Joint KINS-IAEA-ANNuR/ANSN/FNRBA
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Basic Nuclear Reactor Physics and Theory

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Who is the Professor?

- Ph.D of Nuclear Engineering in KAIST
- Currently,
  - Professor and Principal Researcher in INSS/KINS
  - Adjunct Professor in KAIST since 2019

- Major Career ;
  - Started Regulatory Work in KINS from 1984
  - Plant Inspector
  - Research PM (Risk-Informed Regulation, System Safety Performance)
  - Regulatory PM (Regulatory Inspection, Power-Uprate Safety Review)
  - Head, Dept. of Risk Assessment (PRA & Severe Accident)
  - Director, International Nuclear Safety School(INSS)
  - Director, Div. of Operating Reactor Regulation and Div. of Reactor Licensing
  - Vice President of KINS
  - President of KINS

Lecture Objectives

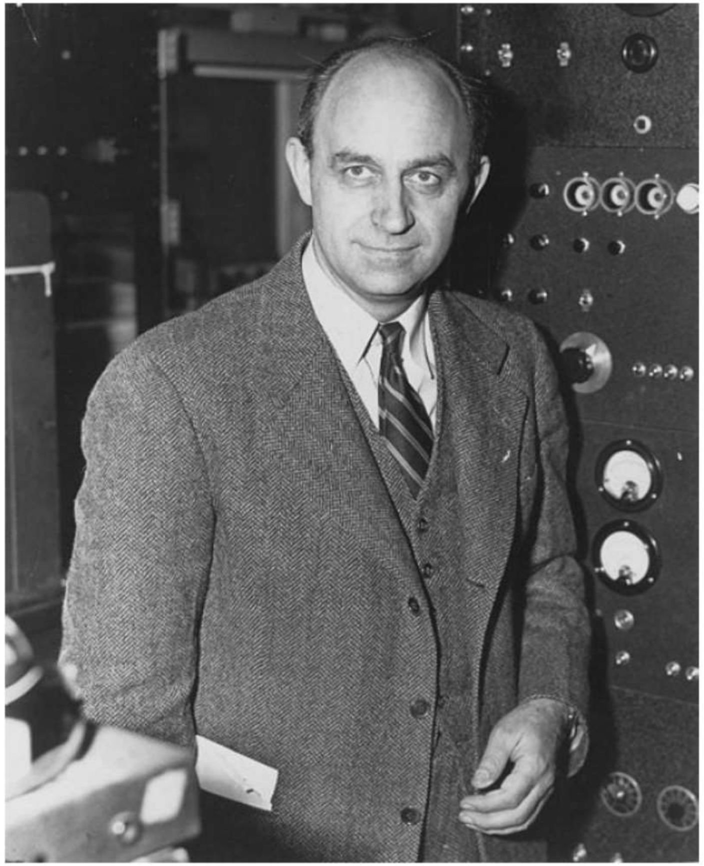
- To learn how a nuclear fission is proceeded and sustained as a heat source in reactor(neutron generation, moderation, absorption and fission)
- To learn how the fission process is affected by the operating parameters(temperature, pressure, power) in a reactor core
- To learn how the fission process is controlled (Reactor kinetics)
- Main topics are as follows;
  - Basics of Nucleus & Nuclear force
  - Nuclear fission, Fissile & Fissionable materials
  - Neutron life cycle in the reactor core (Six factors)
  - Reactivity and (Reactivity) Coefficients
  - Reactor Kinetics
  - Xenon Poison

Lecture Topics

- **History of Nuclear Fission**
  - Discovery of Nuclear Fission
  - Chicago Pile Experiment
- **Nuclear Physics**
  - Principle of Nuclear Fission
  - Criticality & Reactivity
  - Reactivity Coefficients
- **Neutron Control in Reactor**
  - Reactor Kinetics
  - Neutron Flux Shape Control
  - Xenon Poison

Discovery of Fission(1)

- Enrico Fermi’s Experiment (1934)
  - showed neutrons could spilt many kinds of atoms. When he bombarded uranium with neutrons, he did not get the elements he expected. The elements were much lighter than uranium.
- Otto Hahn and Fritz Strassman’s Experiment (1938)
  - Fired neutrons into uranium(atomic no. 92) resulting in generation of lighter elements, such as barium(atomic No. 56) in the leftover materials.



Enrico Fermi, c. 1943-1949 (National Archives and Records Administration)

Enrico Fermi

Discovery of Fission(2)

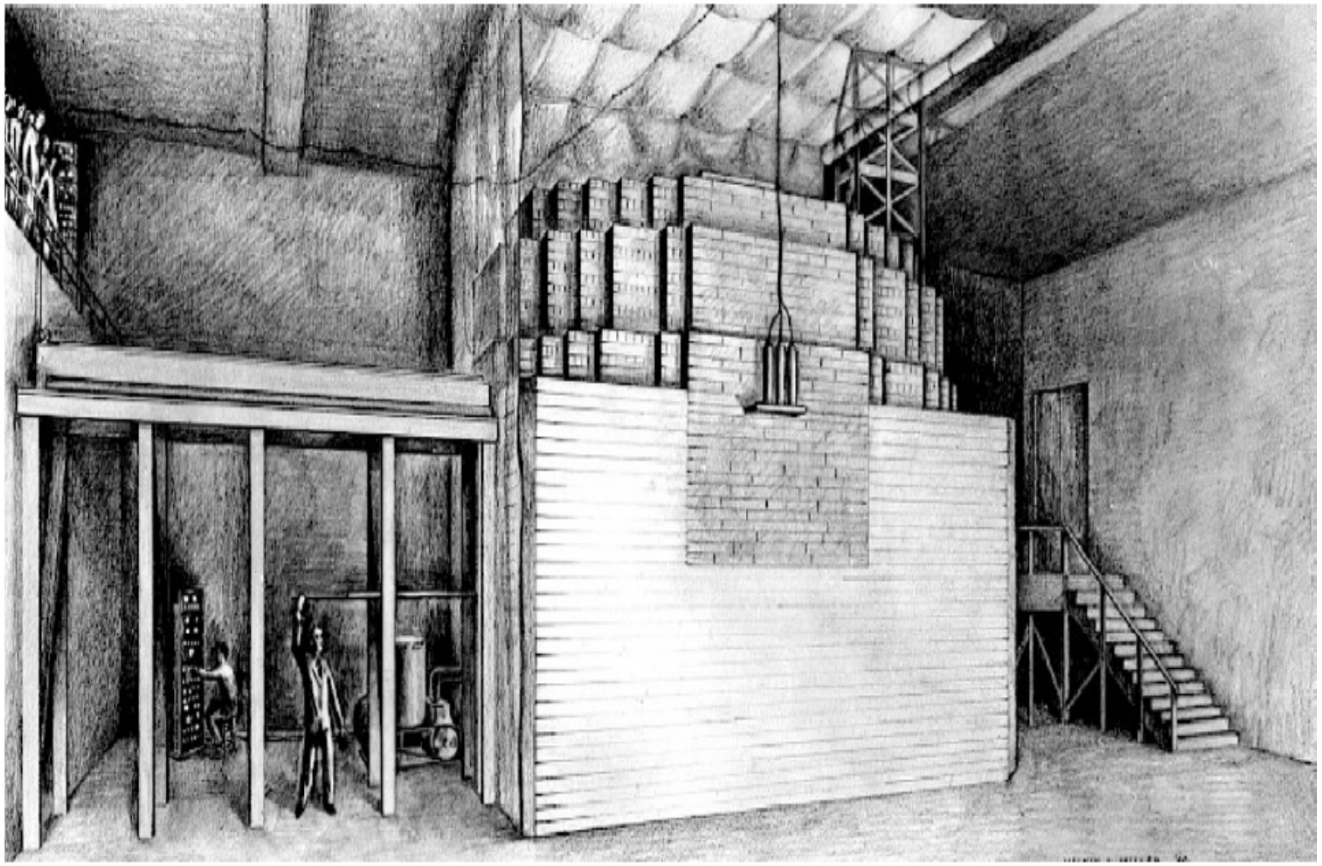
- Lise Meitner and Otto Frisch' s Study
  - From Hahn and Fritz Strassman's experiment, they thought the barium and other light elements in the leftover material resulted from the uranium splitting (fissioning)
  - However, they found that atom masses of the fission products did not equal to the uranium mass. Meitner used Einstein theory to show the lost mass changed to energy. (calculated the energy release from the fission as about 200MeV, and confirmed experimentally in 1939)
  - This proved fission occurred and confirmed Einstein's work. ($E= \Delta MC^2$)

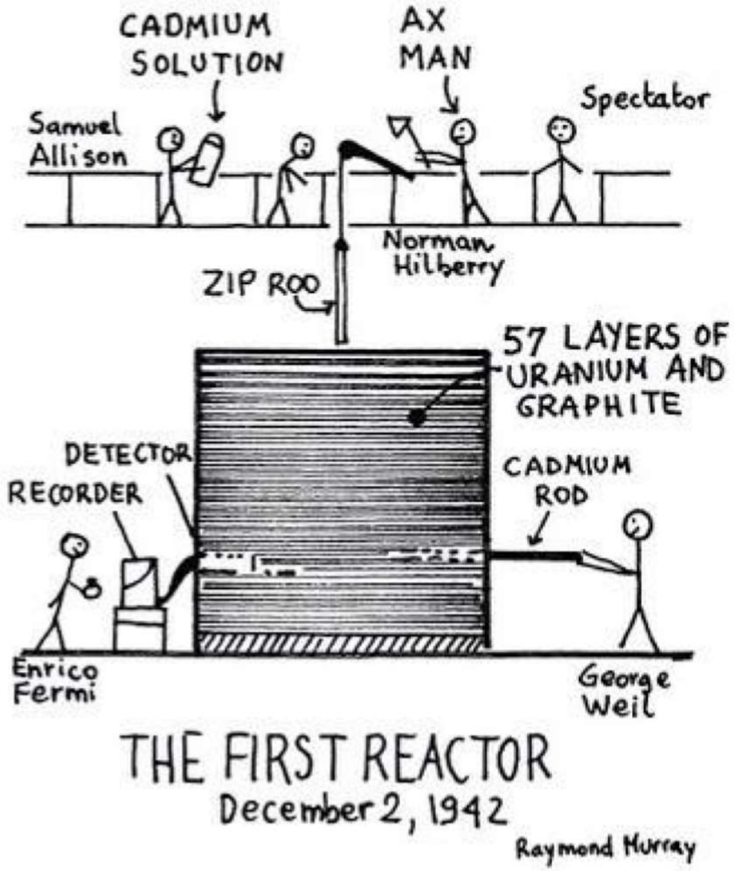
Lise Meitner and Otto R. Frisch

The First Self-Sustaining Chain Reaction

- Many scientists began to believe a self-sustaining chain reaction might be possible if enough uranium could be brought together under proper conditions.
  - The amount of uranium needed to make a self-sustain chain reaction is called a critical mass.
- Early in 1942, a group of scientists led by Fermi gathered at the University of Chicago(UC) to develop their theories.
- By November 1942, they were ready for construction to begin on the world's first nuclear reactor, which became known as ***Chicago Pile-1.***
- In December 2, 1942, the nuclear reaction became self-sustaining when the control rods were withdrawn a few inches.

Picture of Chicago Pile-1(CP-1)





Chicago Pile-1(CP-1)

- The pile consists essentially of a lattice of lumps, partly of uranium metal and partly of uranium oxide imbedded in graphite(57 layers).
- Shape of structure may be roughly described as a flattened rotational ellipsoid having the polar radius 309cm and equatorial radius 388cm. The graphite is supported on a wooden structure and rests on the floor on its lowest point.
- During the construction as a matter of precaution, apparently before reaching this critical layer, some cadmium strips were inserted in suitable slots.
  - They were removed once every day with the proper precautions in order to check the approach to the critical condition. (Critical was estimated to occur at about the 56^{th} layer)
  - The actual construction was carried in this way to the 57^{th} layer, about one layer beyond critical dimensions. When all cadmium is removed the effective reproduction factor of the structure is about 1.0006.

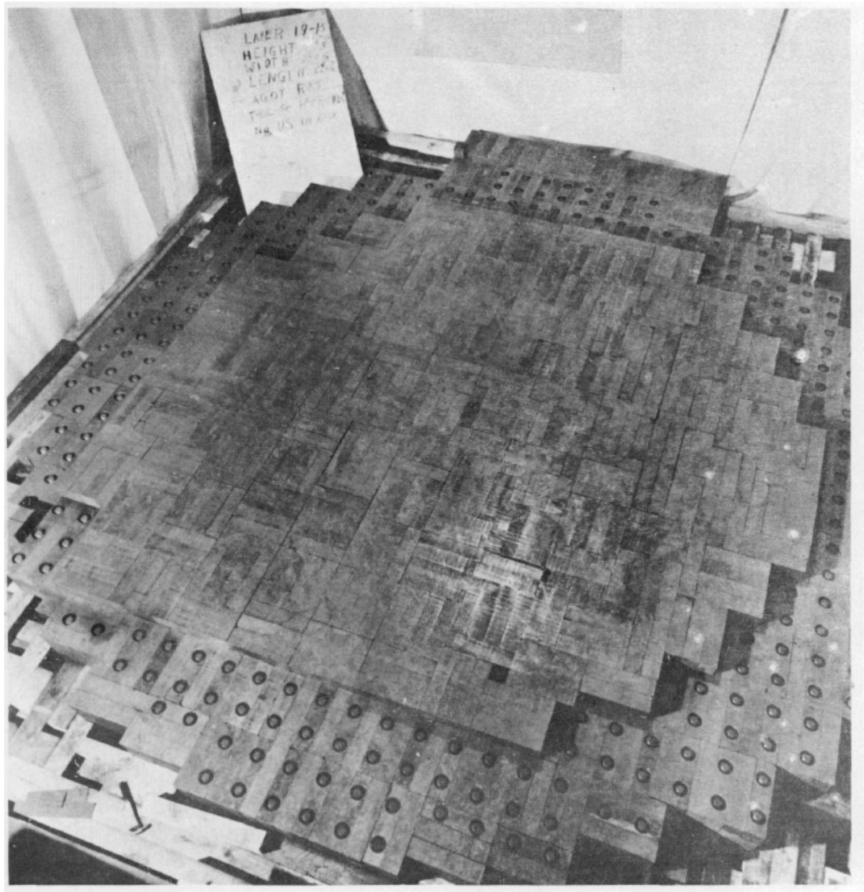


Photo of the 19th and 18th layers

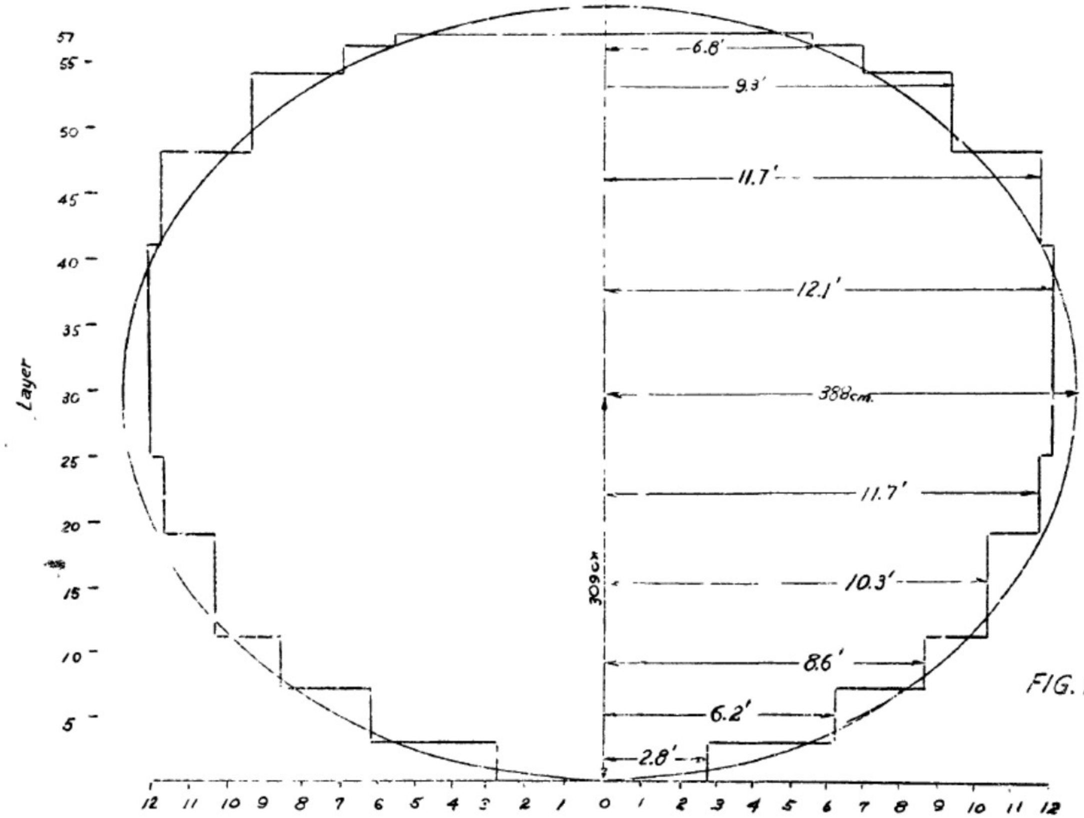


FIG. 1

57 layers, polar radius 309cm and equatorial radius 388cm

Chicago Pile-1(CP-1)

- The controlling of the reaction was obtained by inserting some strips of neutron absorbing materials(cadmium and, in one case boron steel) into the pile
- The pile was provided with two safety rods and one automatic control rod.
  - The safety rods were normally out of the pile during operation (hold by magnet, inserted into the pile by the weights of suitable system if the intensity of the neutron level is above a specific limit.)
  - The automatic control rods might be pushed inside and outside the pile by two electrical motors and be operated either by hand or by an amplifying system to meet the desired level.
- When all the cadmium was completely located outside of the pile, the intensity increased approximately at the rate of a factor 2 every minute.

Lecture Topics

- **History of Nuclear Fission**
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  - Chicago Pile Experiment
- **Nuclear Physics**
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  - Criticality & Reactivity
  - Reactivity Coefficients
- **Neutron Control in Reactor**
  - Reactor Kinetics
  - Neutron Flux Shape Control
  - Xenon Poison

Basics of Nucleus

- **Structure of the Atom**
  - Consists of dense nucleus of protons and neutrons surrounded by electrons travelling in discrete orbits
  - Nuclear Radius : $r = (1.25 \times 10^{-13}\ \text{cm})\ A^{1/3}$
    - A : Atomic mass number
    - H^1 (1.23×10^{-13} cm), U^{238} (7.74×10^{-13} cm)
  - Mass (unit : amu, 1amu : 1.6606×10^{-27} kg)
    - Neu. : 1.008665, Pro. : 1.007277, Ele. : 0.0005486

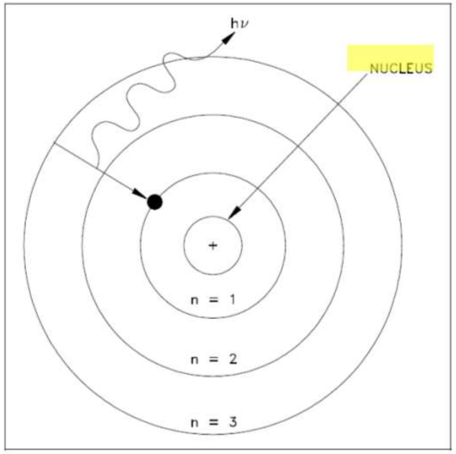


Figure 1 Bohr's Model of the Hydrogen Atom

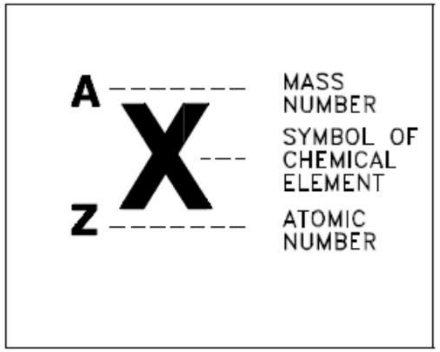


Figure 2 Nomenclature for Identifying Nuclides

- **Nuclides**
  - Each type of atom that contains a unique combination of protons and neutrons
  - Denoted by the chemical symbol of the element(X)
    - Atomic No.(subscript, Z), Mass No.(superscript, A)
    - A = Z(No. of proton or electron)+ N(No. of neutron)
    - Nucleons : protons and neutrons in the nucleus

Properties of Subatomic Particles

Particle	Location	Charge	Mass
Neutron	Nucleus	none	1.008665 amu
Proton	Nucleus	+1	1.007277 amu
Electron	Shells around nucleus	-1	0.0005486 amu

Nuclide	Element	Protons	Electrons	Neutrons
$^{1}_{1}H$	hydrogen	1	1	0
$^{10}_{5}B$	boron	5	5	5
$^{14}_{7}N$	nitrogen	7	7	7
$^{114}_{48}Cd$	cadmium	48	48	66
$^{239}_{94}Pu$	plutonium	94	94	145

Forces Acting in the Nucleus

Three forces in the nucleus

- **Gravitational force (between any two objects that have mass)**
  - The GF between two protons that are separated by a distance of 10^{-20} m is about 10^{-24} Newtons

$$F_g = \frac{G\ m_1\ m_2}{r^2}$$

- **Electrostatic force (between charged particles)**
  - The EF between two protons that are separated by a distance of 10^{-20} m is about 10^{12} Newtons

$$F_e = \frac{K\ Q_1\ Q_2}{r^2}$$

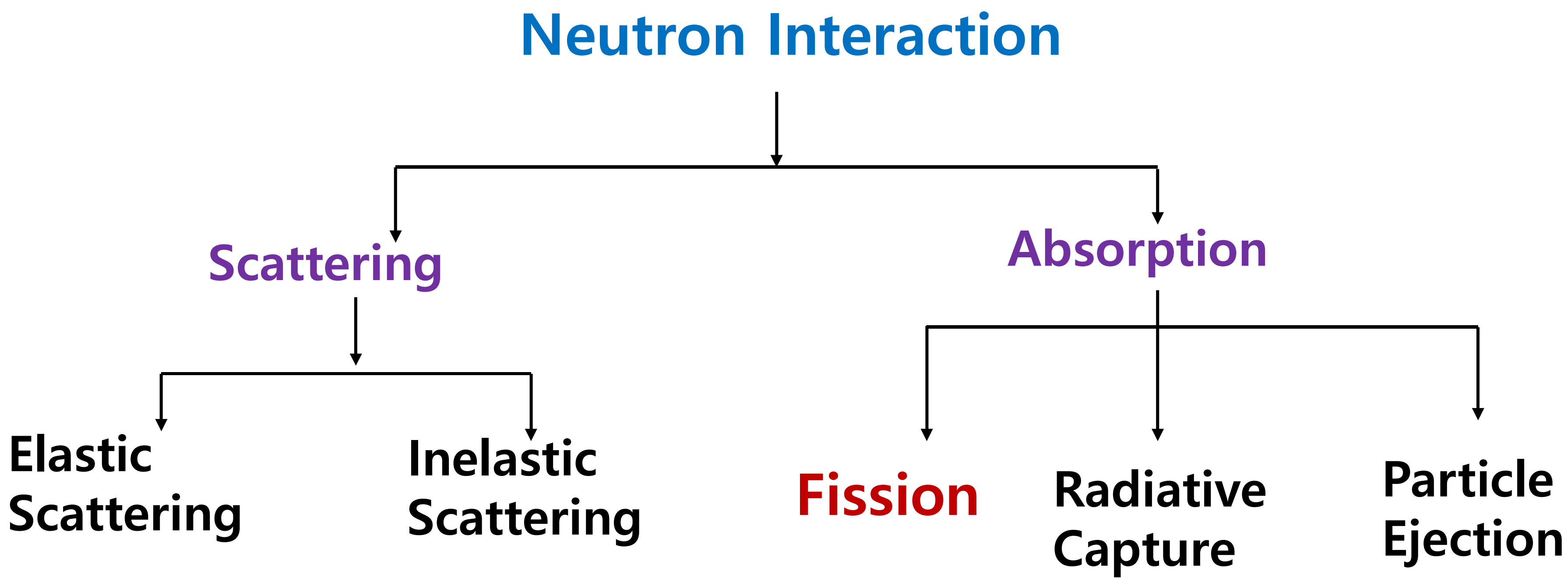
❖ If only the above two forces existed in the nucleus, then it would be impossible to have stable nuclei composed of protons and neutrons (Elec. Force(repulsive) >>>> Grav. Force)

➔ There must be another attractive force acting within the nucleus

- **Nuclear force (between pairs of any nucleons in the nucleus)**
  - A strong attractive force that is independent of charge.
  - It acts equally only between pairs of neutrons, pairs of protons, or a neutron and a proton.
  - The NF has a very short range; it acts only over distances approximately equal to the diameter of the nucleus(10^{-13} cm).

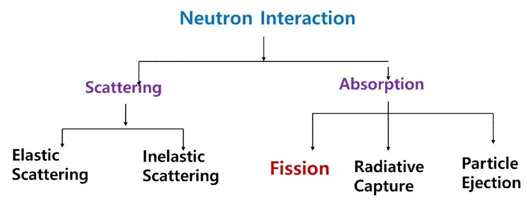
➔ So, The attractive NF between all nucleons drops off with distance much faster than the repulsive EF between protons. (Range of force : EF >>> NF)

Neutron Interactions



Scattering

- In an ***elastic scattering(ES)*** reaction between a neutron and a target nucleus,
  - there is no energy transferred into nuclear excitation. Momentum and kinetic energy of the "system" are conserved although there is usually some transfer of kinetic energy from the neutron to the target nucleus.
  - Two ways : Resonance ES & Potential ES
    - Resonance(unusual) : forming a compound nucleus, followed by the re-emission of a neutron
    - Potential(usual) : the neutron does not actually touch the nucleus and a compound nucleus is not formed
- In ***inelastic scattering***, the incident neutron is absorbed by the target nucleus, forming a compound nucleus.
  - The compound nucleus will then emit a neutron of lower kinetic energy which leaves the original nucleus in an excited state.
  - The nucleus will usually, by one or more gamma emissions, emit this excess energy to reach its ground state.



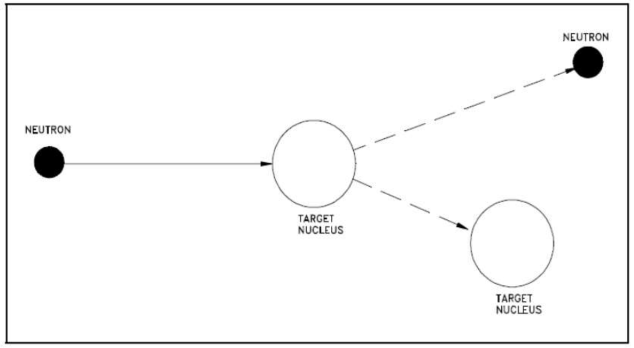


Figure 16 Elastic Scattering

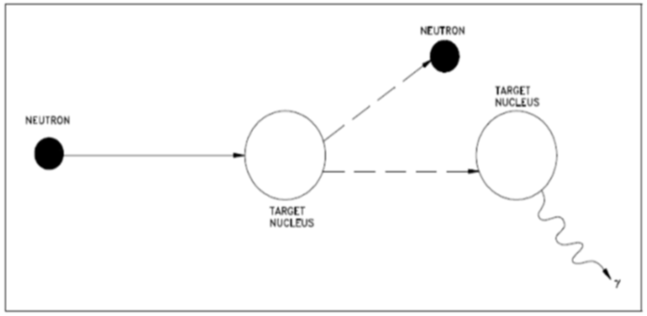
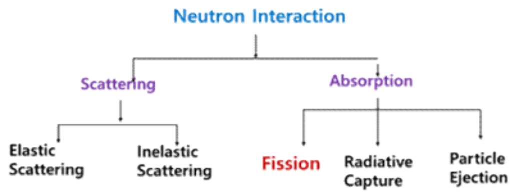


Figure 17 Inelastic Scattering

Absorption Reaction(1)



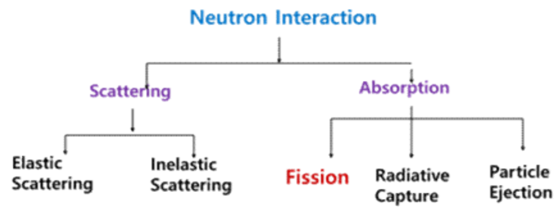
- **Radiative Capture**
  - In radiative capture the incident neutron enters the target nucleus forming a compound nucleus. The compound nucleus then decays to its ground state by gamma emission. An example of a radiative capture reaction is shown below.

$$ {}^{1}_{0}n + {}^{238}_{92}U \rightarrow \left({}^{239}_{92}U \right)^{*} \rightarrow {}^{239}_{92}U + {}^{0}_{0}\gamma $$

- **Particle Ejection**
  - In a particle ejection reaction the incident particle enters the target nucleus forming a compound nucleus. The newly formed compound nucleus has been excited to a high enough energy level to cause it to eject a new particle(alpha or proton) while the incident neutron remains in the nucleus.

$$ {}^{1}_{0}n + {}^{10}_{5}B \rightarrow \left({}^{11}_{5}B \right)^{*} \rightarrow {}^{7}_{3}Li + {}^{4}_{2}\alpha $$

Absorption Reaction(2)



- **Fission**
  - In the fission reaction the incident neutron enters the heavy target nucleus, forming a excited compound nucleus (by the change in binding energy(BE) + kinetic E of the incident neutron)

$$ {}^{1}_{0}n + {}^{235}_{92}U \rightarrow \left({}^{236}_{92}U\right)^{*} \rightarrow {}^{140}_{55}Cs + {}^{93}_{37}Rb + 3\left({}^{1}_{0}n\right) $$

  - High energy level($E_{exc} > E_{crit}$) can make the nucleus split (fission) into two large fragments plus some neutrons.
    - Excitation E(E_{exc}) : the amount of energy a nucleus has above its ground state
    - **Critical E**(E_{crit}) : the minimum excitation energy required for fission to occur
  - The splitting of the nucleus into two separate nuclei is accompanied by the release of a large amount of energy released in the form of radiation and fragment kinetic energy.

Binding Energy

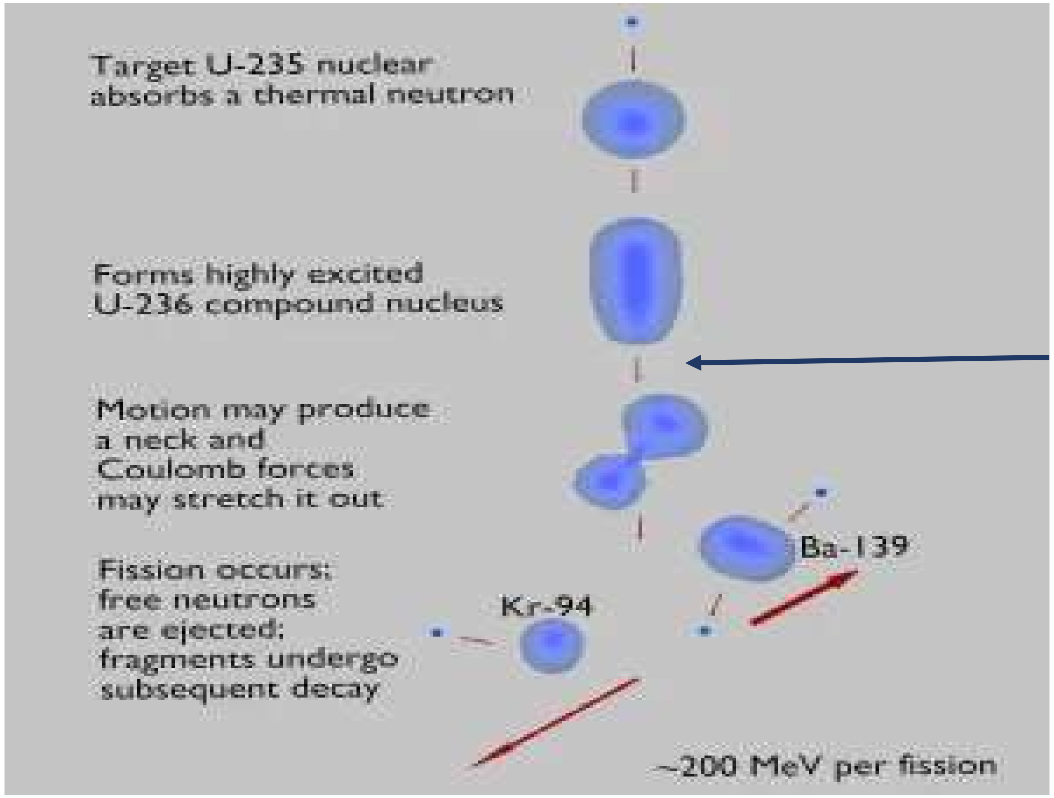
- The loss in mass, or mass defect, is due to the conversion of mass to **binding energy(BE)** when the nucleus is formed. (Einstein's **Theory of Relativity**)
- **BE** is defined as the amount of energy that must be supplied to a nucleus to completely separate its nuclear particles (nucleons).
  - It can also be understood as the amount of energy that would be released if the nucleus was formed from the separate particles.
- Therefore, **BE** is the energy equivalent to the **mass defect.**
  - Energy equivalent of 1 amu is as follows;
    - $E = mC^2 = 1(\text{amu}) \times C^2 (c = 2.998 \times 10^8$ m/sec$) = 931.5$ MeV (*1MeV = 1.6022×10^{-13} joules)
  - B.E of a last neutron

$$^{1}_{0}n + {}^{235}_{92}U \rightarrow \left({}^{236}_{92}U\right)^* \rightarrow {}^{140}_{55}Cs + {}^{93}_{37}Rb + 3\left({}^{1}_{0}n\right)$$

    - ① Mass of U-235 : 235.0439 amu, ② Mass of neutron : 1.008665 amu
    - ③ Mass of U-236 : 236.0456 amu,
    - Mass defect(①+②-③) = 0.006965 amu ➔ ~ 6.49 Mev (rough value)
    - Excitation Energy = 6.49Mev+ Kinetic energy of neutron

Fission : Liquid Drop Model(1)

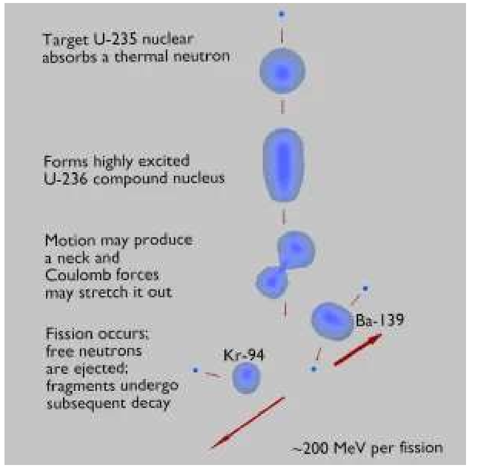
- **The characteristics of the nuclear force(NF)**
  - (a) very short range, with essentially no effect beyond nuclear dimensions(10^{-13}cm)
  - (b) stronger than the repulsive EFs within the nucleus
  - (c) independent of nucleon pairing
  - (d) saturable, that is, a nucleon can attract only a few of its nearest neighbors



($E_{exc} > E_{crit}$)

- **GF** : Gravitational Force
- **NF** : Nuclear Force (⇨⇦)
- **EF**: Electrostatic (⇔)

Fission : Liquid Drop Model(2)



● Fission process explained by the liquid drop model

① When a neutron is absorbed by the target nucleus, a compound nucleus is formed. The compound nucleus temporarily contains all the charge and mass involved in the reaction and exists in an excited state.

- The **excitation energy($E_{exc.}$)** added to the compound nucleus is equal to **BE** (contributed by the neutron) + **KE** (possessed by the neutron).

① Excitation energy(E_{exc})thus imparted to the compound nucleus, which may cause it to oscillate and become distorted.

② If the $E_{exc.}$ is greater than a certain **Critical Energy($E_{cri.}$),** the oscillations may cause the compound nucleus to become dumbbell-shaped.

- When this happens, the attractive Nuclear Forces(short-range) in the neck area are small due to saturation, while the repulsive Electrostatic Forces (long-range) are only slightly less than before.

③ When the repulsive EFs exceed the attractive NFs, **nuclear fission occurs**

Fissile and Fissionable Materials

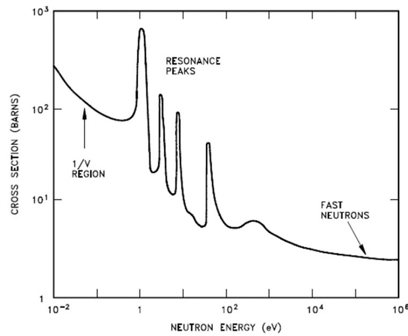
- A ***fissile*** material is composed of nuclides for which fission is possible with neutrons that have zero kinetic energy(Thermal energy).
  - Fission is possible in these materials with thermal neutrons, since the change in binding energy supplied by the neutron addition alone is high enough to exceed the critical energy.
  (U-233, U-235, Pu-239)

TABLE 4

Critical Energies Compared to Binding Energy of Last Neutron

Target Nucleus	Critical Energy E_{crit}	Binding Energy of Last Neutron BE_n	BE_n - E_{crit}
$^{232}_{90}Th$	7.5 MeV	5.4 MeV	-2.1 MeV
$^{238}_{92}U$	7.0 MeV	5.5 MeV	-1.5 MeV
$^{235}_{92}U$	6.5 MeV	6.8 MeV	+0.3 MeV
$^{233}_{92}U$	6.0 MeV	7.0 MeV	+1.0 MeV
$^{239}_{94}Pu$	5.0 MeV	6.6 MeV	+1.6 MeV

- A ***fissionable*** material is composed of nuclides for which fission with neutrons is possible.
  - A material for which fission caused by neutron absorption is possible provided the kinetic energy added with the binding energy is greater than the critical energy.
  (Th-232, U-238, Pu-240)
  - All fissile nuclides fall into this category as well.



- ***Fertile*** materials are materials
  - that can undergo transmutation to become fissile materials
    (Th-232➔ U-233, U-238➔Pu-239)

Calculation of Fission Energy

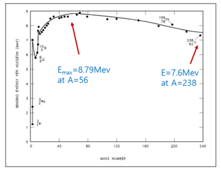


Figure 20 Binding Energy per Nucleon vs. Mass Number

- Let's assume the fission fragments are Rb-93 and Ce-140, then the fission energy? (see Ref.2, p.59)

$$ {}^{1}_{0}n + {}^{235}_{92}U \rightarrow \left({}^{236}_{92}U\right)^{*} \rightarrow {}^{140}_{55}Cs + {}^{93}_{37}Rb + 3\left({}^{1}_{0}n\right) $$

1) Using the BE

➔ (809+1176)-1786 = 199 MeV

2) Using the Mass Defect (more accurate)

- Mass of Reactant(U^{235}+n) : 236.052589 amu
- Mass of Products (Rb^{93}+Cs^{140}+3n) : 235.85208 amu
- Mass diff : 0.200509amu ~ 186.8MeV (instantaneous E)

Binding Energies Calculated from Binding Energy per Nucleon Curve

Nuclide	B.E. per Nucleon (BE/A)	Mass Number (A)	Binding Energy (BE/A) x (A)
${}^{93}_{37}Rb$	8.7 MeV	93	809 MeV
${}^{140}_{55}Cs$	8.4 MeV	140	1176 MeV
${}^{235}_{92}U$	7.6 MeV	235	1786 MeV

- Total released per fission is about 200 MeV for U-235
  - But, about 7% (13 MeV) is released at some time after the instant of fission.
  - When a reactor is shut down, fissions essentially cease, but energy is still being released from the decay of fission products (decay heat).

Instantaneous Energy from Fission

Kinetic Energy of Fission Products	167 Mev
Energy of Fission Neutrons	5 MeV
Instantaneous Gamma-ray Energy	5 MeV
Capture Gamma-ray Energy	10 MeV
Total Instantaneous Energy	187 MeV

Delayed Energy from Fission

Beta Particles From Fission Products	7 Mev
Gamma-rays from Fission Products	6 MeV
Neutrinos	10 MeV
Total Delayed Energy	23 MeV

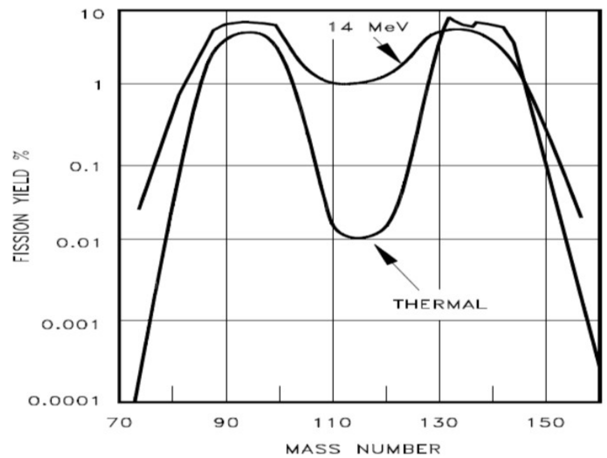
** Neutrino energy is not absorbed in the reactor

Fission Fragments Distribution

- The fission reaction of U-235, is represented as follows;

  ${}_{92}U^{235} + {}_{0}N^{1} \rightarrow {}_{92}U^{236*} \rightarrow {}_{z}X^{A} + {}_{z}Y^{A} + 2.43\ {}_{0}n^{1}$ + Energy

  (About 80% of all U-236 atoms will fission : High Prob. of fission)
- The resultant fission fragments(FPs) have various masses.
  - The most probable pair of FPs for the thermal fission of U-235 have masses of about 95 and 140 which are highly likely numbers
  - More than 40 different FP pairs that result from fission



U-235 Fission fragments distribution

- The FPs are radioactive because they have neutron to proton ratios that are larger than those for stable nuclei.
  - To be stable nuclei, following transmutations are occurred.

    1) Direct neutron emission
       (< 1% of neutron-rich nuclei decay)

    2) Beta emissions along with one or more gamma rays

    ➔ These emissions interact with the surrounding fuel and reactor materials, thereby transferring significant energy in the form of heat.

    (Source of Decay Heat)

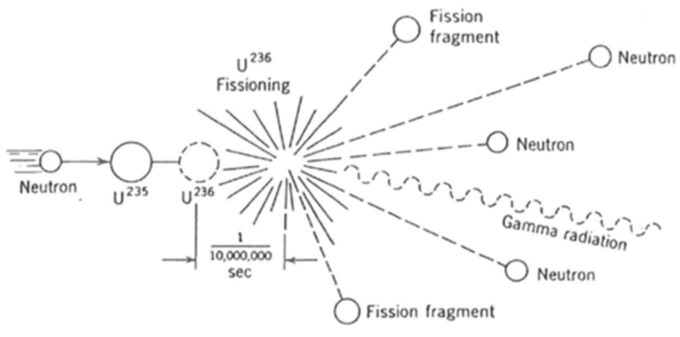


FIGURE 1-1 A fission reaction.

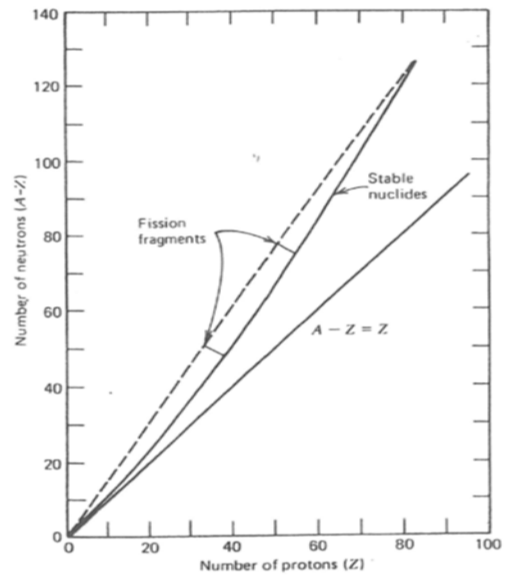


FIGURE 1-2 Fission fragment instability.

Figures excerpted from the Book - Nuclear Power Reactor Safety (E.Lewis)

Estimation of Decay Heat

- One example of the more than 40 different FP pairs that result from fission
  - Xe^{140} and Sr^{94} transmute to stable daughter nuclei with emitting Beta and Gamma rays **(Decay Chains)**
- Each FP generated from decay chains is characterized by a yield and half-life
- The decay heat produced by a reactor shutdown from full power is initially equivalent to about 6 to 7% of the thermal rating of the reactor.
  - The amount of decay heat present in the reactor is dependent on three factors.
    - The pre-shutdown power level
    - How long the reactor operated
    - The amount of time since reactor shutdown

Ex) Thermal energy of Full Power in OPR1000
~ 3,300 MWt

- Decay heat in 1 month after Shutdown

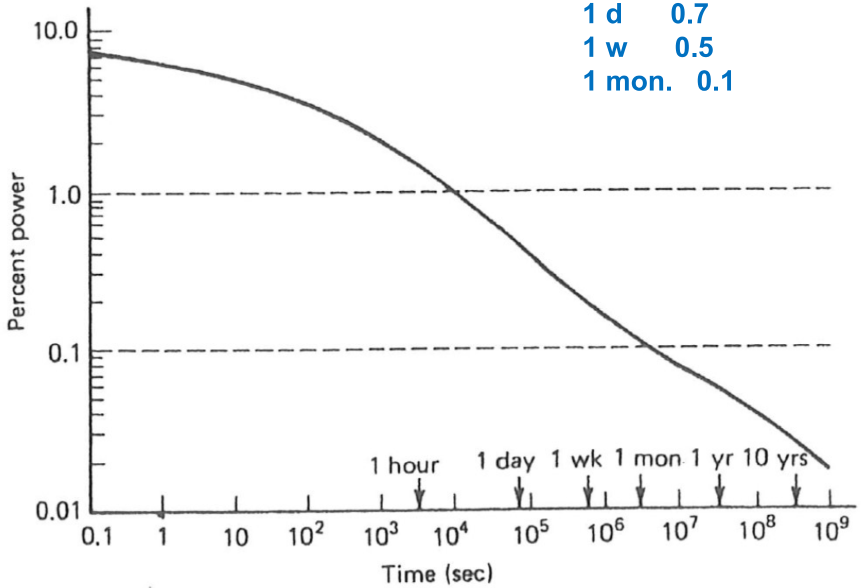
✓ 3,300 x 0.001(=0.1%) = 3.3 MWt = 3,300 KWt

$$ {}_0n^1 + {}_{92}U^{235} \rightarrow {}_{54}X^{140} + {}_{38}Sr^{94} + 2\,{}_0n^1 $$

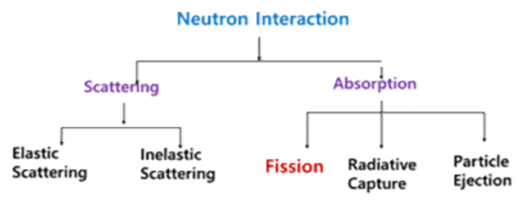
$$ {}_{54}Xe^{140} \rightarrow {}_{55}Cs^{140} + \beta + \gamma \rightarrow {}_{56}Ba^{140} + \beta + \gamma \rightarrow {}_{57}La^{140} + \beta + \gamma \rightarrow {}_{58}Ce^{140} + \beta + \gamma, $$

$$ {}_{38}Sr^{94} \rightarrow {}_{39}Y^{94} + \beta + \gamma \rightarrow {}_{40}Zr^{94} + \beta + \gamma. $$

1 s	6.0% of Full Power
1 min.	4.5
1 h	1.6
1 d	0.7
1 w	0.5
1 mon.	0.1



Nuclear Cross-Sections



- **Neutron Reactions**
  - Scattering : a neutron transfers some of its energy to the nucleus
    - Elastic scattering : a neutron bounces off a nucleus
    - Inelastic scattering : a neutron penetrates the nucleus and form an unstable compound nucleus, and then the neutron is ejected.
  - Neutron Absorption : a nucleus keeps the neutron after the collision
    - Then the captured neutron can cause the nucleus to fission

- **Microscopic Cross Section(σ)** : Unit is the barn(1 barn = 10^{-24}cm^2.)
  - The **probability** of a particular reaction occurring between a neutron and a nucleus.
  - The *microscopic cross section* may also be regarded as the effective area the nucleus presents to the neutron for the particular reaction. The larger the effective area, the greater the probability for reaction.
  - Total Microscopic Cross Section (σ_T)
    - $\sigma_T = \sigma_s + \sigma_a$, $\sigma_s = \sigma_{se} + \sigma_{si}$, $\sigma_a = \sigma_f + \sigma_c + (\sigma_{pe})$, $\boldsymbol{\sigma_T = \sigma_f + \sigma_c + \sigma_{se} + \sigma_{si}}$

- **Macroscopic Cross Section(Σ)** :
  - Whether a neutron will interact with a certain volume of material depends not only on the microscopic cross section of the individual nuclei but also on the number of nuclei within that volume. Therefore, it is necessary to define another kind of cross section known as the macroscopic cross section.
  - The *macroscopic cross section* is the probability of a given reaction occurring per unit travel of the neutron.

    $\Sigma = N\sigma$ where :

    Σ = macroscopic cross section (cm^{-1}), N= atom density of material (atoms/cm^3)

- **σ** represents the effective target area that a single nucleus presents to a bombarding particle.
- **Σ** represents the effective target area that is presented by all of the nuclei contained in 1 cm^3 of the material.

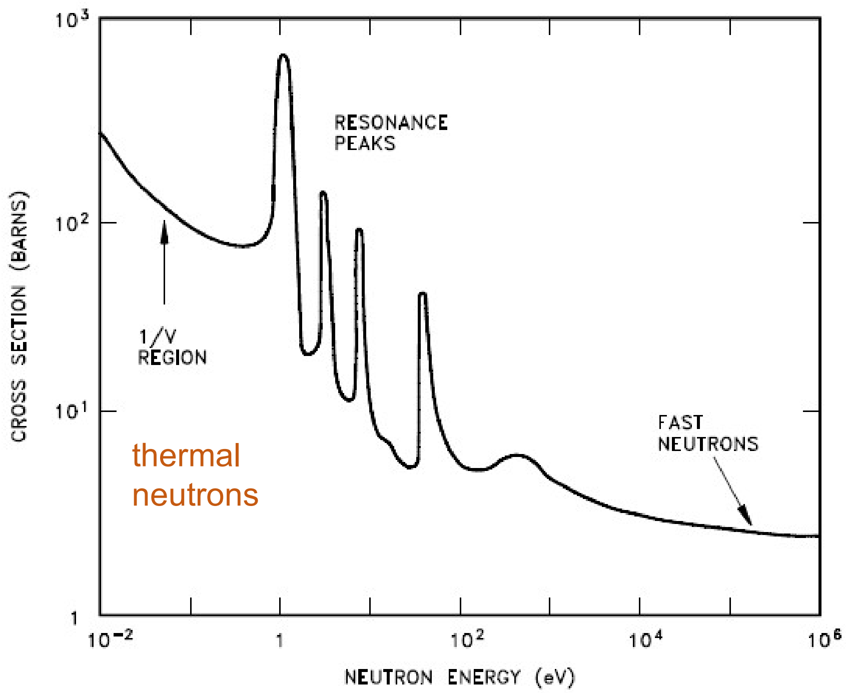


Fig. Typical Neutron Absorption Cross Section(σ_a) vs. Neutron Energy

NUCLIDE	REACTIONS	E_n	σ(barns)
$_{92}U^{235}$	Absorption	.025 ev	690
	Scattering	.025 ev	10
	Fission	.025 ev	580
	Capture	.025 ev	110
	Alpha emission	.025 ev	0
$_5B^{10}$	Absorption	.025 ev	4010
	Capture	.025 ev	0
	Alpha emission	.025 ev	4010
	Scattering	.025 ev	5
	Absorption	1 ev	665
	Absorption	10 ev	200
	Absorption	100 ev	66.5
	Absorption	1000 ev	20.0
	Absorption	10,000 ev	6.65

$${}^{1}_{0}n + {}^{10}_{5}B \rightarrow \left({}^{11}_{5}B\right)^{*} \rightarrow {}^{7}_{3}Li + {}^{4}_{2}\alpha$$

Table. Microscopic Cross Section Values

- **1/v region** : The cross section decreases steadily with increasing neutron energy in a low energy region, which includes the thermal range (E < 1 eV).
- **Resonance region** : The cross sections rise sharply to high values called "resonance peaks" for neutrons of certain energies, and then fall again.
- **Fast neutron region** : For higher neutron energies, the absorption cross section steadily decreases as the energy of the neutron increases.

Neutron Moderation(1)

● Neutron Thermalization

- Fission neutrons are produced at an average energy level of 2 MeV and immediately begin to slow down as the result of numerous scattering reactions with a variety of target nuclei.
- Thermalization is the process of reducing the energy level of a neutron **from** the energy level(~2Mev) at which it is produced **to** an energy level in the thermal energy range (0.025 eV at 20(C^0)). Neutrons in the specific region (< 1 eV) are designated thermal neutrons.
- The process ➔ Thermalization, Slowing down, or Moderation. The material ➔ Moderator

● Candidate of Good Moderator

- Reduces the speed of neutrons in a small number of collisions, but does not absorb them to any great extent.
- Less collisions ➔ Less leakage from the reactor core, and Less resonance absorptions in non-fuel materials.
- The ideal moderating material (moderator) should have the following nuclear properties.
  - **large energy loss per collision, large σ_s , small σ_a**

● Average logarithmic energy decrement(ξ)

- $\xi = \ln E_i - \ln E_f = 2/(A+2/3)$ (Approximately) where, A : Moderator's mass number
  - ξ does not depend upon the initial neutron energy, and it is a constant for each type of material.
  - Relatively accurate for mass numbers(A) greater than 10, but for some low values of A it may be in much error by over 3%. (H=1, H_2O : 0.972, D_2O :0.510, Carbon : 0.158, U-238 : 0.00838)

Neutron Moderation(2)

- **Macroscopic Slowing Down Power(MSDP)**
  - Although the logarithmic energy decrement(ξ) is a convenient measure of the ability of a material to slow neutrons, it does not measure all necessary properties of a moderator. A better measure of the capabilities of a material is the macroscopic slowing down power.
  - MSDP indicates how rapidly a neutron will slow down in the material in question,
    MSDP = $\xi \Sigma_s$ (macroscopic cross section for scattering)

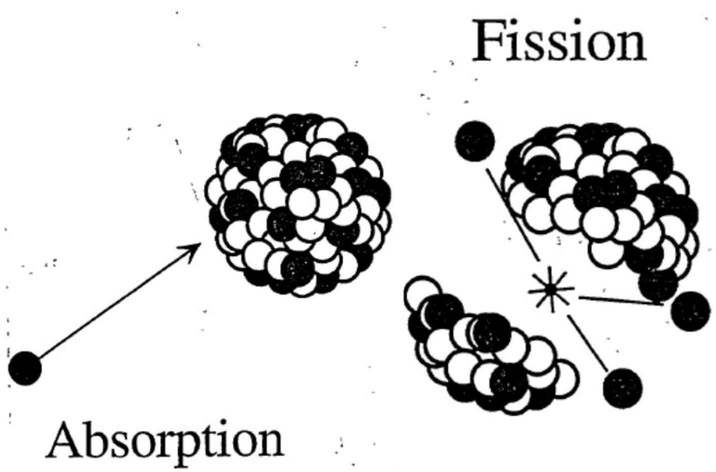
- **Moderating Ratio (MR)**
  - But, MSDP still does not fully explain the effectiveness of the material as a moderator.
  - An element such as boron has a relatively high logarithmic energy decrement and a good slowing down power, but it is a poor moderator because of its high probability of absorbing neutrons.(particle ejection)
  - The most complete measure of the effectiveness of a moderator is the **moderating ratio.**
  - **MR = $\xi \Sigma_s / \Sigma_a$** (ratio of the macroscopic slowing down power to the macroscopic cross section for absorption)
  - **The higher the moderating ratio, the more effectively the material performs as a moderator.**

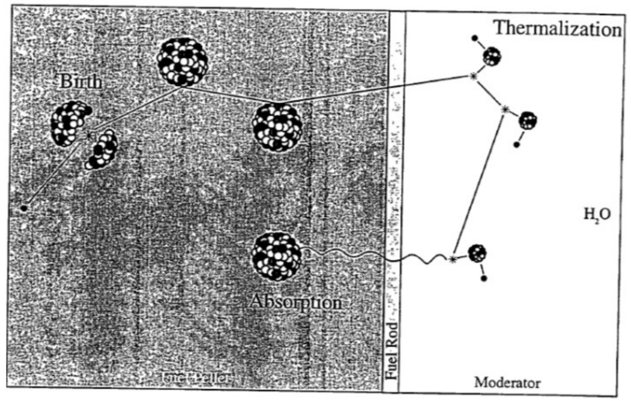
Moderating Properties of Materials

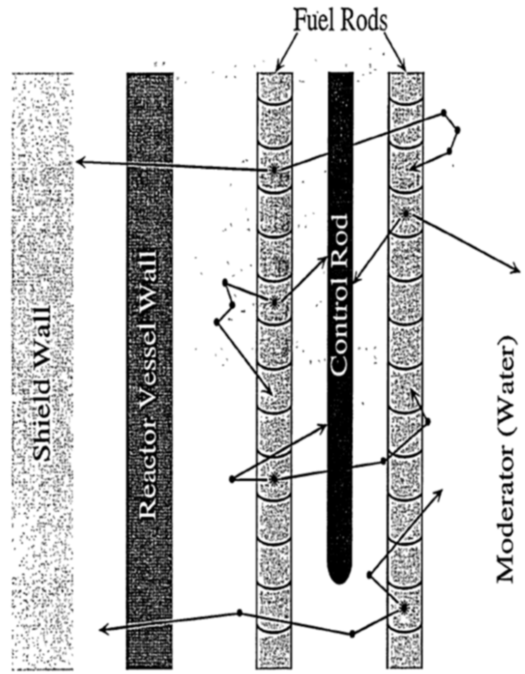
Material	ξ	Number of Collisions to Thermalize	Macroscopic Slowing Down Power	Moderating Ratio	σ_a
H_2O	0.927	19	1.425	62	0.664
D_2O	0.510	35	0.177	4830	0.00133
Helium	0.427	42	9×10^{-6}	51	
Beryllium	0.207	86	0.154	126	
Boron	0.171	105	0.092	0.00086	4010
Carbon	0.158	114	0.083	216	0.0034

D_2O and Carbon : Good Moderator?

Neutron Life Cycle in the Core







Absorption

Fission & 2~3 Fast neutrons

N Generation

- Moderation→ (Leakage)→ Absorption→ **Fission + n**
- Leakage to outside of RV→ Extinguishment
- Fast neutron absorption to U → **Fission +n** (low proba.)
- Absorption to Non-fuel material → Extinguishment

(N+1) Generation

Multiplication Factor(k) = No. of neutrons in (N+1)G/ (N)G

Neutron Life Cycle(1)

- Effective Multiplication Factor(K_{eff})
  - The ratio of (neutron production from fission in in one generation)
    to (neutron absorption + neutron leakage in the preceding generation)
  - It is a significant parameter of the nuclear chain reaction because its value determines the rate of neutron level multiplication.
    - If $K_{eff} > 1$, the reactor is supercritical; the neutron population, the fission rate, and energy production are increasing exponentially.
    - If $K_{eff} = 1$, the reactor is critical; the neutron population is constant, as is the fission rate and the energy production. The nuclear chain reaction is sustained and controlled.
    - If $K_{eff} < 1$, the reactor is subcritical; the neutron population, the fission rate, and the energy production are decreasing exponentially.
  - The initial value of K_{eff} is affected by the material composition and the geometry of the reactor core.

- Under normal operating conditions, a power reactor is kept critical; i.e., the equilibrium value of K_{eff} is 1.0 all the time.
  - Subcriticality leads eventually to shutdown of the reactor.
  - Supercritical operation is necessary during startup and power increase, but care must be taken not to let K_{eff} become too large or else the reactor will be difficult to control.

Neutron Life Cycle(2)

- Six Factor Formula for K_{eff}

- To understand how K_{eff} is affected by changes in core parameters, a six factor formula is used to illustrate the factors that must be considered when deriving K_{eff}.

$$K_{eff} = \varepsilon \cdot L_f \cdot p \cdot L_{th} \cdot f \cdot \eta$$

where

- K_{eff} = effective multiplication factor
- ε = fast fission factor
- L_f = fast non-leakage probability
- p = resonance escape probability
- L_{th} = thermal non-leakage probability
- f = thermal utilization factor
- η = reproduction factor

$$\varepsilon = \frac{\text{number of fast neutrons produced by all fissions}}{\text{number of fast neutrons produced by thermal fissions}}$$

$$\mathcal{L}_f = \frac{\text{number of fast neutrons that do not leak from reactor}}{\text{number of fast neutrons produced by all fissions}}$$

$$p = \frac{\text{number of neutrons that reach thermal energy}}{\text{number of fast neutrons that start to slow down}}$$

$$\mathcal{L}_t = \frac{\text{number of thermal neutrons that do not leak from reactor}}{\text{number of neutrons that reach thermal energies}}$$

$$f = \frac{\text{number of thermal neutrons absorbed in the fuel}}{\text{number of thermal neutrons absorbed in all reactor materials}}$$

$$\eta = \frac{\text{number of fast neutrons produced by thermal fission}}{\text{number of thermal neutrons absorbed in the fuel}}$$

Neutron Life Cycle(3)

- The thermal utilization factor(f) can be calculated from the macroscopic cross section for absorption of reactor materials. The thermal utilization factor is effected by the enrichment of uranium-235, the amount of neutron poisons, and the moderator-to-fuel ratio.

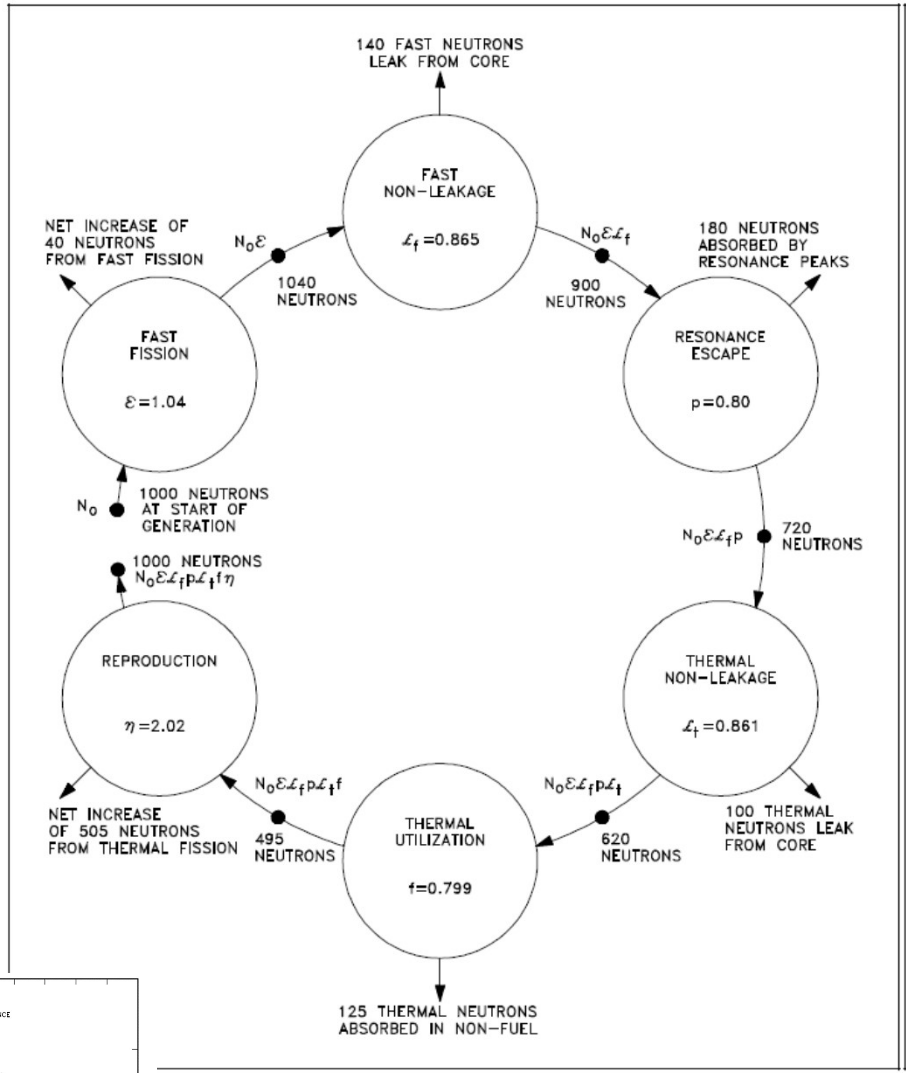
$$f = \frac{\Sigma_a^U}{\Sigma_a^U + \Sigma_a^m + \Sigma_a^p}$$

- The reproduction factor(η) can be calculated based on the characteristics of the reactor fuel. The reproduction factor is effected by the enrichment of uranium-235.

$$\eta = \frac{N^{U\text{-}235}\,\sigma_f^{U\text{-}235}\,\nu^{U\text{-}235}}{N^{U\text{-}235}\,\sigma_a^{U\text{-}235} + N^{U\text{-}238}\,\sigma_a^{U\text{-}238}}$$

- The resonance escape probability(p) is effected by the enrichment of uranium-235, the temperature of the fuel, and the temperature of the moderator.

- An increase in moderator temperature will have the following effects
  - Increase the thermal utilization
  - Decrease resonance escape probability
  - Decrease fast non-leakage probability
  - Decrease thermal non-leakage probability



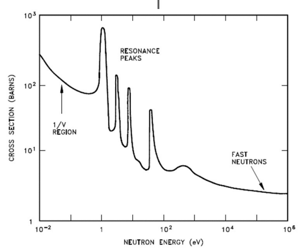


Fig. Neutron Life Cycle with k_{eff} = 1

Reactivity and Reactivity Coefficients

● Reactivity(ρ)

- The measure of the fractional change in neutron population per generation in a reactor and is **a measure of reactor's departure from criticality**.
- The numerical change in neutron population is ($N_o \cdot k_{eff}$ - N_o), where N_o is neutrons in the preceding generation,

$$\frac{N_o k_{eff} - N_o}{N_o k_{eff}} \Rightarrow \rho = \frac{K_{eff} - 1}{K_{eff}} = \frac{\Delta K}{K}$$

- Reactivity(ρ) may be positive, zero, or negative, depending upon the value of k_{eff}.
  - Critical reactor (k_{eff} = 1.0, ρ = 0), Supercritical (k_{eff} > 1.0, ρ = (+)), Subcritical (k_{eff} <1.0, ρ = (-))
    ex) If k_{eff} = 1.002. ρ = (+)0.001996, If k_{eff} = 0.998. ρ = (-) 0.0020
  - The larger the reactivity in the reactor core, the further the reactor is from criticality.

● Reactivity Coefficients

- Temperature Coefficient : Defined as the change in reactivity per degree change in temperature
  → Moderator Temperature Coefficient (MTC)
  → Fuel Temperature Coefficient (FTC)
- Moderator Void Coefficient (VC)
- Pressure Coefficient (PC) : BWR

Moderator Temperature Coefficient(1)

- Moderator temperature increases, the moderator density decreases
  - Fewer moderator molecules are available for slowing down and thermalizing neutrons.
  - This causes neutrons to remain at higher energies longer and travel farther

  ➔ resulting in increased leakage and more non productive capture.
- The effect of a moderator temperature increase on K_{eff}
  - As moderator temperature increases,
    - Increase the thermal utilization factor (f)
    - Decrease resonance escape prob.(p), fast non-leakage prob.(L_f) and thermal non-leakage prob.(L_{th})

    ➔ Generally, resulting in a decrease in K_{eff}

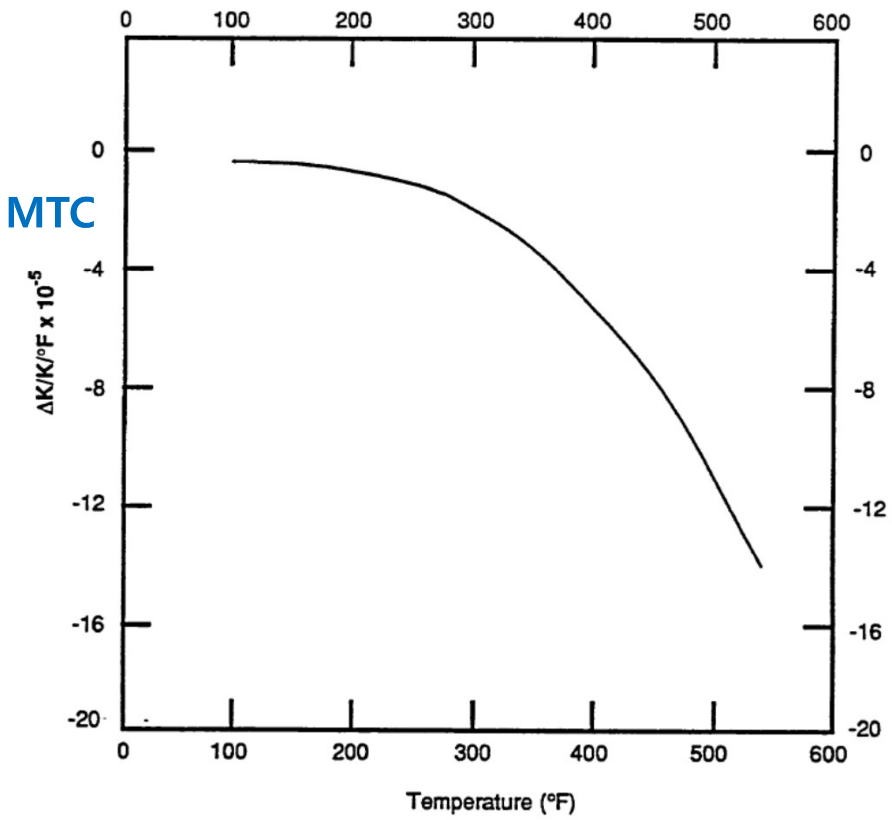
    But, not always. It depend on the neutron poison concentration in the core
- As moderator temperature increases, MTC becomes more negative.
  - because a 1F° increase at a high moderator temperature results in a larger density reduction (larger reduction in molecules available to thermalize neutrons) than a 1F° increase at a lower moderator temp.

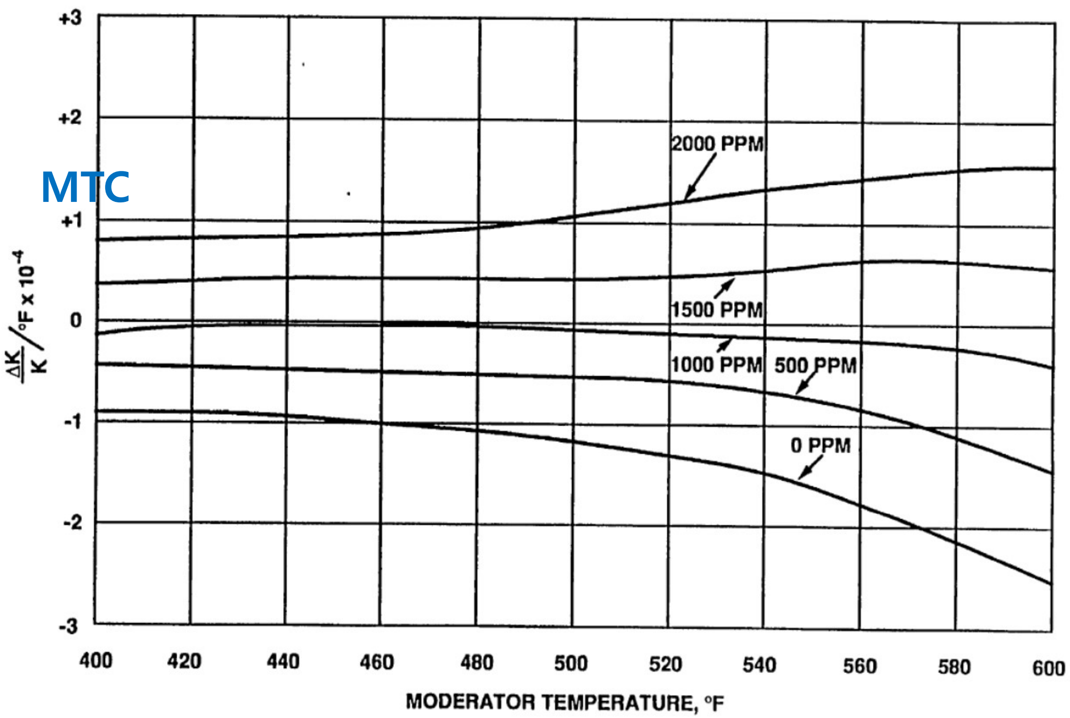
- **MTC should always be negative to contribute to inherent reactor stability.**
  - This means negative reactivity must be inserted when the temperature of the moderator increases.
  - Consider what would happen if the MTC is positive.

  ➔ As the moderator temperature increases, with positive MTC more positive reactivity would be added

  ➔ Reactor power would be self-escalating with a positive MTC

Moderator Temperature Coefficient(2)





MTC behaviors by the change of temperature & neutron poison(Boron) concentration

Fuel Temperature Coefficient(1)

- The **FTC** is the change in reactivity per degree change in fuel temperature. (Fuel Doppler Reactivity Coefficient)
  - In thermal reactor with typical low enrichment and light water moderated, FTC is negative and is the result of the doppler effect, also called doppler broadening.
- The doppler effect phenomenon is caused by an apparent broadening of the resonances due to thermal motion of nuclei as illustrated in the following Figure.
  - Because the temperature of a target nucleus(mainly U-238 and Pu-240) causes it to be in motion, there is a band of neutron velocities or energies where resonance absorption could occur.
  - As the resonance band broadens due to a temperature increase, more neutrons will enter the band as they are slowing down. With more neutrons in the band, more of them will be absorbed. This increased absorption will result in **a negative reactivity addition.**

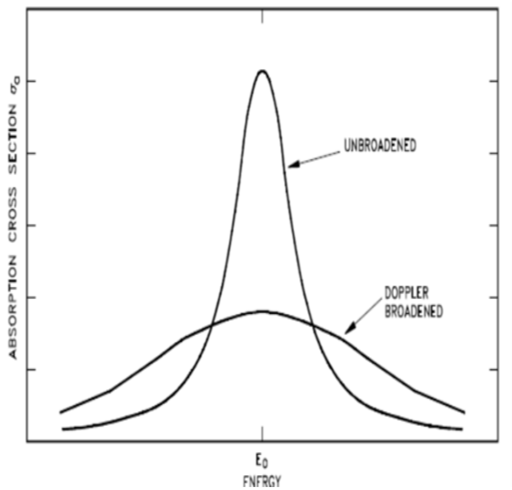


Fig. Effect of Fuel Temperature on Resonance Absorption Peaks

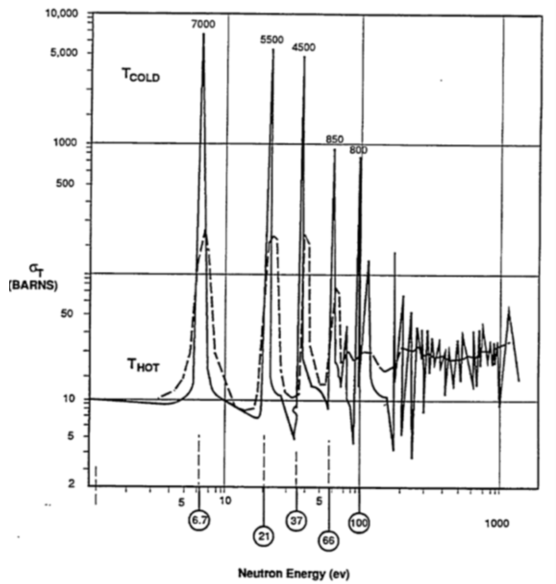


Fig. U-238 Cross Section Curve

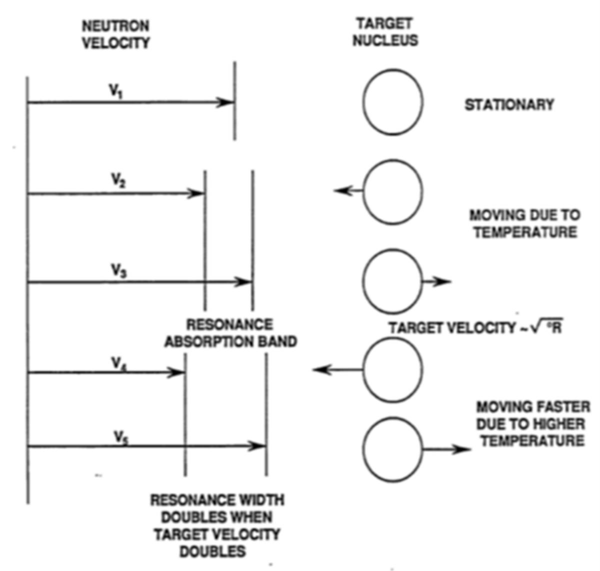
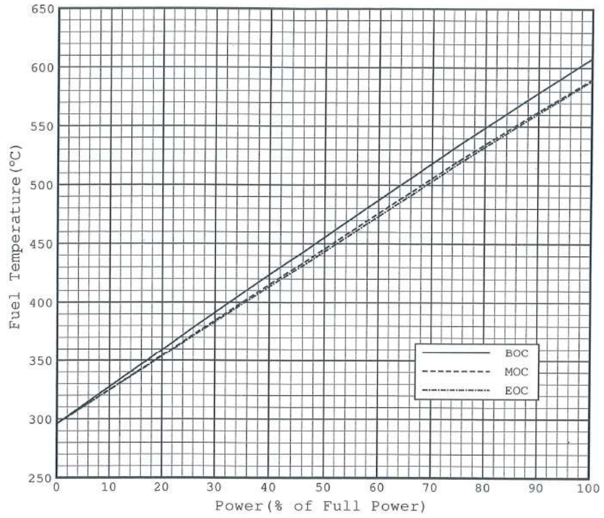


Fig. Resonance Capture

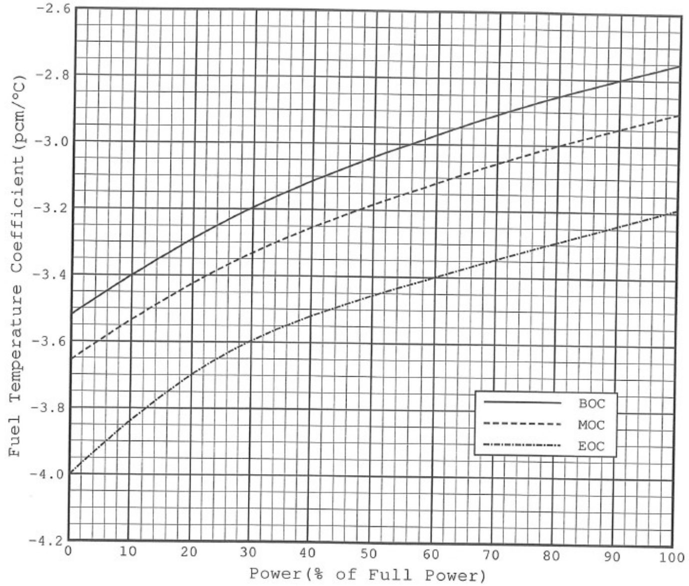
Fuel Temperature Coefficient(1)

- FTC is also called the "prompt" temperature coefficient
  - because an increase in reactor power causes an immediate change in fuel temperature.
    (The time for heat to be transferred to the moderator is measured in seconds)
  - A negative FTC is generally considered to be even more important than a negative MTC because fuel temperature immediately increase following an increase in reactor power.
  - ❖ In the event of a large positive reactivity insertion, the MTC cannot turn the power rise for several seconds, whereas the FTC starts adding negative reactivity immediately.

- As the core ages, FTC becomes more negative.
  - The dominant factor is the buildup of Pu-240, which is a strong resonant neutron absorber.
  - Accordingly, total resonance absorption increases, thereby FTC to become more negative.



Fuel Temp. vs. Power



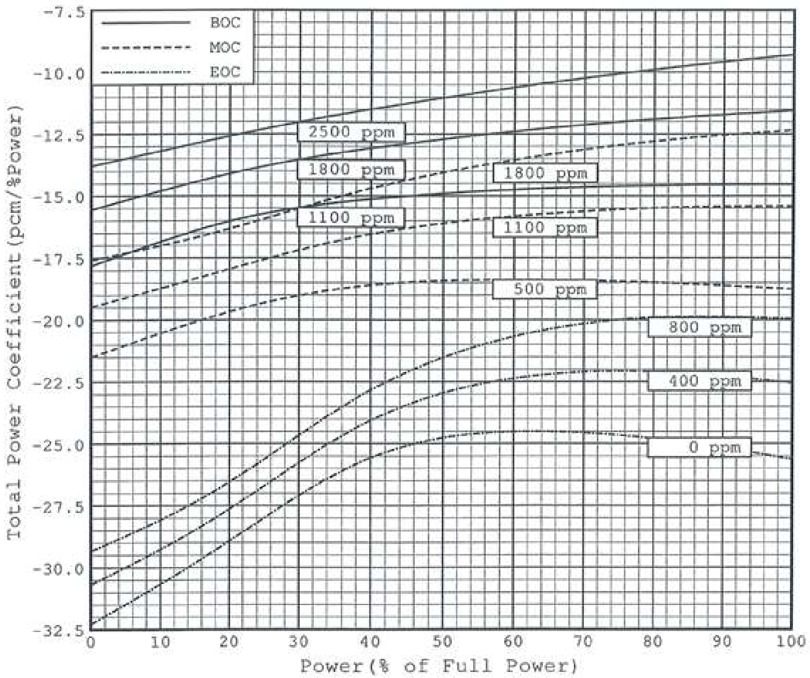
FTC vs. Power (Typical)

Power Coefficient

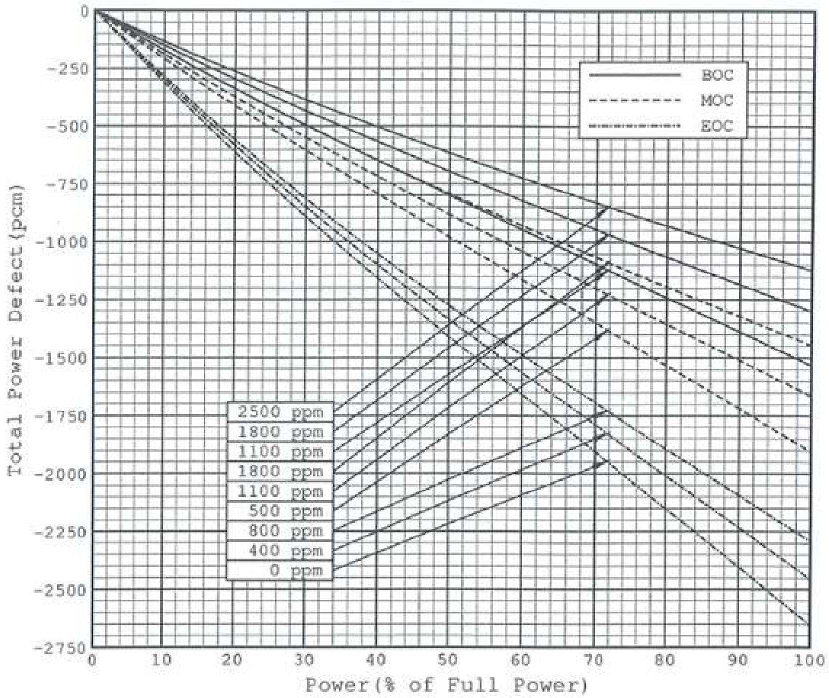
- PC is expressed as the change in reactivity per change in percent power and is **negative at all times** in core life.
  - A change in reactor power normally causes or is accompanied by a change in fuel temperature, moderator temperature, and/or moderator void fraction.
  - Therefore, it is frequently convenient to consider a **power coefficient** that combines the effects of the fuel temperature, moderator temperature, and void coefficients for changes in reactor power.

$$\approx -1 \times 10^{-4} \frac{\Delta K / K}{\%\ \text{power}}$$

- As power is increased, (-) reactivity will be added to the core because the power coefficient is negative.
  - Therefore, an equal amount of positive reactivity must be added to keep the reactor critical.
  - When power is decreased quickly, as it is after a trip or scram, the power coefficient adds ***positive* reactivity**, and negative reactivity insertion is required to make and keep the reactor subcritical
- The total amount of reactivity added by a power change is called **Power Defect**.



PC vs. Power



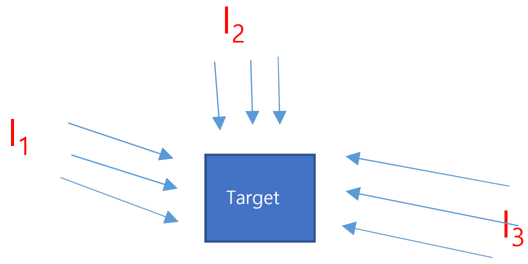
Power Defect vs. Power

Lecture Topics

- **History of Nuclear Fission**
  - Discovery of Nuclear Fission
  - Chicago Pile Experiment
- **Nuclear Physics**
  - Principle of Nuclear Fission
  - Criticality & Reactivity
  - Reactivity Coefficients
- **Neutron Control in Reactor**
  - Reactor Kinetics
  - Neutron Flux Shape Control
  - Xenon Poison

Neutron Flux and Fluence

- Neutron flux (Φ)
  - defined as a measure of the intensity of neutron radiation, determined by the rate of flow of neutrons.
  - Mathematically, this is the equation, $\Phi = n\,v$
    - Φ = neutron flux (neutrons/cm2-sec)
    - N = neutron density (neutrons/cm3)
    - v = neutron velocity (cm/sec)



- Intensity of Neutron
  - The directional beam intensity(I) is equal to the number of neutrons per unit area and time (neutrons/cm2-sec) falling on a surface perpendicular to the direction of the beam.
  - The neutron flux in a reactor as being comprised of many neutron beams traveling in various directions. Then, the neutron flux becomes the scalar sum of these directional flux intensities (added as numbers and not vectors), that is, $\Phi = I_1 + I_2 + I_3 + \ldots I_n$.
- Neutron Fluence
  - The time integral of the neutron flux, expressed as number of particles(neutrons) per cm^2
  - Widely used in measurement of material embrittlement and fuel burn-up

Neutron Classification

- ***Prompt neutrons***
  - The great majority (over 99%) of the neutrons produced in fission are released within about 10^{-14} seconds of the actual fission event. These are called prompt neutrons.
- ***Delayed neutrons***
  - A small portion of fission neutrons(<1%) are produced for some time after the fission process has taken place.
  - These delayed neutrons are emitted immediately following the first beta decay of a fission fragment known as a delayed neutron precursor. An example of a delayed neutron precursor is bromine-87, shown below.

$$ {}^{87}_{35}\mathrm{Br} \xrightarrow[55.9\ \mathrm{sec}]{\beta^-} {}^{87}_{36}\mathrm{Kr} \xrightarrow[\mathrm{instantaneous}]{n} {}^{86}_{36}\mathrm{Kr}_{stable} $$

❖ *Although delayed neutrons are a very small fraction of the total number of neutrons, they play an* ***extremely important role in the control of the reactor***

Delayed Neutron Fraction

- The table-2 lists the characteristics for the six precursor groups resulting from thermal fission of uranium-235.
  - Around 0.7% of neutrons are emitted by the decay of some unstable fission product nuclides which have half-lives of 0.2 ~ 55 sec.
  - The fraction of all neutrons that are produced by each of these precursors is called the delayed neutron fraction(β) for that precursor.

- The fraction of delayed neutrons produced varies depending on the predominant fissile nuclide in use.
  - The total delayed neutron fractions for the fissile nuclides of most interest are as follows:
  - U-233(0.0026), U-235 (0.0065),
  - U-238 (0.0148), Pu-239 (0.0021)

Table-1. Neutrons per fission

REACTOR FUEL	NEUTRONS / FISSION
U-235	2.43
Pu-239	2.90
U-233	2.50
U-238	2.30

Table-2 Delayed Neutron Fraction for U-235 & U-238

	GROUP	HALF LIFE $T_{1/2}$(SEC)	DECAY CONSTANT λ (SEC^{-1})	YIELD (NEUTRONS/FISSION)	FRACTION β
U^{235}	1	55.72	0.0124	0.00052	0.000215
	2	22.72	0.0305	0.00346	0.001424
	3	6.22	0.111	0.00310	0.001274
	4	2.30	0.301	0.00624	0.002568
	5	0.610	1.14	0.00182	0.000748
	6	0.230	3.01	0.00066	0.000273

TOTAL YIELD: 0.0158
TOTAL DELAYED NEUTRONS FRACTION (β): 0.0065

	GROUP	HALF LIFE $T_{1/2}$(SEC)	DECAY CONSTANT λ (SEC^{-1})	YIELD (NEUTRONS/FISSION)	FRACTION β
U^{238}	1	52.38	0.0132	0.00054	0.000192
	2	21.58	0.0321	0.00564	0.002028
	3	5.00	0.139	0.00667	0.002398
	4	1.93	0.358	0.01599	0.005742
	5	0.490	1.41	0.00927	0.003330
	6	0.172	4.02	0.00309	0.001110

TOTAL YIELD: 0.0412
TOTAL DELAYED NEUTRONS FRACTION (β): 0.0148

Effect of Delayed Neutrons on Reactor Control(1)

- U-233(0.0026), U-235 (0.0065),
- U-238 (0.0148), Pu-239 (0.0021).

- Power generation ratio by fissionable materials in the core
  - At BOL, 93% of the power generated in the core is by the U-235, and 7% is by U-238.
  - At EOL, assume 40% of power is generated by Pu-239, 7% by U-238 and 53% by U-235.
  - Then, the total β fraction at BOL and EOL as follows, respectively ;

$$\begin{aligned}\beta &= (.93)(\beta_{U\text{-}235}) + (.07)(\beta_{U\text{-}238}) \\ &= (.93)(.0065) + (.07)(.0148) \\ &= .006045 + .001036 \\ &= .0070 \text{ at BOL}\end{aligned}$$

$$\begin{aligned}\beta &= (.53)(\beta_{U\text{-}235}) + (.07)(\beta_{U\text{-}238}) + (.40)(\beta_{Pu\text{-}239}) \\ &= (.53)(.0065) + (.07)(.0148) + (.40)(.0021) \\ &= .003445 + .001036 + .00084 \\ &= .0053 \text{ at EOL.}\end{aligned}$$

- Neutron population change in a given period of time
  - Even though the fraction of delayed neutrons is very low, 0.7% of all the fission neutrons have a large effect on the operation of the reactor. The following equation can be used to calculate a change in neutron population over a given period of time:

$$N = N_0 e^{(K_{eff}-1)\, t / \bar{\ell}}$$

N = Number of neutrons after time t,
N_0 = Number of neutrons at t = 0,
t = time period (seconds), and
$\bar{\ell}$ = generation time .

Effect of Delayed Neutrons on Reactor Control(2)

- The generation time (mean life time, $\bar{\ell}$)
  - The average the time between the absorption of a neutron in the fuel and the absorption of the resulting fission neutron in some materials.
  - For a prompt neutron and delayed neutrons, the generation times are as follows.
  (assume that the average mean lifetime for delayed neutron precursors is about 10 seconds)

$$\bar{\ell}_p = 10^{-14} \text{ sec (absorption to fission/birth)} + 10^{-4} \text{ sec (birth to absorption)} = 10^{-4} \text{ sec}$$

$$\bar{\ell}_d = 10^{-14} \text{ sec (absorption to fission)} + 10 \text{ sec (fission to birth)} + 10^{-4} \text{ sec (birth to absorption)} = 10 \text{ sec}$$

- The generation time is weighted average of the lifetime for prompt and delayed neutrons.

$$\bar{\ell} = (1-\beta)\bar{\ell}_p + (\beta)(\bar{\ell}_d)$$

Assume BOL conditions $\beta = .0070$

$$\bar{\ell} = (1-.007)(10^{-4}) + (.007)(10)$$

$$\bar{\ell} = .0000993 + .07$$

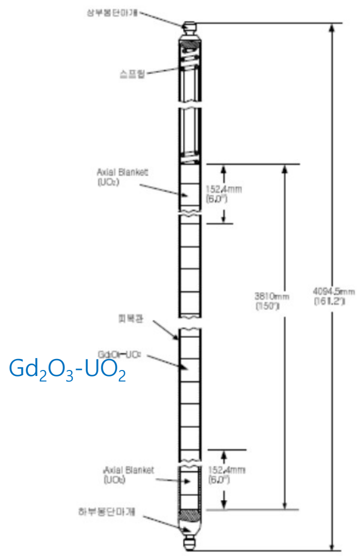
$$\bar{\ell} = .0700993 \text{ seconds} = .07 \text{ seconds}$$

- Example :
  - The reactor would be uncontrollable with a life cycle time due to prompt neutrons alone,
  - but with the addition of delayed neutrons the reactor is controllable.

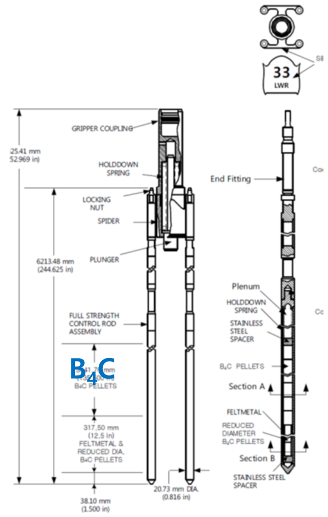
GIVEN:	PROMPT $_0N^1$ ONLY	PROMPT + DELAYED $_0N^1$
$N_0 = 10^6$ NEUTRONS	$N = N_0\, e^{(K_{eff}-1)\, t/\bar{\ell}_p}$	$N = N_0\, e^{(K_{eff}-1)\, t/\bar{\ell}}$
$t = 5$ SEC	$N = 10^6\ e^{(.001)5/10^{-4}}$	$N = 10^6\ e^{(.001)5/.07}$
$\bar{\ell}_p = 10^{-4}$ SEC	$N = 10^6\ e^{50}$	$N = 10^6\ e^{.0714}$
$\bar{\ell}_d = 10$ SEC	$N = 10^6 \times 5.184 \times 10^{21}$	$N = 10^6 \times 1.074$
$\bar{\ell} = 0.07$ SEC		
$K_{eff} = 1.001$	$N = 5.18 \times 10^{27}$ NEUTRONS	$N = 1.074 \times 10^6$ NEUTRONS

Neutron Poisons

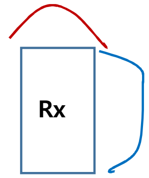
- **A fixed burnable neutron poison (in fuel rods)**
  - a material that has a high neutron absorption cross section that is converted into a material of relatively low absorption cross section as the result of neutron absorption. (the form of compounds of uranium and gadolinium : Gd_2O_3-UO_2)
  - used in reactor cores to compensate for the excess positive reactivity of the fuel
  - Fixed burnable poisons have several advantages over chemical shim(Boric acid).
    - Can be used to shape flux profiles
    - Do not have an adverse effect on moderator temperature coefficient



- **A non-burnable neutron poison (CEA)**
  - a material that has relatively constant neutron absorption characteristics over core life (Hafnium and Boron(B_4C))
  - to shape or control flux profiles in the core to prevent excessive flux and power peaking near moderator regions.
  - Do not have an adverse effect on moderator temperature coefficient
  - (Power Control, Shutdown)
- **Chemical shim (Boron in Coolant)**
  - is a soluble neutron poison(Boric Acid) that is circulated in the coolant during normal operation.
  - Burnable neutron poisons are used in reactor cores to compensate for the excess positive reactivity of the fuel when the reactor is initially started up.
  - Chemical shim has several advantages over fixed burnable poisons.
    - Has a spatially uniform effect
    - Possible to increase or decrease amount of poison in the core during reactor operation
  - (Strong Negative Reactivity Insertion during Emergency Core Cooling Operation)



Neutron Flux Shape Control



- **Core Power Distribution**
  - Three methods used to shape or flatten the core power distribution.
    - Use of reflectors
    - Installation of neutron poisons
    - Axial or radial variation of fuel enrichment
- **Reflector(Fig.1)**
  - A reactor core is frequently surrounded by a "reflecting" material to reduce the ratio of peak flux to the flux at the edge of the core fuel area.
  - Reflector materials are normally not fissionable, have a high scattering cross section, and have a low absorption cross section. (for thermal reactors a good moderator is a good reflector)
- **Installation of Neutron Poison** : Fixed burnable neutron poison(Gd)
- **Axial or Radial Variation of Fuel Enrichment**
  - The simplified example illustrated in Fig.2 shows the effect of using a higher enrichment in the outer regions of the core.
  - In the example illustrated the large central peak is reduced, but the average power level remains the same.
- **Axial Flux Distribution (Fig.3)** – ASI(Axial Shape Index)
  - The thermal flux is largely suppressed in the vicinity of the control rods, and the majority of the power is generated low in the core.
  - This flux profile can be flattened by the use of axial fuel and/or poison zoning. (Part strength control rod, Fixed burnable rod)
- **(Azimuthal) Power Tilt**
  - A non-symmetrical variation of core power in one quadrant of the core relative to the other quadrant
  - The power in one portion might be suppressed by over-insertion of control rods, which, for a constant overall power level, results in a relatively higher flux in the remainder of the core.
- **Planar Radial Peaking Factor(F_{xy})**
  - Ratio of peak to average power density of the individual fuel rods

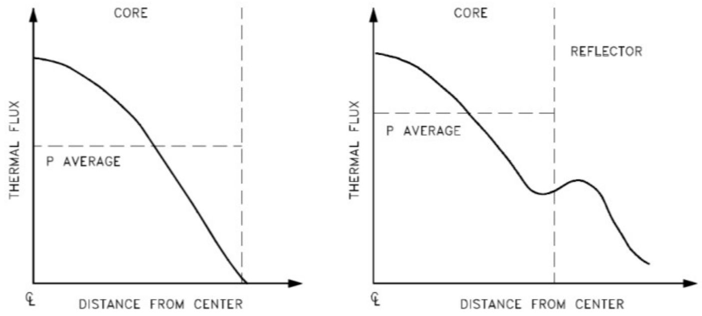


Fig.1 Neutron Radial Flux Shape for Bare and Reflected Cores

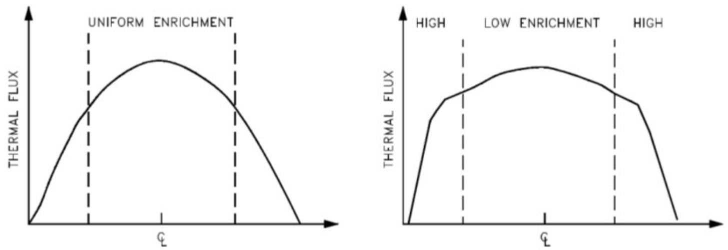


Fig.2 Effect of Non-Uniform Enrichment on Radial Flux Shape

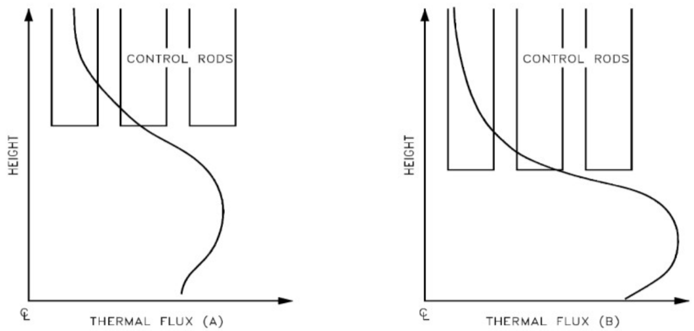


Fig.3 Effect of Control Rod Position on Axial Flux Distribution

Example of Flux Shape Control(1)

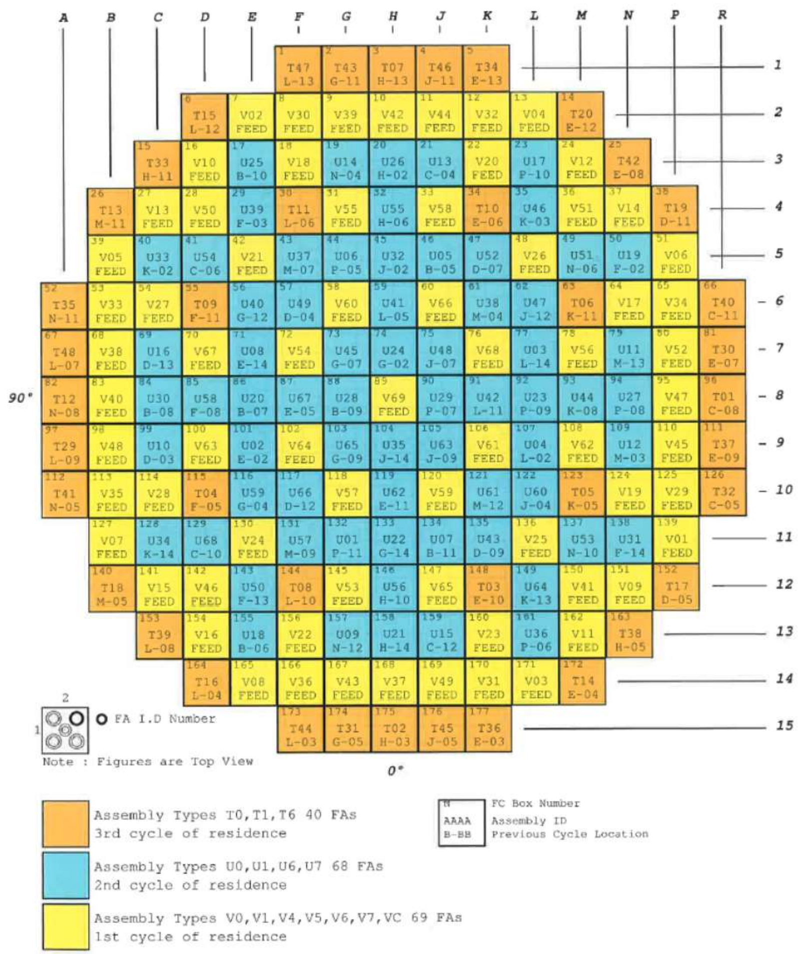


Fig. Core Loading Pattern

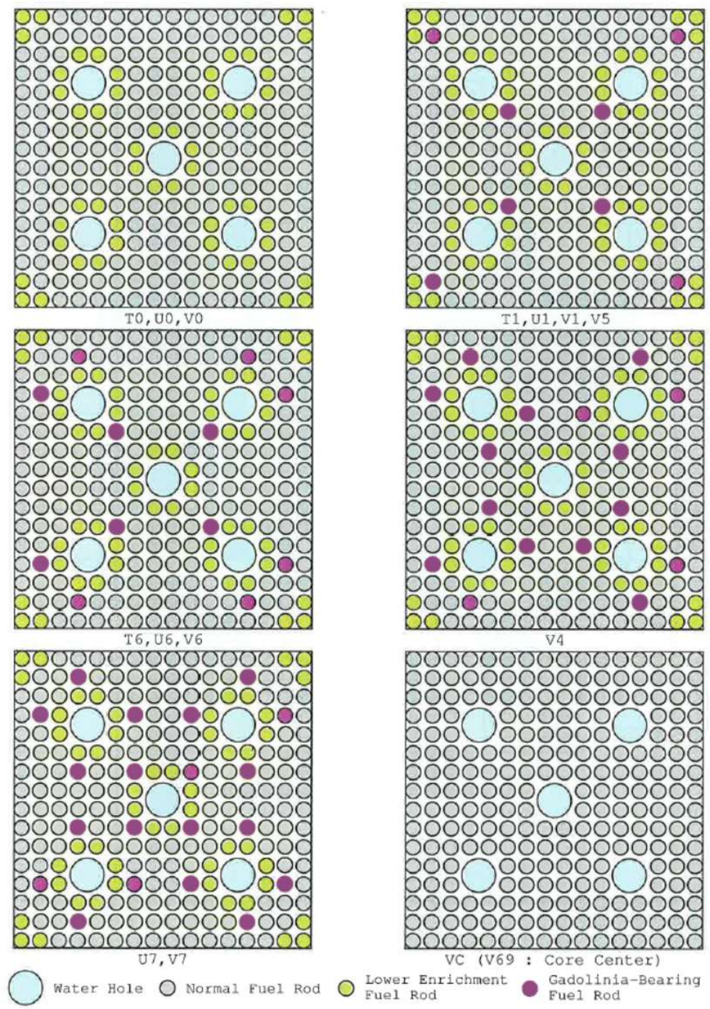


Fig. Enrichment Zoning Pattern and Burnable Poison Rod Arrangement

Xenon Poison

Samarium(Sm)-149 Case ?

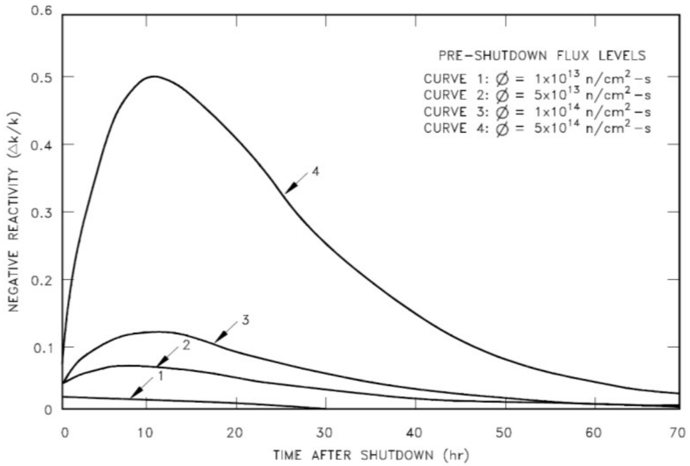
- **Xenon Production and Removal**
  - Xenon-135 has a 2.6×10^6 barns neutron absorption cross section.
  - It is produced directly by some fissions(0.3%), but is more commonly a product of the Tellurium-135 decay chain.
    - 95% of all the Xenon-135 produced comes from the decay of Iodine-135 which is successively produced from tellurium-135(~6% generation from the fission).
  - Therefore, the half-life of Iodine-135 plays an important role in the amount of Xenon-135 present.

$$ {}^{135}_{52}\mathrm{Te} \xrightarrow[19.0\ \mathrm{sec}]{\beta^-} {}^{135}_{53}\mathrm{I} \xrightarrow[6.57\ \mathrm{hr}]{\beta^-} {}^{135}_{54}\mathrm{Xe} \xrightarrow[9.10\ \mathrm{hr}]{\beta^-} {}^{135}_{55}\mathrm{Cs} \xrightarrow[2.3\times10^6\ \mathrm{yr}]{\beta^-} {}^{135}_{56}\mathrm{Ba}\ (\mathrm{stable}) $$

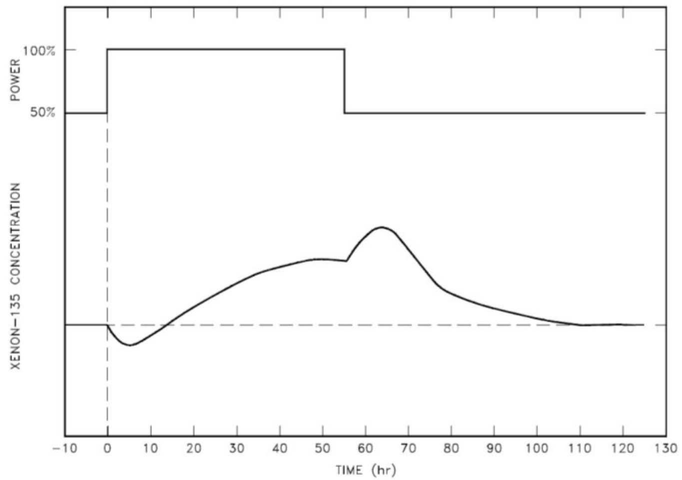
$$ \frac{dN_{Xe}}{dt} = \gamma_{Xe}\,\Sigma_f^{fuel}\,\phi + \lambda_I N_I - \lambda_{Xe} N_{Xe} - \sigma_a^{Xe} N_{Xe}\,\phi $$

$$ \begin{matrix}\text{rate of change of}\\ \text{xenon-135 concentration}\end{matrix} = \begin{pmatrix}\text{xenon-135 yield}\\ \text{from fission}\end{pmatrix} + \begin{pmatrix}\text{iodine-135}\\ \text{decay}\end{pmatrix} - \begin{pmatrix}\text{xenon-135}\\ \text{decay}\end{pmatrix} - \begin{pmatrix}\text{xenon-135}\\ \text{burnup}\end{pmatrix} $$

- **Xenon-135 Behaviors**



Xe-135 Reactivity after Reactor Shutdown



Xe-135 Variations During Power Changes

- After a power increase, Xe-135 concentration will initially decrease due to the increased removal by burn-up.
- Xe-135 will reach a minimum about 5 hours after the power increase and then increase to a new, higher equilibrium value as the production from iodine decay increases.

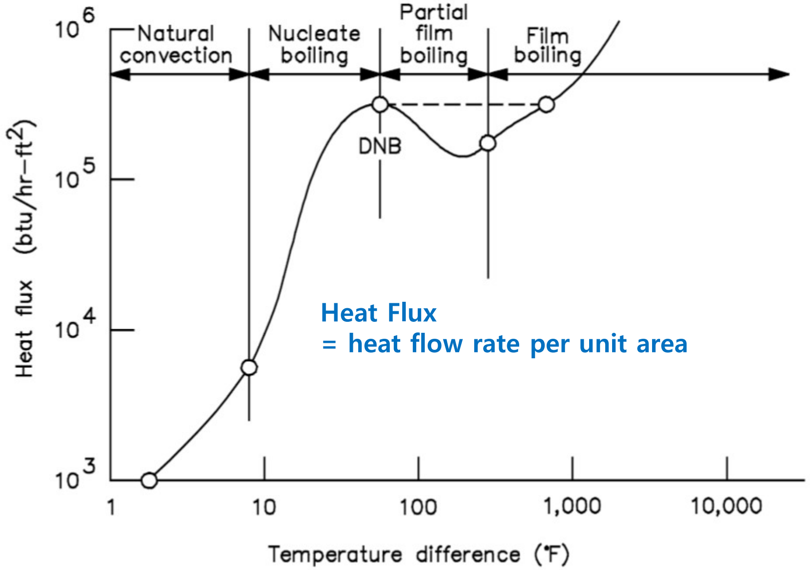
Shutdown Margin

- Several factors may change during and after the shutdown of the reactor that affect the reactivity of the core.
  - Control rod position
  - Soluble neutron poison concentration
  - Power level, Temperature of the fuel and coolant
  - Core Burnup
  - Xenon (& Samarium)
  - Pressure/Flow - BWR

- **Shutdown margin(SDM) means**
  - the instantaneous amount of reactivity by which a reactor is **subcritical** or would be subcritical from its present condition assuming all control rods are fully inserted except for the single rod with the highest integral worth(stuck rod criterion), which is assumed to be fully withdrawn.
  - It is important that there be enough negative reactivity(SDM) capable of being inserted by the control rods **to ensure complete shutdown** at all times during the core lifetime.

DNB & CHF

- **DNB (Departure from Nucleate Boiling)**
  - In practice, if the heat flux is increased,
    the transition from nucleate boiling to film boiling occurs suddenly,
    and the temperature difference increases rapidly, as shown
    by the dashed line in the figure.
  - The process at the transition is so unstable that the heat transfer process actually moves back and forth between nucleate and film boiling.
  - **Departure from nucleate boiling(DNB) :**
    - The point of transition from nucleate boiling to film boiling
  - **Critical heat flux(CHF) :**
    - The point of the heat flux that causes DNB to occur



❖ DNB Ratio(DNBR)

= The ratio of (CHF at a particular core location) to (the local heat flux)

- Ex) Assume CHF is designed as 27 kw/ft.
  - Calculate the DNBR when actual local power of 9 kw/ft.
  - ➔ Then, DNBR = 27/9 = 3 : Film boiling does not occur
  - Local power reaches 27kw/ft, then DNBR= 27/27 =1
  - ➔ Film boiling just starts

- A limit for the DNBR is set at the value greater than 1.0 to provide a margin of safety and to ensure that DNB will never reached by excessive heat generation. **DNBR Limit > 1.3 (typical)**

- Elements decreasing the margin to DNB
  - Decrease in reactor coolant pressure,
  - Decrease in reactor coolant flow rate,
  - Increase in reactor power,
  - Increase in reactor coolant inlet temperature

Limitations for Safe Operation related to Reactor Physics

- **Reactor Shutdown**
  - Fast Suppression of Nuclear Fission
  - Enough Shutdown Margin : Potential Sub-criticality in any time
  - ➔ Control Rods, Boron Control
- **Reactivity Control**
  - Xenon, MTC/FTC, Fuel Burn-up
  - ➔ Control Rods, Boron Control, Emergency Boron Injection
- **Flux Shape Control**
  - Local Heat Flux Control, Prevention of Unbalanced Burn-up
  - ➔ Neutron Detectors, Control Rod, Nuclear Design

❖ **Plant Technical Specifications (Tech. Spec.)**

Reference Materials

1. Power Plant Engineering Course Manual (USNRC)
2. US DOE Fundamentals Handbook, Nuclear Physics and Reactor Theory, Vol1&2
3. US DOE Fundamentals Handbook, Thermodynamics, Vol2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