

INTRODUCTION TO NUCLEAR REACTORS AND NUCLEAR POWER GENERATION

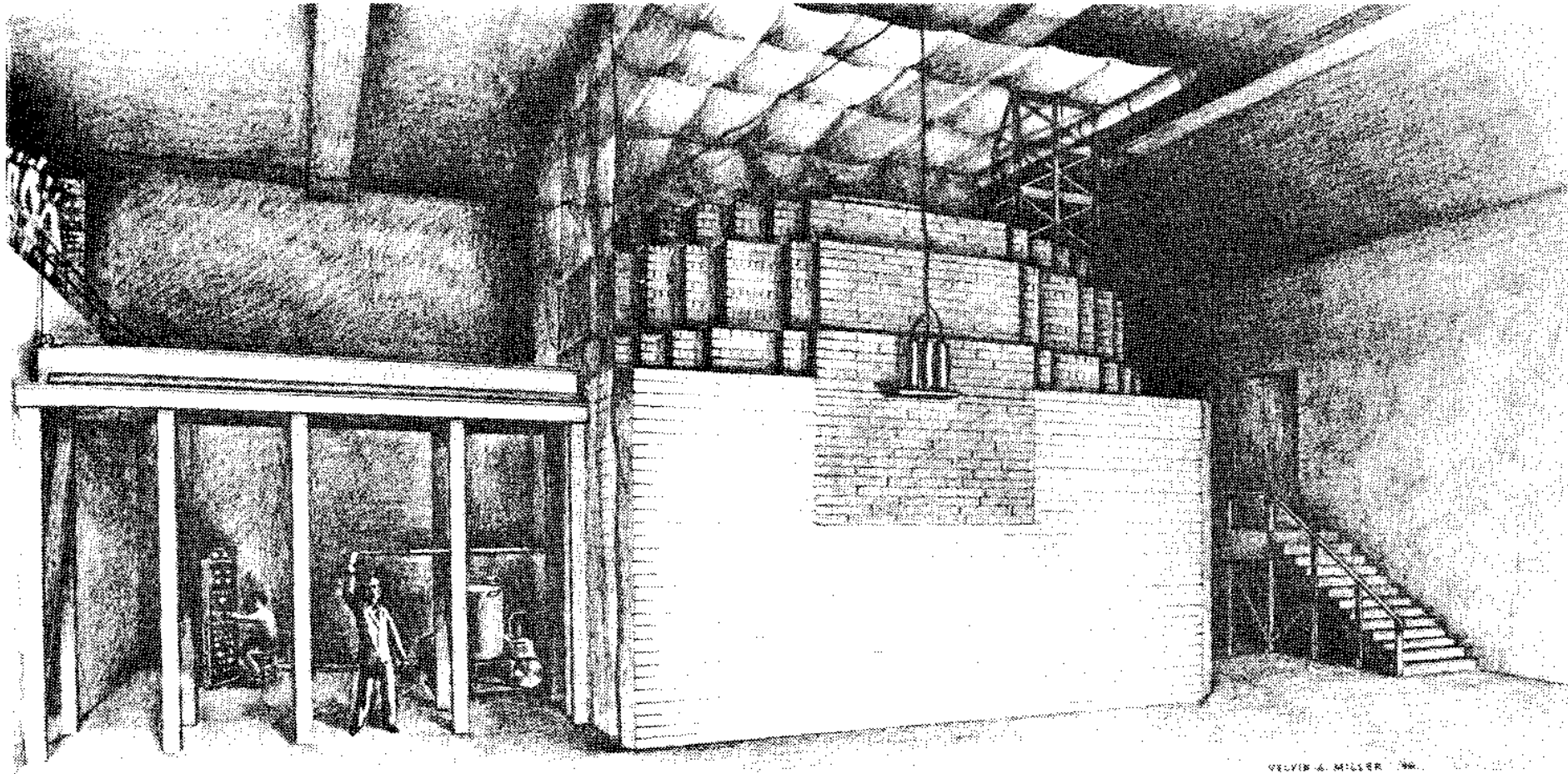
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THE FIRST STEP TOWARD NUCLEAR POWER

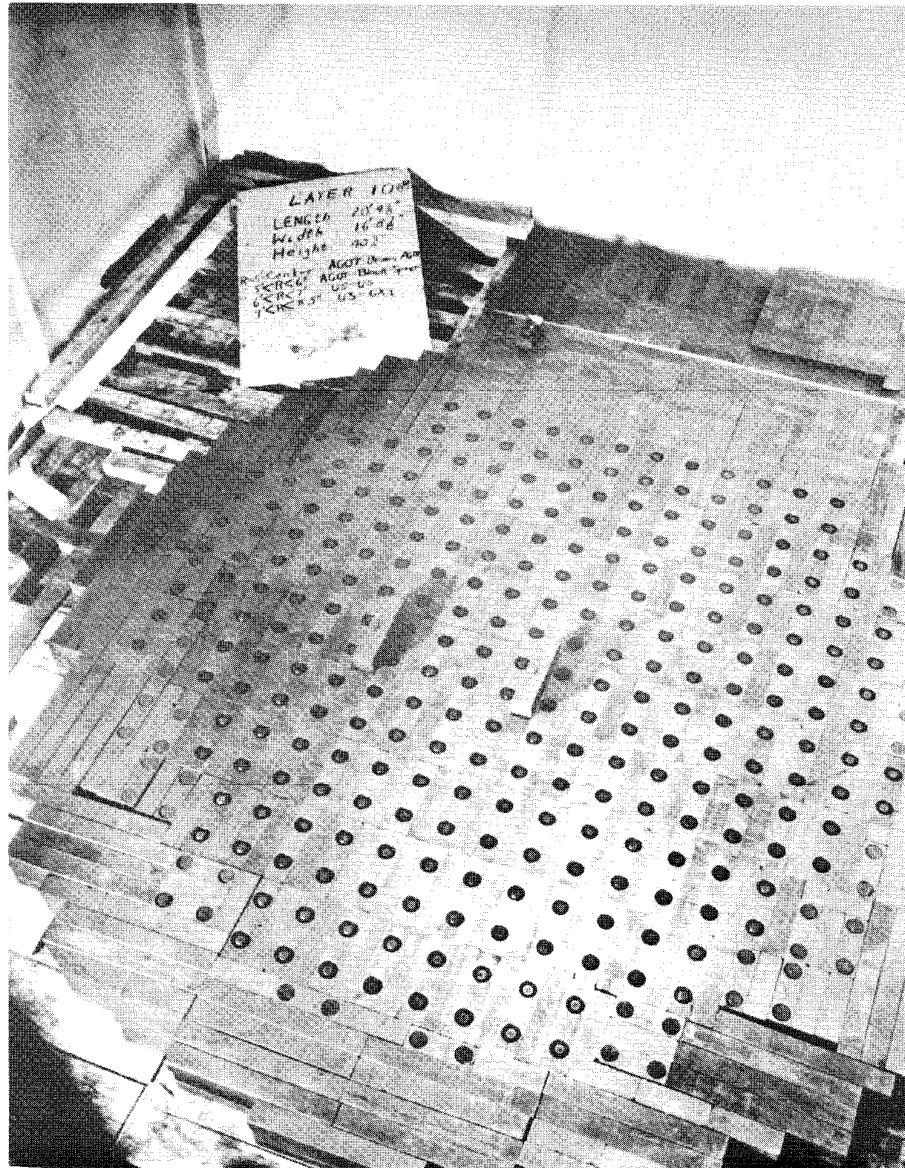
THE FIRST NUCLEAR REACTOR IN THE WORLD



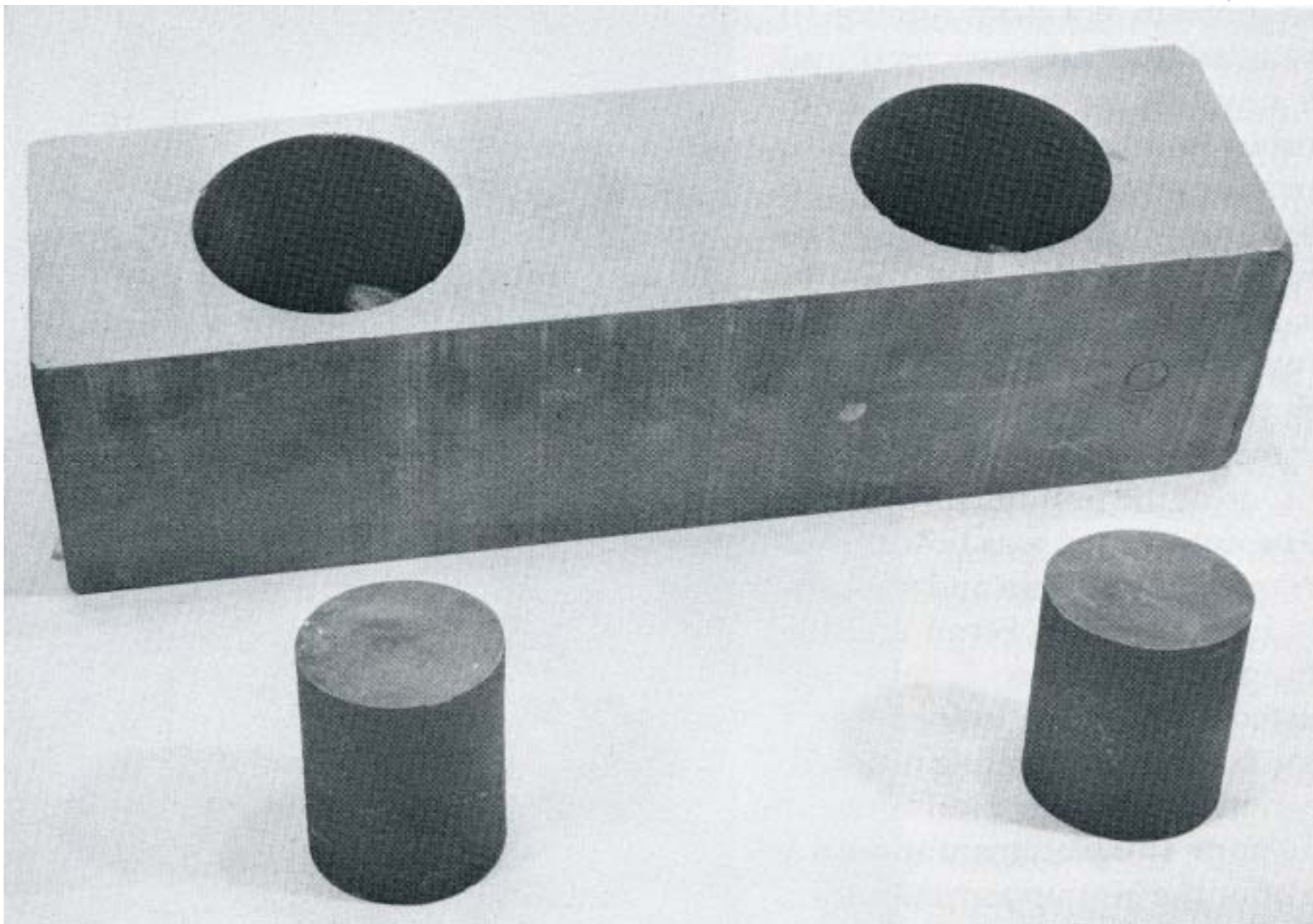
Drawing depicting CP-1

(from p.11 of “Controlled Nuclear Chain Reaction-The first 50 Years”, American Nuclear Society 1992.) 4

REACTOR CORE STRUCTURE OF CP-1



CP-1 during construction, showing the pattern of graphite blocks and uranium pellets
(from p.12 of “Controlled Nuclear Chain Reaction-The first 50 Years”, American Nuclear Society 1992.) 5



A typical block of CP-1 graphite, with uranium pellets

(from p.10 of “Controlled Nuclear Chain Reaction-The first 50 Years”, American Nuclear Society 1992.) 6

NUCLEAR ENERGY

Fissile material (U-235, Pu-239, U-233, etc.) + Neutron

Neutron absorption by fissile material

Nuclear fission

Energy release + Emission of a few neutrons

GREAT HISTORICAL EVENTS FOR NUCLEAR ENERGY

In the year 1905 Discovery of the relation: $E=mc^2$
(Albert Einstein)

In the year 1932 Discovery of neutron
(James Chadwick)

In the year 1938 Discovery of nuclear fission
(Otto Hahn, Fritz Strassmann and Lise Meitner)

Nuclear energy		Energy released			
		Mev	Kcal	KWH	Oil equivalent ton [*]
U-235	1 fission	200	7.7×10^{12}	8.9×10^{18}	7.4×10^{22}
U-235	1gr	5.5×10^{23}	2.1×10^{10}	24000	2
Metalic uranium	1cc	7.5×10^{22}	2.9×10^9	3300	0.3

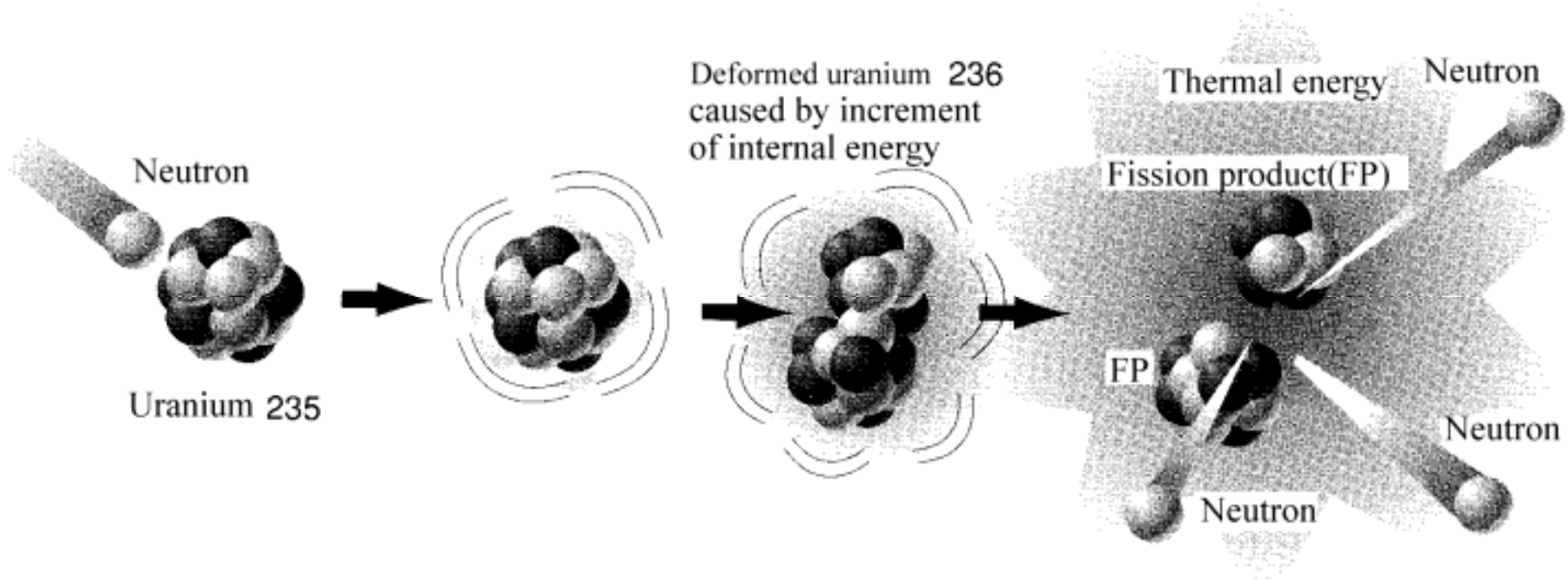
* Oil equivalent ton is a converted number in tons of oil equivalent to energy released from various sources in order to facilitate comparison of the energy.

1(Oil equivalent ton)=12000(KWH)

Magnitude of nuclear energy

PHYSICS OF NUCLEAR FISSION

NUCLEAR FISSION AND NEUTRON RELEASE



Nuclear fission of U-235 and neutron release

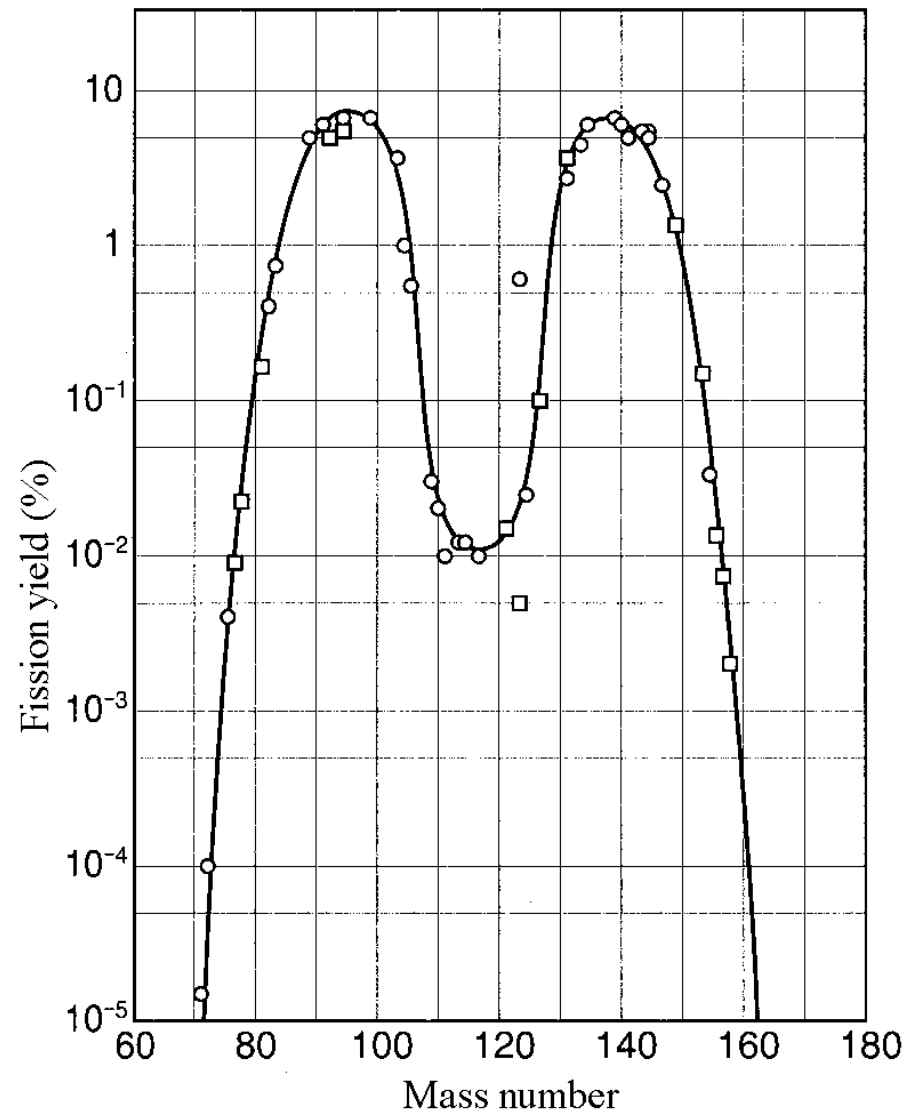
THE NUMBER OF FISSION NEUTRONS

Fissile nuclide	The number of fission neutrons ()
U-233	2.49
U-235	2.43
Pu-239	2.88
Pu-241	2.94

Values in the above table are the average number of fission neutrons released when the fission is induced by a thermal neutron at the energy of 0.025 eV.

Neutron released from nuclear fission is called fission neutron.

FISSION PRODUCTS OF NUCLEAR FISSION



Mass distribution of fission products of U-235 fission by thermal neutron

CROSS SECTION

Probability of neutron reaction with nucleus is expressed by cross section of the nucleus.

Greek letter sigma (σ) is used as the notation of cross section.

σ_f = cross section of (n, f) reaction

σ_c = cross section of (n,) reaction

σ_a = cross section of (n,) reaction

σ_s = cross section of elastic scattering (n, n) reaction

σ_i = cross section of inelastic scattering (n, n') reaction

$\sigma_a = \sigma_f + \sigma_c + \sigma_i$

NEUTRON REACTION WITH FISSILE MATERIAL

There are three different ways of reaction of neutron with fissile atoms.

- (1) A neutron is absorbed to induce nuclear fission of the nucleus.
- (2) A neutron is absorbed but no neutron is emitted.
- (3) A neutron is scattered by the nucleus.

The average number of neutrons generated per one neutron absorbed by a fissile nucleus can be expressed as

$$k = \frac{\nu f}{\sigma_a} = \frac{\nu f}{\sigma_f + \sigma_{sc}}.$$

The k value is more important than the β value for chain reaction.

Note: In the reaction of neutron with fissile material, all other neutron absorption reactions except (n, f) and (n, γ) reactions can be neglected. Accordingly, $k = \frac{\nu f}{\sigma_f + \sigma_{sc}}$ is an appropriate expression for this case.

THE VALUE OF VARIOUS FISSILE MATERIALS

Fissile material		*
U-233	2.30	2.49
U-235	2.07	2.43
Pu-239	2.12	2.88
Pu-241	2.17	2.94

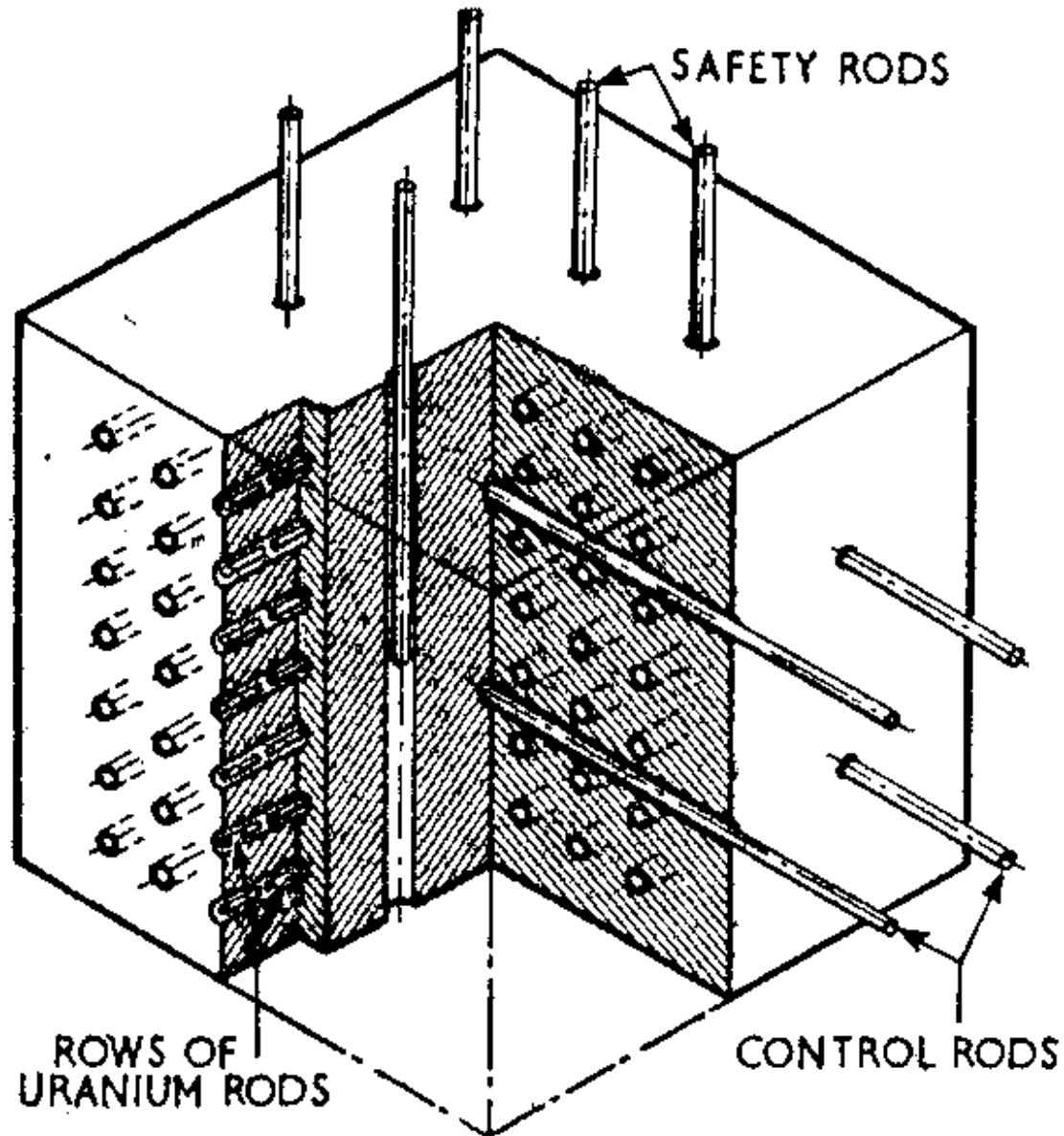
Note: The values in the above table are the average values at the neutron energy of 0.025 eV.

* This column is reproduced for reference from the table shown previously.

SUSTAINED CHAIN REACTION IN NUCLEAR REACTOR

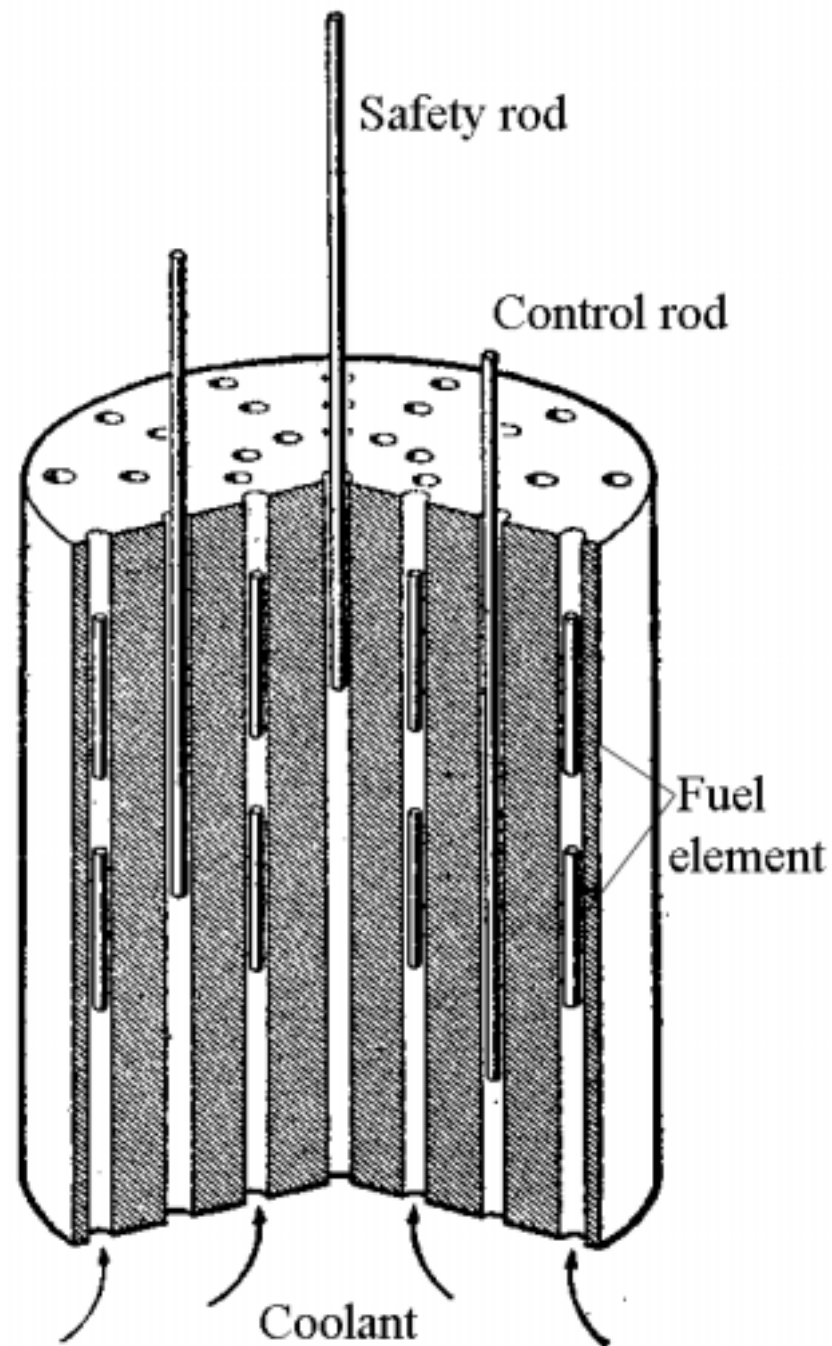
NUCLEAR CHAIN REACTION

- 1) A neutron hits a fissile atom and is absorbed in the nucleus of the atom.
- 2) The nucleus goes into nuclear fission.
- 3) As a result of the nuclear fission, fission products and a few neutrons are released with huge energy.
- 4) Some of the fission neutrons succeed in hitting another fissile atom to induce nuclear fission of the nucleus.
- 5) This process creates a new generation of neutrons successively from the preceding generation of neutrons.



Principle of core arrangement in an early experimental reactor

(from p.8 of "Calder Hall-the story of Britain's first atomic power station" by Kenneth Jay 1956.)



Principle of vertical core arrangement for power reactors

(from p.27 of "Calder Hall-the story of Britain's first atomic power station" by Kenneth Jay 1956.)

REACTOR POWER AND NEUTRON POPULATION

Reactor power is proportional to the number of nuclear fissions per unit time.

The number of nuclear fissions per unit time is proportional to the neutron population at the moment

Accordingly, neutron population represents reactor power.

NEUTRON BALANCE IN CHAIN REACTION

N_A = The number of absorbed neutrons by fissile material

N_C = The number of absorbed neutrons by non-fissile materials

N_L = The number of lost neutrons out of the system

$N_A + N_C + N_L$ = The total number of neutrons consumed

N_A = The Number of released neutrons from induced nuclear fissions

(1) Increase in neutron population

$$N_A > N_A + N_C + N_L$$

(2) Constant neutron population

$$N_A = N_A + N_C + N_L$$

(3) Decrease in neutron population

$$N_A < N_A + N_C + N_L$$

NEUTRON LIFE AND NEUTRON MULTIPLICATION FACTOR

Neutron population of the first generation = $N_A + N_C + N_L$

Neutron Population of the second generation = N_A

Neutron life (Neutron life span): T

= The time required to replace the first generation by the second

Neutron multiplication factor: K

$$= N_A / (N_A + N_C + N_L)$$

CONCEPT OF CRITICALITY

Supercritical condition	$K > 1$
Critical condition	$K = 1$
Subcritical condition	$K < 1$
$K = \eta N_A / (N_A + N_C + N_L)$	

NEUTRON POPULATION CHANGE WITH TIME

(1/8)

Time-dependent neutron population = $n(t)$

Neutron life = T

Neutron multiplication factor = K

Neutron population at the time $t = n(t)$

Neutron population at the time $t + T = n(t + T) = K n(t)$

Neutron population change from time t to time $t + T$

$$\Delta n = n(t + T) - n(t) = K n(t) - n(t)$$

$$\therefore \Delta n = (K - 1)n(t) \quad \cdot \quad \cdot \quad \cdot \quad \cdot \quad \cdot \quad \cdot \quad \cdot \quad (1)$$

NEUTRON POPULATION CHANGE WITH TIME

(2/8)

On the other hand, Δn can be expressed as follows.

$$\Delta n = \frac{dn}{dt} T \quad \cdot \quad \cdot \quad \cdot \quad \cdot \quad \cdot \quad \cdot \quad \cdot \quad (2)$$

From the equation (1) and (2), a differential equation is obtained.

$$\frac{dn}{dt} = \left(\frac{K - 1}{T} \right) n(t)$$

$$\therefore n(t) = n(0) \exp\left\{ \frac{(K - 1)t}{T} \right\}$$

Neutron population will increase or decrease exponentially.

NEUTRON POPULATION CHANGE WITH TIME

(3/8)

Useful concept of neutron doubling time at supercritical condition

Neutron population at the time t :

$$n(t) = n(0) \exp\left\{\frac{(k-1)t}{T}\right\}$$

Neutron population at the time $T + \Delta t$:

$$\begin{aligned} n(t + \Delta t) &= n(0) \exp\left\{\frac{(K-1)(t + \Delta t)}{T}\right\} \\ &= n(0) \exp\left\{\frac{(K-1)t}{T}\right\} \exp\left\{\frac{(K-1)\Delta t}{T}\right\} \end{aligned}$$

When $\exp\left\{\frac{(k-1)\Delta t}{T}\right\} = 2$,

$$\Delta t = \frac{T \ln(2)}{K-1}$$

Then, the neutron doubling time (DT) = $\frac{T \ln(2)}{K-1}$

NEUTRON POPULATION CHANGE WITH TIME

(4/8)

At supercritical condition : $K > 1$ or $K - 1 > 0$

- (1) Neutron population increases exponentially with time.
- (2) Neutron DT is proportional to neutron life T.

Then, neutron population would increase slowly if neutron life is longer with the same value of K.

A reactor with long neutron life would be easily controlled.

NEUTRON POPULATION CHANGE WITH TIME

(5/8)

At subcritical condition : $K < 1$ or $K - 1 < 0$

- (1) Neutron population decreases exponentially with time. Then, finally neutron population would approach zero if no neutron source existed in the system.
- (2) If neutron source existed in the system, neutron population would approach a steady state value*.

* The steady state value depends on both value of K and strength of neutron source in the system.

NEUTRON POPULATION CHANGE WITH TIME

(6/8)

At critical condition : $K = 1$ or $K - 1 = 0$

(1) Neutron population does not change with time.

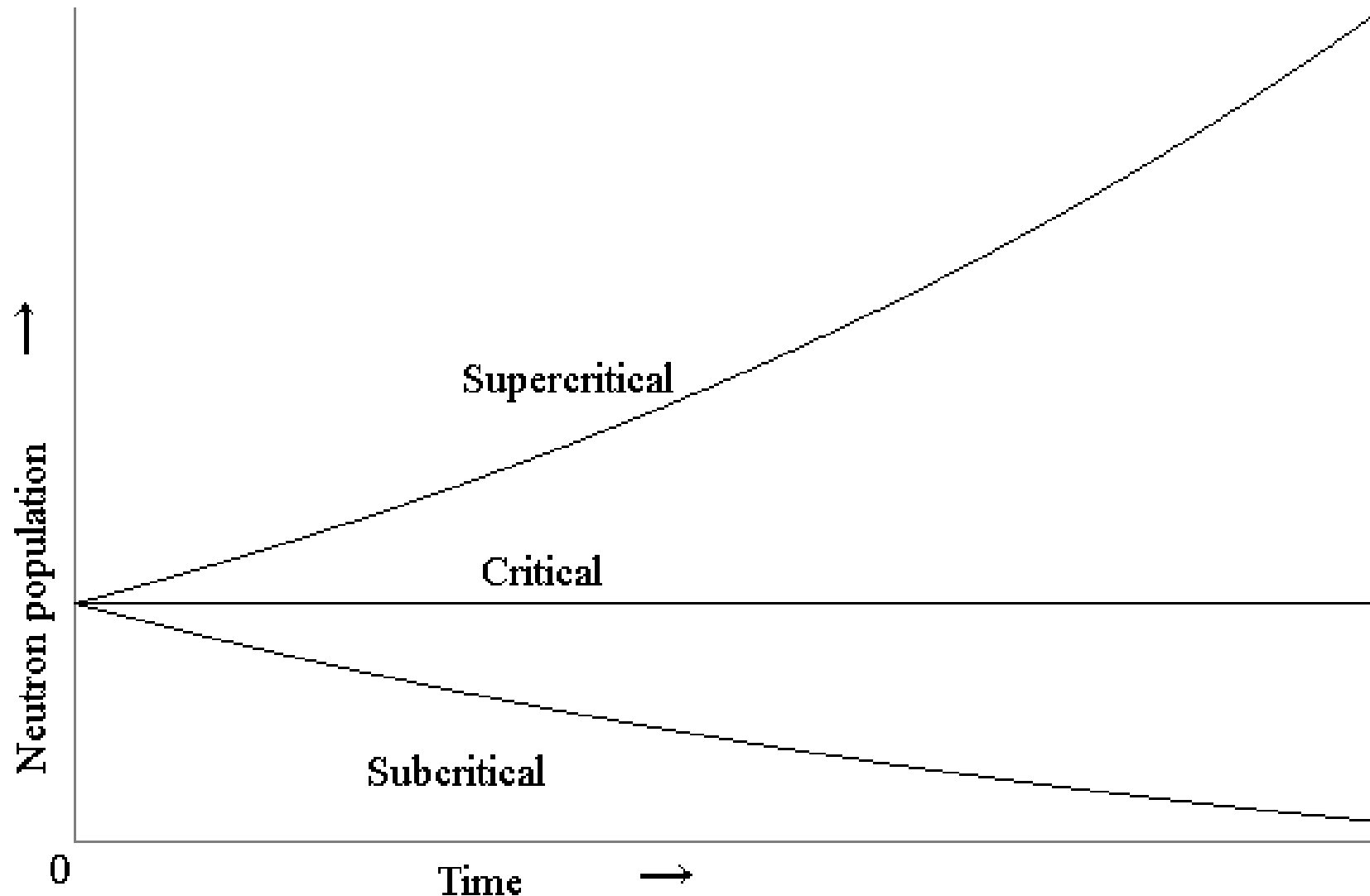
(2) If neutron source existed in the system,
neutron population would increase linearly with time*.

* The increment of neutron population depends on the strength of neutron source.

NEUTRON POPULATION CHANGE WITH TIME

(7/8)

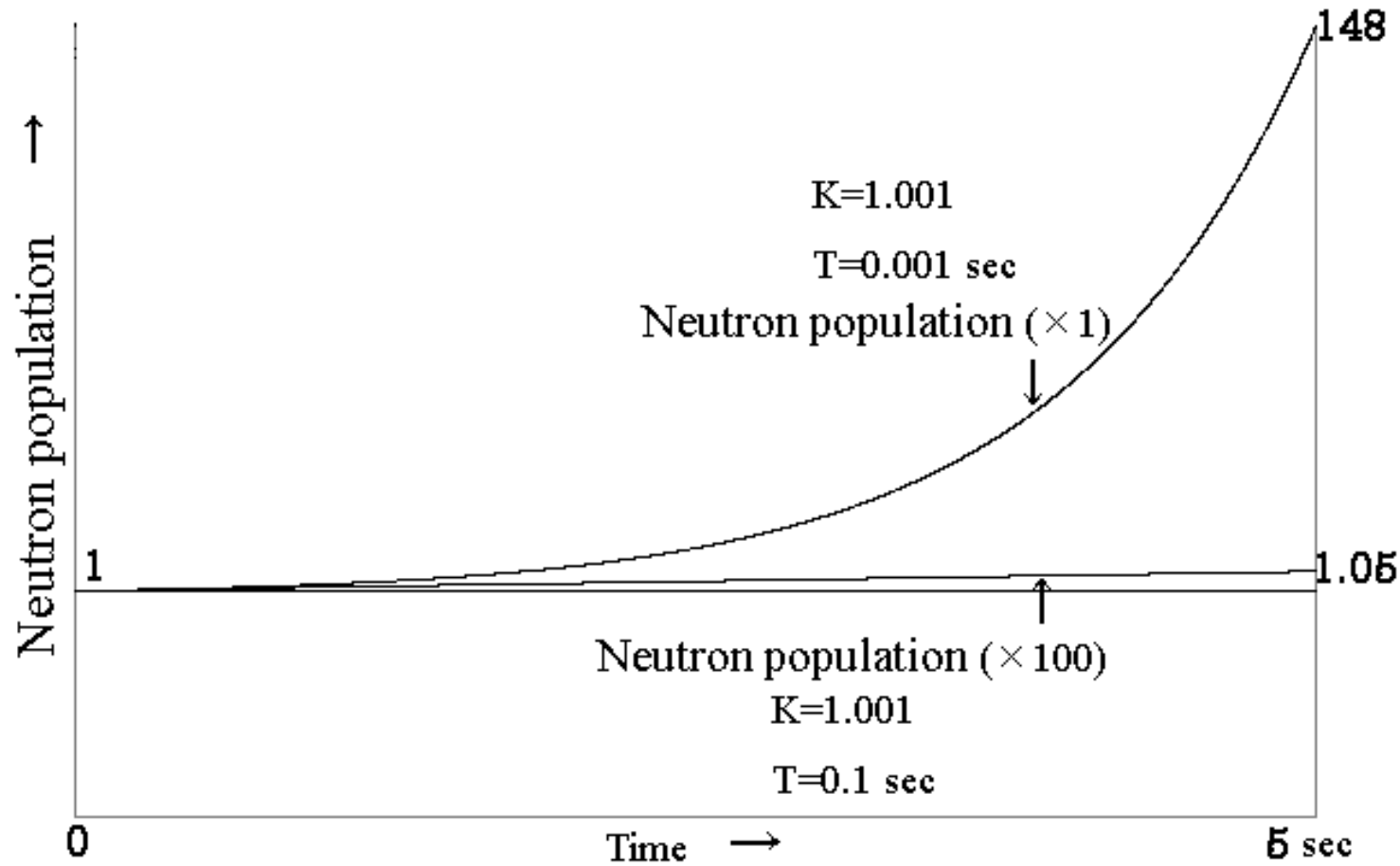
Neutron population changes with time at various criticality conditions.



NEUTRON POPULATION CHANGE WITH TIME

(8/8)

Effect of neutron life on neutron population change



CONTROL OF NUCLEAR REACTOR

HOW TO CONTROL CHAIN REACTION

$$K = N_A / (N_A + N_C + N_L).$$

K could be controlled by changing N_C

Adding or removing neutron absorber

CONTROL ROD

CONTROL ROD

Two important roles of control rod:

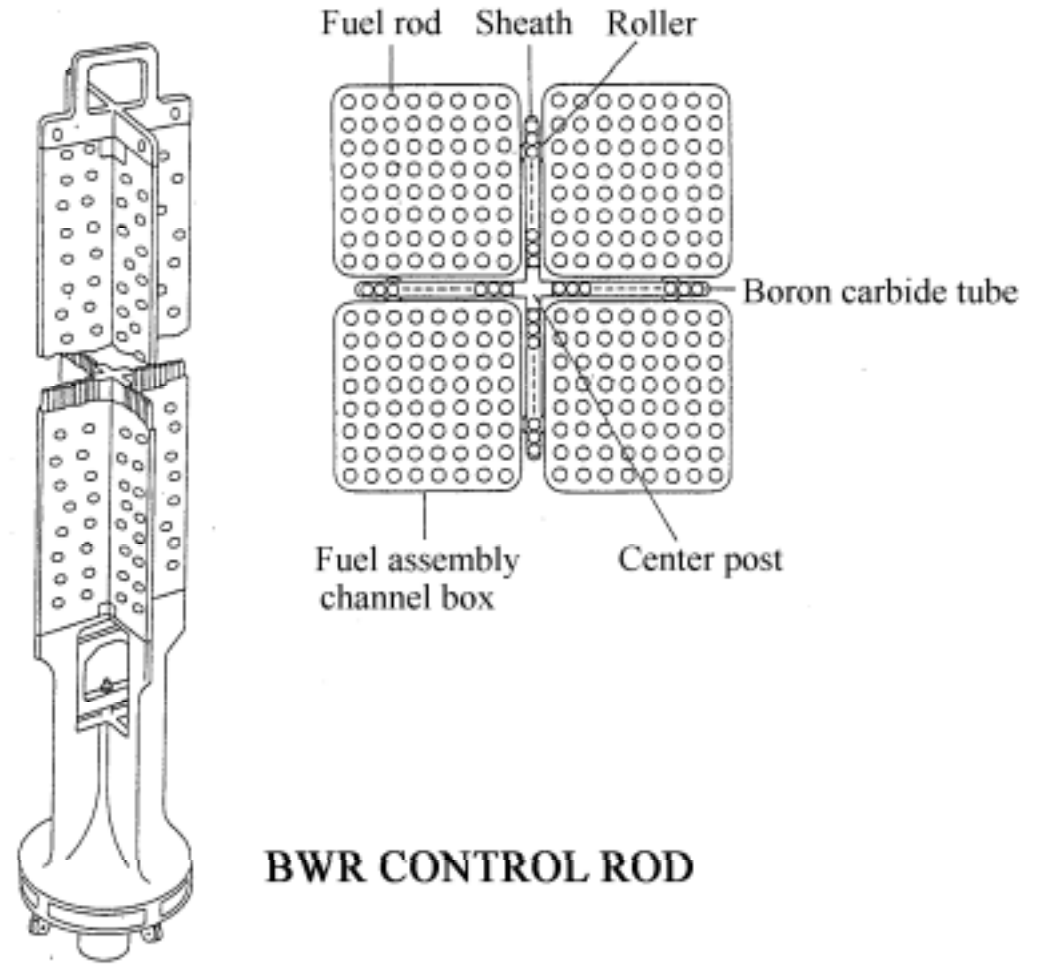
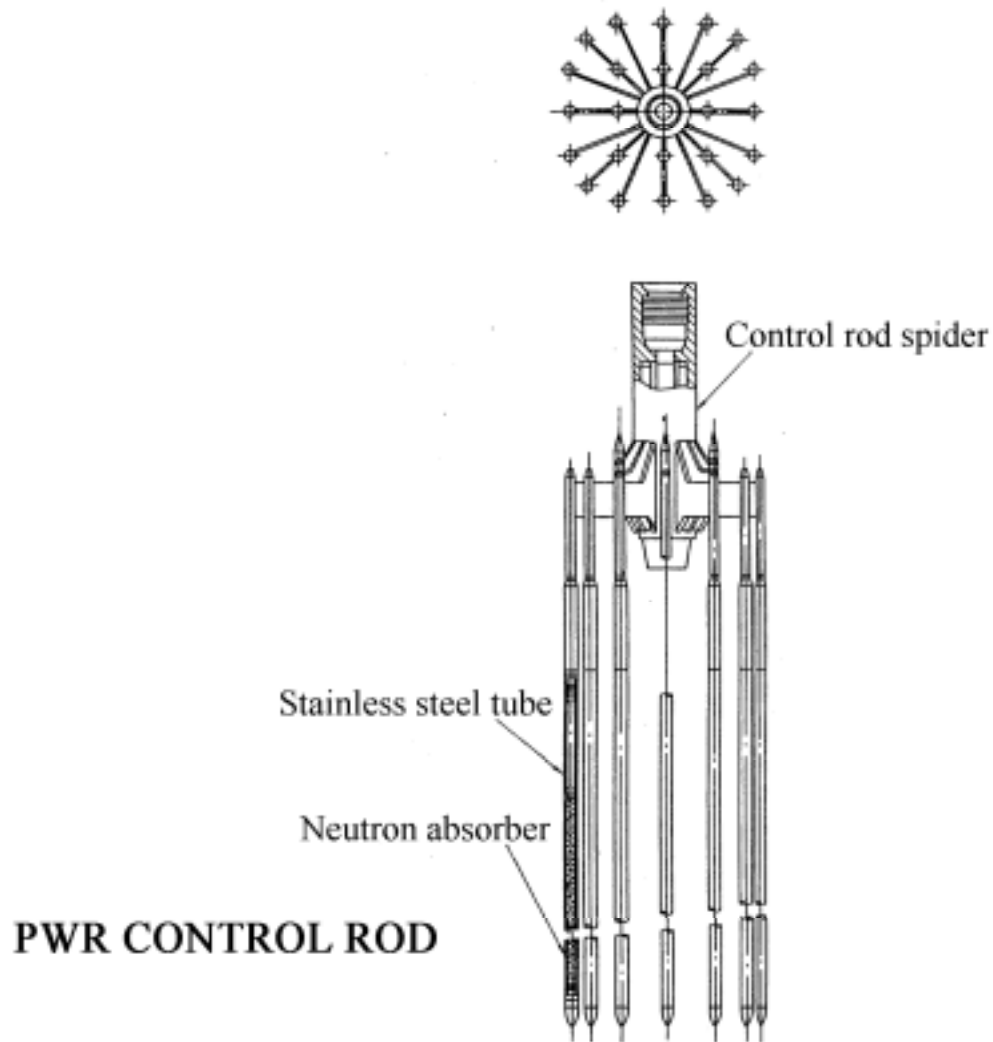
- 1) Control of chain reaction in the reactor
- 2) Safe shutdown of the reactor

Materials for control rod:

Boron (B), Hafnium (Hf), Cadmium (Cd)

Materials for burnable poison:

Gadolinium (Gd), Erbium (Er)



Control rods for light water reactors

REACTOR CONTROL AND NEUTRON LIFE

Practically, neutron life (T) of all kinds of nuclear reactors is about 10^{-1} [sec].

This relatively long neutron life is a key factor to easy control of the reactors.

For example, in case of $K=1.001$, which is a common size of K adopted to increase the reactor power, neutron doubling time is

$$t = T \cdot \ln(2) / (K - 1) = 0.1 \times 0.693 \div 0.001 = 69.3 \text{ [sec]}$$

This means that it takes more than a minute to double the reactor power.

NEUTRON LIFE AND DELAYED NEUTRONS (1/5)

Neutron life (T) consists of slowing-down time in moderator (T_S) and diffusion time as thermal neutrons (T_D).

That is, $T = T_S + T_D$.

After moderation process, thermalized neutrons walk around in the reactor for a while before hitting fissile atoms to induce nuclear fission.

This process is called “neutron diffusion”, and the time consumed for the process is called diffusion time (T_D).

The diffusion time is actually very much longer than slowing-down time. That is, $T_S \ll T_D$. Then, $T \approx T_D$

NEUTRON LIFE AND DELAYED NEUTRONS (2/5)

The diffusion time depends on reactor design.

Approximate value of diffusion times are shown for typical reactor types.

Reactor type	Diffusion time (T_D)
GCR	1×10^{-3}
HWR	1×10^{-3}
BWR	4.3×10^{-5}
PWR	1.5×10^{-5}

Note: GCR=Graphite-moderated gas-cooled reactor, HWR=Heavy water reactor, such as CANDU.

NEUTRON LIFE AND DELAYED NEUTRONS (3/5)

More than 99% of fission neutrons are released promptly just after the nuclear fission. Less than 1% of fission neutrons, however, come out very late. Such neutrons are called “delayed neutron”. The ratio of delayed neutrons to all fission neutrons is denoted by β .

The time required for birth of prompt neutrons is almost zero, but the time required for birth of delayed neutrons is very long.

As a result, the average time required for birth of all fission neutrons becomes very long, that is about 10^{-1} [sec] though the percentage of delayed neutrons is less than 1% as already mentioned.

NEUTRON LIFE AND DELAYED NEUTRONS (4/5)

Fissile nuclide	value
U-233	0.0026
U-235	0.0065
Pu-239	0.0021

NEUTRON LIFE AND DELAYED NEUTRONS (5/5)

We understand that neutron life (T) should be expressed as follows.

$$T = 10^{-1} + T_S + T_D \approx 10^{-1} + T_D \approx 10^{-1} \text{ [sec].}$$

The effect of delayed neutrons almost unexpectedly prolongs neutron life up to about 10^{-1} [sec], which was used in the previous calculation.

Observing the preceding discussion, we can conclude that the delayed neutron effect on the neutron life is practically not dependent on reactor design but fissile material used for fuel.

PROMPT CRITICALITY (1/2)

When K satisfies the following equation,

$$(1 - \beta)K = 1$$

Such a value of K is denoted by K_{PC} .

$$K_{PC} = 1/(1 - \beta) = 1 + \beta, \text{ because } \beta < 1.$$

If $K > K_{PC}$, it is called super-prompt critical.

If $K = K_{PC}$, it is called prompt critical.

At prompt critical condition, neutron chain reaction could be sustained without delayed neutron participation.

Physically this is an important threshold point.

PROMPT CRITICALITY (2/2)

Thanks to the delayed neutron effect, almost all nuclear reactors are easily controlled.

Above the prompt critical condition, effective neutron life becomes T_D because neutron multiplication process can be completed and developed without waiting for delayed neutron participation.

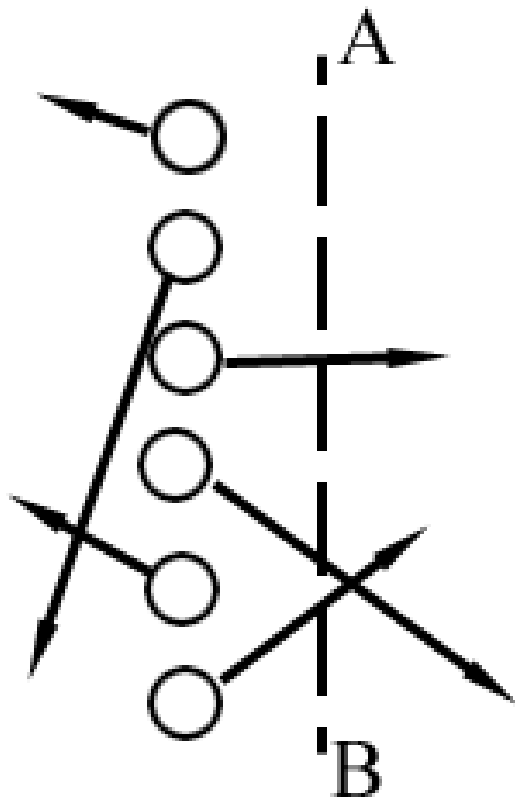
That is, T becomes the order between $10^{-3} \sim 10^{-5}$ [sec] from the order of 10^{-1} [sec].

As a result, even if K became very slightly over K_{PC} , neutron doubling time would be very short. That is, reactor power would increase too fast to control.

Such situation is called “run-away” of the reactor.

NEUTRON BEHAVIOR IN NUCLEAR REACTOR

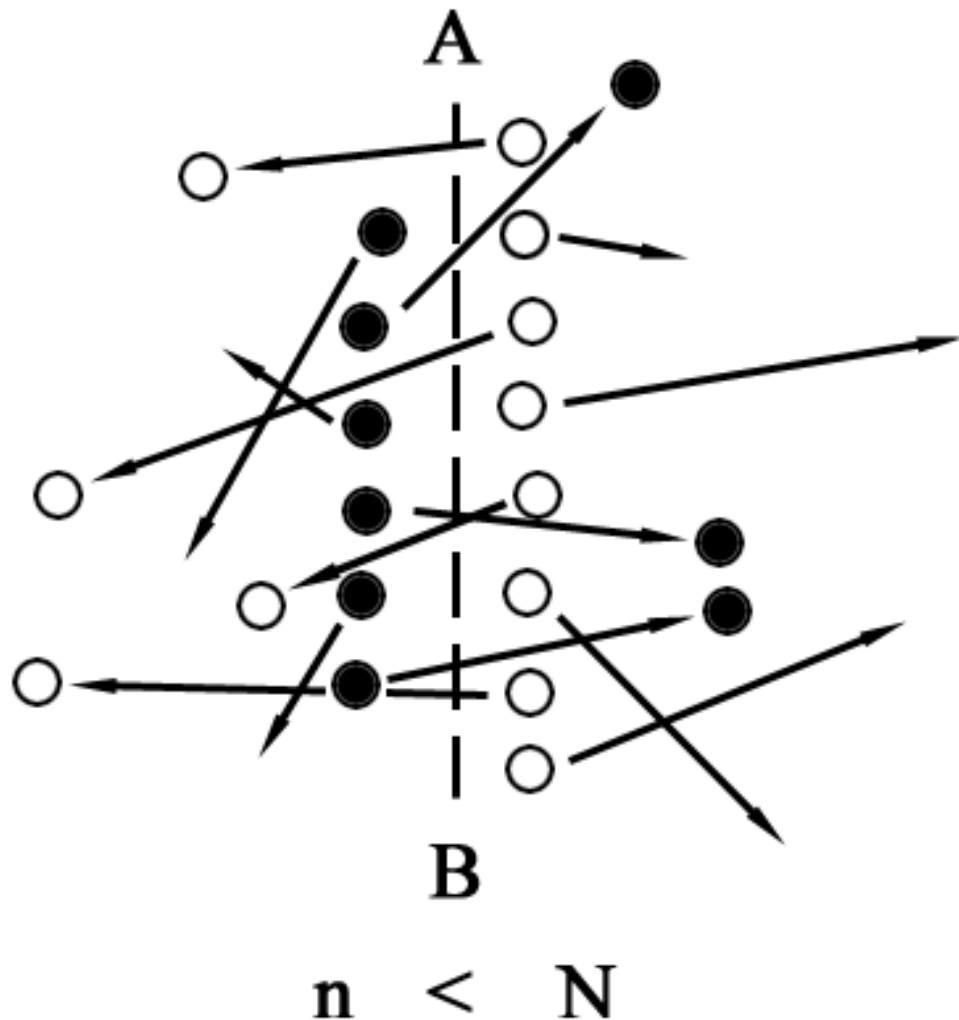
RANDOM WALK OF NEUTRON



Neutrons in a reactor move almost at random by collision with moderator.

In the homogeneous distribution of neutrons, the number of neutrons passing through the plane (A-B) from left to right per second is equal to the number of neutrons passing through the plane from right to left.

NEUTRON CURRENT (1/2)



If neutron density is different across the plane (A-B), the number of neutrons passing the plane from left to right is different from the number of neutrons passing the plane from right to left.

NEUTRON CURRENT (2/2)

N_L = Neutron density at the left side of the plane (A-B)

N_R = Neutron density at the right side of the plane (A-B)

The number of neutrons passing the unit area of the plane from the left to the right per second = $N_L/2$

The number of neutrons passing the unit area of the plane from the right to the left per second = $N_R/2$

Net neutron current from the left to the right = $N_L/2 - N_R/2$
= $(N_L - N_R)/2$

If $N_L < N_R$, $(N_L - N_R)/2 < 0$

BASIC BEHAVIOR OF NEUTRONS IN NUCLEAR REACTOR

We have seen that neutrons collectively flow out of a high density area into the adjacent low density area though all the neutrons move at random.

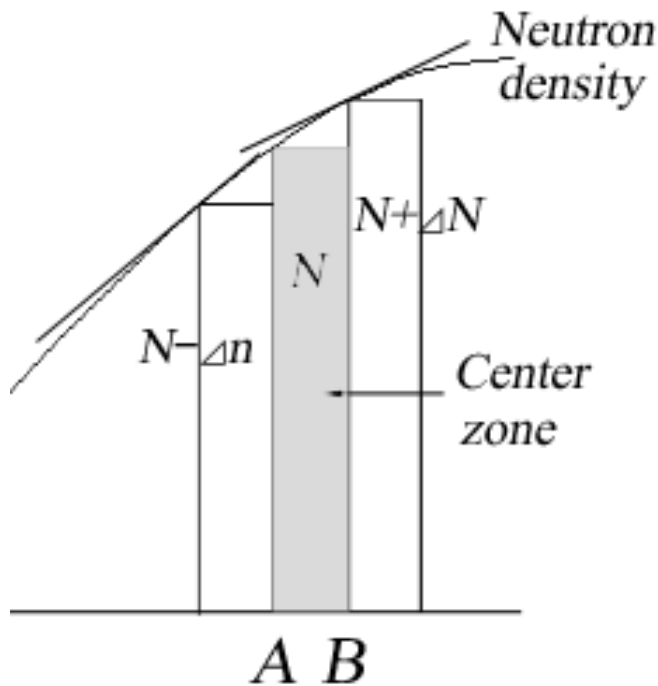
In nuclear reactor, neutron density near the surface of reactor is lower than that in the inner region.

Neutrons flow into the near-surface area from the inner region of the reactor.

Neutrons leak out of the reactor surface.

In the steady state, the total number of lost neutrons from the surface is equal to the total number of excess neutrons generated in the reactor.

NEUTRON POPULATION BALANCE BETWEEN ZONES



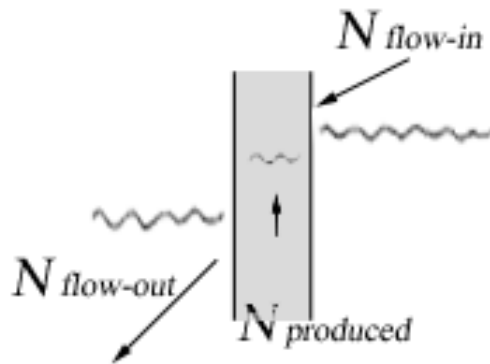
Let us think about the steady-state neutron population balance in the center zone in the figure.

Neutrons flow into the center zone from the right zone while neutrons flow out of it to the left zone.

Excess neutrons are produced by sustained chain reaction in the zone.

Neutron population balance in the zone is

$$N_{\text{FLOW-IN}} + N_{\text{PRODUCED}} = N_{\text{FLOW-OUT}}$$



$$k_{\infty} > 1$$

NEUTRON DISTRIBUTION IN NUCLEAR REACTOR

(1/3)

Production capability of excess neutrons per volume depends on both fuel material and reactor design.

The index of this capability is shown by k .

The number of excess neutrons produced in a neutron life per volume can be expressed by $(k - 1)N$, where N is neutron density at the point.

Remember the definition of K previously mentioned. It is easily understood that the following relation between K and k .

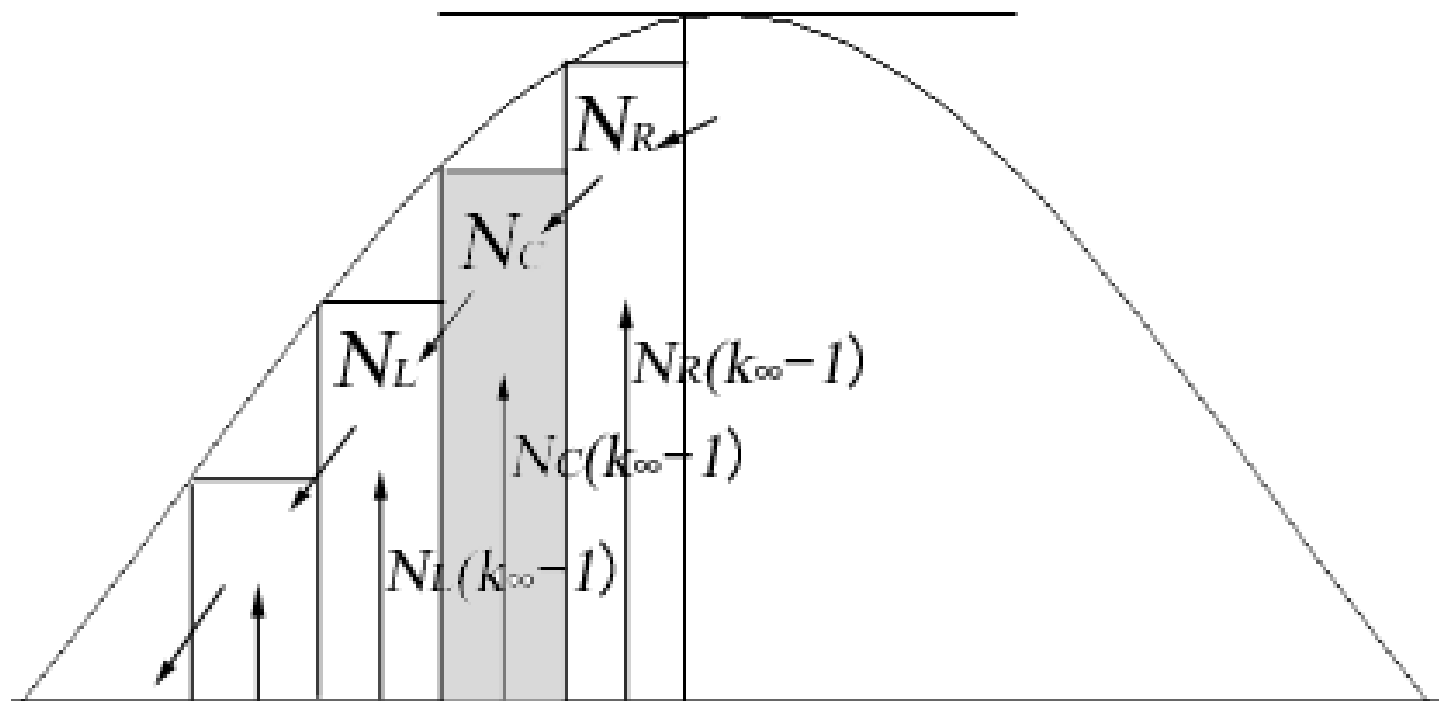
$$K = k \frac{(N_A + N_C)}{(N_A + N_C + N_L)}$$

$$K < k$$

NEUTRON DISTRIBUTION IN NUCLEAR REACTOR

(2/3)

If the reactor structure is uniform, the k value is constant everywhere, so that the number of excess neutrons produced is proportional to the neutron density at the point. Let us observe the case of this situation.



NEUTRON DISTRIBUTION IN NUCLEAR REACTOR

(3/3)

Observations on the figure of neutron distribution

- 1) Neutron density is highest at the center of the reactor.
- 2) Neutron density decreases from the center toward the periphery.
- 3) The decrease rate of neutron density increases toward the periphery. In other words, the gradient of neutron density distribution increases toward the periphery.
- 4) The rate of increase of the gradient, however, decreases toward the periphery.
- 5) Neutrons leak out of the reactor surface at the rate which is proportional to the gradient of neutron density distribution at the surface.

FAVORABLE SHAPES OF NUCLEAR REACTOR (1/3)

We have seen that nuclear reactor loses neutrons from the surface. If the reactor is critical, the total number of lost neutrons from the surface must be compensated with the total number of excess neutrons generated in the reactor.

From the view point of neutron economy, we can say that nuclear reactor should be shaped to minimize the surface area to the volume of the reactor.

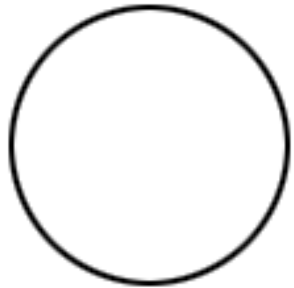
Consequently, sphere is ideal shape for nuclear reactor.

FAVORABLE SHAPES OF NUCLEAR REACTOR

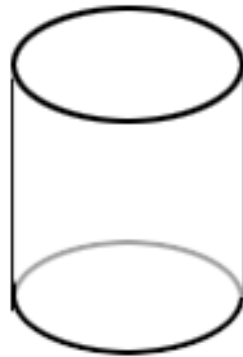
(2/3)

Though sphere is the best shape for nuclear reactor from the theoretical point of view, cylinder is the most favorable shape for power reactors from the engineering point of view.

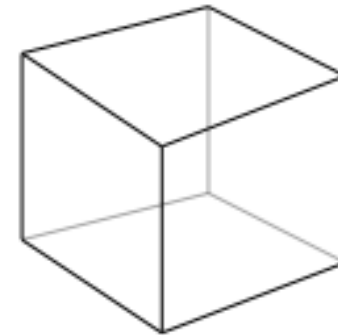
For some experimental reactors, cubic shape is chosen because of simplicity.



(a)



(b)



(c)

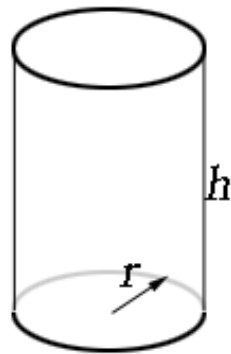
(a) Sphere: Ideal shape (b) Cylinder: Practical shape (c) Cube: Special shape

FAVORABLE SHAPES OF NUCLEAR REACTOR (3/3)

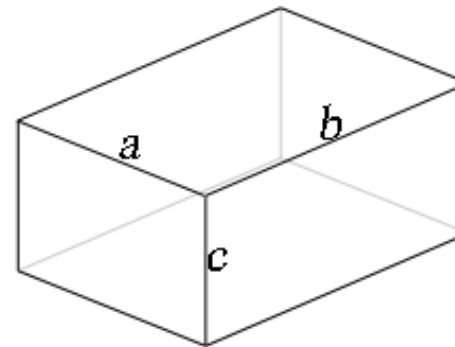
In cylindrical shape, the surface area is the smallest to a given volume when the height is equal to the diameter of the cylinder. This is a reason why most of the power reactors are designed to have such a geometrical proportion. Such a cylinder is named the ideal cylinder.

In rectangular shape, the surface area is the smallest to the given volume when it is cubic.

Note: It might be good mathematical drill to prove the above two statements.



Ideal cylinder
 $h=2r$



Cube
 $a=b=c$

RELATION BETWEEN SURFACE AREA AND VOLUME

A simple mathematical relation, $S = kV^{2/3}$, exists between the surface area (S) and the volume (V) of all the objects such as sphere, ideal cylinder or cube, where k is a proper constant to the specific shape.

$S=kV^{2/3}$	Sphere	Ideal cylinder	Cube
k	4.84	5.54	6

From the above table, it is understood that spherical shape has the smallest surface area to the given volume.

EFFECTIVE REACTOR CORE SIZE

Reactor type		BWR	PWR	GCR	CANDU
Electric Power (MWe)		1100 MWe (BWR-5, Tokai-2)	1160 MWe (4-loop, Tsuruga-2)	166 MWe (Tokai-1)	935 MWe (Darlington)
Effective core size	Diameter, D (m)	4.75	3.37	11.67	7.06
	Height, H (m)	3.71	3.66	6.76	5.95
Nondimensional core size	D/M^*	54	51	37	35
	H/M^*	42	56	22	30

* M is the migration length derived from M^2 of the reactor on sheet 83.

CRITICAL MASS (1/2)

The total number of neutrons leaking out of the reactor (N_{LEAK}) is proportional to the surface area of the reactor (S).

According to the relation of $S = kV^{2/3}$,

$$N_{\text{LEAK}} \propto V^{2/3}.$$

Then, the total number of neutrons leaking out of the reactor (N_{LEAK}) per reactor volume ($N_{\text{LEAK}}/V = n_{\text{LEAK}}$) is proportional to $V^{-1/3}$.

That is,

$$n_{\text{LEAK}} \propto V^{-1/3}.$$

The production rate of excess neutrons per reactor volume is constant, while the number of leakage neutrons per reactor volume decreases with the increase of the reactor volume.

CRITICAL MASS (2/2)

Based on the above observation, it is understood that the total number of leakage neutrons from the reactor could be smaller than the total number of excess neutrons produced in the reactor by increasing the reactor size.

There is a size at which the total number of leakage neutrons from the reactor (N_{LEAK}) is balanced with the total number of excess neutrons produced in the reactor (N_{EXCESS}). That is,

$$N_{\text{LEAK}} = N_{\text{EXCESS}}$$

This reactor size is called critical size or critical mass.

The smaller the number of excess neutrons per volume, the larger the critical size.

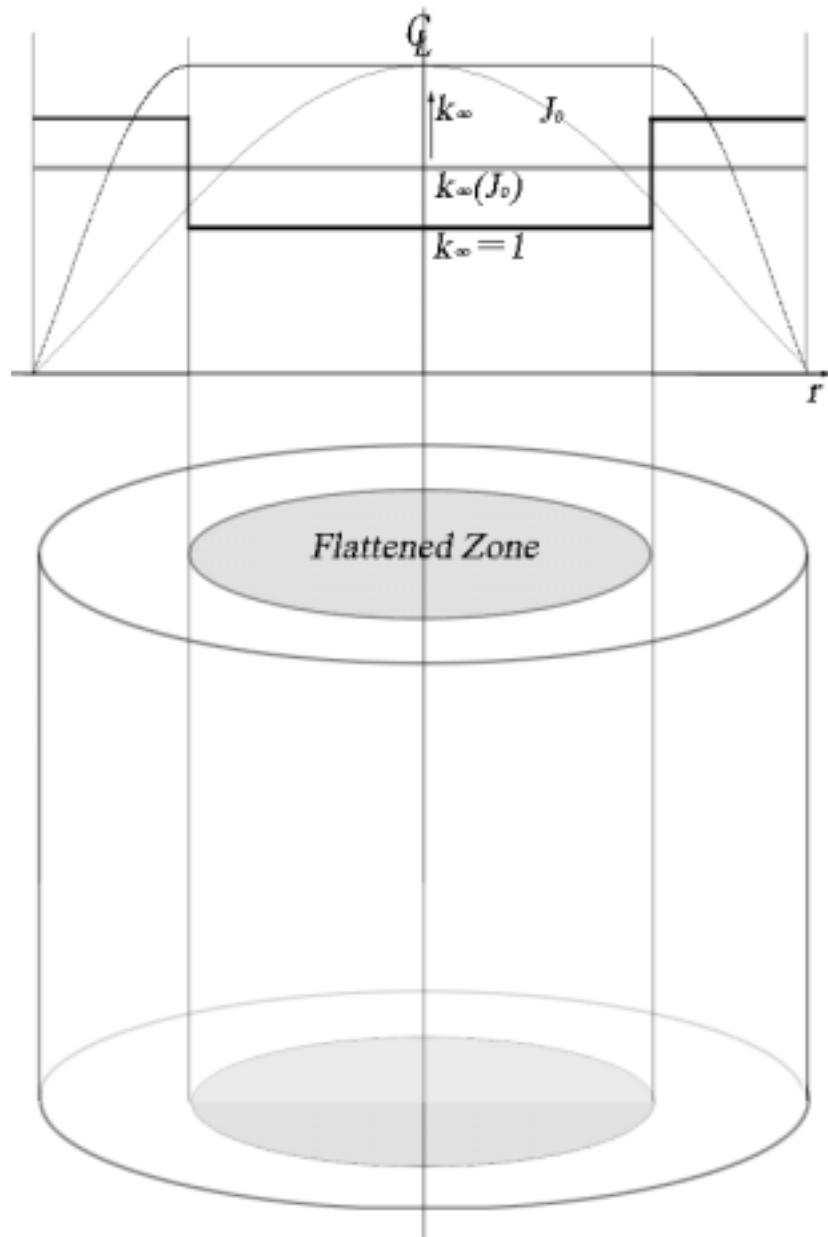
PRACTICAL SIZE OF POWER REACTORS

Power reactors are designed to be much larger than the critical size.

In general, the size of power reactors is several times larger than the critical size.

Accordingly the excess neutrons have to be absorbed by neutron absorbers such as control rods to achieve neutron balance with the leakage rate; otherwise the reactor would be supercritical.

POWER FLATTENING



From the safety viewpoint, a maximum allowable thermal power is set to fuel element.

In order to achieve the largest power production from a given reactor, it is desirable to make as many fuel elements as possible produce the maximum allowable thermal power.

Practically it is achieved by power flattening, or flattened neutron distribution.

REFLECTOR

From the viewpoint of neutron economy, it is desirable to reduce the amount of neutron leakage out of reactor.

For this purpose, nuclear reactors are wrapped by a layer of moderator material. This layer is called reflector.

Neutrons coming out of reactor surface are partly scattered back by the reflector to the inside of reactor.

ESSENCE OF THERMAL REACTOR THEORY

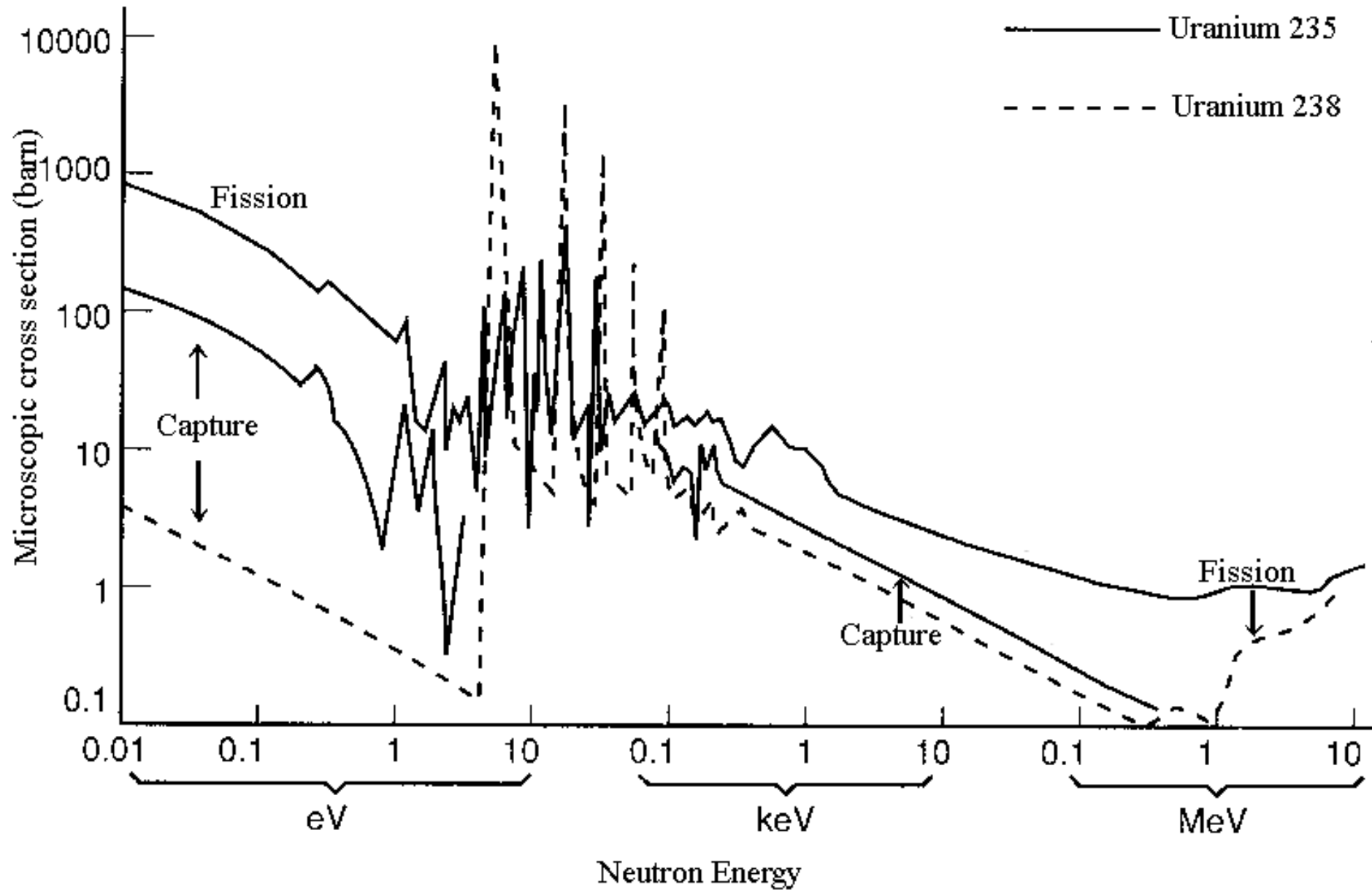
ROLE OF MODERATOR

Nuclear chain reaction can not be sustained in a lump of natural uranium. The fact is understood by observing the neutron cross sections of U-235 and U-238 as well as the energy spectrum of fission neutrons.

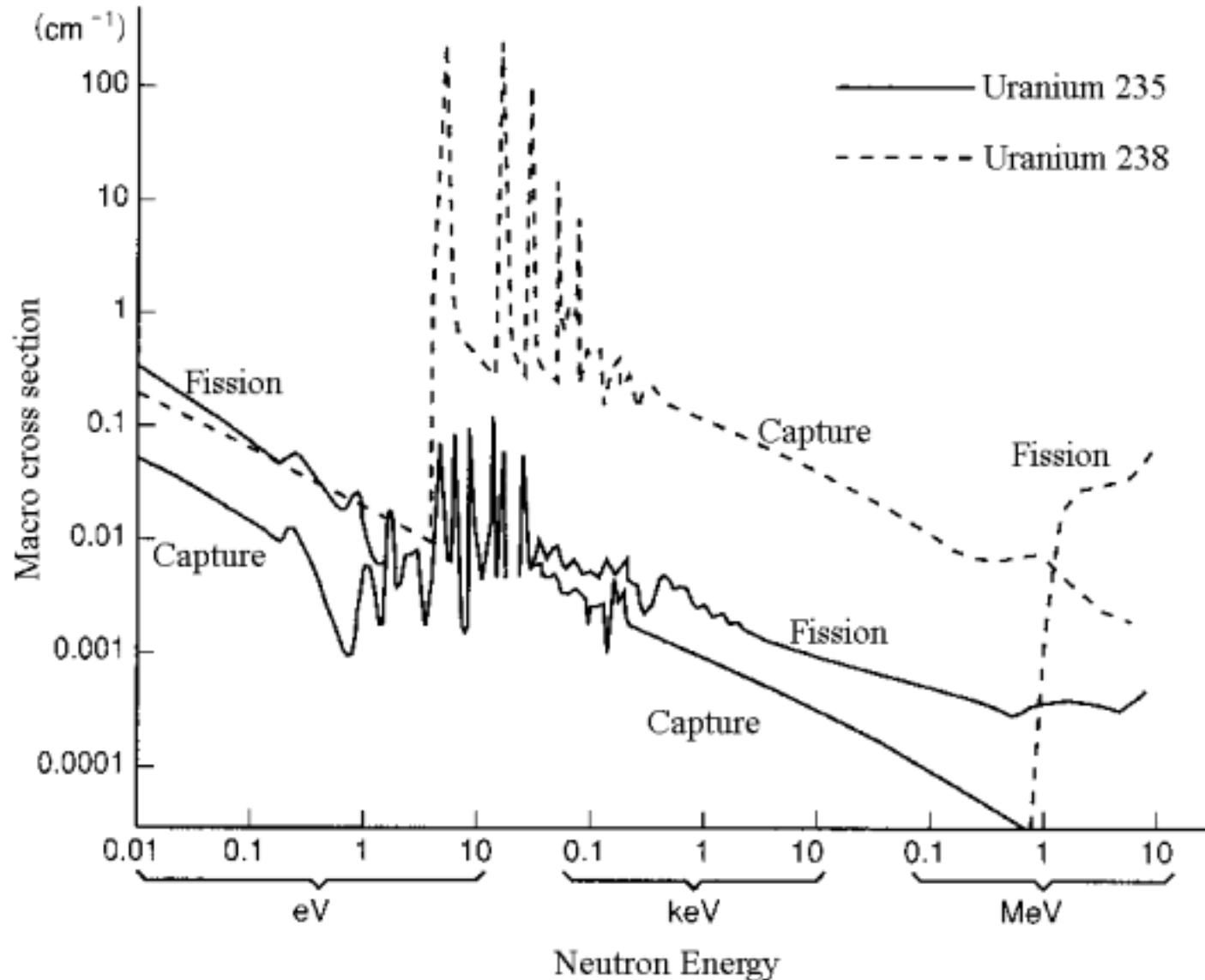
In a natural uranium system, nuclear chain reaction could be sustained if neutrons involved in the nuclear reaction had low kinetic energy.

Role of moderator is to reduce neutron kinetic energy to achieve the sustained chain reaction.

(Further discussion will be given later.)



Neutron cross sections of U-235 and U-238
 (日本原子力学会編「原子力がひらく世紀」159頁より)



Neutron cross sections of natural uranium

(日本原子力学会編「原子力がひらく世紀」159頁より)

Note: Natural uranium consists of 0.7% of U-235 and 99.3% of U-238. Considering this fact, neutron cross sections of natural uranium can be constructed as the above graph from the figure on sheet 66.

KINETIC ENERGY OF FISSION NEUTRONS

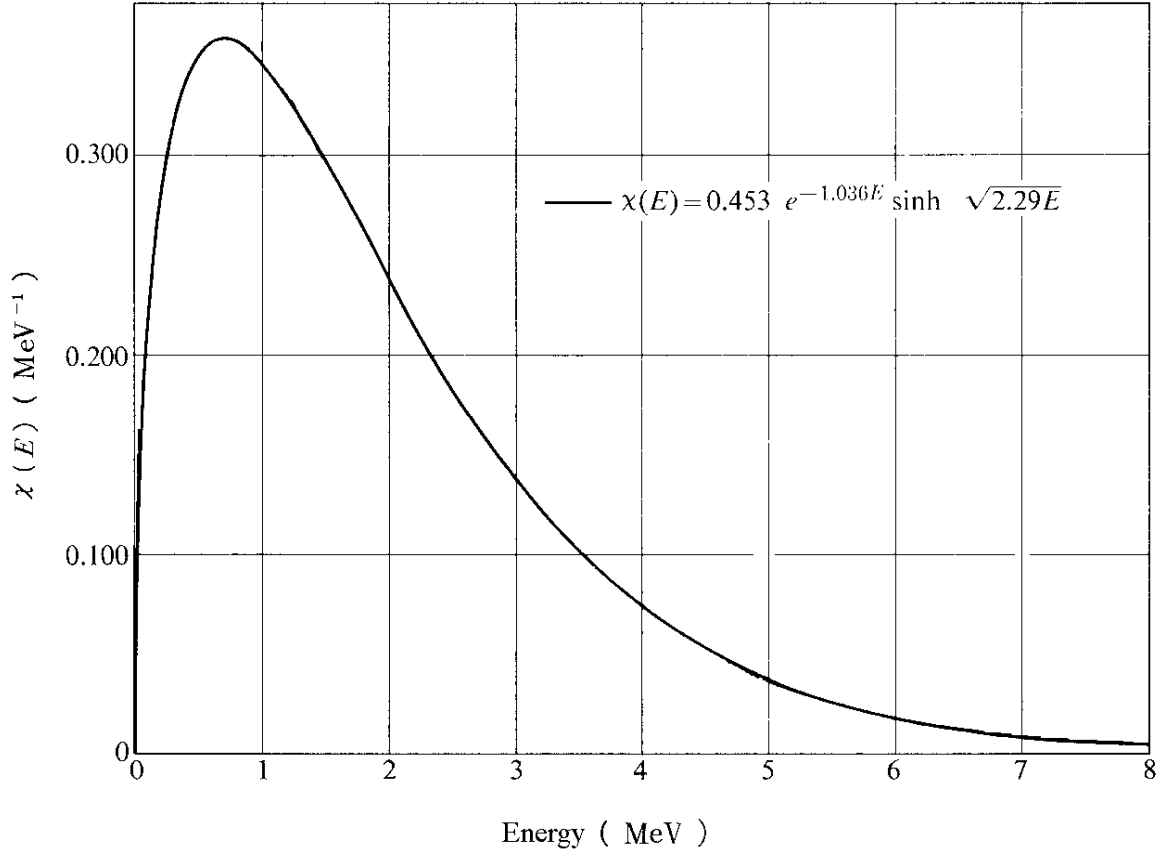
Kinetic energy of fission neutrons distributes from zero MeV to several MeV shown in the figure on sheet 69.

The average kinetic energy of fission neutrons is 1.98 MeV.

The most probable kinetic energy of fission neutrons is 0.73 MeV.

Average energy of delayed neutron released from thermal fission of U^{235}

Group	Energy (keV)
1	250
2	560
3	405
4	450
5	—
6	—



Energy spectrum of prompt fission neutrons

EFFECT OF U-238 IN CHAIN REACTION

99.3% of natural uranium is U-238.

U-238 is fissionable only for high energy neutrons. Very few fission neutrons can induce fission.

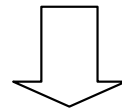
Most of fission neutrons absorbed by U-238 are lost without making nuclear fission.

Avoiding this situation, fission neutrons should be slowed down quickly to the level of eV, where the adverse effect of U-238 is comparatively low. (See the figure on sheet 67.)

PREREQUISITES TO MODERATOR

Neutrons are slowed down by collisions with moderator atoms.

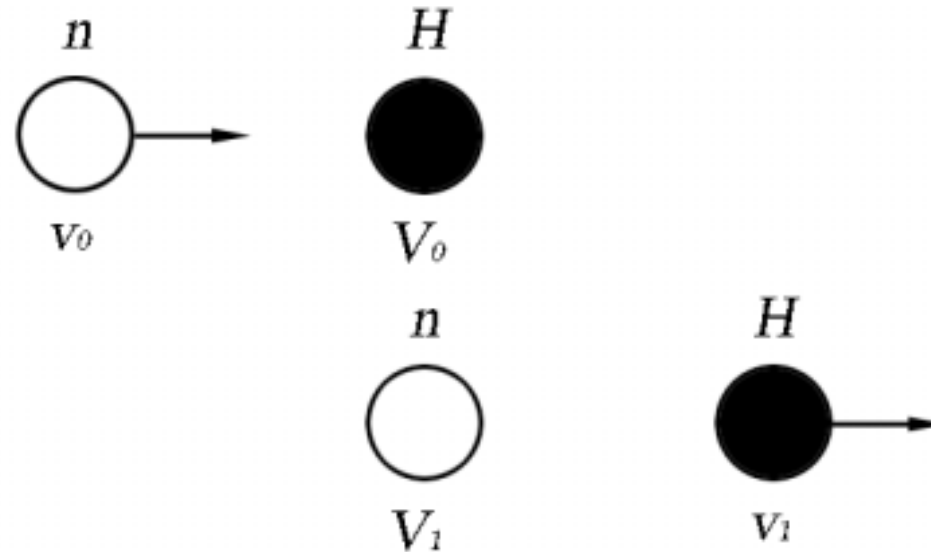
During slowing-down process, neutrons are lost by absorption and leakage.



- (1) The necessary number of collisions with moderator atoms for thermalizing fission neutrons should be as small as possible.
- (2) The neutron collision probability with moderator atoms should be as large as possible.
- (3) The neutron absorption probability by moderator atoms should be as small as possible.

NEUTRON COLLISION WITH HYDROGEN ATOM

Neutron collision with hydrogen nucleus ($A=1$)



Speed before collision :

neutron : v_0

hydrogen nucleus : $V_0 = 0$

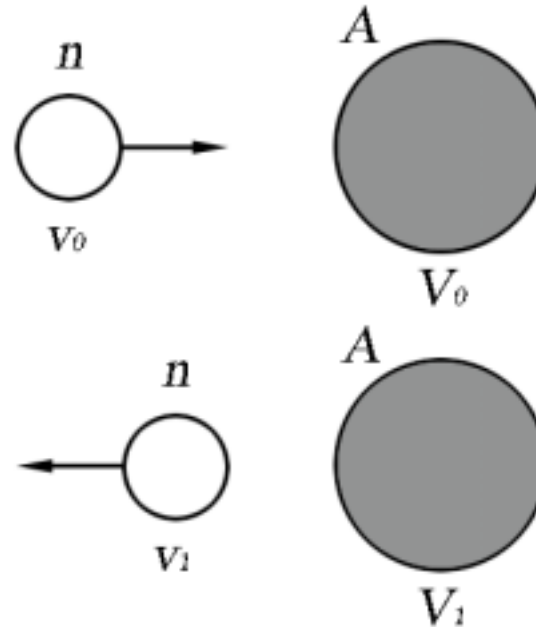
Speed after collision :

neutron : $-\{(A - 1)/(A+1)\} \cdot v_0$ $v_1 = 0$

hydrogen nucleus : $\{2/(A+1)\} \cdot v_0$ $V_1 = v_0$

NEUTRON COLLISION WITH NUCLEUS

Neutron collision with moderator nucleus ($A > 1$)



Speed before collision :

neutron : v_0

target nucleus : $V_0 = 0$

Speed after collision :

neutron : $v_1 = - \left\{ \frac{A - 1}{A + 1} \right\} \cdot v_0$

at A , $v_1 = - v_0$

target nucleus : $V_1 = \left\{ \frac{2}{A + 1} \right\} \cdot v_0$

at A , $V_1 = 0$

NEUTRON COLLISION WITH MODERATOR ATOM (1/2)

Any moderator atom has larger mass than a neutron except hydrogen atom. (A hydrogen atom has the same mass as a neutron.)

A moving neutron is brought to a standstill by a head-on collision with a standing hydrogen atom. (This is the extreme case of neutron energy loss by a collision with moderator atom.)

In general, a moving neutron reduces its speed to some extent by a collision with a moderator atom depending on the mode of collision.

Note: It is implicitly assumed that the kinetic energy of the moderator atom is small enough to be neglected.

NEUTRON COLLISION WITH MODERATOR ATOM (2/2)

The ratio of before to after collision in neutron energy is almost independent of the neutron energy before collision.

For practical convenience the natural logarithm of this ratio is defined and denoted by a Greek letter ξ .

Note: More exactly speaking, ξ is the mean value of individual ξ value corresponding to all the possible collision modes.

A larger ξ means that the average energy loss of neutron by a single collision with the moderator atom is larger.

In other words, ξ is a good index of moderator prerequisite property.

THREE INDICES TO MODERATOR (1/2)

- (1) Larger ξ : This means that energy loss of neutron by a single collision with the moderator atom is large.
- (2) Larger λ_{tr} : This means that a neutron needs not to move so far to achieve a collision.
This means that neutron loss by leakage during moving around can be limited.
- (3) Smaller Σ_a : This means that neutron loss by reacting moderator atoms during slowing-down process is smaller.

THREE INDICES TO MODERATOR

(2/2)

In order to reveal the essential quality of moderator, two new indices are devised on the three basic indices ξ , s and a .

- 1) ξs : The product of ξ and s is more appropriate to express the neutron slowing-down capability of moderators.
- 2) s/a : The ratio of s to a is the most appropriate to express the overall quality of moderator because this value means neutron slowing-down capability per unit neutron loss during the process.

The set of three indices such as ξ , s and s/a are frequently used to show the quality of moderator.

MAJOR MATERIALS FOR MODERATOR

Moderator	Mass number		Necessary number of collisions [†]	$s^{\dagger\dagger}$	s/a
H ₂ O	-	0.92	16	1.35	71
D ₂ O	-	0.509	29	0.176	5,670
Be	9	0.209	69	0.158	143
Graphite	-	0.158	119	0.064	170
Paraffin	-	0.913	21	1.63	62.4
²³⁸ U (reference)	238	0.008	1,730	0.003	0.0092

[†] The necessary number of collisions with moderator atoms for slowing-down from fission neutrons (2.0MeV) to near thermal neutron (1ev)

^{††} cm⁻¹

MODERATION PROPERTIES OF MODERATOR MATERIAL AND SIMPLE SUBSTANCE

Moderator		Moderator properties				
Simple substance	Moderator material	Mass number		Necessary number of collisions [†]	$s^{\dagger\dagger}$	s / a
H		1	1	14	82.02	247
	H₂O	-	0.92	16	40.35	71
D		2	0.725	20	5.54	10672
	D₂O	-	0.509	29	5.29	5670

† The necessary number of collisions with moderator atoms for slowing-down from fission neutrons (2.0MeV) to near thermal neutron (1ev).

†† barn

NEUTRON MIGRATION (1/2)

Neutrons move around in the reactor during the slowing down process.

- *We introduce a new parameter, τ , to express the total traveling distance of the neutron during the slowing down process.*
- *τ has the dimension of area (cm^2) though it is proportional to the mean value of the total traveling distance of neutrons during their slowing down process.*

Thermalized neutrons move around in the reactor before being absorbed.

- *We introduce a new parameter, L^2 to express the square of crow-flight distance of the neutron before being absorbed.*
- *L^2 has the dimension of area (cm^2), which is proportional to the mean value of the square of crow-flight distance of neutrons before being absorbed.*

τ is called Fermi age, and L^2 is called diffusion area.

NEUTRON MIGRATION (2/2)

How quickly thermal neutrons are absorbed mainly depends on the absorption capability of fuels in the reactor. As a result, the value of diffusion area (L^2) *depends on the reactor structure rather than moderator itself.*

In comparison with L^2 , *the value* of Fermi age (τ) is determined by the property of moderator.

Fermi age (τ) and diffusion area (L^2) *have the same dimension of area though the definition of them are different.*

- *We introduce a new parameter M^2 , which is called migration area.*

$$M^2 = \tau + L^2$$

- Square root of migration area is called migration length (M).

MIGRATION AREA AND REACTOR SIZE

Larger migration area means that neutrons move around larger area.

When neutrons move around larger area, the probability of the neutron leakage out of reactor tends to be higher.

Let's imagine two reactors which have the same geometrical size but different migration area. The reactor having larger migration area loses more neutrons than the other does.

As a result, the size of the reactor having larger migration area tends to be larger.

NEUTRON MIGRATION CHARACTERISTICS IN THE REACTOR

		Fermi age (cm^2)	Diffusion area L^2 (cm^2)	Migration area M^2 (cm^2)
<i>Light Water</i>		<i>37</i>	<i>7.3</i>	<i>44.3</i>
PWR		40	3.24	43.24
BWR	void : 0%	50	4	54
	void : 40%	72	5	77
<i>Heavy Water</i>		<i>111</i>	<i>12100</i>	<i>12211</i>
CANDU		155	241	396
<i>Graphite</i>		<i>312</i>	<i>2809</i>	<i>3121</i>
GCR				972
HTGR		300	144	444

Note : These reactor-specific values are general ones of the reactor.

THERMAL REACTOR

Fission neutrons lose their energy by successive collisions with moderator atoms. In other words, high speed neutrons reduce their speed by successive collisions with light atoms in the moderator.

Neutron motion finally reaches thermal equilibrium with moderator atoms. Neutrons reached thermal equilibrium with moderator are called “*THERMAL NEUTRON*”.

Chain reaction is sustained mainly by thermal neutrons in most of the modern power reactors, which is called “*THERMAL REACTOR*”.

HETEROGENEOUS STRUCTURE

(1/3)

Lattice structure is the most distinctive feature of thermal reactors. It gives heterogeneity to reactors.

In the thermal reactors, low energy neutrons suffer strong resonance absorption by U-238 during slowing-down process.

Neutron loss by U-238 resonance absorption during slowing-down process should be minimized to achieve a good neutron economy.

The basic idea for minimizing the resonance absorption is to separate the neutron slowing-down region from fuel (U-238).

HETEROGENEOUS STRUCTURE

(2/3)

If lumps of fuel were placed in moderator in lattice, fission neutrons could easily leave fuel lumps and enter the moderator surrounding fuel lumps.

Once neutrons enter the moderator, the neutrons will lose energy quickly by collisions with moderator atoms. During the slowing-down process, probability of neutron collision with U-238 atoms is low because of the separation of fuel lumps from moderator.

Lattice structure is an essential means to reduce the resonance absorption of neutrons in thermal reactors.

HETEROGENEOUS STRUCTURE

(3/3)

Remarks on the heterogeneity of thermal reactors

The idea of lattice structure was acquired at very early stage of nuclear reactor development. (see the picture on sheet 5)

In July 1941, before the construction of CP-1 was started, the study of graphite-uranium lattice piles was begun at Columbia in USA. It was one of the most important steps toward the success in sustained chain reaction.

If U-238 were not a strong resonance absorber, thermal reactors could be designed less heterogeneously. Actually thorium cycle thermal reactors can be designed as a semi-homogeneous system.

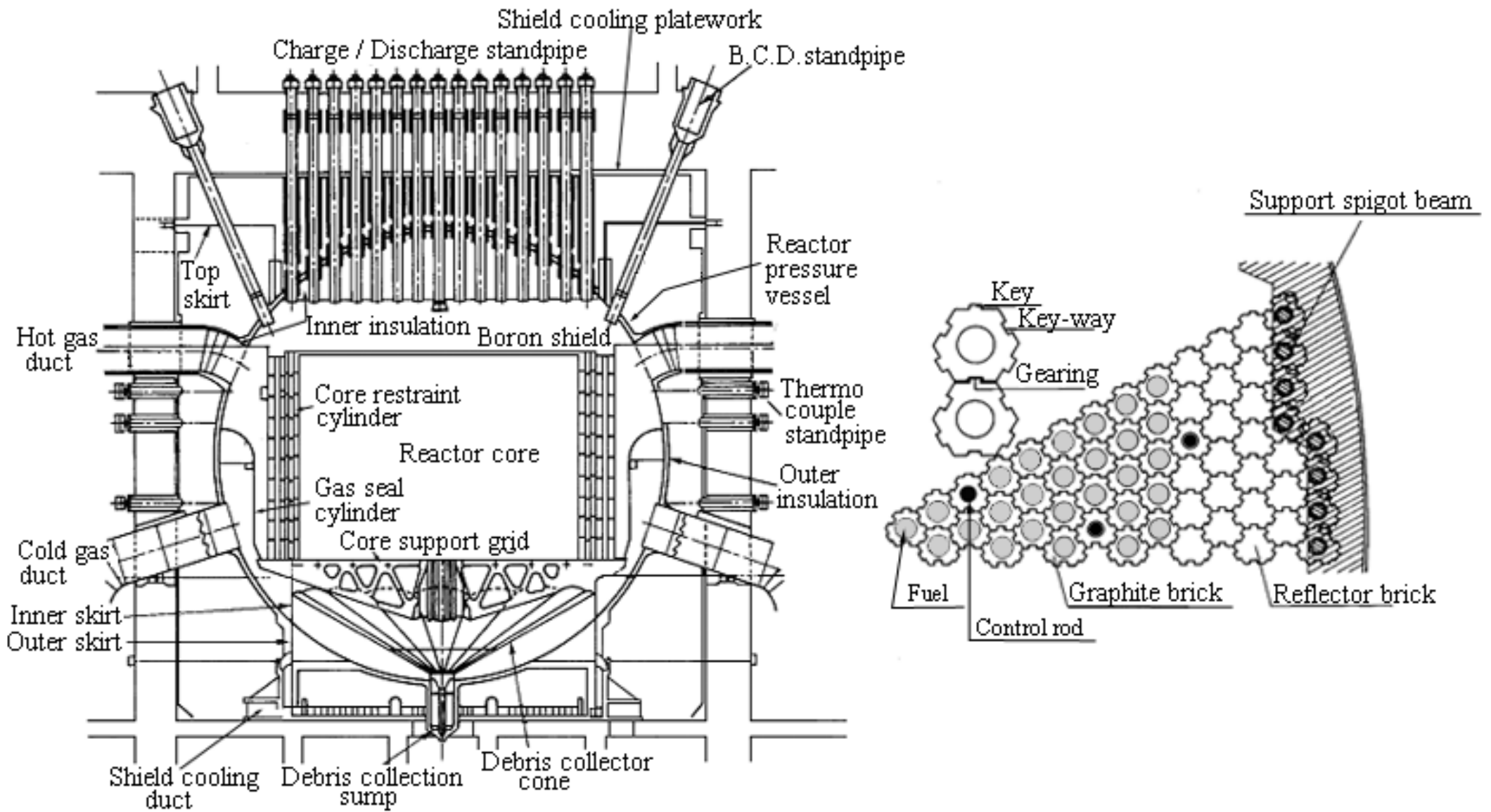
ENRICHED URANIUM

Enriched uranium contains more U-235 compared with natural uranium. For example, typical low enriched uranium for light water reactors consists of 3 ~ 5% of U-235 and 97 ~ 95% of U-238.

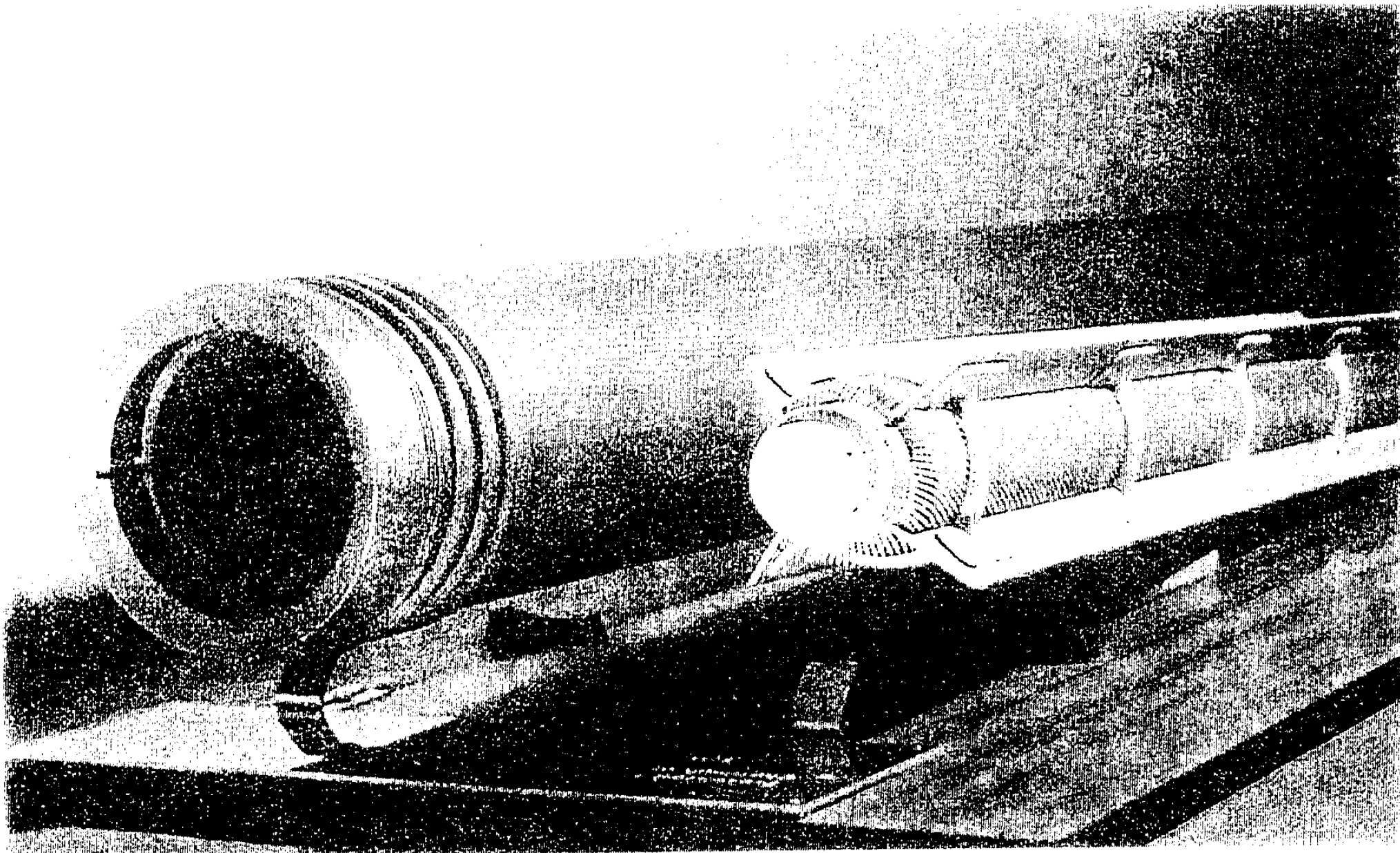
Enriched uranium is more favorable to fuel because the adverse effect of U-238 is compensated by enhanced U-235 density.

Enriched uranium could give more freedom to reactor design.

As a result, less heterogeneous reactors could be designed. It means that compact reactor core design is allowed. More absorption material could be introduced in reactors. It means that strong fuel cladding is allowed.



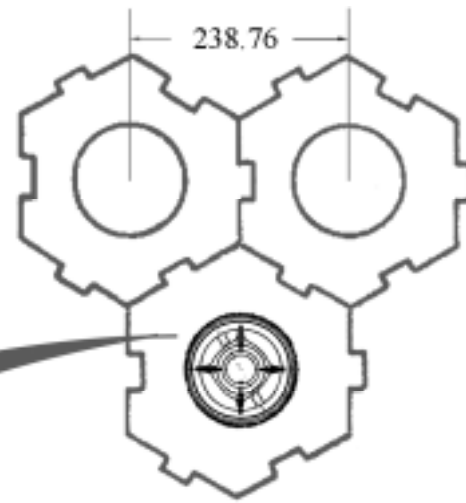
Heterogeneous structure of graphite-moderated gas-cooled reactor core



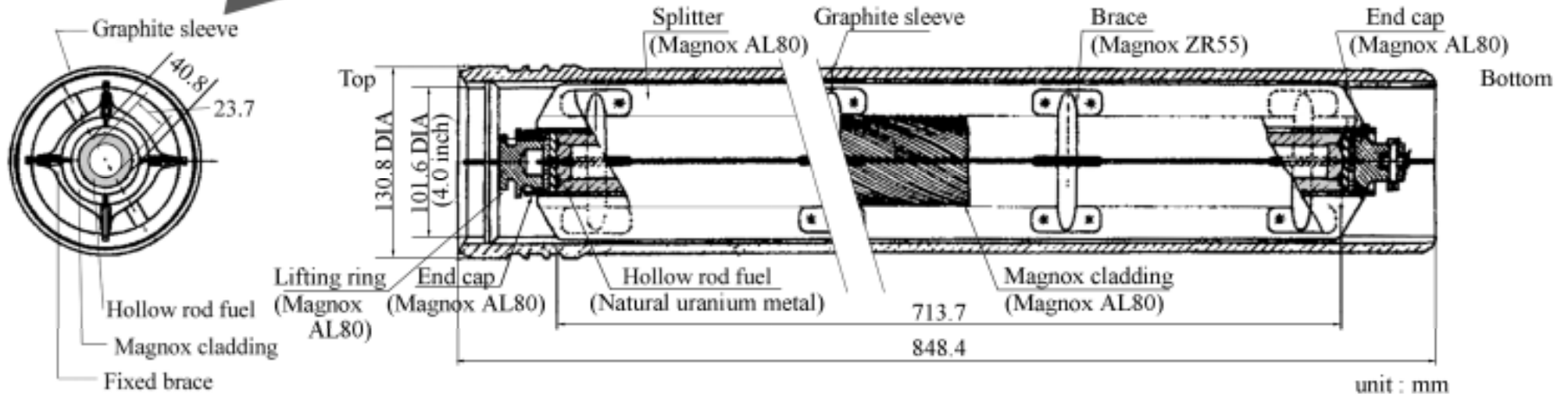
Fuel element of graphite-moderated gas-cooled reactor (from the top)



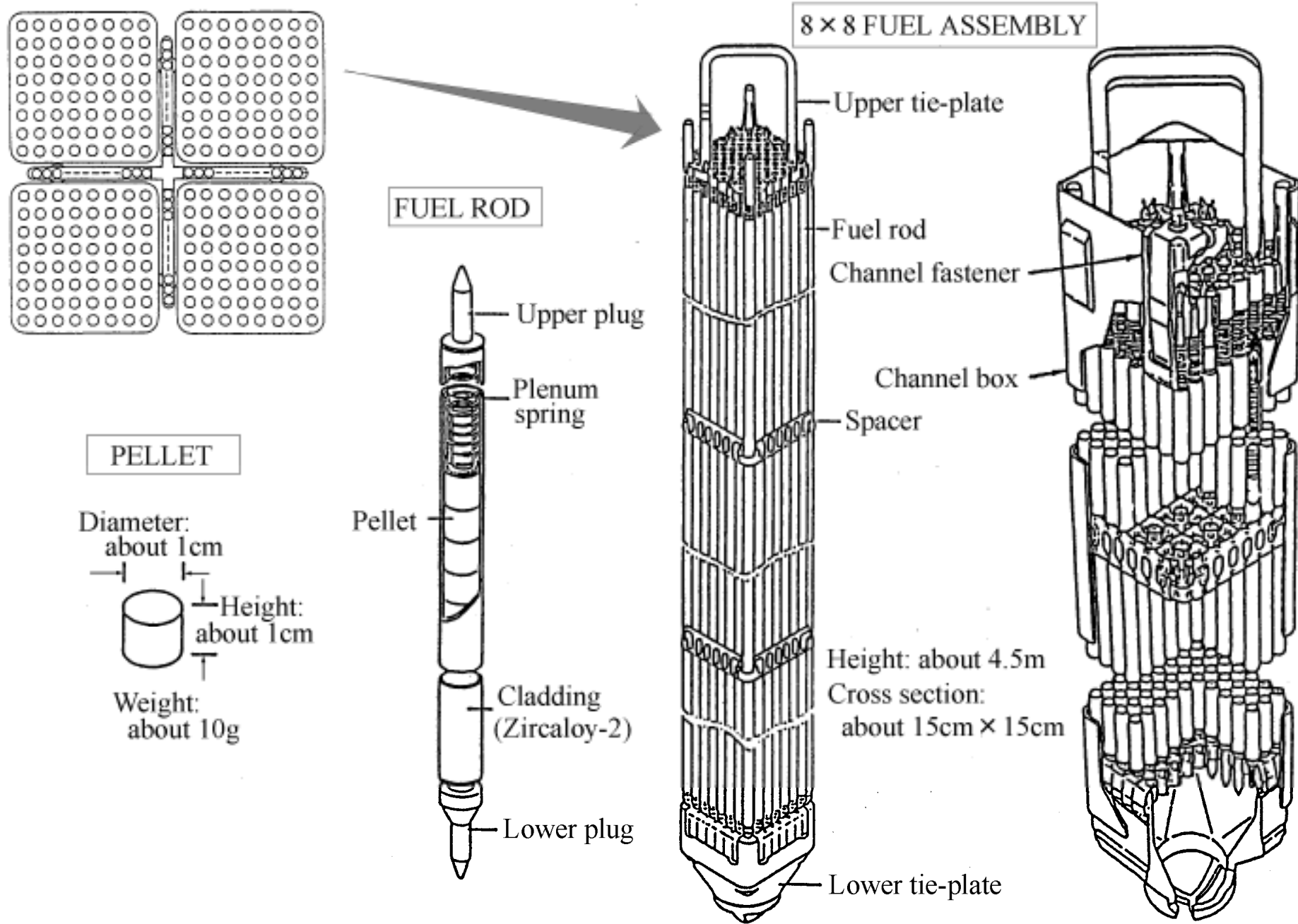
Fuel element of graphite-moderated gas-cooled reactor (from the bottom)



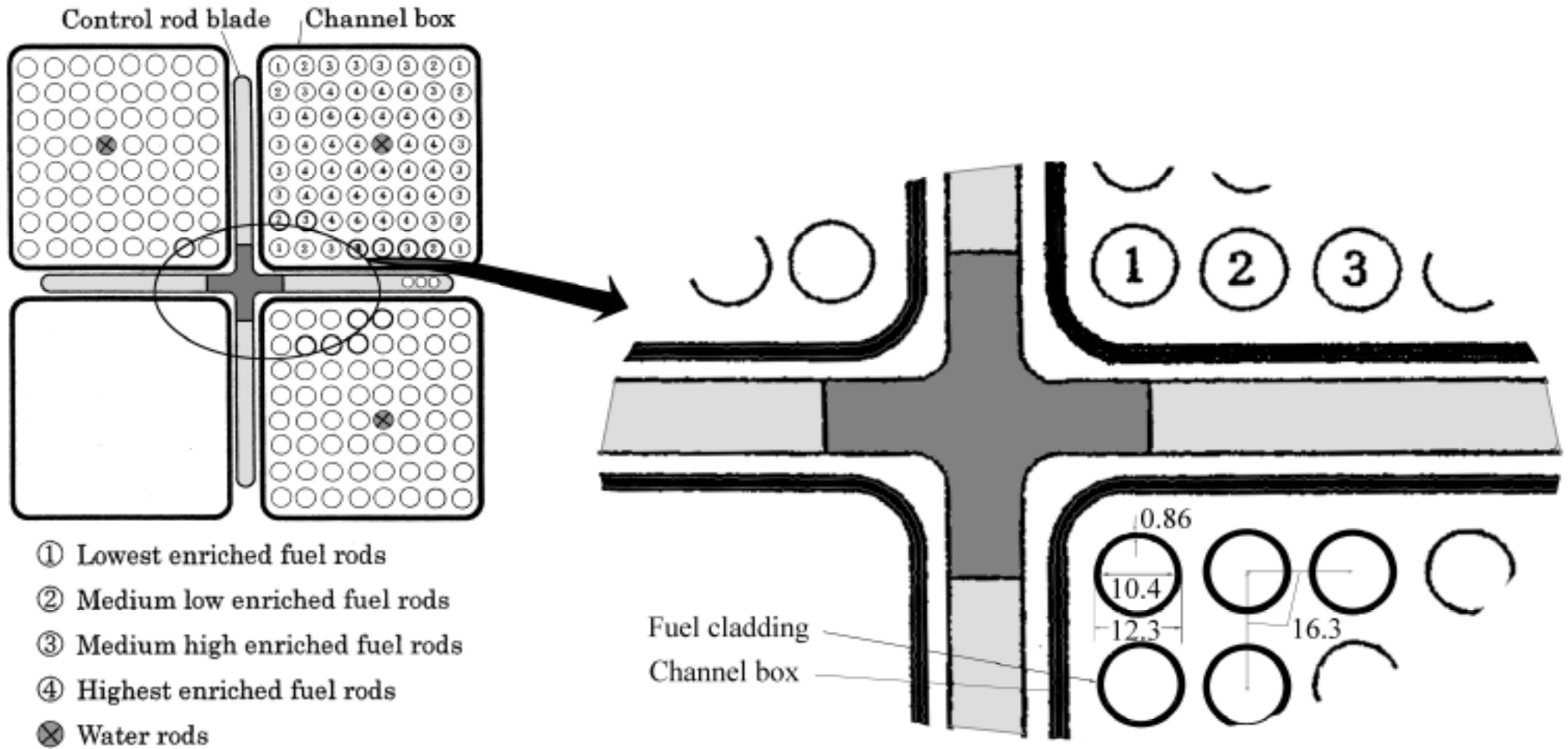
Triangular lattice structure



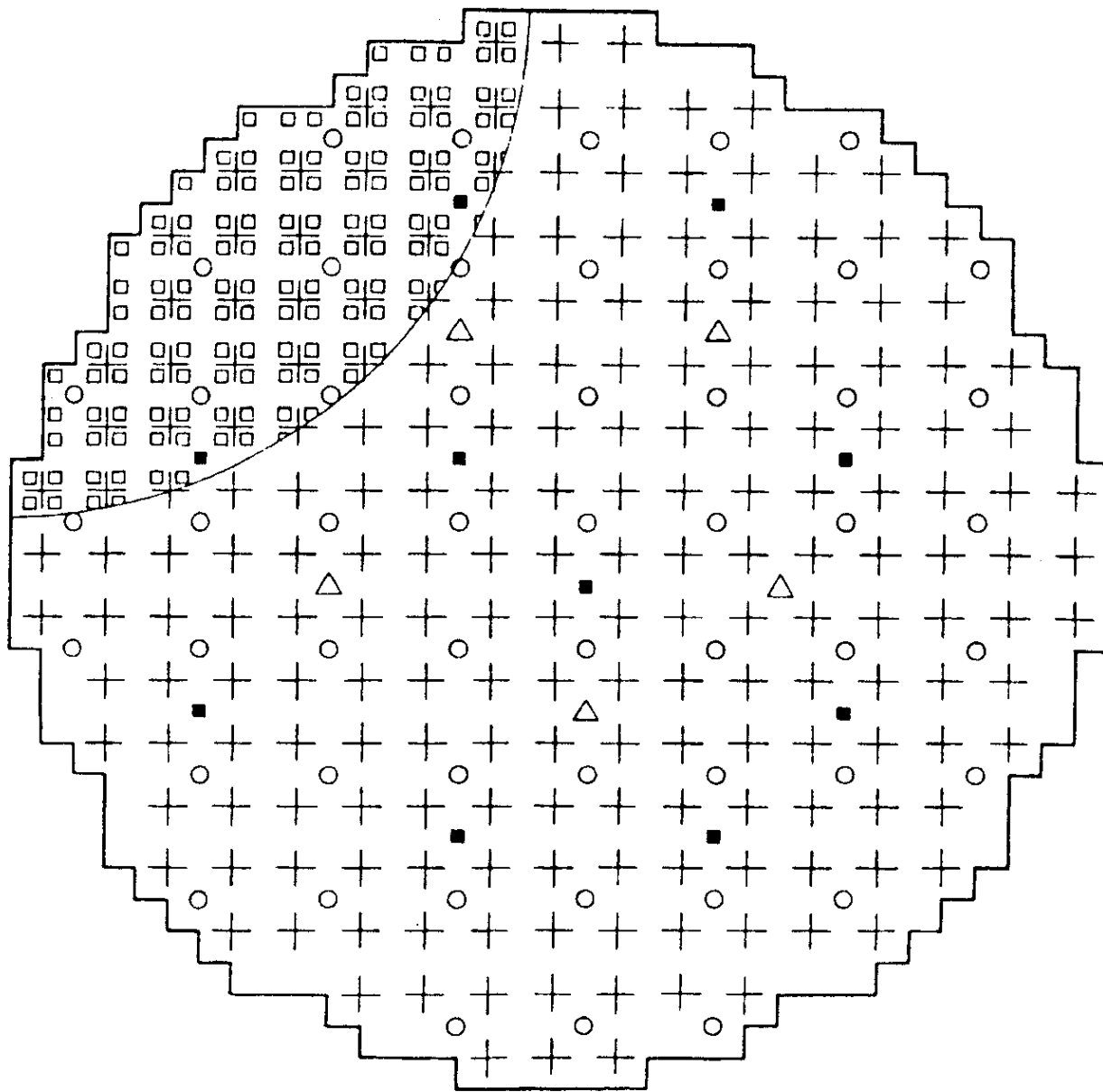
Hollow rod fuel structure (4.0 inch)



Fuel element of BWR

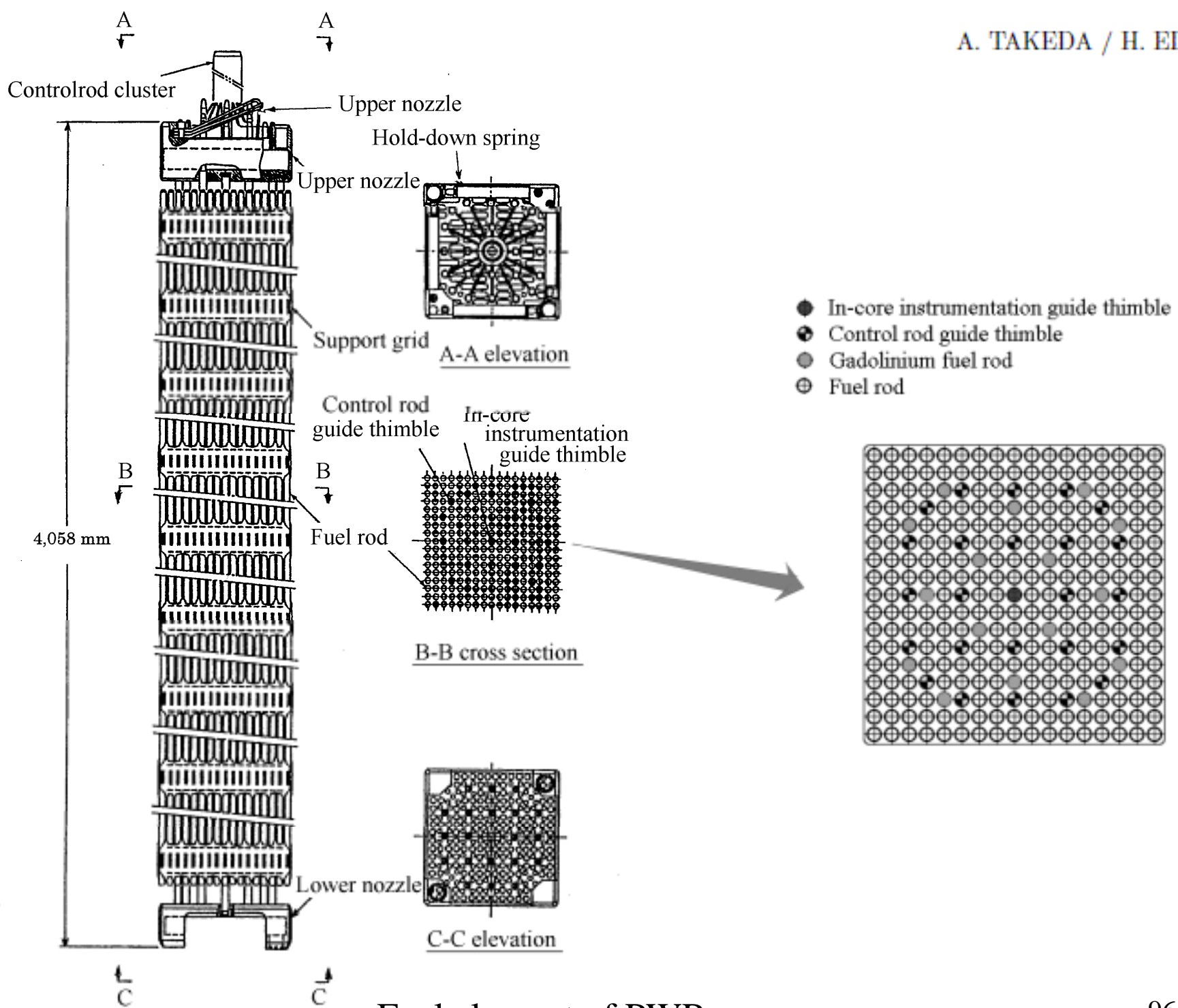


Square lattice structure (BWR)

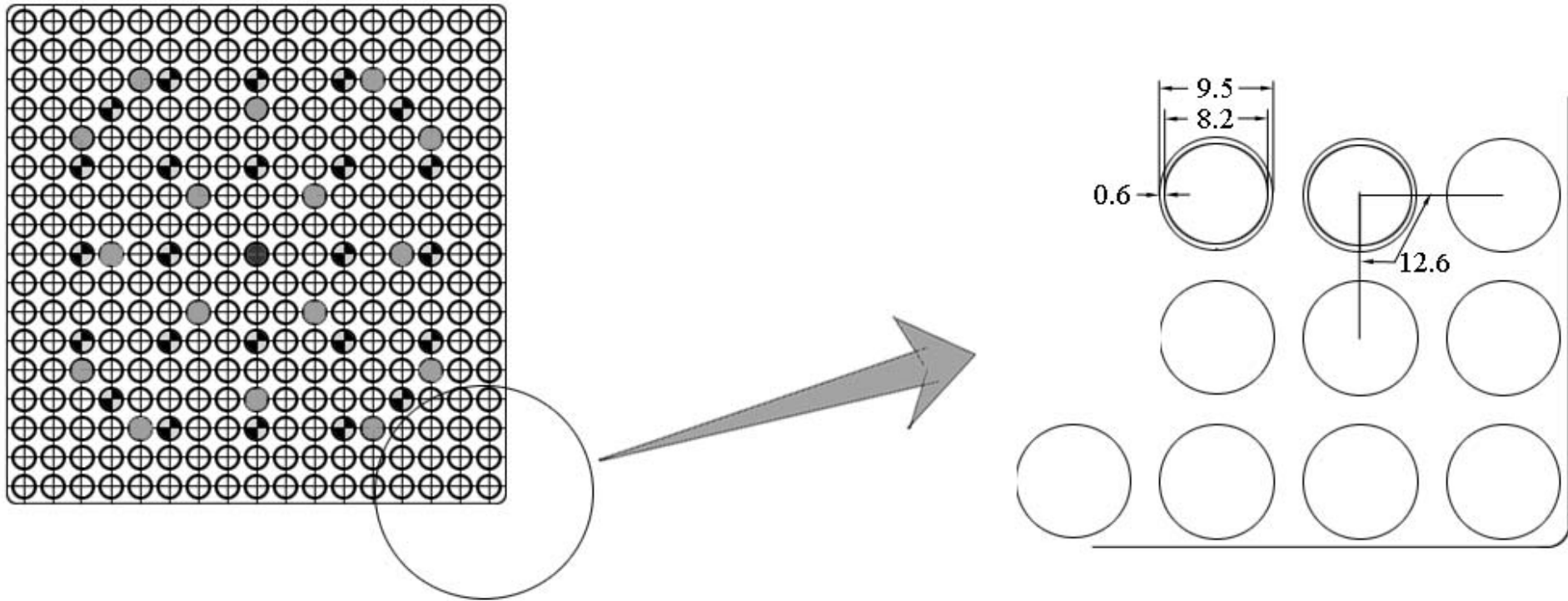


□	Fuel assembly	872	○	Power range Monitor (PRM)	52×4
+	Control rod	205	■	Source range monitor (SRNM)	10
			△	Neutron source	5

Heterogeneous structure of BWR

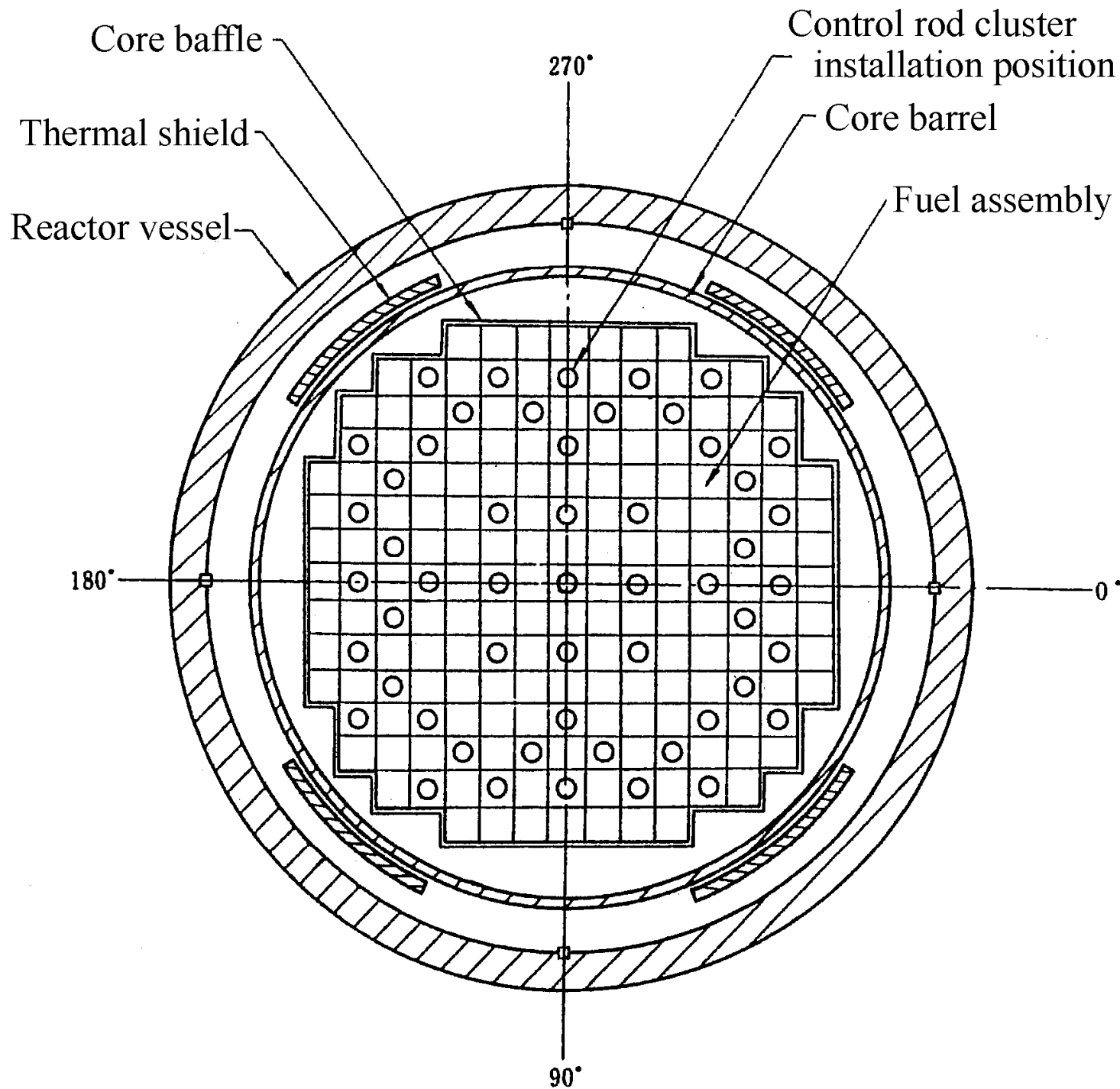


Fuel element of PWR



- In-core instrumentation guide thimble
- ⊕ Control rod guide thimble
- Gadolinium fuel rod
- ⊕ Fuel rod

Square lattice structure (PWR)



Heterogeneous structure of PWR

HETEROGENEITY OF VARIOUS THERMAL REACTORS

	PWR	BWR	GCR
Fuel diameter (mm)	8.2	10.4	40.8 / 23.7*
Lattice pitch (mm)	12.6	16.3	238.8
Lattice	Square	Square	Triangle
Enrichment (%)	4.1	3.5	0.7 (Natural)
Moderator	Light water	Light water	Graphite

* Inner diameter of hollow rod fuel

FEASIBLE THERMAL REACTORS

	Moderator		
	Heavy water	Graphite	Light
Natural uranium (Oxide)		×	×
Natural uranium (Metal)			×
Enriched uranium (Oxide)			

Note: Selection of coolant will change the above feasibility and make the problem more complicated.

THERMAL POWER DENSITY AND REACTOR TYPE

In general, when we want to get larger thermal power out of a reactor, the reactor is made larger, because thermal power density of the reactor belonging to a reactor type does not vary so much with the design.

Thermal power density of reactor, on the contrary, varies greatly with the reactor type.

Thermal power density of reactor is closely related with fuel design and the properties of coolant and moderator.

Higher thermal power density of reactor is appreciated because it makes it possible to achieve good economy of the plant.

REACTOR THERMAL POWER DENSITY

	GCR	CANDU	BWR	PWR
Early design (kW/l)	0.56	9.5	40.6	70.5
	(Bradwell)	(Pickering-A)	(BWR-2, Tsuruga-1)	(2-loop, Mihama-1)
Matured design (kW/l)	0.76	11.9	50.6	104.9
	(Wylfa)	(Darlington)	(BWR-6, Kashiwazaki-1)	(4-loop, Tsuruga-2)

THERMAL POWER DENSITY OF GRAPHITE-MODERATED REACTORS

GCR			AGR	HTGR		RBMK
CO ₂ cooled,			CO ₂ cooled,	He cooled, Th cycle,		Light water cooled,
Natural U metal fuel			Slightly enriched oxide fuel	coated particle fuel		Channel-type BWR
Bradwell	Wylfa	Tokai-1	-	Fort St Vrain	THTR	Ignalina
(Early design)	(Matured design)	(Hollow rod fuel)		(Block type fuel)	(Ball type fuel)	(RBMK-1500)
0.56	0.76 (kW/l)	0.81 (kW/l)	2.6 (kW/l)	6.3 (kW/l)	6.0 (kW/l)	6.3 (kW/l)

STRUCTURE OF NUCLEAR REACTORS

BASIC STRUCTURE OF NUCLEAR REACTORS

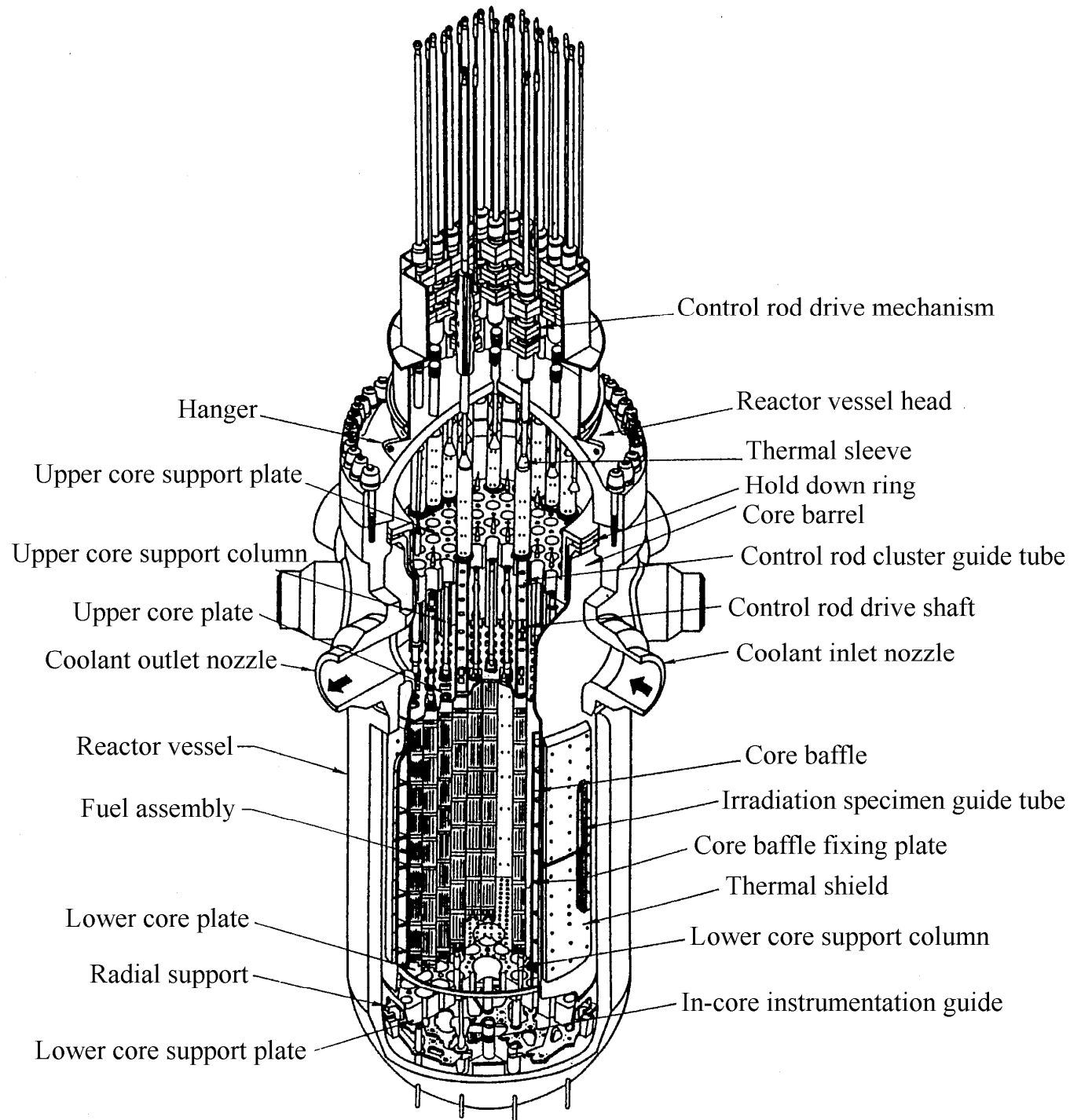
Fuel, moderator and coolant are the three basic components of nuclear reactors.

Fuel	Natural uranium, Enriched uranium, Plutonium Metallic fuel, Oxide fuel
Moderator*	Graphite, Heavy water, Light water
Coolant	Carbon dioxide, Helium, Heavy water, Light water Liquid metal

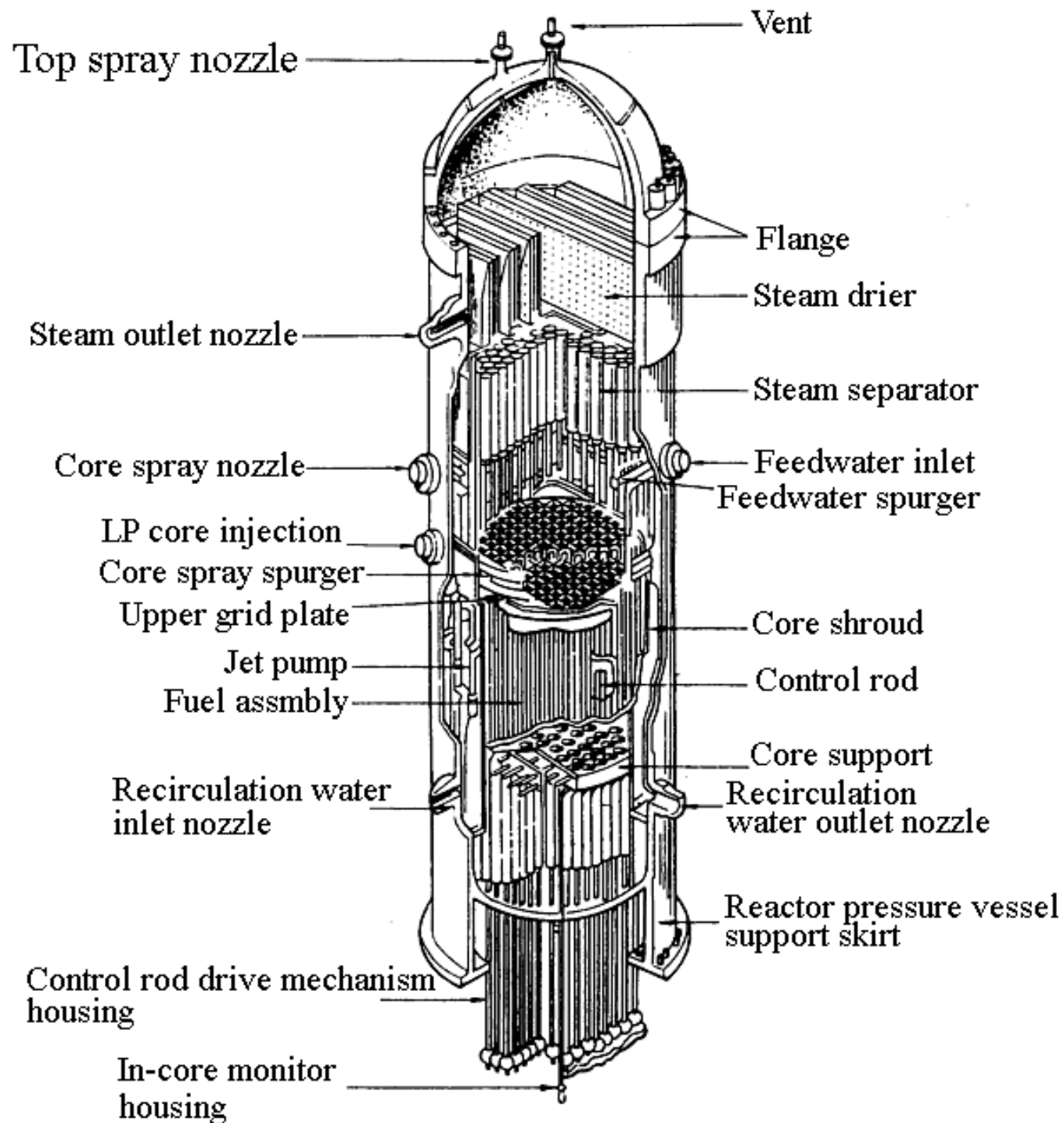
* Moderator is specific to thermal reactors while fast reactor has no moderator.

TYPICAL NUCLEAR POWER REACTORS IN THE WORLD

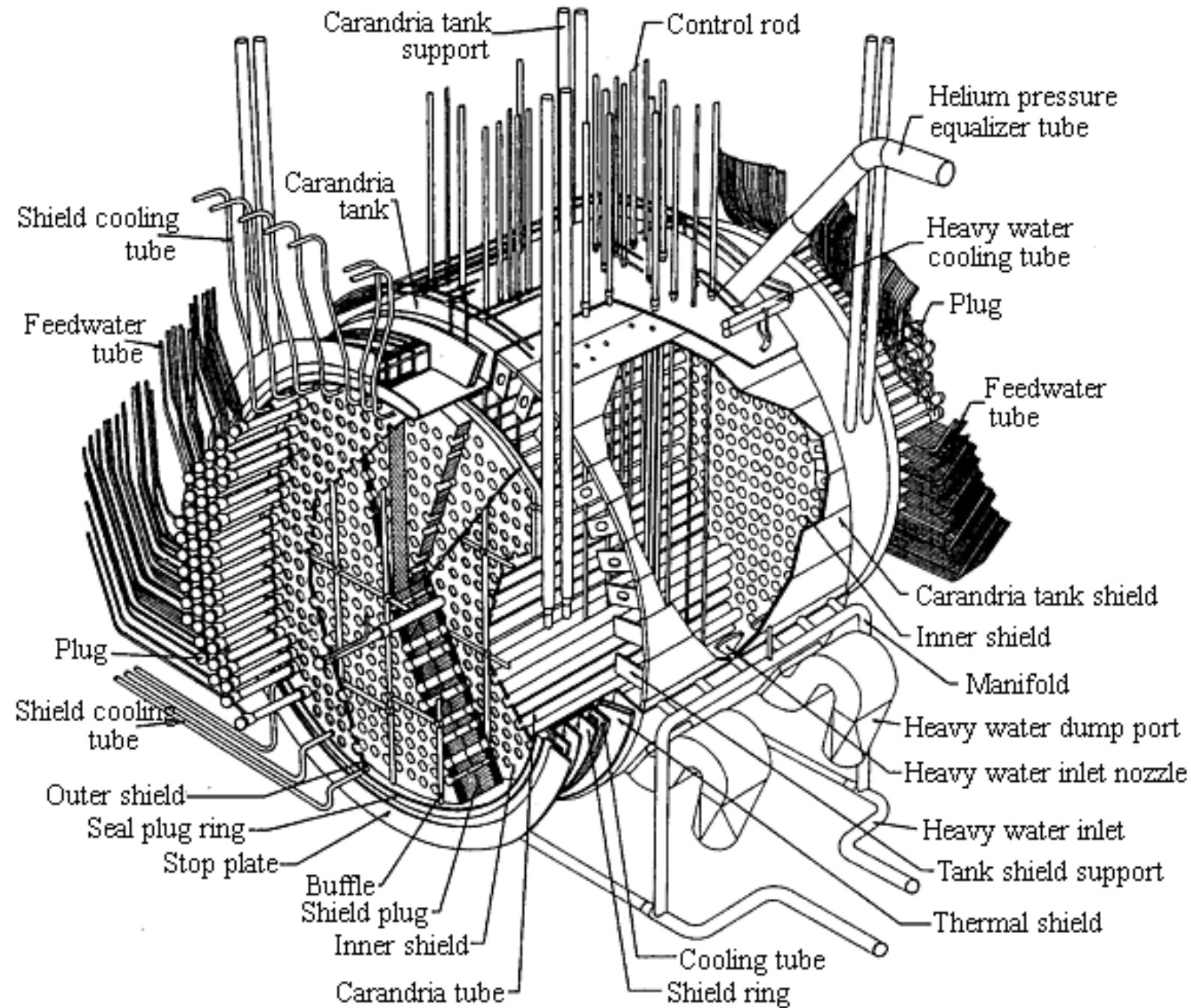
	Fuel	Moderator	Coolant
PWR	Enriched uranium, oxide fuel	Light water	Light water
BWR	Enriched uranium, oxide fuel	Light water	Light water
CANDU	Natural uranium, oxide fuel	Heavy Water	Heavy Water
GCR	Natural uranium, metallic fuel	Graphite	Carbon dioxide
RBMK-type	Enriched uranium, oxide fuel	Graphite	Light water
Fugen-type	Enriched uranium, oxide fuel Pu-fertilized uranium	Heavy Water	Light water
HTGR	Enriched uranium, oxide fuel Th / uranium-233	Graphite	Helium



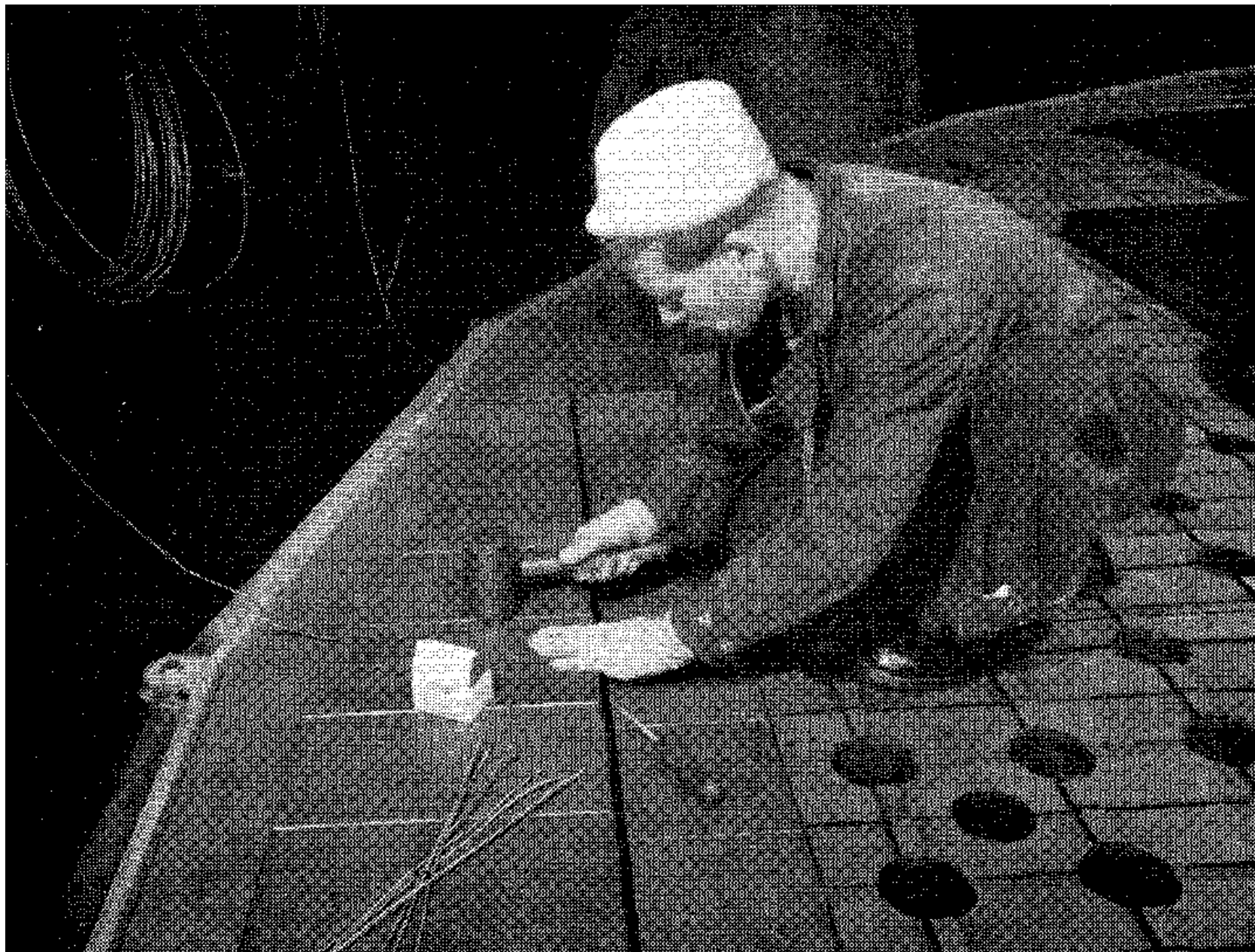
An example of PWR



An example of BWR

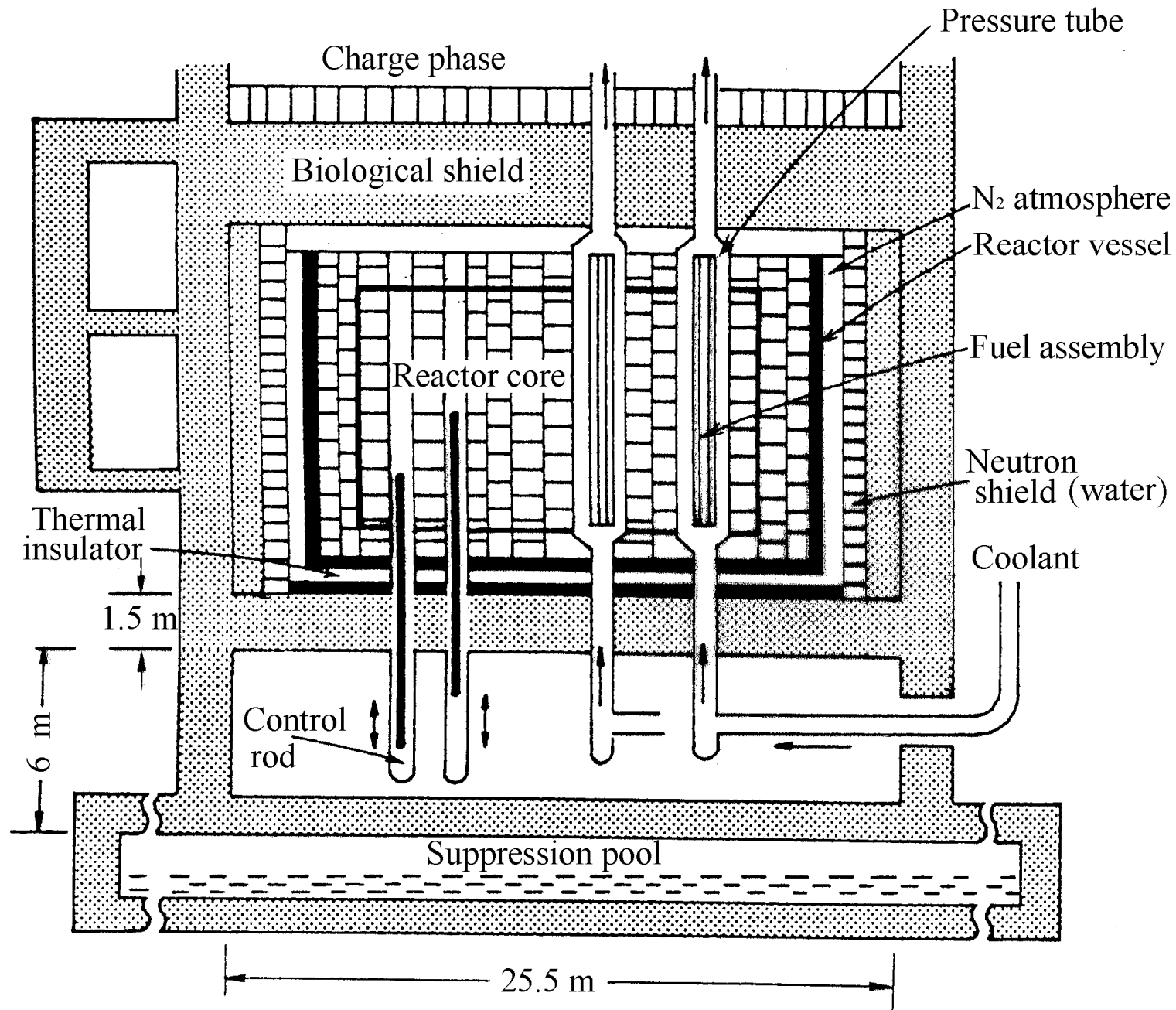


An example of CANDU

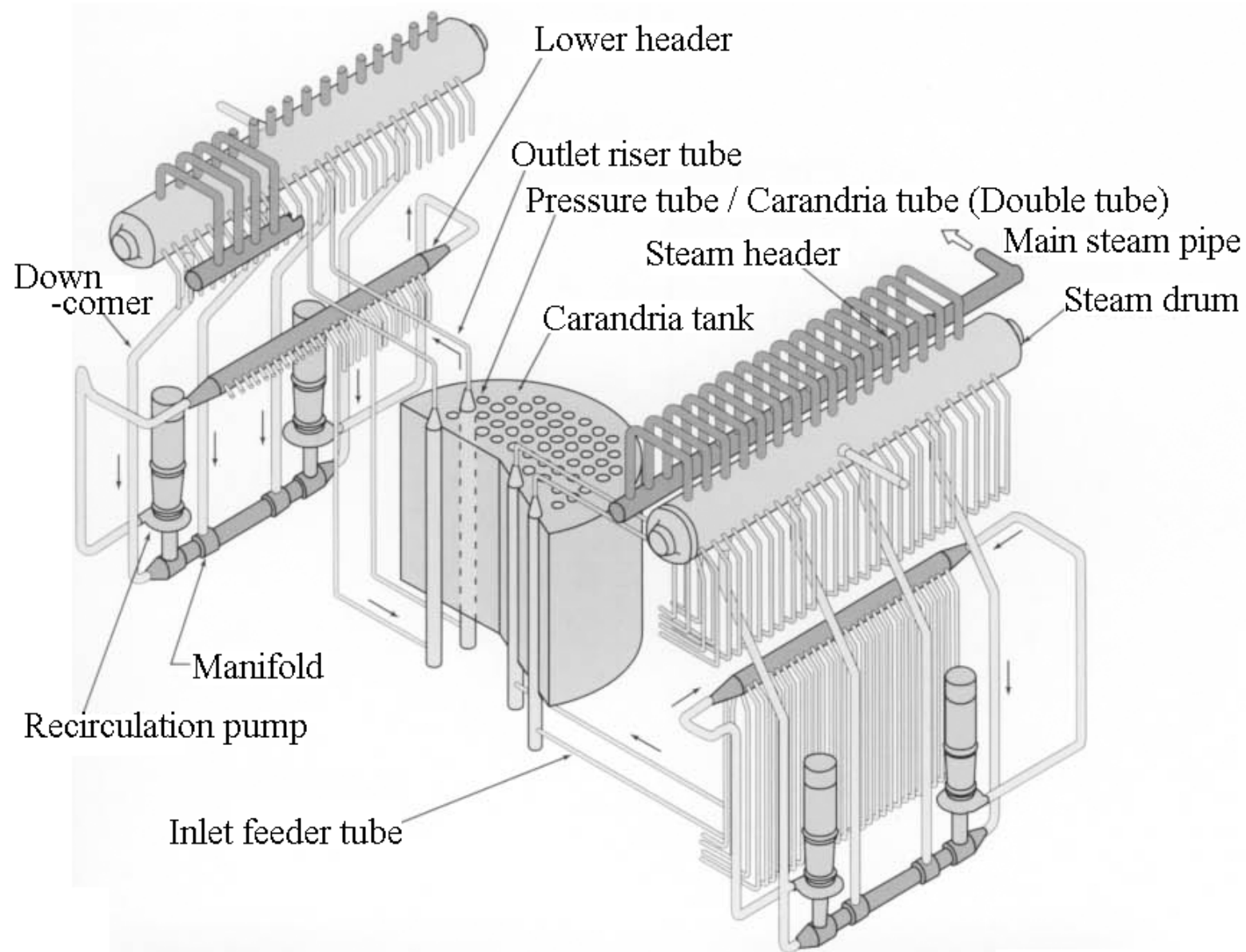


An example of GCR

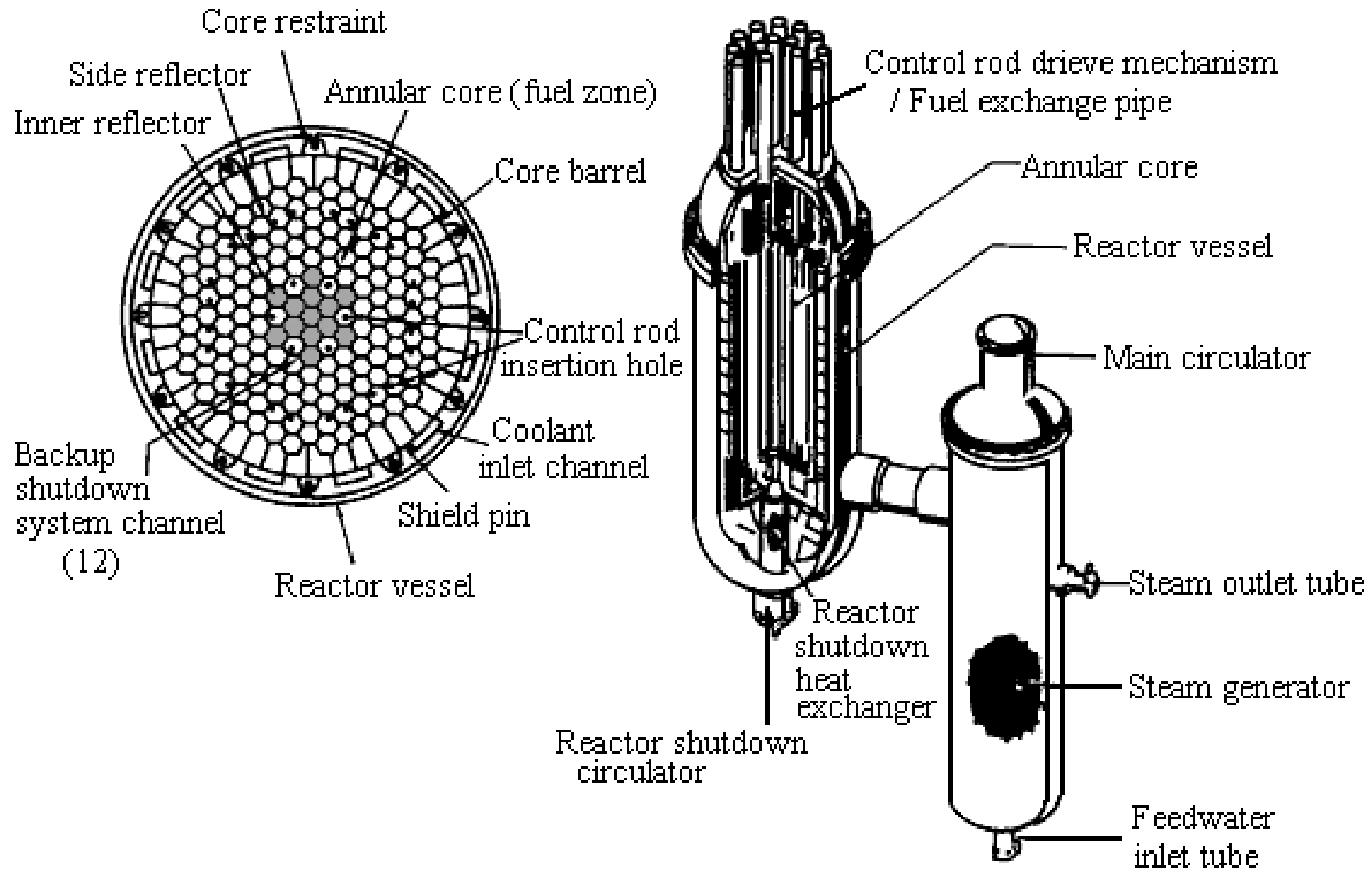
(from p.34 of “Calder Hall-the story of Britain’s first atomic power station” by Kenneth Jay 1956.)



An example of RBMK



An example of Fugen-type reactors



An example of Modular HTGR

NUCLEAR POWER PLANTS

BASIC COMPOSITION OF NUCLEAR POWER PLANT

Nuclear power plants consist of two parts: heat generation system and electricity generation system.

In a sense, nuclear power plants are similar to conventional thermal power plants.

	Nuclear power plant	Thermal power plant
Heat generation	Nuclear reactor	Boiler
Electricity generation	Turbine and generator	

NUCLEAR REACTOR AND BOILER

As a heat supplier, difference between boiler and nuclear reactor should be recognized. It makes things difficult and complicated.

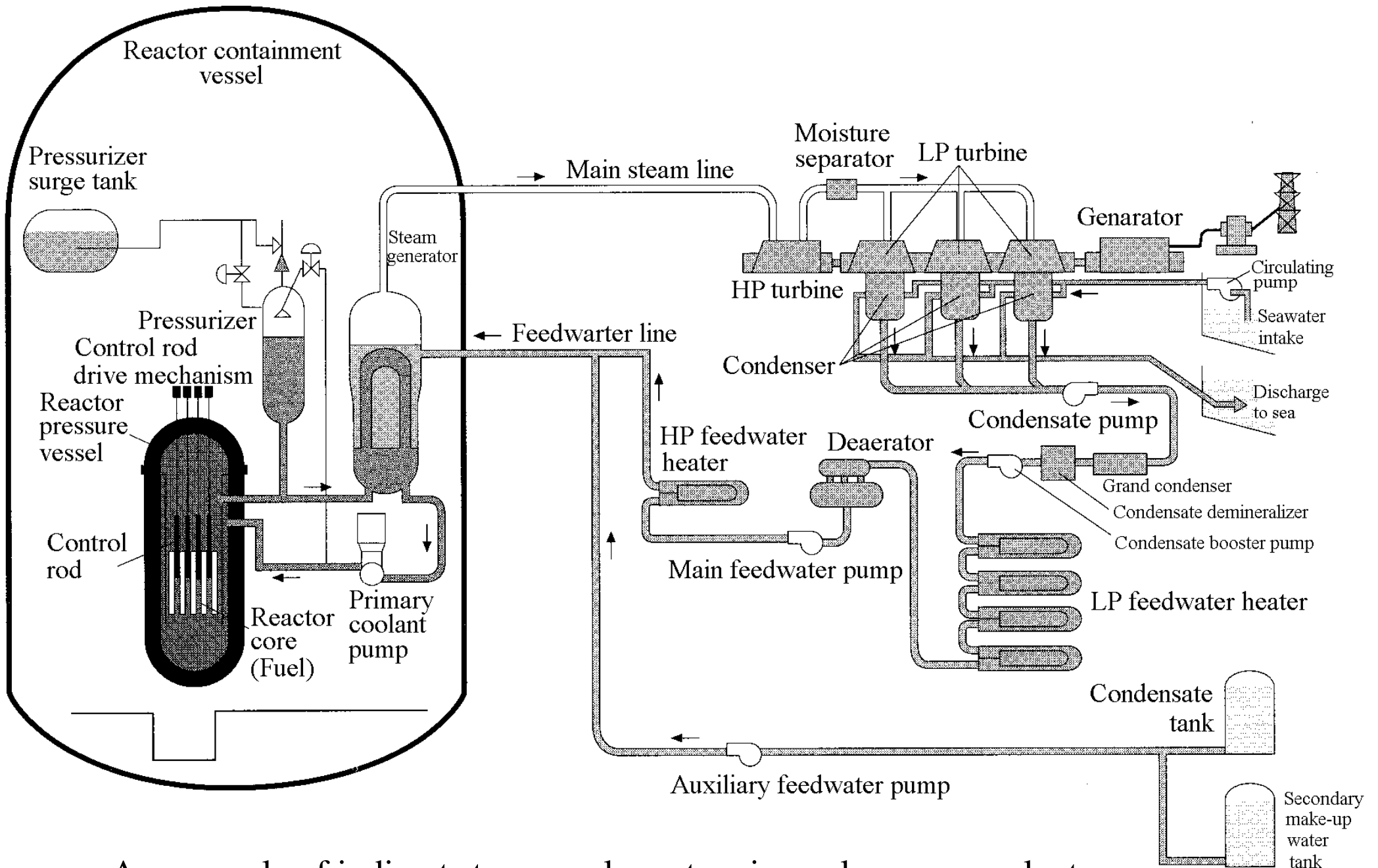
Boilers make steam directly, which can drive a steam turbine.

Nuclear reactors generate heat, which is taken out of the reactor by coolant. After this point, there are two different ways to drive a turbine.

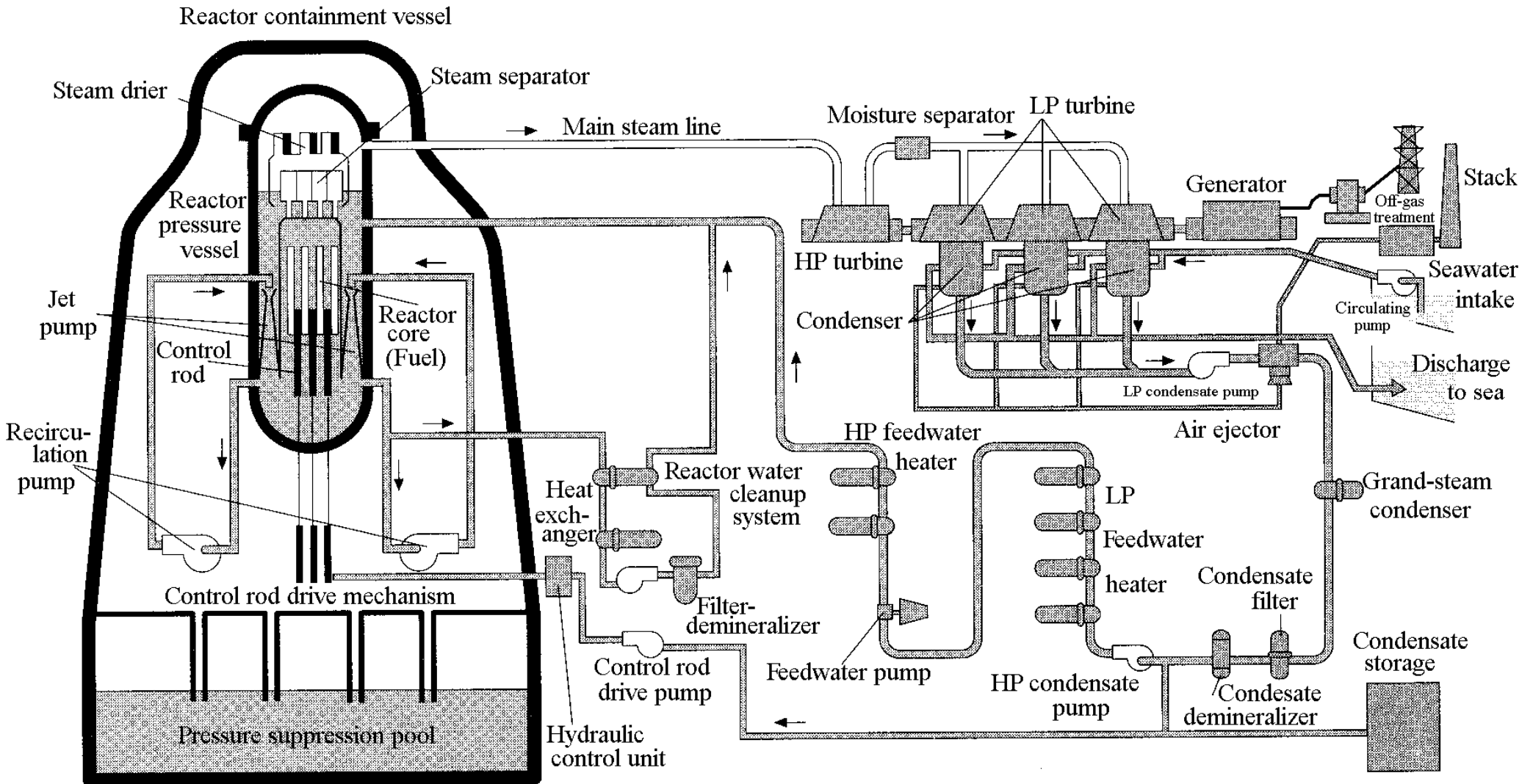
- (1) Heated coolant transfers the energy to water to generate steam.
- (2) Heated coolant is directly used to drive turbine.

NUCLEAR POWER PLANT SYSTEMS

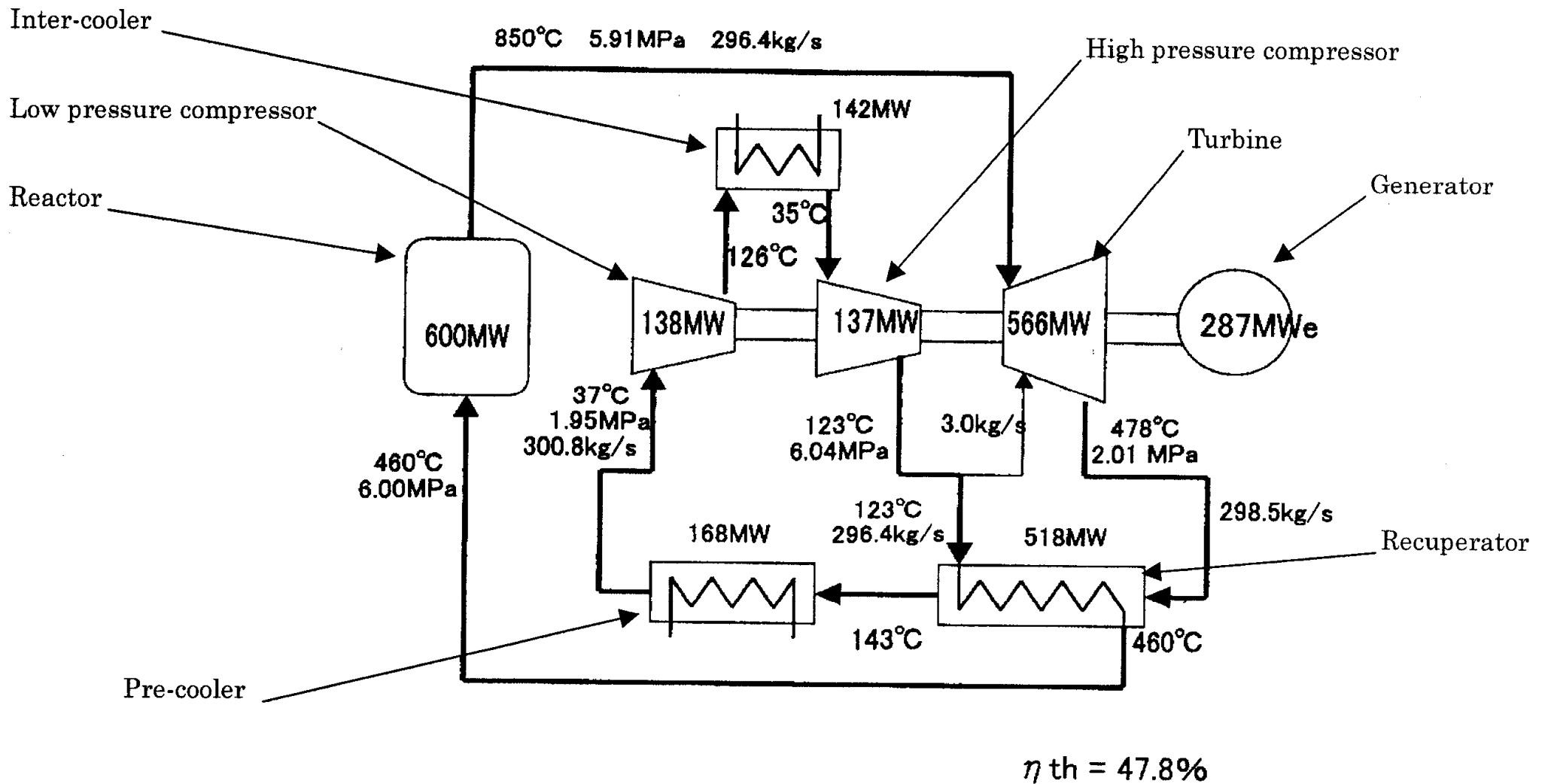
		Coolant	Turbine drive medium
(1)	GCR	Gas (CO ₂)	Steam
	PWR	Water	Steam
	CANDU	Heavy water	Steam
	HTGR	Gas (He)	Steam
	FBR	Liquid metal (Na)	Steam
(2)	BWR	Water	Steam [Phase change]
	RBMK	Water	Steam [Phase change]
	GT-MHTR	Gas (He)	Gas (He) [No phase change]



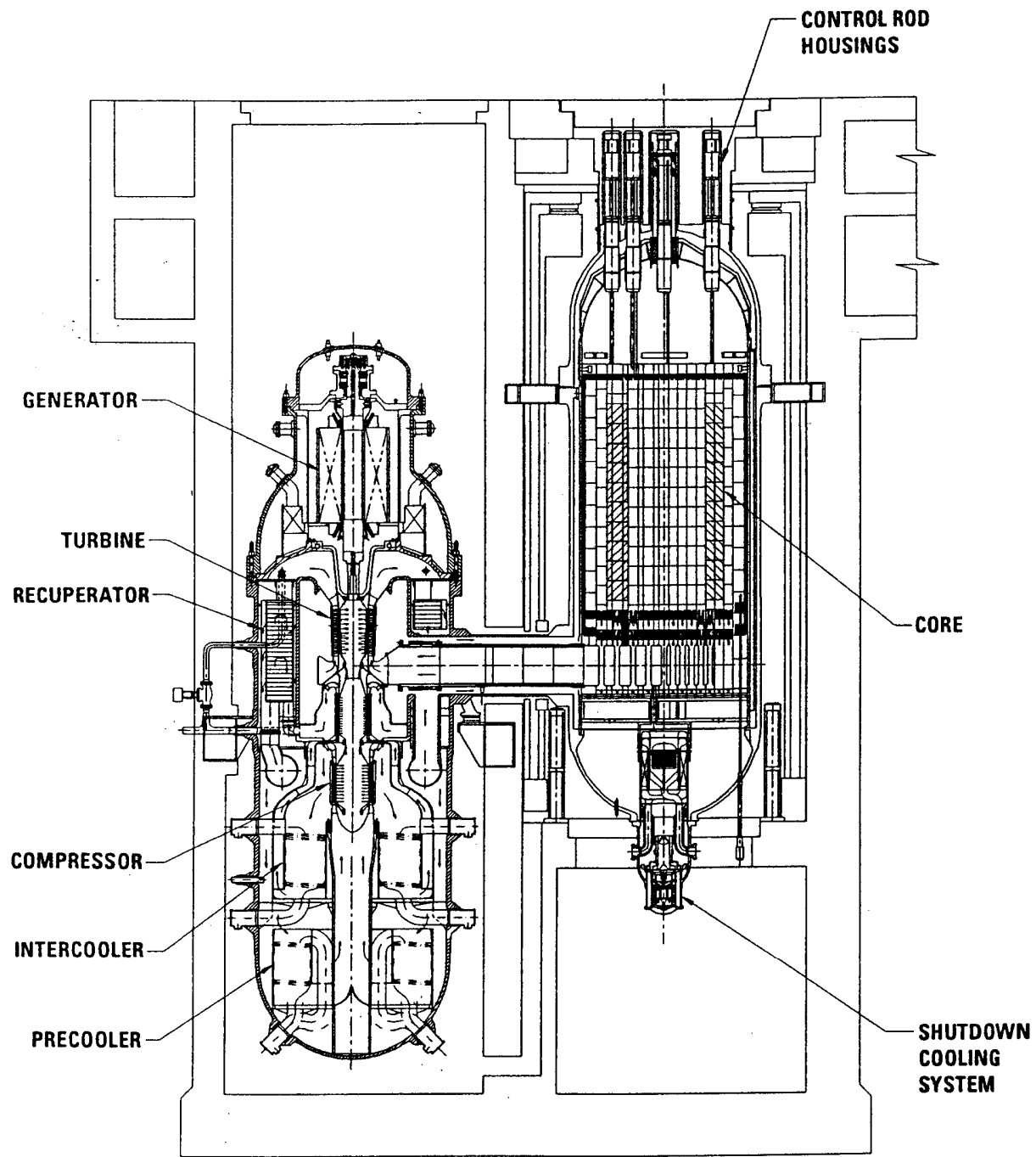
An example of indirect steam cycle system in nuclear power plant (PWR)



An example of direct steam cycle system in nuclear power plant (BWR)



An example of direct cycle gas turbine system in nuclear power plant (GT-MHTR)



Drawing of GT-MHTR

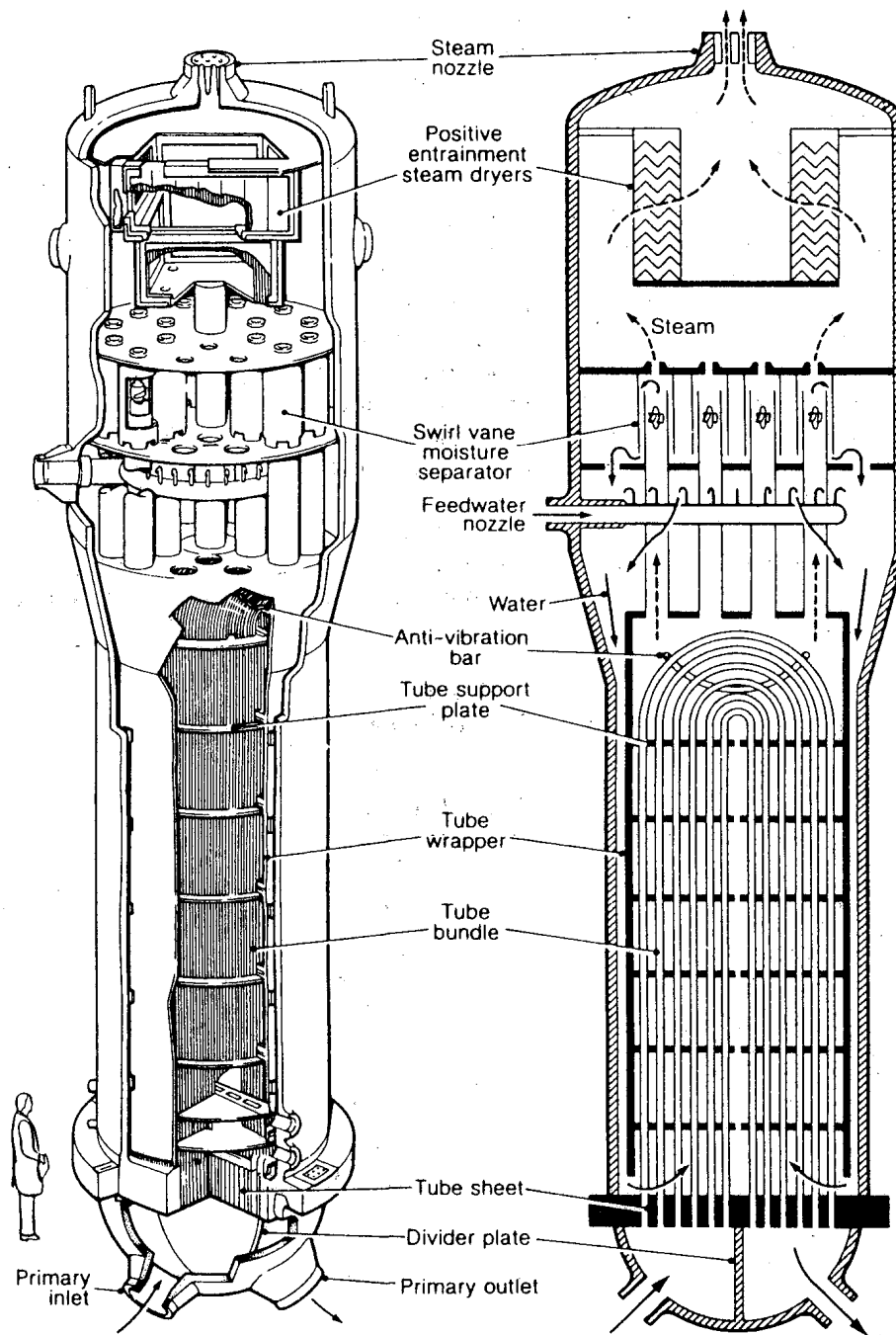
STEAM GENERATOR

Steam generator is one of the most important components in the nuclear power plants.

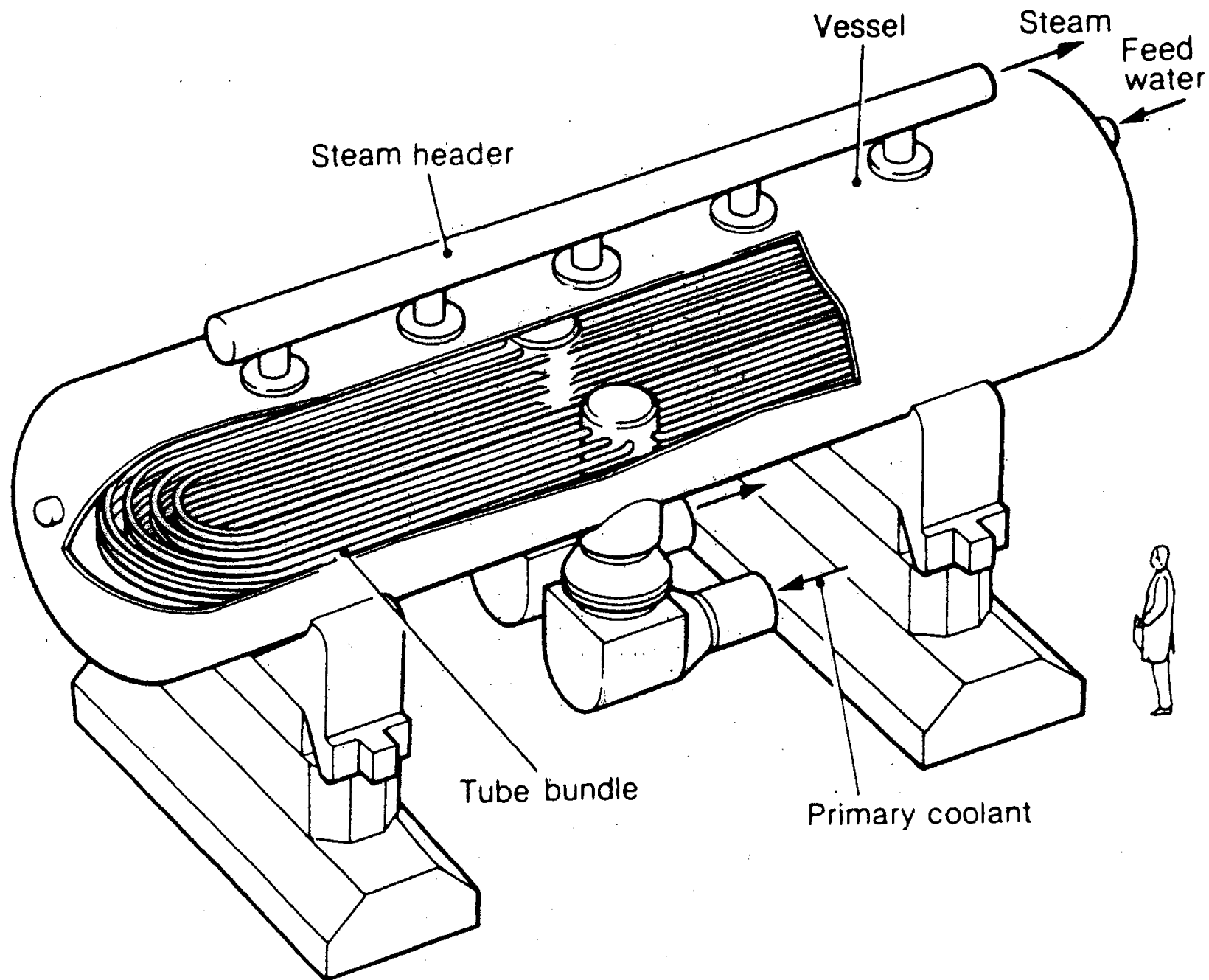
Steam generator is a heat exchanger to generate steam for turbine.

Steam generator isolates radioactive coolant from turbine system.

Steam generator is a simple device but not easy equipment.



An example of steam generator
(PWR)



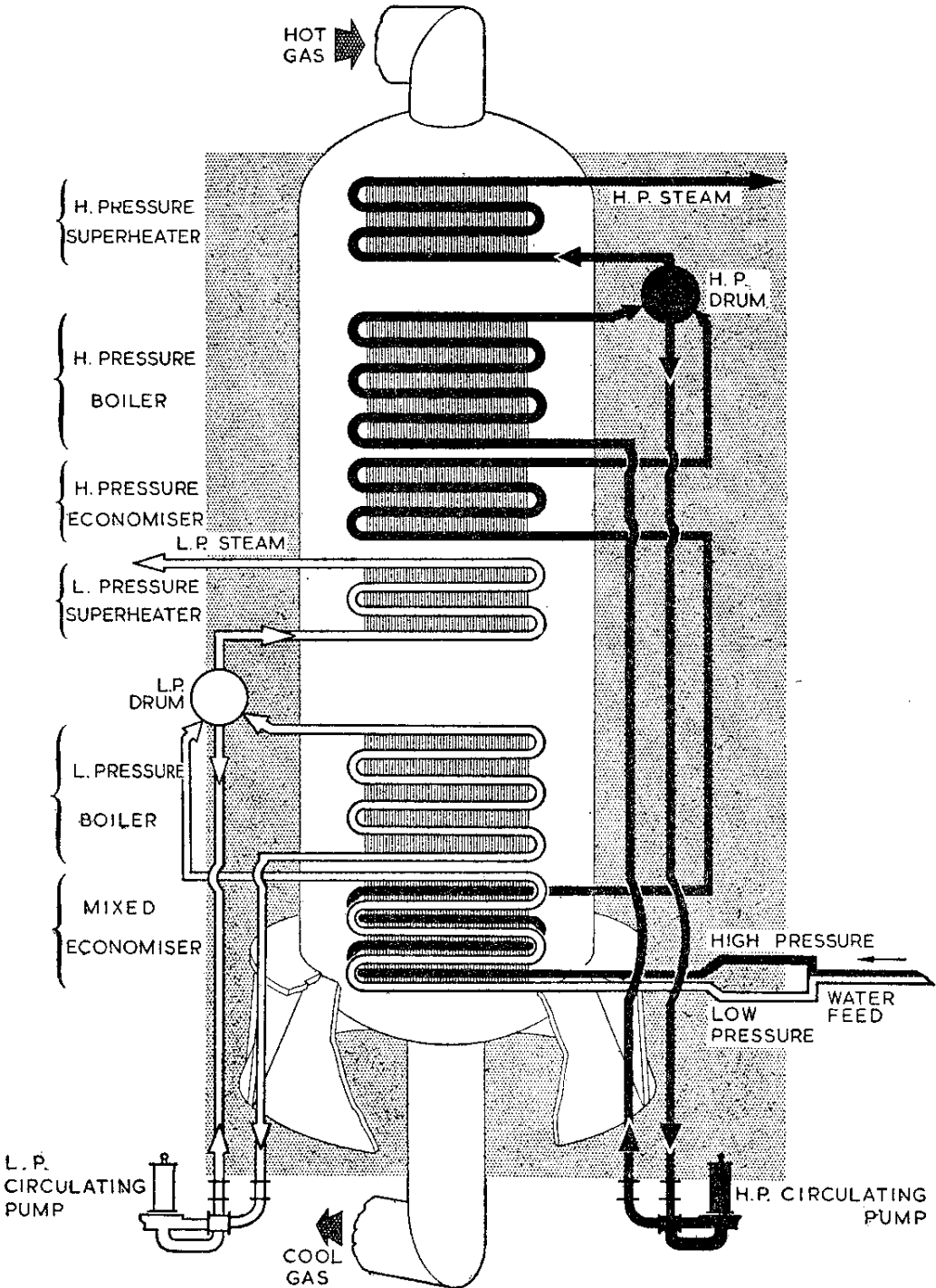
An example of steam generator (VVER)

DIFFERENCE BETWEEN STEAM GENERATOR AND BOILER

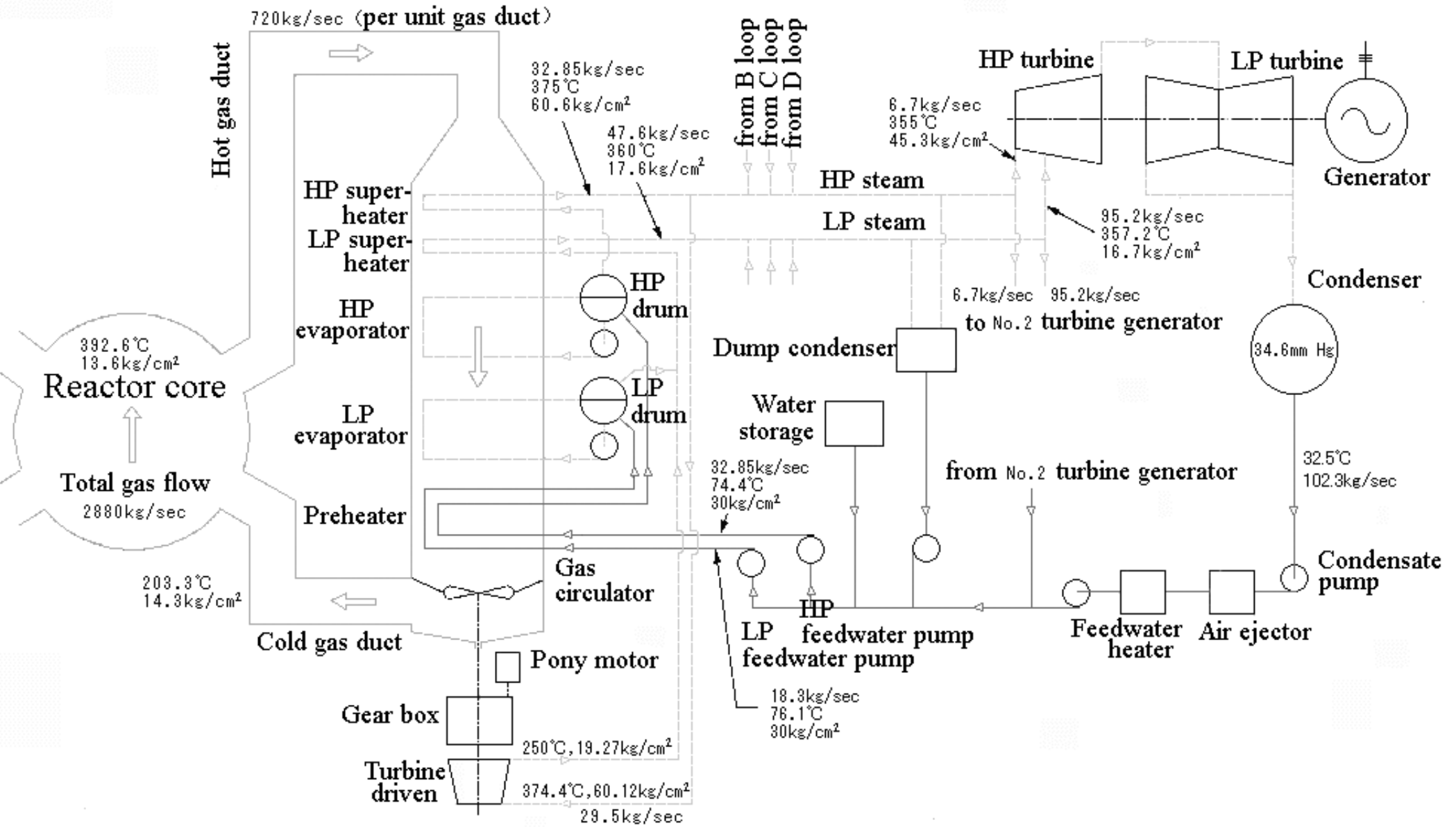
Economizer, evaporator, and super-heater are three basic sections of conventional boilers.

Most of the steam generators in nuclear power plants do not have such distinguished sections. In other words, most of the steam generators have no super-heater section.

Steam generators in gas-cooled nuclear power plants, however, have such distinguished three sections. More importantly, they have super-heater section.



Steam Generator of Calder Hall Type Gas-cooled Nuclear Power Plant



Steam Cycle of Calder Hall Type Gas-Cooled Nuclear Power Plant

PLANT PARAMETERS IN NUCLEAR POWER PLANTS

		Nuclear power plants			Typical thermal power plant
		GCR	PWR	BWR	
Turbine inlet steam	Pressure (kg/cm ² g)	45.3	60.1	66.8	246
	Temperature ()	355	275.5	282.4	538
	Moisture (%)	0	0.4	0.4	0
Turbine exhaust steam	Weight flow rate (kg/kwh)	4.3	5.7	5.8	3.1
	Volume flow rate (m ³ /kwh)	0.04	0.18	0.17	0.04

STEAM CONDITION OF NUCLEAR POWER PLANT

(1/2)

Most of the modern nuclear power plants generate “saturated steam” to drive turbine for electricity generation, though gas-cooled reactors generate “super-heated steam”.

In general, the steam condition of the nuclear power plants is not good enough to compete with modern conventional thermal power plants in respect of thermal efficiency.

The difference in steam condition between nuclear power plants and conventional thermal power plants should be recognized.

STEAM CONDITION OF NUCLEAR POWER PLANT

(2/2)

Efforts have been made to cope with the disadvantage of steam condition in nuclear power plants.

Historically, addition of conventional super-heaters to BWR was the first trial. Then, nuclear super-heat was tried for BWR. Both of them, however, failed.

Reheat of turbine steam had been put to practical use but little economical merit was observed.

DISCUSSION ON NUCLEAR STEAM

There might be two different views for nuclear steam condition.

Thermo-dynamics view:

Poor steam condition is serious drawback to nuclear power. It must be removed by introducing high temperature reactors.

Fuel cycle view:

Development of a new type of reactor is not fundamental solution to the problem. Efficiency of nuclear energy use should be maximized by the best use of fuel resources. In other words, more effort should be made toward completion of fuel cycle.

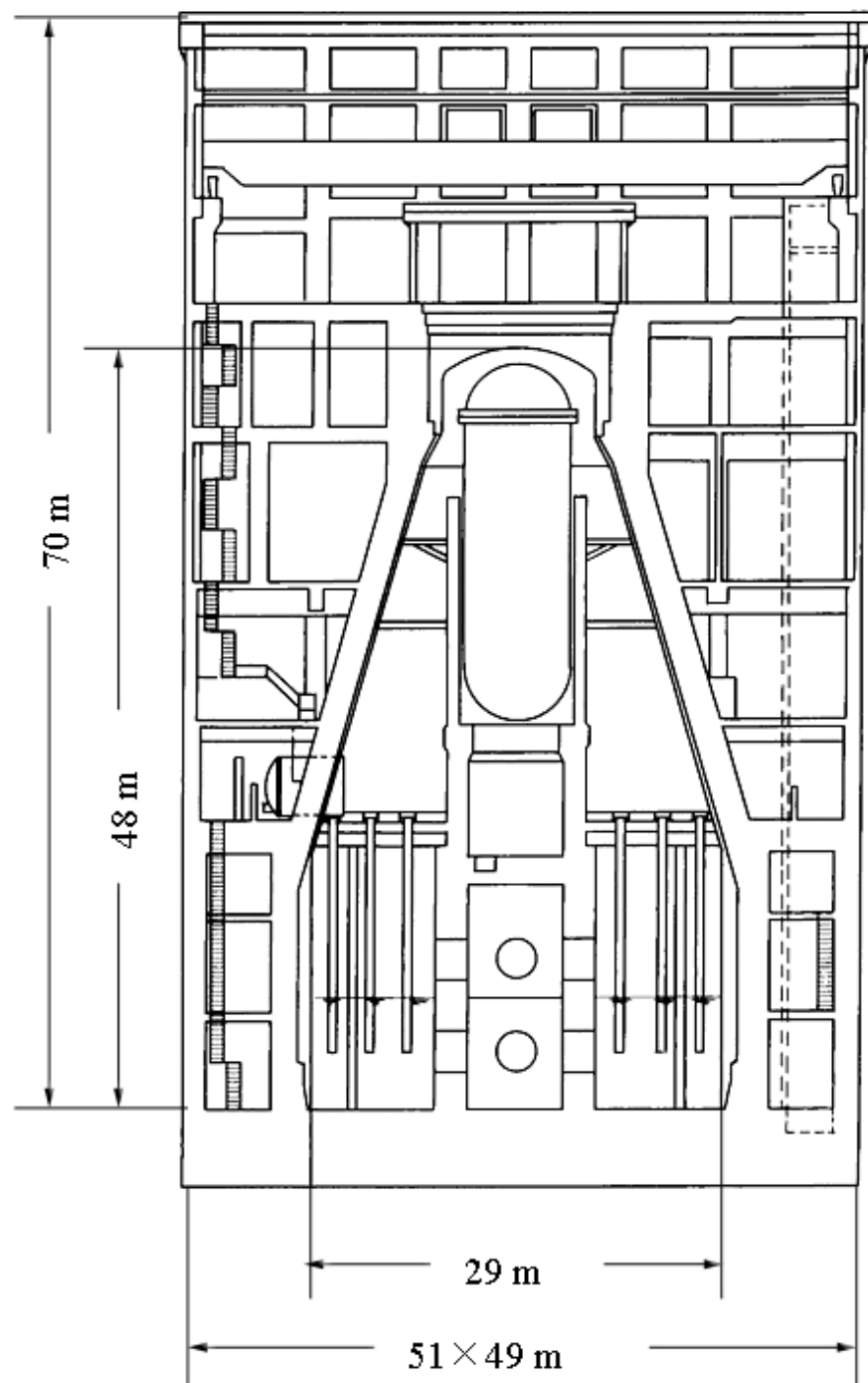
CONTAINMENT VESSEL

Containment vessel is one of the most important safety equipment of nuclear power plants.

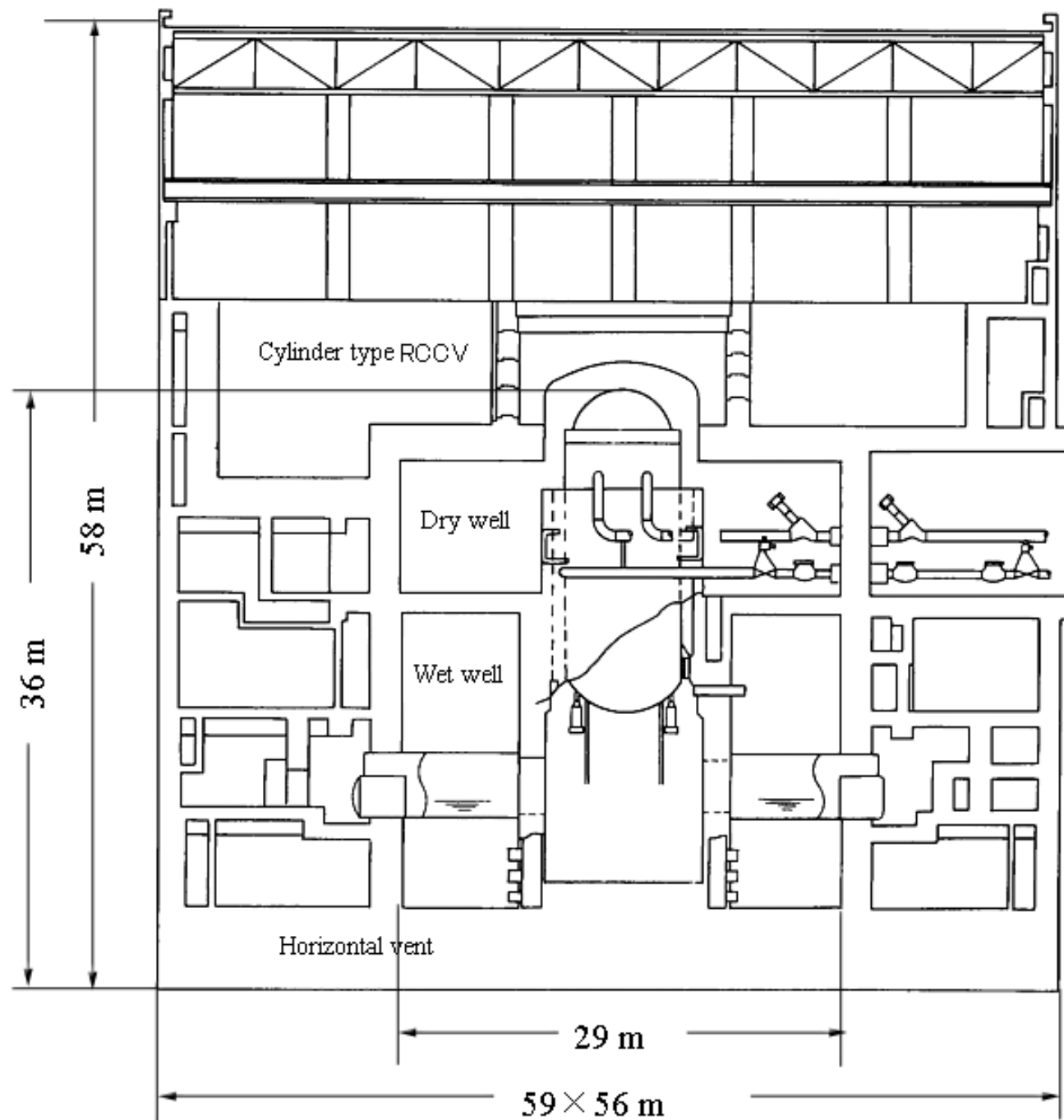
The containment vessel contains all the accidents to protect the peoples around the plant.

A few different types of containment vessel are seen.

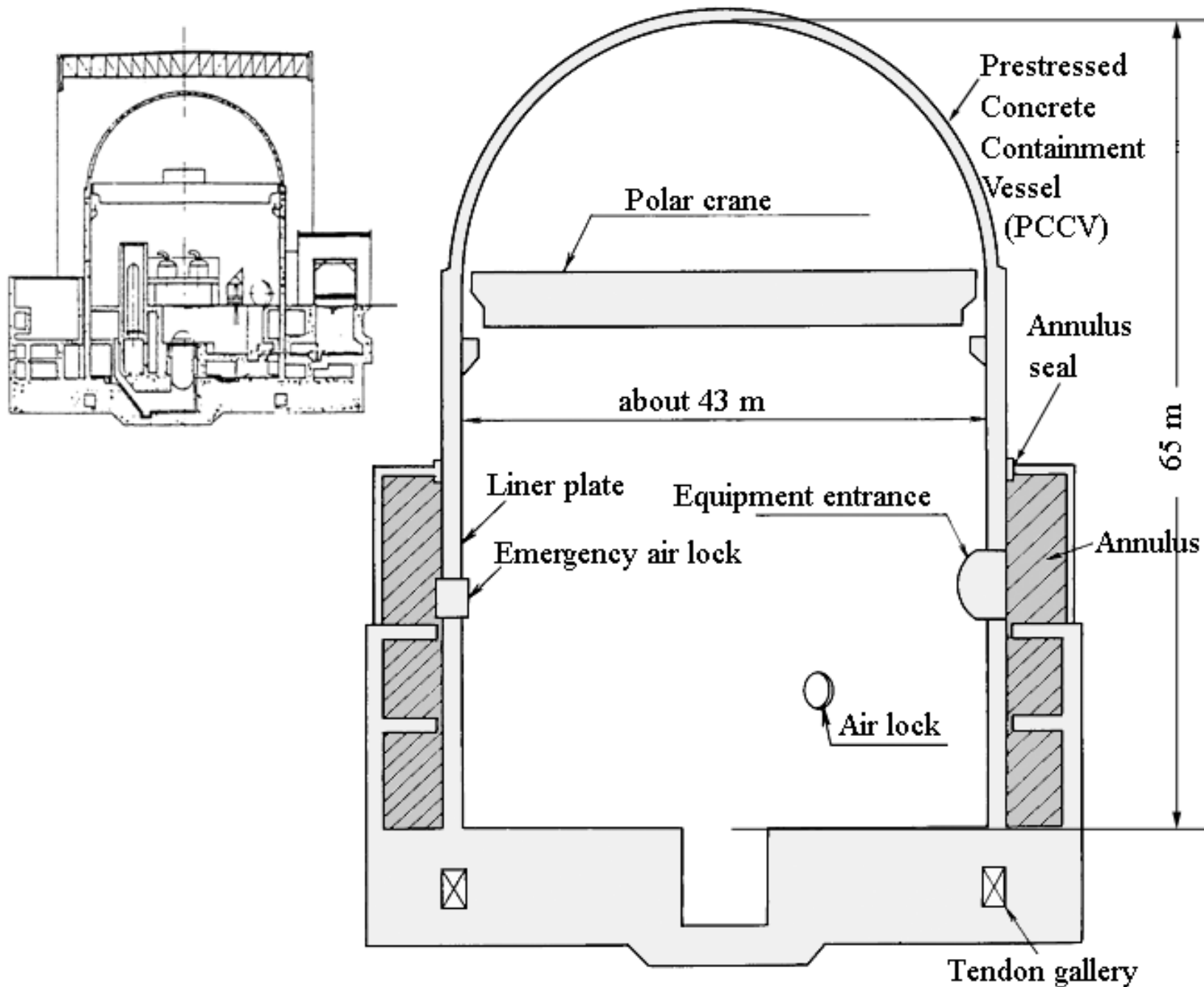
- (1) Steel containment vessel
- (2) Reinforced concrete containment vessel
- (3) Pre-stressed concrete containment vessel



An example of steel containment vessel (BWR)



An example of reinforced concrete containment vessel (BWR)



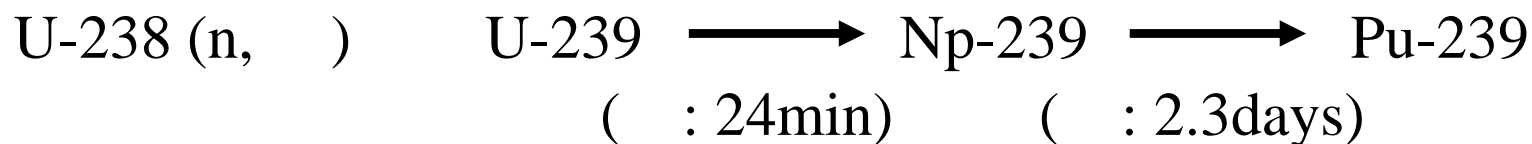
An example of pre-stressed concrete containment vessel (PWR)

NUCLEAR FUEL AND NATURAL RESOURCES

FUEL CONSUMPTION IN REACTOR

During the reactor operation, fissile material (U-235) is consumed and various fission products are produced in the reactor. Most of the fission products are neutron absorber. As a result, the reactivity of the reactor tends to reduce.

On the other hand, U-238 changes into fissile material by neutron absorption. That is,



Pu-239 is a fissile material and compensates for consumed U-235.

In consequence, nuclear characteristics of the reactor gradually changes with operation.

EFFECTIVE FUEL UTILIZATION

Some of the generated Pu-239 is consumed in the reactor to generate energy but the rest of Pu-239 is contained in discharged fuels.

The plutonium in the discharged fuels can be extracted to use as fuel by a chemical treatment, called “fuel reprocessing”.

As a fissile material, Pu-239 has favorable nuclear characteristics not only to thermal neutrons but also to fast neutrons.

Especially for fast neutrons, Pu-239 is superior to U-235.

Note: Utilizing this merit, fast reactors are designed to use extracted plutonium as fuel. Fast reactors with natural or depleted uranium blanket can produce more Pu-239 than consumed Pu-239 as fuel. Such fast reactors are called the fast breeder reactor (FBR).

FUEL CYCLE

From the viewpoint of effective utilization of natural resources, spent fuels should be reprocessed and re-used to support as many nuclear reactors as possible. In other words, fuel cycle should be closed.

At present, however, most of the spent fuels are not reprocessed but stored as a radioactive material in intermediate repositories.

NATURAL RESOURCES FOR NUCLEAR POWER

There are two kinds of natural resources for nuclear power.
They are uranium and thorium.

So far, only uranium resources are used for nuclear power. This is because natural uranium contains fissile material of U-235 but thorium does not contain fissile material.

Note: Th-232 absorbs a neutron to become U-233, which is fissile like U-235. Materials such as Th-232 and U-238 are called “fertile material”. U-235 is only fissile material in nature but there are two fertile materials in nature. Those are U-238 and Th-232.

CONCLUDING REMARKS

HOW DIFFERENT IS ATOMIC BOMB FROM NUCLEAR REACTORS ?

Nuclear reactor is so designed that prompt criticality is never reached. Accordingly, reactor power changes slowly due to the delayed neutron effect. In addition, Nuclear reactor has large thermal capacity because of their structure.

Atomic bomb is made by only pure fissile material and so designed to reach super-prompt critical condition ($K \approx 2 \sim 3$) within an extremely short period of time.

Fissile material in atomic bomb is promptly compressed to reach very high density so that neutron life becomes very short ($T \approx 10^{-8}$ {sec}). This value should be compared with $10^{-3} \sim 10^{-5}$ {sec} of thermal reactor life time at super-prompt critical condition.

Note: T value of fast reactors becomes about 4.4×10^{-7} {sec} above prompt critical condition.

NUCLEAR SAFETY

Nuclear safety is the most important issue not only for plant operators but also for the peoples on the earth.

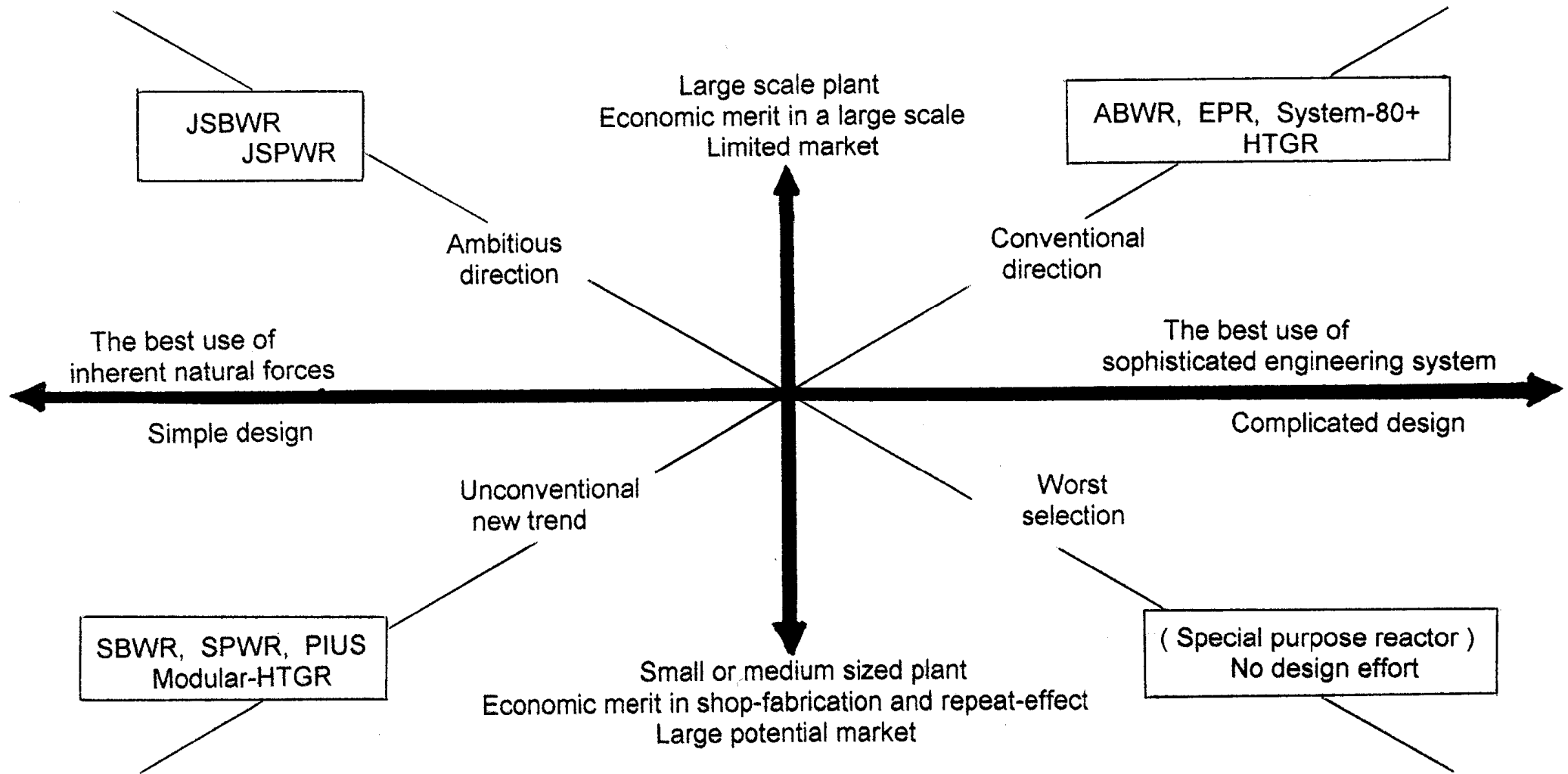
Nuclear safety is assured basically by two technological approaches.

- (1) The plant is equipped with sophisticated engineering safety system.
- (2) The plant is designed to have appropriate safety features.

Nuclear safety principle is expressed by three key concepts.

- (1) Fail safe
- (2) Multiple protection
- (3) Diversification

CURRENT TRENDS IN NUCLEAR POWER PLANTS



THE END

For those who use this educational material for giving a lecture, an instruction manual in Japanese is prepared, in which you can find the author's intention and points to be discussed over each sheet.

You can get the instruction manual by accessing

<http://www4.point.ne.jp/atom/nuclear/lecture.htm>.