The Design of a Nuclear Reactor

Joseph A Nathan and Vijay A Singh

The aim of this largely pedagogical article is to employ pre-college physics to arrive at an understanding of a system as complex as a nuclear reactor. We focus on three key issues: the fuel pin, the moderator, and lastly the dimensions of the nuclear reactor.

1. Introduction

Design considerations have engaged human minds since time immemorial. Consider the example of fire. The control and use of fire marks a dividing line between man and other mammals. The flintstones used to ignite a spark underwent a number of changes. The variety of arrangements of logs, twigs and dried leaves leading to a sustained combustion would make for a fascinating study. Some of these arrangements have been handed down to us from prehistoric times and can still be seen in rural kitchens. The present article is also concerned with design. It employs pre-college physics to arrive at an understanding of a system as complex as a nuclear reactor (NR). Parts of it were selected as an academic task for students around the globe in the recently concluded International Physics Olympiad in Mumbai, India [1].

Around 1938, the German physicist Otto Hahn discovered that uranium nucleus could be split into two smaller nuclei by bombarding it with neutrons, later termed as 'neutron induced fission'[1]. A back-of-the-envelope calculation by Lise Meitner and her nephew Otto Robert Frisch showed that in fission, about 200 MeV energy is released. Further experiments revealed that around 2–3 neutrons are emitted per fission (fission neutrons) having energies between 1–2 MeV; lower the energy of the


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[1] See Amit Roy, Story of Fission: Unlocking Power of the Nucleus, *Resonance,* Vol.21, No.3, 2016.

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Fissile materials are those that undergo fission when a neutron of any energy enters the nuclei.

neutrons inducing fission, more the probability of fission. To harness this fission energy, the fission neutrons are used to induce fission in other uranium nuclei to sustain a chain of fissions (chain reaction).

Naturally occurring uranium contains 0.72% of ^{235}U and 99.28% of ^{238}U. While ^{235}U, categorised as fissile, undergoes fission with neutrons of any energy, ^{238}U, categorised as fissionable, undergoes fission with high energy neutrons only. But due to high inelastic scattering of high energy neutrons from ^{238}U, the energy of most of the fission neutrons is reduced below the threshold energy to cause fission in ^{238}U. So, sustaining a chain reaction even in 99.28% of ^{238}U is not possible. Now to sustain fission in 0.72% of ^{235}U, the energy of the fission neutrons should be reduced to increase the probability of fission. Enrico Fermi[2] achieved the sustained chain reaction in 0.72% of ^{235}U in the first man-made NR named Chicago Pile-1 in 1942 [2,3], by making the fission neutrons collide with low-mass number element called moderators (graphite) to lose energy. For efficient neutron utilisation, they should have a low affinity to absorb neutrons. Later, in the year 1954, USSR's Obninsk Nuclear Power Plant [4] became the world's first to generate around 5 MW of electric power. At present, India has 21 NRs that are operated with various reactor technologies which produce 5780 MW of electric power.

[2] See *Resonance*, Vol.18, No.8, 2014.

Fissionable materials are those that undergo fission only when a neutron with high energy enters the nuclei.

Reactors are categorized broadly into two types: thermal and fast reactors. Fast reactor technology is at the developmental stage and almost all commercial reactors are thermal.

2. Thermal Reactors

In the thermal reactors, fission is caused by neutrons with their energy thermalised to the temperature of the moderator. The fuel for these reactors is UO_2. There are several types of thermal reactors from the compact to the very large reactors and with a variety of technologies. We mention the two most popular types of thermal reactors.

Boiling Water Reactors: The most successful and compact reactors are the boiling water reactors. They are called pressure vessel reactors where the moderator is light (normal) water, which also acts as coolant. The water surrounds the fuel bundles just like in the case of a heater rod immersed in water. The heat from the fuel bundles is transferred directly to the water which is allowed to boil. This same boiling water acts as the moderator for the fission neutrons. The steam generated from boiling is collected and sent to turbines. For this reason, they are most efficient and the use of light water makes them most compact. But since the reactor has to be vacuum sealed for the collection of steam, refueling is one of the major issues. After a sufficient duration of operation, the reactor requires to be shut down for refueling. Further, absorption of neutrons in light water is high. In order to compensate for the loss of neutrons, the fuel UO_2 requires enrichment of ^{235}U, sometimes as much as 20%, and this is a difficult technology.

Pressurised Heavy Water Reactors: The second most popular are the pressure tube reactors, where coolant and the moderator are physically separated, though they both are heavy water[3]. While the coolant is used to remove heat from the fuel, the moderator is used to reduce the energy of the high energy neutrons. The name 'pressure tube reactors' is because the coolant in these reactors is pressurised and the tubes through which the coolant flows has to withstand this pressure. Since the moderator and coolant are kept separate, the design of these reactors for the coolant flow is not simple. They are nevertheless very popular since they can use natural uranium without any enrichment.

[3] See Uday Maitra and Richard N Zare, Fall and Rise of D_2O Ice Cube in Liquid H_2O, *Resonance*, Vol.21, No.5, pp.453–456, 2016.

3. Fast Reactors

In fast reactors fission is sustained by high energy ($\leqslant$1Mev) neutrons inducing fission. So the fuel is different from that of thermal reactors.The fuels ^{233}U and ^{239}Pu for fast reactors are not available in Nature but created respectively when ^{232}Th and ^{238}U transmutes after absorbing a neutron. The reactors in which the fuel for fast reactors are produced are called fast breeder reactors. Fast reactors do not have moderators and coolant has high mass number. When high energy neutrons induce fission, the fission neutron emission increases hence ^{233}U or ^{239}Pu can be bred by placing ^{232}Th or UO_2 inside the reactor. The coolant in this reactor is liquid sodium, a material difficult to handle. Also, the neutron population changes at a faster rate which requires a fast acting system for control. It is these difficulties along with the production of ^{233}U and ^{239}Pu which has delayed the commercial success of fast reactors.

4. A Typical Nuclear Reactor

A typical NR (*Figure* 1) consists of a cylindrical tank of height H and radius R filled with D_2O (heavy water) called moderator. Cylindrical tubes, called fuel channels are kept axially in a square array. In each fuel channel

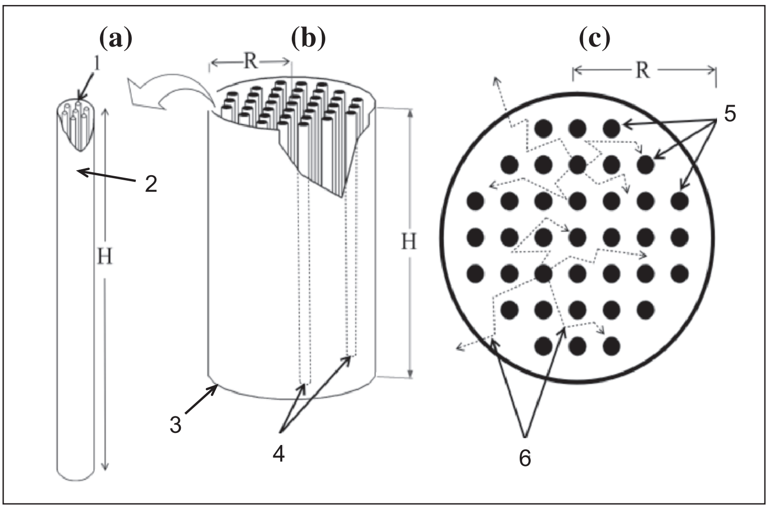

Figure 1. Schematic sketch of the Nuclear Reactor (NR) **(a)**. Enlarged view of a fuel channel. (1: Fuel pins; 2: Pressurised coolant (D_2O) flows inside channel at an average temperature of 555 K.)
(b) A view of the NR. (3: Filled with moderator (D_2O) maintained at 353 K; 4: Fuel channels.)
(c) Top view of NR. (5: Square arrangement of fuel channels; 6:Typical neutron paths.)
Other components (for example, control rods) are not shown.

there are fuel bundles which are stacked one above the other to a height H. A fuel bundle contains a cluster of cylindrical fuel pins of solid natural UO_2. Fission neutrons, coming outward from a fuel channel, collide with the moderator, losing energy, and reach the surrounding fuel channels with low enough energy to cause further fissions (see *Figures* 1a–c). Heat generated from fission in the pin is transmitted to the surrounding coolant fluid flowing along its length inside the fuel channel. This cylinder is kept horizontal (unlike what is shown in *Figure* b) to avoid pumping the coolant against gravity. The coolant in these type of reactors is again D_2O and is pressurised to avoid boiling. For this reason, such an NR is called Pressurised Heavy Water Reactor (PHWR). The heat from the coolant is removed in a heat exchanger and is used to produce steam which runs the turbines to produce electricity. Here, we shall study some of the physics behind the (i) design of the fuel pin, (ii) role of a moderator and finally (iii) dimensions of a NR of cylindrical geometry. However, the study can be extended to other types of reactors.

5. Fuel Pin

The fuel pin is the elementary entity of the NR. We shall arrive at the upper limit on the radius of the fuel pin starting from the energy released in a typical nuclear fission. Consider a typical fission reaction of a stationary ^{235}U after it absorbs a neutron of negligible kinetic energy:

$$^{235}\mathrm{U} + {}^{1}\mathrm{n} \longrightarrow {}^{94}\mathrm{Zr} + {}^{140}\mathrm{Ce} + 2\,{}^{1}\mathrm{n} + \Delta E\,.$$

We estimate ΔE (in MeV) – the total fission energy released. Ignoring charge imbalance, this energy released can be calculated using Einstein's famous mass–energy formula ($E = mc^2$),

$$\Delta E = [m(^{235}\mathrm{U}) + m(^{1}\mathrm{n}) - m(^{94}\mathrm{Zr}) - m(^{140}\mathrm{Ce}) - 2m(^{1}\mathrm{n})]\mathrm{c}^2\,.$$

Substituting in terms of unified atomic masses (u)

Nuclear Masses	**Data for UO_2**
$m(^{235}\text{U}) = 235.044$ u	Molecular weight $M_w = 0.270$ kg mol^{-1}
$m(^{94}\text{Zr}) = 93.9063$ u	Density $\rho = 1.060 \times 10^4$ kg m^{-3}
$m(^{140}\text{Ce}) = 139.905$ u	Melting point $T_m = 3.138 \times 10^3$ K
$m(^{1}\text{n}) = 1.00867$ u	Thermal conductivity $\lambda = 3.280$ W m^{-1} K^{-1}

Table 1. Relevant data for UO_2 and nuclear masses [5], [6].

(see *Table* 1), we get

$$\Delta E = 208.684 \text{ MeV}.$$

Let N be the number of ^{235}U atoms per unit volume in natural UO_2. The number of UO_2 molecules per m^3 of the fuel N_1 is given in terms of its density ρ, the Avogadro number N_A and the average molecular weight M_w as (see *Table* 1)

$$N_1 = \frac{\rho N_A}{M_w} = \frac{10600 \times 6.022 \times 10^{23}}{0.270} = 2.364 \times 10^{28} \text{m}^{-3}.$$

Each molecule of UO_2 contains one uranium atom. Since only 0.720% of them are ^{235}U, $N = 0.0072 \times N_1 = 1.702 \times 10^{26}$ m^{-3}.

For the study of interaction of neutrons with a target material, we define the following three quantities.

1) Neutron flux (ϕ) is the total number of neutrons crossing a unit area per second.

2) The microscopic cross-section σ is the number of (specific) interactions of neutrons per second per unit atom of the target material.

3) The macroscopic cross-section Σ is the product of σ and the atom density N of the target material.

Let us assume that the neutron flux $\phi = 2.000 \times 10^{18}$ m^{-2} s^{-1} on the fuel is uniform. The microscopic fission cross-section of a ^{235}U nucleus is $\sigma_\text{f} = 5.400 \times 10^{-26}$ m^2.

We will now estimate Q (in Wm^{-3}), the rate of heat production in the pin per unit volume, given that 80.00% of the fission energy is available as heat. Heat energy available per fission $E_f = 0.8 \times 208.7$ MeV $= 2.675 \times 10^{-11}$J. The total cross-section per unit volume is $N \times \sigma_{\mathrm{f}}$. Thus, the heat produced per unit volume per unit time,

$$Q = N \times \sigma_{\mathrm{f}} \times \phi \times E_{\mathrm{f}} = 4.917 \times 10^{8}\ \mathrm{Wm}^{-3}.$$

To estimate the radius of the fuel pin, we consider the steady-state temperature difference between the center (T_{c}) and the surface (T_{s}) of the pin which can be expressed as $T_{\mathrm{c}} - T_{\mathrm{s}} = kF(Q, a, \lambda)$, where k is a dimensionless constant and a is the radius of the pin. The functional dependence $F(Q, a, \lambda)$ of the temperature difference can be obtained by solving the Fourier equation for heat conduction. However, there is an appealing method using dimensional analysis. The dimensions of $T_{\mathrm{c}} - T_{\mathrm{s}}$ is temperature. We write this as $T_{\mathrm{c}} - T_{\mathrm{s}} = [K]$. One can similarly write down the dimensions of Q, a and λ. Equating the temperature to powers of Q, a and λ, one could state the following dimensional equation:

$$K = Q^{\alpha} a^{\beta} \lambda^{\gamma} = [\mathrm{M\ L^{-1}T^{-3}}]^{\alpha}[\mathrm{L}]^{\beta}[\mathrm{M\ L^{1}T^{-3}K^{-1}}]^{\gamma}\,.$$

By equating powers of temperature, we get $\gamma = -1$ and $\alpha + \gamma = 0$ by equating powers of mass or time. Therefore, $\alpha = 1$. Next, equating powers of length yields $-\alpha + \beta + \gamma = 0$ which gives $\beta = 2$. Thus we have

$$T_{\mathrm{c}} - T_{\mathrm{s}} = \frac{Qa^2}{4\lambda},$$

where $k = 1/4$, and this dimensionless factor can be obtained from the detailed solution of the heat conduction equation. Factors like the efficiency of the heat exchanger, the steam pressure requirement of the turbine, etc., empirically decide the inlet temperature of the coolant as 533 K. To increase the heat transfer ability throughout the cycle, the coolant is pressurised so that there is no boiling. An optimised flow rate of the coolant through the fuel channel increases the temperature by about 44 K making the maximum outlet temperature of the coolant 577 K [6]. We can then estimate the upper limit $a_{\rm u}$ on the radius a of the pin. The melting point of UO_2 is 3138 K. This sets a limit on the maximum permissible temperature $(T_{\rm c} - T_{\rm s})$ to be less than $(3138 - 577 = 2561$ K) to avoid 'meltdown'. Thus one may take a maximum of $(T_{\rm c} - T_{\rm s}) = 2561$ K. Noting that $\lambda = 3.28$ $\mathrm{Wm}^{-1}K^{-1}$ (*Table* 1), we have

$$a_u^2 = \frac{2561 \times 4 \times 3.28}{4.917 \times 10^8}\,.$$

This yields $a_u \backsimeq 8.267 \times 10^{-3}$ m, which constitutes an upper limit on the radius of the fuel pin. It is interesting to note that the Tarapur 3 & 4 NR in Western India has a fuel pin radius of 6.090×10^{-3} m.

6. Moderator

Consider the 2D elastic collision between a neutron of mass 1 u and a moderator atom of mass A u. Before collision, all the moderator atoms are considered to be at rest in the laboratory frame (LF). Let $\overrightarrow{v_b}$ and $\overrightarrow{v_a}$ be the velocities of the neutron before and after collision respectively in the LF. Let $\overrightarrow{v_m}$ be the velocity of the center of mass (CM) frame relative to LF and θ be the neutron scattering angle in the CM frame. All the particles involved in collisions are moving at non-relativistic

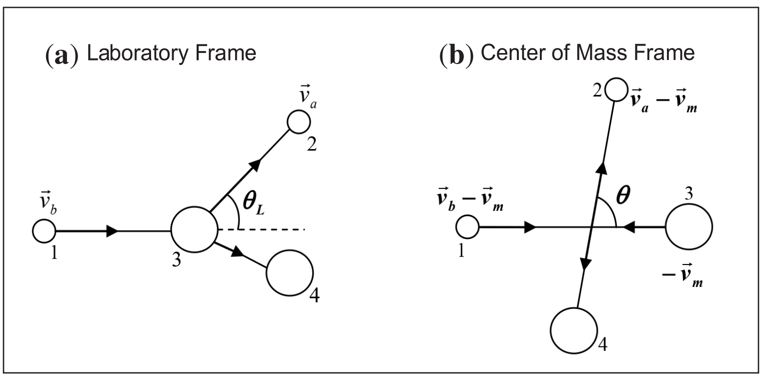

Figure 2. The collision of the neutron labeled 1 with moderator atom labeled 3 in the LF (**a**) and CM frame (**b**). The scattered neutron and moderator atom are labeled 2 and 4 respectively.

speeds. Let E_b and E_a be the kinetic energies of the neutron, in the LF, before and after the collision respectively. We will estimate E_a/E_b and rewrite it in terms of a dimensionless parameter $\alpha \equiv [(A-1)/(A+1)]^2$.

The collision is shown schematically with θ_L as the scattering angle in LF and θ as the scattering angle in CM frame (*Figure* 2). From the definition of the CM frame, $v_m = v_b/(A+1)$, before the collision $v_b - v_m = Av_b/(A+1)$ and v_m will be the magnitude of the velocities of the neutron and moderator atom respectively. In elastic collision, the particles are scattered in opposite directions in the CM frame and so the speeds remain the same $v = Av_b/(A+1)$ and $V = v_b/(A+1)$. Since $\overrightarrow{v_a} = \overrightarrow{v} + \overrightarrow{v_m}$, $v_a^2 = v^2 + v_m^2 + 2vv_m \cos\theta$. Substituting the values of v and v_m,

$$\frac{v_a^2}{v_b^2} = \frac{E_a}{E_b} = \frac{A^2 + 2A\cos\theta + 1}{(A+1)^2} = \frac{1}{2}\left[(1+\alpha) + (1-\alpha)\cos\theta\right].$$

A neutron undergoes a number of collisions in an NR. An average over θ of the above expression, gives the average fraction of energy lost per collision. Since this value will be a function of α only, it is a constant. So on an average, the neutron loses the same fraction of energy per collision. Then the energy E_n after n collisions is

$$E_n = E_0 e^{-\xi(A)n}$$

where E_0 is the initial neutron energy and $\xi(A)$ is a constant which is a property of the mass number of the moderator atom. Using the property of logarithms, the above expression can be written as

$$\xi(A) = \frac{\ln\frac{E_0}{E_1} + \ln\frac{E_1}{E_2} + \ldots + \ln\frac{E_{n-2}}{E_{n-1}} + \ln\frac{E_{n-1}}{E_n}}{n} = \overline{\ln\frac{E_b}{E_a}},$$

where the last step follows for large n and the bar denotes averaging. Assuming scattering is isotropic in the

CM frame, the weight function is $\sin\theta$. Thus

$$\xi(A) = \frac{\int_0^\pi \ln \frac{E_b}{E_a} \sin\theta \; d\theta}{\int_0^\pi \sin\theta \; d\theta} = 1 + \frac{\alpha}{1-\alpha} \ln \alpha.$$

The moderator in NR is D_2O and assuming naively that the expression we derived holds for a molecule we have $A = 20$ and $\xi = 0.10$. However, an appropriate average of the deuterium and oxygen weighted with the scattering cross-section yields $\xi = 0.51$.

The fission neutrons have an average energy of 1 MeV and the average temperature of the moderator is 353 K = 0.03042 eV. So the number of collisions required to reduce the average energy of the fission neutron from 1 MeV to 0.03042 eV is

$$n = \frac{\ln \frac{E_0}{E_n}}{\xi} = \frac{\ln \left(\frac{10^6 \text{ eV}}{0.03042 \text{ eV}} \right)}{0.51} \backsimeq 34.$$

We are now in a position to estimate the distance between two fuel channels. For simplicity, consider a neutron beam traveling along the x-axis and undergoing only forward scattering. These assumptions will give only a very crude estimate of the distance between two fuel channels. The intensity of the beam $I(x)$ is attenuated, $I(x) = I(0)\; e^{-\Sigma_s x}$, where $I(0)$ is the intensity at $x = 0$ and Σ_s represents the scattering constant. The reader will recall that this attenuation formula holds for light, and indeed for any beam phenomena where scatterings/absorptions are independent events. Let x be the distance traveled by a neutron between two scatterings. Then $\overline{x}$, the average distance a neutron travels without any scattering is

$$\overline{x} = \frac{1}{I(0)} \int_0^\infty I(x)\; dx = \frac{1}{\Sigma_s}.$$

So the average distance a neutron travels in the moderator after it leaves a fuel channel is $n\overline{x} = n/\Sigma_s$. Since the neutron travels in three dimensions, the distance traveled in one dimension is approximately $n/(3\Sigma_s)$, which is also an estimate of the distance between two fuel channels. Substituting $n = 34$ and $\Sigma_s = 45$ m^{-1} for D_2O yields the distance between two fuel channels to be 0.262 m. The actual distance in the Tarapur 3 &4 NR is 0.286 m.

7. The Nuclear Reactor

In the steady state, the NR maybe operated at any constant neutron flux Ψ provided the leakage of neutrons from it is compensated by an excess production of neutrons in it. For a cylindrical reactor, an elaborate calculation yields that the leakage rate is

$$k_1 \left[\left(\frac{2.405}{R} \right)^2 + \left(\frac{\pi}{H} \right)^2 \right] \Psi.$$

The excess production rate is $k_2\Psi$ which is proportional to the flux. The constants k_1 and k_2 depend on the material properties of the NR. Let us consider an NR with $k_1 = 1.021 \times 10^{-2}\ m$ and $k_2 = 8.787 \times 10^{-3}\ m^{-1}$ [6]. Noting that for a fixed volume the leakage rate is to be minimized for efficient fuel utilization, we can obtain the dimensions of the NR in the steady state. For constant volume $V = \pi R^2 H$,

$$\begin{aligned} \frac{d}{dH} \left[\left(\frac{2.405}{R} \right)^2 + \left(\frac{\pi}{H} \right)^2 \right] &= \frac{d}{dH} \left[\frac{2.405^2 \pi H}{V} + \frac{\pi^2}{H^2} \right] \\ &= \frac{2.405^2 \pi}{V} - 2\frac{\pi^2}{H^3} = 0, \end{aligned}$$

gives $\left(\frac{2.405}{R}\right)^2 = 2\left(\frac{\pi}{H}\right)^2$. For steady state,

$$1.021 \times 10^{-2} \left[\left(\frac{2.405}{R} \right)^2 + \left(\frac{\pi}{H} \right)^2 \right] \Psi = 8.787 \times 10^{-3}\ \Psi.$$

This yields $H = 5.866$ m and $R = 3.175$ m. The height and radius of the Tarapur 3 and 4 NR are 5.940 m and 3.192 m respectively.

Finally, we will estimate the critical mass M_F of UO_2 required to operate the NR in steady state. For this, we need to find the number of fuel channels F_n present in the critical volume of the NR calculated above. Though this can be done by rigour, we adopt a quick and easier method. The fuel channels are in a square arrangement with nearest neighbour distance 0.286 m (see previous section); so the effective area per channel is 0.286^2 m^2 = 8.180×10^{-2} m^2. The cross-sectional area of the core is $\pi R^2 = 3.142 \times (3.175)^2 = 31.67$ m^2; so the maximum number of fuel channels that can be accommodated in the cylinder is the integer part of $31.67/0.0818 = 387$. The effective radius of a fuel channel (if it were solid) is 3.617×10^{-2} m [6]. So the mass of fuel,

$$M_F = 387 \times (\pi \times 0.03617^2 \times 5.866) \times 10600 = 9.892 \times 10^4 \text{kg}.$$

The total volume of the fuel is $387 \times (\pi \times 0.03617^2 \times 5.866) = 9.332$ m^3. If the reactor works at 12.5 % efficiency, then using result of Section 2 for Q (the heat produced per unit volume), we have that the power output of the reactor is $9.332 \times 4.917 \times 10^8 \times 0.125 = 573$ MW. We note that the Tarapur 3 & 4 NR has 392 channels and the mass of the fuel in it is 10.15×10^4 kg. It produces 540 MW of power.

8. Concluding Remarks

Design is given short shrift in physics education both in high school and in college. The simple pendulum is simply a point mass with a massless rod. The same holds true in the discussions of the Venturi meter (Bernoulli's principle), engines (invariably the idealized Carnot engine), telescope (ray optics), Van de Graff generator electrostatics) or cyclotron (Lorentz force). These devices and machines are viewed as unidimensional. Generally, they are used to illustrate a single physics principle. The truth is that each of these can be used to illustrate a number of principles from physics and chemistry. We believe that the holistic nature of physics is done a disservice by this exercise.

In the present example we have employed nuclear physics, thermal conduction, mechanical collisions and a whole host of concepts, all at the pre-college level, to unravel the working of a nuclear reactor. Starting from types of reactors and a single fission reaction we arrived at an estimate of the radius of a fuel pin in Section 5. Once again, starting from an elastic collision and a number of admittedly empirical inputs we arrived at the average distance between fuel channels in Section 6. In Section 7 we use the optimisation technique to arrive at the size of the reactor and the fuel mass. This is an illustration of the bottom-up approach. Other notable examples are the Atomic Probe Microscope problem in IPhO – 2004 held in Korea and the Accelerometer and Air-Bag problem in IPhO – 2007 in Iran [7]. Many more attempts of this nature need to be undertaken by physics educationists. The student must know that what they learn is not piecemeal – it is connected and socially useful at the same time.

Acknowledgement

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Suggested Reading

[1] The design of a nuclear reactor was one of the three thematic questions in the International Physics Olympiad 2015. The Olympiad was held in Mumbai, India from July 5 - July 12 2015. A total of eighty three nations and 382 students participated. India's performance was a success and secured medals which included Four Silver and One Bronze.

[2] *Reactors Designed by Argonne National Laboratory: Chicago Pile 1*, Argonne National Laboratory. 21 May 2013.

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[7] See for example www.ipho.org/problems-and-solutions.html