Introduction to Nuclear Reactors, Fuels, and Materials

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Nanotechnology in Nuclear Fuels and Materials R&D
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Intro to Nuclear Goals

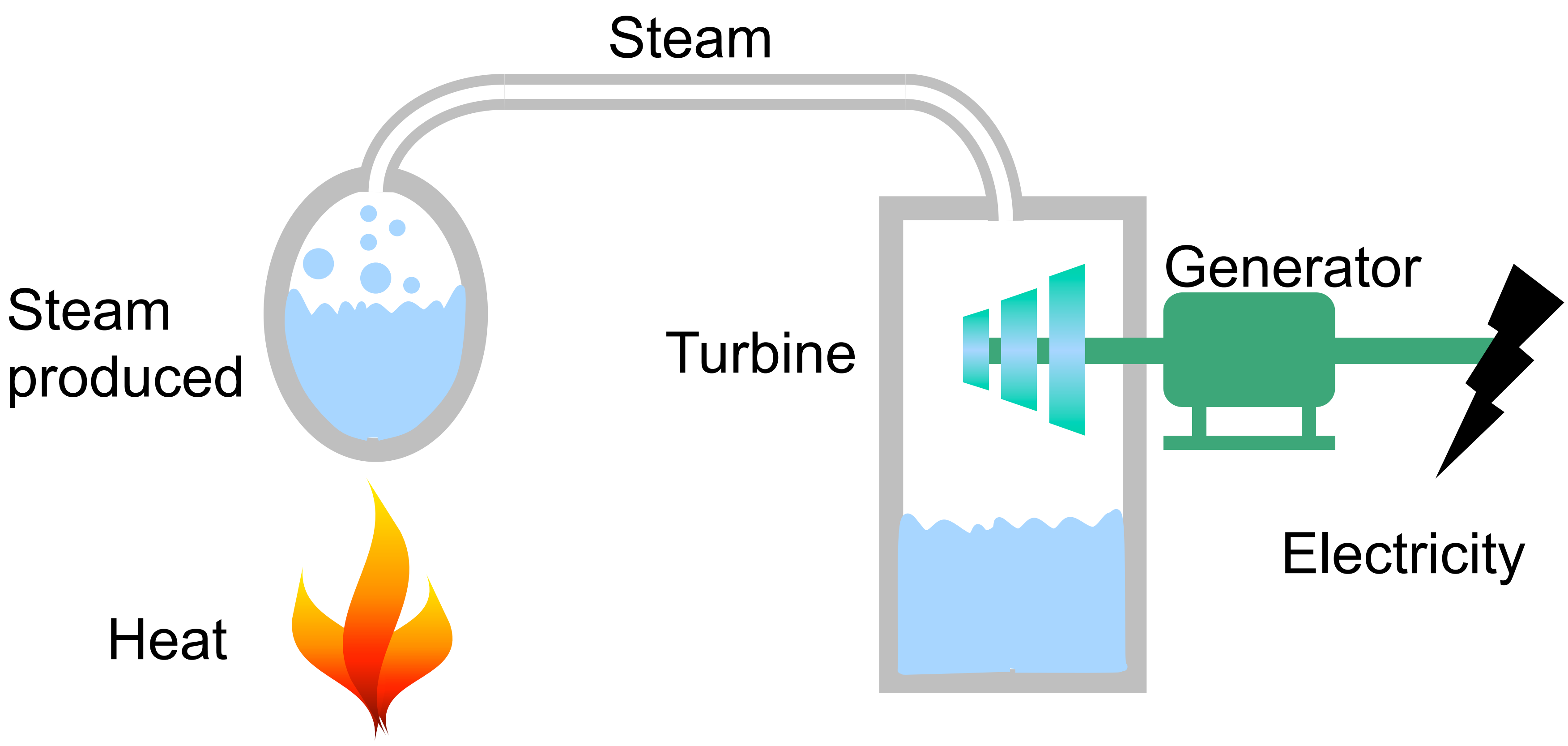
What will be covered:

- What goes in a nuclear power plant
- What goes on in a nuclear power plant
- Challenges in nuclear fuels and materials

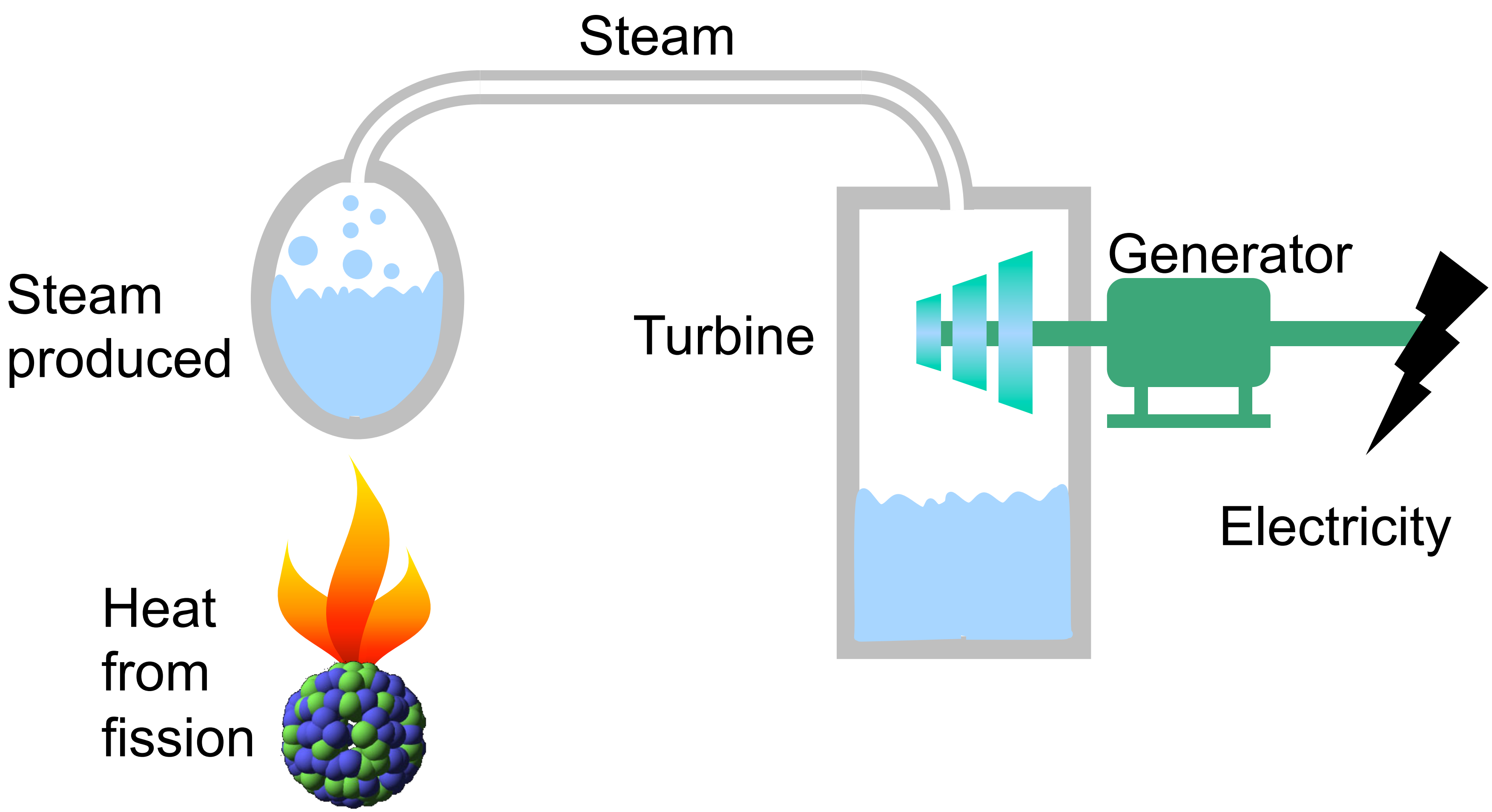
Key lessons:

- Fuels and materials change during irradiation
- Design changes must consider neutronic, thermal, mechanical, chemical properties (a change in 1 property may affect the others)

What is a power plant?



What is a nuclear power plant?

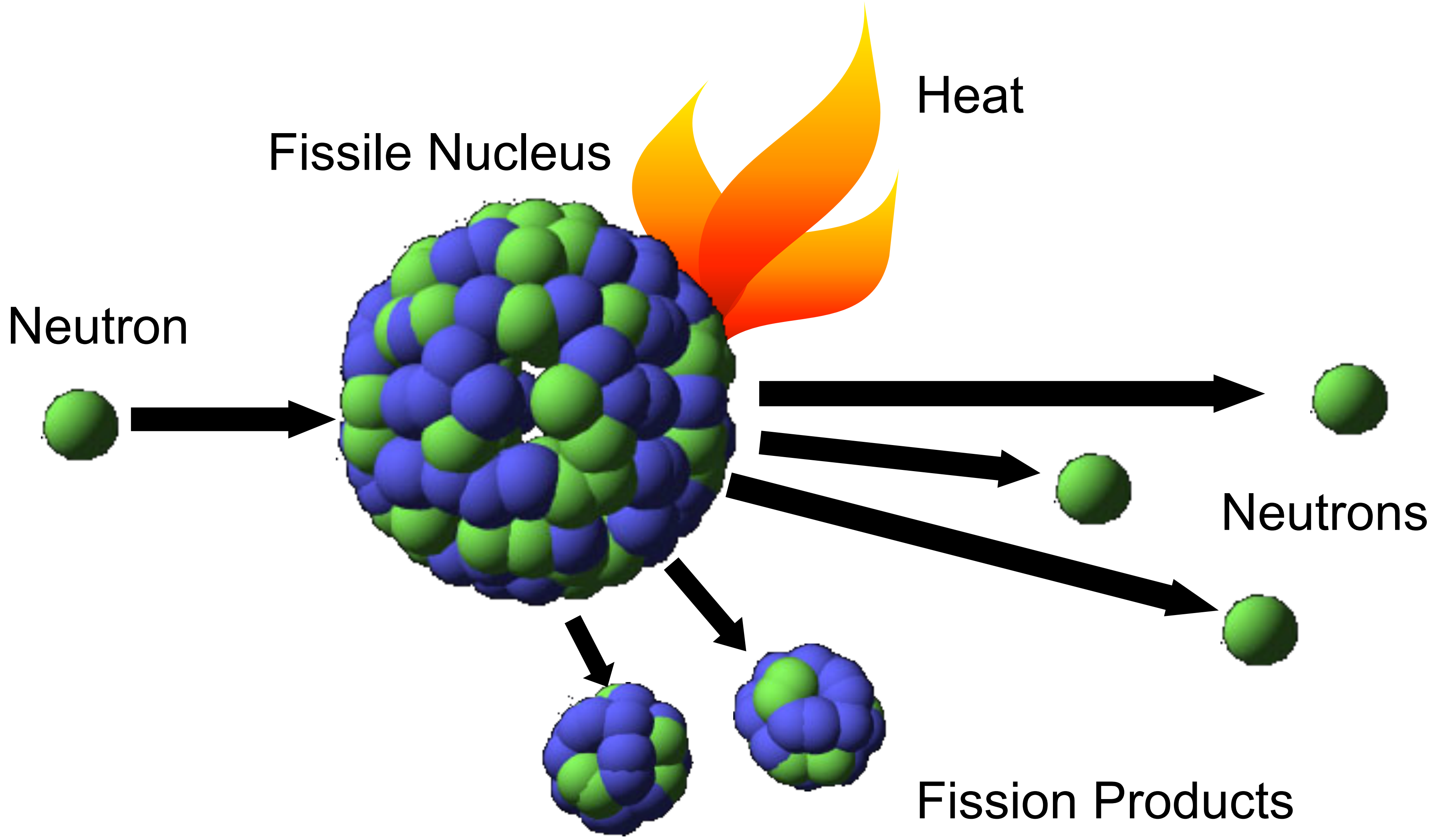


What goes in a nuclear reactor?

Fuel	Coolant	Moderator	Materials	Control
ceramic metallic particle dispersion nitride	water sodium gas liquid metal	water graphite	steel zircaloy graphite advanced alloys	fixed (control rods) soluble boron Ag-In-Cd

- Continual effort to improve nuclear power:
  - more efficient (better fuel utilization, better heat removal)
  - more economical (construction and operations)
  - safer (better accident/off-normal response)

Fission



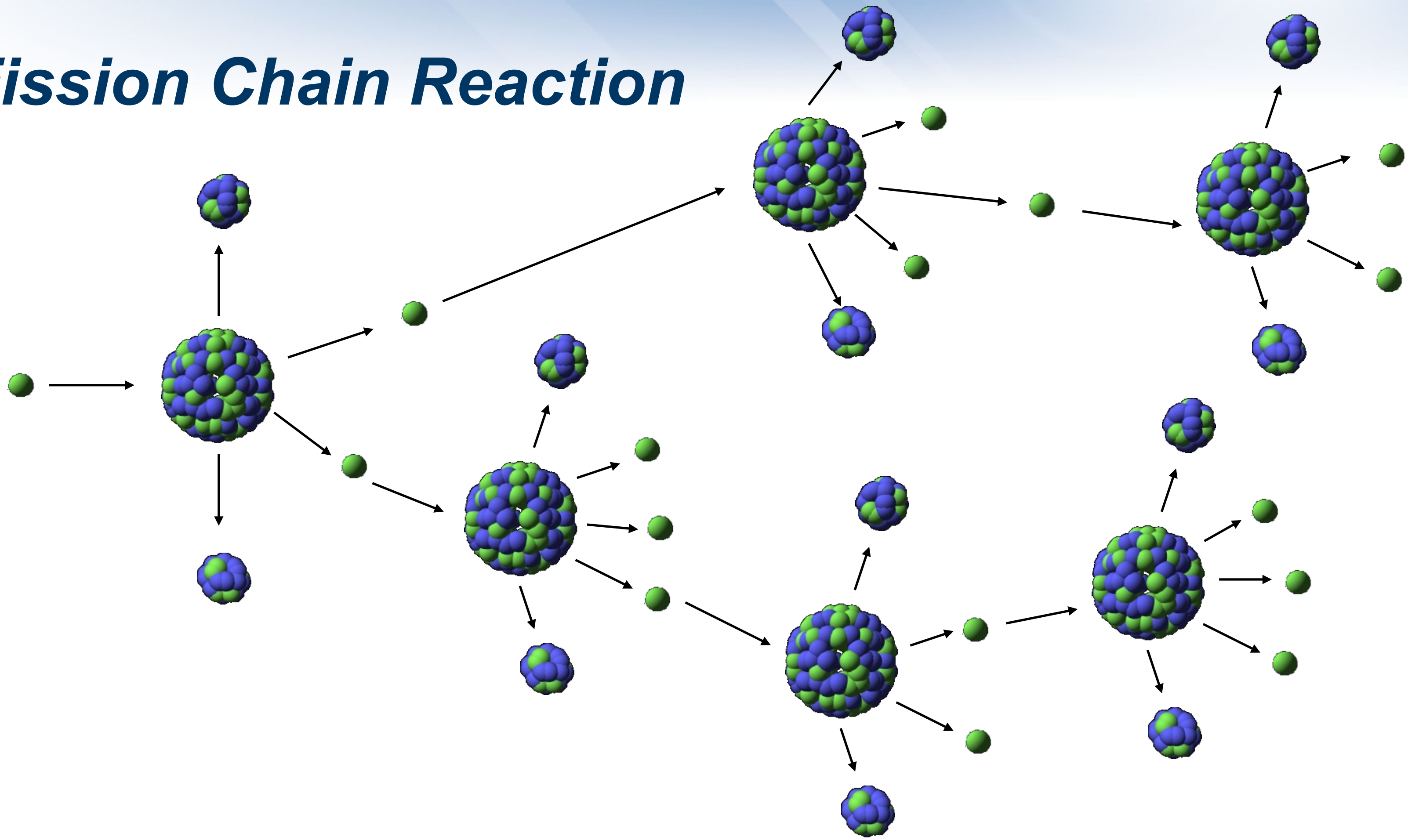
Fission

$$n + {}^{235}U \xrightarrow{fission} 2\,FP + 2.5\,n + energy$$

n = neutron
FP = fission products

- Fuel is consumed during fission
- Fuel is changing composition during irradiation
- Neutrons are consumed and created during fission

Fission Chain Reaction



- Neutrons are needed to sustain a fission chain reaction
- Reactor is critical when neutrons consumed = neutrons created
- Every neutron does NOT lead to fission

Cross-sections

- Cross-section describes the probability of a specific interaction occurring
- Interactions include
  - fission
  - scattering
  - absorption
- Cross-sections depend on
  - isotopes and particles involved
  - energy of the particles (neutrons)

- Neutrons created during fission have high energy
- Cross-section (probability) for fission in ^{235}U is increased for low-energy neutrons

^{235}U Fission Cross-section

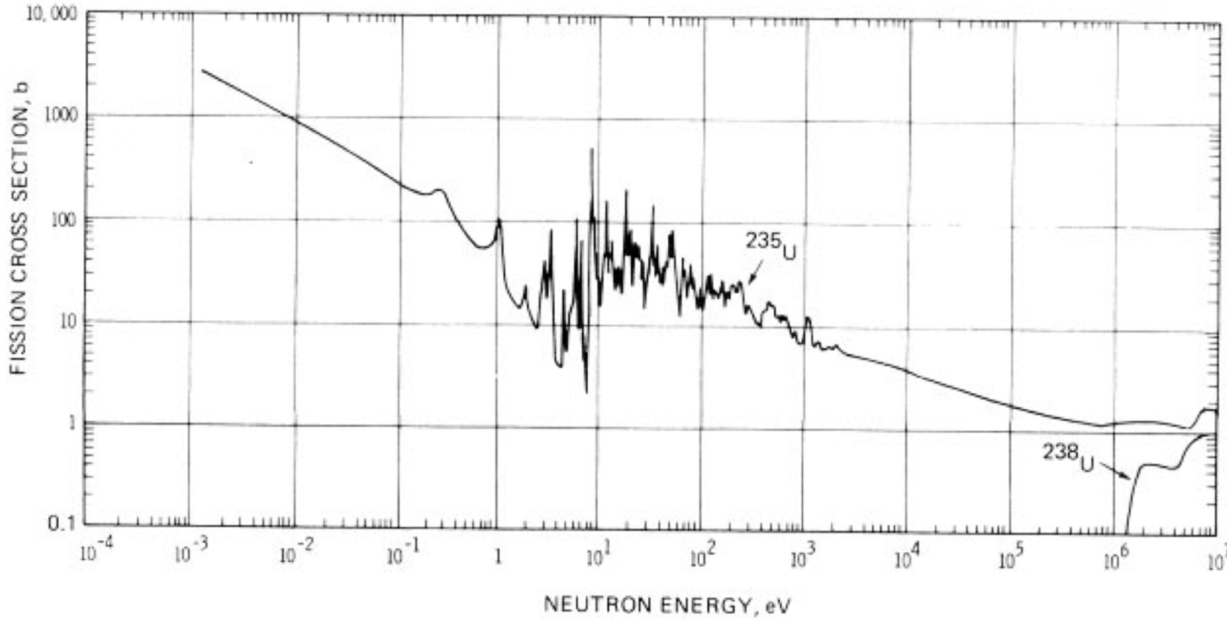


FIGURE 2-12
Microscopic fission cross section for fissile ^{235}U and fissionable ^{238}U. (Data from Hughes/BNL-325, 1955.)

- Low energy neutrons are "thermal"
- High energy neutrons are "fast"
- Fission neutrons are born "fast"
- Neutrons slow down via scattering collisions

^{235}U Fission Cross-section

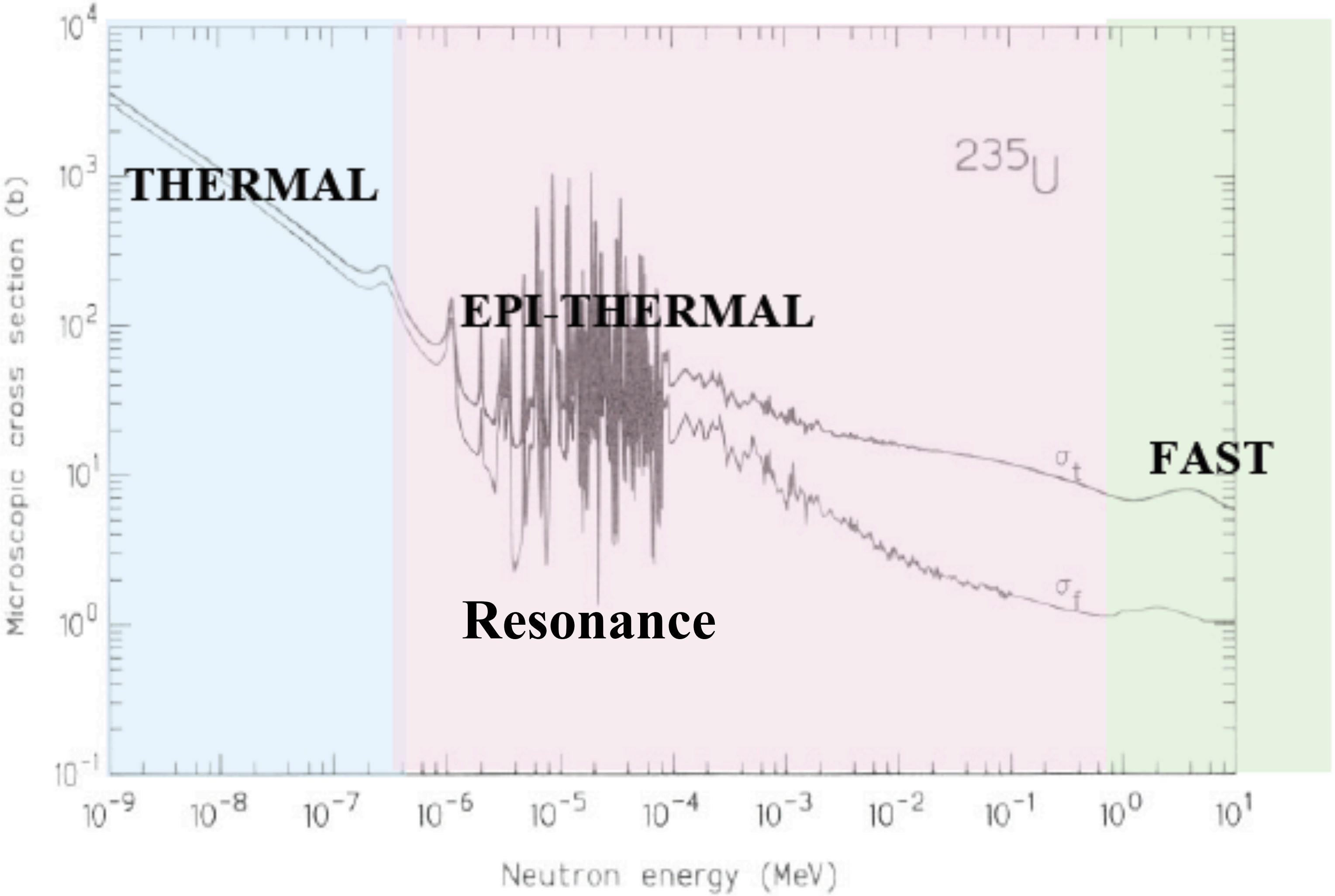


Figure 7.10. The total and fission cross section for ^{235}U based on NJOY-processed ENDF/B (version V) data.

What can fission?

- Fissile isotopes: fission is possible with neutrons of any energy
- Fissionable isotopes: fission is possible with high energy neutrons (E>1 MeV)

- Fissile: ^{233}U, ^{235}U, ^{239}Pu, ^{241}Pu
  - only ^{235}U occurs naturally, 0.7% of natural U
  - $$^{238}U + n \rightarrow {}^{239}U \xrightarrow{\beta^-} {}^{239}Pu$$
  - $$^{232}Th + n \rightarrow {}^{233}Th \xrightarrow{\beta^-} {}^{233}Pa \xrightarrow{\beta^-} {}^{233}U$$

- Fissionable: fissile + ^{232}Th, ^{238}U, ^{240}Pu

- Transmutation

^{238}U Cross-sections

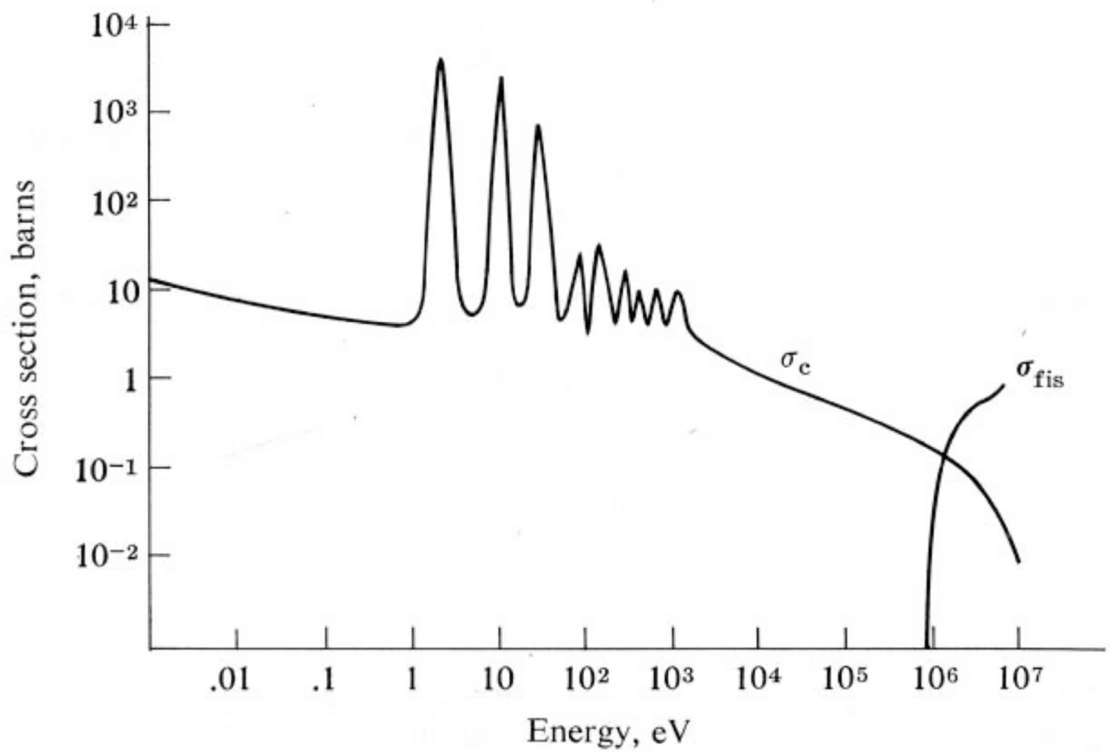


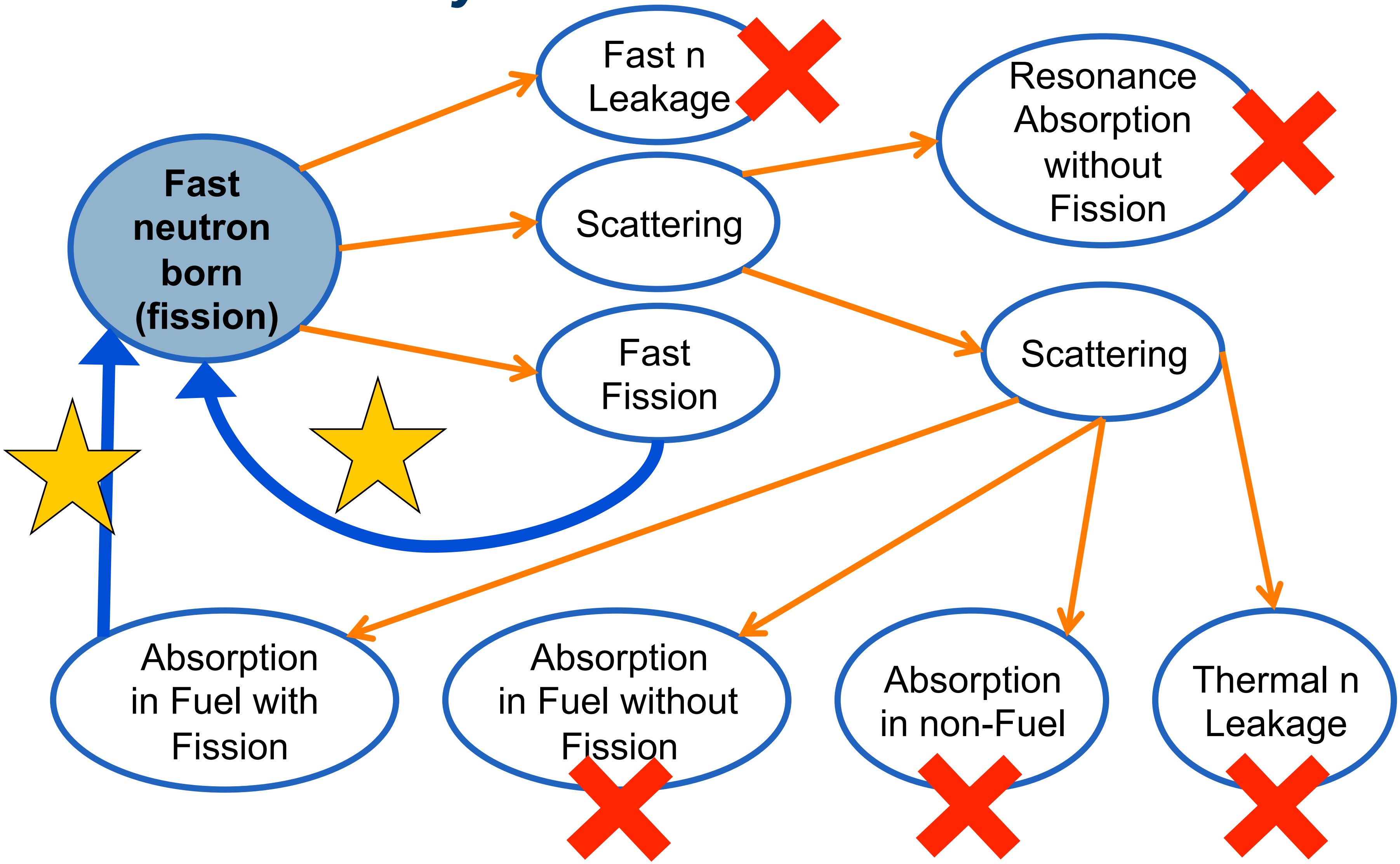
FIG. 9.2 *Absorption (capture and fission) cross sections for ^{238}U.*

- ^{238}U only fissions with high energy ("fast") neutrons
- 99.3% of natural uranium is ^{238}U
- Reactors designed to use "fast" neutrons can fission natural U (abundant fuel supply)

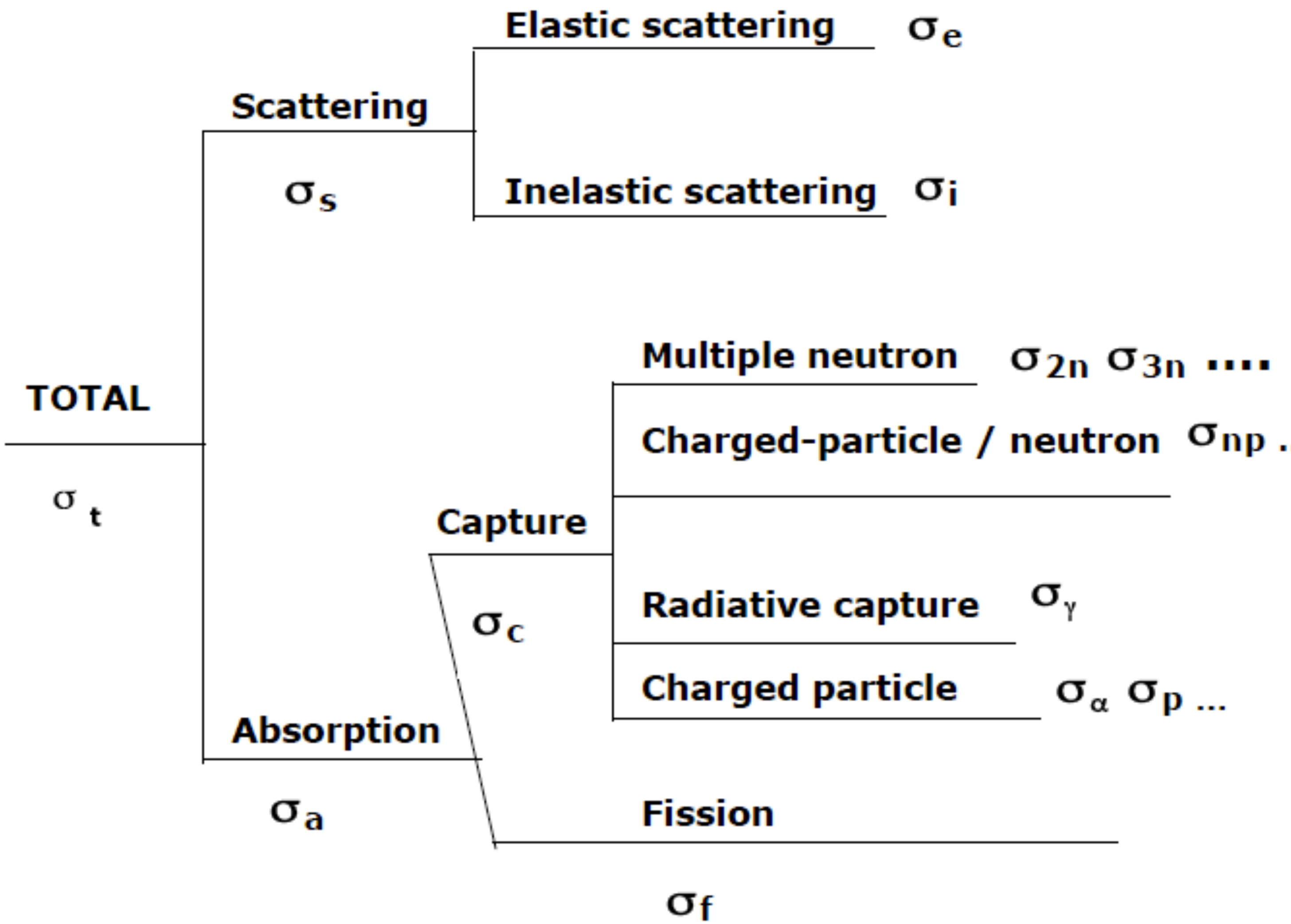
Neutron Life Cycle

- Every neutron interaction does NOT cause fission
- Interactions include:
  - no interaction (neutron may escape from the system)
  - scattering (neutron loses energy = moderation)
  - absorption in non-fissile material
  - absorption in fissile material without fission
  - absorption in fissile material with fission
- Design choices optimize probability for fission
  - materials, geometry, energy spectrum
- Use absorbing materials to control chain reaction

Neutron Life Cycle



The interaction of radiation and matter is a statistical process, described in terms of probability



Implications for Reactor Design

- Choose fuels with high probability for fission
  - fast neutron energy
  - thermal neutron energy
- Select other materials with low probability for absorption
  - coolant
  - moderator
  - fuel pin cladding
  - core structural materials
- Select control materials with high probability for absorption
  - soluble poison (boron)
  - control rods (Ag-In-Cd, B_4C)
- Design geometry of the reactor to minimize leakage
  - add reflectors (scattering material) to keep neutrons in the reactor

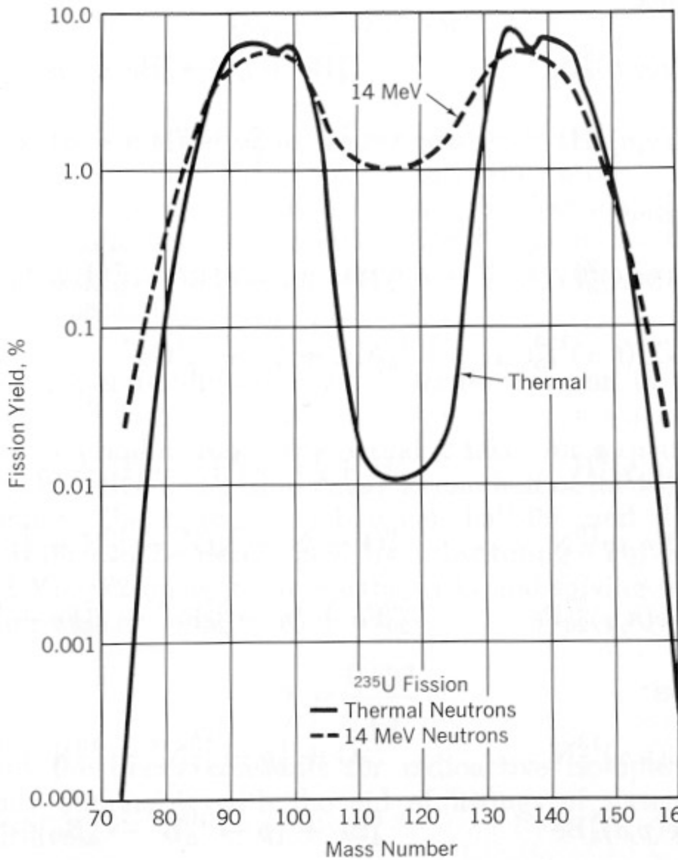
Fission

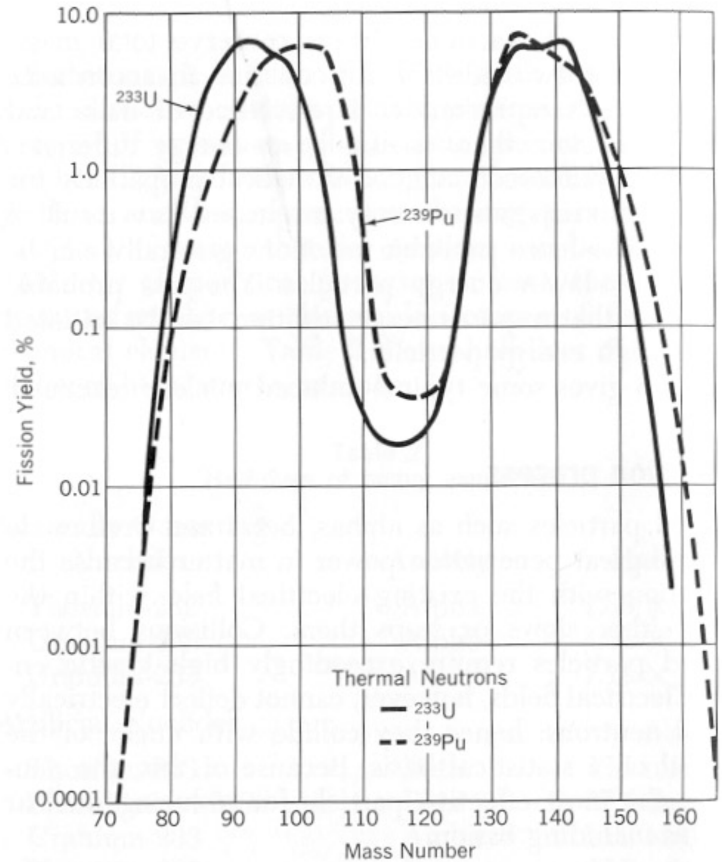
$$n + {}^{235}U \xrightarrow{fission} 2\,FP + 2.5\,n + energy$$

n = neutron
FP = fission products

- Fuel is consumed during fission
- Fuel is changing composition during irradiation
- Neutrons are consumed and created during fission
- Some neutrons are delayed, produced some time after fission
- Radioactive decay produces heat long after chain reaction is stopped

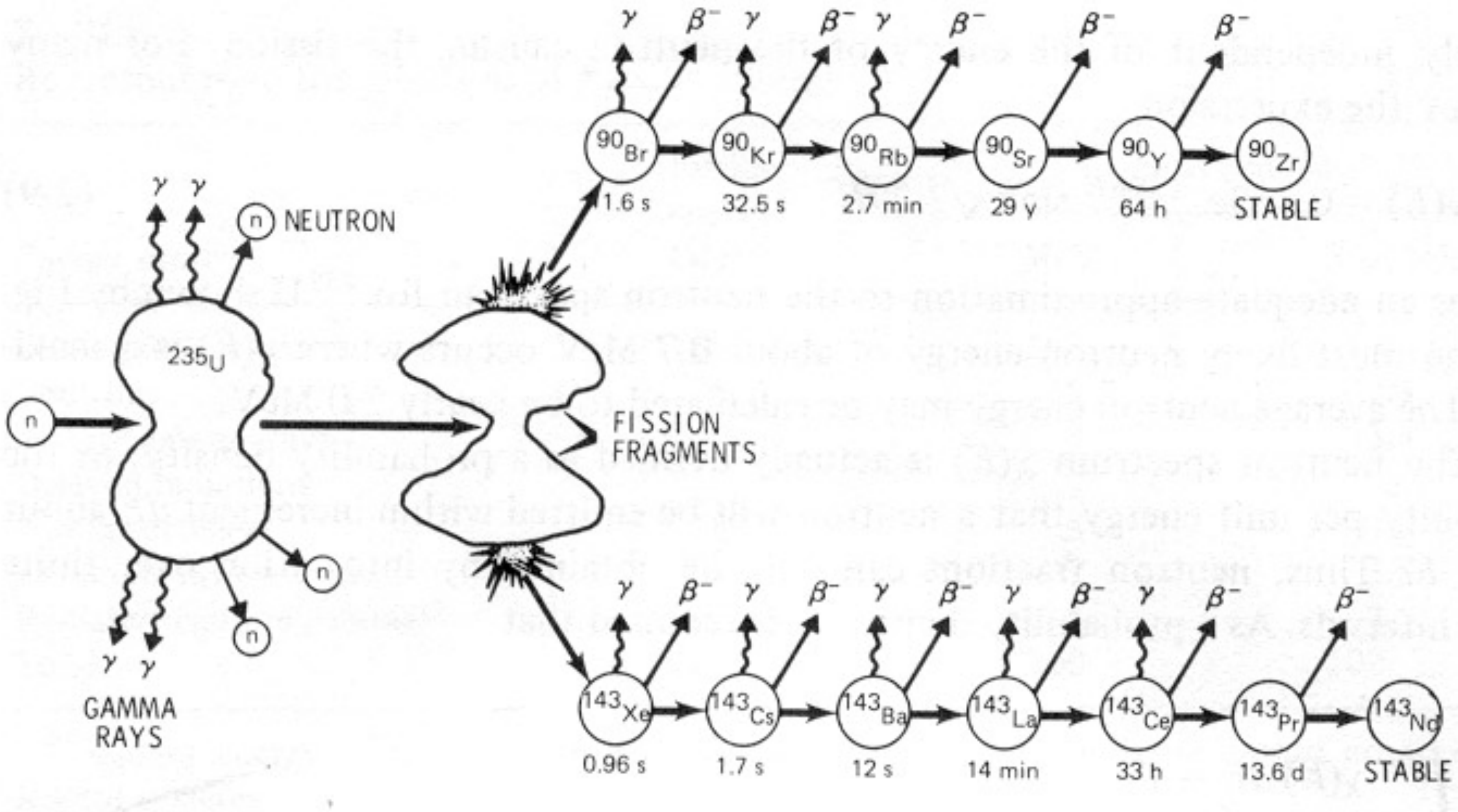
Mass Distribution of the Fission Products





- Fission products may be: solid (75%), gaseous (25%), radioactive, chemically unstable
- Fission products born in fuel matrix, may migrate
- Fission products carry a lot of energy

Representative Fission Product Decay

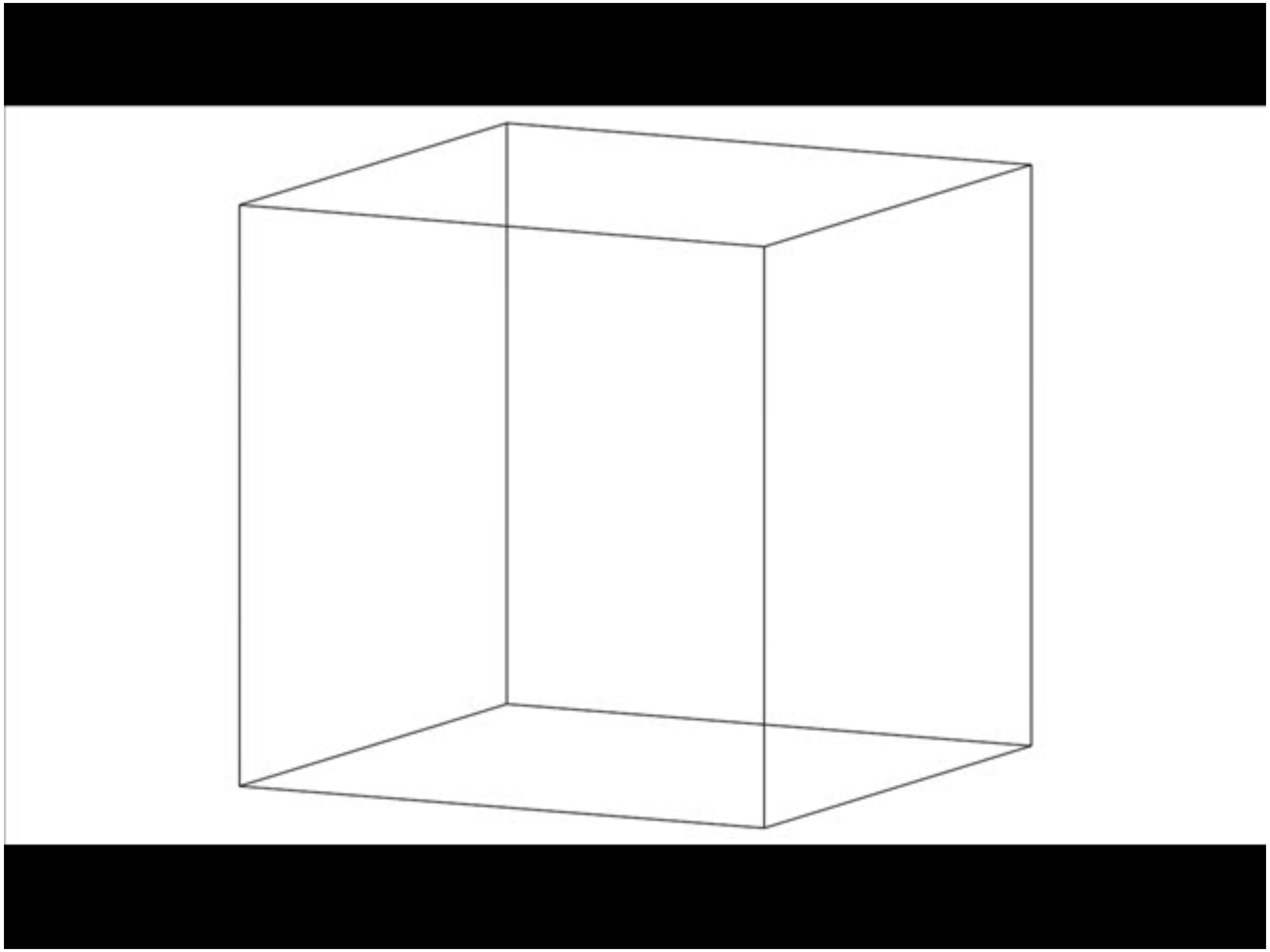


A reactor core is constantly changing

- Fuel "burns up"
  - fissile atoms are replaced by 2 fission product atoms
  - neutrons are produced

- Fission products and neutrons are energetic
  - collisions with other atoms damages other materials
  - damage cascades
  - defects are created
  - damage measured in displacements per atom (dpa)
  - some damage is "healed" by self-annealing
  - defects may move due to diffusion or chemical or thermal gradients

Cascade



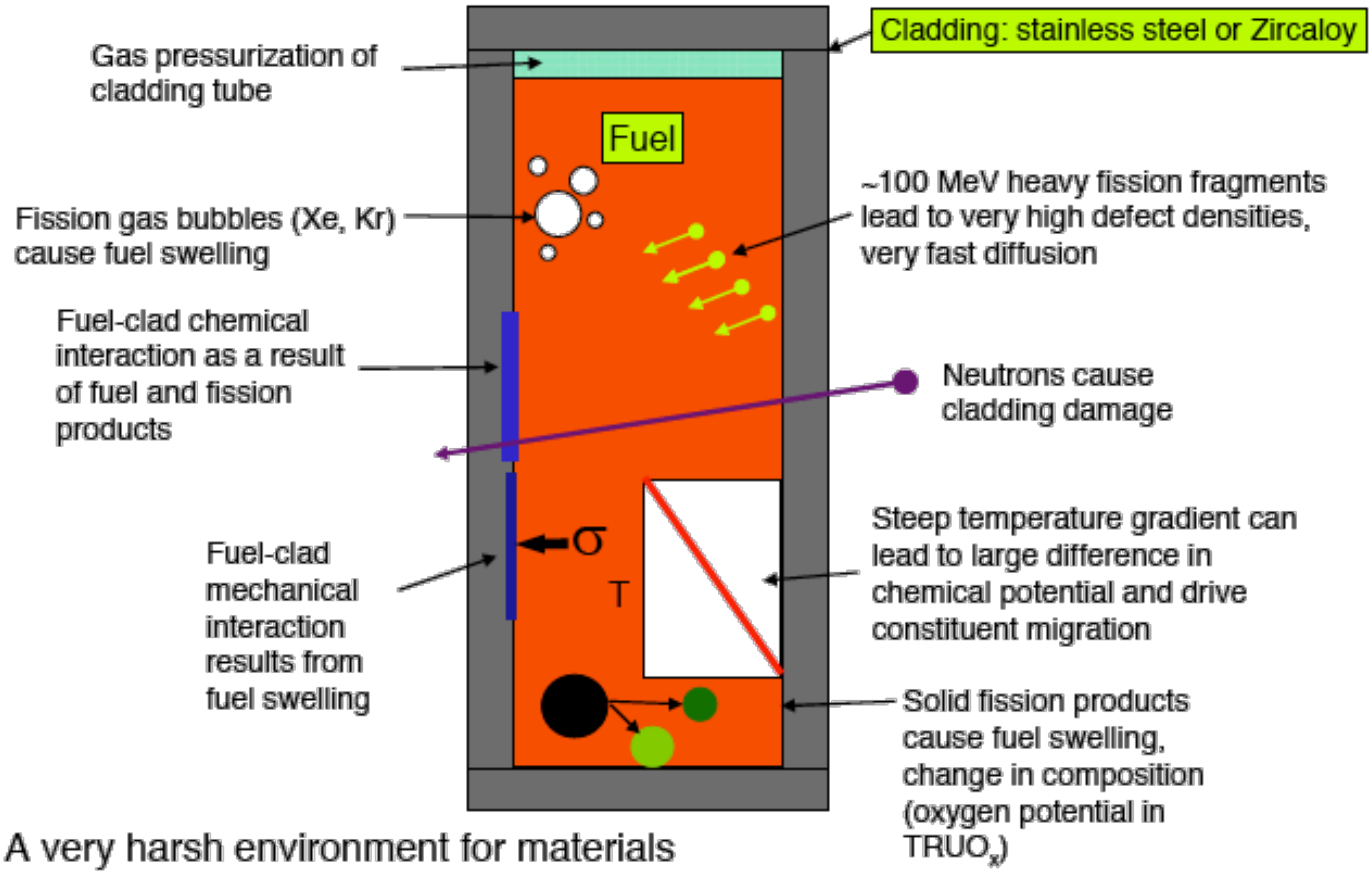
Challenges in Nuclear Fuels & Materials

- During irradiation of nuclear fuel, many complex and interrelated phenomena occur

- These phenomena degrade the nuclear fuel eventually requiring its discharge from the reactor

- In-situ instrumentation would benefit nuclear power plants and fuel development R&D efforts

Fuel Environment

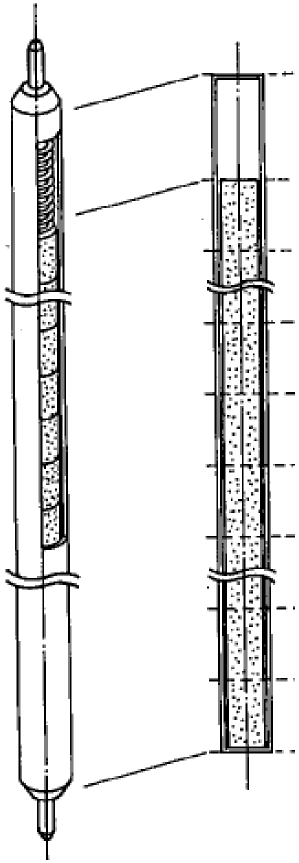


- A very harsh environment for materials

Graphic from Mitch Meyer, INL

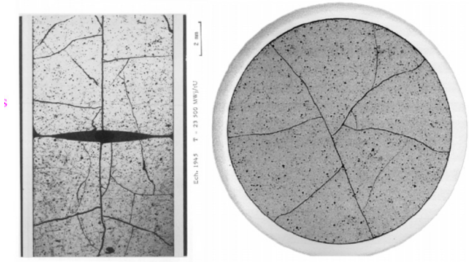
Fuel Behavior During Irradiation

At beginning of life, a fuel element is quite simple …



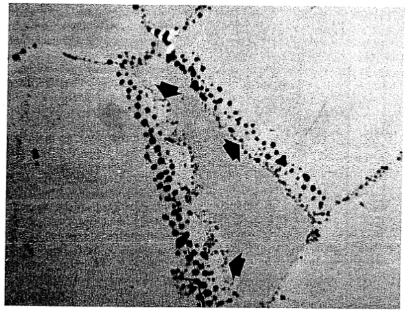
Nakajima et al, Nuc Eng Des, **148**, 41 (1994)

but irradiation brings about substantial complexity



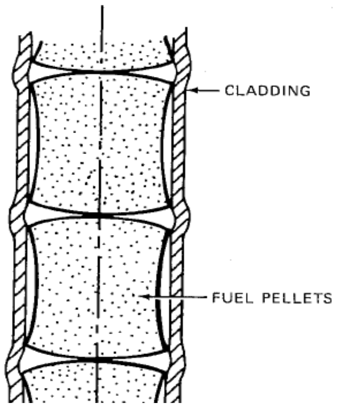
Michel et al, Eng Frac Mech, **75**, 3581 (2008)

Fuel Fracture



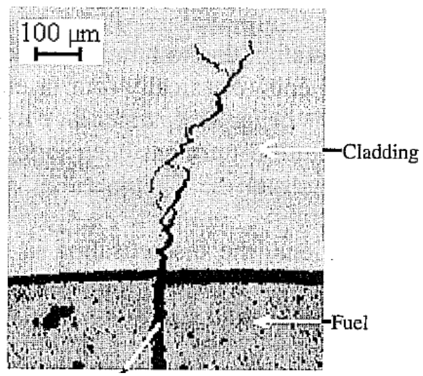
Olander, p. 323 (1978)

Fission Gas



Olander, p. 584 (1978)

Multidimensional Contact and Deformation

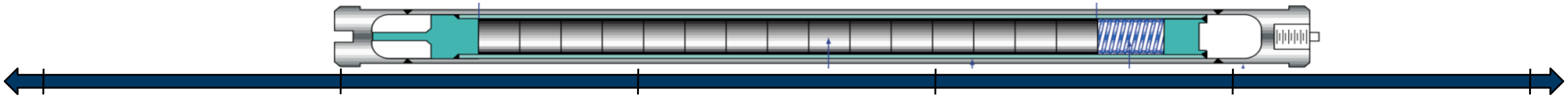


Bentejac et al, PCI Seminar (2004)

Stress Corrosion Cracking Cladding Failure

Idaho National Laboratory

Microstructure Evolution in LWR Fuel



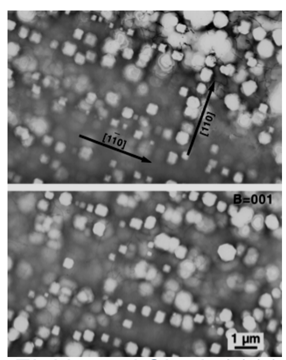
Insertion in reactor **Removal from reactor**

Early Life
- Thermal expansion
- Fracture
- Point defect and fission gas generation
- Fuel densification

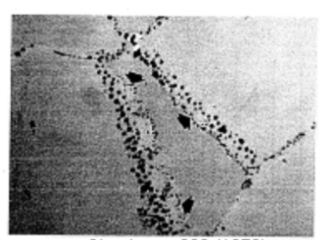
Mid Life
- Point defect diffusion
- Point defect clustering
- Fission gas segregation to GB and voids
- Bubble nucleation

Late Life
- Fission product swelling
- Bubble percolation and fission gas release
- Cladding creep
- Fuel creep

Fuel Failure
- Pellet/cladding interaction
- Cladding corrosion
- Cladding fracture

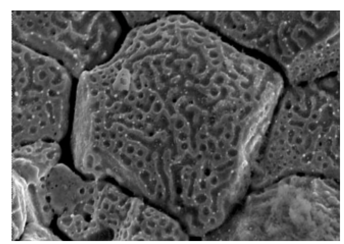


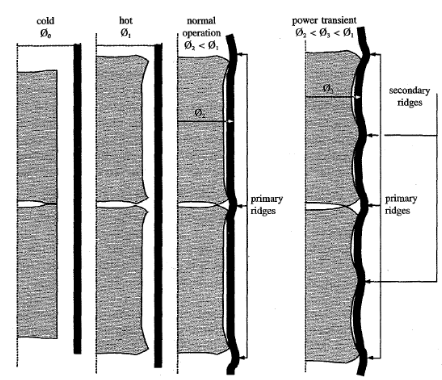
Zinkle and Singh 2000

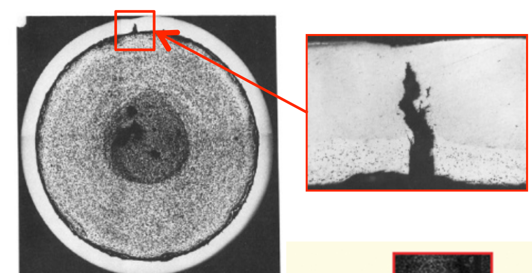


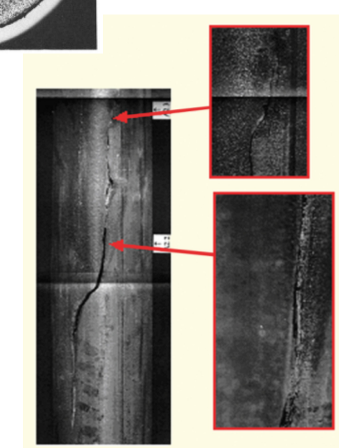
Olander, p. 323 (1978)

Brohan 2000

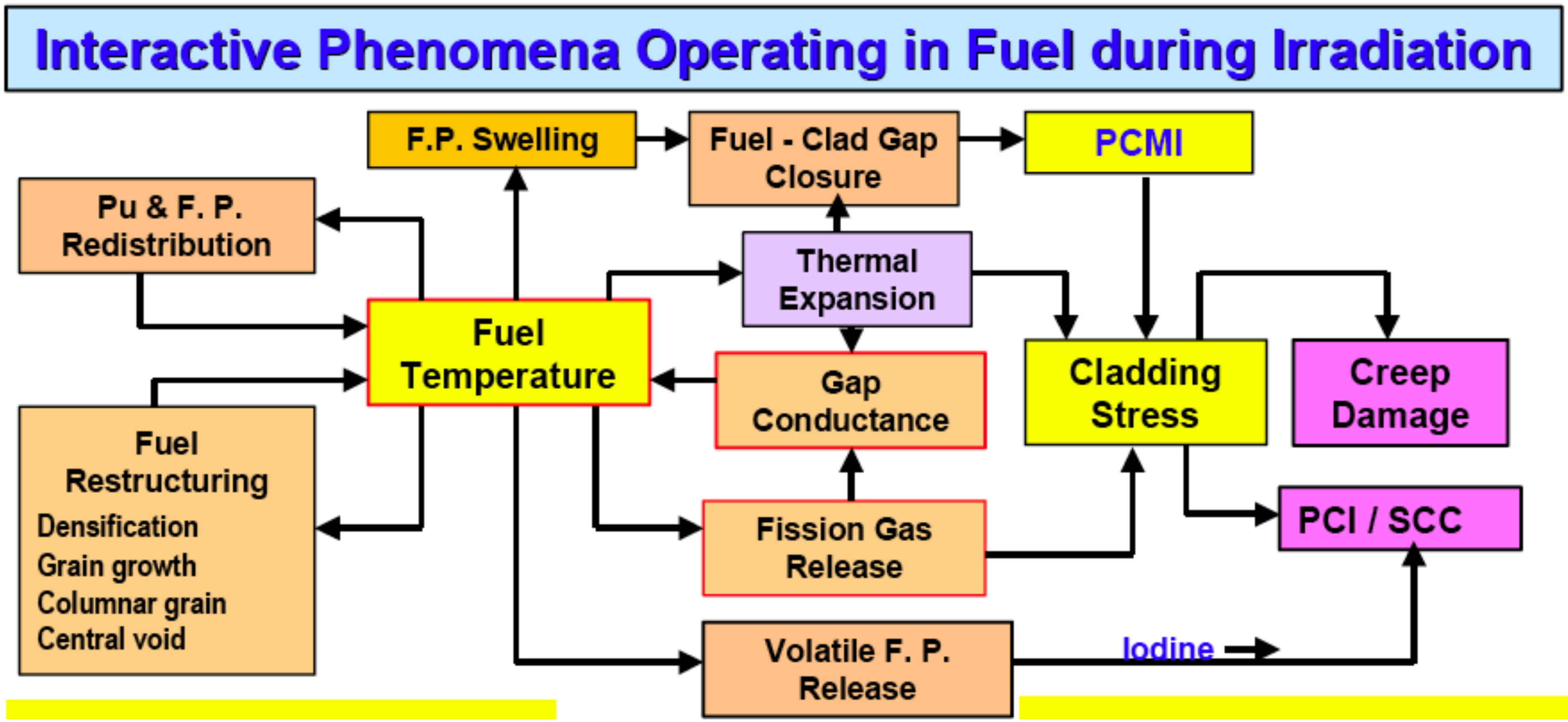






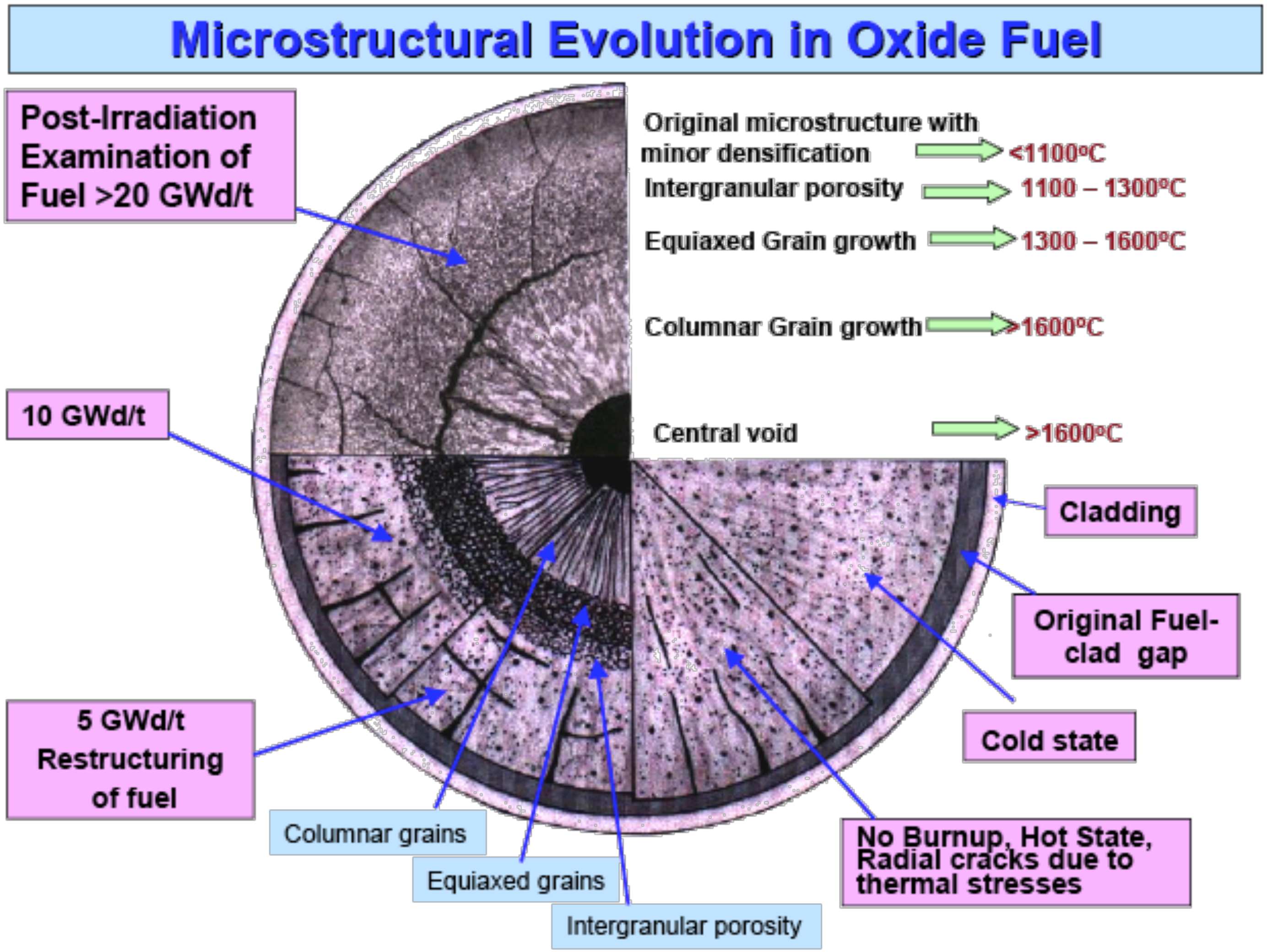


Fuel Environment



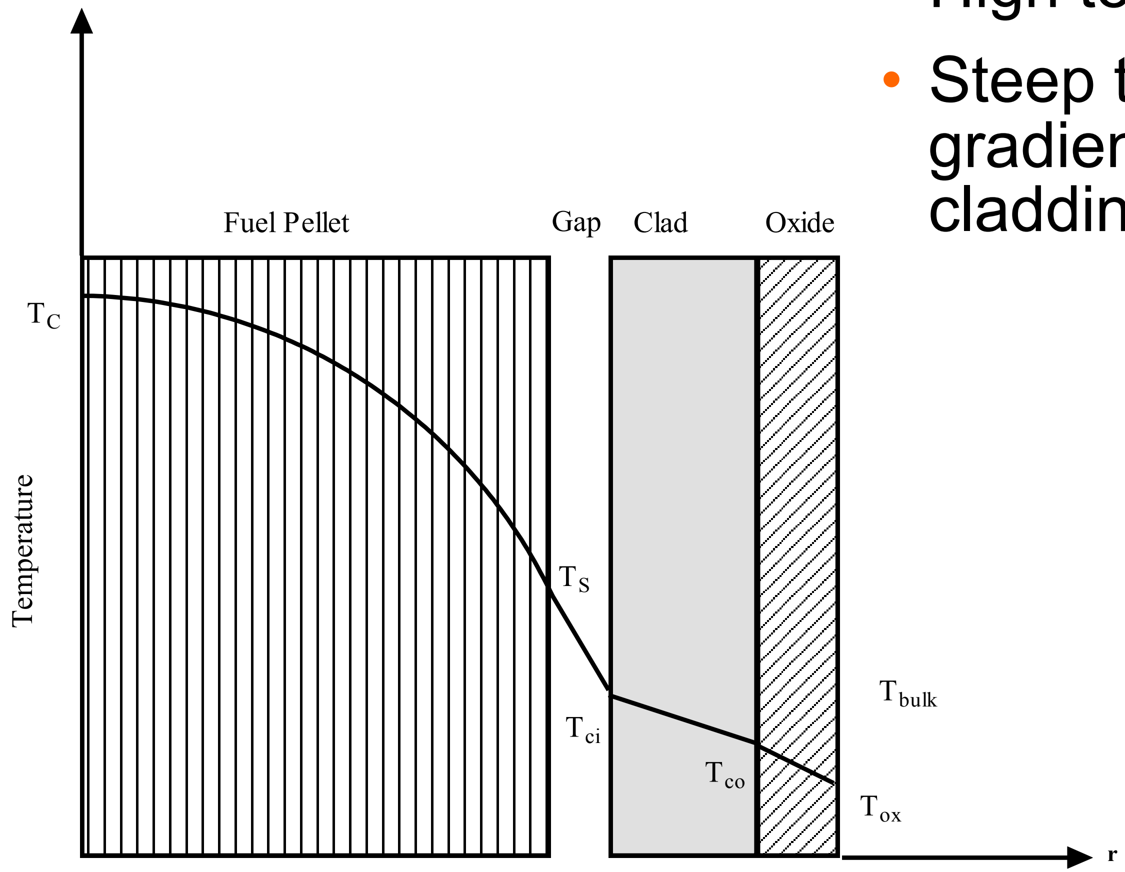
Graphic from S. Banerjee, Bhaba Atomic Research Center

Fuel Environment



Graphic from S. Banerjee, Bhaba Atomic Research Center

Typical Temperature Distribution

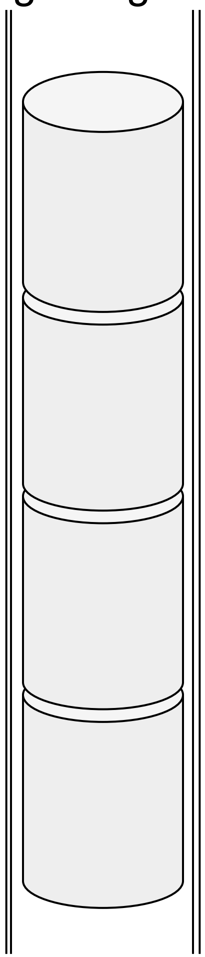


- High temperatures in fuel
- Steep temperature gradients in fuel and cladding

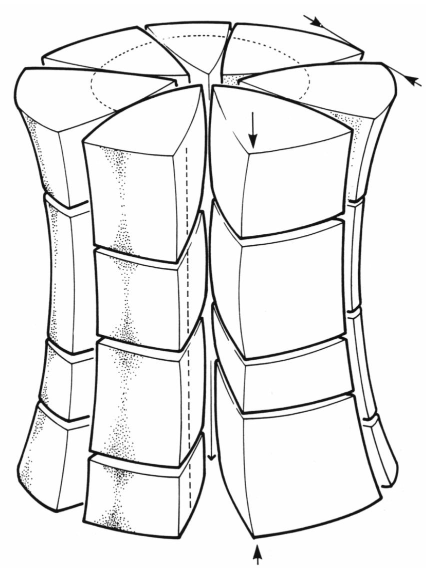
Graphic from Rosa Yang, EPRI

Fuel Response to Irradiation

Beginning of life

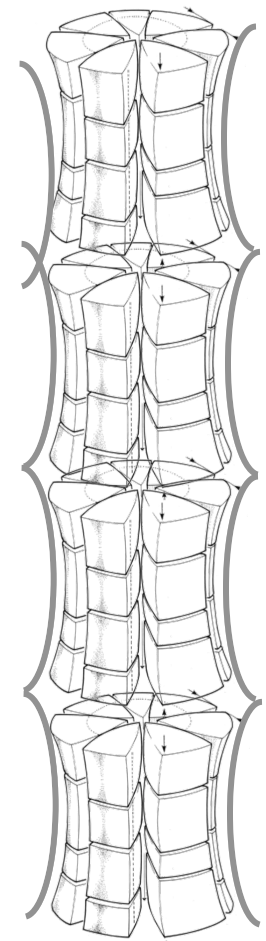


Cracking due to thermal expansion coefficient differences at varying temperatures



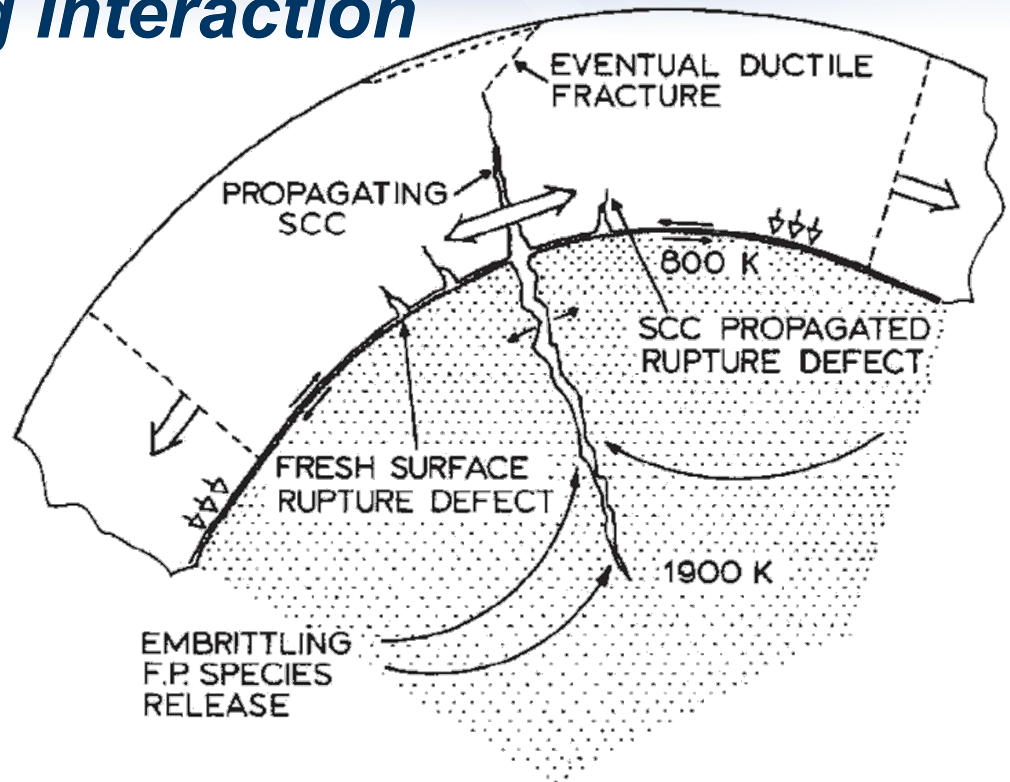
- Pellet swelling
- Gaseous fission products (Xenon, Krypton...)
- Density and porosity evolve with burn-up

After 1 cycle



Graphic from Rosa Yang, EPRI

Fuel-Cladding Interaction



- FCCI : Fuel-Cladding Chemical Interaction
- FCMI: Fuel-Cladding Mechanical Interaction
- PCI: Pellet-Cladding Interaction
- IASCC: Irradiation-Assisted Stress Corrosion Cracking

Graphic from Roberts (Structural Materials in Nuclear Power Systems, 1981)

Fuel-Cladding Interaction

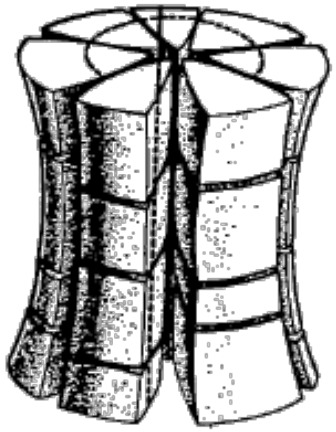
- Fuel-Cladding Mechanical Interaction (FCMI)
  - Fuel swelling and/or cladding creepdown closes gap
  - Continued swelling/creep stresses cladding
- Fuel-Cladding Chemical Interaction (FCCI)
  - Once in contact, fuel and cladding can react chemically
  - Reaction often produces a brittle layer that thins the cladding wall
  - As cladding wall thins, cladding stresses increase

Cladding Performance

- Cladding integrity assures fission product containment
  - Breach of cladding referred to as fuel “failure”
  - Failure generally precludes continued use of fuel element/bundle
- Cladding integrity degrades during irradiation
  - Temperature, pressure and neutron flux cause “creep”
    - High coolant pressure causes creepdown (LWRs)
    - High fission gas release causes outward creep (LMRs)
  - Radiation damage causes swelling (embrittlement)
  - Corrosion by coolant
  - Interaction with fuel

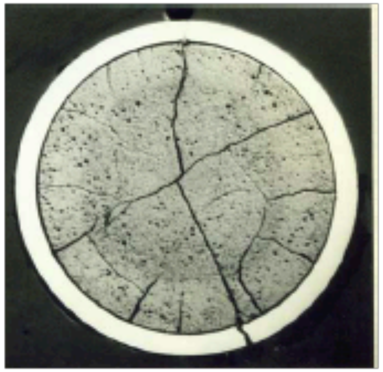
Fuel-Cladding Interaction

Pellet Clad Interaction at High Burnup

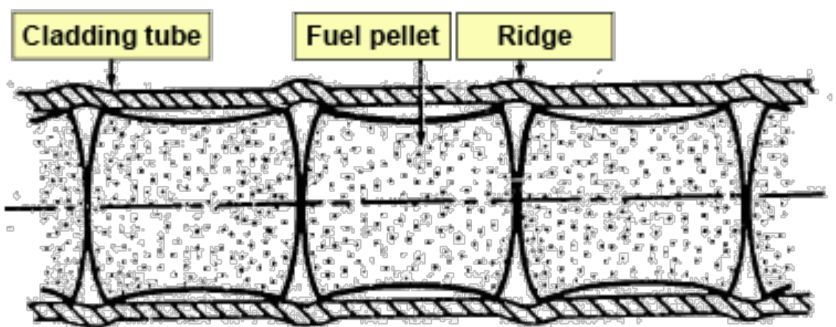


Hour-glassing of fuel pellet due to radial thermal gradient

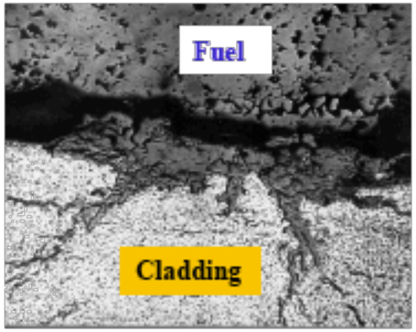
Protection against PCI/SCC failure is required for pushing the fuel burnup.



PCI/SCC Failure



Circumferential ridges in fuel pins

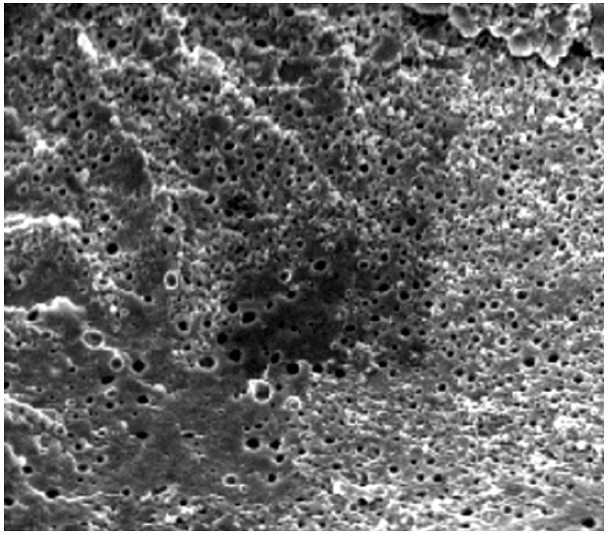


Incipient PCI/SCC cracks

Graphic from S. Banerjee, Bhaba Atomic Research Center

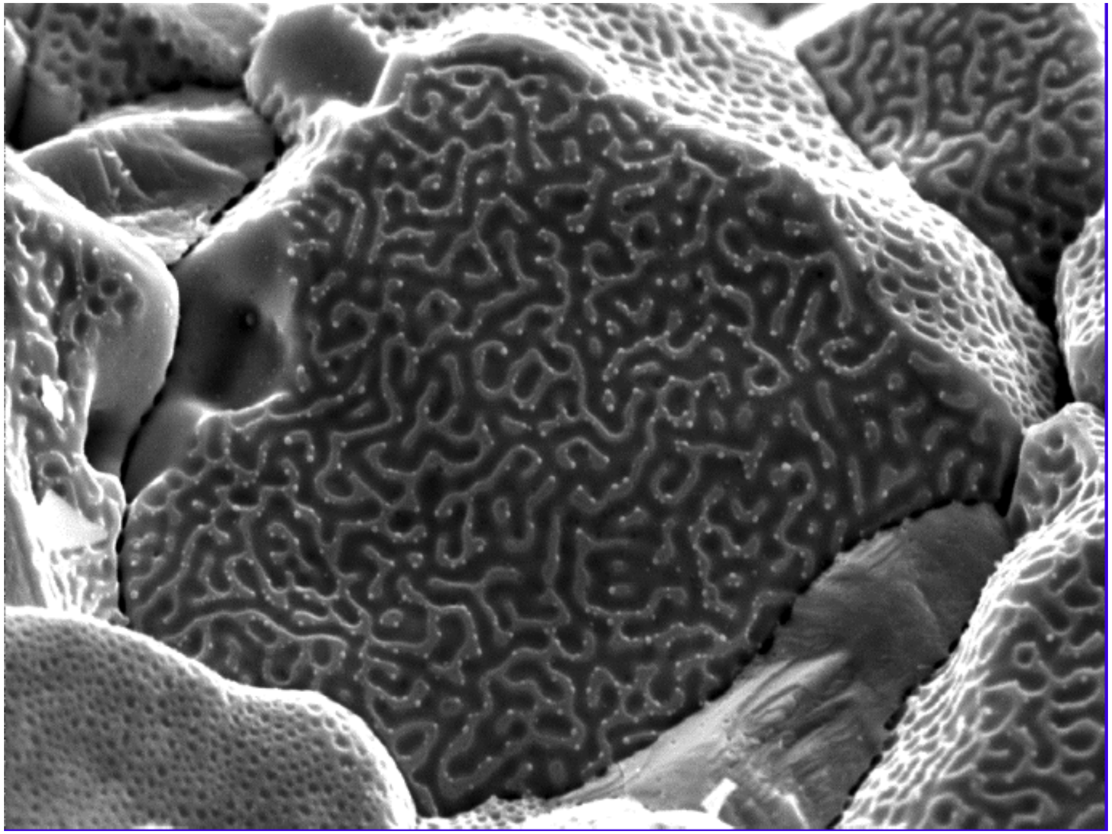
Swelling/Fission Gas Release

- Fission Products
  - Two atoms replace every U (or Pu) atom that fissions
  - 25% of fission products are gas atoms (Kr, Xe)
- Fuel Swelling
  - Fuel swells due to generation of fission products
  - Gas atoms coalesce into bubbles, accelerating swelling
  - Fuel swelling tends to reduce or close gap
- Fission Gas Release
  - Some fission gas escapes fuel
  - Pressurizes plenum
  - Percent of gas escaping fuel
    - < 10% in LWR fuel
    - > 50% in fast reactor fuel



Bubbles in metallic fuel

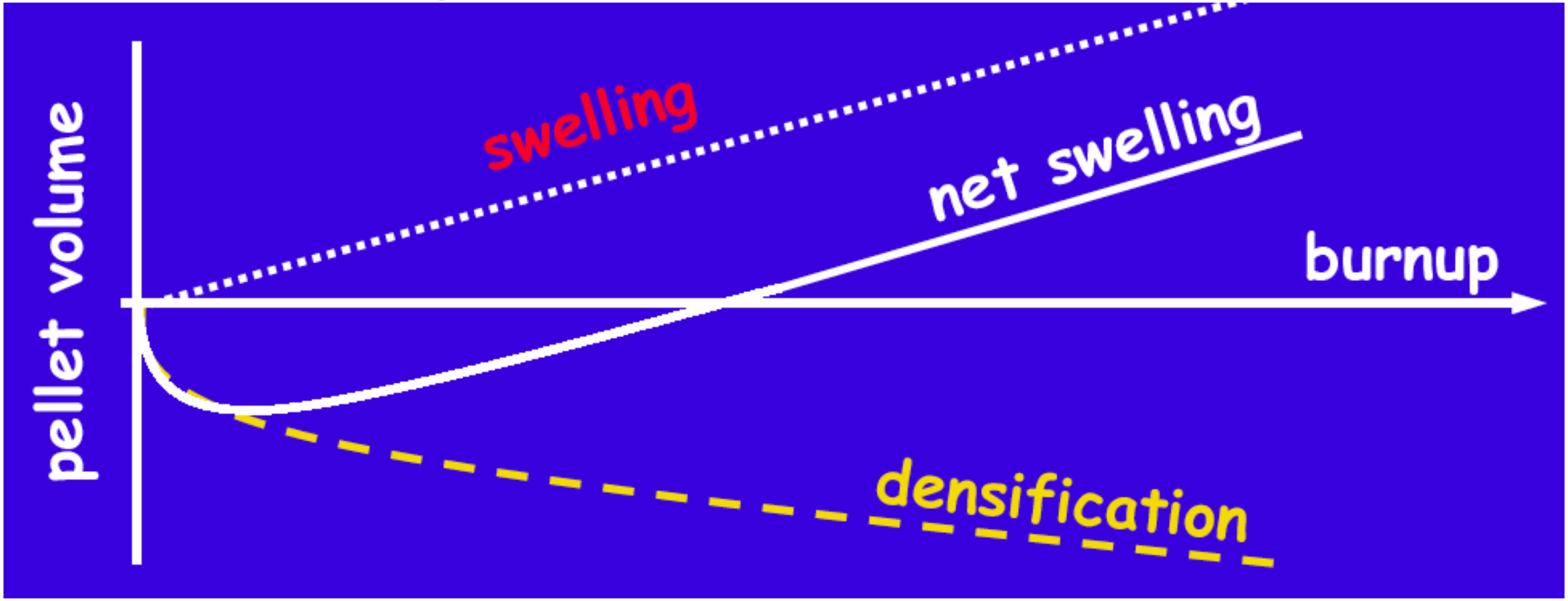
Fission Gas Release



- Fission gas pockets can eventually link and escape to plenum
- Gap can be sized to prevent fuel-cladding mechanical interaction
- Fission product gases decrease gap thermal conductivity

Graphic from Dr. Rebecca Weston, BNFL

Fuel-Cladding Interaction



- Densification as pores in ceramic fuel sinter
- Swelling due to solid and gaseous fission products
- Xenon and Krypton insoluble in UO_2

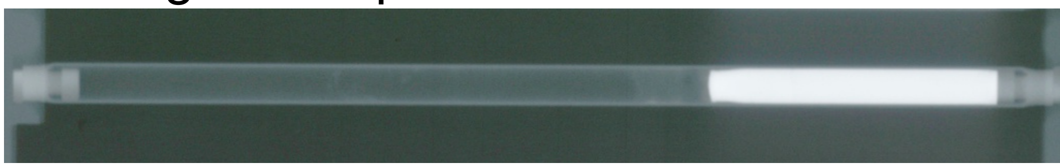
Graphic from Dr. Rebecca Weston, BNFL

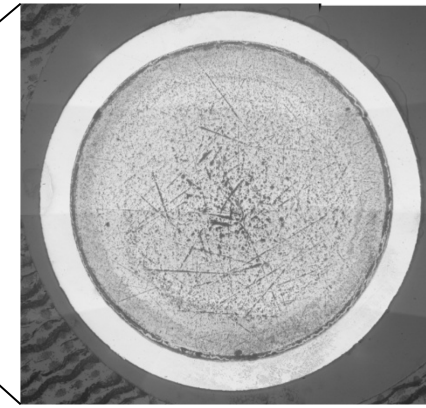
Life-Limiting Phenomena

- Cladding breach ends a fuel element’s use
- Cladding breach occurs due to:
  - Embrittlement of zirconium cladding due to corrosion/ hydriding by water coolant and stresses induced by FCMI (LWRs) → motivates development of corrosion-resistant cladding alloys
  - Creep rupture of cladding due to fission gas pressurization, accelerated by cladding thinning due to FCCI (LMRs) → motivates development of creep-resistant cladding alloys
- Burnup limit set to preclude cladding breach during irradiation

Current Fuel Development Efforts

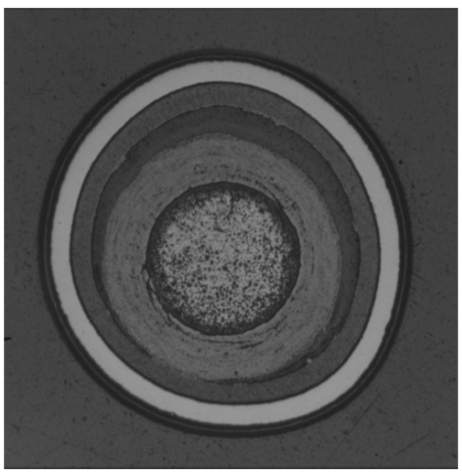
- Fuel Cycle R&D TRU-bearing metallic and ceramic fuels for high burnup



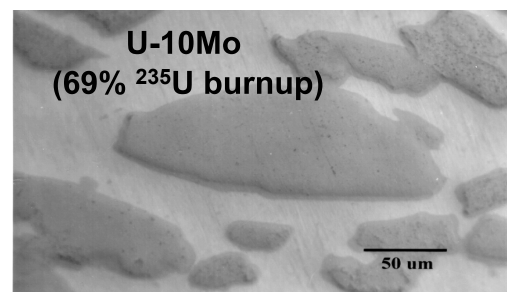


AFC-1H (U-29Pu-4Am-2Np-30Zr)
33.2 at% fissile burnup
(3.91E21 f/cm^3)

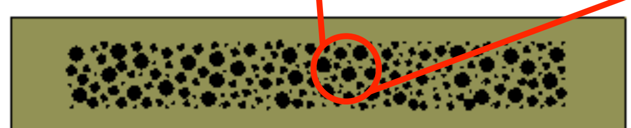
- High Temperature Gas Reactor TRISO Fuels



AGR-1 LEU (19.8% U^{235})
TRISO-coated Uranium Oxy-Carbide
12% FIMA burnup



- Reduced Enrichment Dispersion Fuels



Fuels and Reactors

Fuel Types

- A nuclear fuel is a removable component that is introduced into a reactor core and contains the U (or Pu) to be fissioned in a reactor
- Nuclear fuels differ widely from reactor to reactor
  - Geometrical configuration of fuel and cladding
    - Fuel rods
    - Fuel plates
    - TRISO pellets
  - Materials used for U-bearing (or Pu) fuel
    - Ceramic compounds
    - Metallic alloys
  - Materials used for cladding
- Fuel-clad system is designed to
  - Produce and transfer heat to the coolant while
  - Prevent fission products from reaching the coolant

Cladding Performance

- Survival of cladding must be predictable
  - Wastage
    - Corrosion by the coolant
    - Fuel cladding chemical interaction (FCCI)
  - Strain
    - Fission gas pressurization
    - Swelling and associated creep of constrained components
    - Fuel cladding mechanical interaction (FCMI)
  - Microstructural stability during accident transients

Fuel Element Materials

- Fuel Materials
  - Oxides: UO_2, $(U,Pu)O_2$
  - Carbides: UC, (U,Pu)C
  - Nitrides: UN, (U,Pu)N
  - Metal Alloys: U-Pu-Zr-Mo
  - Others: UAl_x, U_3Si_2, U/Zr hydride, UCO
- Bond (Gap) Materials
  - Helium gas
  - Liquid sodium
- Cladding Materials
  - Zirconium Alloys for LWRs
  - Stainless Steels
  - Aluminum Alloys for Research and Test Reactors
  - Refractory Alloys for High Temperature Applications (i.e., W, Ta, Nb, Mo, V)

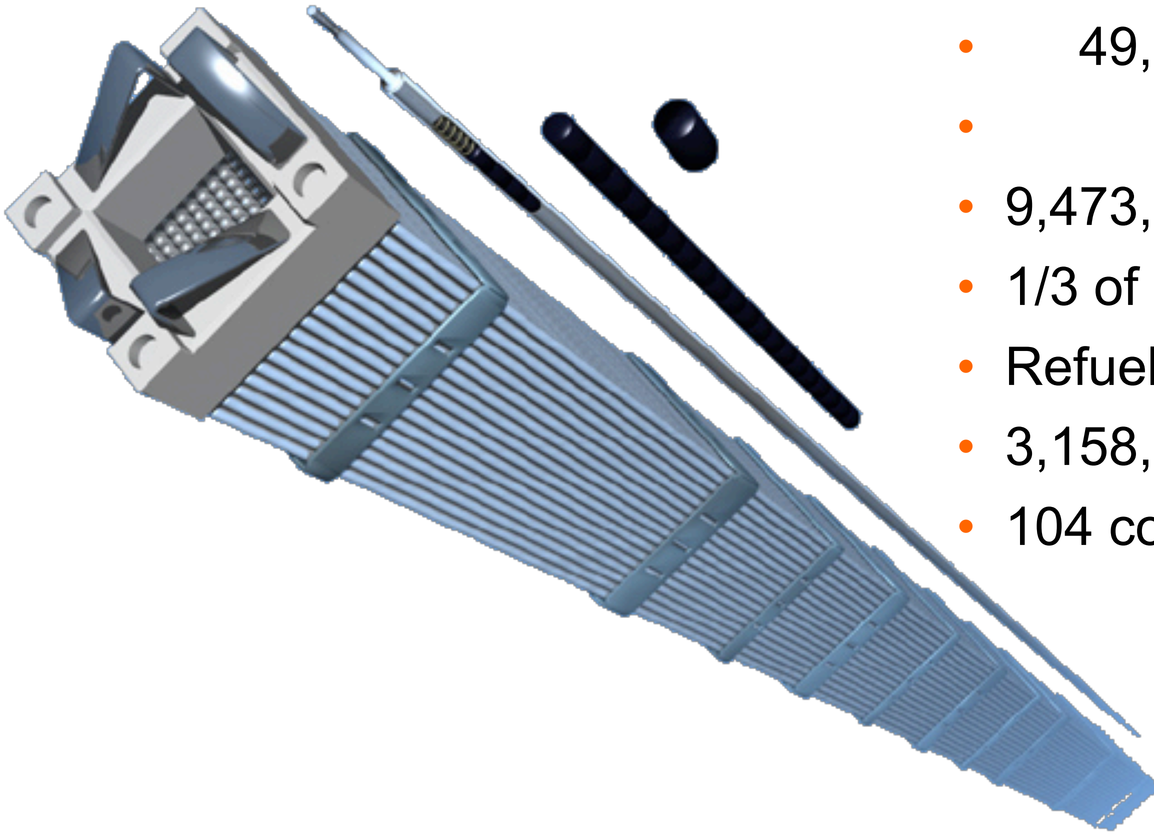
Table 10.1 Nuclear Fuels

Fertile Fuels		Fuels Fissionable by Thermal Neutrons
		U-235
U-238	⇒	Pu-239
Th-232	⇒	U-233

Uranium solid ceramic pellets

- Typical pellets:
  - 0.25" (6.3 mm) diameter
  - 0.50" (12 mm) height
- Pellet fabricated geometry
  - dish and chamfer
  - accommodate swelling
- U.S. Commercial Reactors
  - fuel enriched to 5% ^{235}U
  - ^{238}U breeds ^{239}Pu
  - ^{239}Pu fissions occur

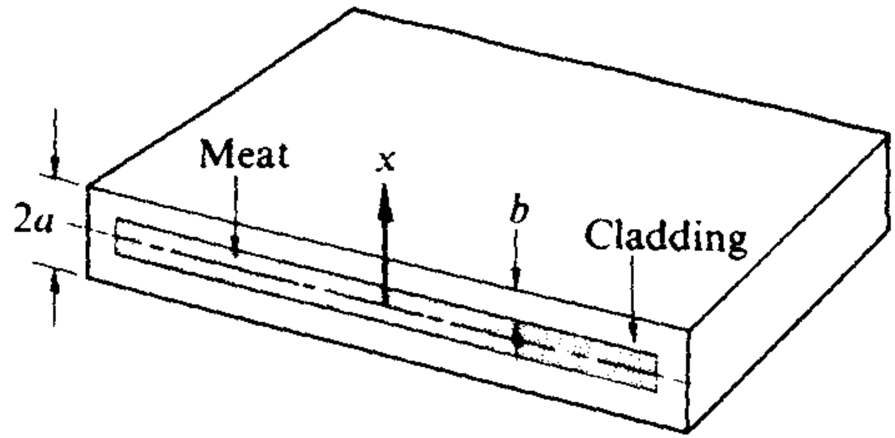
Fuel pellets grouped into fuel pins and assemblies



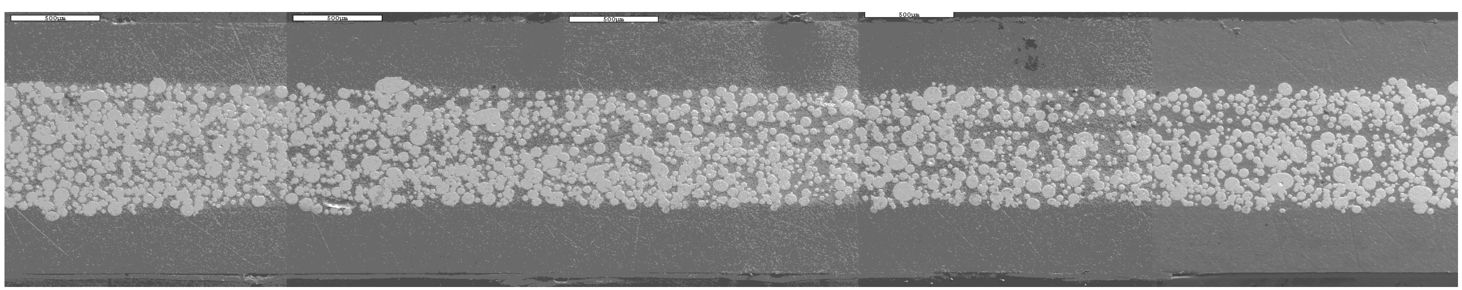
- Fuel height ~ 93 in. (3.7 m)
- 186 pellets/pin
- 264 pins/assembly
- 49,000 pellets/assembly
- 193 assemblies/core
- 9,473,000 pellets/core
- 1/3 of core replaced per refuel
- Refuel every 18 months
- 3,158,000 pellets/reload/reactor
- 104 commercial reactors in U.S.

Plate Fuel

- Research and test reactors (ATR, HFIR)
- Dispersion fuels (i.e., fuel particles in metal matrix)
- Solid fuel foils
- No plenum

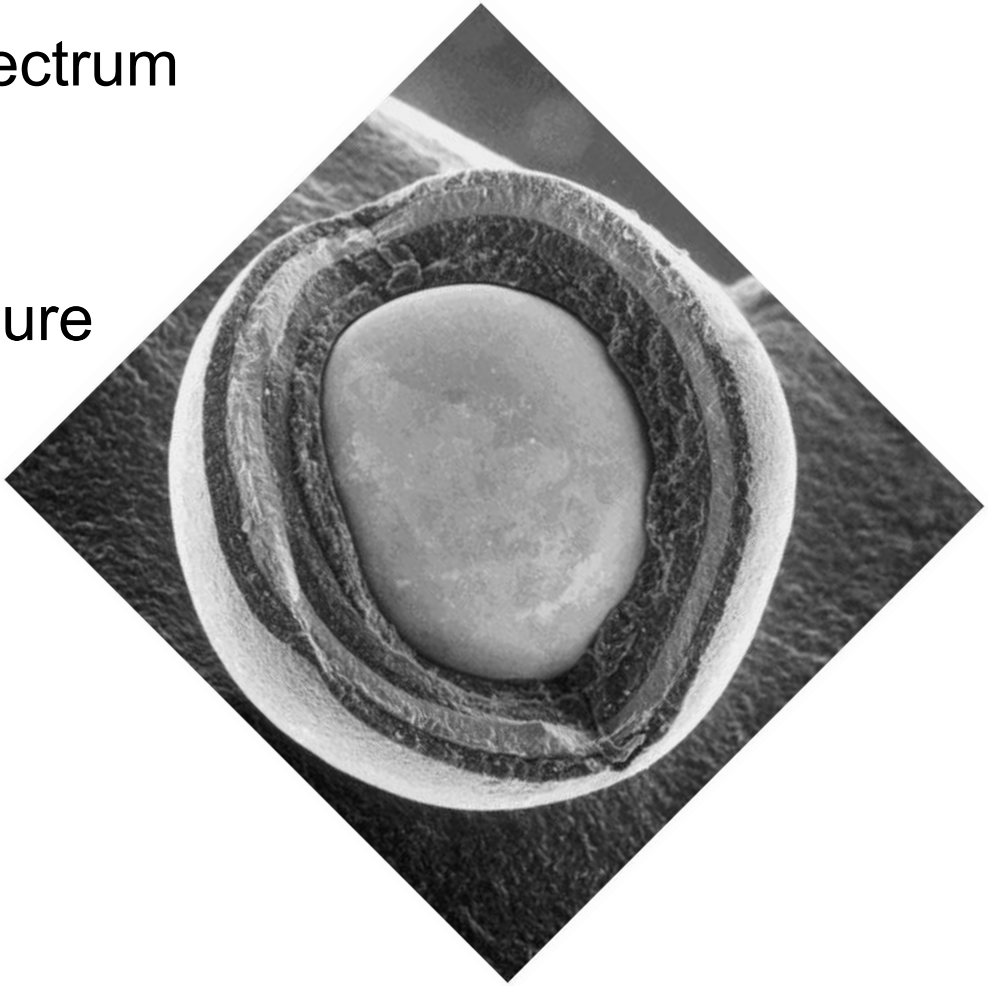


Cladding
Dispersion Fuel
Cladding



TRISO Particle Fuel

- Fuel particle for thermal spectrum gas-cooled reactors
- ~0.5 to 0.9 mm sphere
- Particles act as small pressure vessels
- Fission products remain inside coatings

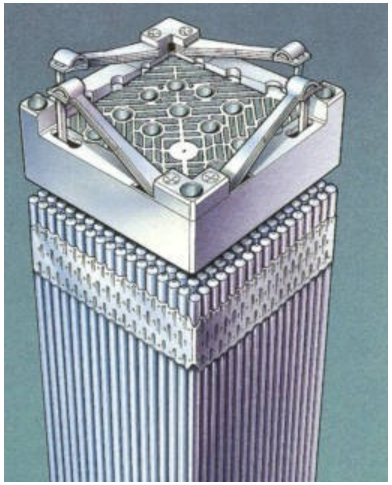


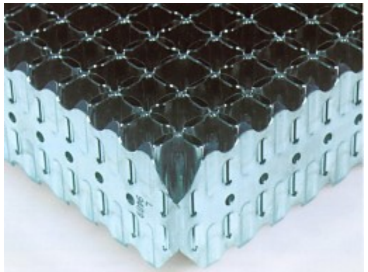
Fuel Assembly Performance

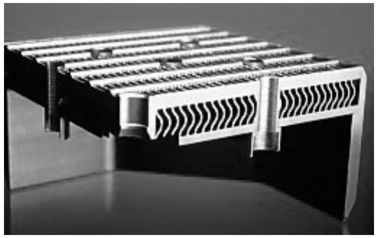
- Design Functions
  - Provide support and protection for the fuel-pin bundle and other components of the subassembly
  - Provide a controlled path for the primary coolant
  - Provide a compact structural unit that can be easily moved in and out of the core by a refueling machine
  - Interact with adjacent subassemblies, retaining ring, and core support plates in a manner that assures safe and predictable reactor geometry
- Design Issues
  - Swelling, creep, fatigue, toughness
  - Reduced limits for elements

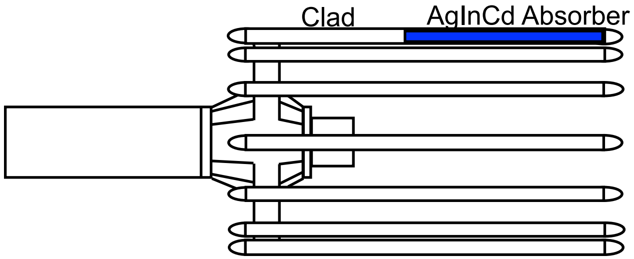
Fuel Assembly – PWR

- Fuel Assembly
  - Fuel rods
    - Cladding
    - Pellets
  - Grids (Spacers)
  - Guide tubes
  - Nozzles (tie plates)
- Control Rods





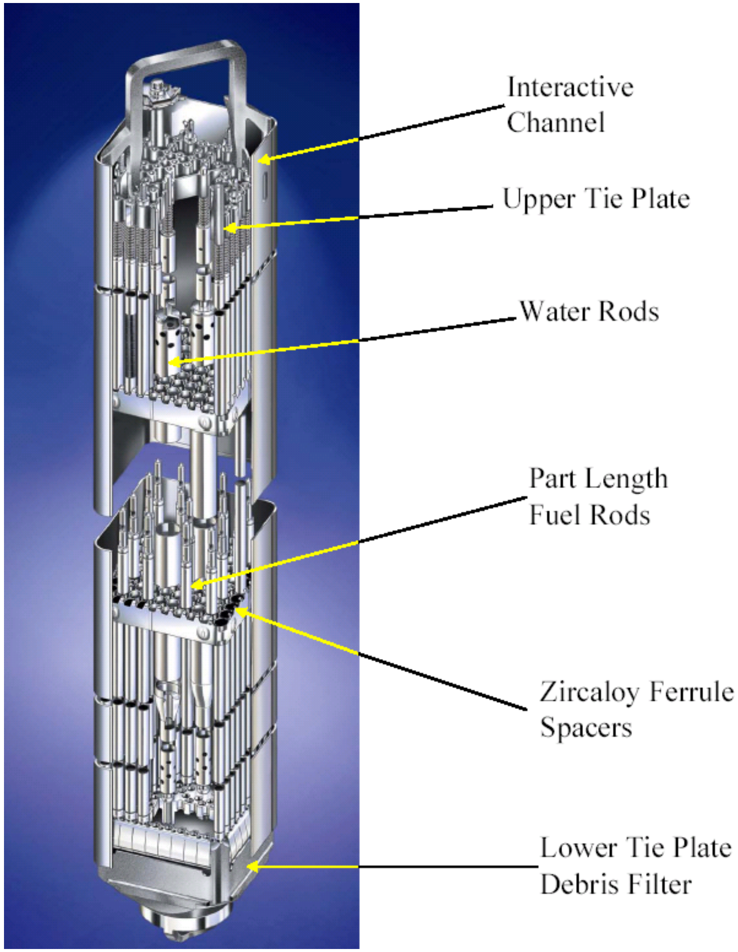




Graphic from Rosa Yang, EPRI

BWR Fuel

- Fuel Assembly
  - Fuel rods
    - Cladding
    - Pellets
  - Spacers
  - Water Rods / Box
  - Tie Plates
  - Channel
- Control Blades



Rosa Yang, EPRI

TRISO Fuel Particles: Prismatic Fuel System

- TRISO fuel particles pressed into compacts with graphite matrix

- Fuel assembly has channels for fuel compacts and channe for gas (coolant) flow

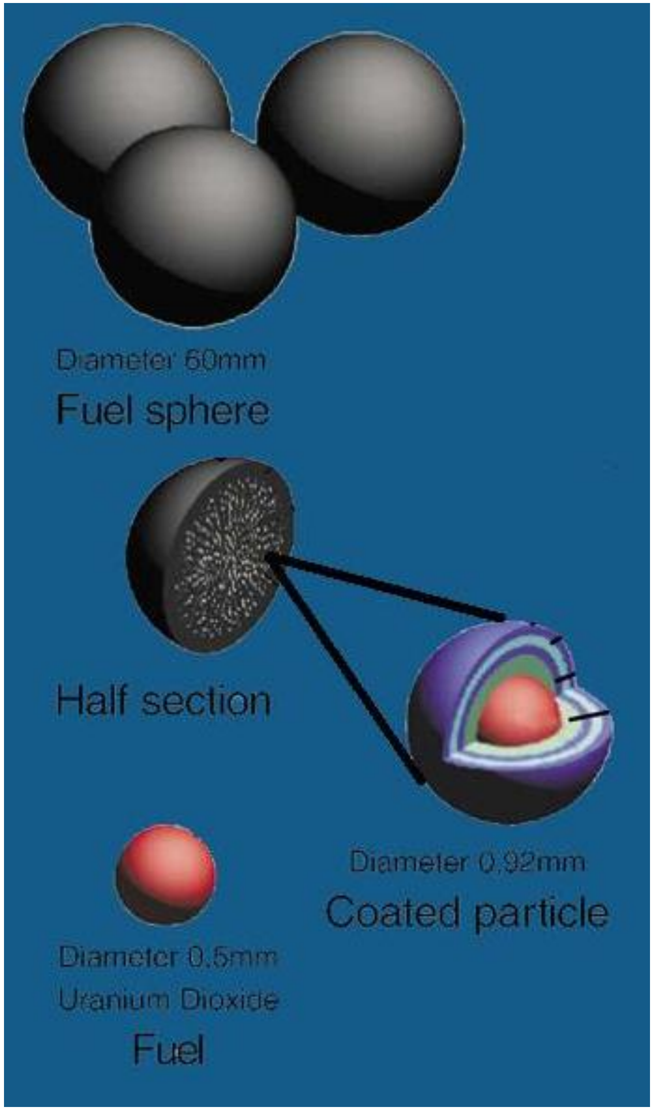
TRISO Coated Particles →

Compacts →

Fuel Assembly

TRISO Fuel Particles: Pebble Bed Modular Reactor

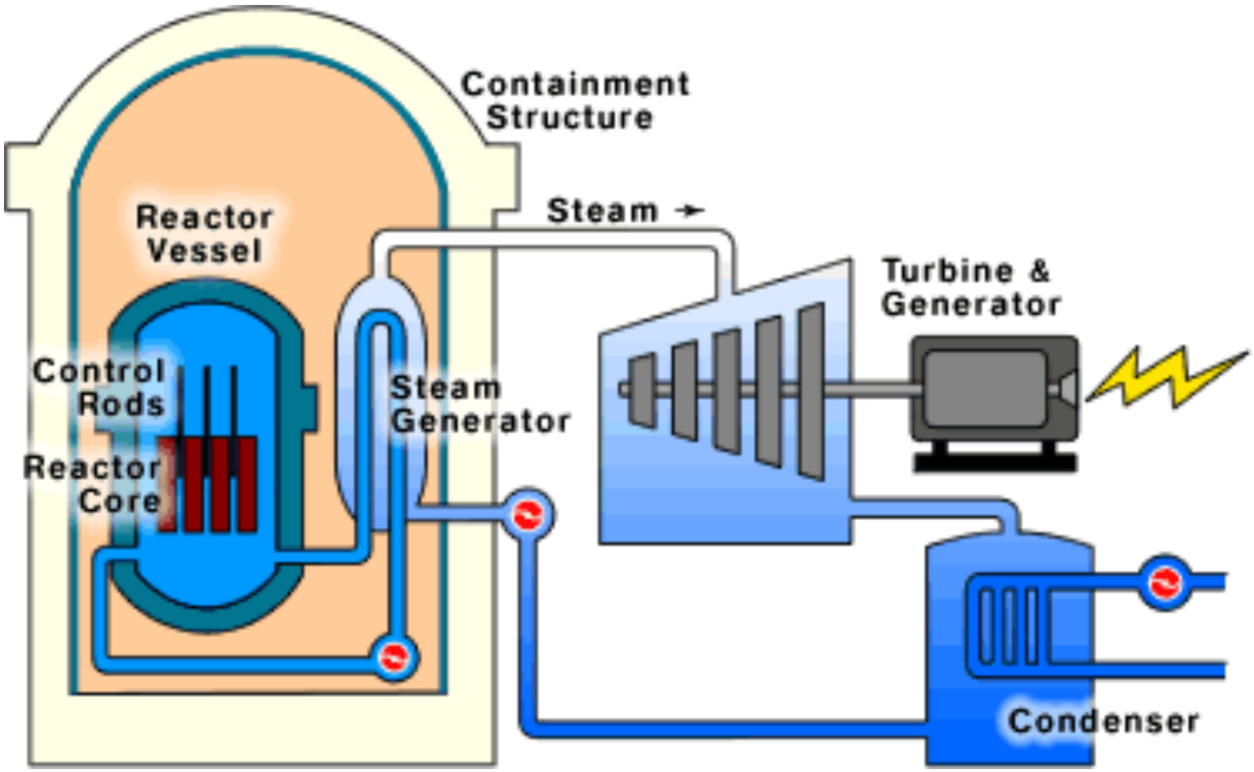
- Fuel element for the Pebble Bed Modular Reactor
- 11,000 TRISO particles in every pebble



Reactor Types

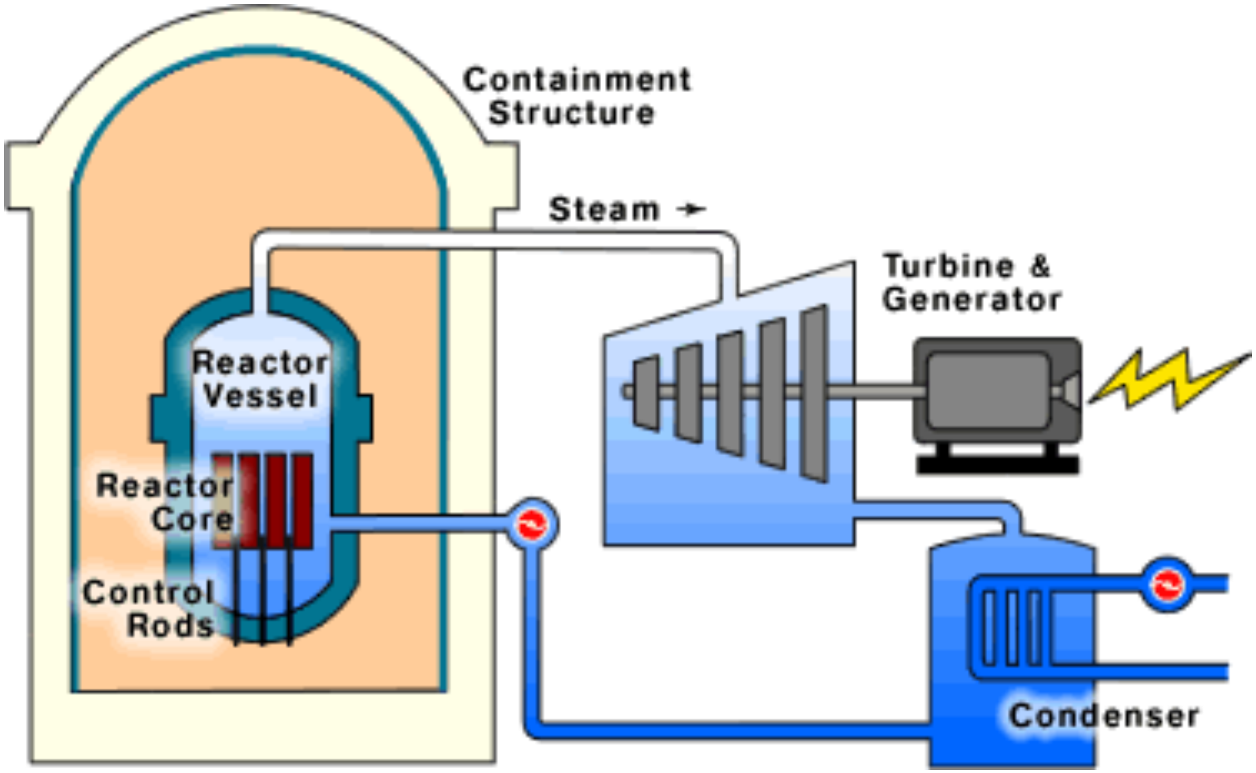
Pressurized Water Reactor

- Coolant water is pressurized to prevent boiling
- 325°C outlet temperature
- >1000 Mw_e
- Soluble poison and chemistry adjustment in coolant water
- Most common reactor design



Boiling Water Reactor

- Coolant water boils
- 288°C outlet temperature
- >1000 MW_e
- Cannot use soluble poison or chemistry adjustment
- Lower pressure system



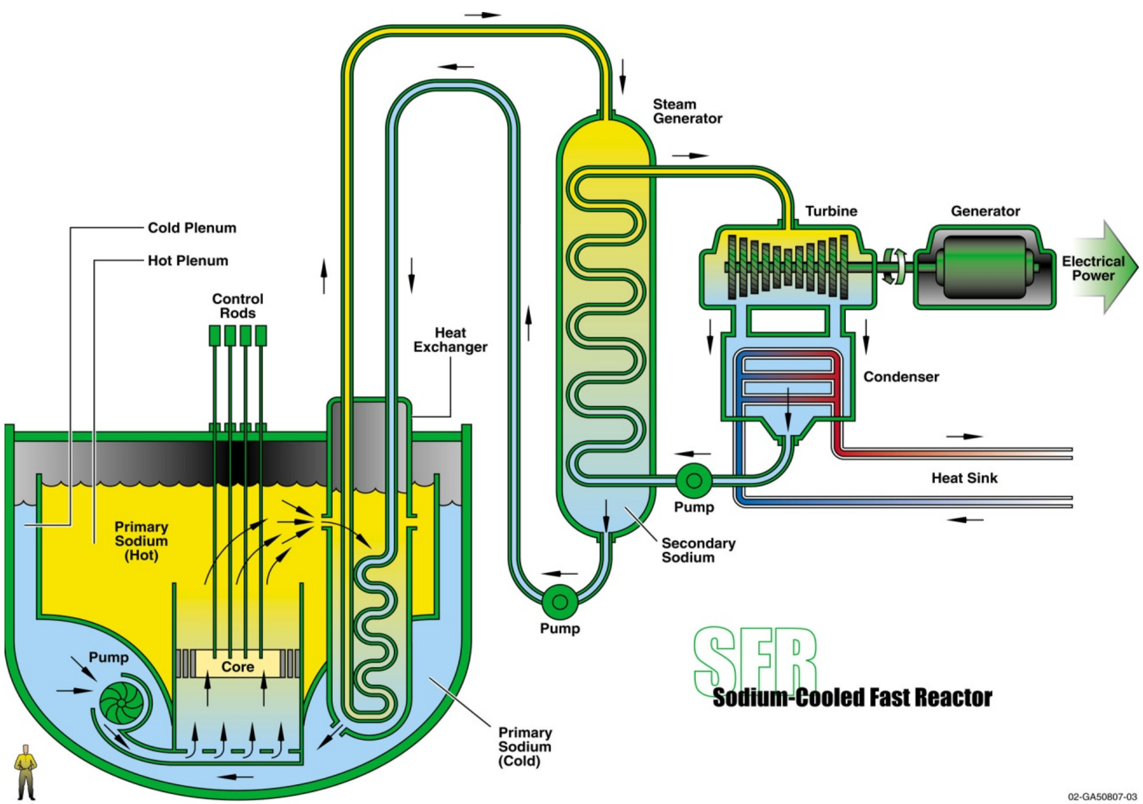
Sodium-Cooled Fast Reactor (SFR)

Characteristics

- Sodium coolant
- 550°C outlet temperature
- 150 to 500 MW_e
- Metal fuel with pyro processing
- MOX fuel with advanced aqueous

Benefits

- Consumption of LWR actinides
- Efficient fissile material generation



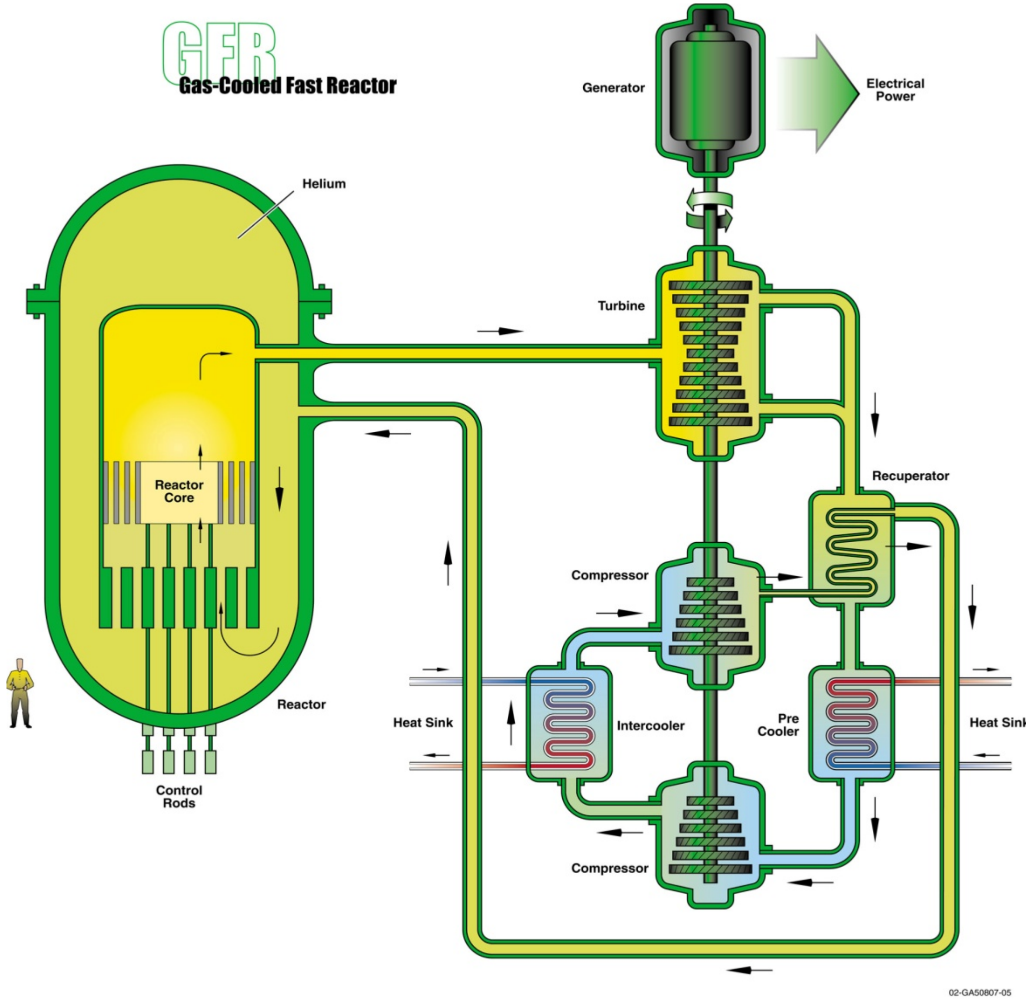
Gas-Cooled Fast Reactor (GFR)

Characteristics

- He coolant
- 850°C outlet temperature
- Direct gas-turbine conversion cycle 48% efficiency
- 600 MW_{th}/288 MW_e
- Several fuel options and core configurations

Benefits

- Waste minimization and efficient use of uranium resources



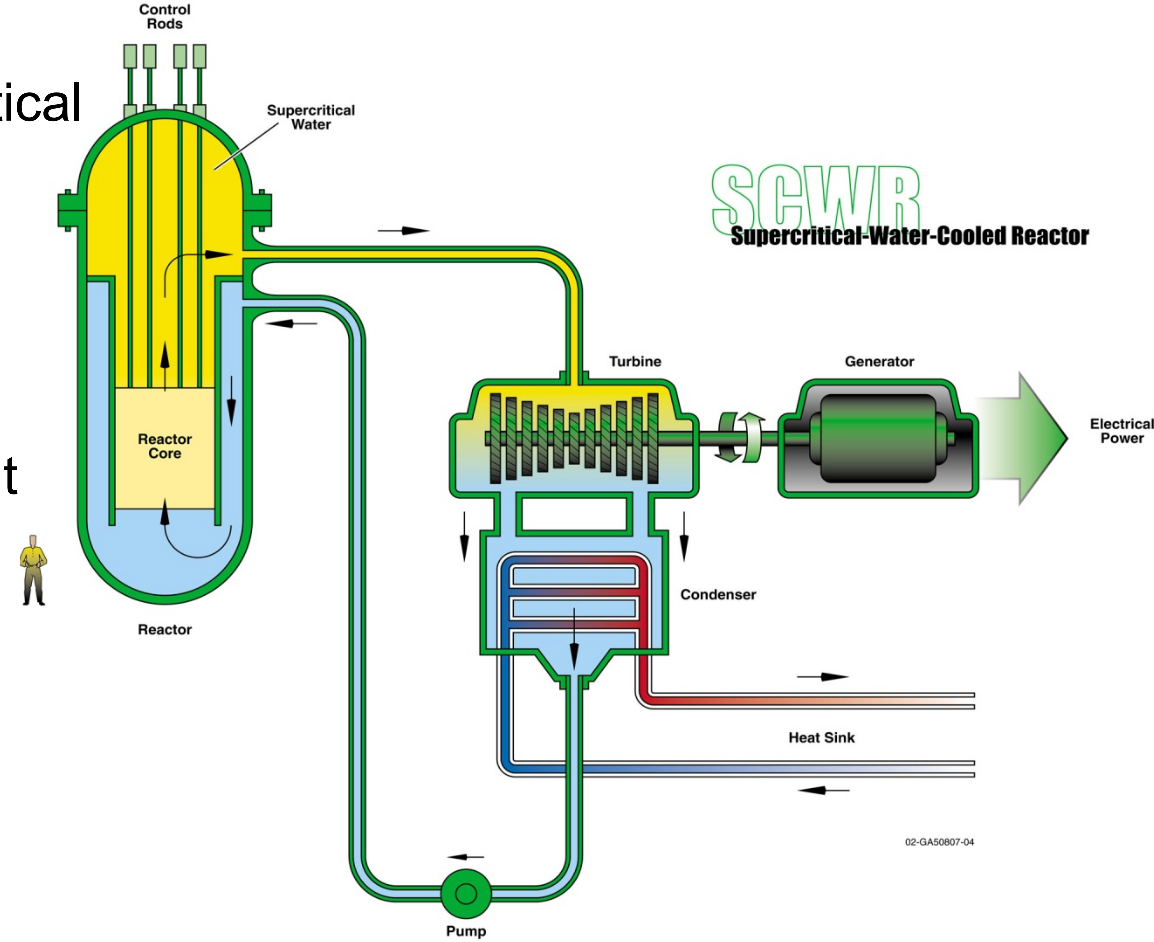
Supercritical-Water-Cooled Reactor (SCWR)

Characteristics

- Water coolant at supercritical conditions
- 550°C outlet temperature
- 1700 Mwe
- >20 MPa
- Simplified balance of plant

Benefits

- Efficiency near 45% with excellent economics
- Thermal or fast neutron spectrum



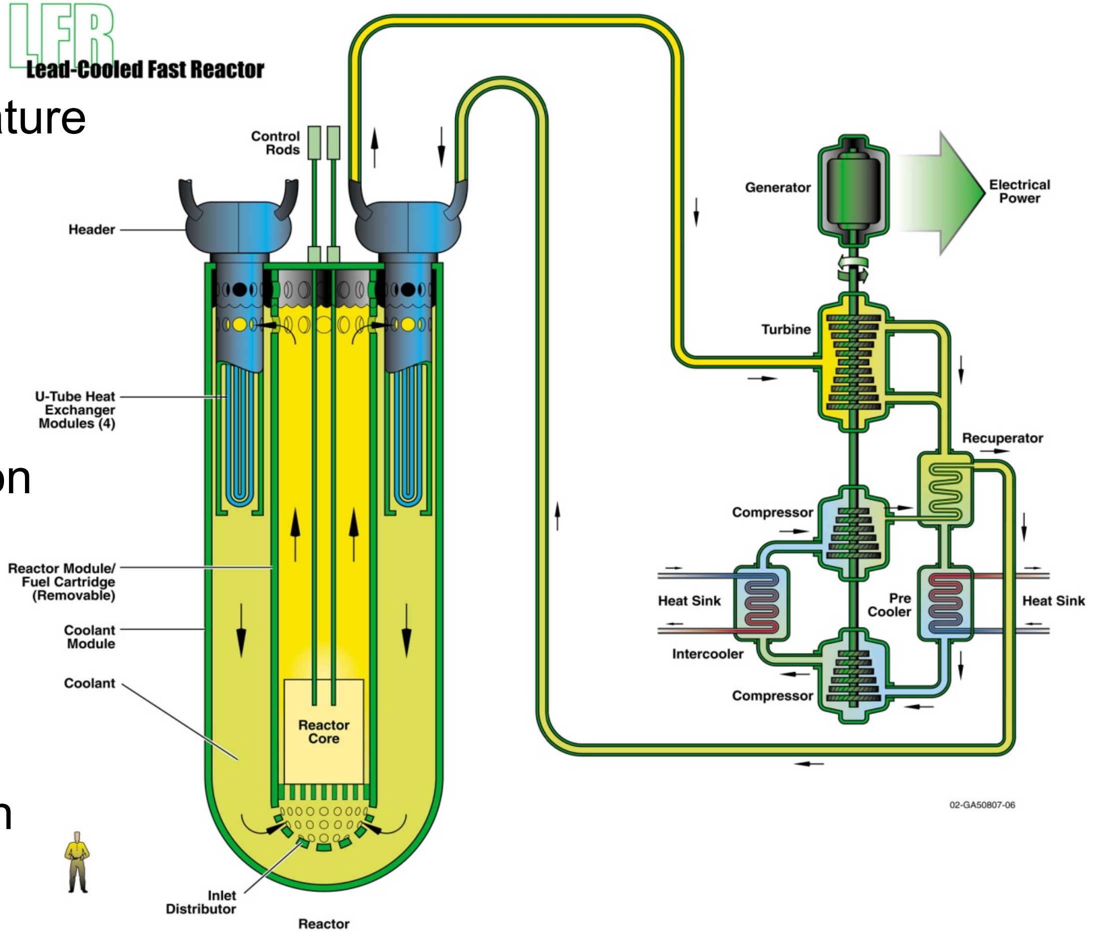
Lead-Cooled Fast Reactor (LFR)

Characteristics

- Pb or Pb/Bi coolant
- 550°C to 800°C outlet temperature
- 120–400 MW_e
- 15–30 year core life

Benefits

- Distributed electricity generation
- Hydrogen and potable water
- Cartridge core for regional fuel processing
- High degree of passive safety
- Proliferation resistance through long-life cartridge core



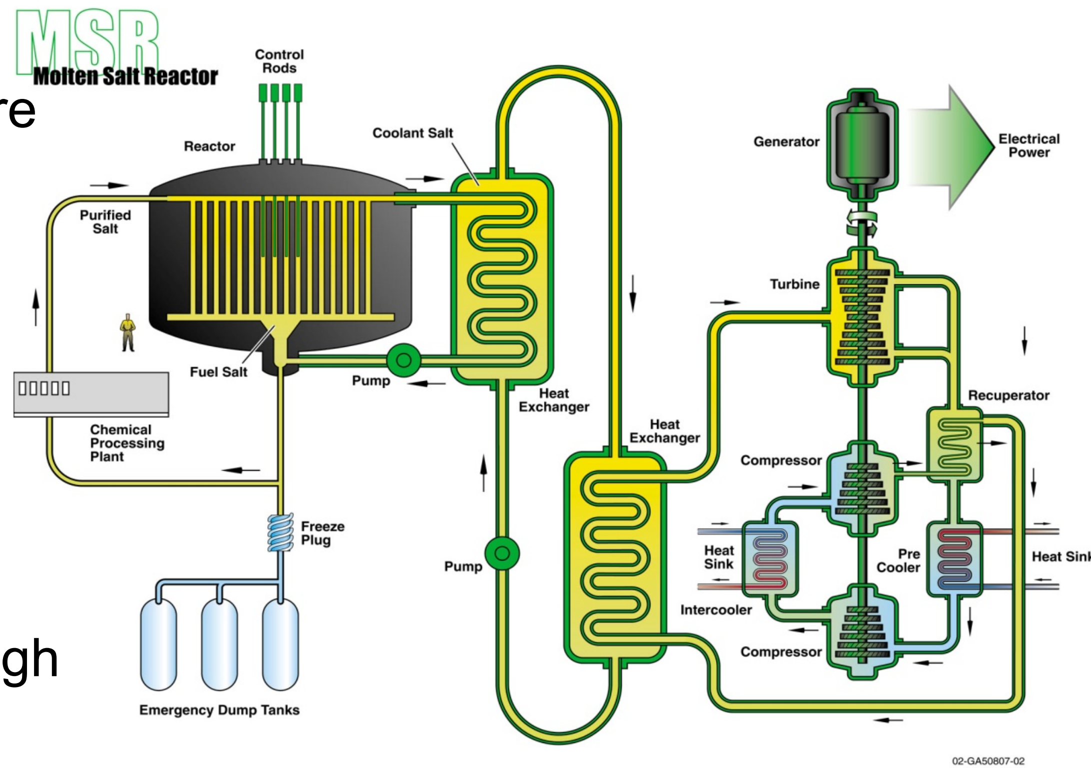
Molten Salt Reactor (MSR)

Characteristics

- Fuel: liquid Na, Zr, U and Pu fluorides
- 700–800°C outlet temperature
- 1000 MW_e
- Low pressure (<0.5 MPa)

Benefits

- Waste minimization
- Avoids fuel development
- Proliferation resistance through low fissile material inventory



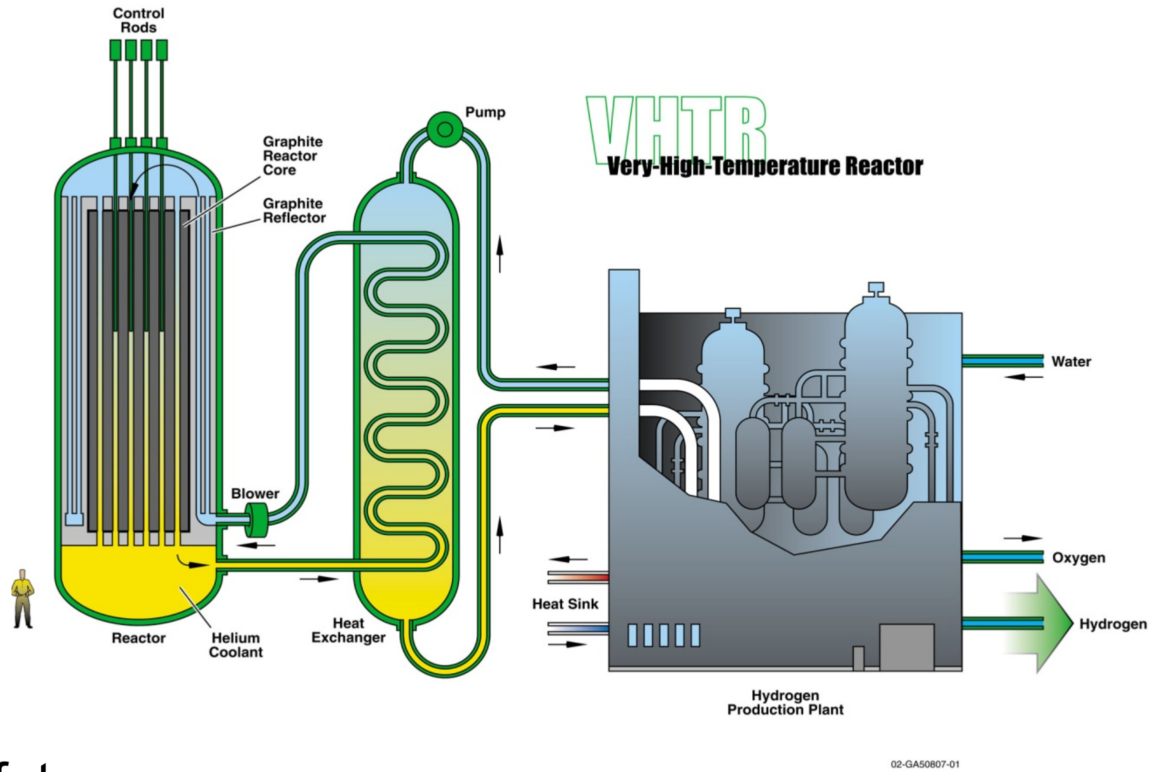
Very-High-Temperature Reactor (VHTR)

Characteristics

- He coolant
- 1000°C outlet temperature
- 600 MWe
- Solid graphite block core based on GT-MHR

Benefits

- High thermal efficiency
- Hydrogen production
- Process heat applications
- High degree of passive safety



Introduction to Nuclear Reactors, Fuels, and Materials

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