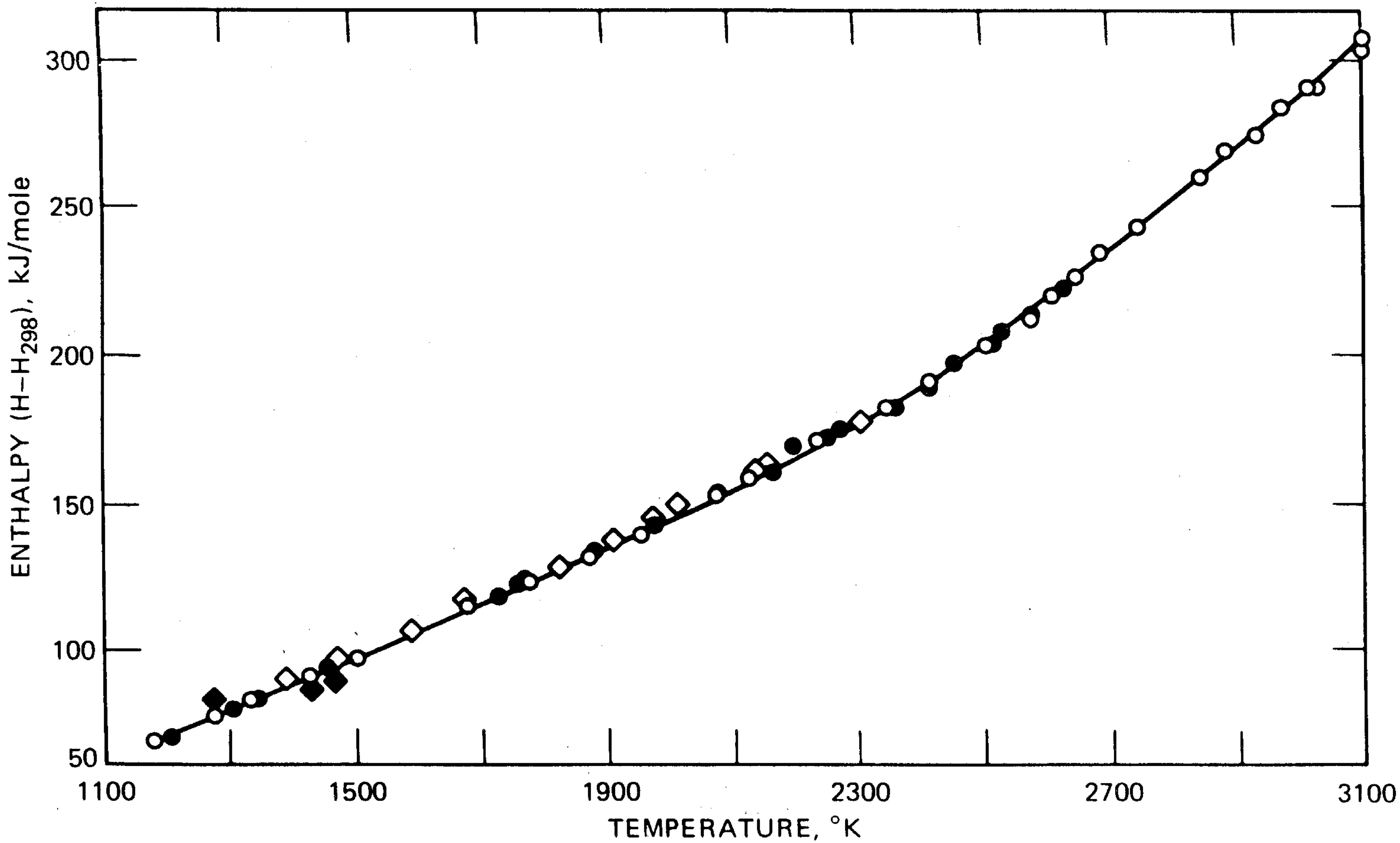


Fig. 10.10 Enthalpy—temperature data for stoichiometric UO_2. [From R. A. Hein, L. H. Sjodahl, and R. Szwarc, *J. Mater.*, 25: 99 (1968).]