Outline of Nuclear Fuel Engineering

Part 1. Fuel behavior during normal operation；
Structure and design of fuel
Behavior of fuel pellet and cladding,
High-burnup fuel, MOX fuel, Fuel failure

Part 2. Irradiation test；
Test reactor, Instrumentation, PIE

Part 3. Fuel behavior under accident
RIA, LOCA

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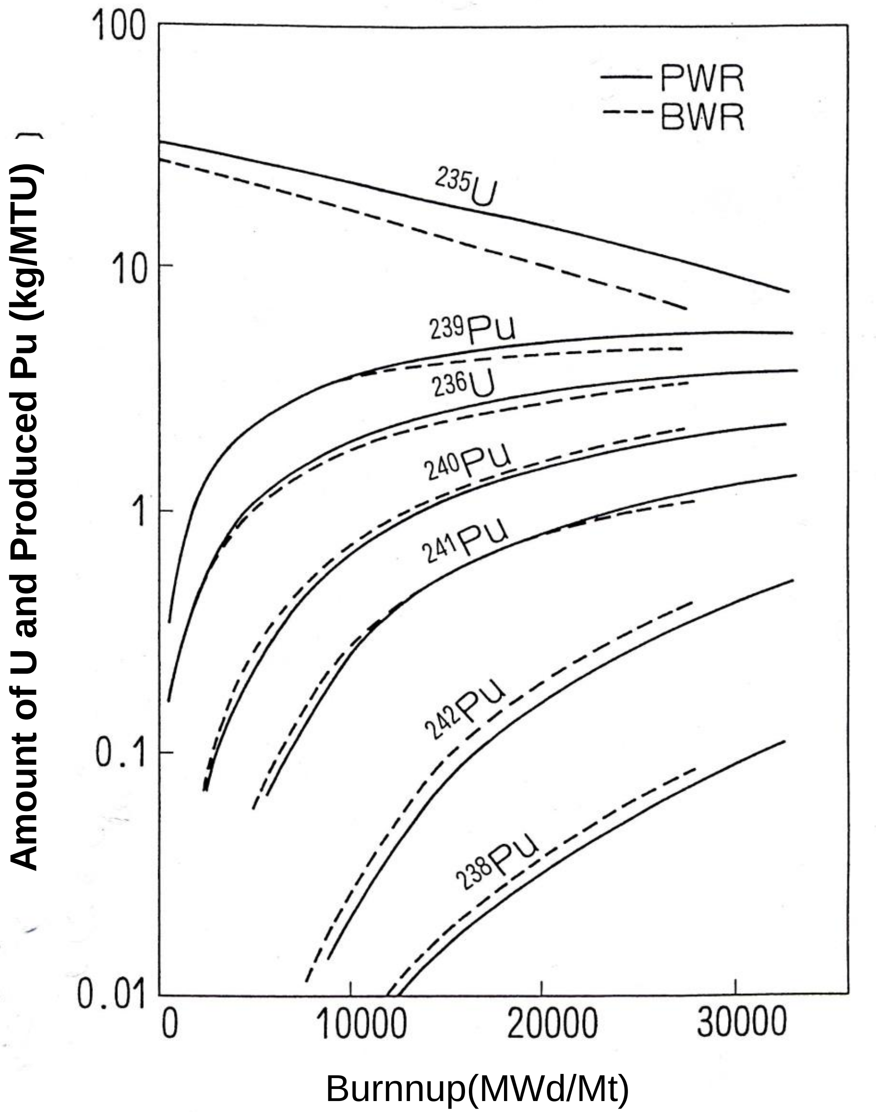
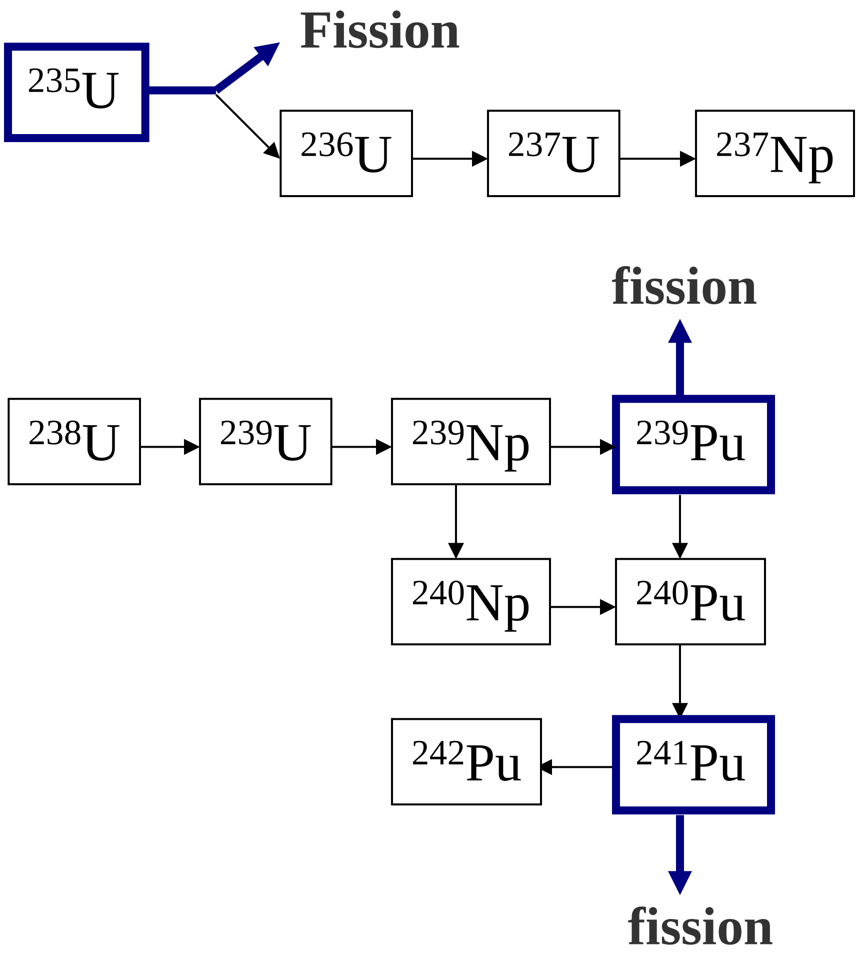
Major nuclear fission reaction in reactor



Production of Pu in UO_2 fuel for LWR



With burnup increase, the fission ratio of ^{239}Pu and ^{241}Pu to ^{235}U increases.

Role of nuclear fuel pellet

(1) To produce energy continuously

→ Atomic density of nuclear fissile material, and nuclear properties of chemical compound must be advantageous for continuity of nuclear reaction.

(2) To transfer the produced energy to coolant

→ To transfer the energy as thermal energy to coolant, effectively, nuclear fuel have to have high thermal conductivity.

(3) Work as the first wall for radioactivity such as FPs

→ To be stable physically and chemically at high temperature use, defects formation by irradiation must be low.

In fuel, there exists radial temperature gradient, and thermal conductivity K(T) is a function of temperature. The maximum linear hear rate P(W/m) of fuel rod is determined by thermal conductivity integral∫K(T)dt.

$$\int_{Ts}^{Tc} K(T)dt = P/4\pi$$

Fission energy and burnup

❑Breakdown of fission energy of ^{235}U

- Fission fragment ~168 MeV
- neutron ~ 5 MeV
- prompt γ -ray ~ 5 MeV
- β decay of FPs ~ 7 MeV
- γ decay of FPs ~ 6 MeV
- neutrino ~ 11 MeV
- decay of reacted products ~ 7 MeV

total ~210 MeV

At one nuclear fission, about 210MeV is produced, and about 195MeV is used as thermal energy.

About 30% of fission energy is converted to electricity at a nuclear power plant.

❑ Burning of nuclear fuel material and heat generation （burnup）

1%of 1t of fuel material burned :

$1t \times 0.01 = 0.01t = 10kg$
$10{,}000g / 238 = 42mol$
$42 \times 6.0 \times 10^{23} = 2.5 \times 10^{25}$ atom

1%burnup（%FIMA）

$= 2.5 \times 10^{25} \times 195$ MeV/tHM
$= 2.5 \times 10^{25} \times 195 \times 10^{6} \times 1.6 \times 10^{-19}$ J/tHM
$= 7.8 \times 10^{14}$ Ws/tHM
$\approx 9{,}000$ MWd/tHM
≈ 9 GWd/tHM

HM ••• Heavy Metal

FIMA ••• Fission per Initial Metal Atom

Type of commercial Reactor and fuel form

Reactor		Fuel		Cladding	Moderator	Coolant
kind	type	composition	enrich			
LWR	BWR	UO_2	low enrich	zircaloy-2	H_2O	H_2O
	PWR	UO_2	low enrich	zircaloy-4	H_2O	H_2O
HWR	ATR	UO_2/MOX	low enrich	zircaloy-2	D_2O	H_2O
	CANDU	UO_2	natural	zircaloy-4	D_2O	D_2O
GCR	Calder Hall	U alloy	natural	magnox	Graphite	CO_2
	AGR	UO_2	low enrich	SUS	Graphite	CO_2
FBR	LMFBR	MOX	depleted	SUS	none	Na

Composition of natural Uranium

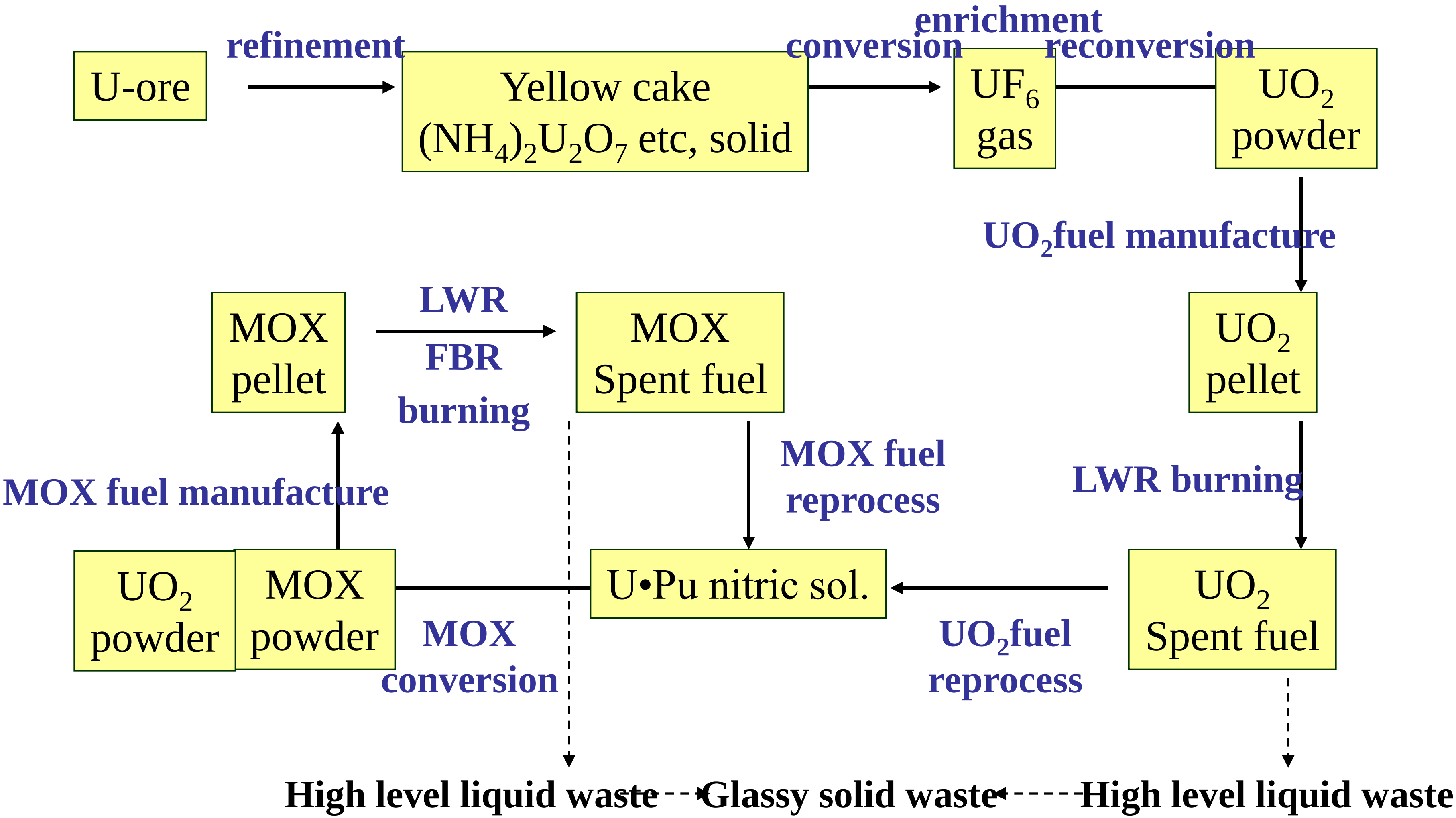
U-235 0.72%

U-238 99.28%

U-234 0.006%

About 3-5% enriched UO_2 is used for LWRs

Physical/chemical form of nuclear fuel material



Flow of nuclear fuel cycle

Fundamental property of nuclear fuel material

	UO_2	metallic U	UC	UN	PuO_2
melting point （C）	2847±30	1132	2507±25	2830±40 (1atmN_2)	2390±35
density (g/cm^3)	10.96	19.05	13.62	14.32	11.46
heavy metal density (g/cm^3)	9.66	19.05	12.97	13.52	10.11
thermal conductivity (W/mK)	about 3 (1000 C) low	35 (397 C) very high	21.6 (1000 C) high	25 (1000 C) high	about 2.7 (1000 C) low
Phase transition	none	2	none	none	none
Crystal structure (at room temp.)	fluorite FCC	Hexagonal	NaCl FCC	NaCl FCC	fluorite FCC

Characteristics of reactor core and fuel in PWR and BWR

	PWR	BWR
Coolant pressure Coolant temperature Power density of core	157kg/cm^2（Genkai No.3） Inlet 289C, Outlet 325C 104.9kW/l Compact core	70.7kg/cm^2（Kashiwazaki No.2） Inlet 279C, Outlet 286C 50.0kW/l
Fuel assembly Cladding	No partition（canless） Open core coolant channel Fixed enrichment in assem. 14x14, 15x15, 17x17 （not replaceable） zircaly-4、Nb added alloy Low temp. anneal（450-480C）	partition（channel box） Separated coolant channel Various enrichment 7x7 → 8x8 → 9x9 （replaceable） Zircaloy-2 High temp. anneal（580C）
Reactivity control long term short term emergency	 Boron concentration Gd_2O_3 added pellet Control rod cluster Control rod cluster	 Control rod Gd_2O_3 added pellet Recirculation fluid mass Control rod

Part 1.　1.1 What is fuel behavior?

- Fuel behavior = comprehensive interaction of many factors
  - ･temperature　　･neutron flux
  - ･fission：heat generation, Fission Products accumulation
  - ･micro and macro

- Why fuel behavior is important?

①To keep the integrity of fuel rod and assembly until the designed burnup(life limit)

②Complicated interactions : -experience is important-

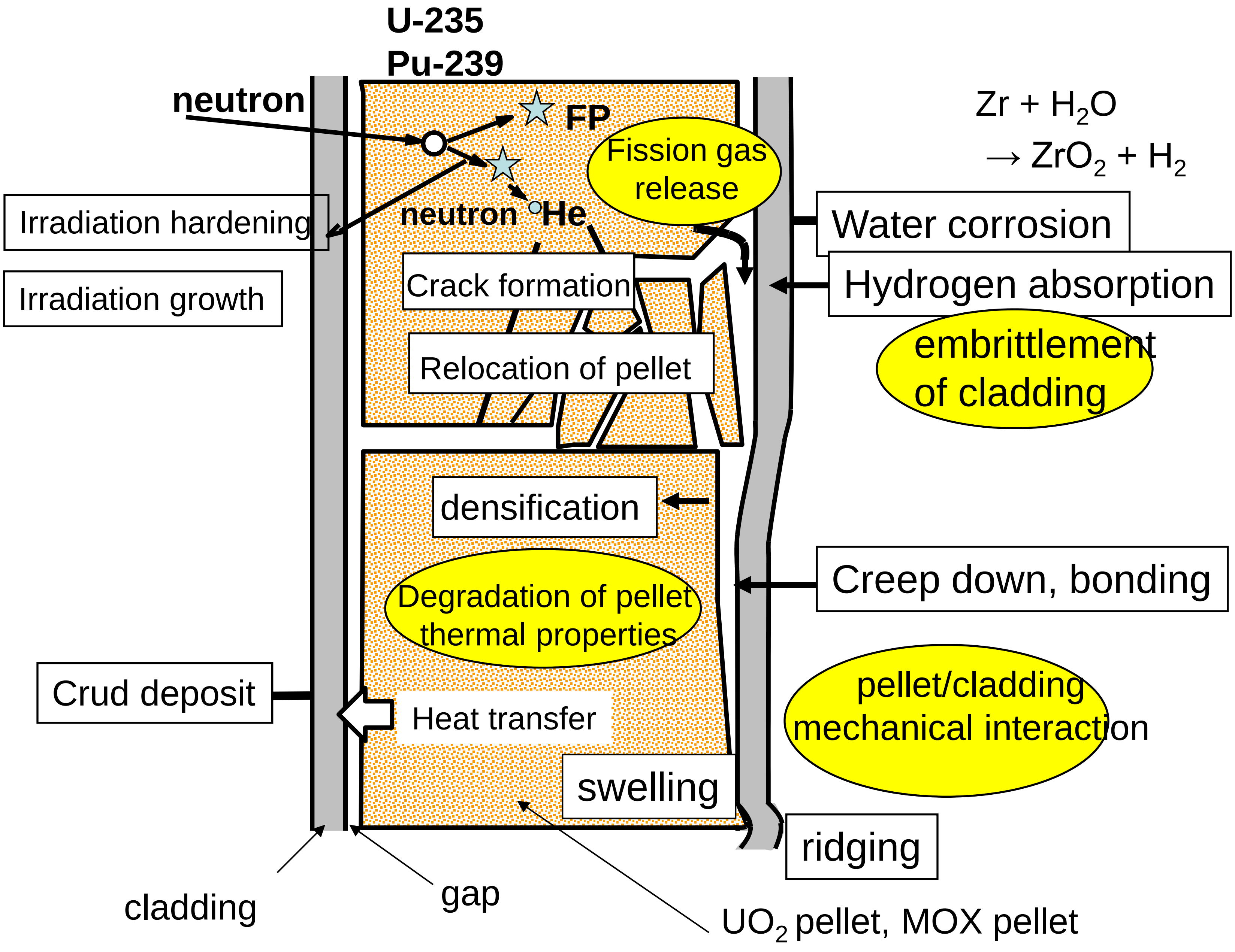
･it is sometimes difficult to predict with theoretical/ deterministic model

･cannot be treated as reactor physics by theoretical model

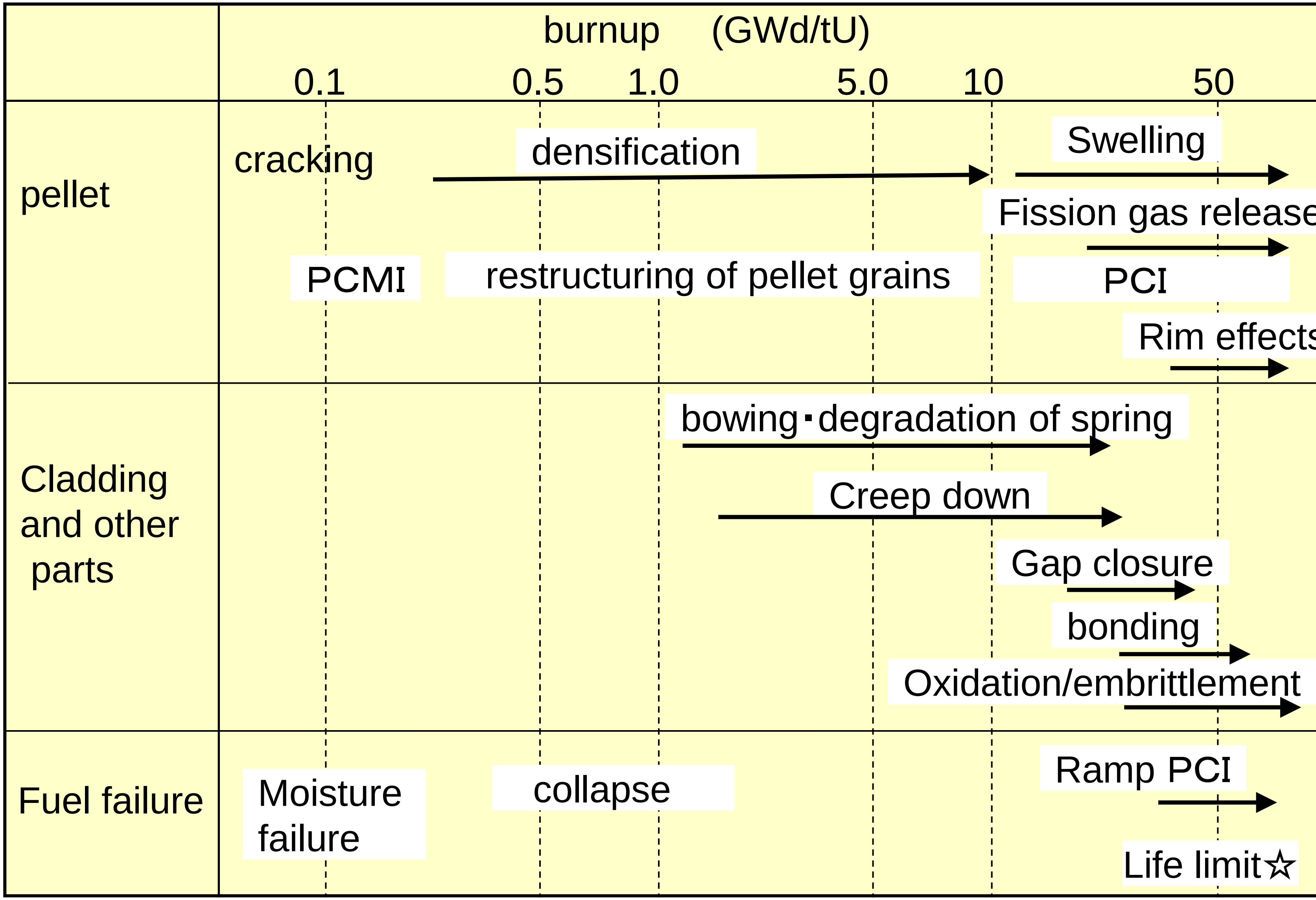
･metallurgical structure sensitivity, production hysteresis

･many unknown factors, irradiation experiment is important.

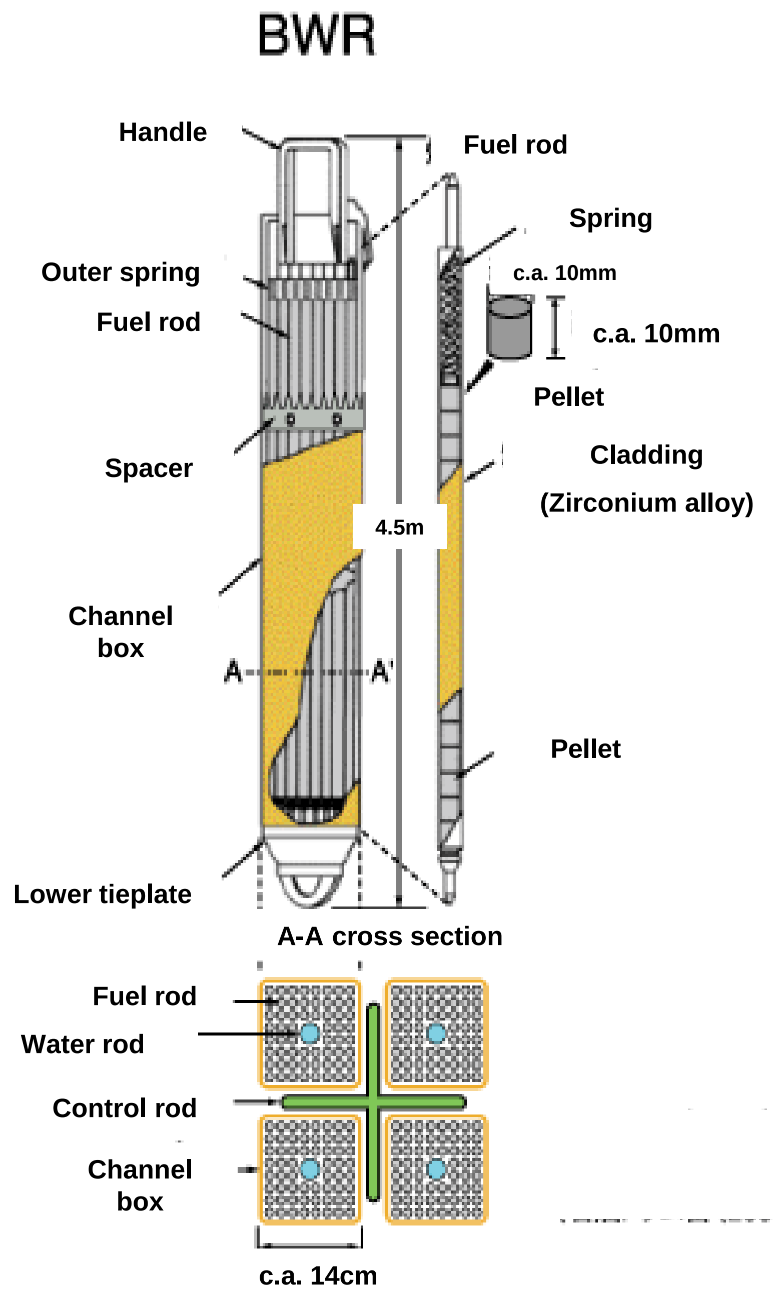
Phenomena occur during rector operation

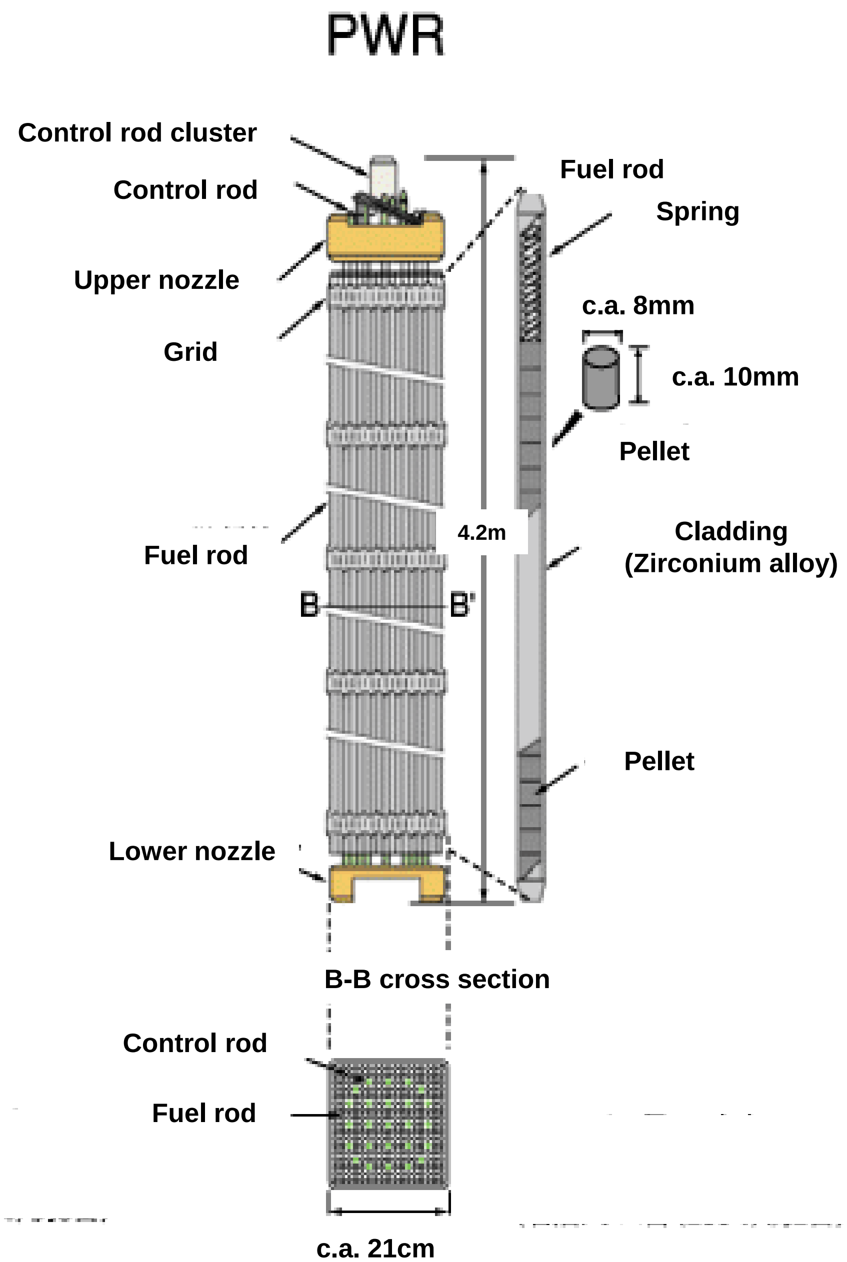


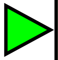
Changes in fuel with burnup increase



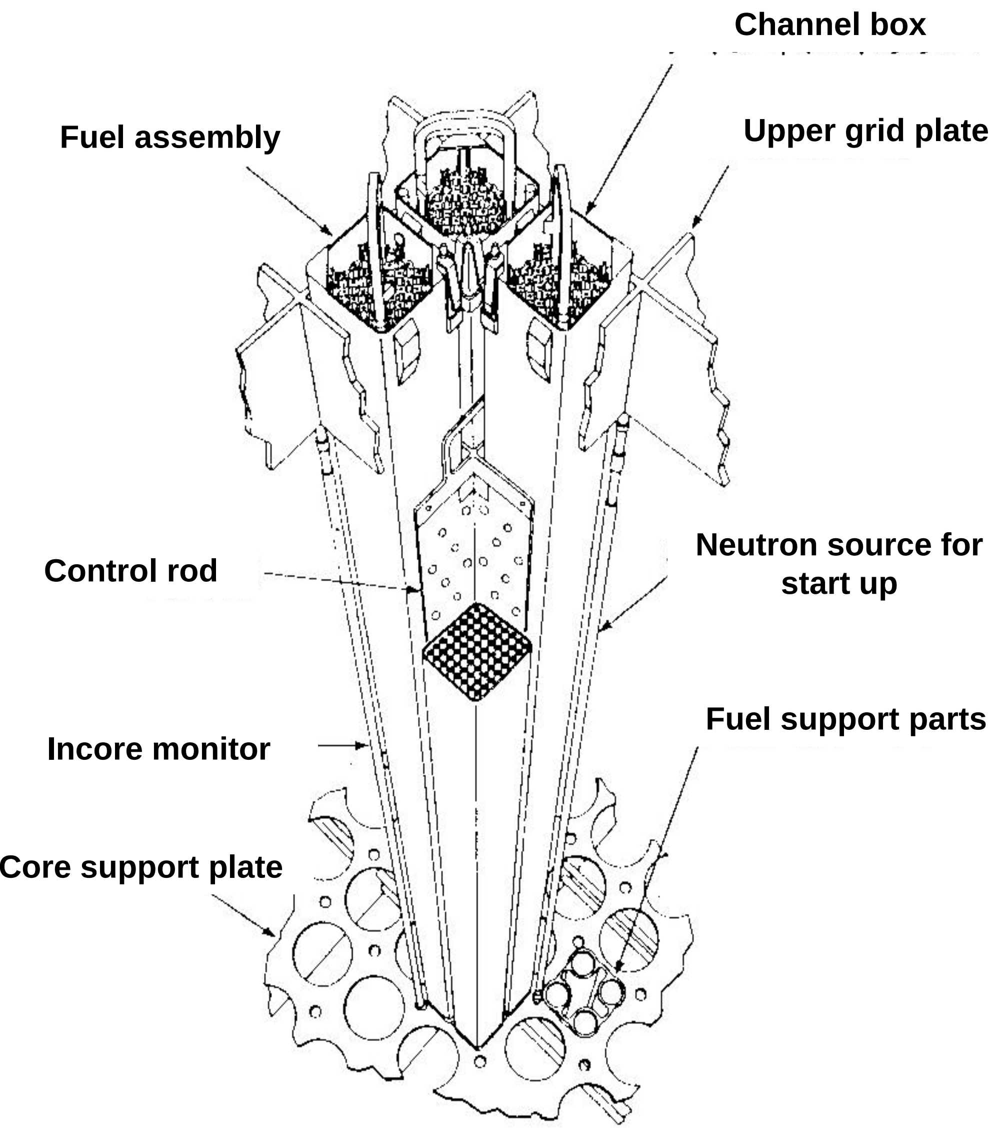
1.2 Structure of fuel assembly





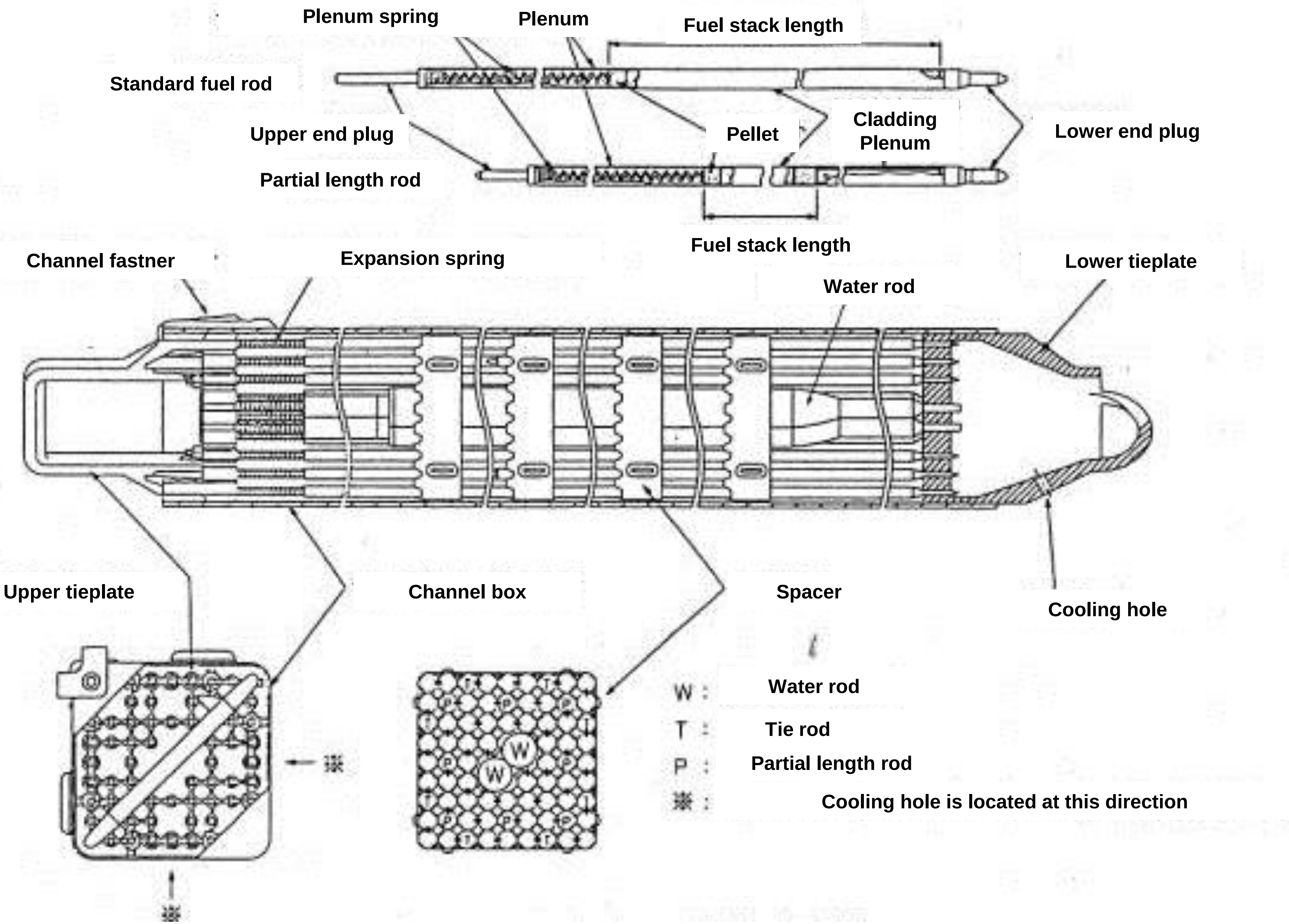
BWR

Structure of BWR core



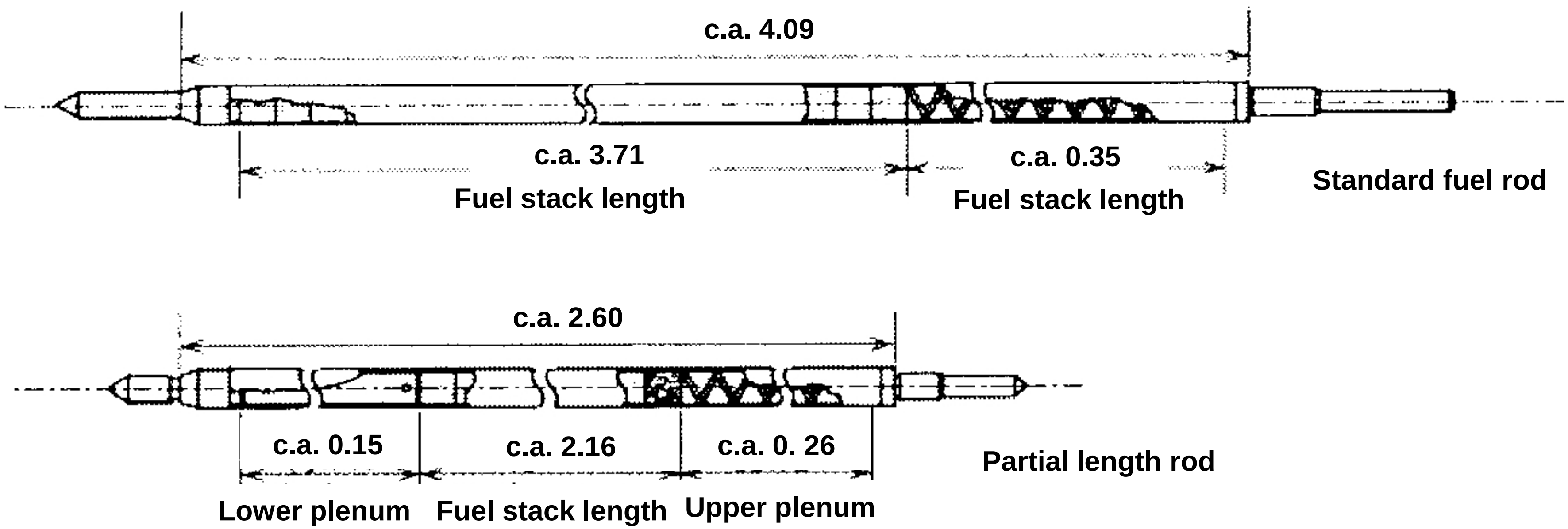
- core consists of units which have one control rod surrounded by 4 fuel assemblies

- Fuel assembly sits on core support and the top of the assembly is supported with upper lattice plate



BWR fuel assembly 9x9 A-type

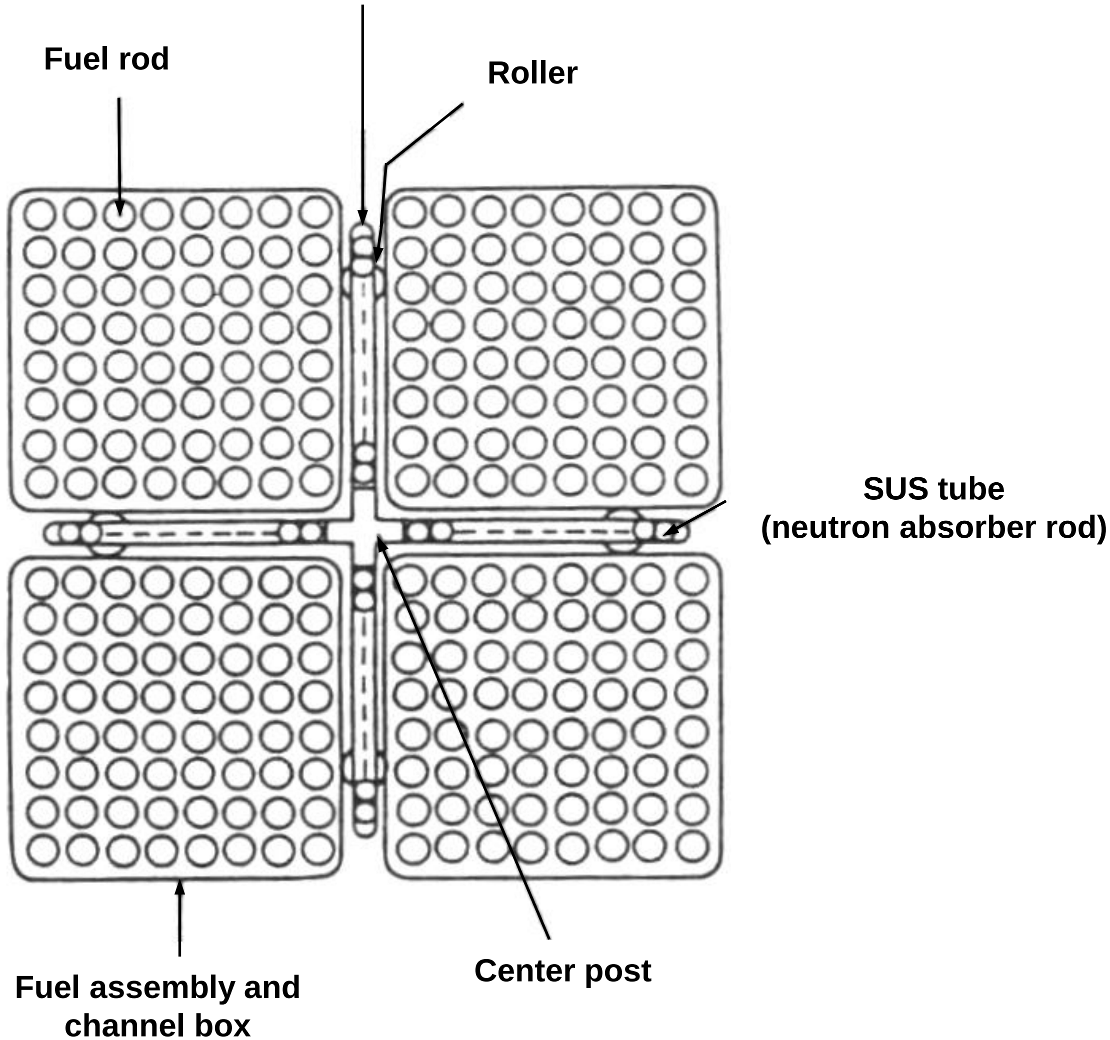
Fuel rod：9 x 9 BWR fuel（A-type）

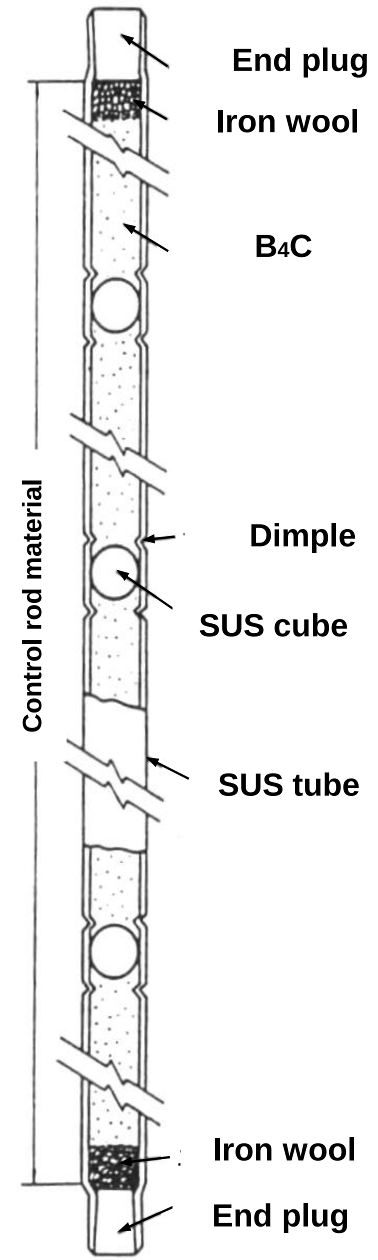


Number in this figure is size for BWR/4, BWR/5 and ABWR (m)

- Pellets are installed in Zr liner cladding and welded with both end plugs with 10bar He filler gas.
- Important design parameters: pellet density, gap size, plenum volume, cladding thickness, He pressure, etc.

Cross section of Control Rod（BWR）

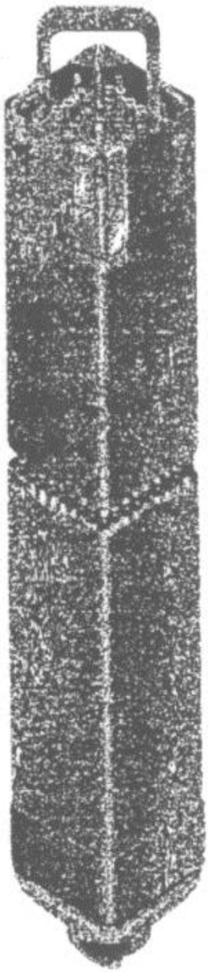




SUS tube (neutron absorber rod)

Detailed figure

Channel box (BWR)



- a side about 14cm, thickness 2-3mm, length about 4m : rectangular tube made of zircaloy-4

Role of channel box

- keep the coolant flow space for assembly
- keep the space for CR insertion
- protection of fuel during handling
- Use for shipping test in reactor(leak fuel test)

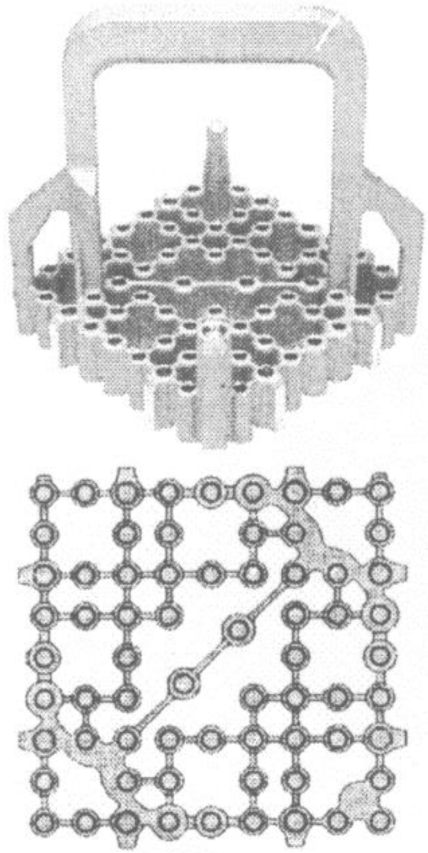
Water rods(Water channel)

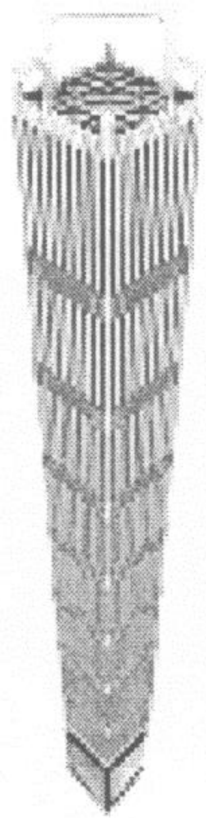
In BWR, coolant water in a fuel assembly surrounded by channel box is under boiling conditions, and outside of channel box is under non-boiling conditions. Therefore, peripheral rods in the fuel assembly is easier to fission due to enough thermal neutrons than the rods in the central part far from non-boiling region. The water rods were introduced to enhance the neutron moderation in the central region of assembly and to increase the fission of the fuel. The non-boiling coolant water flows in the water rods(or water channel) located in the center of the fuel assembly.

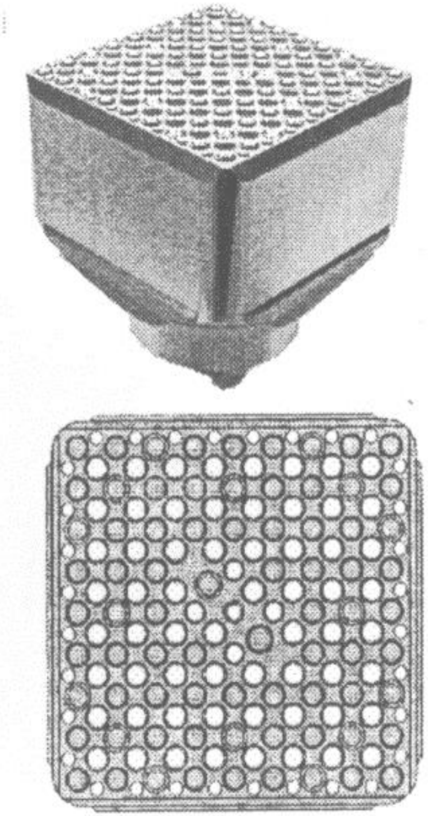
The diameter of water rods increased with the enrichment increase for high burnup use of fuel, to increase the water/ fuel volume ratio within the channel.

Upper tieplate

Lower tieplate(BWR)







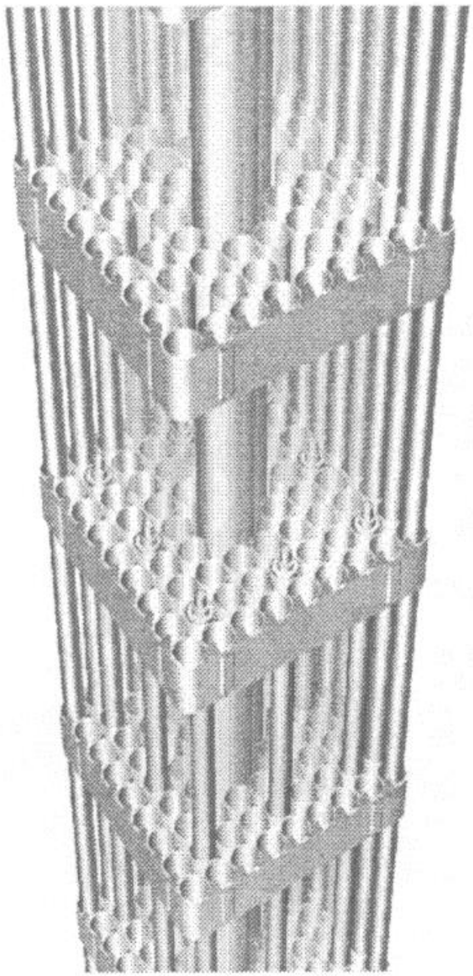
Low pressure drop upper tieplate

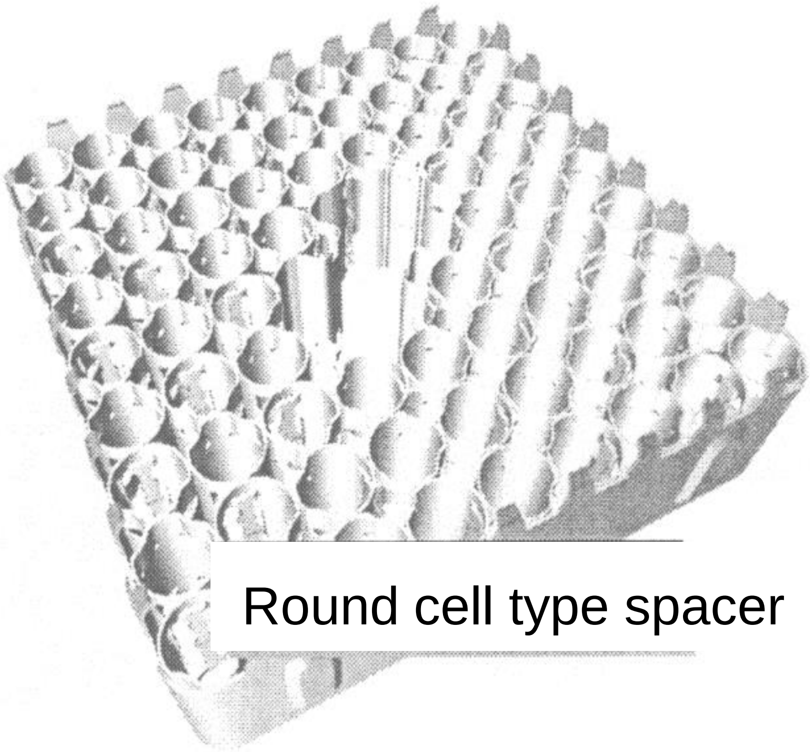
High pressure drop upper tieplate

Reduce the pressure drop in two phase region

Increase the pressure drop in single phase flow region

Stabilize the coolant flow

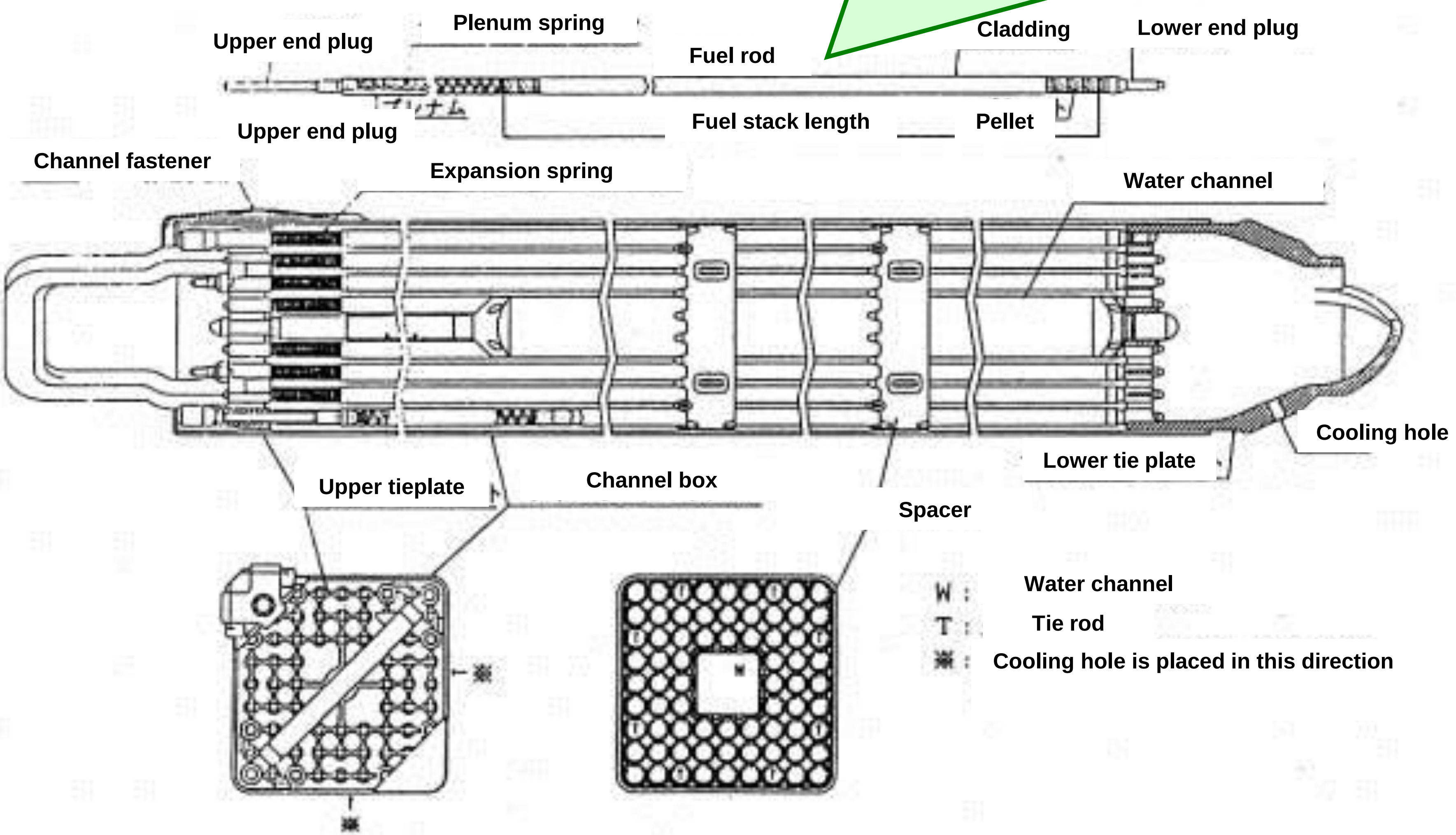




Round cell type spacer

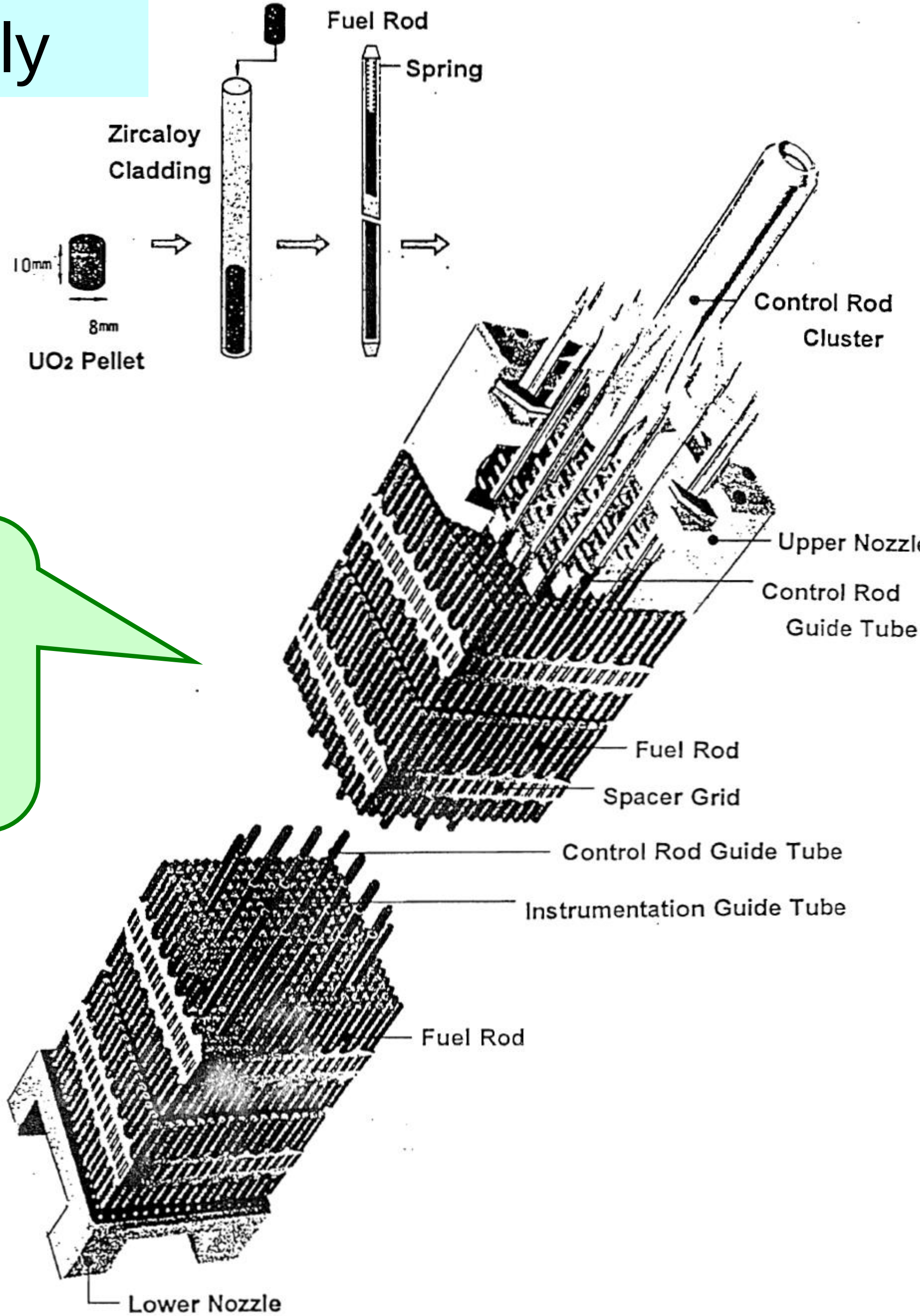
- To keep the horizontal space between fuel rods(BWR)
- Spacer is fixed to water rods and axial position of spacer is fixed.

In BWR fuel assembly, water channel, tie rods and spacer are main framework.



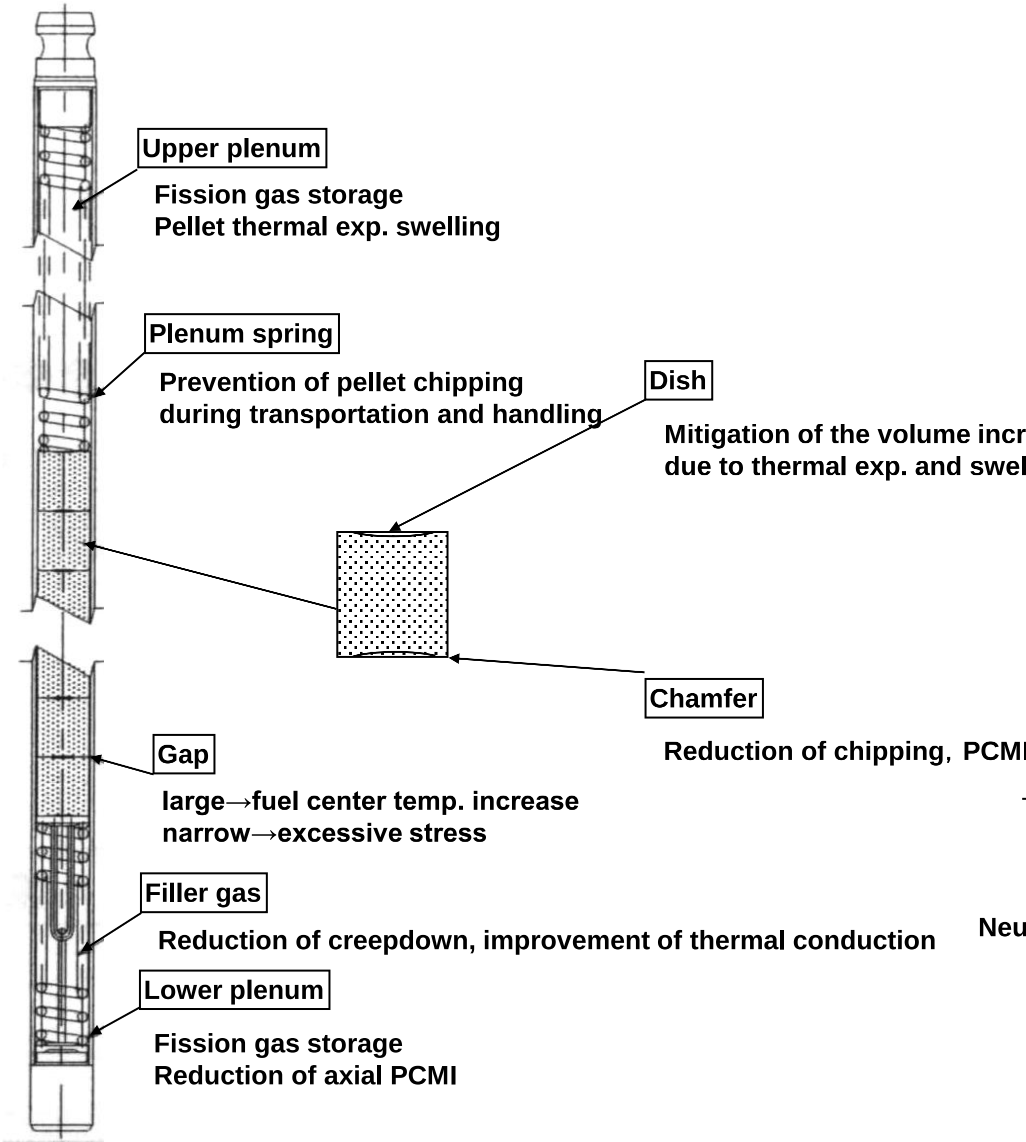
BWR fuel assembly 9x9 B-type

PWR fuel assembly

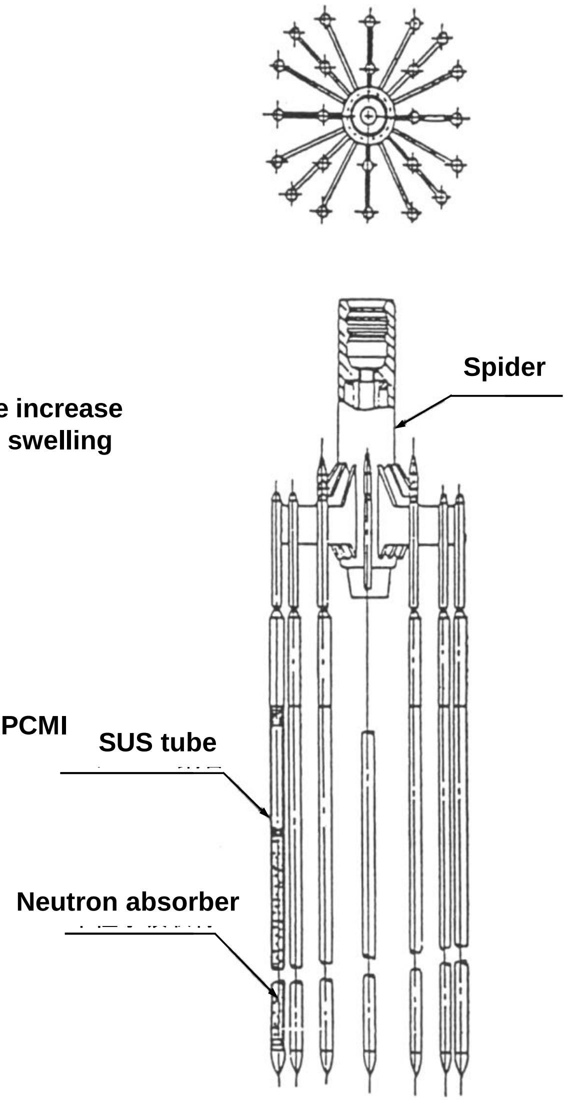


In PWR fuel assembly, thimble tubes and grids are main framework

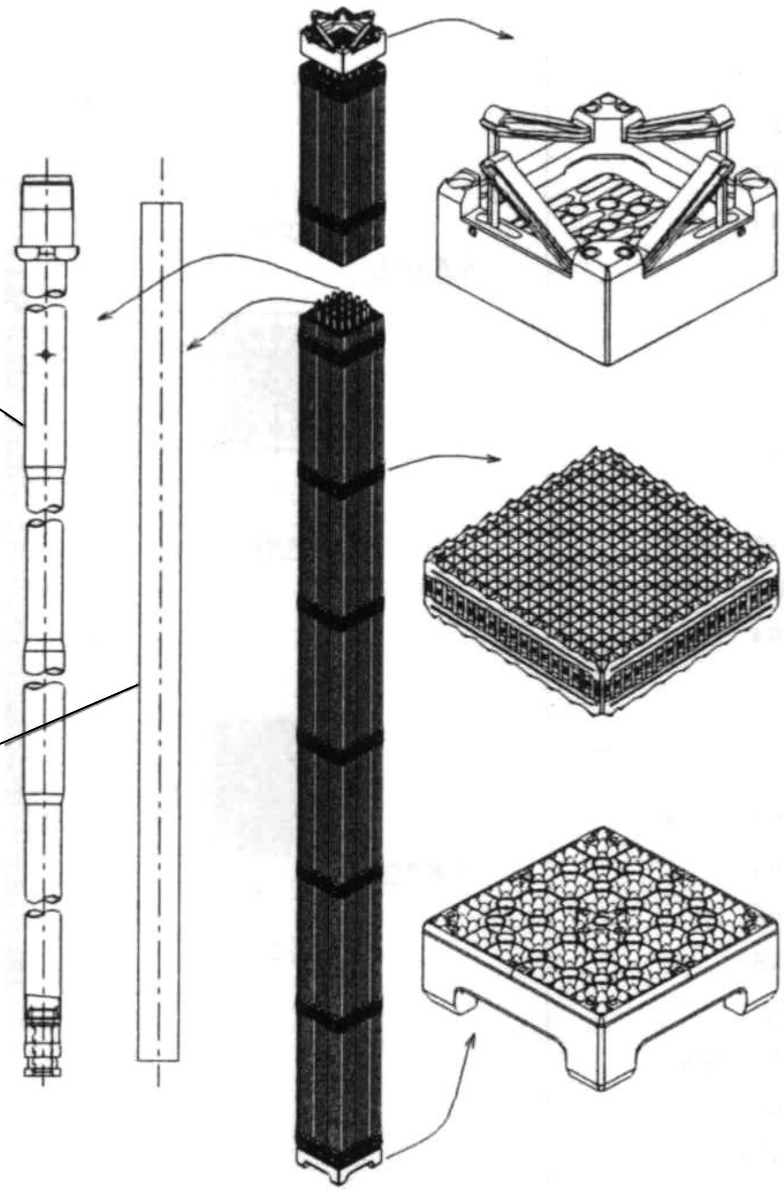
Structure of fuel rod



Structure of Control rod cluster



Structure of fuel assembly(PWR)



Control rod guide tube

Insertion root of CR
Mitigation of shock of CR drop
Fixing of grid
Structure material

Instrumentation guide tube

Guide and support of neutron detector

Upper nozzle

Structure material
Positioning in core
Fuel handling
identification
Prevent of assembly lifting

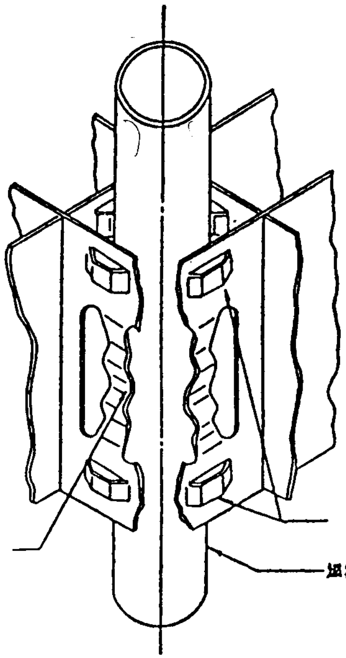
Grid .

Support fuel rod
Keep the space between rod
Adjust axial rod elongation
Adjust fuel diameter change
Reduce fuel rod bending
Increase removal of rod heat

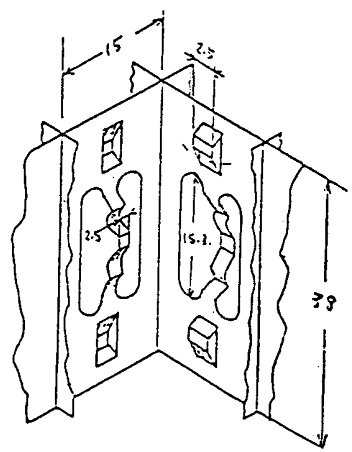
Lower nozzle

Structure material
Positioning in core
Prevention of debris

Structure of grid (spacer)

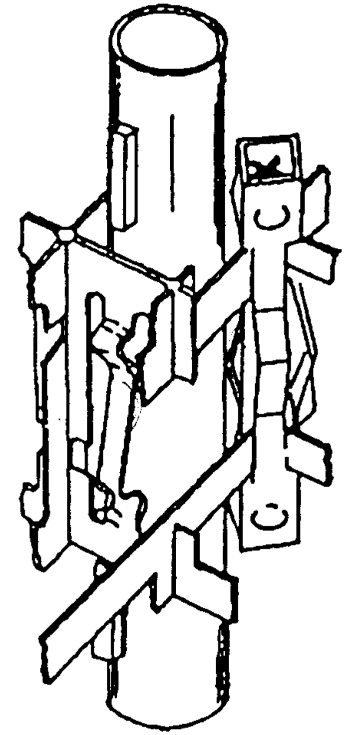


R type



Nuclear ship
MUTSU

(a) PWR grid



(b) BWR spacer(old type)

BWR uses round cell type spacer

Contact to fuel rod with 3 point support.

The axial elongation/shrinkage of fuel rods are allowed

Mechanical design of fuel rod

Requirement

・Cladding should not be failed during abnormal transient in operation (cladding should not be failed systematically with penetrated defects by mechanical load)

(criteria for judgment) conditions to prevent the fuel failure due to over strain, “average circumferential strain of cladding should be below 1%”

Fuel design evaluation for BWR

To evaluate the rod power(Linear Heat Rate) to give 1% plastic strain of cladding: LHR for margin safety

In the safety evaluation during the abnormal transient in operation, the evaluated maximum LHR should not exceed the LHR for margin safety.

Guideline for fuel design

Guidline12. Fuel design

1. Fuel assembly should be designed to keep the integrity regardless the various factors which occur during the in-rector operation.

2. Fuel assembly should be designed not to deform excessively during transportation or handling.

“various factors which occur during the in-rector operation” means pressure difference between outside and inside of the rod, irradiation of fuel or other material, changes of pressure/temperature due to change of load, chemical effects, static/dynamic load, deformation of pellet, gas composition change in fuel rod, etc.

Evaluation of BWR fuel rod design

BWR fuel rod is designed under the following standards based on the guideline for safety design judgment.

（1） average circumferential plastic strain of cladding should be less than 1%.

（2） the stress in the cladding should be less than acceptable stress.

（3） Accumulated fatigue coefficient should be less than 1.

$$C = \Sigma (n_i / N_i) < 1.0$$

Design criteria for PWR fuel

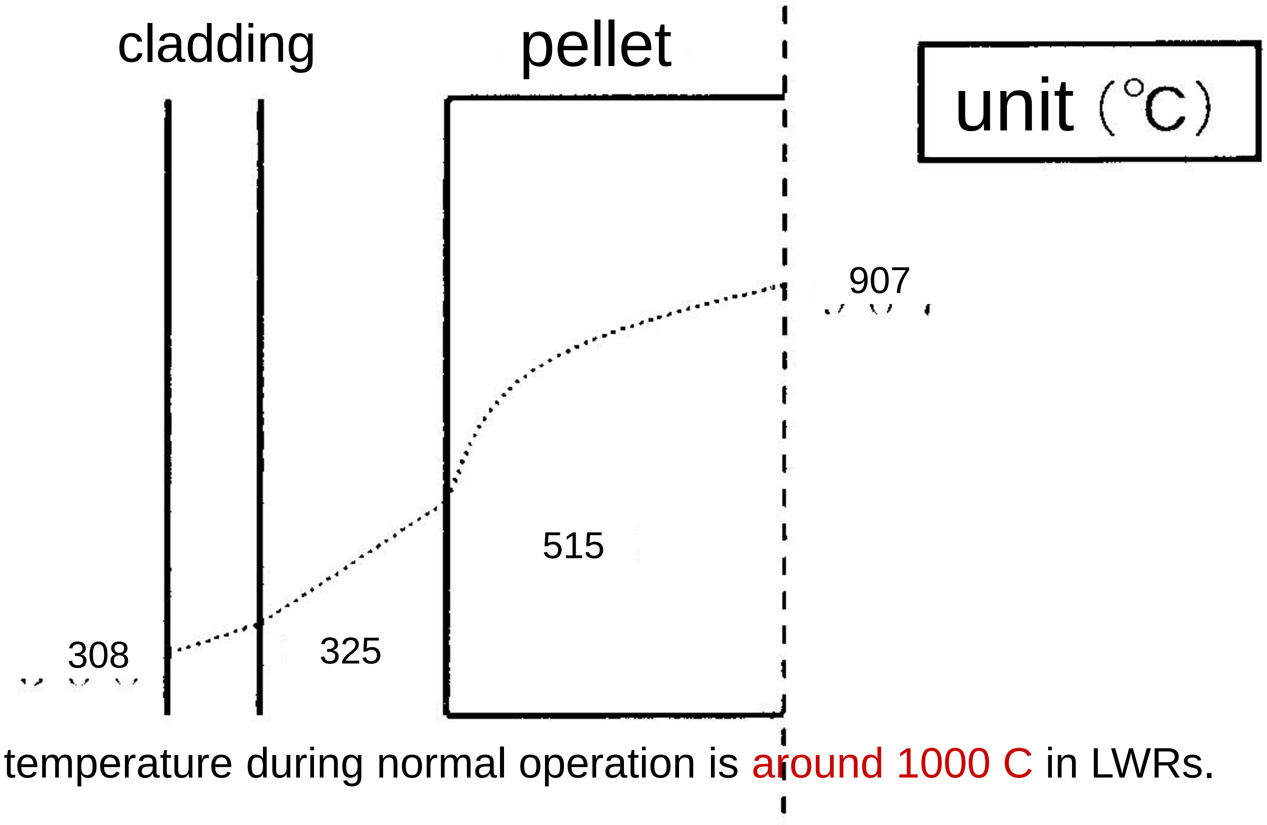
(1) Design criteria for fuel element

Fuel element should be designed to satisfy the following criteria during normal operation and abnormal transients in operation.

① Fuel center temperature should be below the melting point of UO_2
② Pressure in the rod should not exceed the limit to open the closed gap between cladding/pellet due to the creep deformation of cladding outward, during normal operation.
③ The stress in the cladding should be less than yield strength of Zirconium alloy(zilcaloy-4)
④ The circumferential extension strain of cladding during abnormal transient should be less than 1%.
⑤ Accumulated fatigue coefficient should be less than designed life limit.

Example of temperature calculation
(radial temperature distribution)

Example of calculation
Calculation input：17 x 17 4loop core　LHR of rod: 179W/cm



The fuel center temperature during normal operation is around 1000 C in LWRs.

1.3 special unit for nuclear fuel

(Burnup)

Definition：produced thermal energy per fuel unit weight（MWd/tU,MWd/tUO$_2$）

In case of Enrichment 3%、density 95%、200MeV/fission

Thermal unit nuclear unit

10,000MWd/tU=8,814 MWd/tUO$_2$=1.066 %FIMA=2.475 x 10^{20} fissions/cm^3

FIMA: fraction of initial metal atom

(Linear Heat Rate)

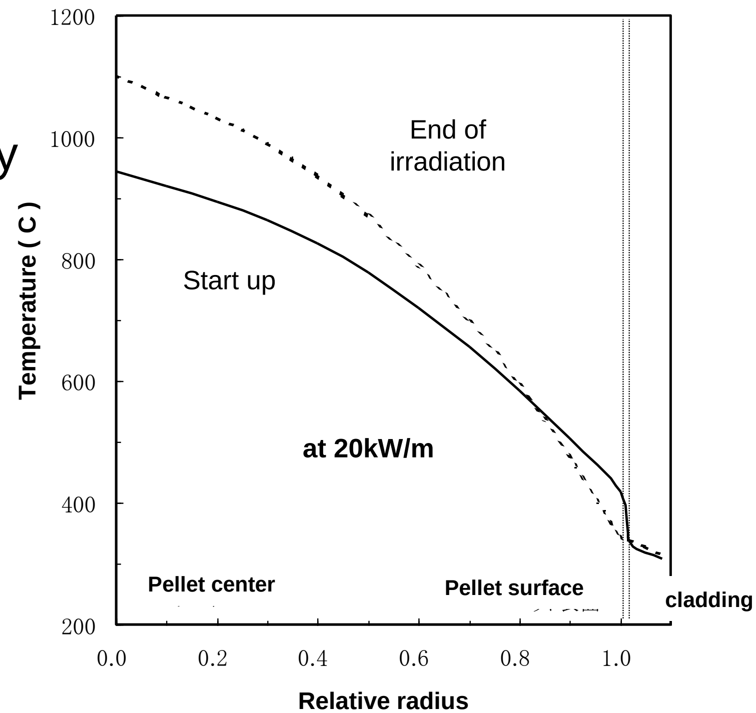
Definition：thermal energy generation rate per unit fuel rod length（kW/m, W/cm）

Relation between LHR and fuel center temperature

$$\frac{q}{4\pi} = \int_{T_S}^{T_C} k(T)dT$$ (thermal conductivity integral)

$$q = \int_{r_{in}}^{r_{out}} 2\pi r F(r) dr$$

q：LHR（kW/m）
k(T)：thermal conductivity（W/m.K）
Ts：fuel surface temperature（K）
Tc：fuel center temperaure（K）
F(r)：function for radial power distribution



1.4 Factors affecting fuel behavior and their interactions

➢Temperature distribution in fuel rod and changes due to irradiation(burnup)

☆Temperature depends on gap width and LHR.

(1) Temperature⇔deformation⇔gap

pellet deformation（thermal expansion, densification, creep）

cladding deformation（ridging, creep down）

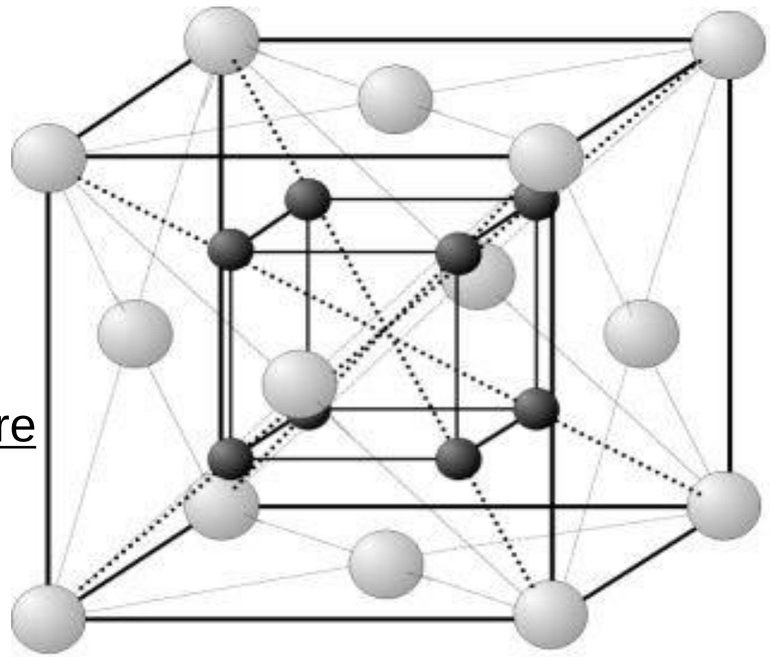
(2) Temperature⇔structure change（accumulation of fission products: thermal properties change）

Density, crystal structure and melting point of UO_2 and PuO_2

・crystal structure (CaF$_2$ type) → (U,Pu)O_2 forms solid solution in all composition

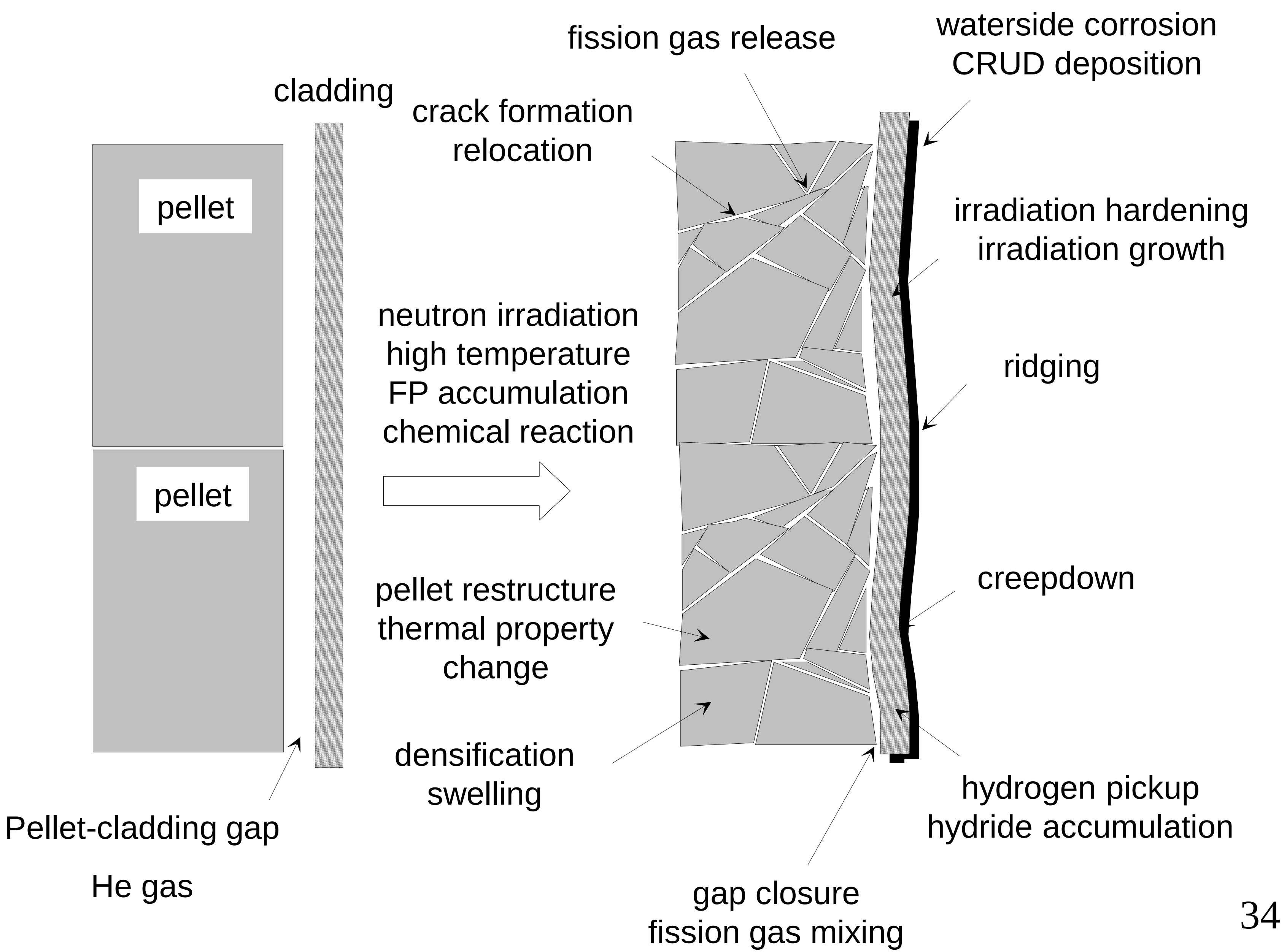
material	U(Pu)fraction (wt%)	Density (g/cm^3)	crystal structure		Melt. T (C)
			type	a_0 (nm)	
UO_2	88.2	10.96	**CaF$_2$ type**	5.470	2865
PuO_2	88.2	11.46	**CaF$_2$ type**	5.396	2390

CaF$_2$ type crystal structure



(: U^{4+}, : O^{2-})

How is fuel on power conditions：general images



(1)-1: Cracking, thermal expansion and relocation of pellet

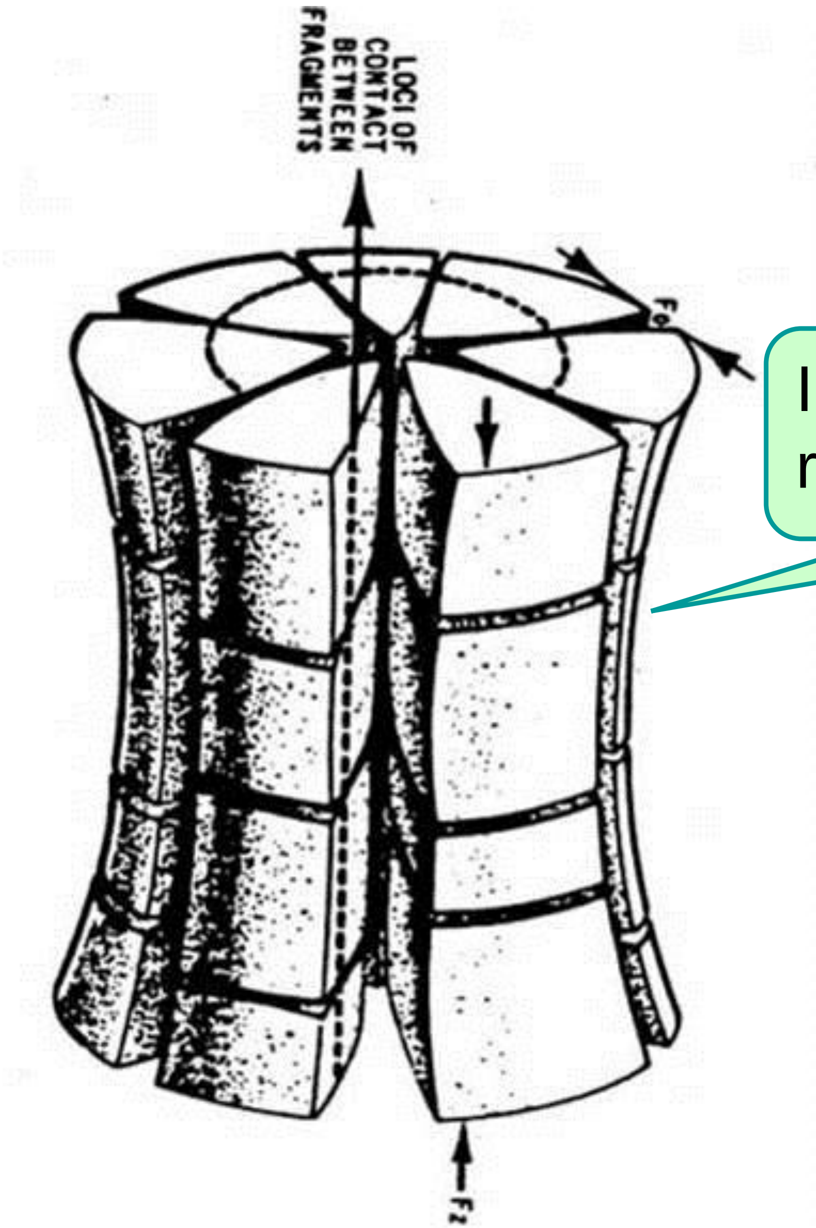
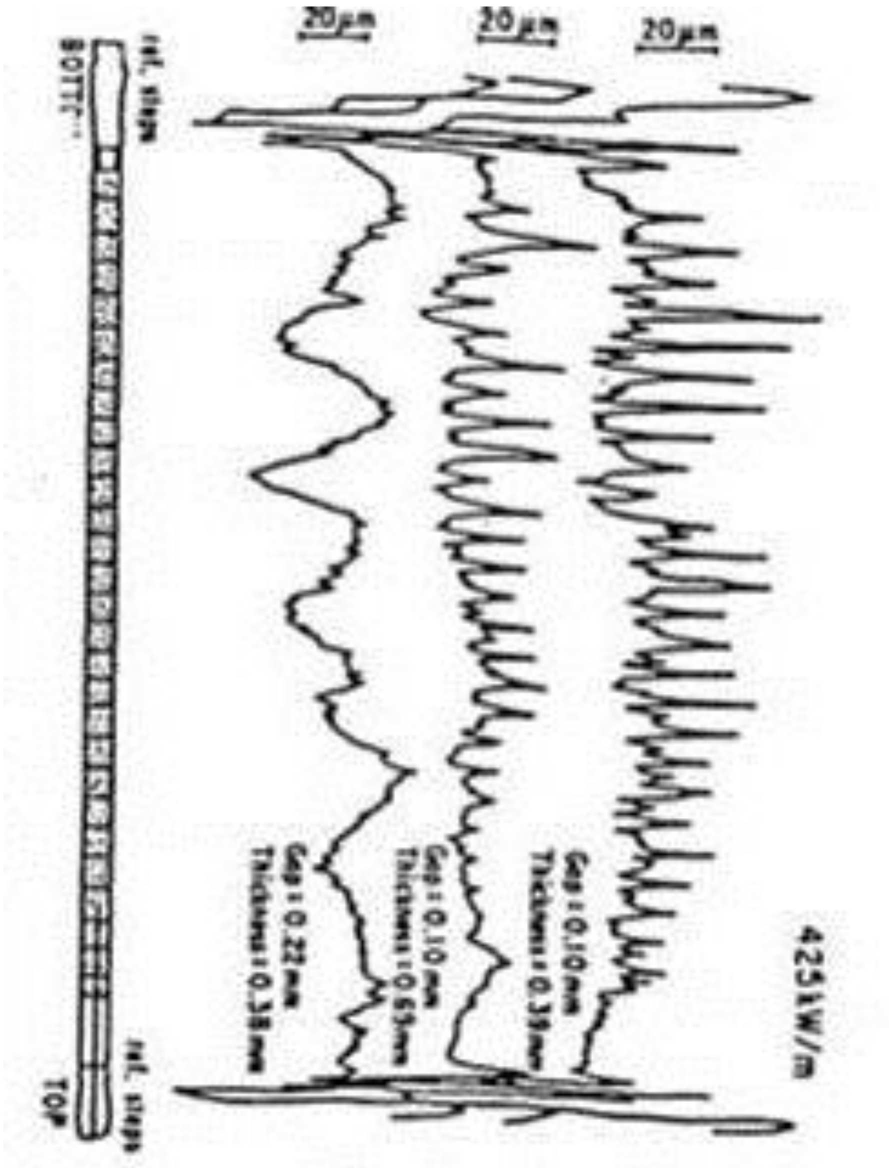
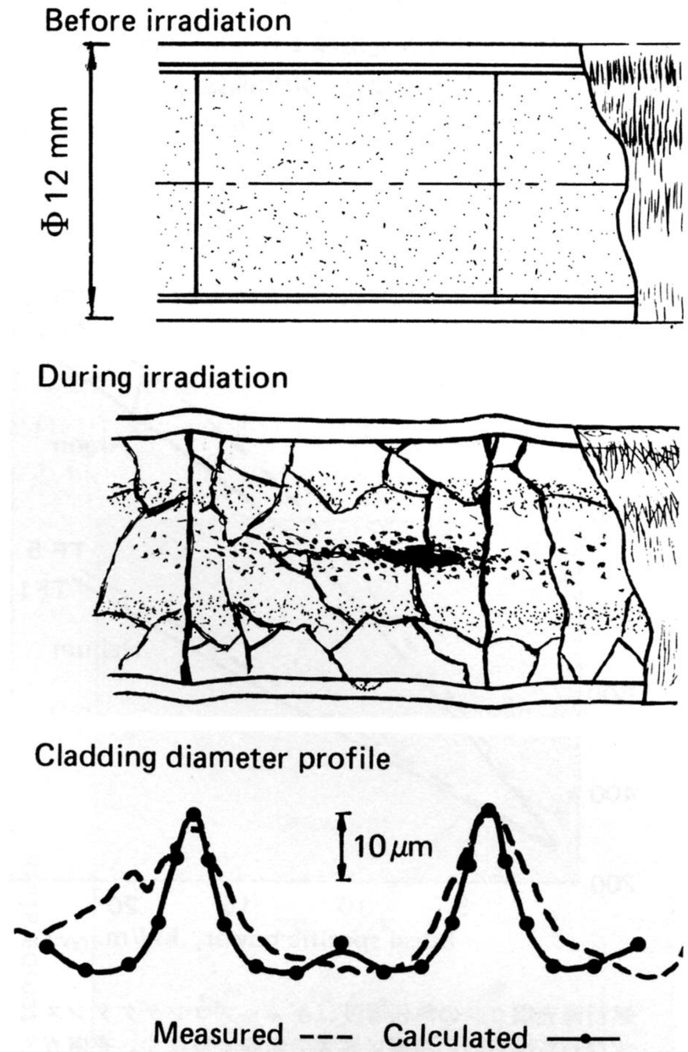


Image of cracking and relocation of pellet

(1)-2 **PCMI** : Pellet-Clad Mechanical Interaction

Thermal expansion of pellet results in the deformation of cladding due to PCMI.

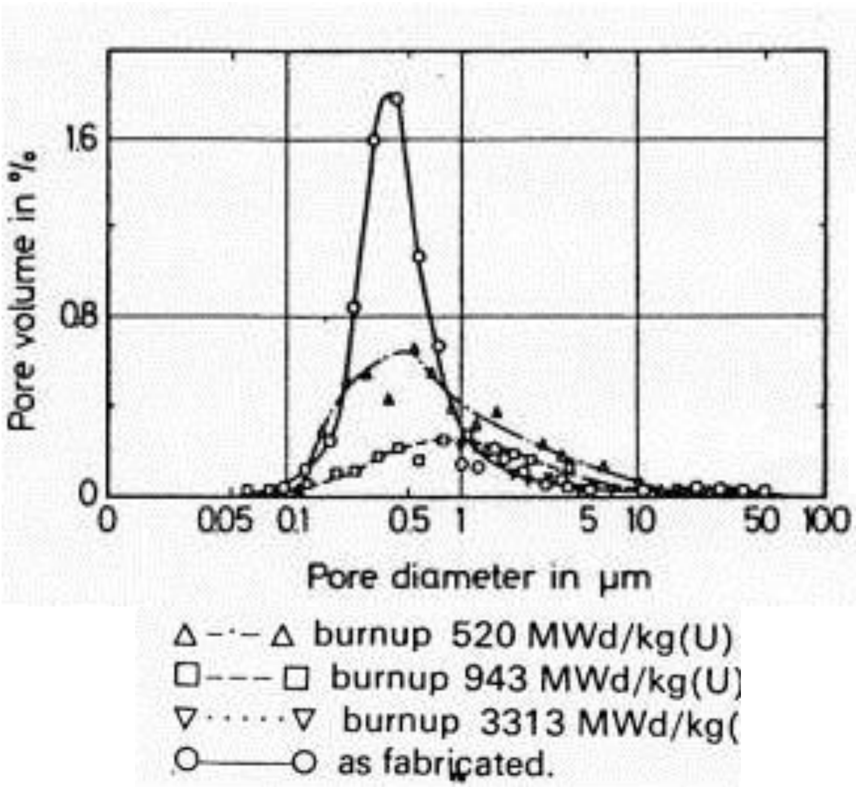




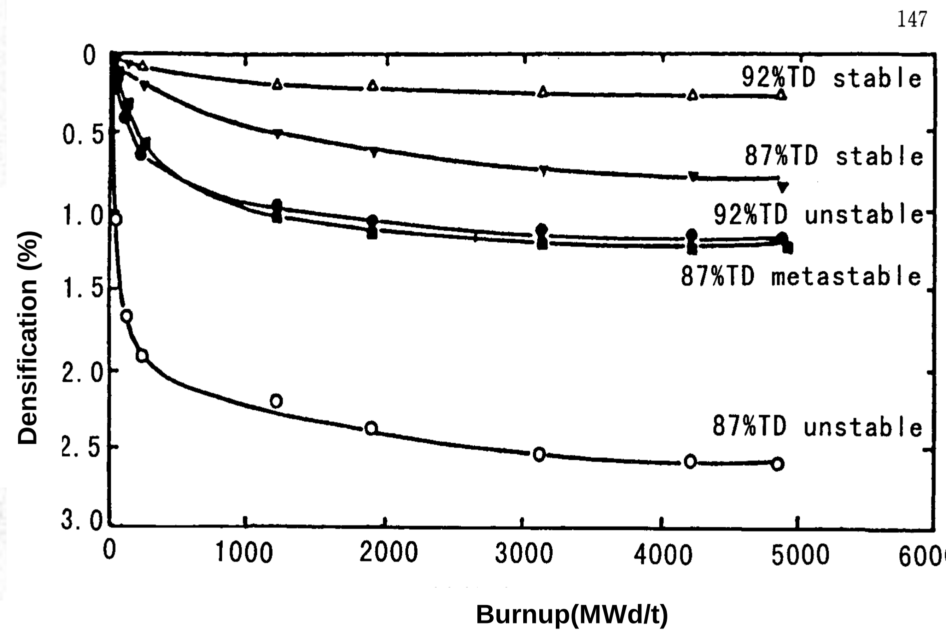
(1)-3. Densification of pellet

Micro porosity from manufacture shrink or disappear at the beginning of irradiation

☆countermeasure：increase of pellet density at manufacturing（93→95−97%TD）



Pore size distribution change in UO_2 due to irradiation



UO_2 pellet stack length change (shortening) due to irradiation

Densification/Swelling

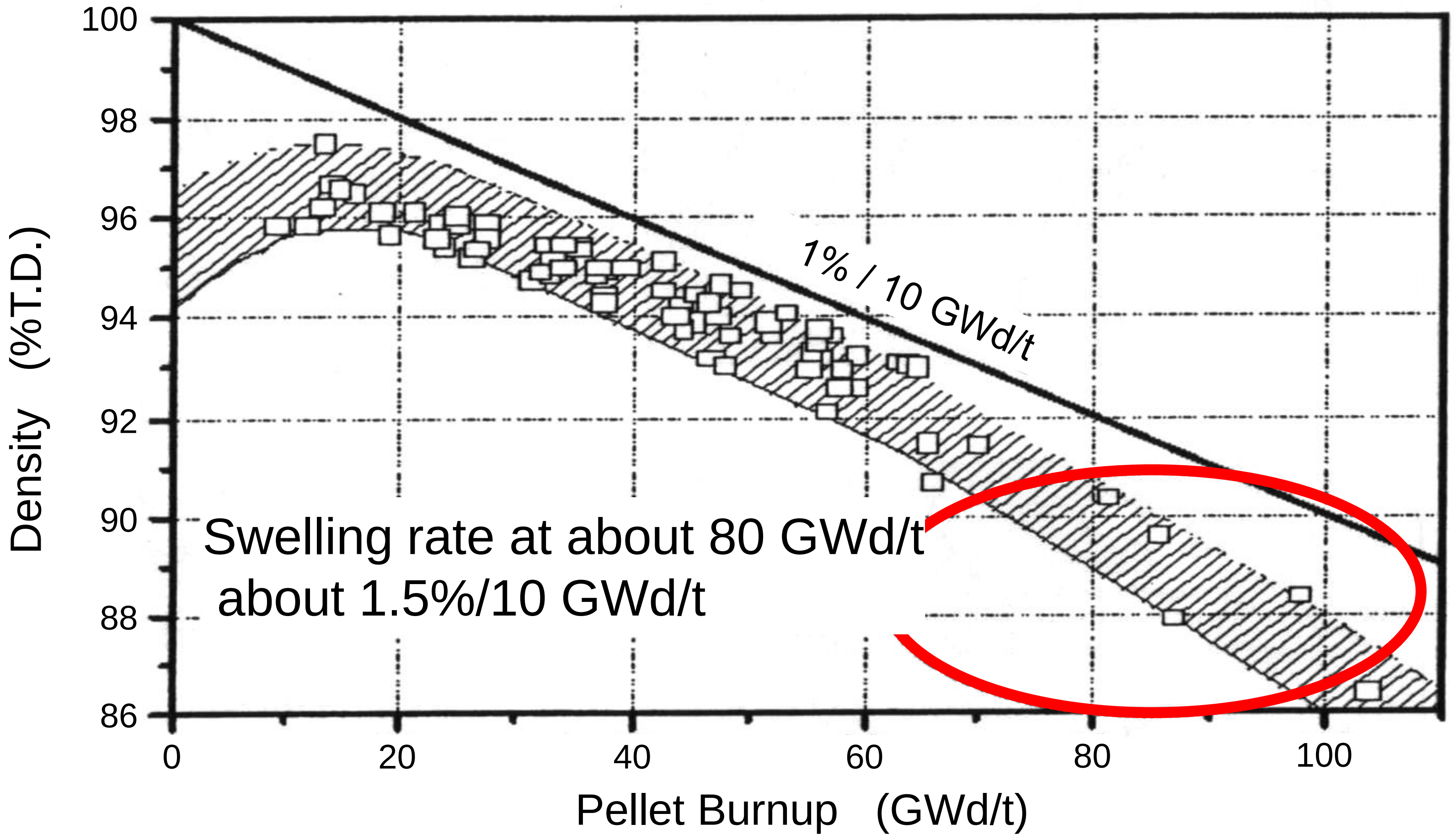
(**Densification**)

- Pores from manufacturing shrink/disappear due to the fission fragments passage, and the density of UO_2 pellet increases.
- Mainly occurs, at low temperature less than 1000 C and low burnup less than 10GWd/t.

(**Swelling**)

- Density of UO_2 decreases due to FPs(Fission Products) accumulation
- Solid FP swelling
  - Precipitation or solution of solid FPs in UO_2 matrix
  - Independent of irradiation conditions(temperature)
- FP gas bubble swelling
  - Gas bubbles grow in grain or grain boundary
  - Strongly depends on irradiation conditions(temperature,rod power)

UO_2 swelling at high burnup



(2) Accumulative effects of fission and structure change of pellet

①change of radial power density distribution in pellet

－accumulation of FPs（solid atoms, gas atoms）and irradiation defects－

②formation of fission gas bubbles and fission gas release

③swelling

④degradation of thermal conductivity of pellet

⑤formation of rim structure

⑥change of O/M ratio：consumption of oxygen due to inner surface oxidation of zircaloy cladding

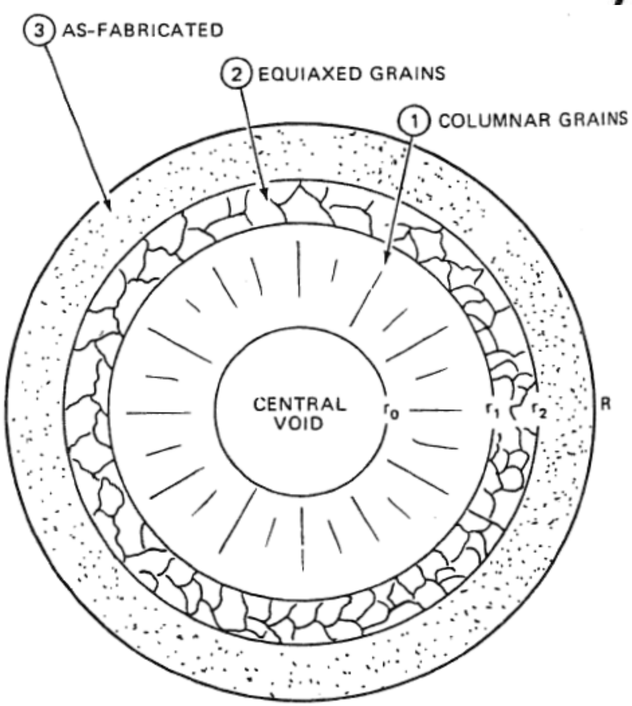
Restructuring of pellet

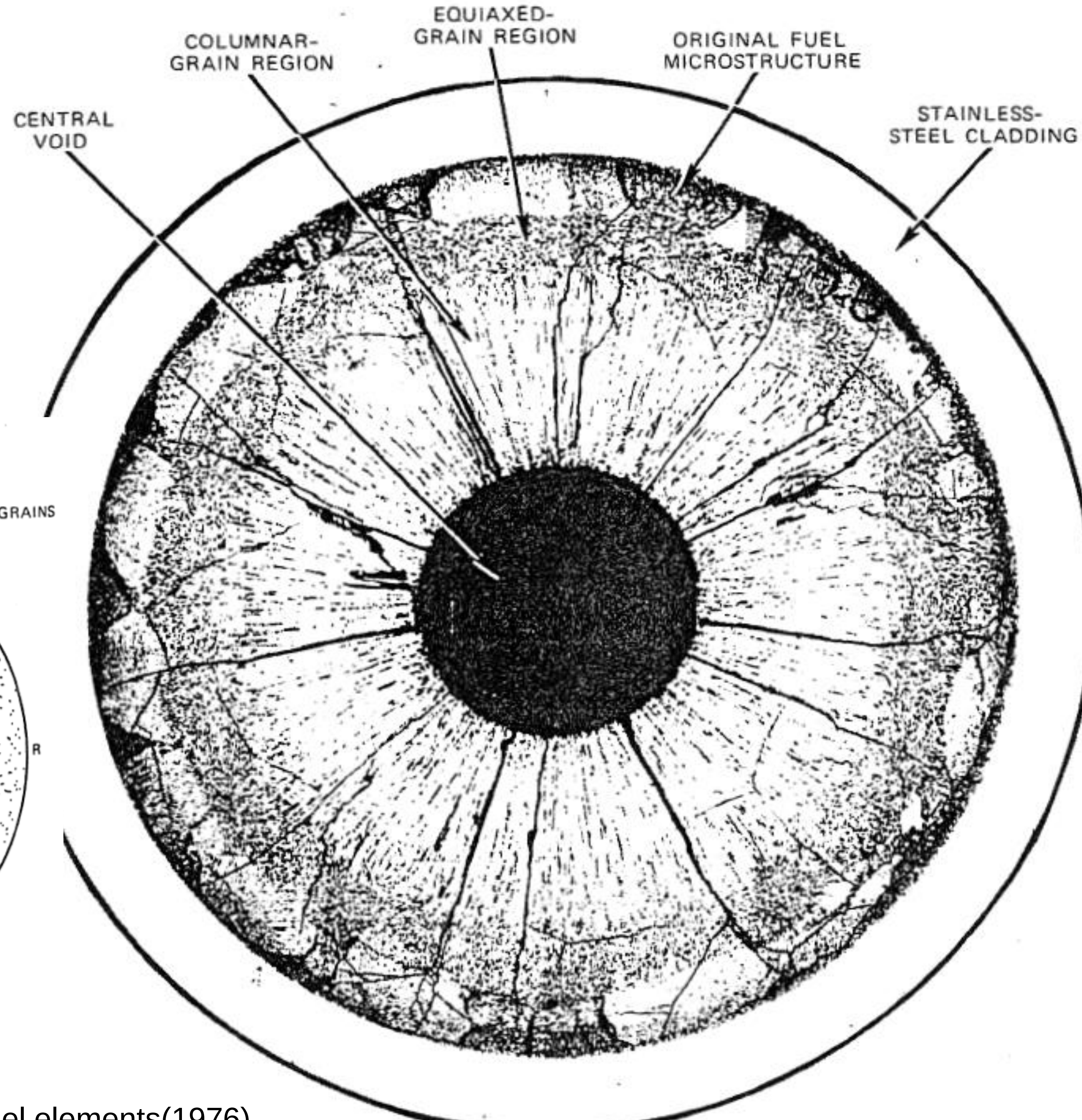
Cracking and relocation of pellet

- Cracking during power up (from outer to inner)
- Cracking during power down(from inner to outer, stop at circumferential cracks)
- Phenomenon of gap width reduction due to radial movement of pellet fragments(relocation)

- **Structure change**
- equi-axed grain growth
- columnar grain growth
- fuel center melting

Restructuring at high power irradiation (FBR fuel)

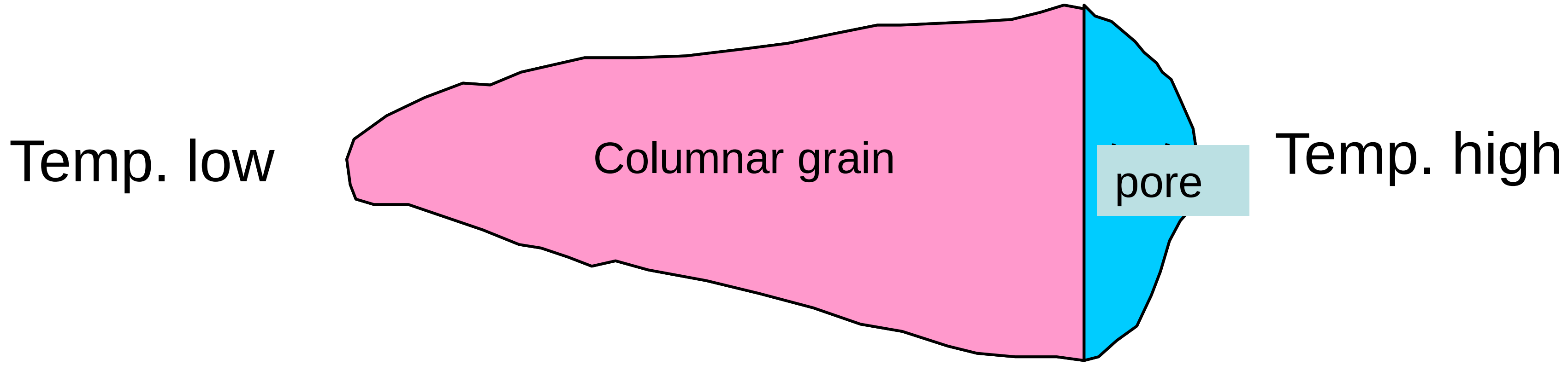




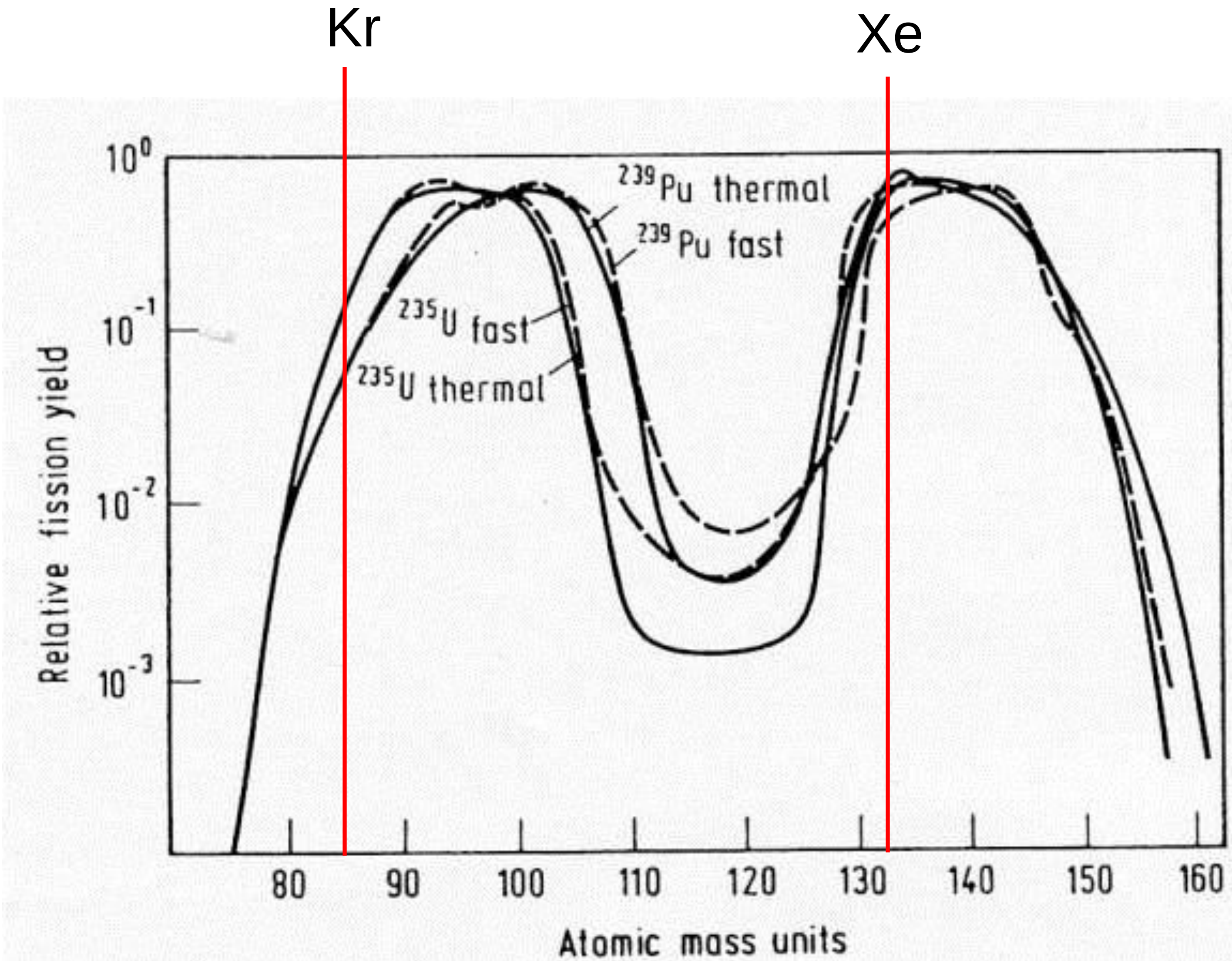
D.R.Olander
Fundamental aspects of nuclear fuel elements(1976)

Formation mechanism of columnar grain

- Vapor pressure of UO_2 is high at high temperature
（above about 1700 C）.
- Pores coalesce and form large lenticular pore.
- UO_2 evaporates at hotter side of lenticular pore and deposits at colder side.
- Total pores move to hotter side.
- Columnar grain formed behind the movement of pores.



Chain yields as a function of mass number of the chain for fast- and thermal neutron flux and for 235-U and 239-Pu



Fission yields and the state of fission products

Fission yield of stable and long life FPs
(^{235}U, thermal)

	%Yield		%Yield
Se	0.4	Sn	0.1
Br	0.3	Sb	0.1
Kr	**3.8**	Te	2.5
Rb	1.3	I	1.0
Sr	6.2	Xe	**20.0**
Y	4.8	Cs	20.0
Zr	36.9	Ba	6.7
Mo	25.0	La	6.6
Tc	6.1	Ce	12.3
Ru	8.3	Pr	5.9
Rh	4.9	Nd	20.5
Pd	1.4	Pm	2.3
Ag	0.2	Sm	1.9
Cd	0.1	Eu	0.2

:fission gas :volatile FP

(total 200%)

Chemical state of FPs and actinoids in UO_2 pellet (D.R. Olander)

Chemical group	Physical state
Zr and Nb*	Oxide in fuel matrix; some Zr in alkaline earth oxide phase
Y and rare earths†	Oxide in fuel matrix
Ba and Sr	Alkaline earth oxide phase
Mo	Oxide in fuel matrix or element in metallic inclusion
Ru, Tc, Rh, and Pd	Elements in metallic inclusion
Cs and Rb	Elemental vapor or separate oxide phase in cool regions of fuel
I and Te	Elemental vapor; I may be combined with Cs as CsI
Xe and Kr	Elemental gas

Xe and Kr is fission gas and about 25% of FPs

Fission gas release（FGR）

Individual physical processes that contribute to fission gas release in fuel（D.R.Olander）

1. Production of gases(Xe,Kr) by fission
2. Nucleation of gas bubbles in matrix
3. Growth of gas bubbles by atomic migration of fission gas atoms to existing bubbles.
4. Re-solution of the gas atom within the bubble.
5. Migration of the bubbles, either as a randum-walk process in the absence of directed forces acting on the bubble or as biased motion when such forces are present.
6. Coalescence of bubbles moving either in a random or directional fashion.
7. Interaction of bubbles with the crystal defects(dislocation and grain boundaries).
8. Release of fission gases, either to external surface or internal surface such as grain boundaries. When the bubbles on grain boundaries become sufficiently large and numerous, they can link up and release gas to one of the external surface.

(9.)Release of fission gas by direct flight of energetic fission fragments out of an external surface.(small and significant only at low temperature.)

Schematic illustration of mechanism of fission gas release

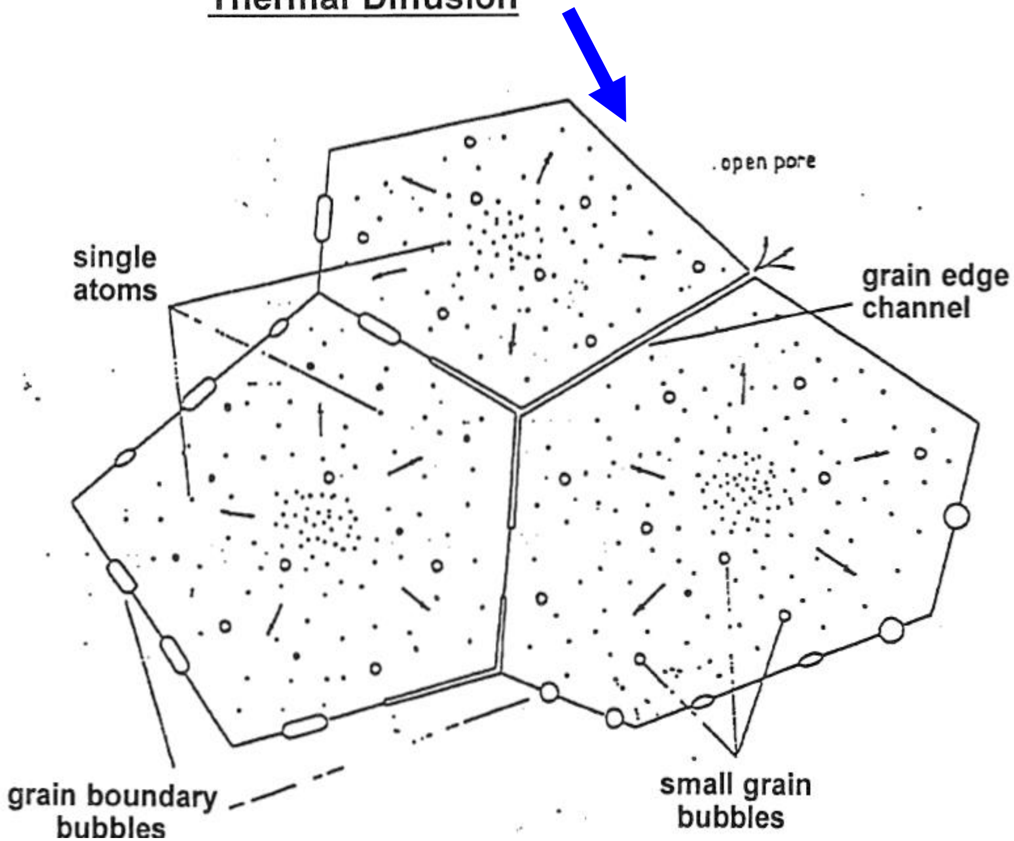
Mechanism of FGR :

- **Very low release at low temperature below about 1000 ℃**

  Recoil, Knockout, Athermal Diffusion

- **Dominant release at high temperature above 1000 ℃**

  Thermal Diffusion



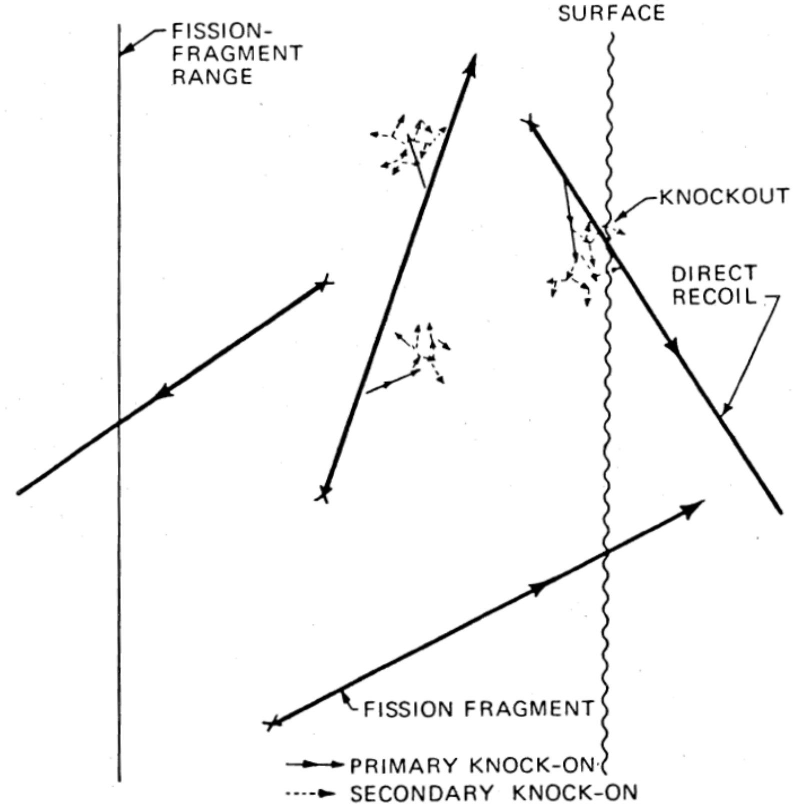
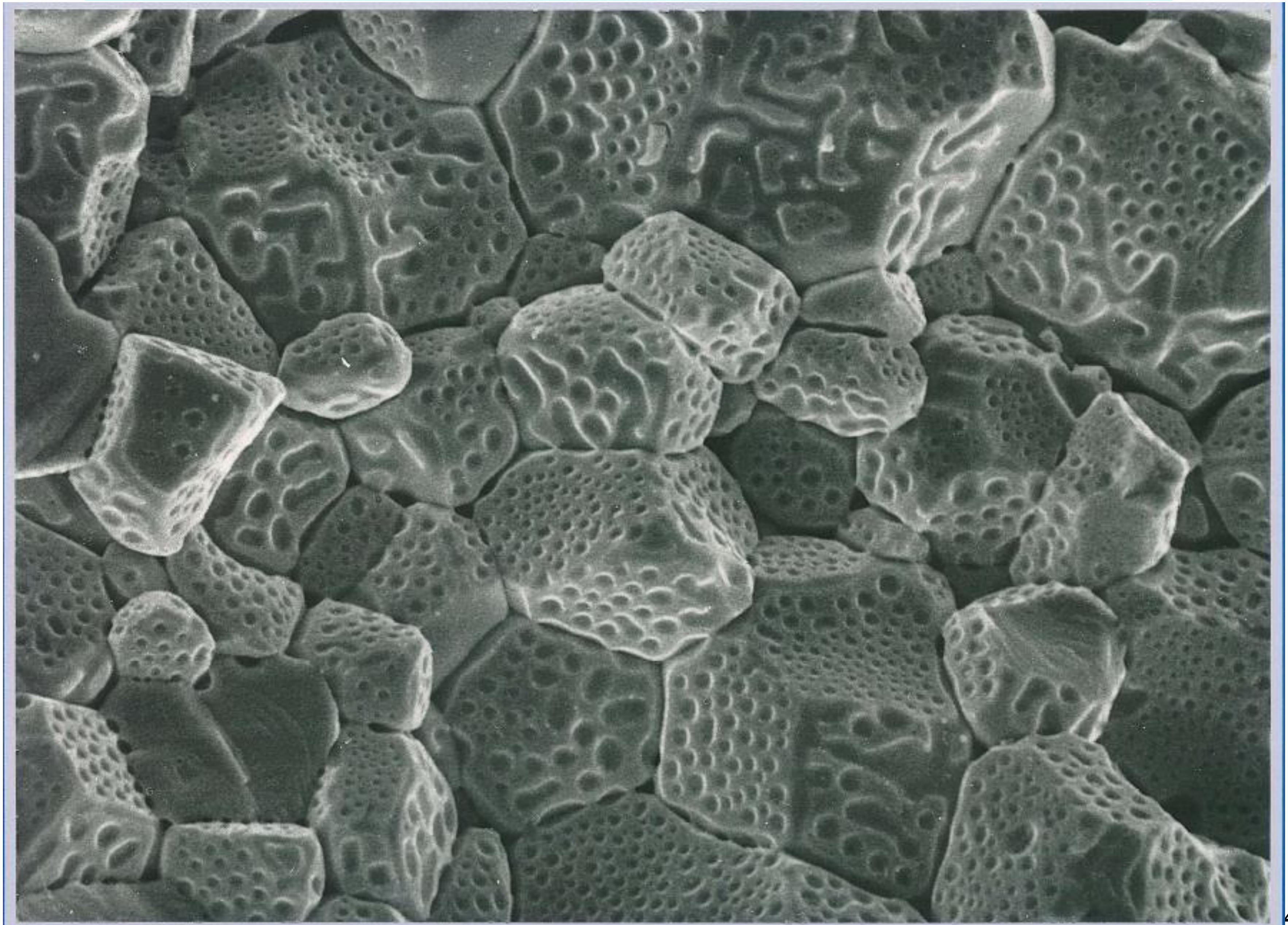
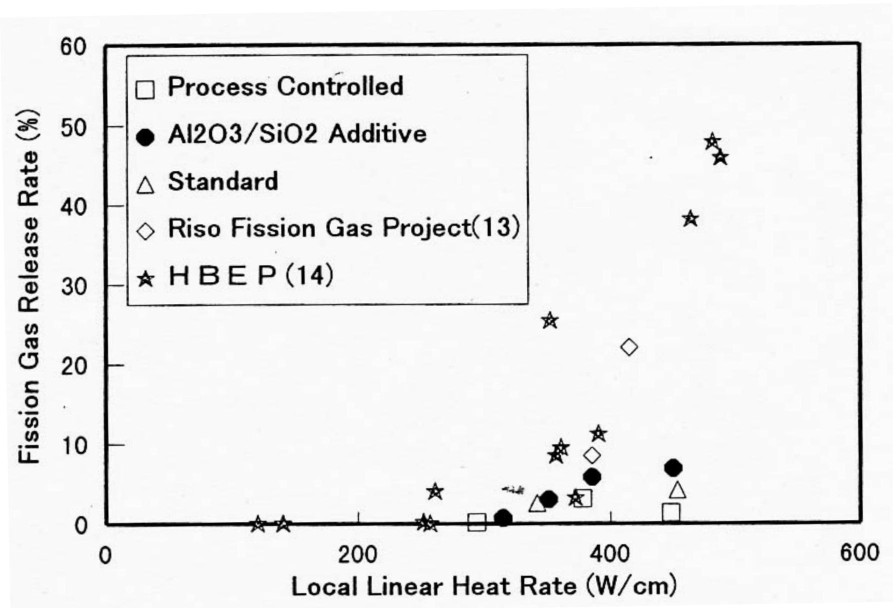


Fig. 15.7 Fission-gas release by direct recoil and knockout.

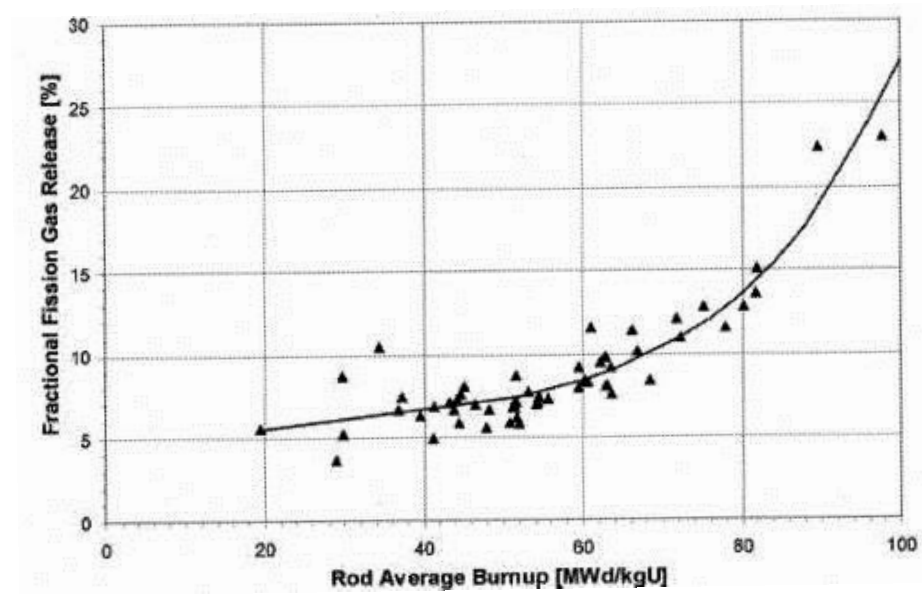
Fission gas bubbles on grain boundary



Fission gas release



LHR dependence of FGR



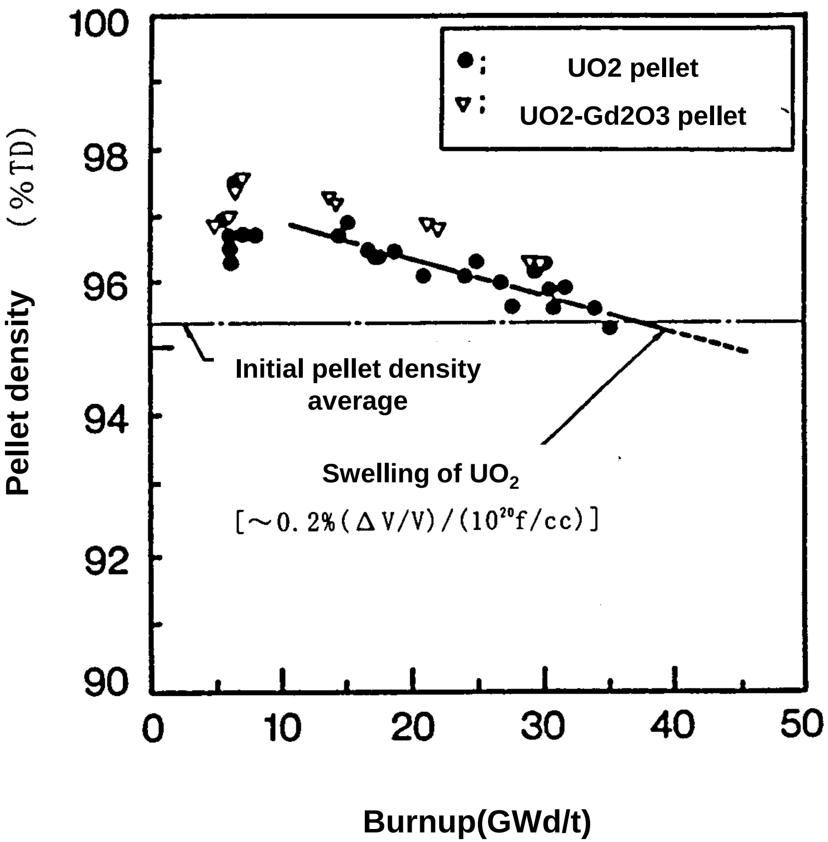
Burnup dependence of FGR

Fission gas release increases at high LHR and at high burnup.
It results in the inner pressure increase of the rod.

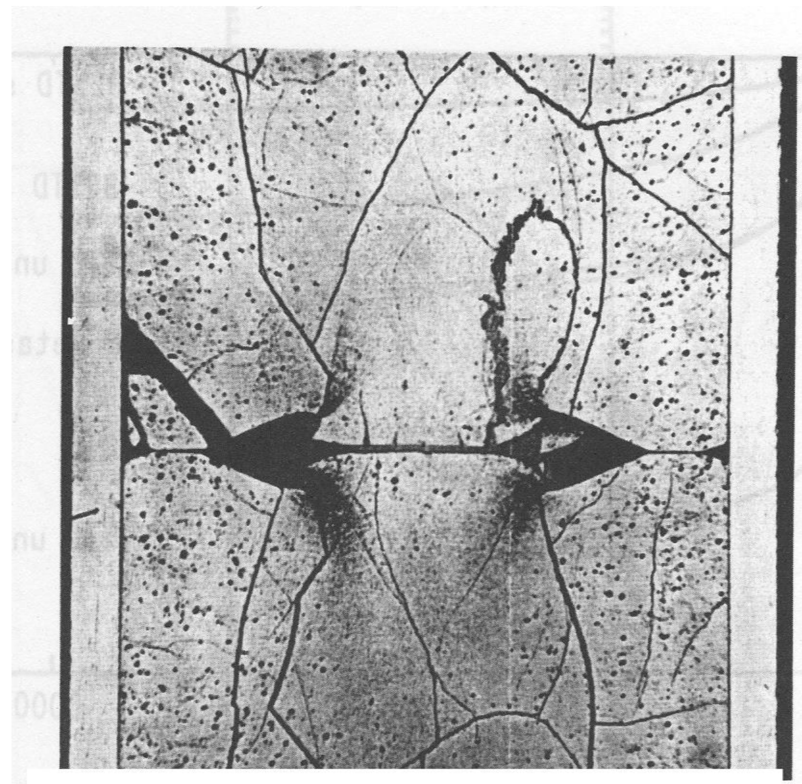
Plenum is set up at the end of fuel rod to accommodate the inner pressure increase of the fuel rod.

(2) ③Swelling

Swelling = solid（atom）swelling + gas bubble swelling



Decrease of pellet density due to swelling



Collapse of dish space due to swelling (gas bubble swelling)

(2) ④Degradation of thermal conductivity of pellet

Irradiation effects on thermal conductivity

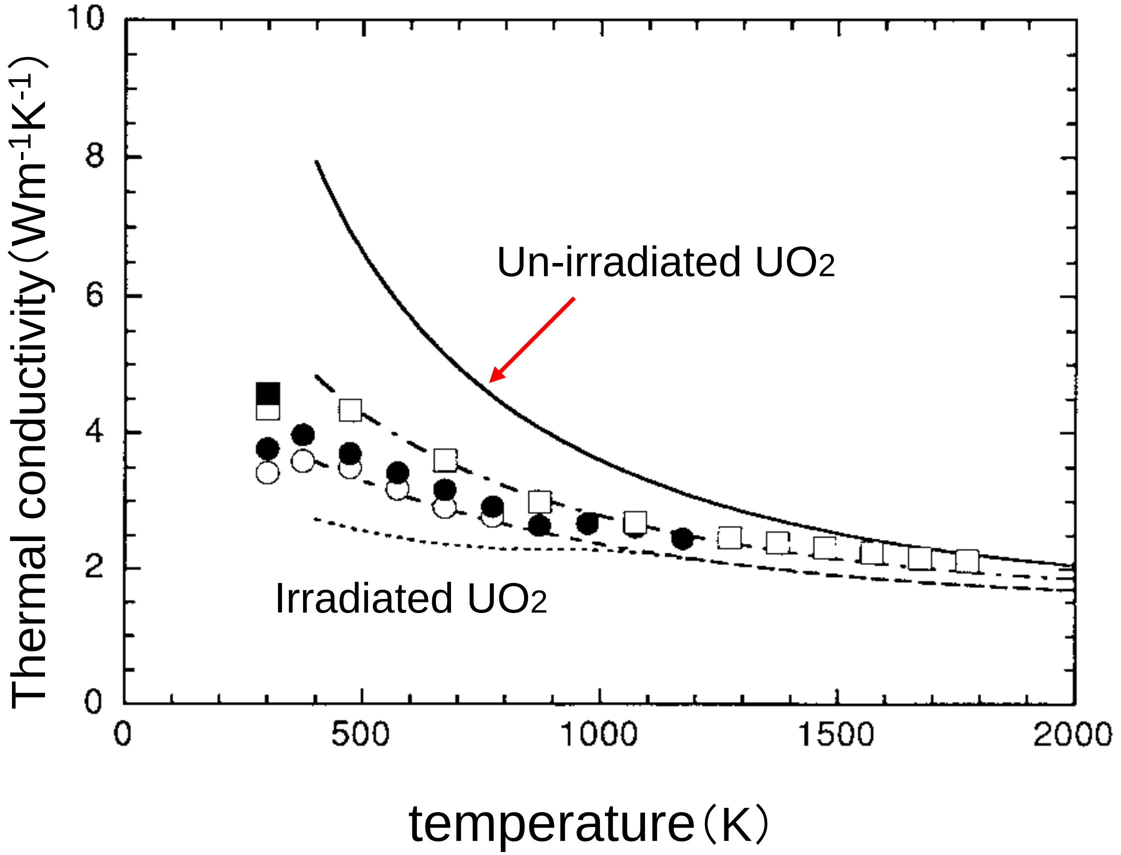
Effects of irradiation

・Irradiation defects,

(lattice defects)

・Accumulation of FPs

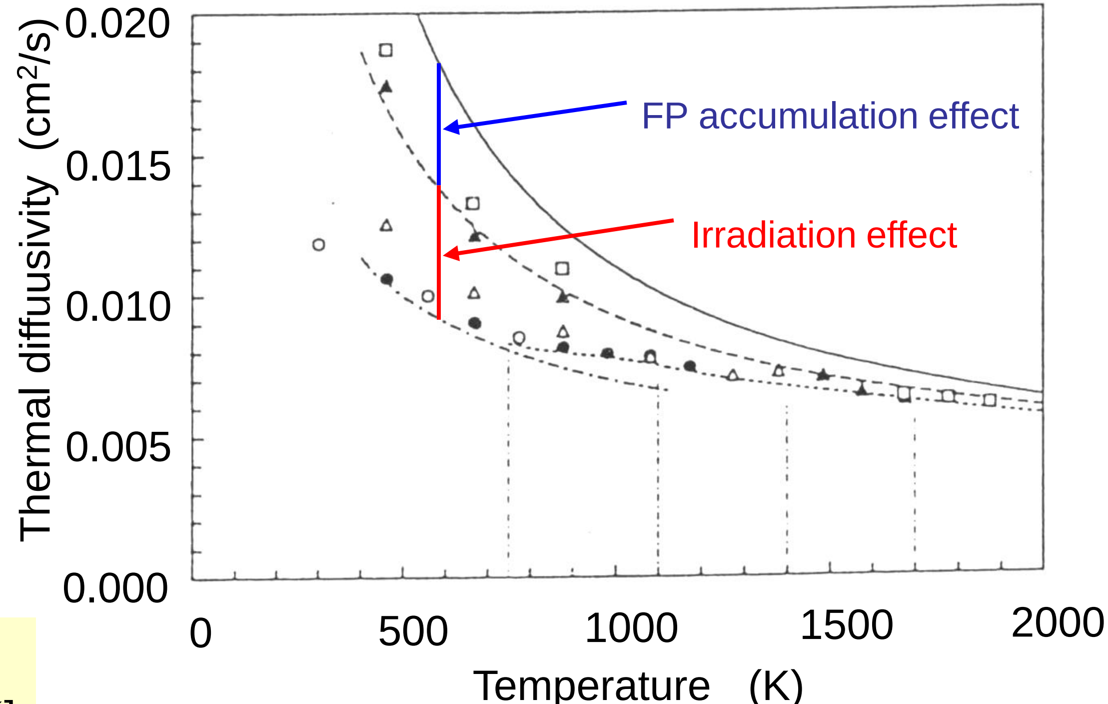
Reduce the thermal conductivity at low temperature region.



Comparison of thermal conductivity model of irradiated UO_2 and experimental data

Irradiation effects disappear after repeated measurement at high temperature

Effects of irradiation defects and accumulated PFs can be separated.



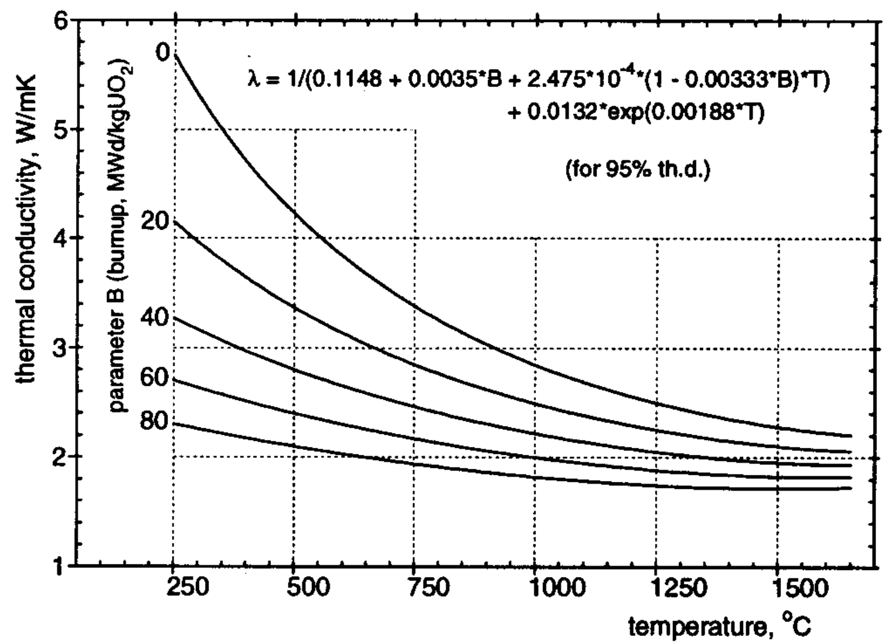
Thermal diffusivity $\kappa=\lambda/c\rho$ [m^2/s]

λ : thermal conductivity [J/s·m·K]

c : specific heat [J/kg·K]

ρ : density [kg/m^3]

Thermal Diffusivity
(Temperature conductivity)



Thermal conductivity degradation model based on the inpile data at Halden rector.

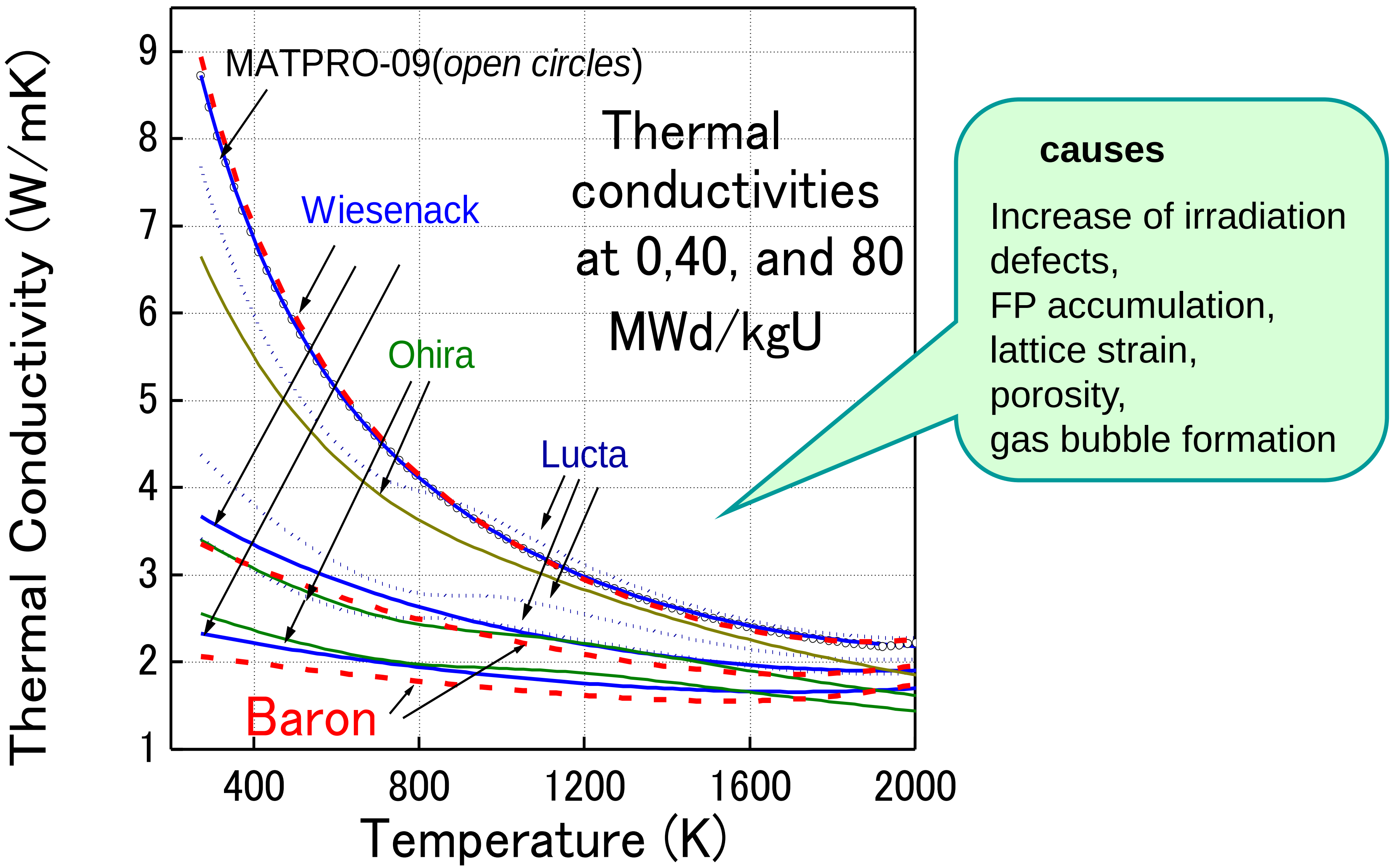
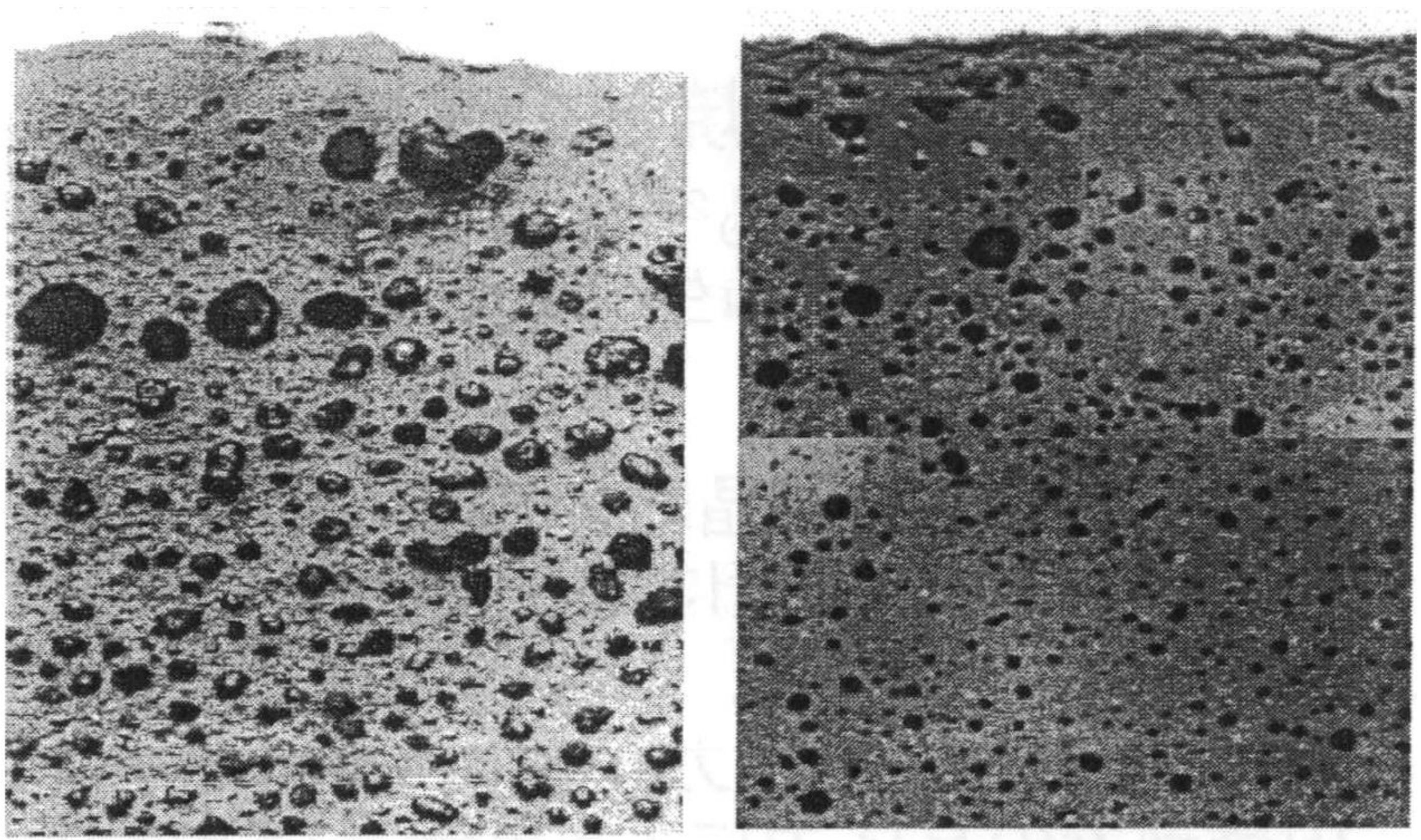


Fig.1 Comparison of empirical equations of pellet thermal conductivity degradation with burnup.

(2)⑤Rim structure

Bubbles and small grains

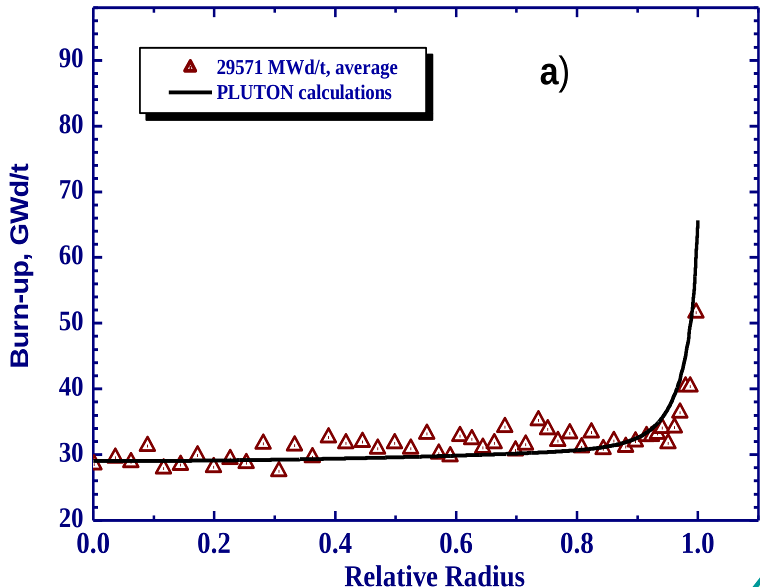
Formed at pellet rim at high burnup. Appears at local burnup above 70GWd/t.
Porous structure in which large gas bubbles of several μm size are surrounded by subdivided small grains(about 0.1μm).

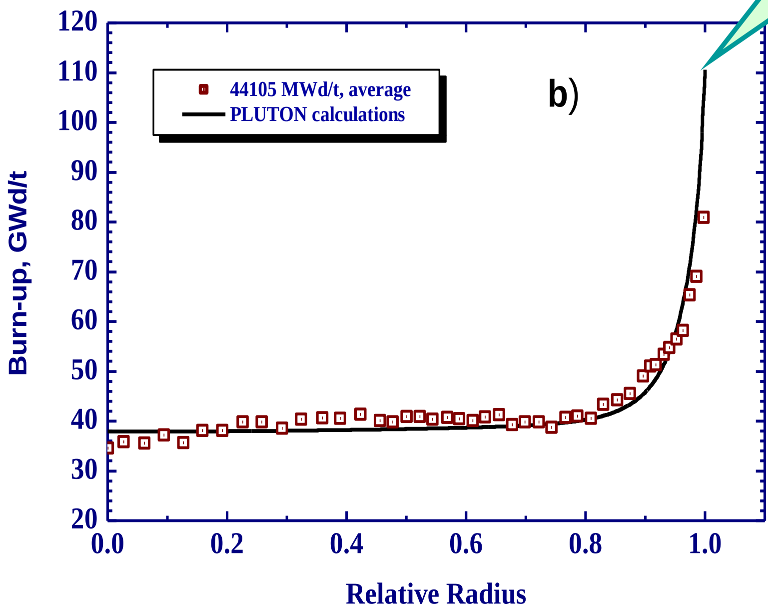


102 Mwd/kgU 20 μm 67 Mwd/kgU

R. Manzel, et al., ANS Top. Mtg. Park City (2000)

Rim structure appears at outer periphery(rim) of pellet where local burnup is very high.





FP accumulation

Accumulation of lattice strain energy

Athermal recrystalization process

Mechanism of rim structure formation

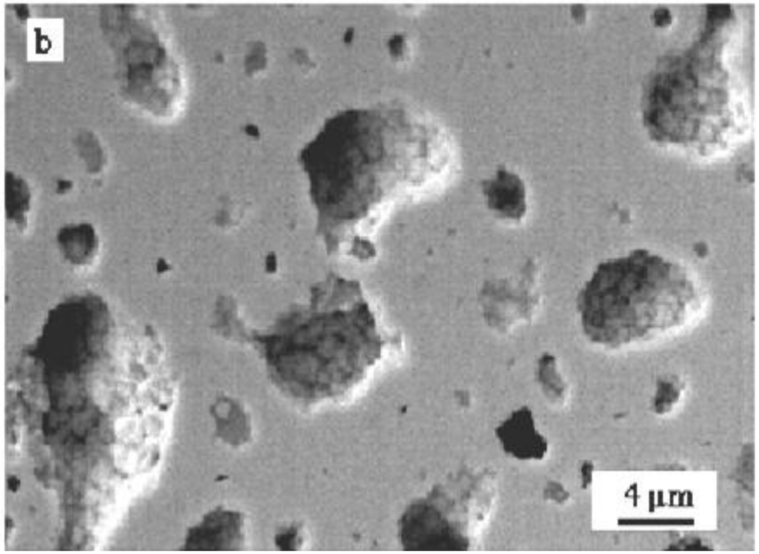
Formation of extremely highburnup region at the periphery of pellet

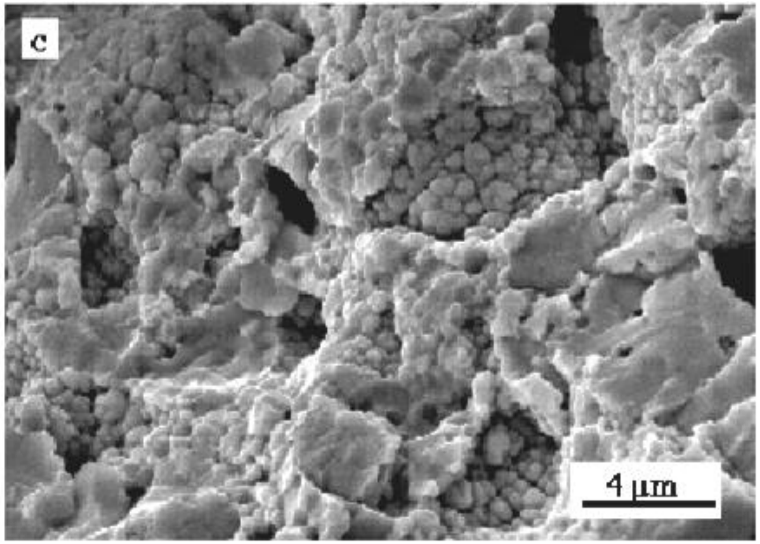
- Production of ^{239}Pu due to resonance neutron absorption of ^{238}U at rim and burning of ^{239}Pu

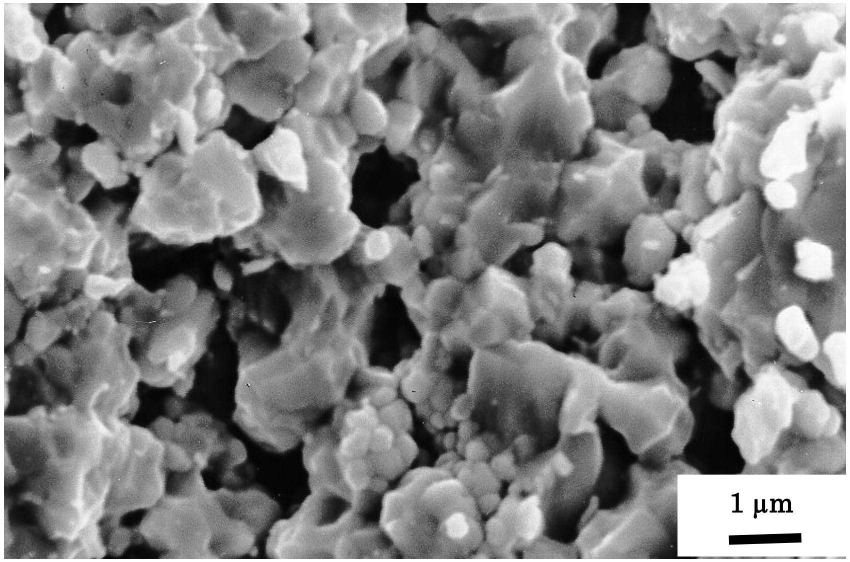
Formation of rim structure(proposed model)

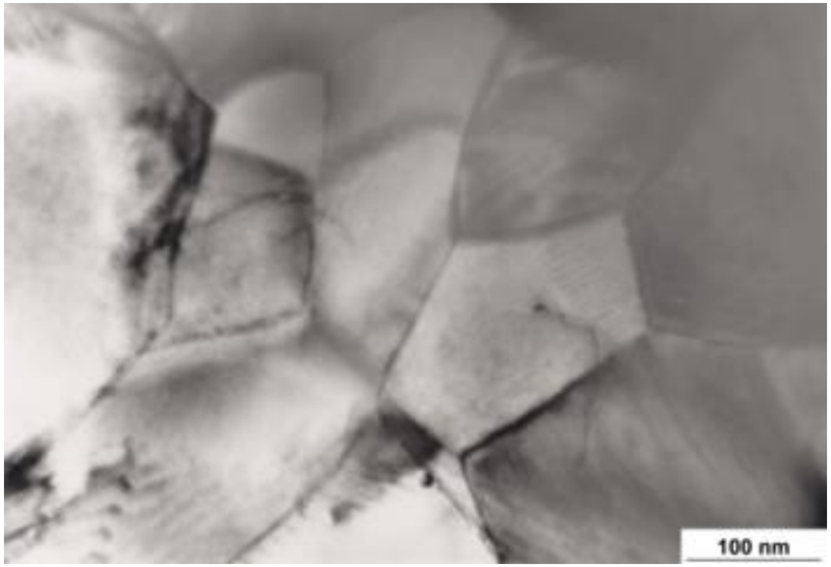
- Subdivision of grain(polygonization) by irradiation defects, and accumulation of gas bubble (Matzke)
- Subdivision of grain and recrystalization by irradiation defects, sweep of gas bubble and accumulation of gas bubble (Une et al.)
- Recrystalization due to high pressure in the gas bubbles. (Spino et al.)

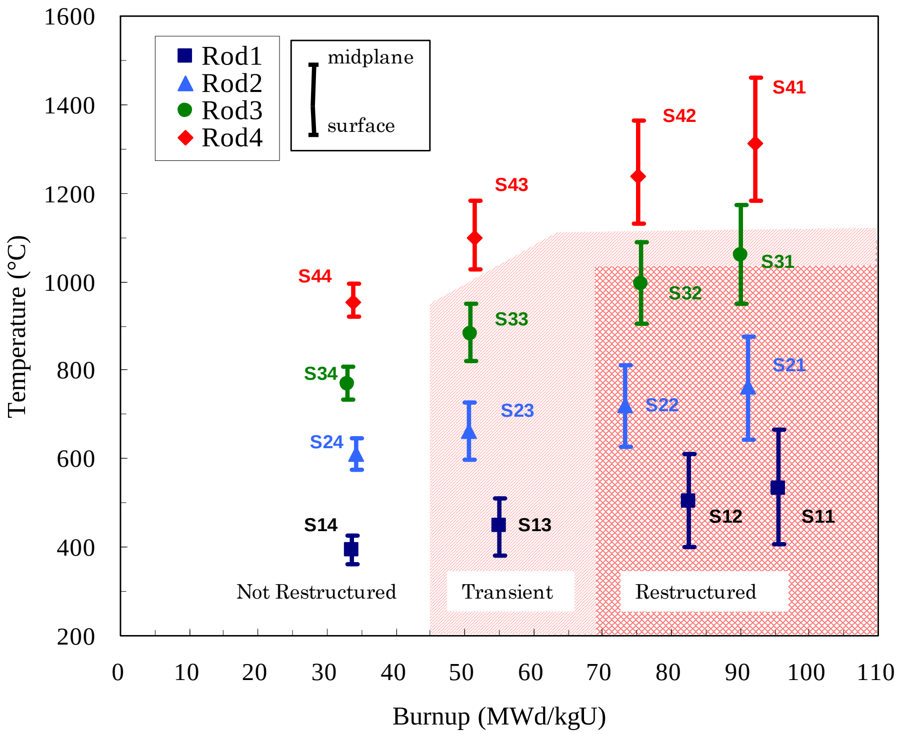
Observation of Rim structure（SEM,TEM)



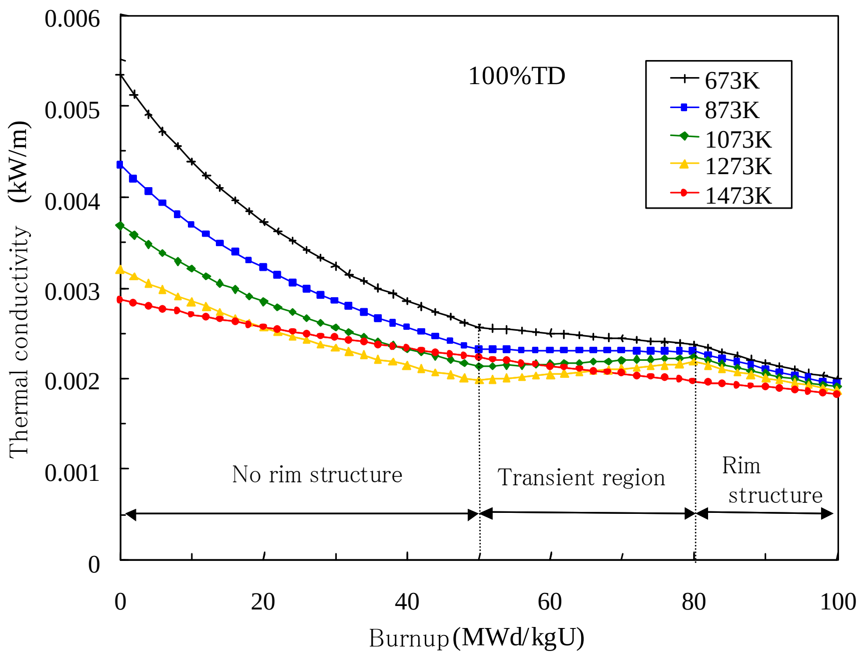








Formation conditions of rim structure (temperature and burnup)



Effects of rim structure formation on thermal conductivity

1．5 cladding

Characteristics needed for fuel cladding

- High corrosion resistance at outer surface
- Adaptability for pellet deformation(swelling, thermal expansion)
- SCC resistance of inner surface
- Small thermal neutron absorption cross section
- Mechanical strength as structural material
- High film coefficient of heat transfer from pellet to coolant
- Integrity and easy machining as parts

One of the main role of cladding is containment of fission products (FPs) with high activity. (The second wall)

Thermal neutron absorption

element	**neutron cross section(barn)**	**Melting T（℃）**
Al	0.23	660
Mg	0.06	651
Si	0.13	1414
Zr	0.18	1852
Sn	0.60	232
Zn	1.06	420
Nb	1.10	2469
Mo	2.50	2610
Fe	2.53	1535
Cr	2.90	1890
Cu	3.69	1085
Ni	4.60	1455
V	5.10	1890
Ti	5.60	1675
（zircaloy）	0.23	1855
（SUS）	～3	～1500

→ **low thermal neutron cross section and high melting point are important**

Composition of various zirconium alloy

Alloy name	Sn	Fe	Ni	Cr	Nb	usage
Zircaloy-2	1.2 ～1.7	0.07 ～0.20	0.03 ～0.08	0.05 ～0.15	—	BWR （H_20 mod.） ATR （D_2O mod）
Zircaloy-4	1.2 ～1.7	0.18 ～0.24	—	0.07 ～0.13	—	PWR （H_2O mod）
Zr-1.0Nb	—	—	—	—	1.0	VVER
Zr-2.5Nb	—	—	—	—	2.5	CANDU
Mg-Al-Be （Magnox）						Gas cool reactor （graphite moderator）
SUS316						FBR

(wt%)

In FBR, mechanical strength at high temperature is important and thermal neutron cross section is not so important.

New zirconium alloy for high burnup use

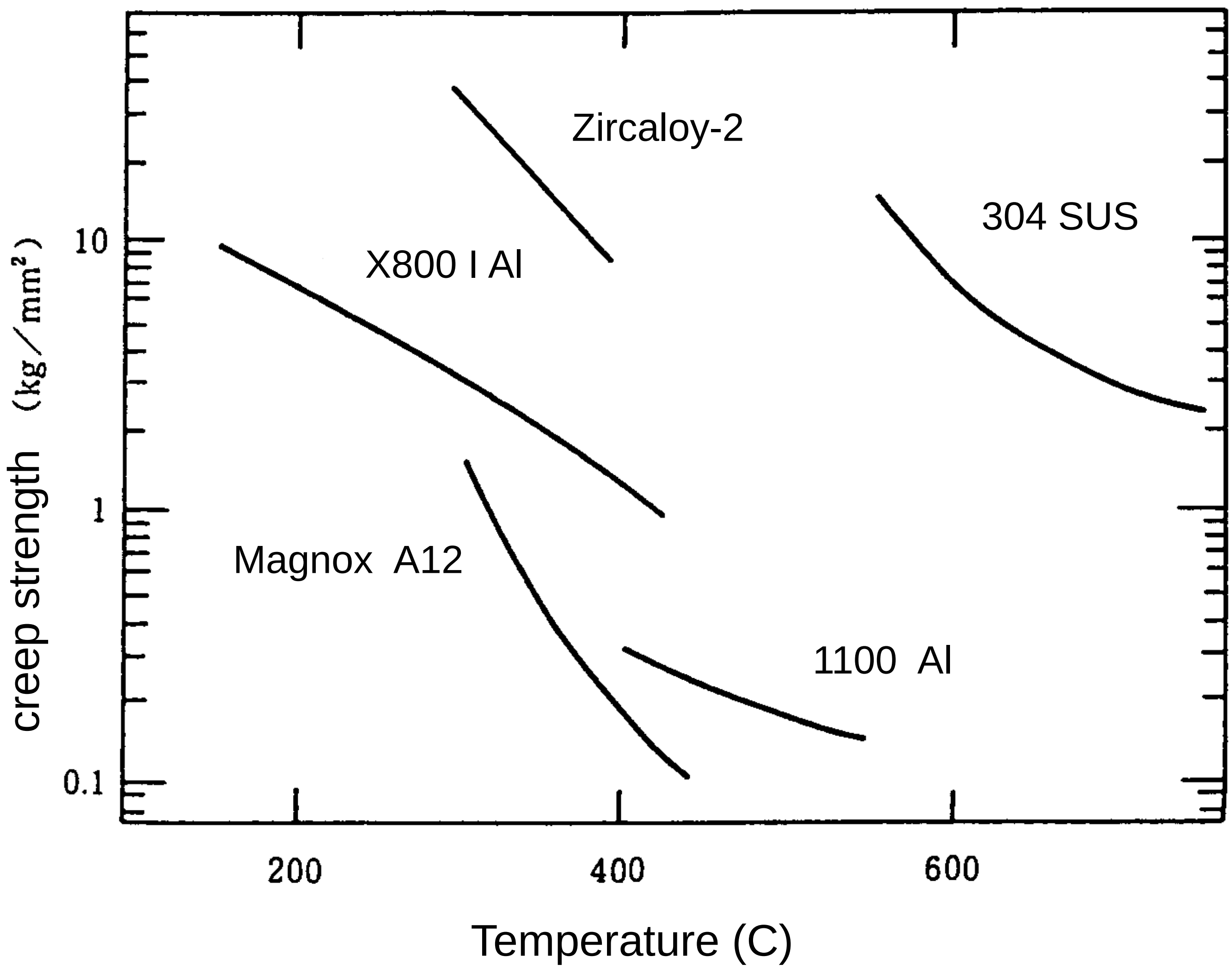
1. BWR

Alloy name	country	Sn	Fe	Ni	Cr
High-FeNi Zry-2	Japan	（low Sn, high Fe,Ni within zircaloy-2）			
Zr linar		（inner：pure Zr、outer：Zry-2）			
Current Zry-2		1.2～1.7	0.07～ 0.20	0.03～ 0.08	0.05～ 0.15

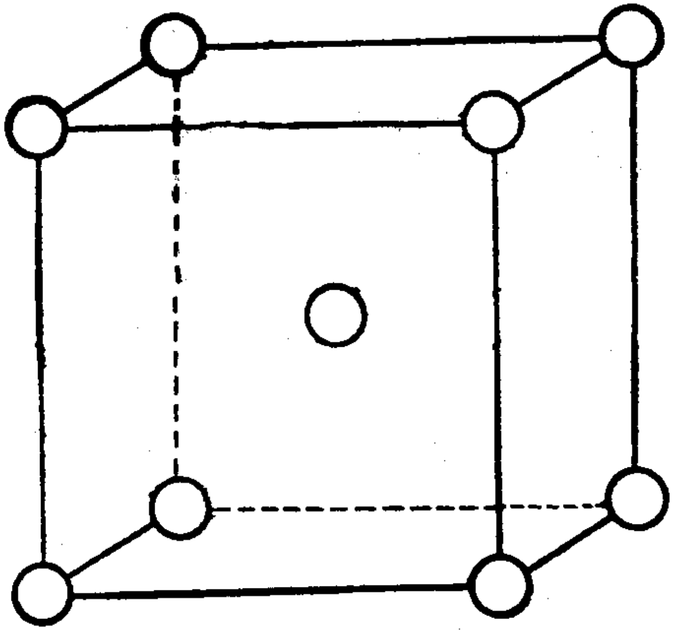
2. PWR

Alloy name	country	Sn	Fe	Ni	Cr	Nb
ZIRLO TM	USA	1.0	0.1	-	-	1.0
Duplex（outer） （inner Zry-4）	Germany	0.8	0.21	-	0.1	-
MDA	Japan	0.8	0.21	-	0.1	0.5
NDA	Japan	1.0	0.27	0.01	0.16	0.1
M5 TM	France	-	-	-	-	1.0
Current Zry-4		1.3	0.2	-	0.1	-

Creep strength of cladding materials

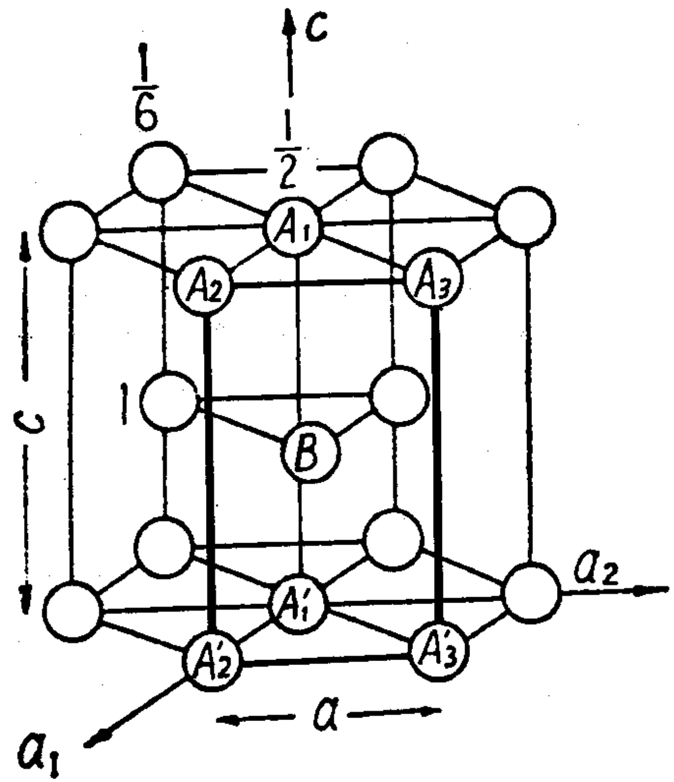


Crystal structure of Zr



Body centered cubic structure (βphase)

higher than 900C

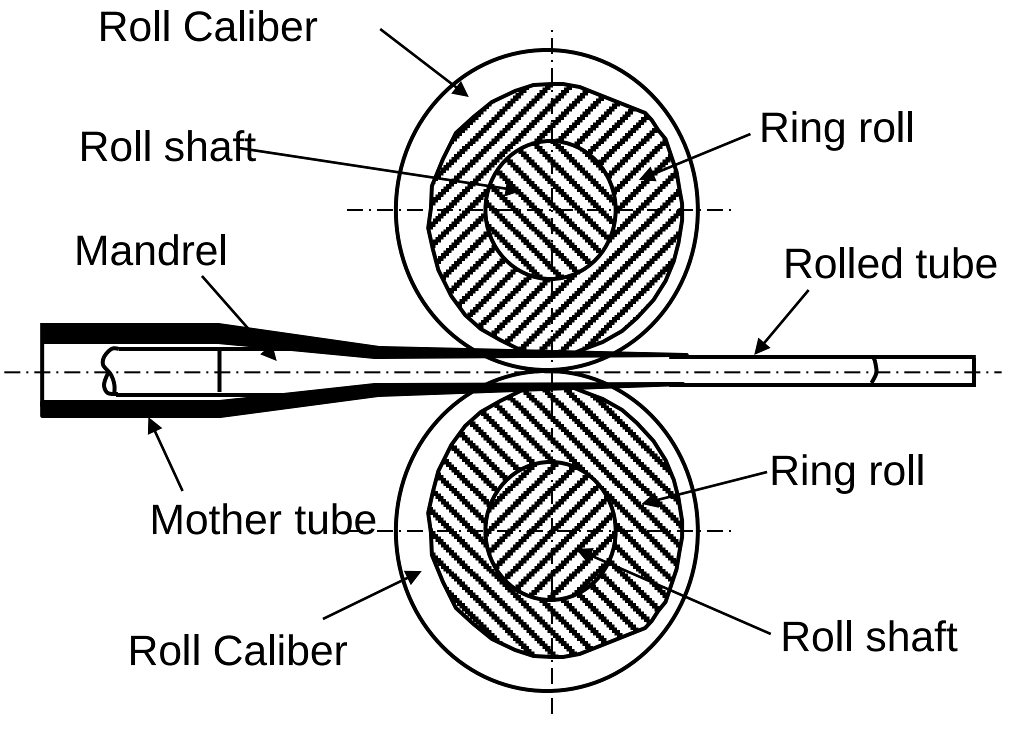


Hexagonal close-packed structure-HCP (αphase)

lower than 900C

Production of zircaloy cladding

Pilger Mill



Characteristic（compared with usual drawing process）

・Thickness of tube can be reduced with small reduction of tube diameter.
（high Q value such as c.a.2 is possible）

Q value

Q = reduction rate of thickness /reduction rate of diameter

Mainly reduction of thickness
→ higher Q

Q value

Q= reduction rate of thickness

/reduction rate of diameter

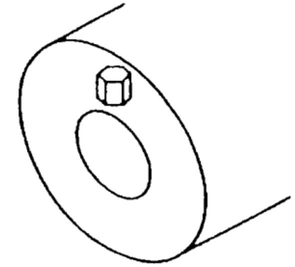
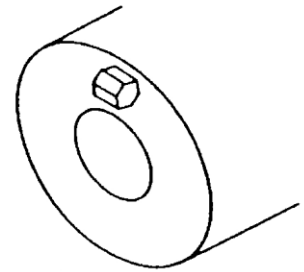
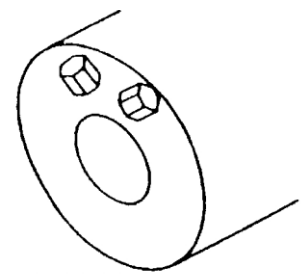
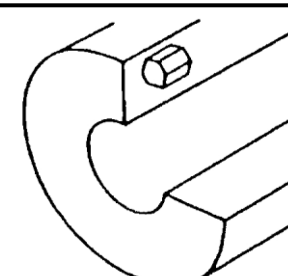
$$Q = \frac{(t_0 - t)/t_0}{(d_0 - d)/d_0}$$

t_0 : th before process t : th after process

d_0 : dia before process d :dia after process

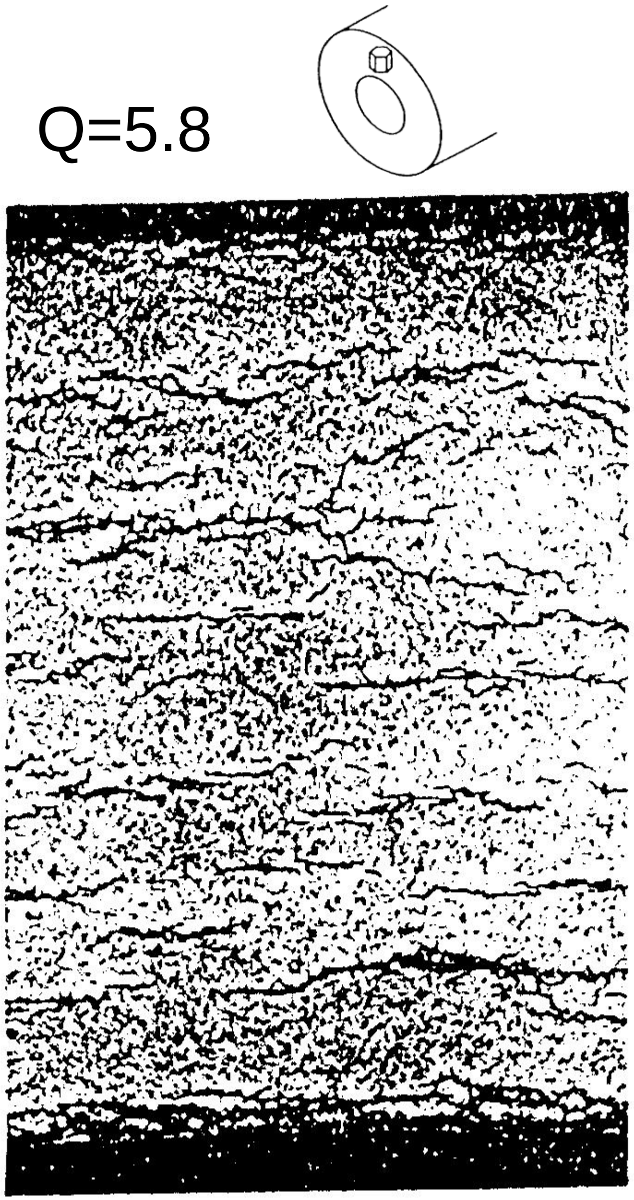
Mainly thickness reduction → high Q

Mainly diameter reduction→ low Q

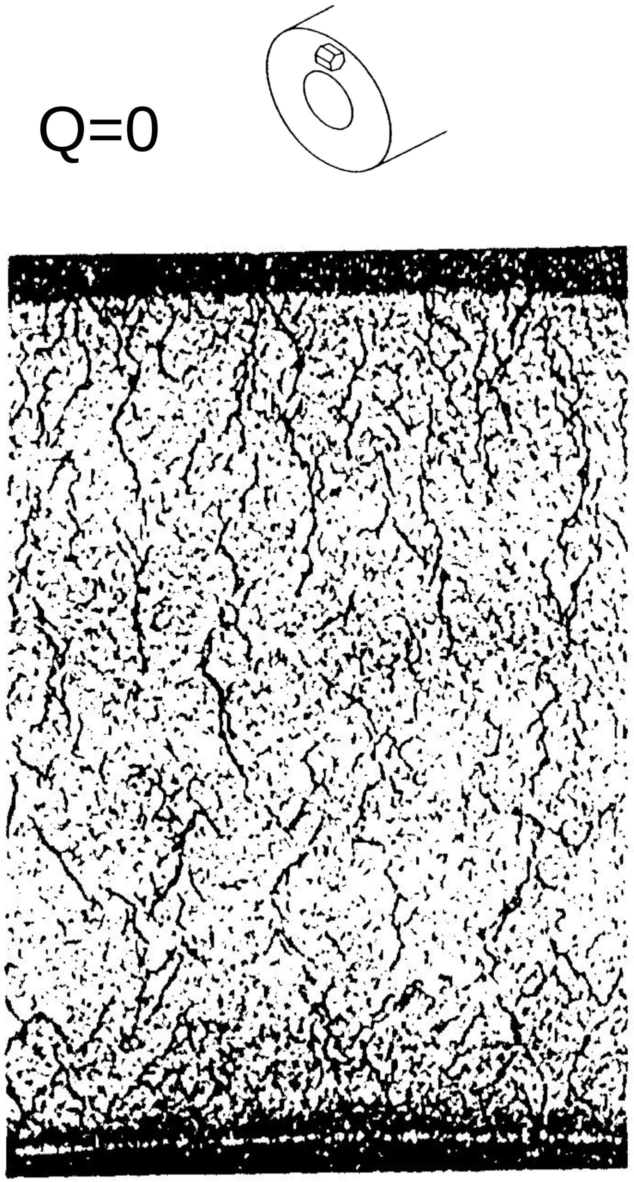
	Axis of crystal	Q valur
A		Q: very large
B		Q: very small
C		Q: adiquit （Q＝about 2）
D (imaginary)		Not occur by usual process

Q value･texture and hydride precipitation

Q=5.8



Q=0

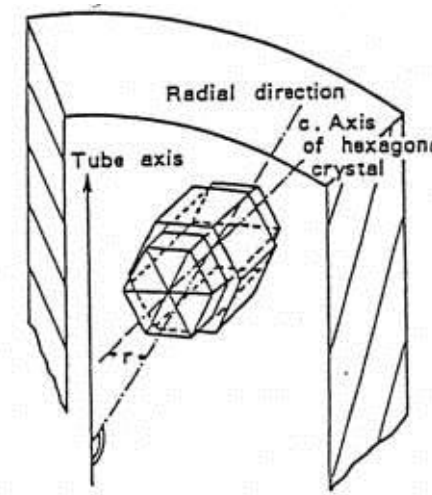


(1) influence of fast neutron irradiation on zircaloy cladding

① creep down of cladding

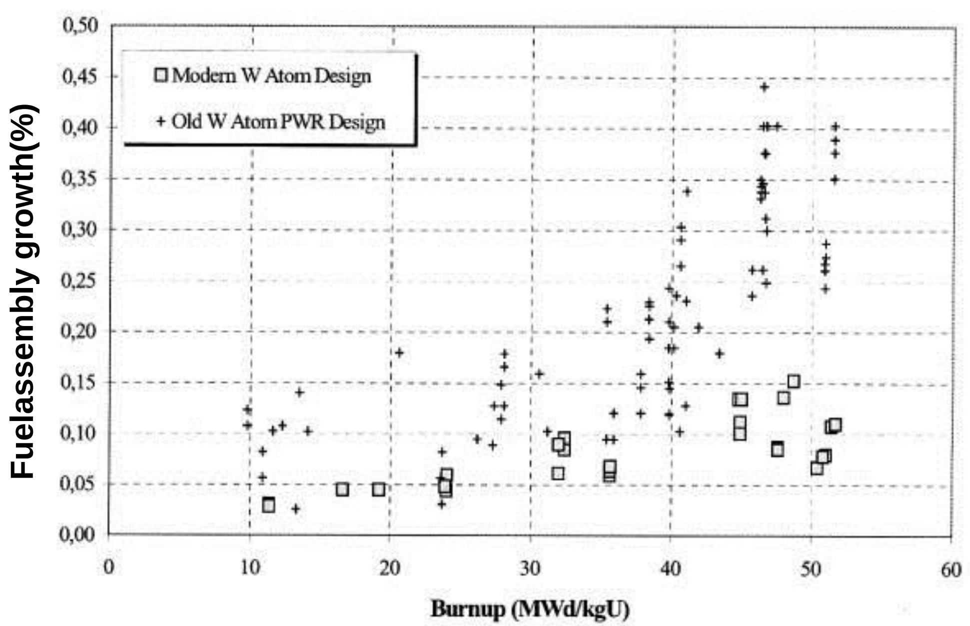
The creepdown in PWR is more remarkable than that in BWR, because of higher coolant pressure and temperature.

		PWR	BWR
Inner pressure	**at fabrication**	32 kg/cm^2G	3-5 kg/cm^2G
	In operation	90 kg/cm^2G	6-10 kg/cm^2G
Outer pressure	**In operation**	157 kg/cm^2G	73 kg/cm^2G

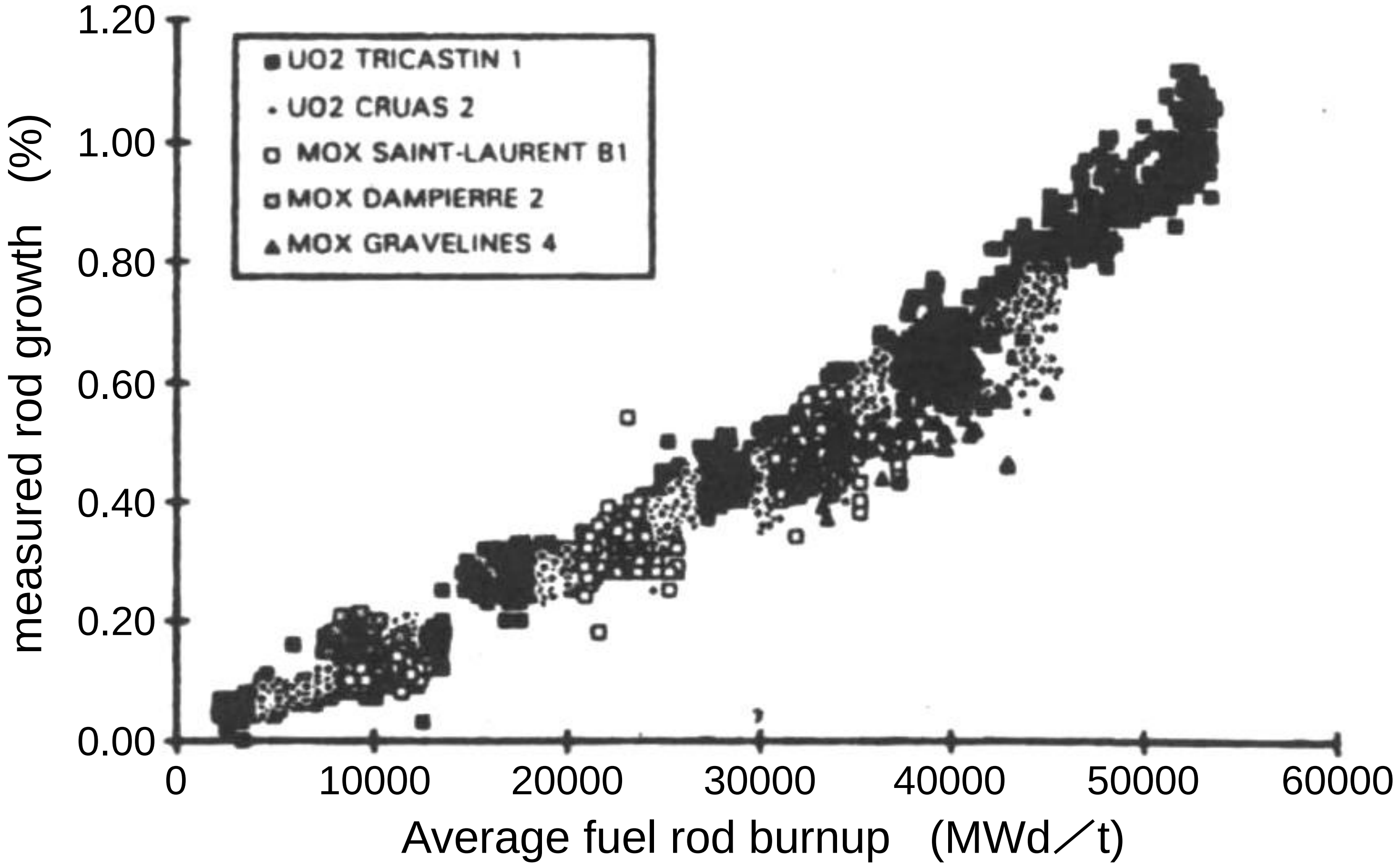


Texture of zircaloy cladding

② irradiation growth of cladding



Irradiation growth of cladding



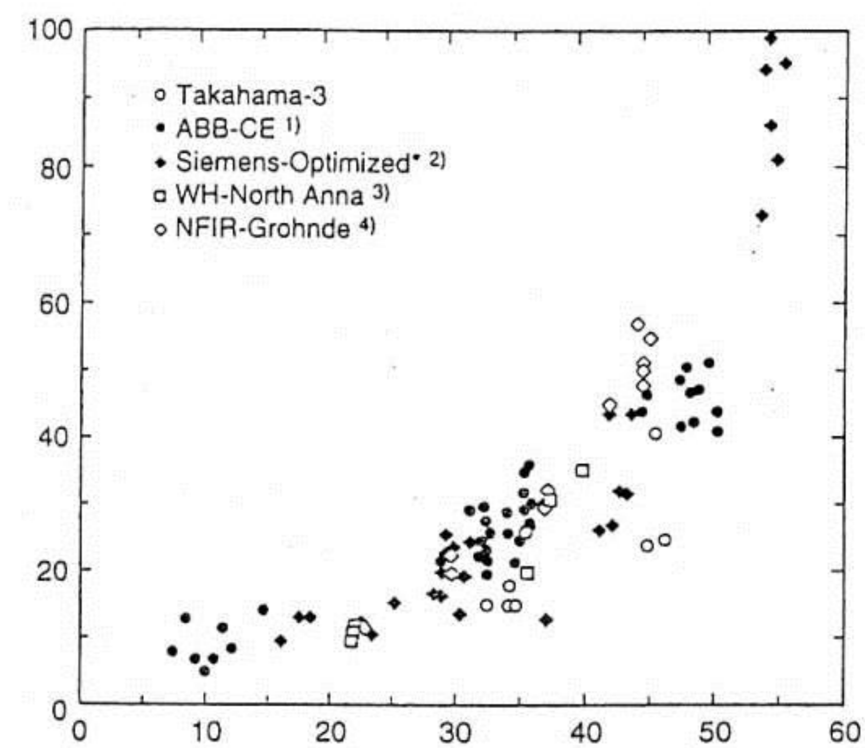
(2) Water side corrosion of cladding and accumulation of hydride

$$Zr + 2H_2O = ZrO_2 + 2H_2$$

↓ Oxide film → ↓ Decrease of metal thickness

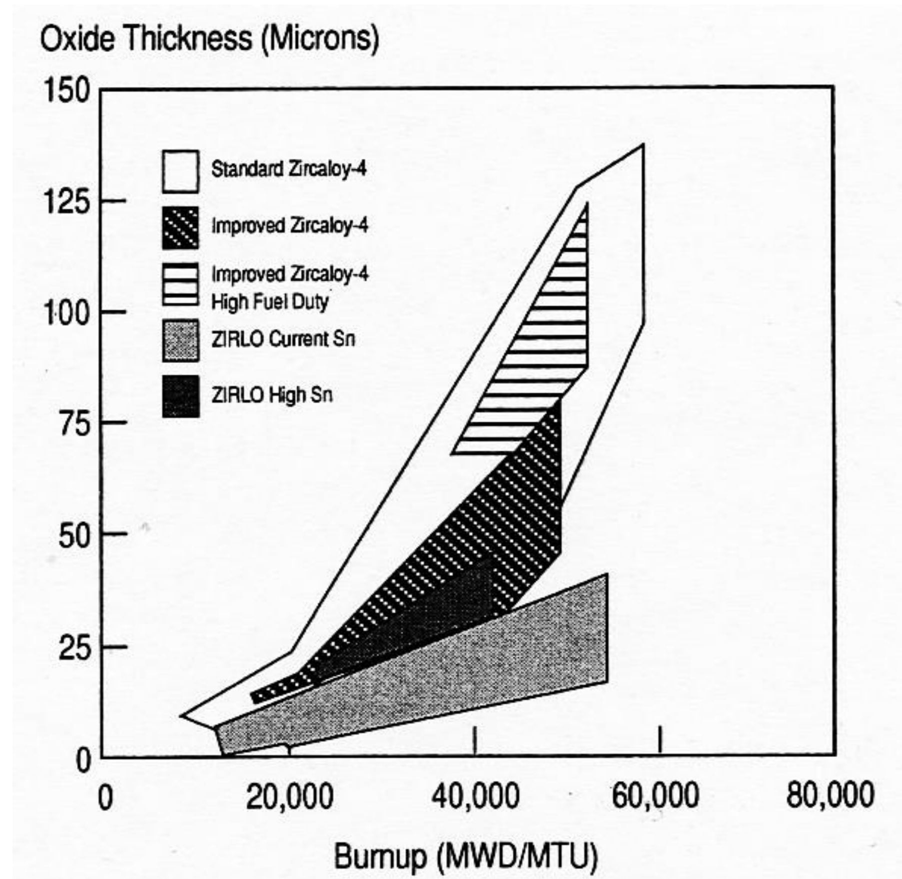
↓ Hydrogen pickup → ↓ Hydrogen embrittlement

↓ Decrease of reliability of fuel rod



Oxidethickness (μm)

Burnup(GWd/t)



Improvement of PWR cladding

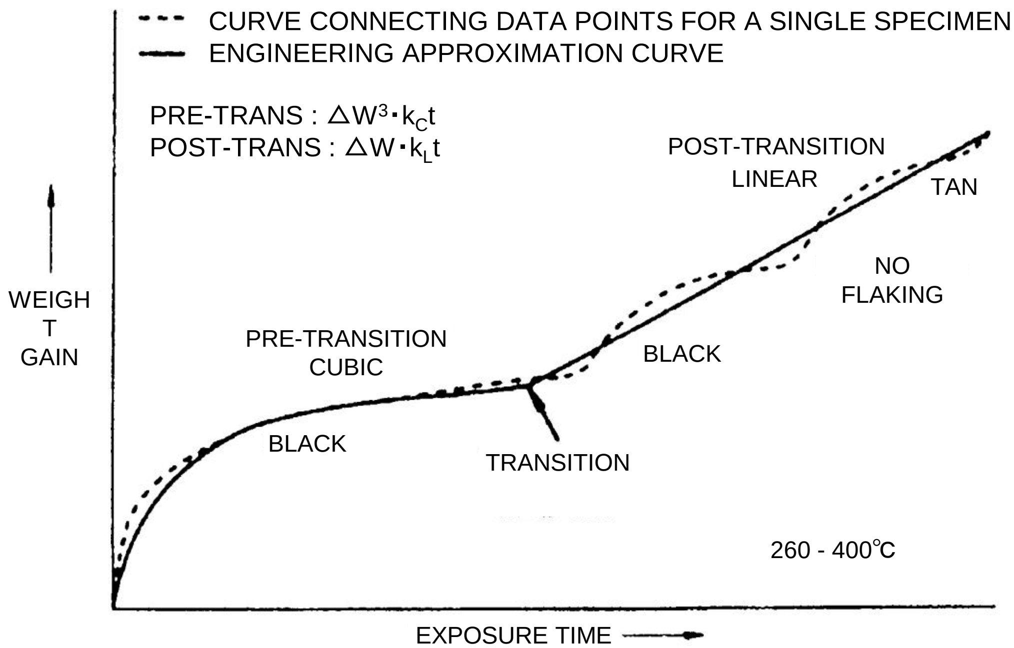
Typical weight gain curve of cladding oxidation

(estimation)

Fine oxide film formed at interface of oxide and metal, works as protective film.

Repeated formation and destruction of protective film result in linear weight gain.

The destruction of protective oxide film is caused by the compressive stress in the film due to volume increase by oxidation.



Typical weight gain curve of out pile corrosion tests of zircaloy-2 and -4

Uniform corrosion and nodular corrosion

Example of nodular corrosion

Uniform corrosion；

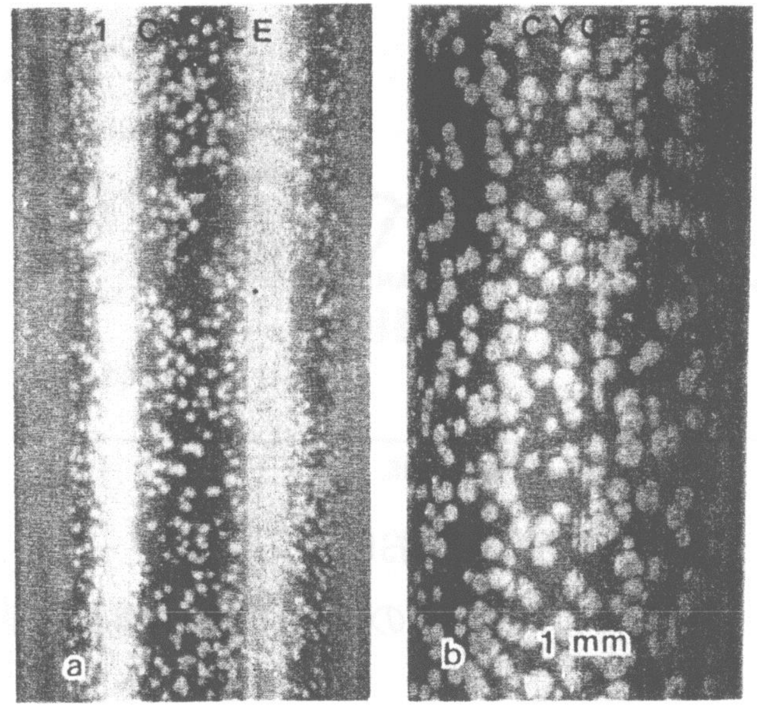
uniform oxide film over the whole tube

Nodular corrosion；（BWR condition）

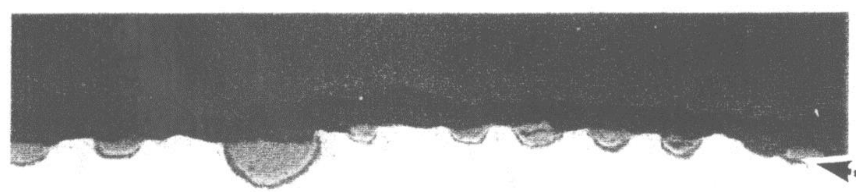
spotted corrosion

nodular ;

nodule ;



ZIRCALOY-2, DIFFERENT RODS



nodule

claddin

100μm

100μm

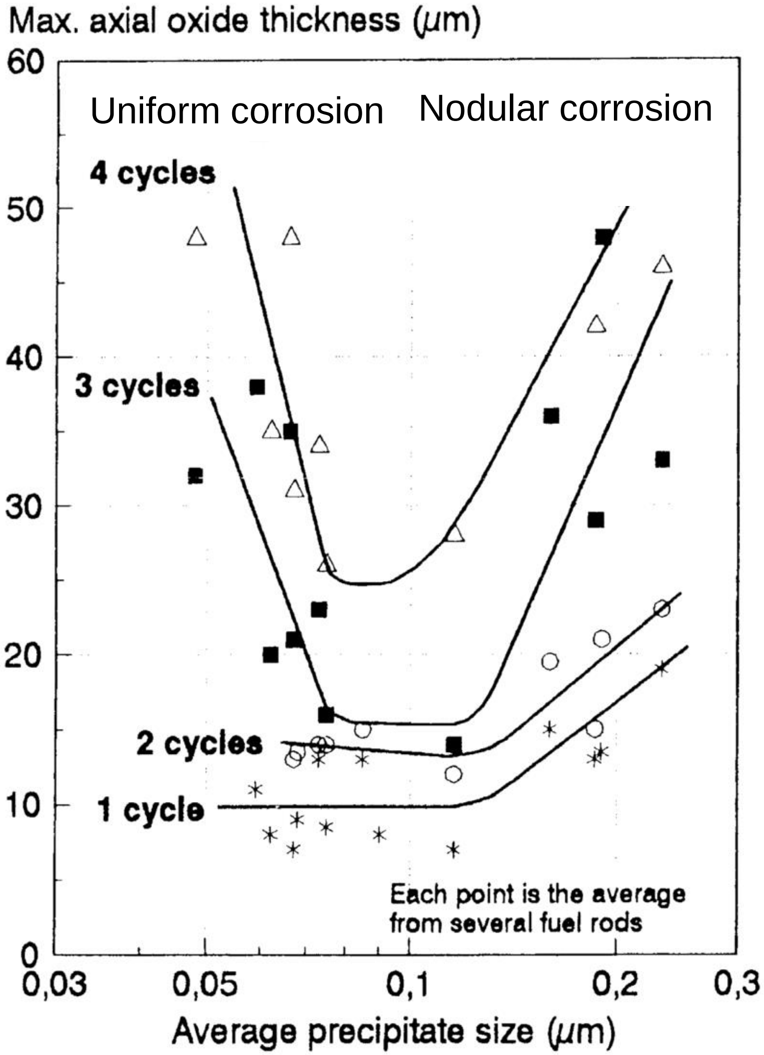
Improvement of water corrosion resistance of zirconium alloy

Method		BWR		PWR
		nodular	uniform	uniform
design	Current composition	Zircaloy-2		Zircaloy-4
		Low Sn High Fe･Ni	Low Sn	Low Sn･low C High Si
	New alloy	Same as above （+Mo，Nb addition）		ZIRLO, MDA, NDA (after 2004) M5
process	Heat treatment	Heat treatment ΣAi optimum	ΣAi optimum	ΣAi optimization

ZIRLO : Zr-1.0Sn-1.0Nb0.1Fe, MDA : Zr-0.8Sn-0.2Fe-0.1Cr-0.5Nb

NDA : Zr-1.0Sn-0.27Fe-0.16Cr-0.10Nb-0.001Ni

ΣAi and corrosion (3) BWR in reactor



・Management of accumulated heat treatment history

→（**Accumulated annealing parameters**, ie, Σ**Ai** or **A-time**）

$$\Sigma Ai = \Sigma\, t_i \cdot \exp(-40000 / T_i)$$

ti : i th heat treatment time

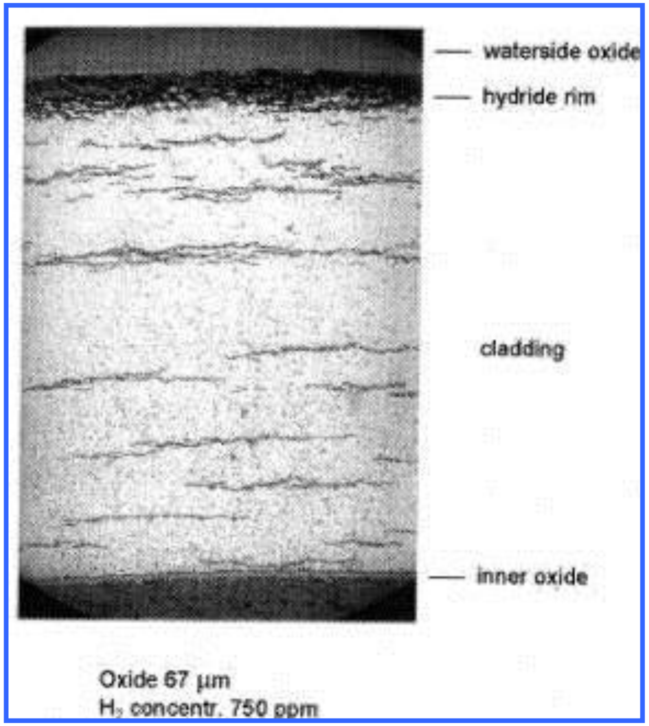
Ti : i th heat treatment temperature

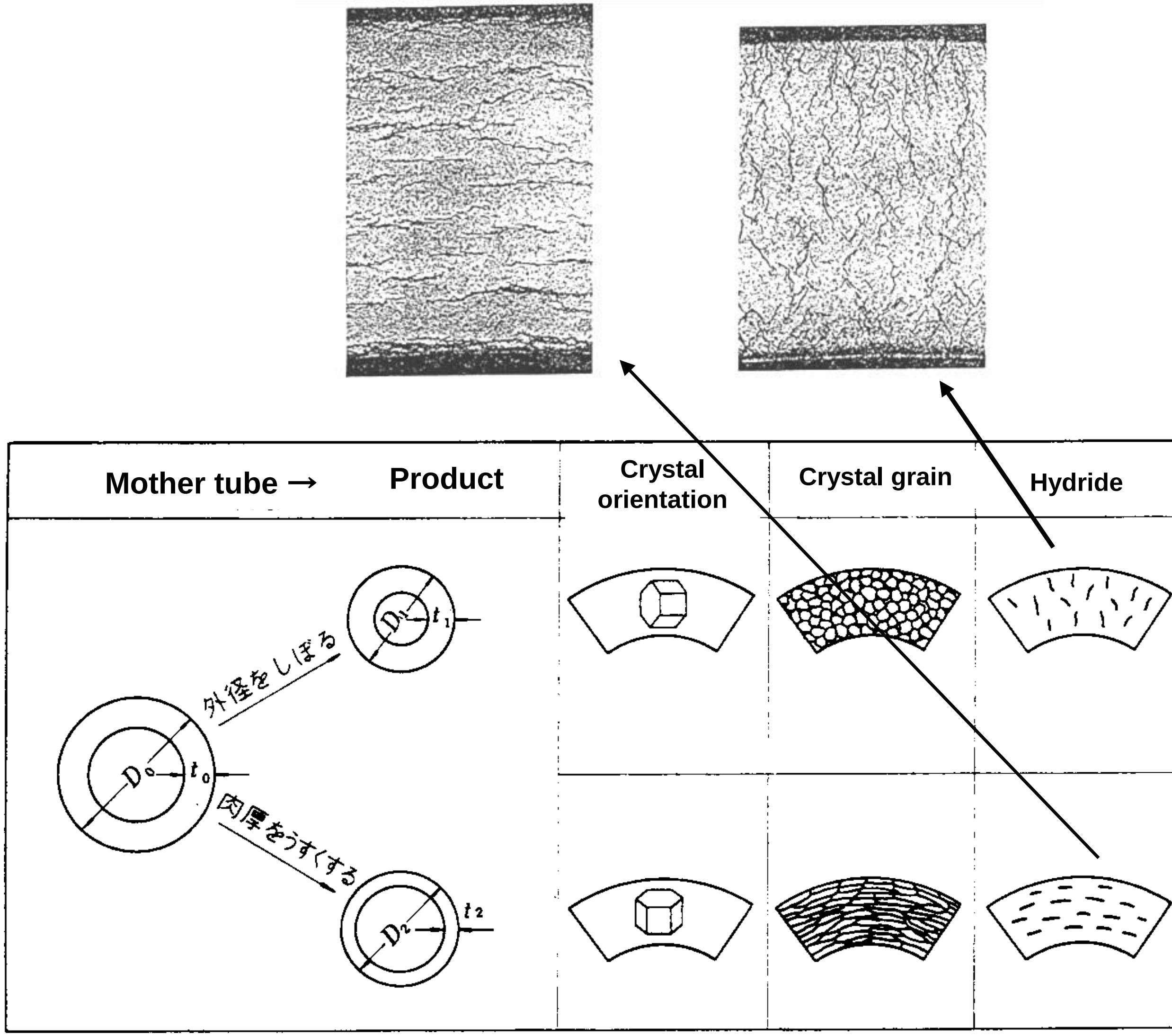
Average precipitate size corresponds to ΣAi

Fig. Effect of the precipitate size on corrosion of experimental fuel rods with Zircaloy-2 cladding in one test assembly in a BWR

※F. Garzarolli et al., ASTN STP 1245, 1994, p.709

Accumulation of hydride due to water corrosion





Zirconium hydride is very brittle, and it degrades the integrity of cladding.

Zirconium hydride precipitates on the special plane of crystal.

1.6 Fuel behavior of MOX (use for LWRs)

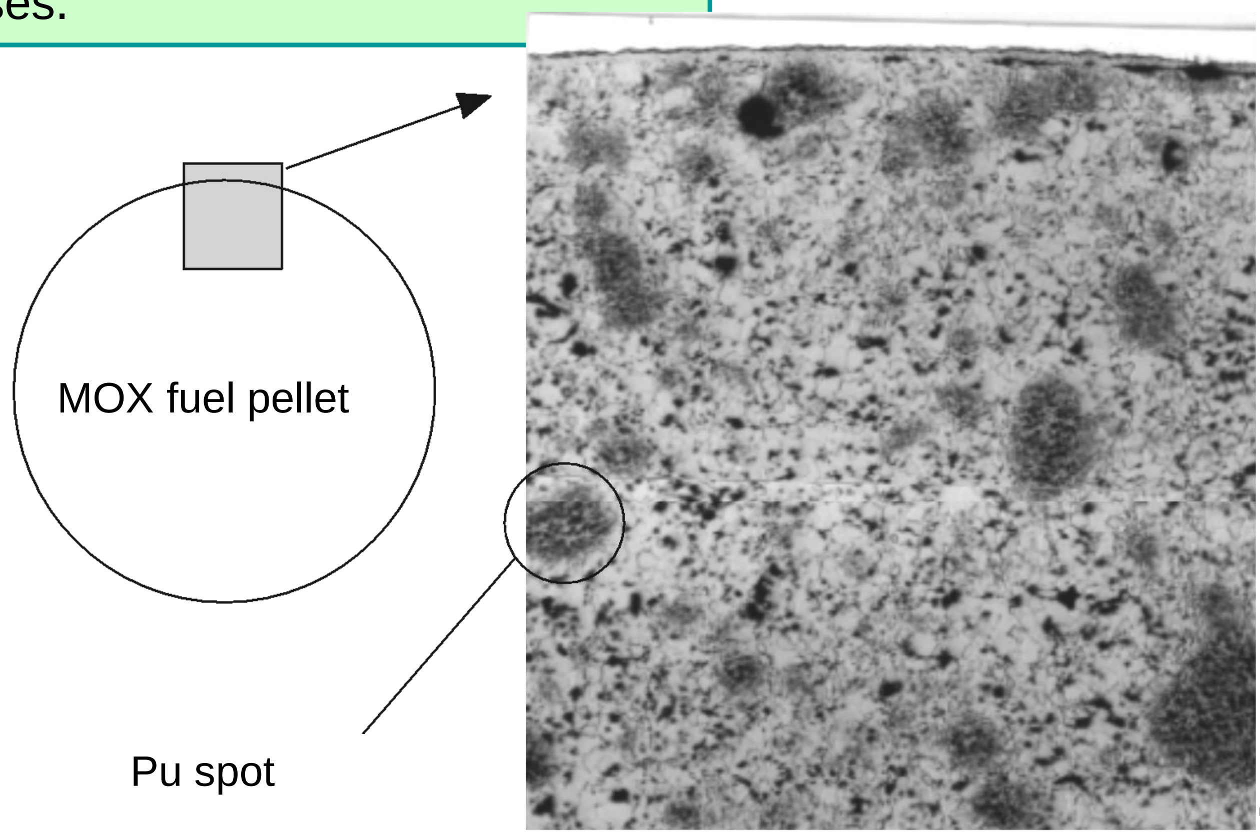
MOX for LWRs contains several % fissile Pu；

①Thermal conductivity is a little lower than that of UO_2.

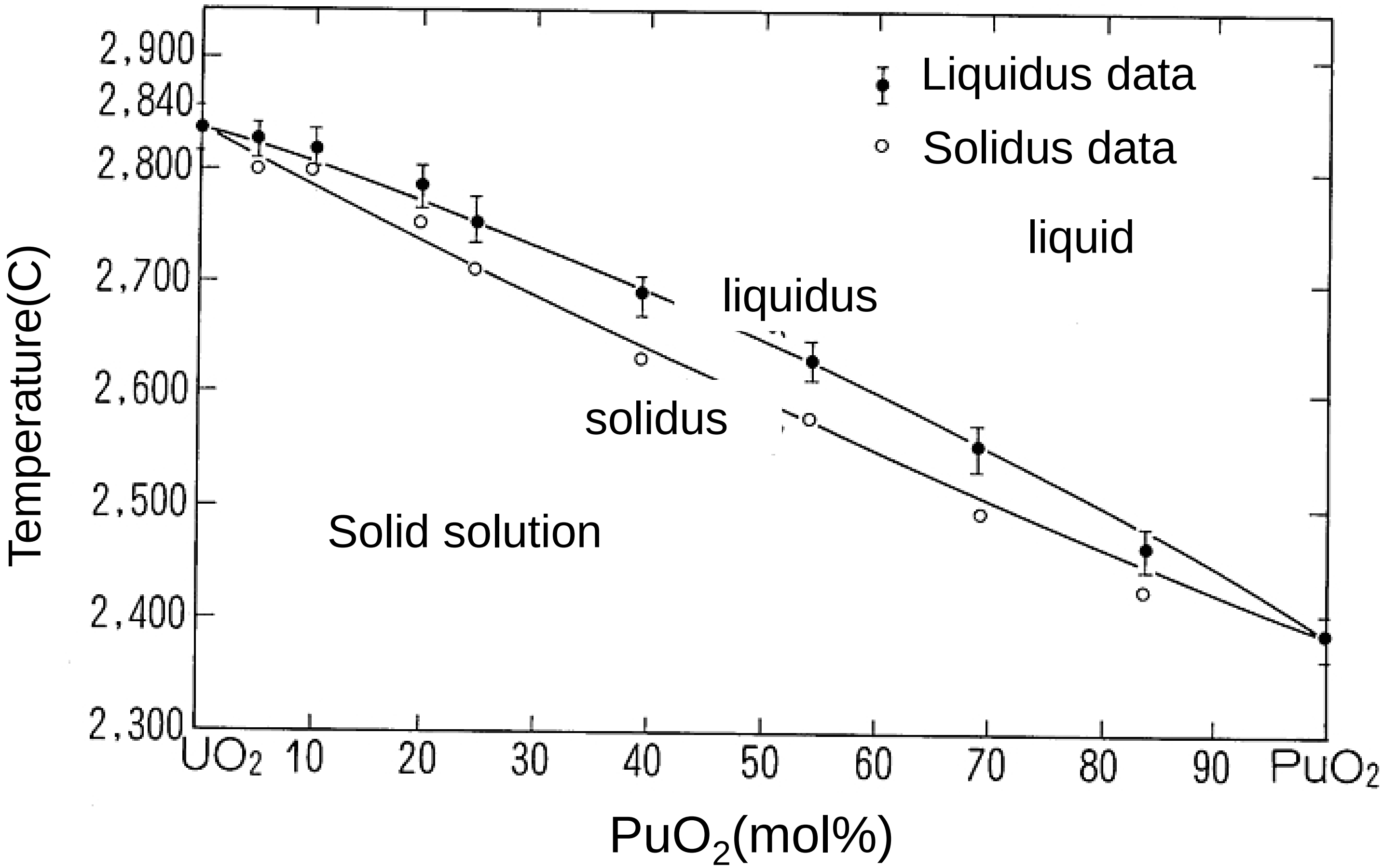
②Fission gas release is a little larger than that of UO_2

③Creep rate is larger than that of UO_2.

④Melting point decreases.



Phase diagram of UO_2-PuO_2



Methods of manufacture

(1) Necessary attention for MOX pellet manufacturing

●Pu is more poisonous than U

- ･high radioactivity
- ･pickup in bone

→ Internal exposure due to α-ray is sever problem

●critical mass for Pu is small

･sphere metal	^{235}U：22.8 kg	↔	^{239}Pu：5.6 kg
･solution	^{235}U：0.82 kg	↔	^{239}Pu：0.51 kg

●heat generation by Pu

●safegards、（physical protection）special nuclear fuel

- ･strict confinement
- ･shielding of neutron, γ-ray
- ･criticality control
- ･accountancy and control

→ More strict conditions than UO_2

Process of MOX fuel production

Special for MOX manufacturing

Preparation of powder for pellet manufacturing

- Mixing of UO_2 and PuO_2 powder
- control of Pu fissile concentration

Manufacture of pellet

- press of green pellet
- sintering under deoxidizing atmosphere
- check of pellet size, inspection

Processing of fuel rod

- Loading of pellet into cladding
- end plug welding
- inspection

Fabrication of fuel assembly

- assembling of fuel rods and components
- inspection

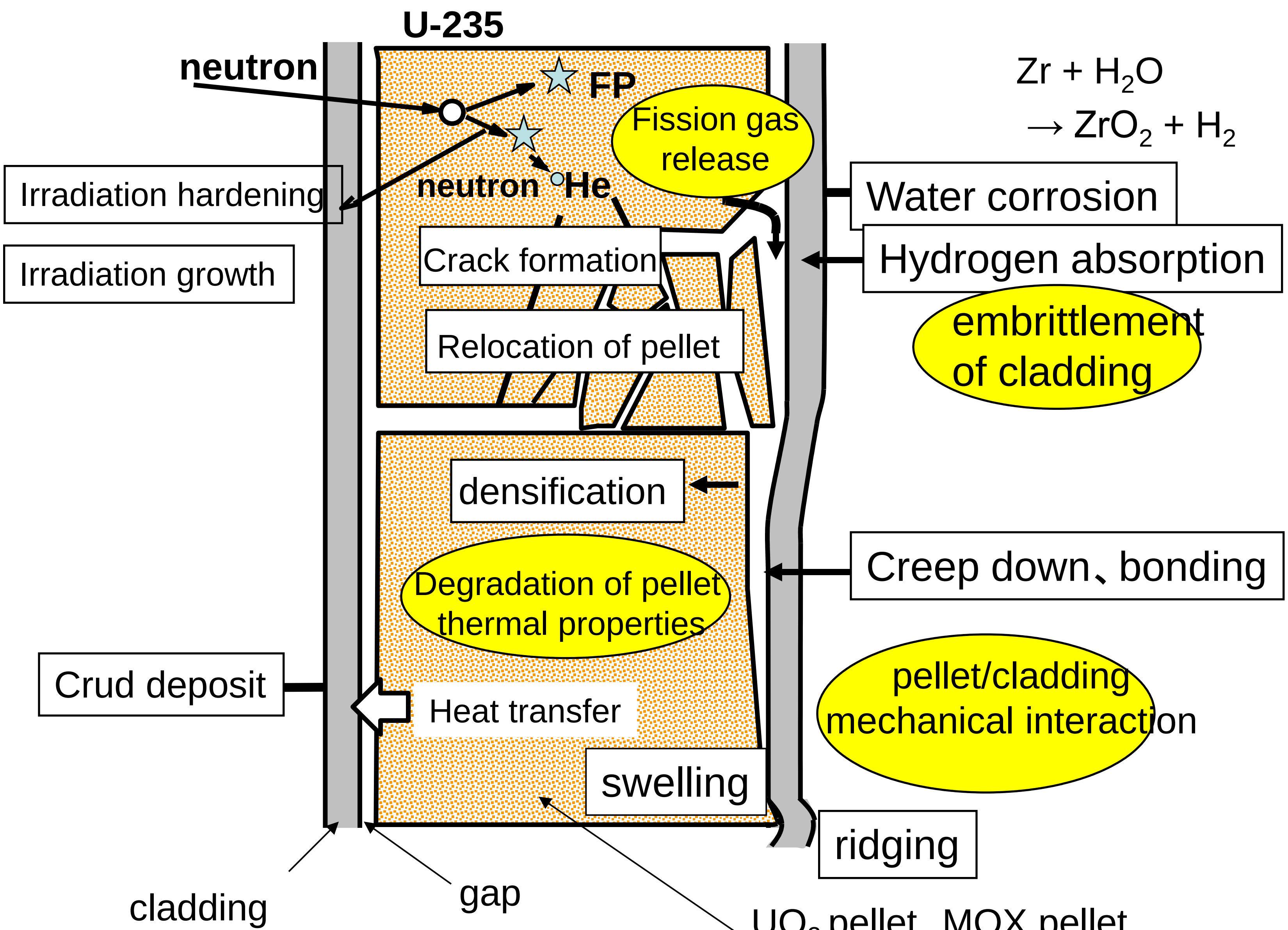
1.7 Fuel failure

Cause	PWR	BWR
Manufacturing Defect	○	○
Local Hydriding	○	○
Dry-out due to channel box deformation		○
CILC*		○
Extraordinary Corrosion	○	○
P C I**	○	○
Debris Fretting	○	○
Baffle Jet Fretting	○	
Grid Fretting	○	

CILC: Crud Induced Localized Corrosion*

*PCI**: Pellet Clad Interaction*

Phenomena occur during rector operation



Causes of fuel failure of PWR and BWR in USA
（number of table corresponds to number of failed assembly）

Causes of Fuel Failure in US PWRs
Over the 1989-1999 Time Period

Number of failed Assemblies

Number of Failed PWR Assemblies											
Failure Cause	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999*
Handling Damage		6	2			1	1		2		
Debris	146	11	67	20	13	6	10	1	10	3	
Grid Fretting	14	18	9	33	36	9	33	52	21	57	5
Primary Hydriding		1		4							
Crudding/Corrosion							4		4		
Cladding Creep Collapse							1				
Other Fabrication	1	15	1	5	3	1	15	5			1
Other Hydraulic					1					2	
Inspected/Unknown					36	36	13	9	10	2	1
Uninspected	43	58	35	61	14	3	12	3	8		3
Yearly Totals	204	109	114	123	103	56	89	70	55	64	10

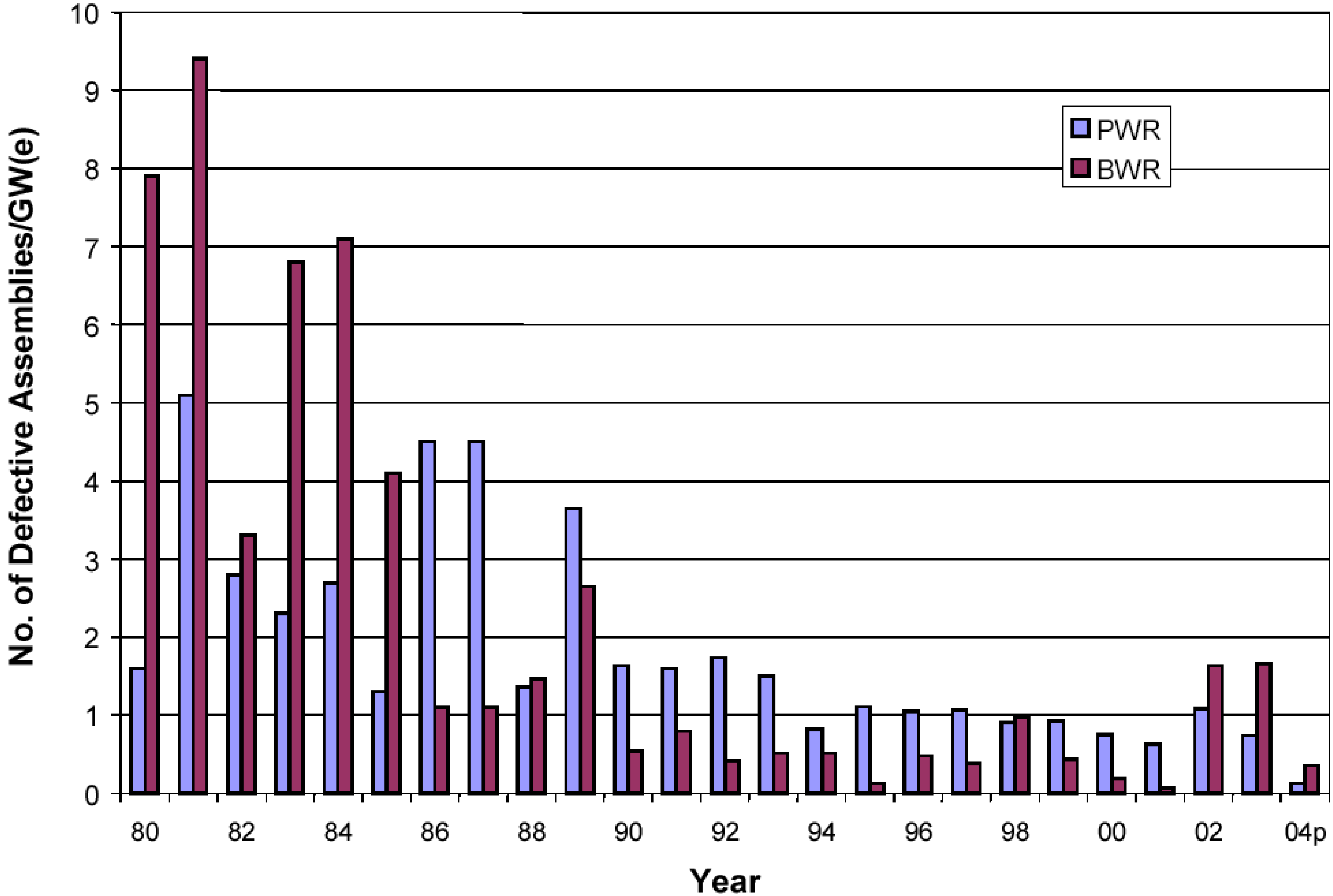
* 1999 data is preliminary and incomplete

Causes of Fuel Failure in US BWRs
Over the 1989-1999 Time Period

Number of failed assemblies

Failure cause	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999*
CILC	52	5	3						3	46	
Crudding/Corrosion											7
Fabrication	3	3	1	1	1	2					
PCI		1			2		2	2	1	1	
Debris	2	2	17	2	6	4		2	3	5	3
Uninspected or Inspected/Unknown		4	3	9	7	9	2	10	1	1	1

* 1999 data is preliminary and incomplete

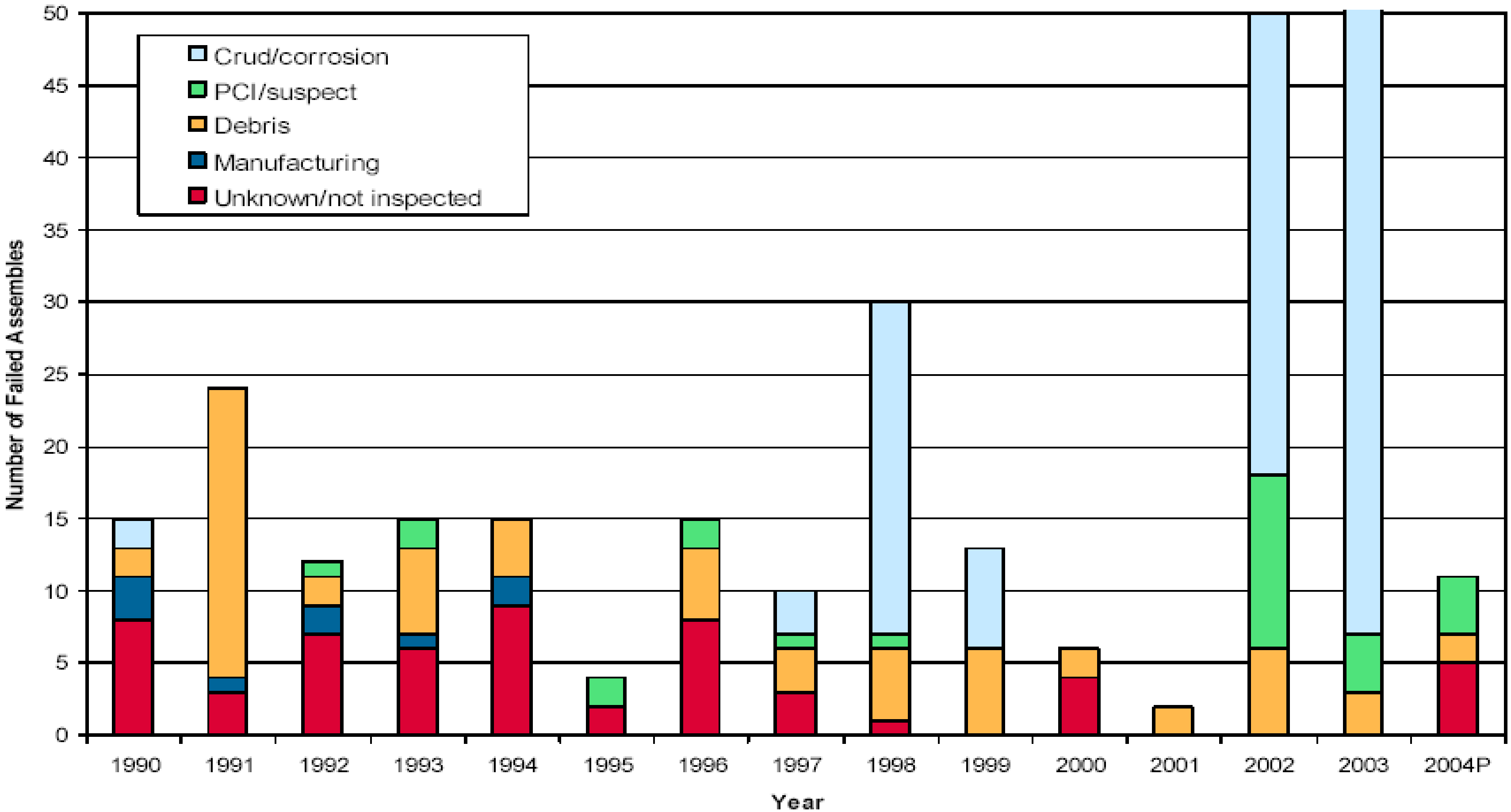


Causes of fuel failure of PWR and BWR in USA

(number in table means number of failed assembly per GW) ANS Meeting 2004 R.Yang

Figure 1. Trend in US fuel failure rates

(2004 results are incomplete)

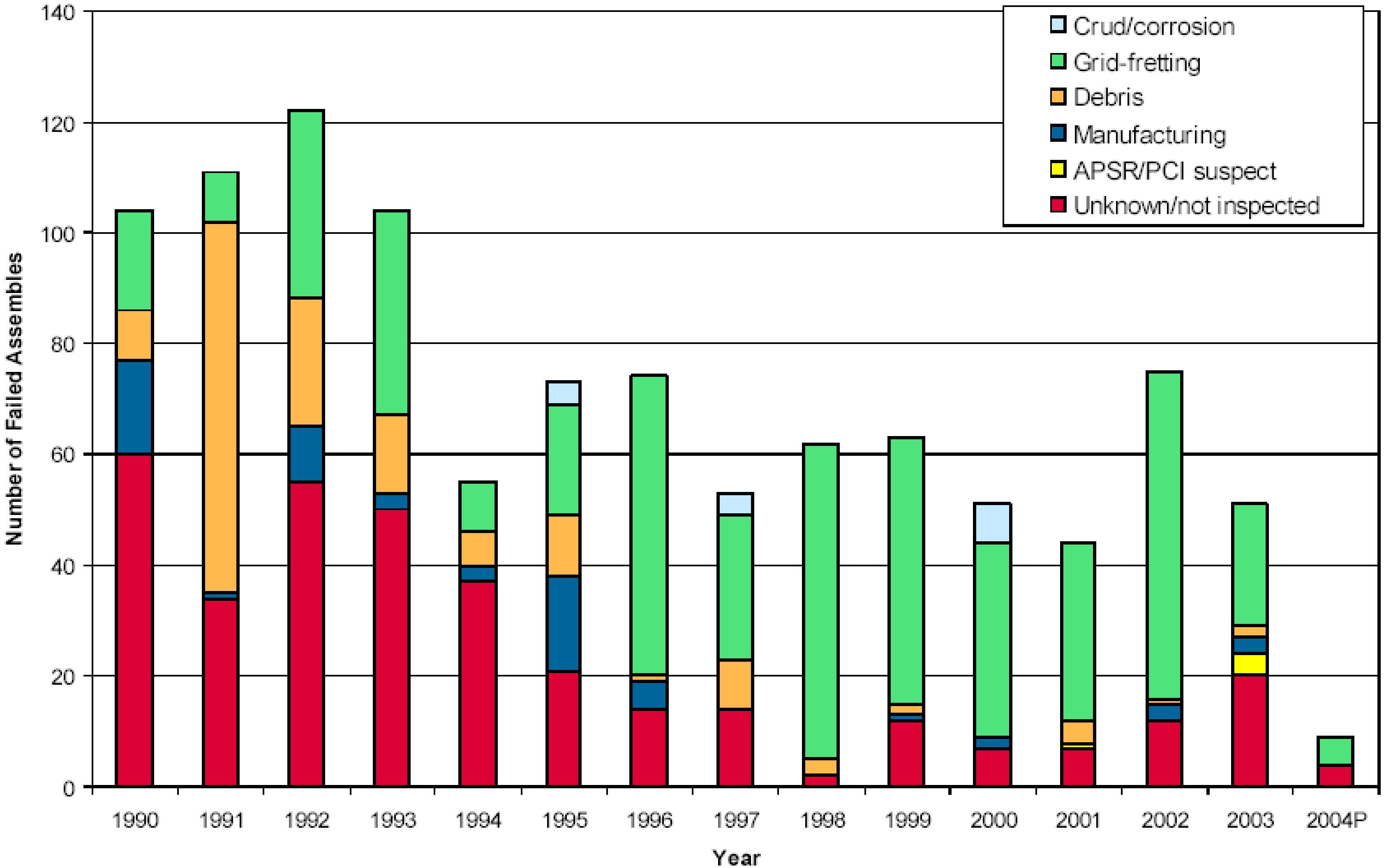


Causes of fuel failure of BWR in USA

ANS Meeting 2004 R.Yang

Figure 2. Trend in US failure root causes

(2004 results are incomplete)



Causes of fuel failure of PWR in USA
ANS Meeting 2004 R.Yang

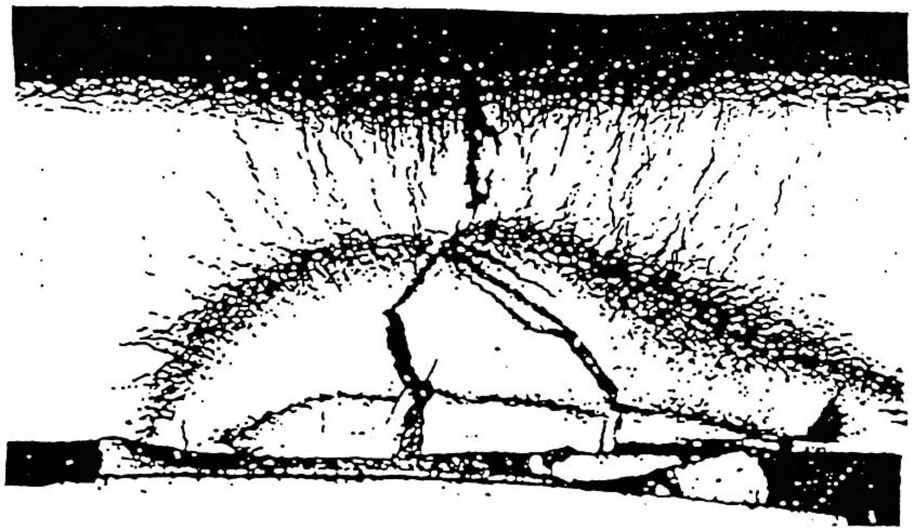
Figure 3. Trend in US PWR failure root causes
(2004 results are incomplete)

Cause and countermeasure of fuel failure

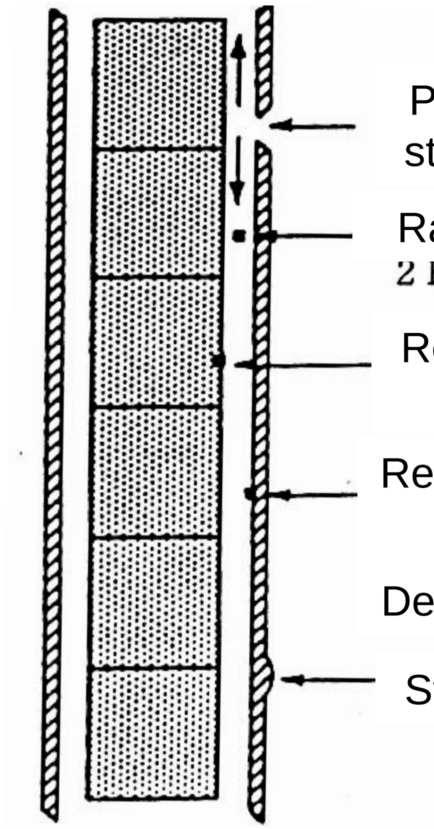
Cause of failure	Countermeasure to prevent the failure
· Manufacturing defect (BWR/PWR)	quality control(QC),improvement of inspection method, improvement of welding
· Local hydriding (BWR/PWR)	control of moisture in the rod during fablication, hydrogen absorber(Zr-Ti-Ni alloy,BWR)
· CILC (Crud Induced Localized Corrosion,BWR)	removal of copper-containig material, control of water chemistry
· Extraordinary corrosion (BWR/PWR)	control of water chemistry, improvement of cladding
· PCI(SCC) (BWR/PWR)	improvement of pellet shape(L/D,dish/chamfer) Zr-liner cladding(BWR),operation mode
· Debris fretting (BWR/PWR)	removal of debris(debris filter,check of maintenance work)
· Baffle jet fretting (PWR)	modification of baffle flow
· Grid fretting (PWR)	improvement of the space between the lower grid
· Collapse of cladding (PWR)	improvement of pellet density, increase of initial gas pressure

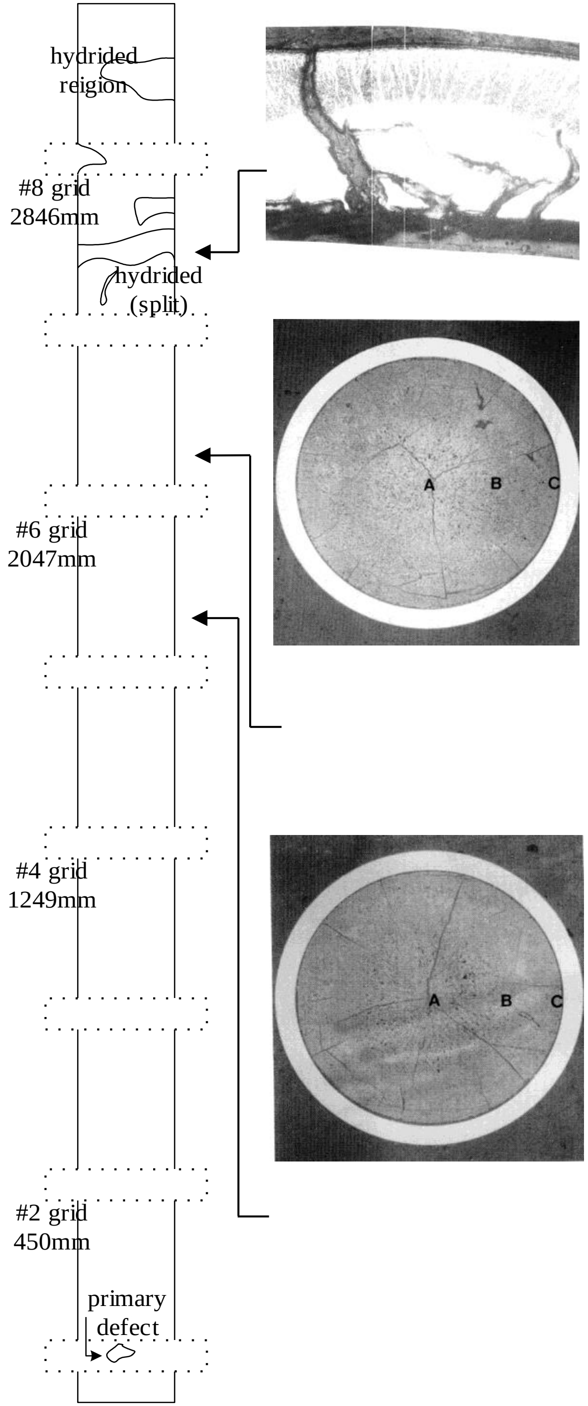
(1) Exsamples of fuel failure

Causes：PCI, hydridation of cladding, grid fretting, Nodular corrosion etc



Sun-burst failure

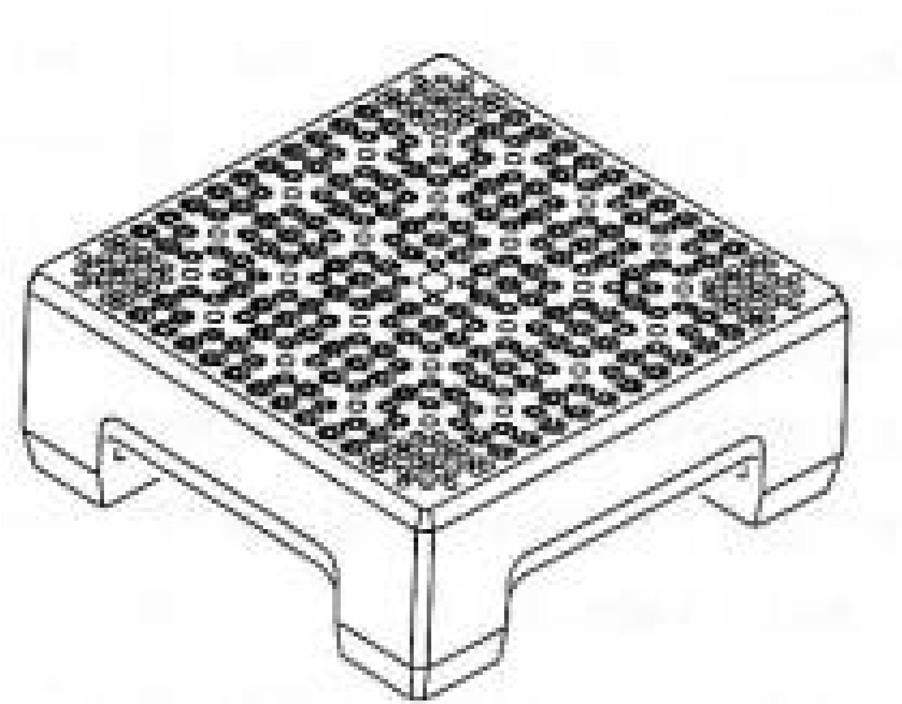




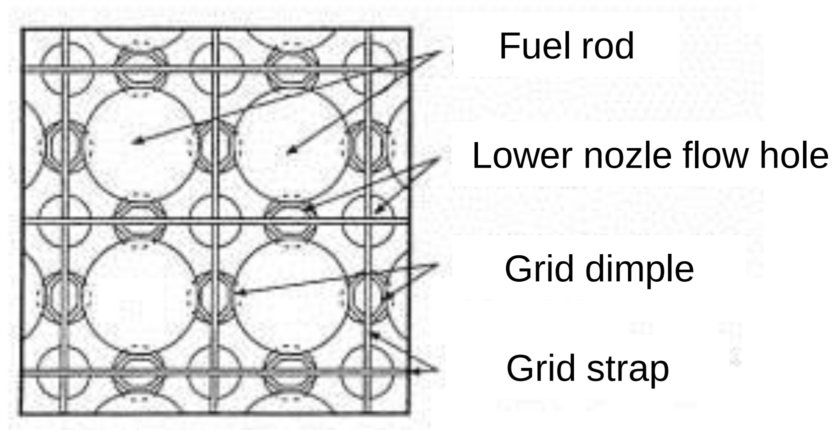
Secondary failure occurred in Korean PWR fuel

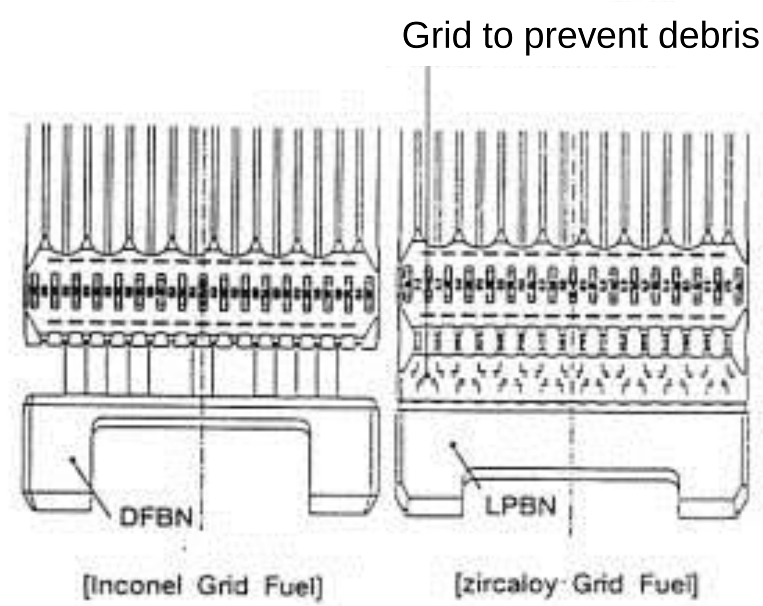
Water invaded from primary defect at lower part of rod and made secondary failure at upper position.

Countermeasure for PWR fuel failure by debris fretting



Lower nozzle to prevent debris for PWR assembly



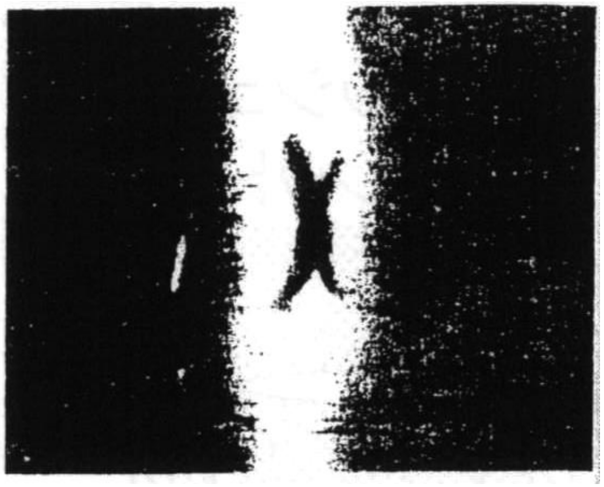


Pellet Cladding Interaction (PCI)

PCI／SCC failure

Long axial crack

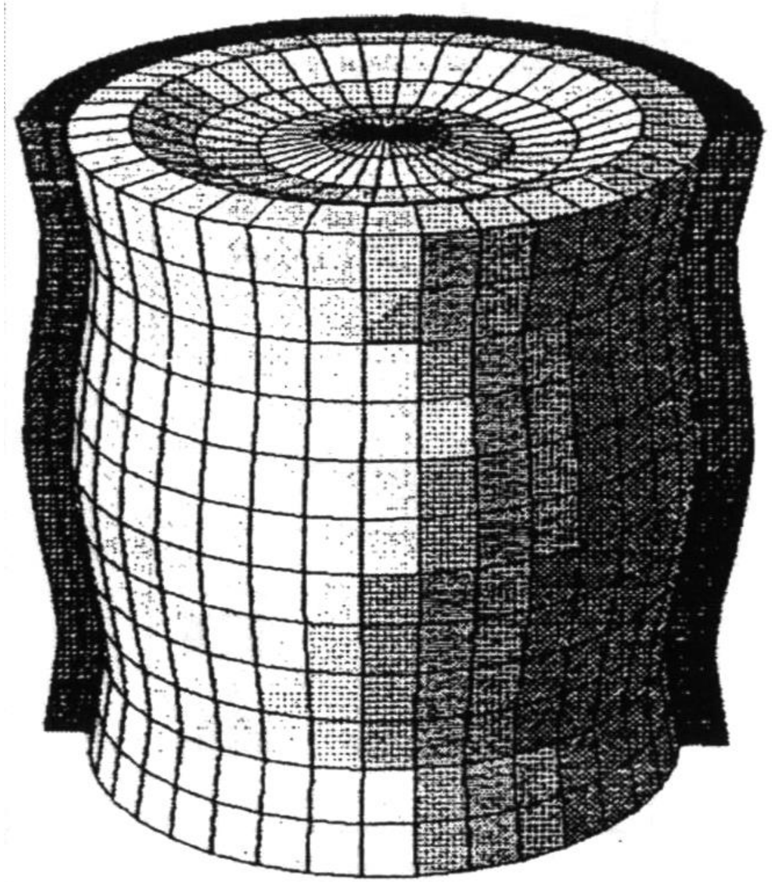
X mark due to plastic deformation
around penetrated defect



PCI failure is stress corrosion cracking
due to large local stress of cladding and
corrodants such as Iodine during power ramp.

Ridge deformation （enlarged deformation）

FUEL DEFORMATION
L.H.R.= 49 (KW/m)

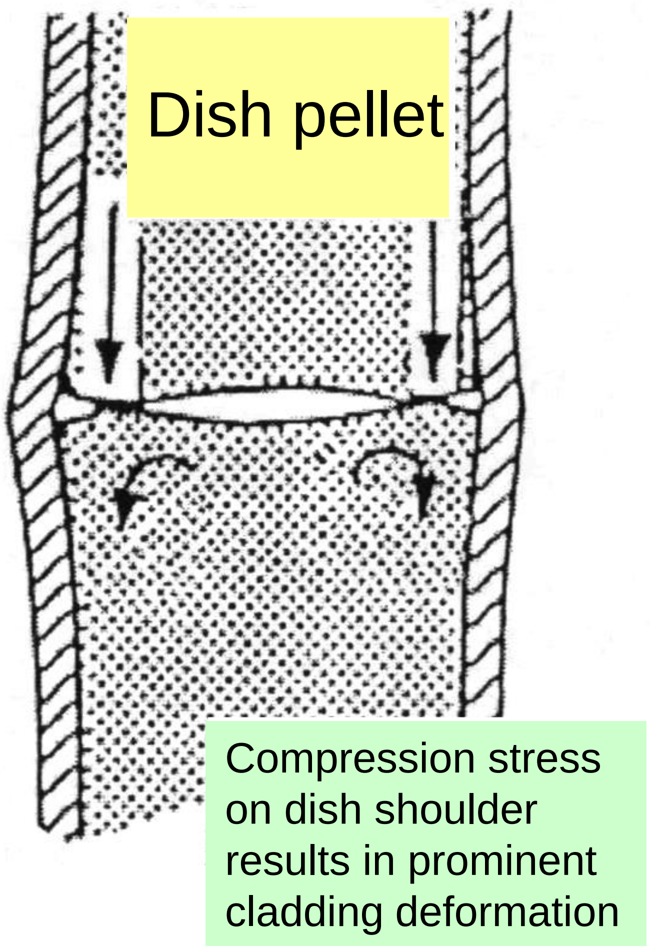


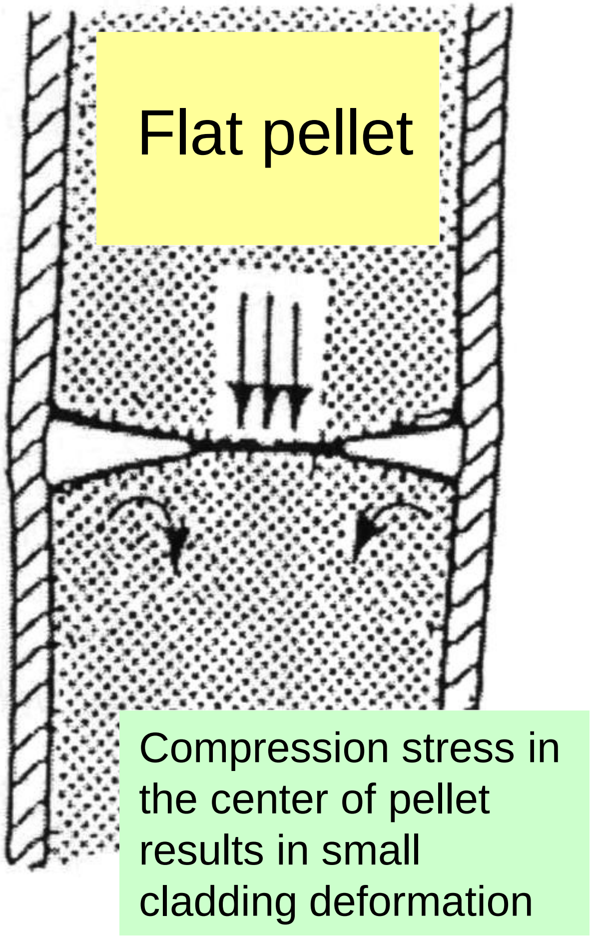
Calculation results by FEMAXI-III

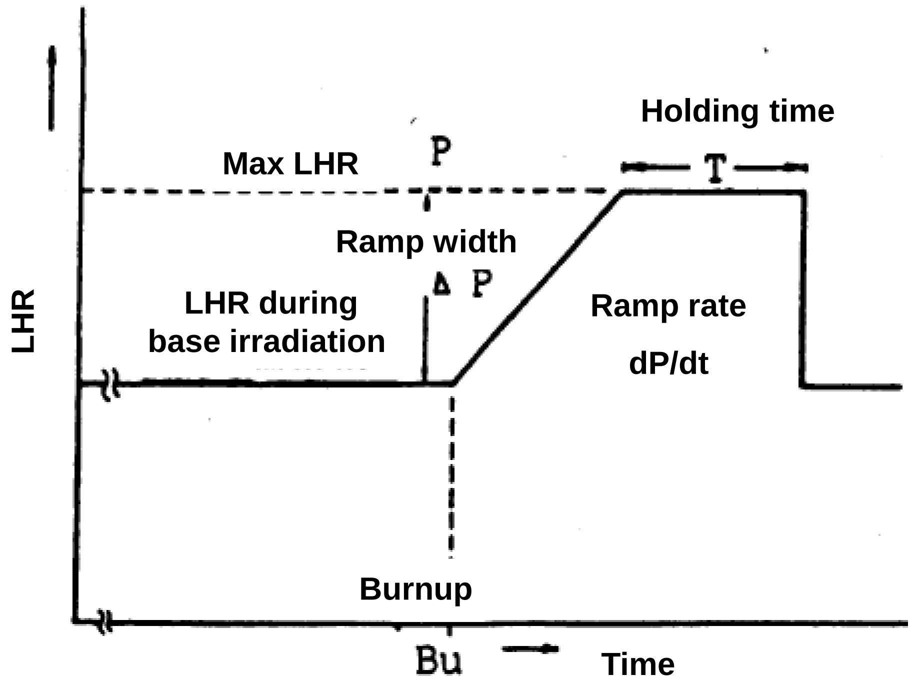
M. Ichikawa, et al., BNES Conf. Stradtford-upon Avon (1985)

Pellet shape and ridging deformation

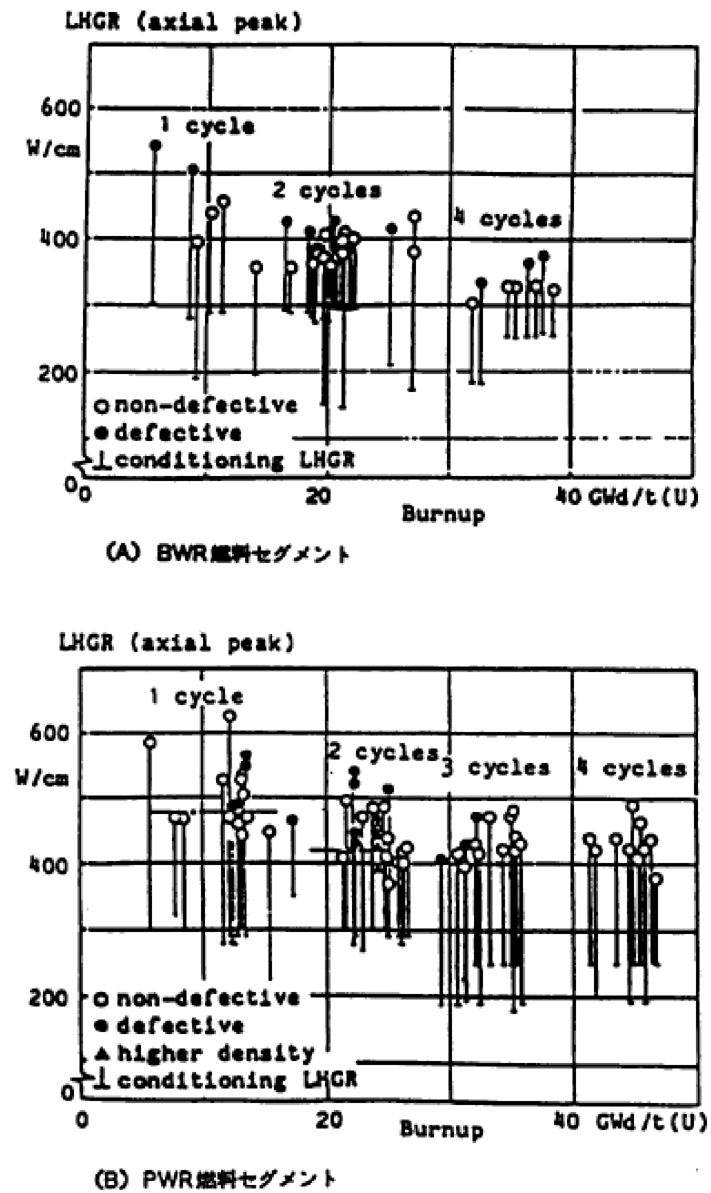
Calculation results of FEMAXI-III code







Power history of Ramp test and main paramet



Power Ramp test results of KWU

(a) BWR (b)PWR

1．8　development of high burnup fuel

- To Improve of economics： reduction of fuel cycle cost
  high burnup use、 reduction of spent fuel

- Effective use of resources： use of plutonium as MOX fuel in LWR

BWR Step I II III	Take out average burnup 33 GWd/t 39.5 GWd/t 45 GWd/t	Maximum assembly burnup 40 GWd/t 50 GWd/t 55 GWd/t (from 1999)
PWR former Step I II	Maximum assembly burnup 39 GWd/t 48 GWd/t 55 GWd/t (from 2004)	

Subjects for high burnup fuel development

1. Integrity and reliability of fuel

◆Reduction of PCMI→Zr liner, soft pellet

◆Waterside corrosion→improvement of alloy, heat treatment

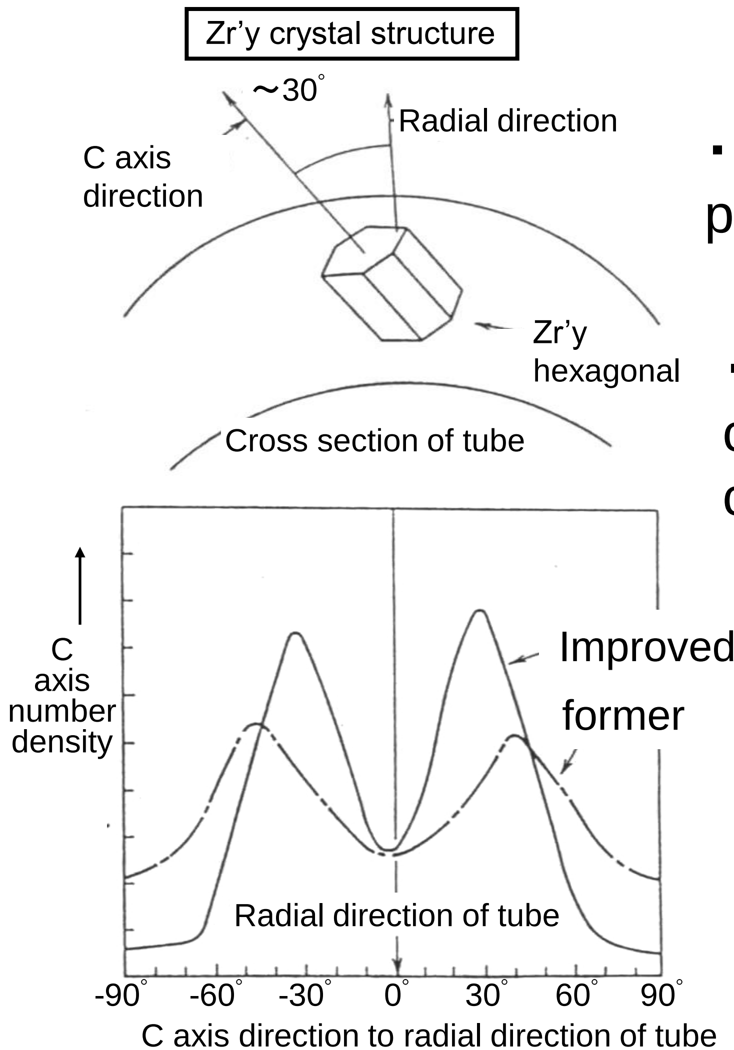
◆FGR→large grain fuel

◆Formation of rim structure and reduction of thermal conductivity, increase of pellet temperature

→affects to PCMI, FGR, swelling, etc

2.Increse of initial enrichment and control of excess reactivity → Introduction of high concentration gadolinia added pellet

Texture adjusted cladding for PCI-（principle）



・SCC propagates through the bottom plane of hexagonal structure, easily.

・To form texture that bottom plane comes to circumferential direction, ie, c axis comes to radial direction.

・If c axis comes radial direction fully, the deformation of cladding is hard. So, c axis is set about 30° from radial direction.

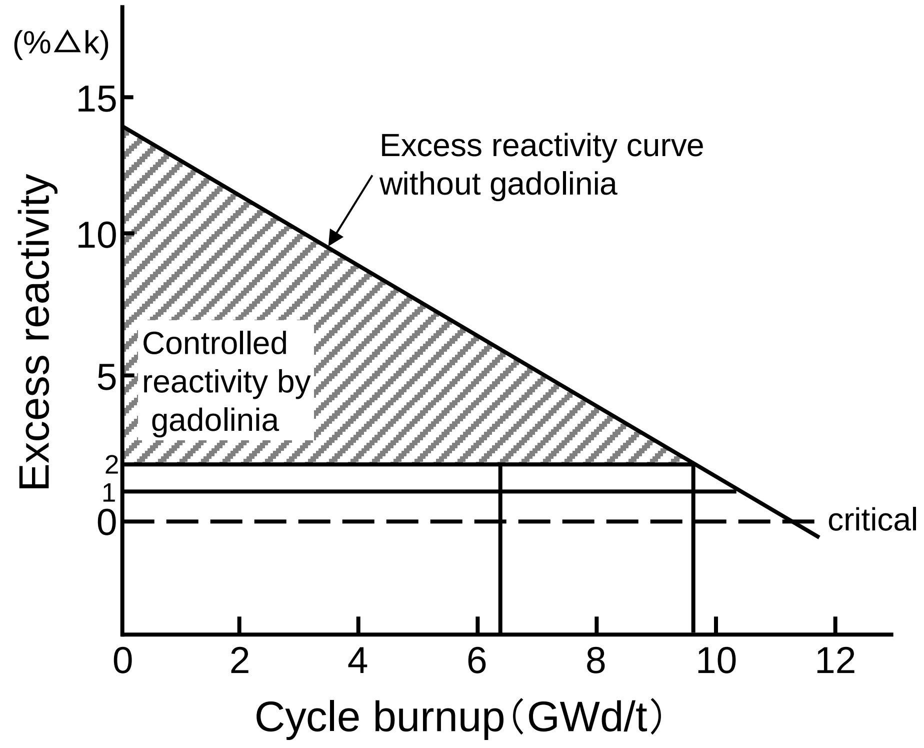
Development of high concentration gadolinia added fuel

High burnup use → increase of initial enrichment
↓
control of initial excess reactivity ← gadolinia addition

The thermal neutron absorption cross section of burnable poison must be much larger than that of U-235.

fission cross section of U-235		580b
absorption cross section of B-10		3,837b
〃	Gd-155	60,900b
〃	Gd-157	25,400b

problem: increase of content results in reduction of melting point and thermal conductivity.



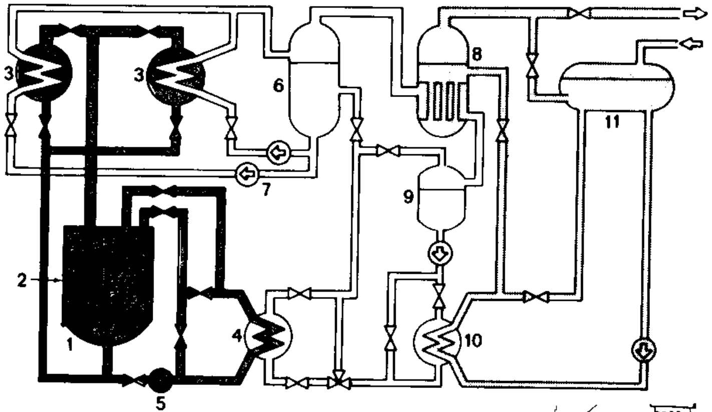
Part 2. Irradiation Test
（Test reactor and instrumentation, PIE technology）

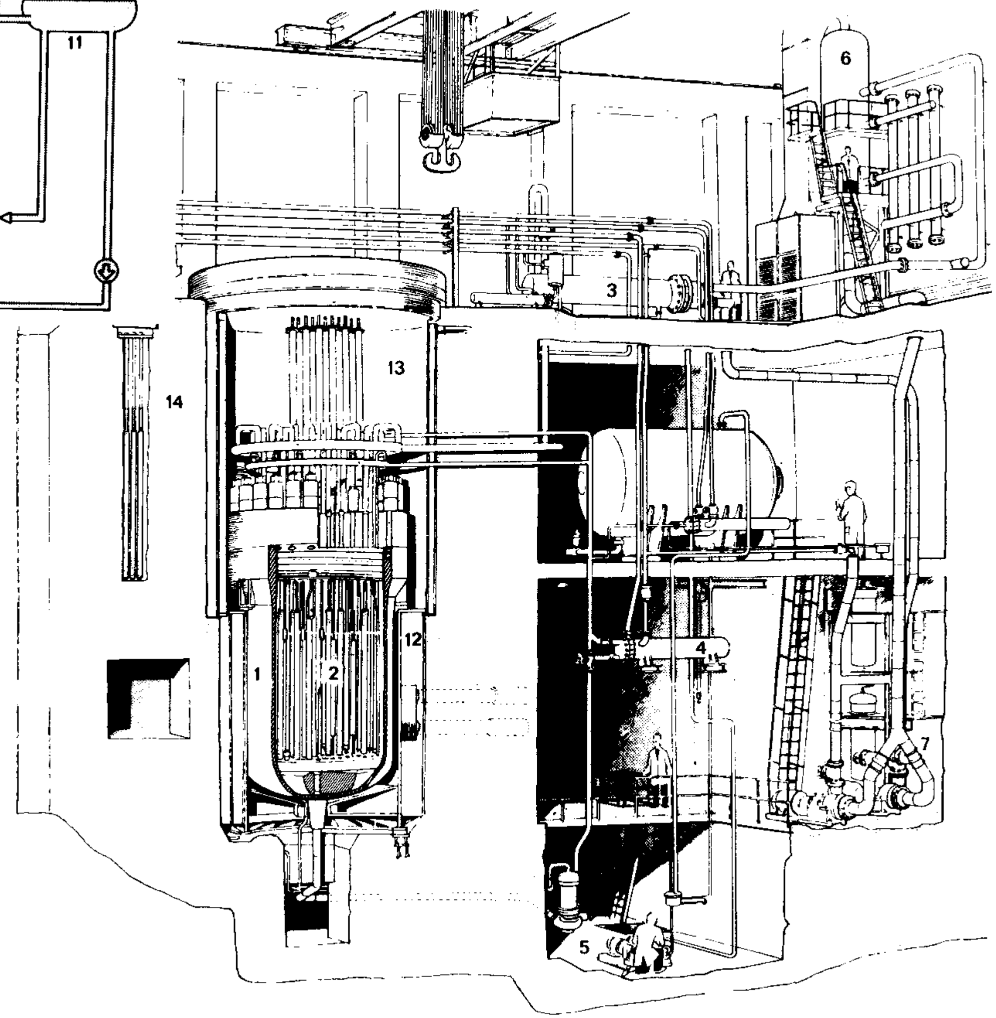
- Norway: OECD Halden Project, Halden reactor(HBWR)：
heavy water boiling reactor for fuel and material test for LWR.
In-pile instrumentation and in-pile test technology is advanced.
- ☆fuel behavior data during irradiation is measured by on-line instrumentation.
  - ①fuel center temperature
  - ②cladding deformation, PCMI,
  - ③fission gas release（rod inner pressure increase）

- JAEA-JMTR(the Japan Materials Testing Reactor)
  -under renewal (will restart in 2011)

- commercial reactors

2.1 Halden reactor(HBWR)

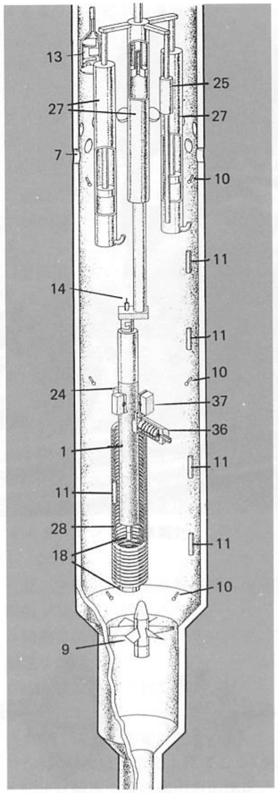




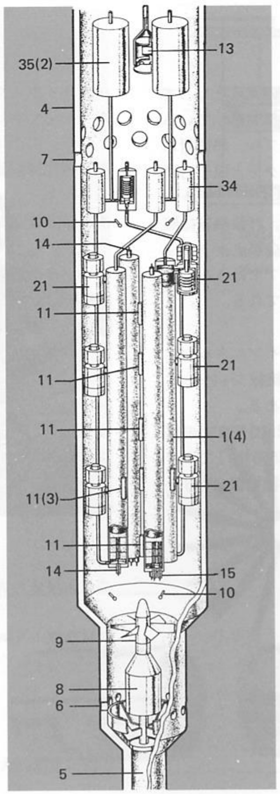
1. Reactor vessel
2. Reactor core
3. Heat exchanger
4. D_2O subcooler
5. Subcooler pump
6. Steam drum
7. La Mont pumps
8. Steam generator
9. Hot well
10. H_2O subcooler
11. Feed water tank
12. Shield coolant circuit
13. Magnetic jack control rod drive
14. Fuel storage pit

Reactor is located in rock mountain. Heavy water boiling reactor. 34 bar 240C 20MW

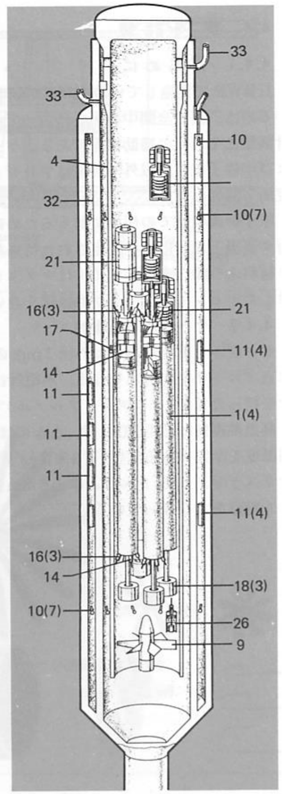
Instrumented fuel assembly -1-



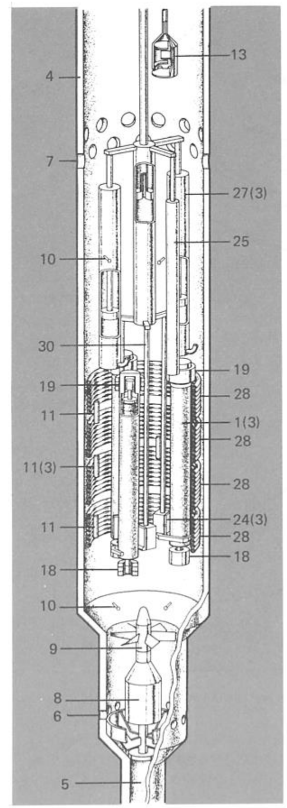
Fuel Rod Gap Meter Rig

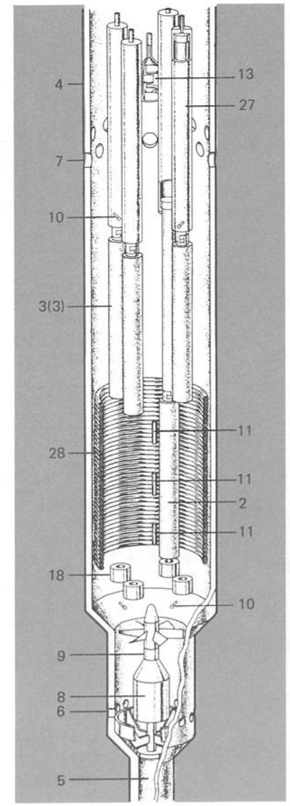


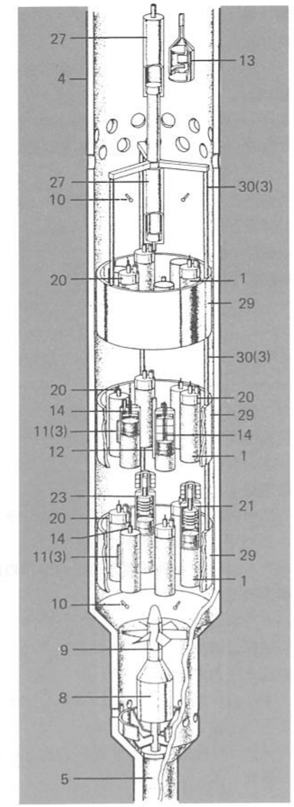
Gas Transport Rig



Flow Starvation Rig
PWR/BWR Forced Circulation Loop







1. Fuel rod
2. Fuel rod in operation
3. Fuel rod, stand by
4. Shroud
5. Forced circulation subcooled inlet
6. Coolant inlet ports, natural circulation
7. Coolant outlet ports
8. 3-way solenoid valve (forced/natural circulation)
9. Coolant turbine flow meter
10. Coolant thermocouples
11. Flux sensor (Va)
12. Flux sensor (Co)
13. Fuel failure indicator, steam sampler
14. Centre fuel thermocouple
15. Fuel perifery thermocouple
16. Cladding thermocouple
17. Shroud thermocouple
18. Cladding extensometer
19. Fuel column extensometer
20. Fission gas pressure gauge, membrane
21. Fission gas pressure gauge, bellows
22. Coolant pressure gauge
23. Cladding extensometer and failure detector
24. Fuel rod diameter gauge
25. Position indicator
26. Water level detector
27. D_2O hydraulic drive cylinder
28. He-3 tubular coil
29. Neutron absorbing shield
30. Lifting rod
31. Fuel rod rotating mechanism
32. Pressure flask
33. D_2O circulation and pressurizing tube
34. Bellows controlled valve
35. Gas bottles
36. Fuel rod compression mechanism
37. Compression sensor

1. Fuel rod
2. Fuel rod in operation
3. Fuel rod, stand by
4. Shroud
5. Forced circulation subcooled inlet
6. Coolant inlet ports, natural circulation
7. Coolant outlet ports
8. 3-way solenoid valve (forced/natural circulation)
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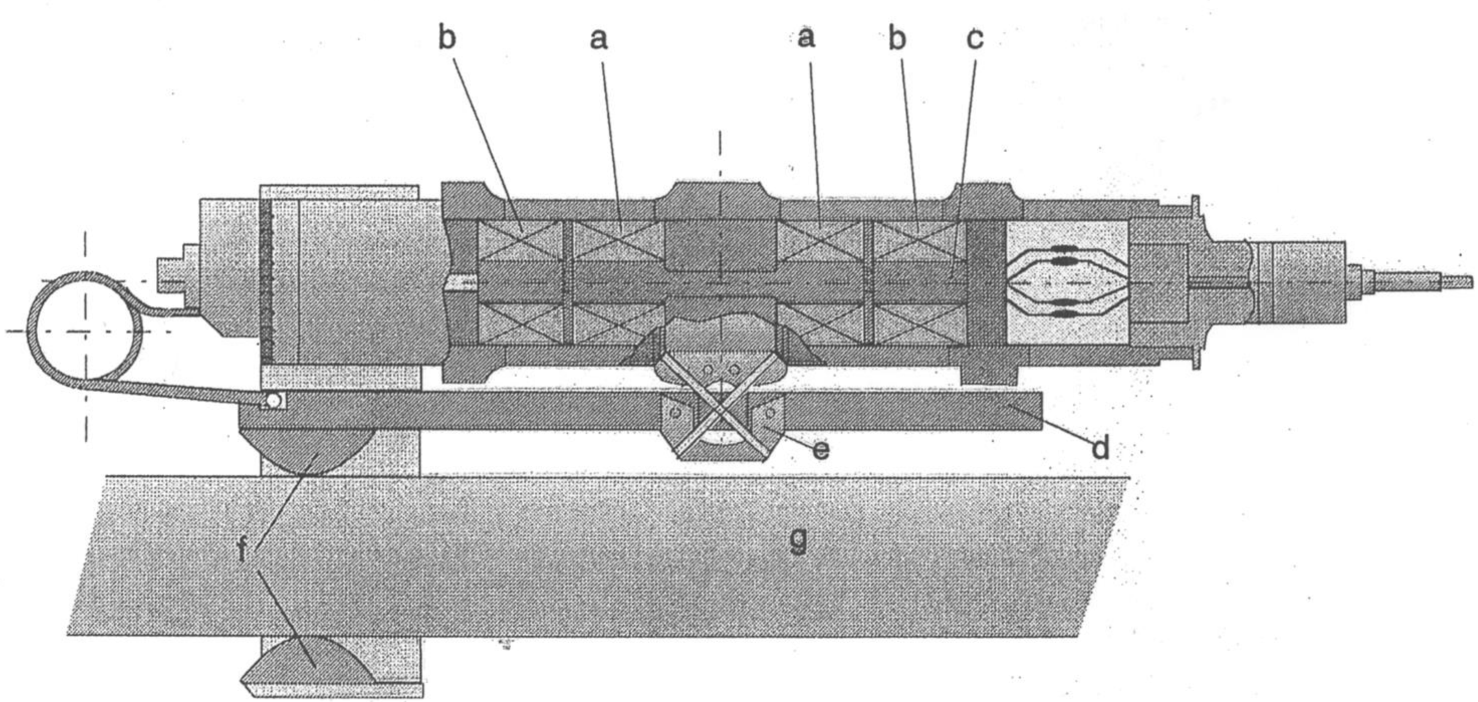
19. Fuel column extensometer
20. Fission gas pressure gauge, membrane
21. Fission gas pressure gauge, bellows
22. Coolant pressure gauge
23. Cladding extensometer and failure detector
24. Fuel rod diameter gauge
25. Position indicator
26. Water level detector
27. D_2O hydraulic drive cylinder
28. He-3 tubular coil
29. Neutron absorbing shield
30. Lifting rod
31. Fuel rod rotating mechanism
32. Pressure flask
33. D_2O circulation and pressurizing tube
34. Bellows controlled valve
35. Gas bottles

Fuel behavior during irradiation by various in-pile instrumentations, power change by He-3,

Diameter gauge for fuel rod

Standard Diameter Gauge



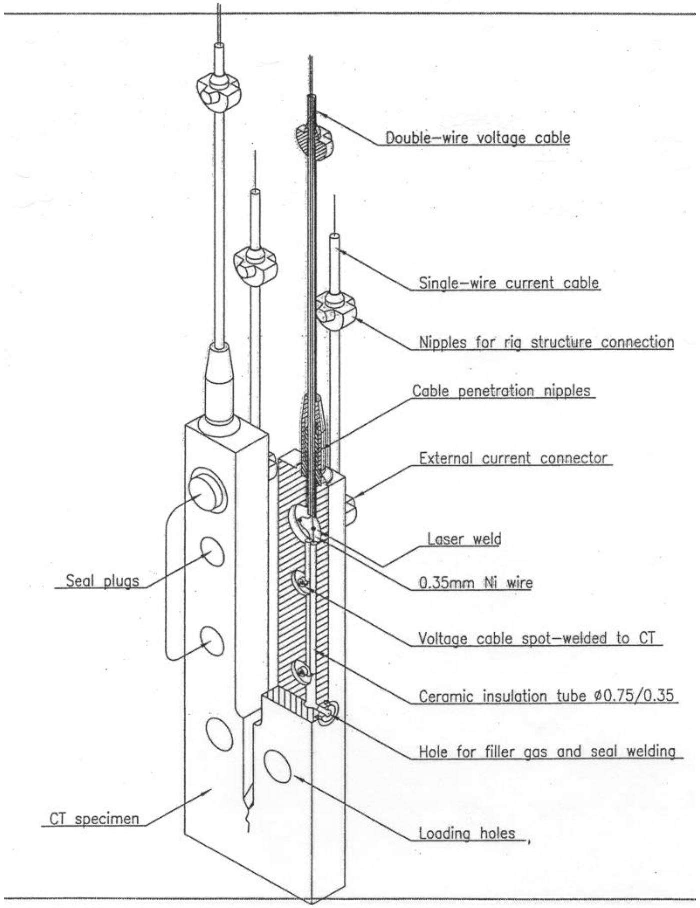
a: Primary coil
b: Secondary coil
c: Ferritic bobbin
d: Ferritic armature
e: Cross spring suspension
f: Feelers
g: Fuel rod

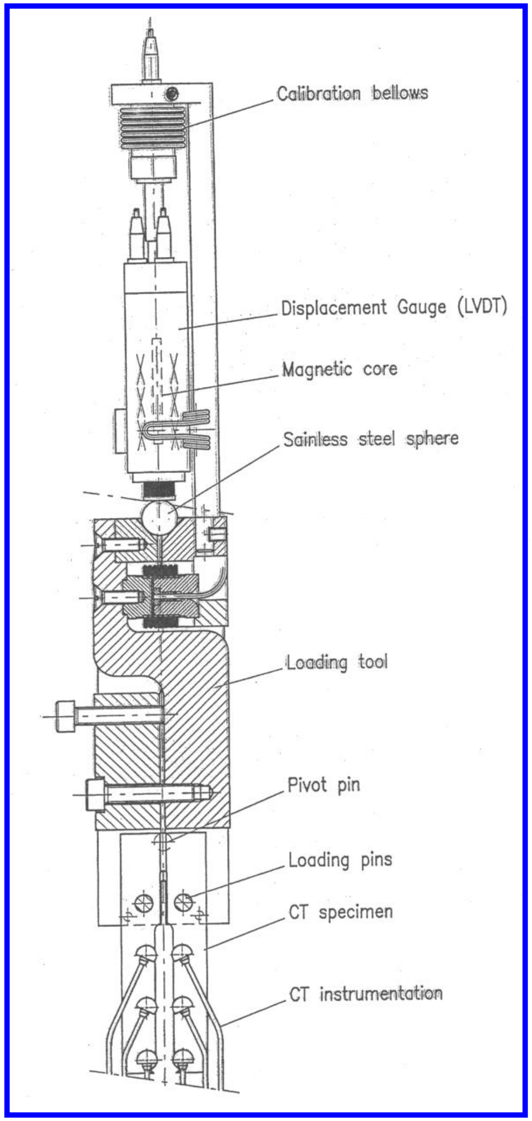
Precision of diameter measurement = about 1µm

IASCC test sensor

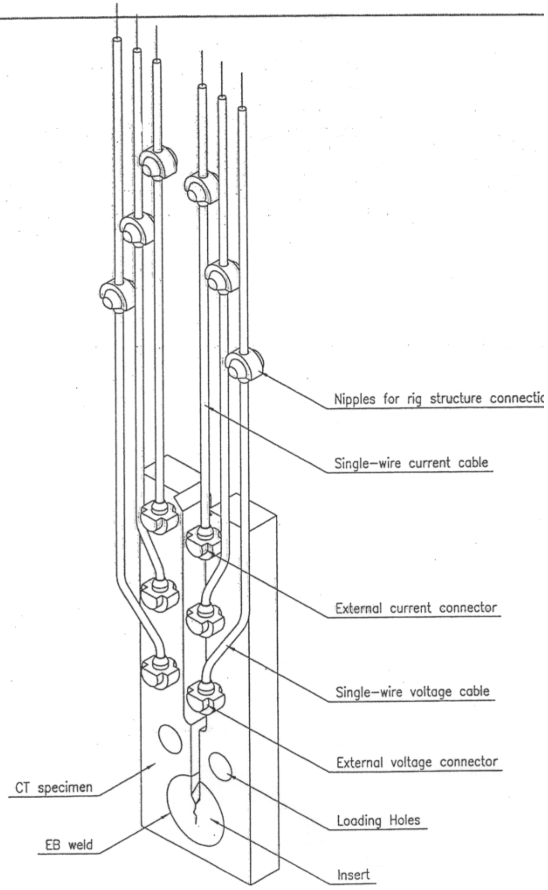
（crack propagation rate is detected by potential change）

Principle layout of the in-core type CT specimen





CT specimen with active insert and external connectors



2.4 JAEA-JMTR (the Japan Materials Testing Reactor)

Main performance of JMTR

1. Irradiation area of core
   - Simultaneous irradiation positions : about 60
   - Low gamma irradiation area

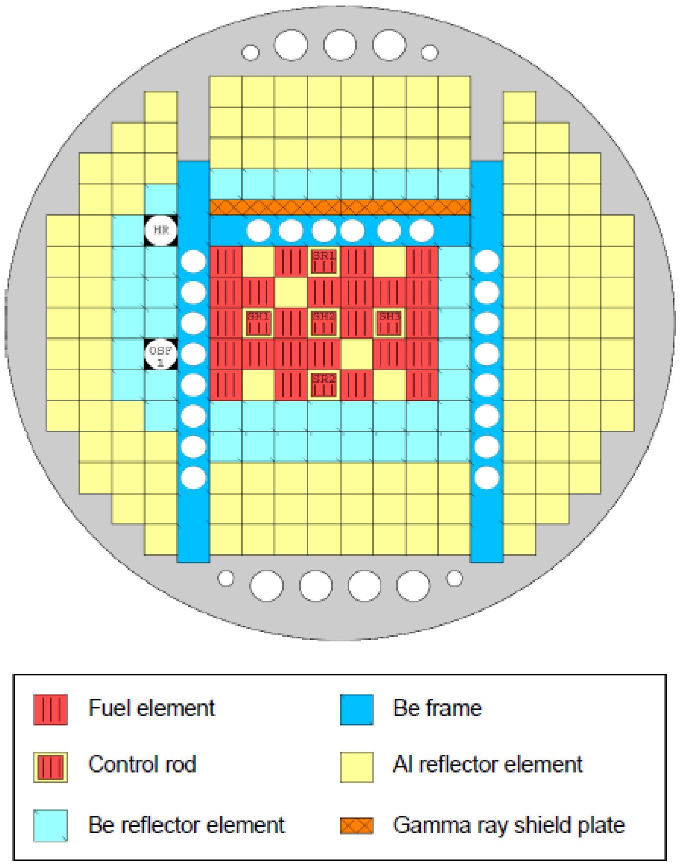
2. Neutron flux
   - Fast : max 4×10^{18} (n/m^2/s)
   - Thermal : max 4×10^{18} (n/m^2/s)

3. Neutron fluence /y (at 180 days operation*)
   - Fast and thermal : max 3×10^{25} (n/m$^{2)}$
   - dpa (SUS) : max 4 (dpa)

4. Dimensions of irradiation capsule
   - ϕ 40mm × 750mm (outer diameter : max 65mm)

5. Irradiation temperature
   - Controlled from 50 to 2000 °C

*: results of Oct.2003~Spt.2005



Configuration of standard core

Irradiation Test with the JMTR

Temp. control
Saturation temp.
Automatic constant temp.
High temp.

Environmental control
Water chemistry, load

Special instrumentation
Displacement, crack propagation
Re- instrumentation (temp., pressure)

Power ramping
Fuel power control by ^{3}He gas

Neutron control
Spectrum adjustment
Pulse irradiation

Re-irradiation
Assembling in hot cell

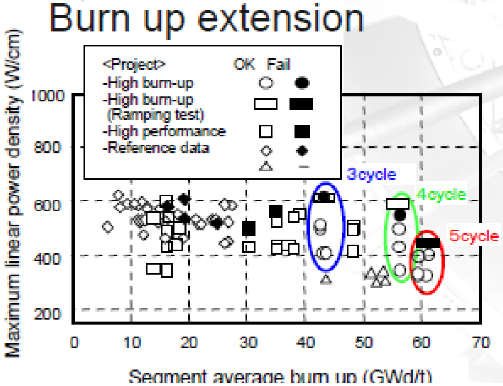
IASCC test of LWR core internals

Life time extension of LWR

shroud

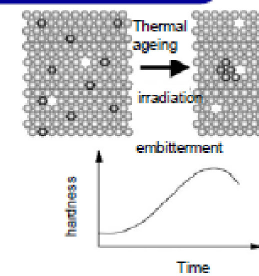
Power ramping test of LWR fuels

Burn up extension



High accuracy temp. control

Research of radiation damage



High temp. irradiation

Development of HTTR

ITER operation, He production rate/dpa simulation

Development of fusion reactor

2.5 Post Irradiation Examination (PIE)

①whole fuel rod

・rod puncture test = inner pressure, FGR, gas composition

⇔final data for fission gas release model verification

・γscan data（axial distribution of burnup, transportation of FP）

・profilometry of cladding（outer radius, elongation, oxide film：PCMI during irradiation, creep, oxidation)

②Pellet

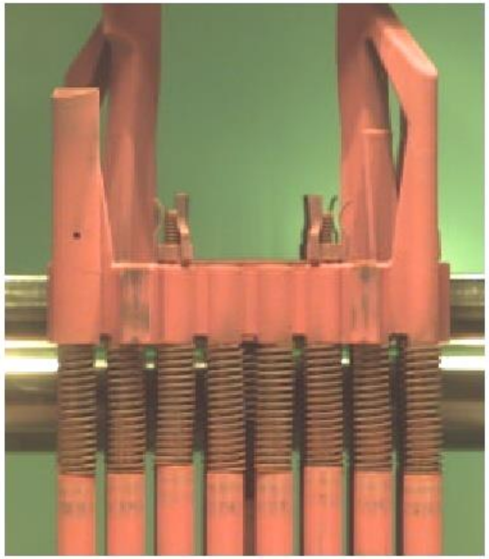
・Metallography/ceramography：density, cracking , grain size distribution（grain growth）, restructuring, pore distribution, rim structure, bonding

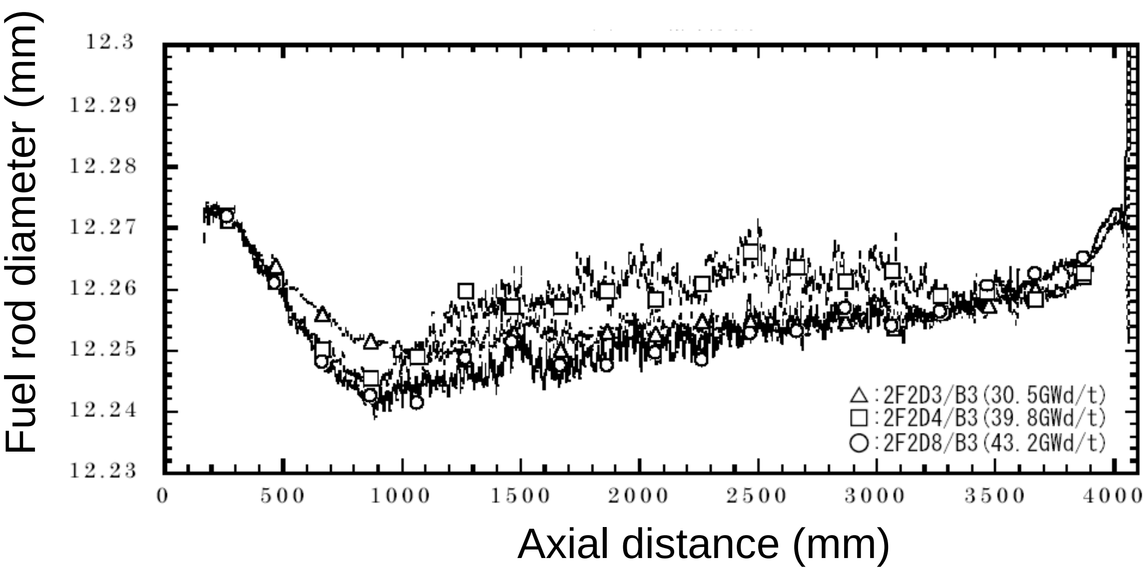
・element analysis：burnup ditribution, FP distribution, Pu accumulation/distribution,

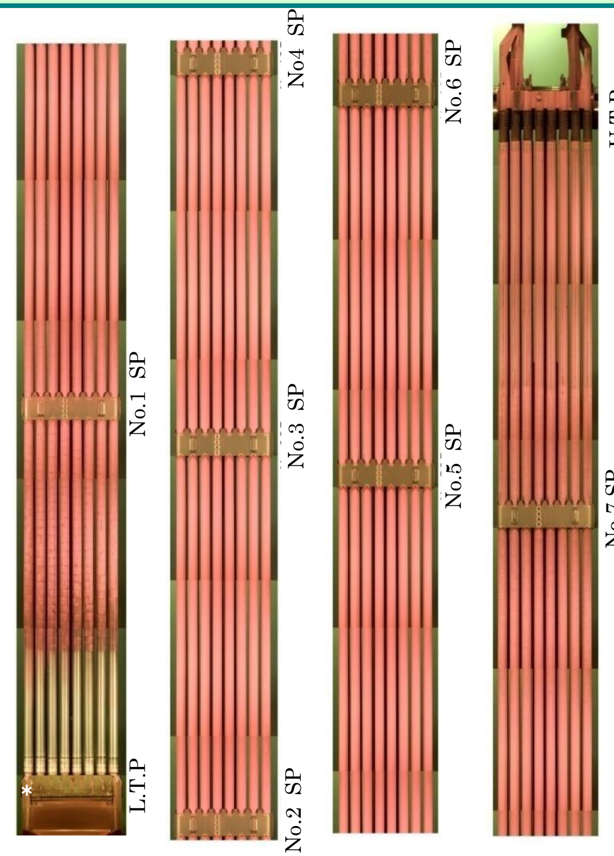
・gap size ：bonding formation

・measurement of thermal conductivity(thermal diffusivity)

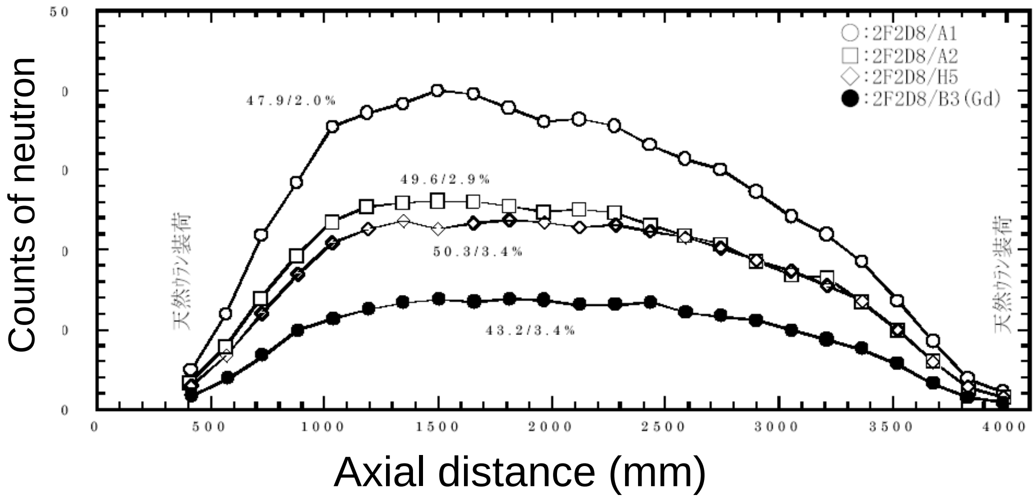
Visual examination, diameter change of fuel rod

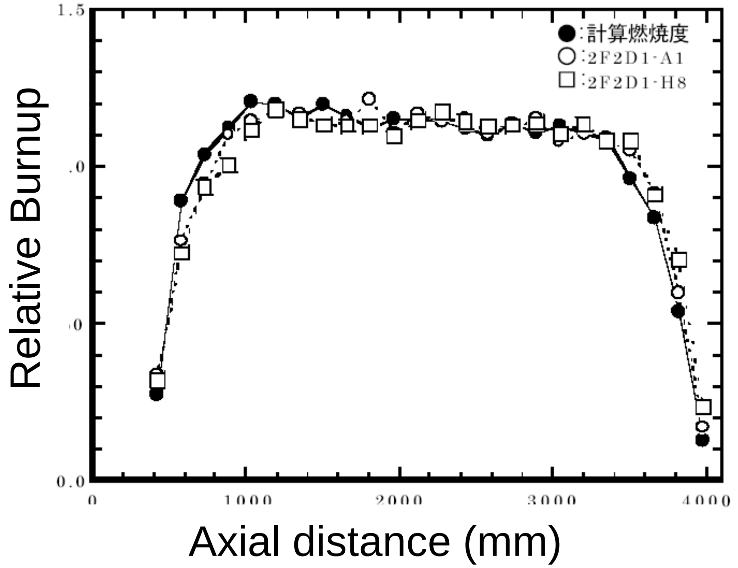




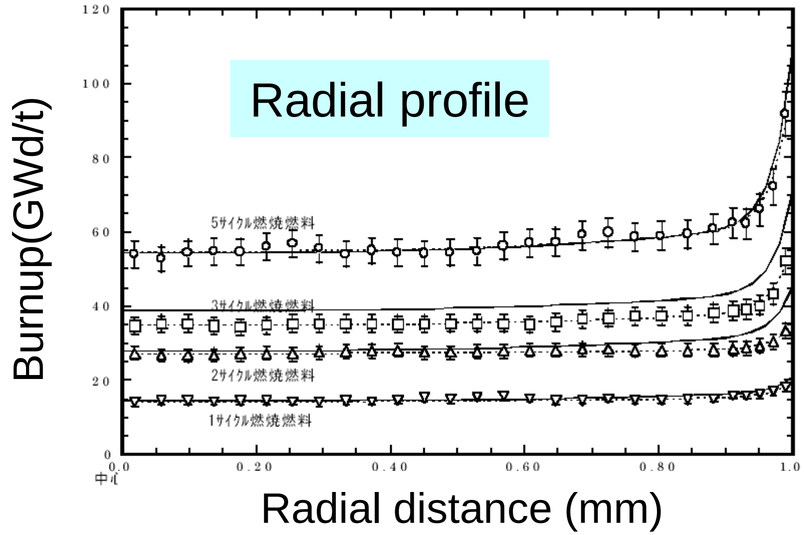


Measurement of burnup distribution in fuel rod

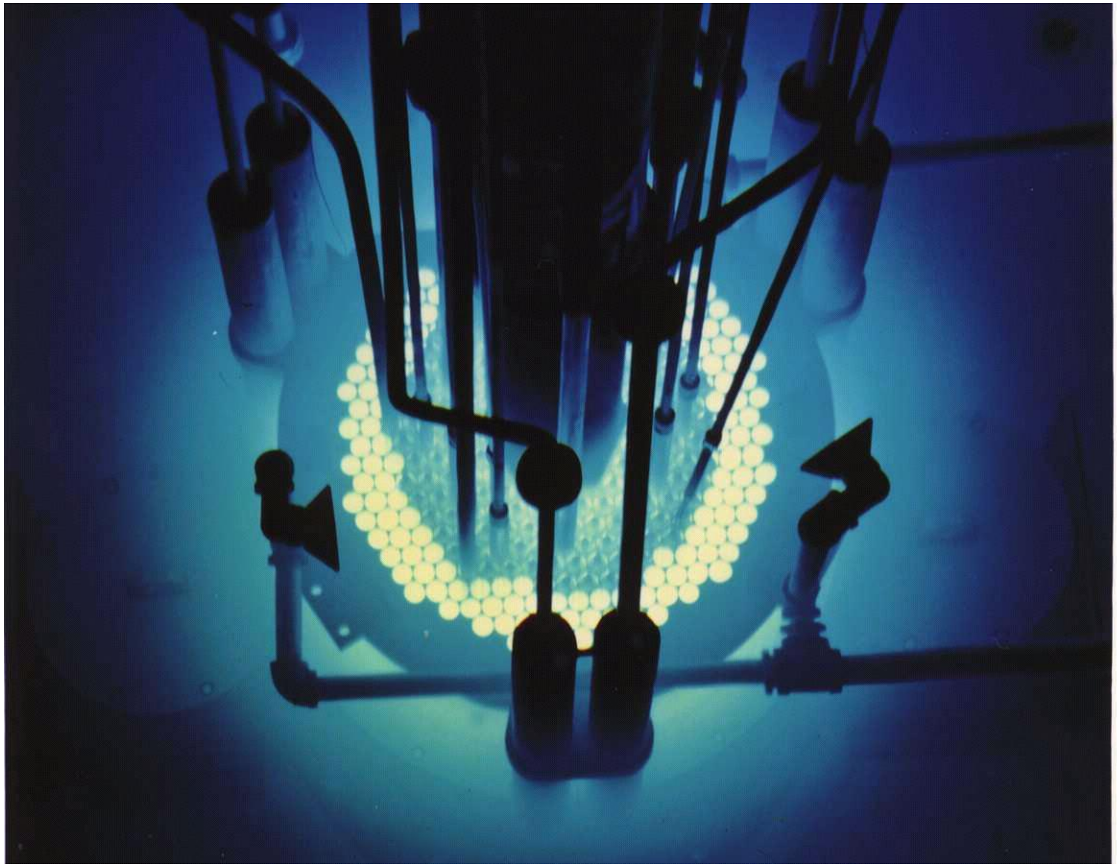




Axial profile



Fuel Behavior under RIA



Cherenkov radiation in the NSRR
(Nuclear Safety Research Reactor) of JAEA

Reactivity Initiated Accident (RIA)

Control rod ejection

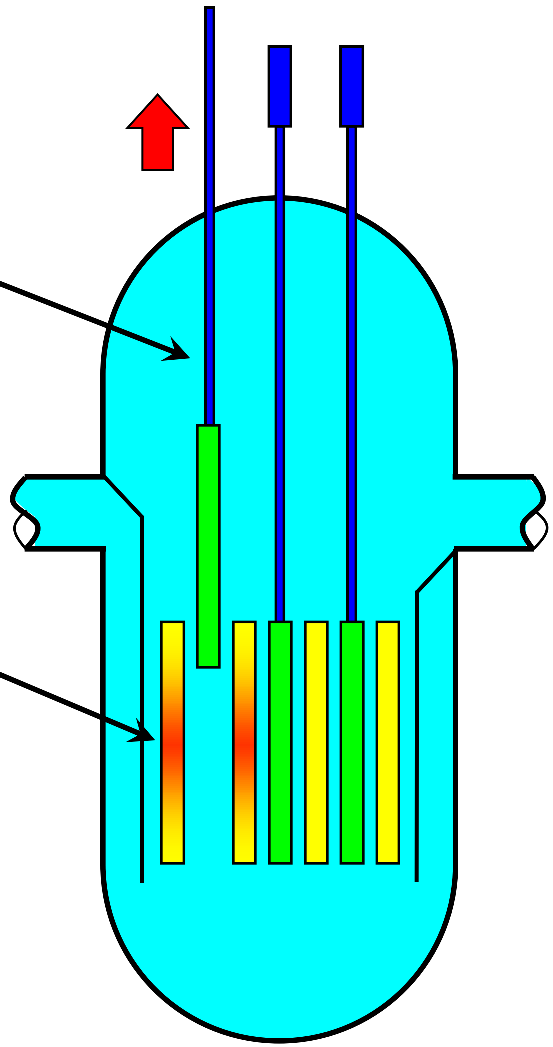
↓

Fuel temperature increase

↓

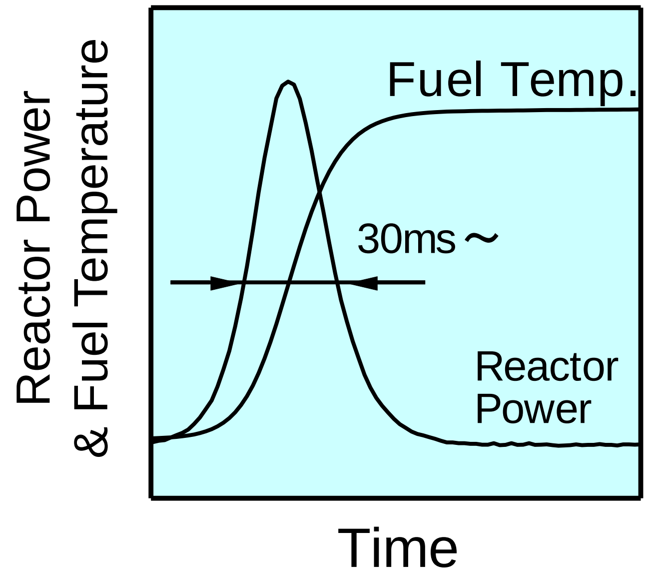
Fuel failure

Possible damage on reactor vessel



Reactor vessel

Power burst in RIA



RIA is a typical DBE (design basis event).

Safety management against RIA

RIA is a very rapid event.
There is nothing we can do after the RIA occurrence.
Safety management against RIA should be made in the design stage.

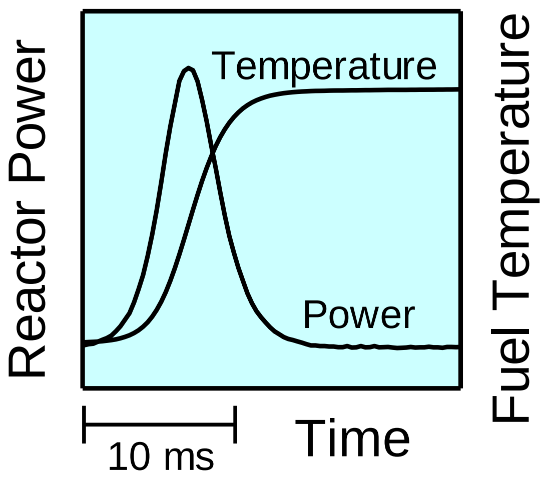
Core designing and safety assessment need the following information:

- Fuel behavior in RIA
- Fuel failure modes and conditions
- Influence of fuel failure

⇨ Needs for experimental data

RIA-simulating experiments
at NSRR (Nuclear Safety Research Reactor)

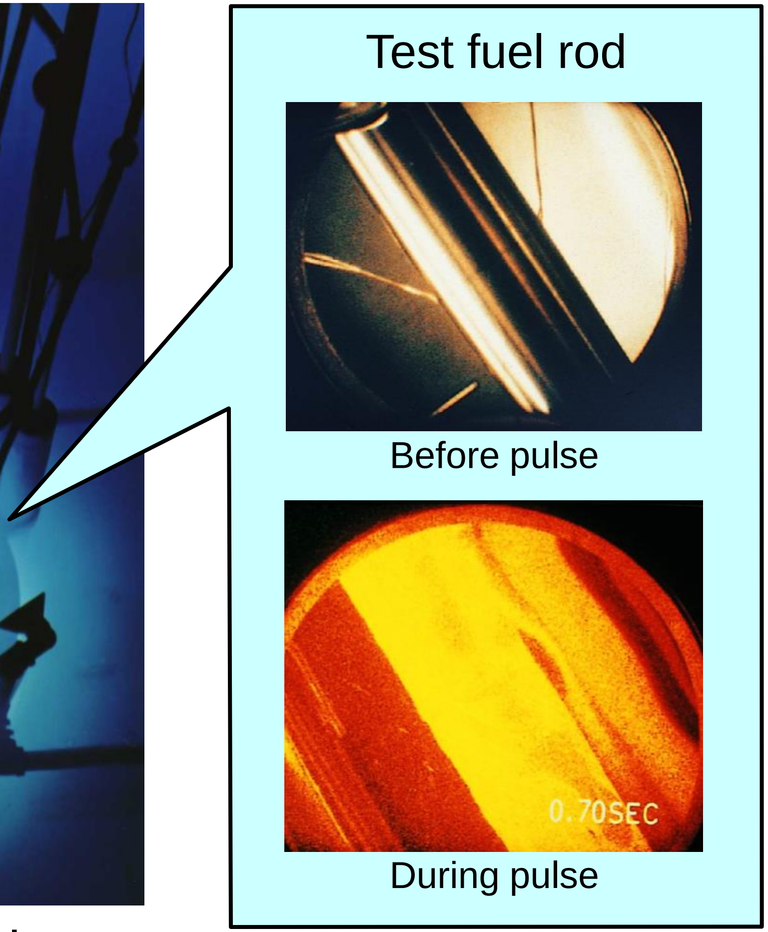
RIA-simulating
pulse operation



Max power: ~23 GW

Pulse width: ~4 ms

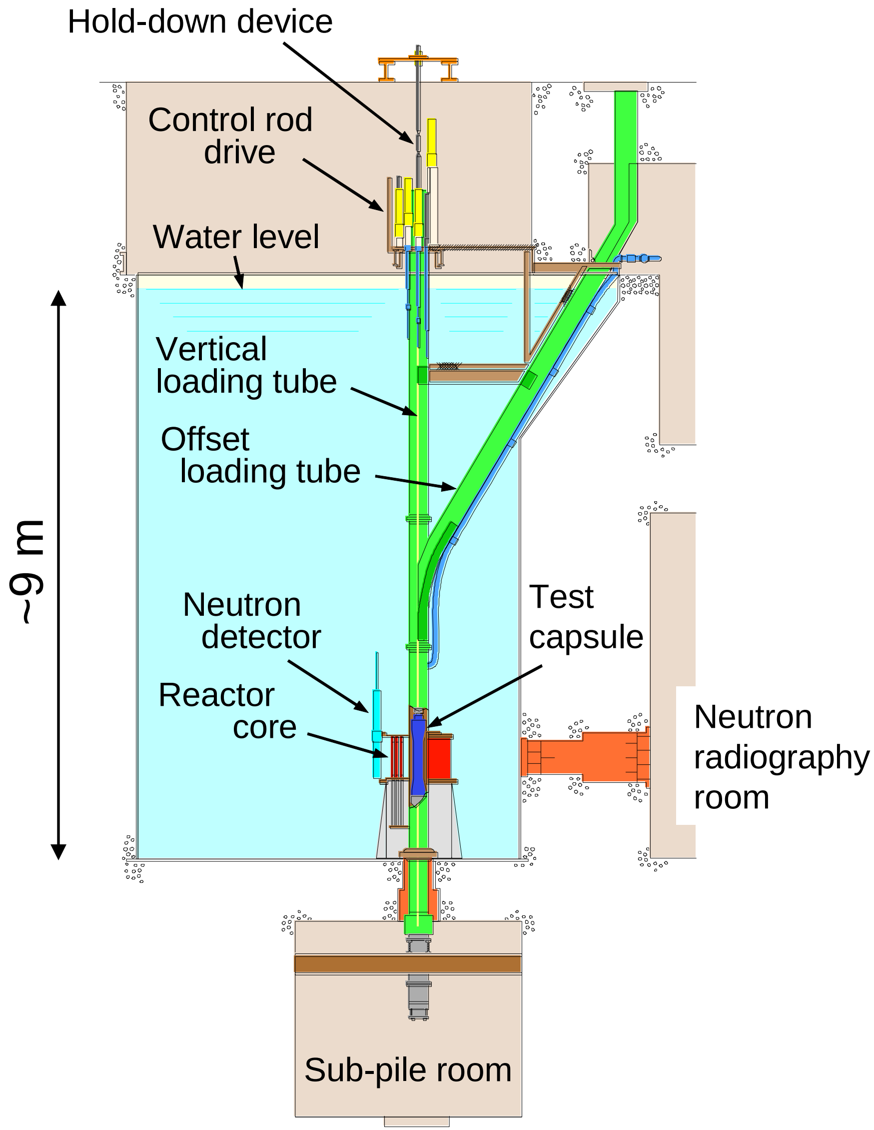
Cherenkov radiation
at NSRR pulse operation

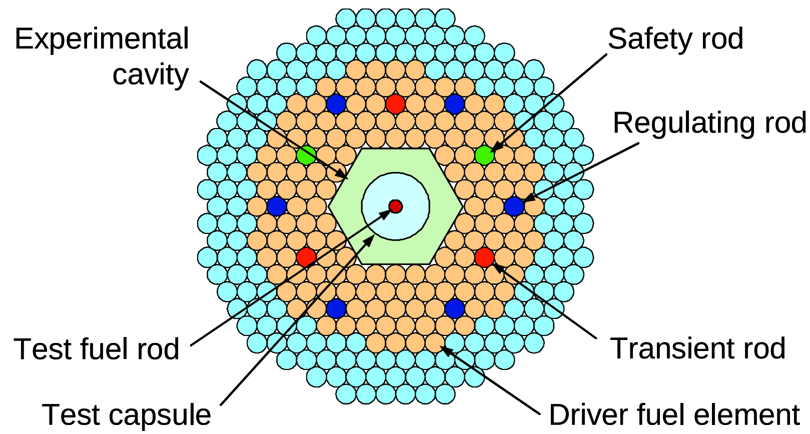


Periscope view
(Fresh fuel tests)

Outline of NSRR facility

Modified TRIGA-ACPR (annular core pulse reactor) constructed in 1975

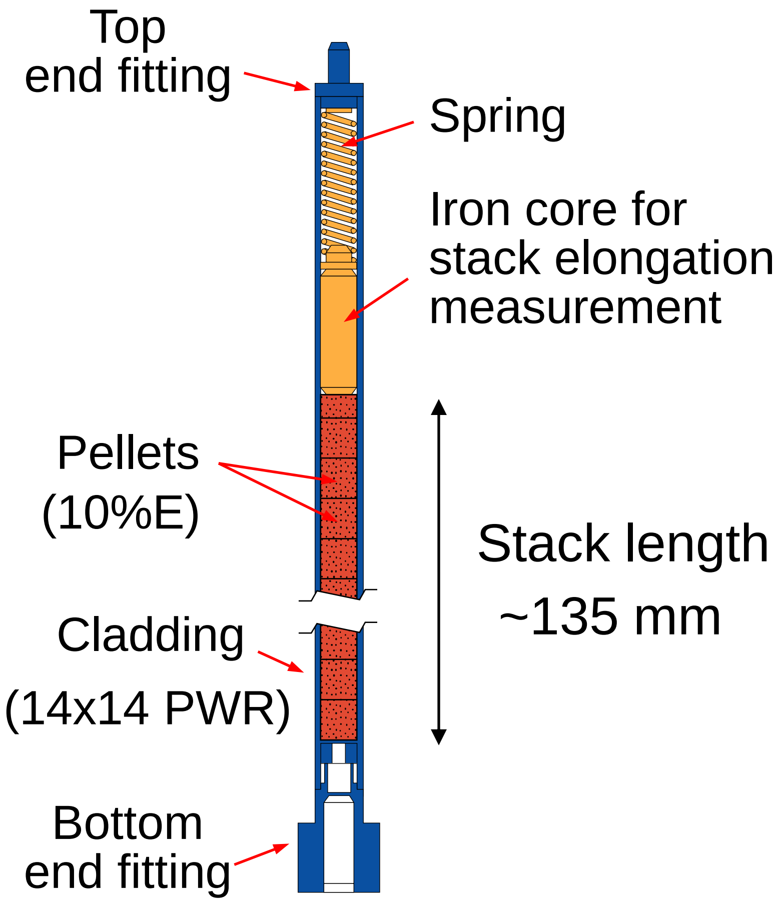




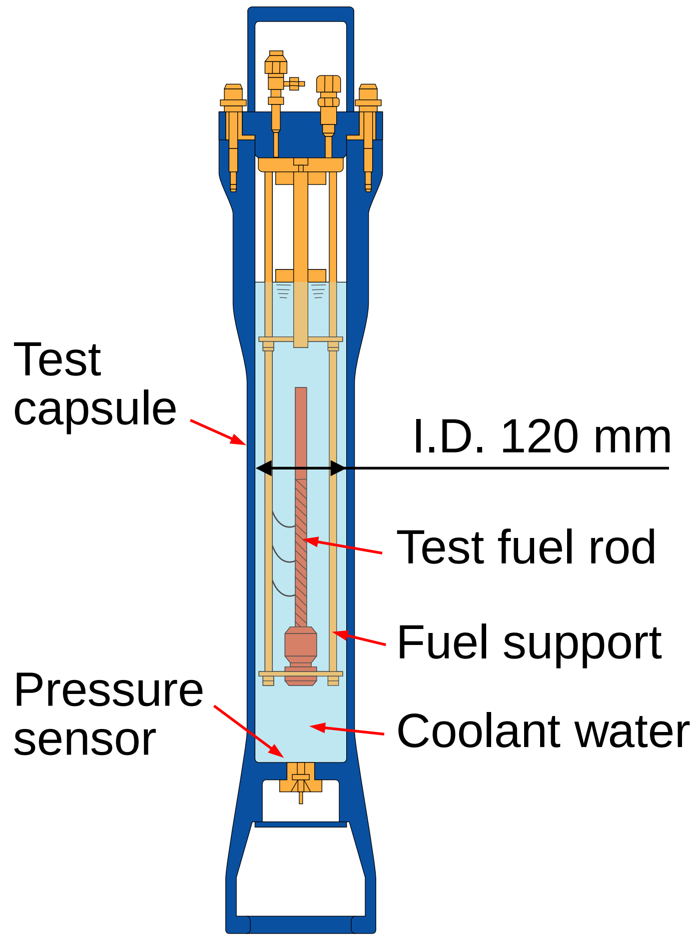
Core horizontal cross section

Reactor core	
Effective height:	~38 cm
Equivalent diameter:	~60 cm
Moderator:	ZrH, H_2O
Driver fuel rod	
Fuel materials:	$U\text{-}ZrH_{1.6}$
Enrichment:	20%
Cladding:	SUS 304
Dimensions:	3.75cmD x 60cmL
Number of rods:	157

NSRR experiment with fresh fuel

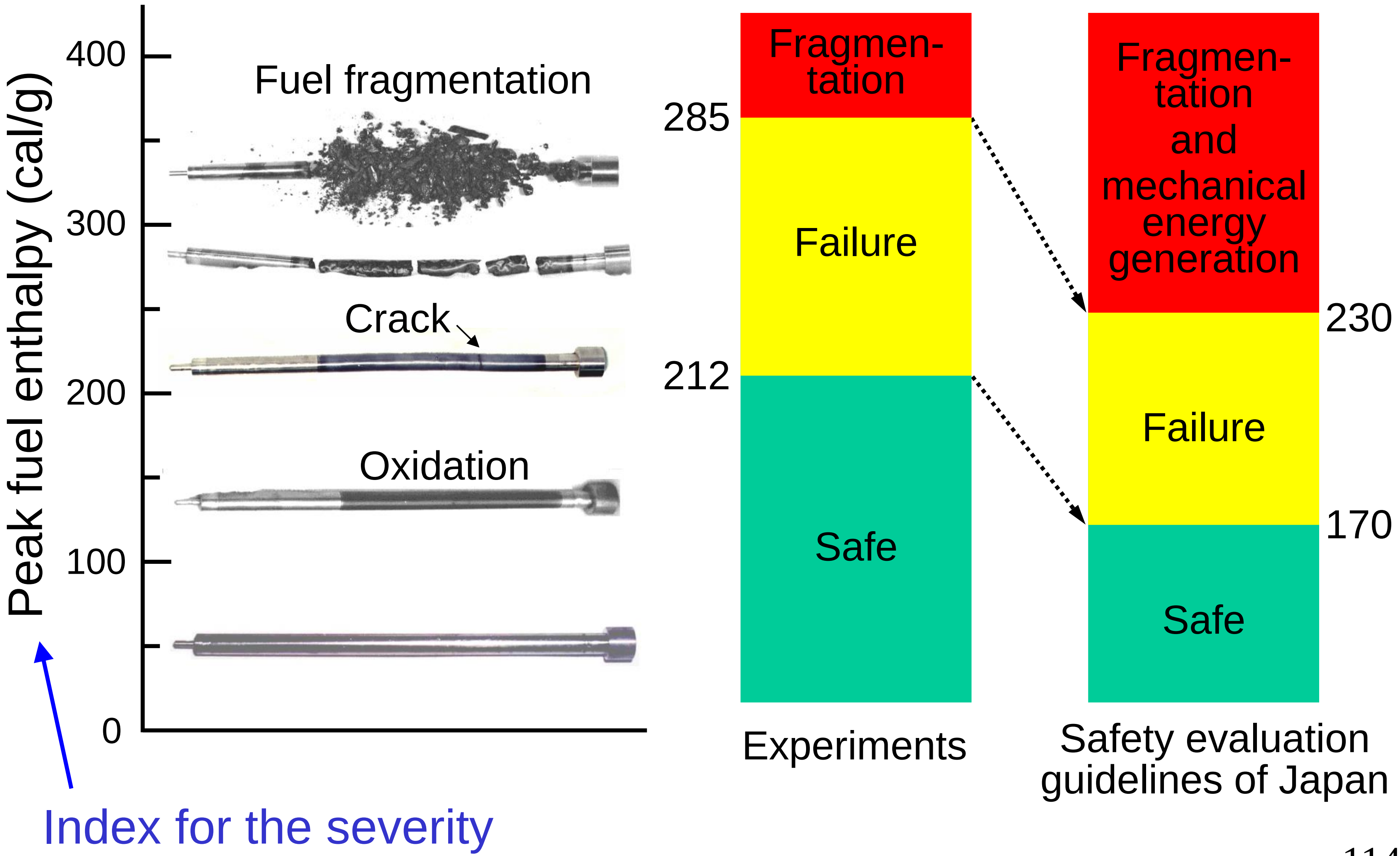


NSRR standard test rod



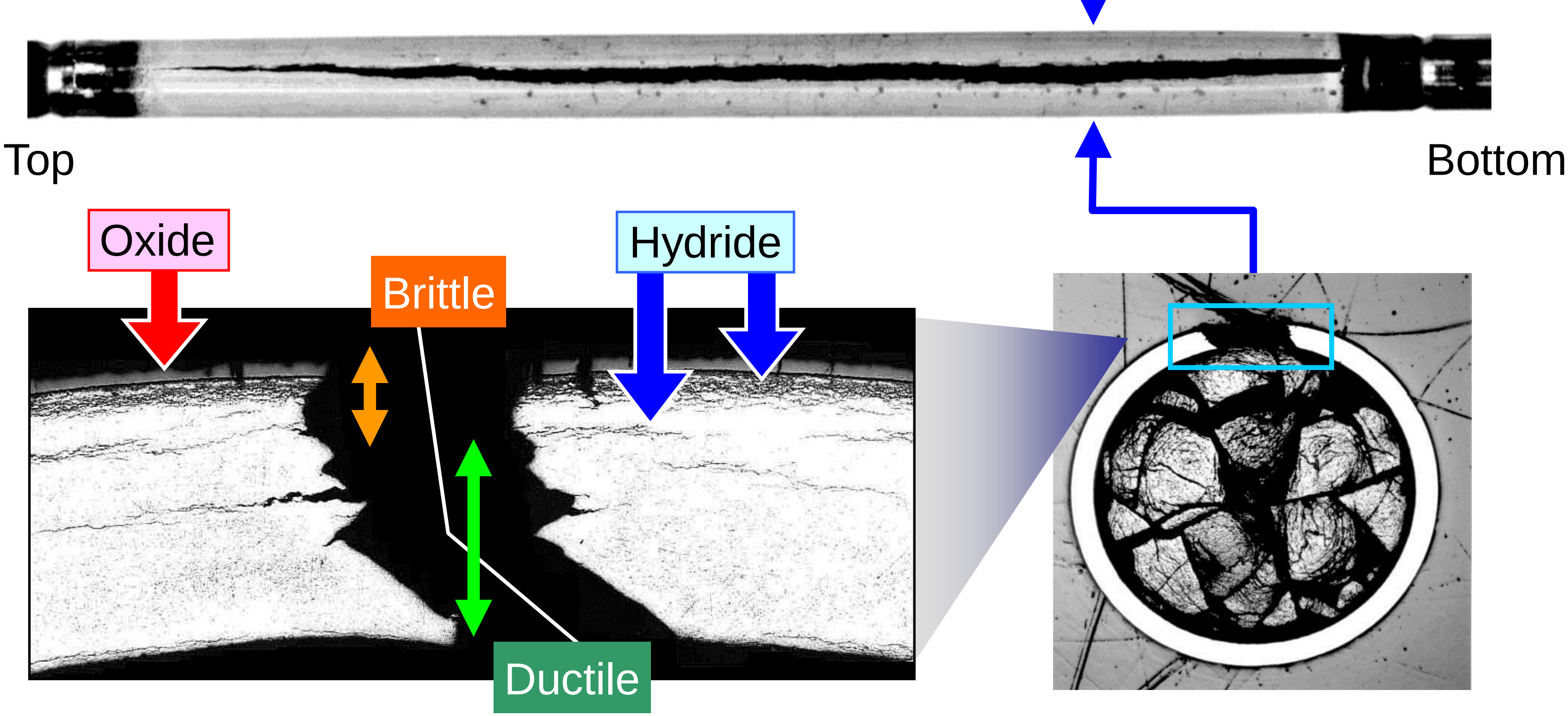
Test capsule

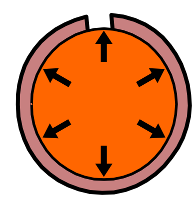
Results from fresh fuel experiments



PCMI failure of high burnup fuel rod

48 GWd/t PWR fuel rod
Failure at 60 cal/g

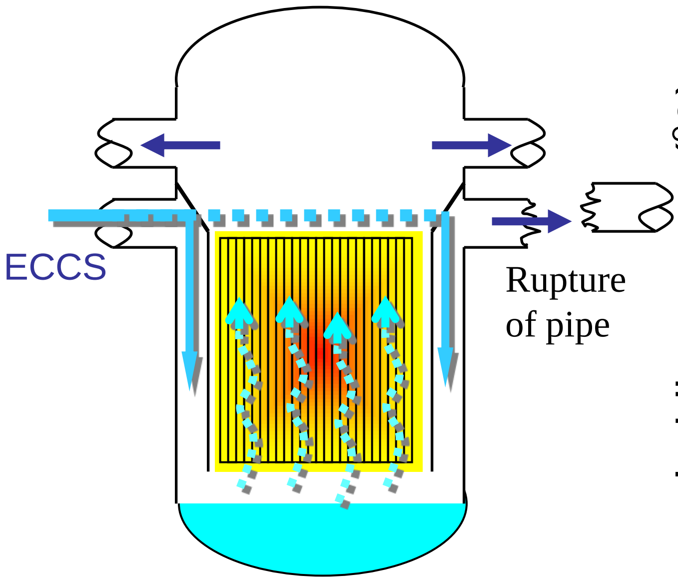




Hydrogen-embrittled cladding failed due to the pellet thermal expansion.

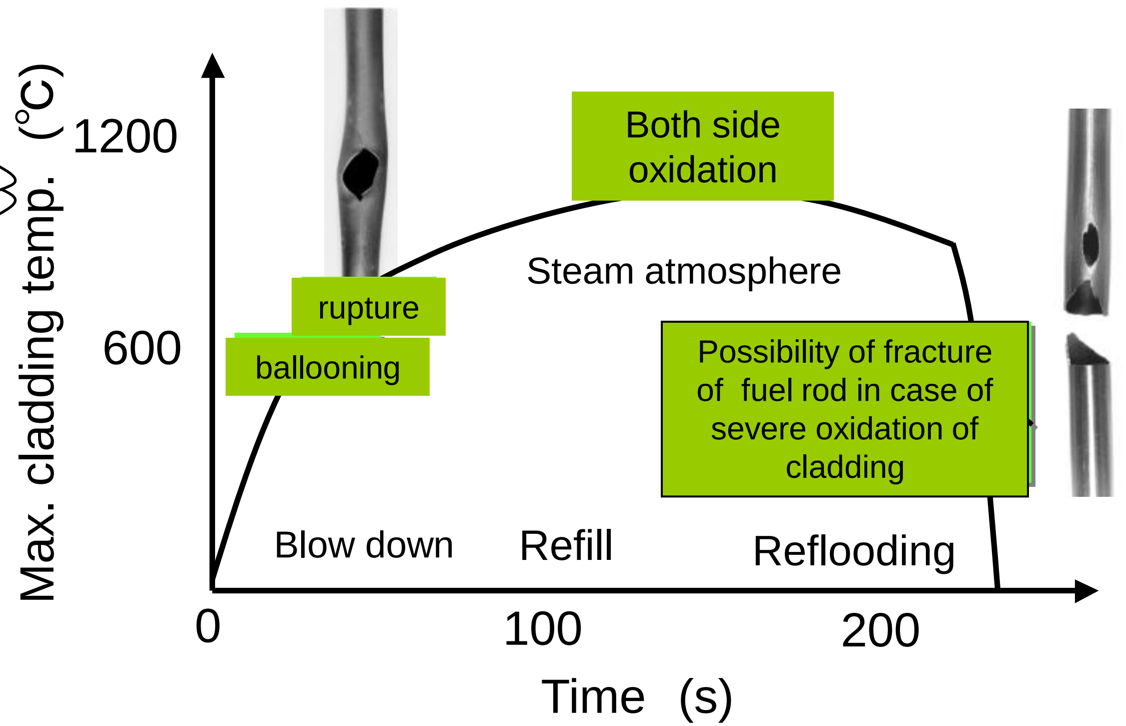
PCMI (pellet-cladding mechanical interaction) failure

Fuel behavior under LOCA and safety criteria



Rupture of primary coolant pipe ⇨ Loss of coolant ⇨ Increase of fuel temp.

⇨ Start of ECCS ⇨ Core cooling



LOCA criteria（ECCS evaluation standards）

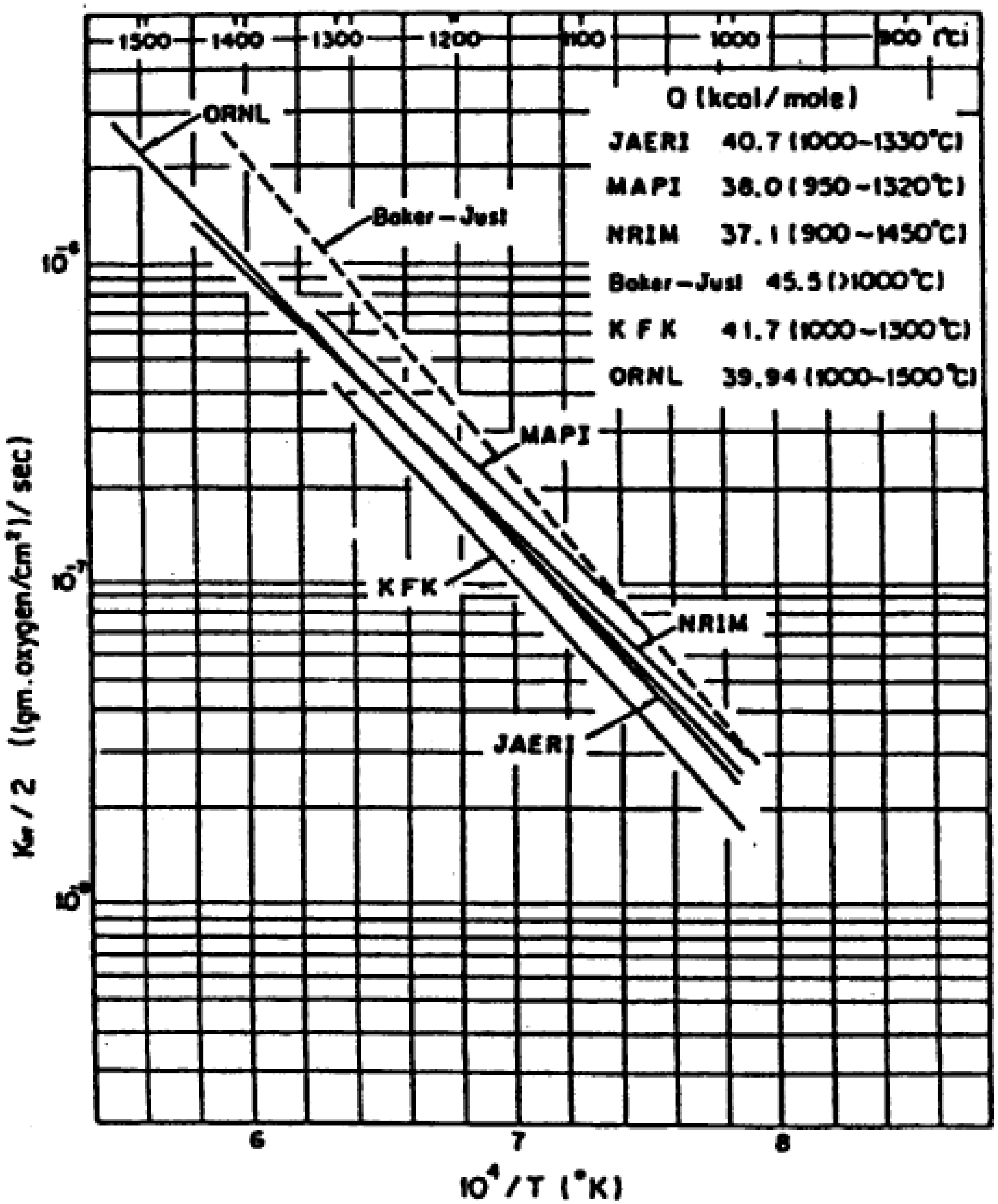
Evaluation of function and performance of ECCS to assure the coolability of reactor core

Standards for cladding embrittlement

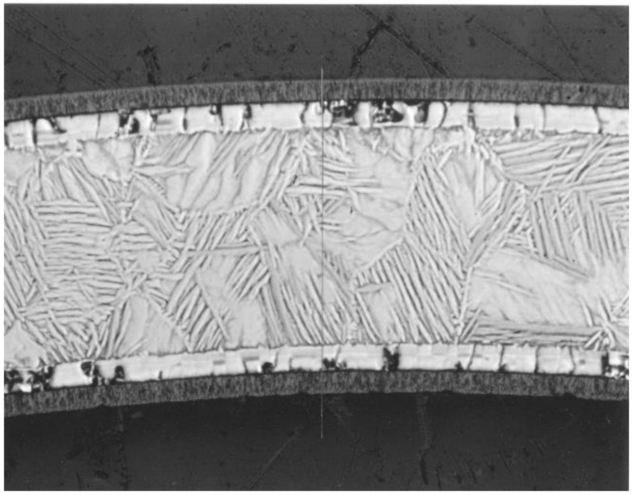
- Cladding temperature less than 1200 C
- Equivalent Cladding Reacted(ECR) less than 15% of cladding thickness

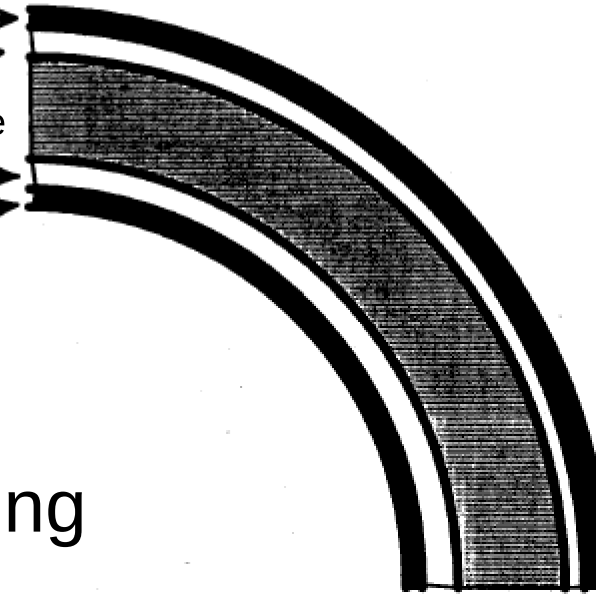
Equivalent Cladding Reacted（ECR）

- The oxidation amount of fuel cladding calculated as stoichiometric ZrO_2 must be less than 15% of cladding thickness before the severe oxidation.
- (calculated with Baker-Just equation for zircaloy oxidation)

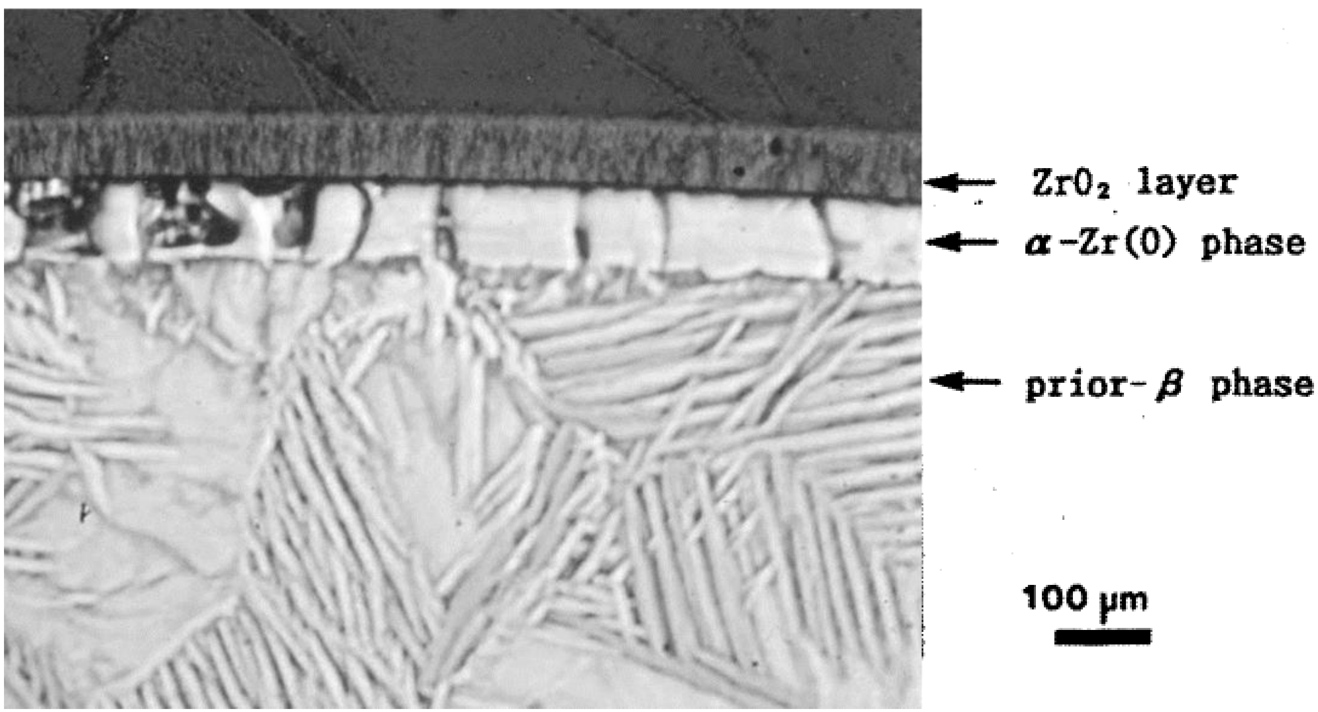


Temperature dependence of Zircaloy- steam reaction rate coefficient





Cross section of oxidized cladding



Cross section of zy-4 oxidized 3min at 1200C