E-LEARNING MODULES FOR NUCLEAR REACTOR HEAT TRANSFER

By

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Abstract

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E learning in engineering education is becoming popular at several universities as it allows instructors to create content that the students may view and interact with at his/her own convenience. Web-based simulation and what-if analysis are examples of such educational content and has proved to be extremely beneficial for engineering students. Such pedagogical content promote active learning and encourage students to experiment and be more creative. The main objective of this project is to develop web based learning modules, in the form of analytical simulations, for the Reactor Thermal Hydraulics course offered by the College of Engineering at UT Arlington. These modules seek to comprehensively transform the traditional education structure. The simulations are built to supplement the class lectures and are divided into categories. Each category contains modules which are sub-divided chapter wise and further into section wise.

Some of the important sections from the text book are taken and calculations for a particular functionality are implemented. Since it is an interactive tool, it allows user to input certain values, which are then processed with the traditional equations, and output results either in the form of a number or graphs.

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Chapter 1

Introduction

In today's world the internet has taken over most of the communication world. Internet has made the world smaller and smaller as we know. Education in this world has been more than just books. The communication between faculty and a student has been more effective with the help of internet. Web based interactive method of learning has made gaining knowledge easier, more fun and most importantly more effective.

1.1 E-learning

E-Learning is learning utilizing electronic technologies to access educational curriculum outside of a traditional classroom. While e-learning has become a primary form of distance learning, it is also transforming the method of education provided in educational institutions. Many students who enroll for courses will come up with several queries while studying and since instructor will not be able to provide solutions there can be online repository which can sometimes help students clarify their doubts. E-learning provide campus based support services. It has addressed the main problem of geography being a barrier for student-institution relationship.

1.1.1 Definitions of e-learning:

1. Classroom Course – Course activity is organized around scheduled class meetings. Some of the courses may involve some sort of computer usage—for example, a software simulation or laboratory or design software for art or engineering

applications—but the course is still anchored to the normal time spent in face-to-face classes. For the purposes of clarity in these definitions, courses use technology inside the classroom.

2. Synchronous Distributed Course - Web-based technologies are used to extend classroom lectures and other activities to students at remote sites in real time. These courses use web tools or other synchronous e-learning media to provide access to a classroom like experience for students at off-campus locations while otherwise maintaining a normal face-to-face classroom schedule.

3. Online Course – All course activity is done online; there are no required face-to-face sessions within the course and no requirements for on-campus activity. Purely online courses totally eliminate geography as a factor in the relationship between the student and the institution. They consist entirely of online elements that facilitate the three critical student interactions: with content, the instructor, and other students. While these courses may appeal to on-campus students, they are designed to meet the needs of students who do not have effective access to campus.

1.1.2 Advantages of e-learning

1. Learning 24/7, anywhere
2. It makes tracking of course progress a breeze
3. User can set his/her own pace
4. Connecting topics while learning is easy with e learning
5. Information reaches faster and its more Flexible

6. Cost effective and saves time
7. Low environmental impact

1.2 E-learning in engineering education

E-learning in engineering side plays an important role in knowledge transfer. The simple concepts as well as complicated definitions are visualized in a better, easy way. Many universities like Penn State, MIT, Carnegie Mellon, UIUC offers e-learning courses or often distance learning, either in the form of a degree, a certification or just for the knowledge purpose. There is a need for E-Learning on engineering domain to be more interactive. So more and more user engaging modules are being developed on the E-learning side every day. Visualization of engineering simulations are more dynamic in E-learning. Industries and companies are also replacing their conventional classroom training with E-learning modules. There are many third party tools like Youtube, Coursera, Khan academy which are integral part of gaining knowledge.

1.3 E-learning in thermal and fluid sciences

There are various modules which are computer based and web based interactive modules which are developed towards thermal and fluid sciences. Any calculation which involves analytical equations or numerical techniques can be easily solved using computers and results can be displayed which helps the user better understand the problem and visualize the trend. Some of the modules which are developed in the thermal area are for example analysation of heat transfer interactions in thermal-fluids systems. This was developed by MIT and is a supplementary teaching tool for the Thermal fluid engineering - 1 course offered at MIT. Many flow simulations are developed to help users analyse, visualize and extract some required data from the module created. Various theories have been developed on fluid side, design, and interactive “What-if” analysis on various design parameters are performed through an active and dynamic learning environments. It is shown that, if used appropriately, web-based learning has the potential to enhance both learning effectiveness and teaching efficiency in the field of engineering education.

Chapter 2

Overview

2.1 Introduction to nuclear energy and nuclear power plant

Nuclear energy is defined as the energy stored inside the nucleus. The process by which the energy gets released is called the nuclear fission. When a heavy atom like U-235 is bombarded with a neutron, U-235 is converted to U-236 which is highly unstable in nature. This atom then splits into 2 parts called fission fragments releasing large amount of energy along with some neutrons. These neutrons further participate in subsequent nuclear fission and so on. The process is shown in the figure below.

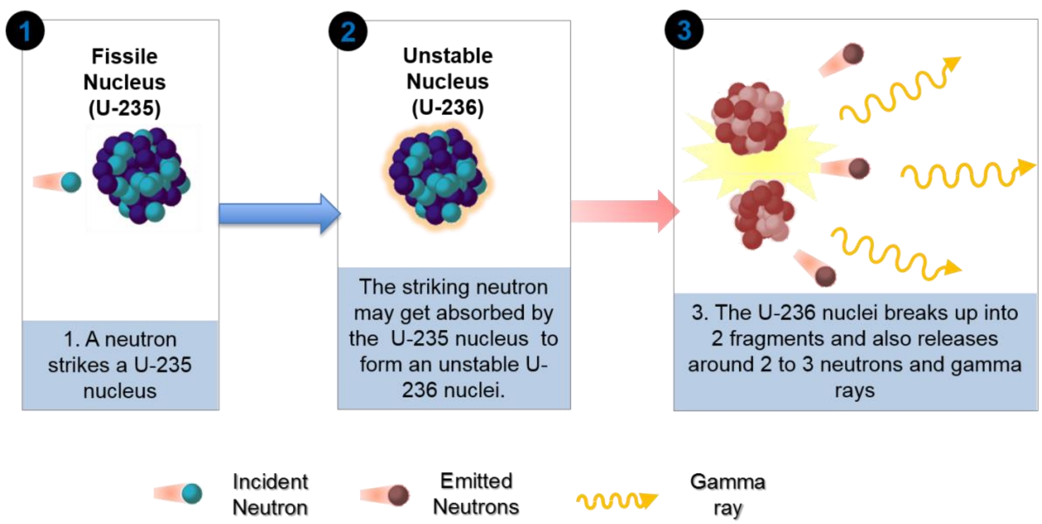

Figure 2.1 Release of nuclear energy

In a typical nuclear reactor, the fuel will be in the form of the cylindrical pellets. The pellets are then stacked on one another to form fuel rods. The fuel rods are then assembled into fuel assembly. These fuel assemblies are placed in the reactor which constitutes to thousands of fuel rods. These fuel assemblies collectively will generate large amount of energy. The energy released will be removed by coolants which will then be used to run the turbines to generate electricity.

The present project handles the calculation of the heat generation, heat transfer and heat removal inside the reactor portion and provides user a perspective how the outcome changes when any of the parameter is changed.

2.3 Thermal challenges in nuclear power plant

Some of the thermal and fluid parameters that are considered during the safety analysis are:

- Fuel pellet materials melts around 3000 °C
- The fuel rod cladding material (Zircaloy) melts at 2200 °C
- The generated heat flux should not exceed a critical value (CHF for BWR and (Departure from Nucleate Boiling) DNB for PWR)

All the modules which are created in the present project are in support to address these thermal challenges which area most commonly faced

2.4 Development of modules:

For developing the modules a powerful tool called u solver has been used. The tool has two parts.

- Spread sheet to perform the business logic, i.e. calculations.
- Html canvas which helps in creating the user interface and aesthetics part of the module.

User can enter input values by entering values in input box, slider, spinner etc. Each of the input box will be connected to a particular cell in the spread sheet so the input values from the user will be captured and further calculation are performed based on the input values. The calculated values are stored in a cell in spreadsheet and are again connected to the output text box in canvas which will be reflected.

The spread sheet have hundreds of ready-made functions that we can use which are very powerful and robust to perform. The formulas are used to perform calculations.

Chapter 3

Modules developed

The modules developed are in parallel to the course work done in class. Some of the important sections from each chapter of the textbook followed in the class are selected and developed. The modules are divided into 4 parts:

1. Fundamentals
2. Heat Generation
3. Heat Transfer
4. Heat Removal

Below are the explanations for each of the modules that are created for the different parts.

3.1 Energy from fission:

Typically when the nuclear fission happens there is a neutron which strikes the radioactive element which splits producing two different lighter nuclei, neutrons and large amount of energy. The total rest masses of the fission products (M_p) from a single reaction is less than the mass of the original fuel nucleus (M). The excess mass is the invariant mass of the energy that is released as (gamma rays) and kinetic energy of the fission fragments. This mass is also called as mass defect.

Figure 3.1 Energy from fission

In this module user selects from pre-defined list of nuclear fission products and accordingly the other fission product is automatically selected and number of neutrons released is calculated. The mass defect $\Delta m = M - M_p$ is calculated. From the Einstein energy to mass equivalence equation, one amu of mass is equivalent to 931 MeV of energy. So multiplying the Δm with 931 MeV will give us the total energy released.

This energy will be further divided into:

a) Energy carried as Kinetic Energy by fission fragments – 80.5%

b) Energy carried by Neutrinos – 5%

c) Gamma Energy by fission products – 3%

d) Rest of energy – 11.5%

The module generates two graphs. One is in the form of pie chart which shows how the energy is split and what the different forms of energy are and what percentage of energy each form carries. The second graph generated shows the total energy and how much the total energy is broken up into individual forms.

3.2 Half-life of radioactive elements:

Half-life of any radioactive material is defined as the time taken for the activity of a given amount of a radioactive substance to decay to half of its initial value. A radioactive material will have an exponential decay. The decay is given by

$$N(t) = N_o e^{-\lambda t}$$

Where,

a) No is the initial quantity of the substance that will decay (this quantity may be measured in grams, moles, number of atoms, etc.),
b) N(t) is the quantity that still remains and has not yet decayed after a time t,
c) λ Is a positive number called the decay constant of the decaying quantity.

Theoretically it takes infinite time for a radioactive material to decay completely. Any material which has reduced to 1/10th of the original amount can be neglected in many practical purposes.

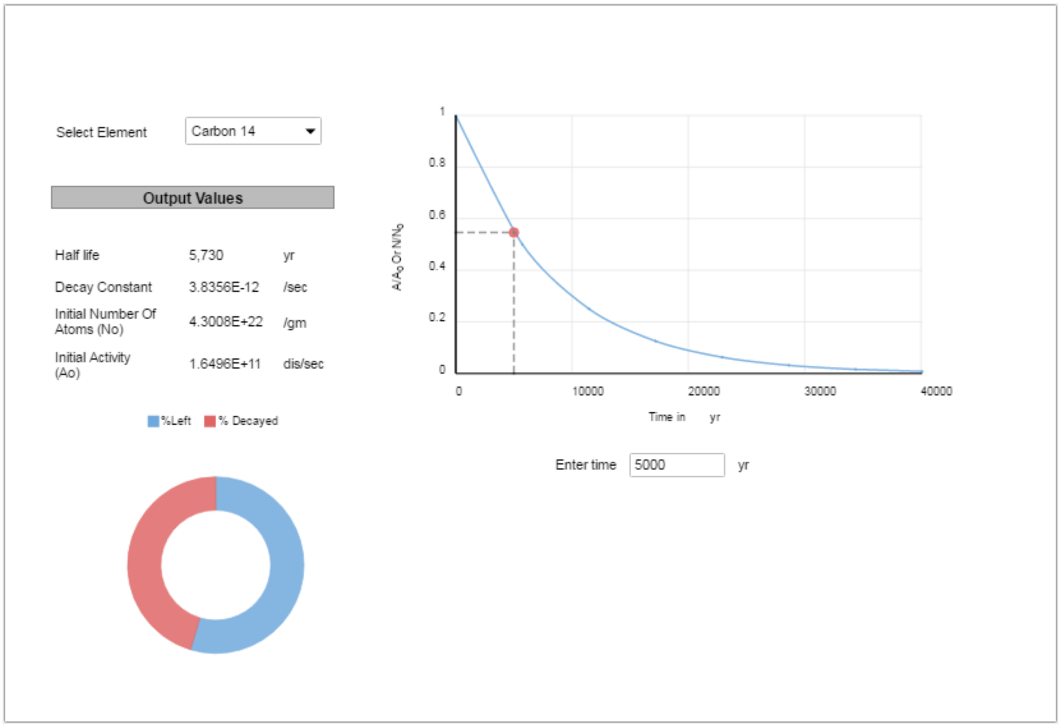

Figure 3.2 Half-life of radioactive elements

In this module user will have an option of selecting a radioactive material from the drop down provided. When user selects a particular element, λ value is calculated and subsequently Number of atoms per gram and Initial activity (A_o) is calculated.

Two graphs are generated for the same. One graph depicts the N(t) / N_o versus time elapsed. Other graph is a pie chart which will give the percentage of decayed material towards percentage of material left. A text box is provided where user can enter the time and the graphs will change and provides the values at that particular time.

3.3 Energy Spectrum of Fission Neutrons:

Newly born fission neutrons constitutes for about 2% of fission energy in the form of kinetic energy. Fission neutrons are divided into two categories, prompt and delayed. The distribution of the number of neutrons with energy in the fission spectrum is well represented by a mathematical function,

$$n(E)dE_n = \sqrt{\frac{2}{\pi e}} sinh\sqrt{2E_n} e^{-E_n} dE_n$$

Where n(E) = the number of neutrons having kinetic energy E_n per unit energy interval dE_n. Prompt neutrons are the first to get released at the time of fission (within 10^{-4} sec after fission occurs). Their energies are comparatively small but play a major role in nuclear reactor control.

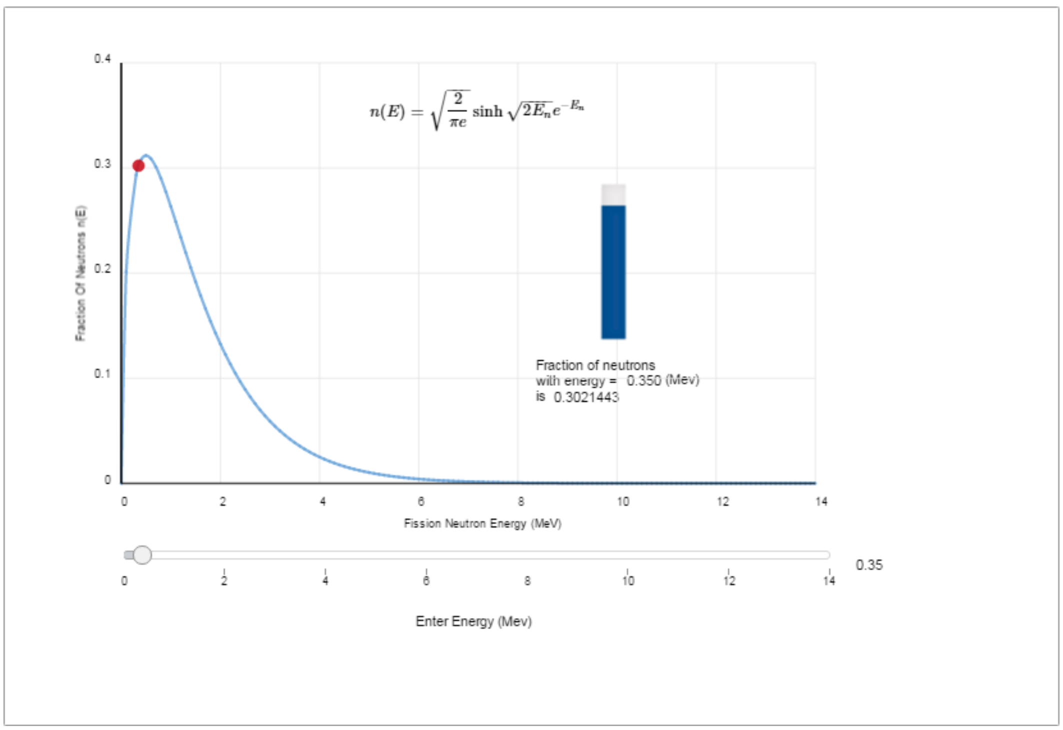

Figure 3.3 Energy spectrum of fission neutrons

The module developed provides energy distribution for the prompt neutrons in graphical form. Users have option to slide through the fission neutron energy and the corresponding fractions of neutrons are displayed for the particular energy user has selected.

3.4 Thermal Neutrons

Fission neutrons are scattered or slowed down by the materials in the core. An effective scattering medium, called a moderator is one which has small nuclei with high neutron scattering cross sections and low absorption cross sections. When neutrons are slowed down in a medium, the lowest energies that they can attain are those that put them in thermal equilibrium with the molecules of that medium. These neutrons are called thermal neutrons.

Particles or molecules at a particular temperature possess a wide range of kinetic energies and corresponding speeds. The speed corresponding to the maximum density (V_m) is called the most probable speed. The distribution follows the Maxwell distribution law.

$$n(V)dV = 4\pi n\left(\frac{m}{2\pi kT}\right)^{1.5} V^2 e^{-\frac{mv^2}{2kt}} dV$$

Where,

n(V) = Number of particles present in given volume of medium, with speeds between V and V + dV

n = Number of particles in same volume of medium

m = Mass of particle

k = Boltzmann's constant

T = Absolute temperature

The most probable speed is found out by differentiating the right hand side of equation with respect to V and equating the derivative to zero. Solving the equation will be reduced to

$$V_m = \left(\frac{2kT}{m}\right)^{0.5}$$

And the corresponding kinetic energy is given by expression

$$KE_m = kT$$

By using the proper value of neutron mass and Boltzmann constant we get

$V_m = 1.2839 \text{ X } 10^2 \ T^{0.5}$ (m/sec) ……………….for a neutron only

$KE_m = 8.6164 \text{ X } 10^{-5}$ T (eV) …………………for any particle

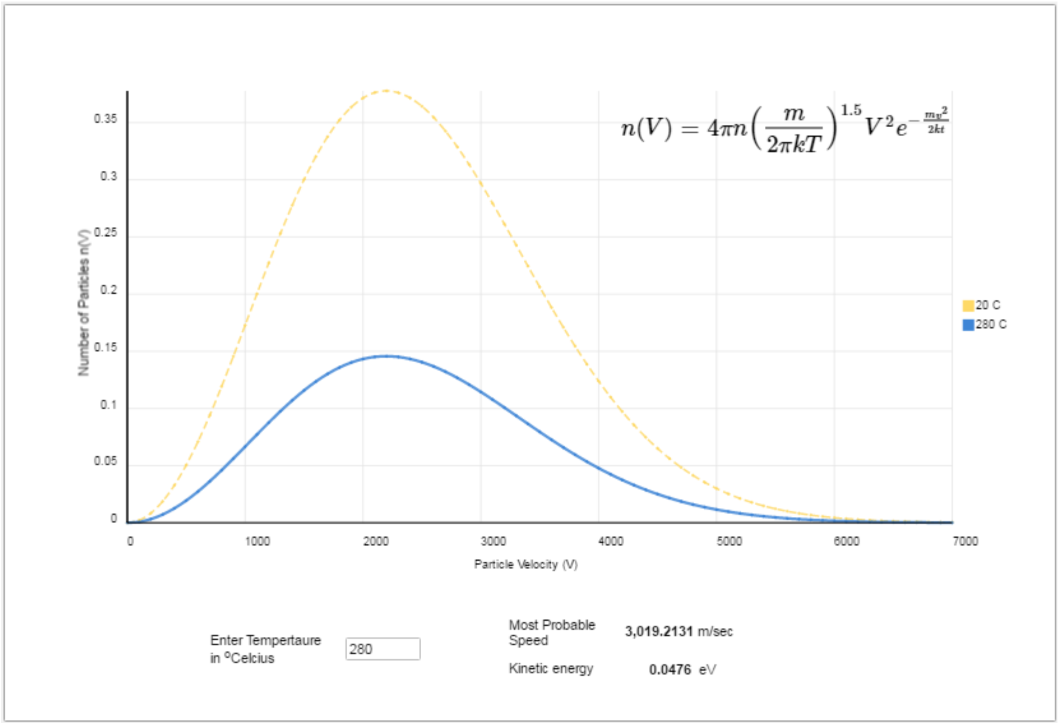

Figure 3.4 Thermal Neutrons

The module developed will have two distribution curves. One corresponding to 20 °C and other with the user entered temperature. For a particular temperature entered by the user the most probable speed and the kinetic energy corresponding to that speed is displayed. User should be able to have a feeling about how the graph varies with the temperature and compare the most probable speeds with different temperatures.

3.5 Nuclear cross sections

In nuclear reactions, the probability with which the neutrons collide or interact with nuclei is proportional to an effective, rather than actual cross sectional area. This probability is called microscopic cross section or simply the cross section (σ).

Below figure shows the energy of released neutrons versus nuclear cross section. The energy of the released neutrons will fall into the high energy level. These neutrons must be brought into lower energy levels which are called thermal neutrons. It can be seen that thermal neutrons have higher probability or higher neutron cross section and hence it is very important that the neutrons required for subsequent nuclear fission reaction loose energy and do fall in this region.

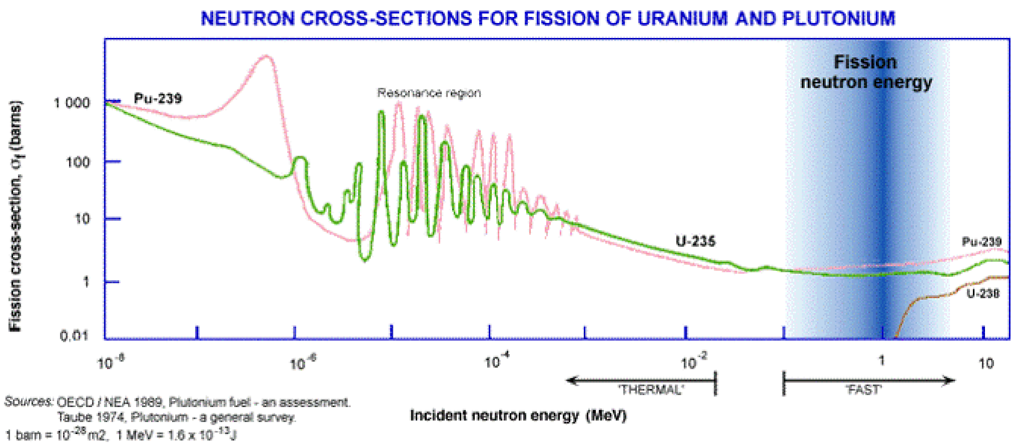

Figure 3.5 Neutron cross-sections for fission of uranium and plutonium

Since the nuclear dimensions are very small, square centimeters would be too large. So the unit of the cross section was defined as barn which has a value equal to 10^{-24} cm^2. There can be many microscopic cross sections as there are possible reactions. The most important are absorption cross section and scattering cross sections. Sometimes the total cross section is also used which is the sum of all possible cross sections.

The product of microscopic cross section and nuclear density is called macroscopic cross section (∑). It has the unit $(cm)^{-1}$.

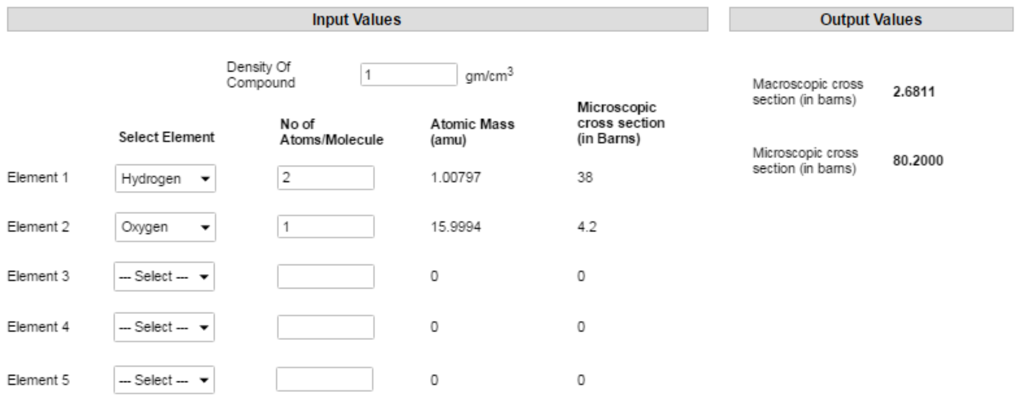

Figure 3.6 Nuclear cross section

The module lets the user calculate the microscopic cross section and the macroscopic cross section of any compound given. The user will be able to select up to 5 elements in a compound and the module picks the corresponding atomic mass and the corresponding microscopic cross section for that particular element selected. The user should provide the number of atoms present in the compound for each element and the density of the compound. The module calculates the total microscopic cross section and the macroscopic cross section of the compound.

3.6 Nuclear Flux distribution

Nuclear flux is defined as the number of neutrons passing through a unit area from all directions per unit time. Since flux defines the density of the neutrons at a given point, the reaction rate between neutrons and nuclei is therefore proportional to it.

Neutron flux is the product of Neutron density and Neutron velocity i.e. ϕ = nV neutrons/sec cm^2. The reactor equation is given as $\nabla^2\phi + B^2\phi = 0$ where B is the geometrical buckling.

For parallelepiped the reactor equation is given as $\frac{\partial^2\phi}{\partial x^2}+\frac{\partial^2\phi}{\partial y^2}+\frac{\partial^2\phi}{\partial z^2}+B^2\phi=0$

For sphere the reactor equation is given as $\frac{\partial^2\phi}{\partial r^2}+\frac{2}{r}\frac{\partial\phi}{\partial r}+B^2\phi=0$

For cylindrical the reactor equation is given as $\frac{\partial^2\phi}{\partial r^2}+\frac{2}{r}\frac{\partial\phi}{\partial r}+\frac{\partial^2\phi}{\partial z^2}+B^2\phi=0$

The solution to the above equations will yield the neutron flux distribution in terms of spatial co-ordinates.

- For parallelepiped : $\phi_{co}\cos\left(\frac{\pi x}{a_0}\right)\cos\left(\frac{\pi y}{b_0}\right)\cos\left(\frac{\pi z}{c_0}\right)$
- For Sphere : $\frac{\phi_{co}}{\pi r/R_e}\sin(\frac{\pi r}{R_e})$
- For cylinder : $\phi_{co}\ \cos\left(\frac{\pi z}{H_e}\right)\ \ J_0(\frac{2.405r}{R_e})$

ϕ is positive at any point in the reactor and equal to zero if the point of interest is outside the reactor. Where, ϕ_{co} is the flux at the geometrical center of reactor core.

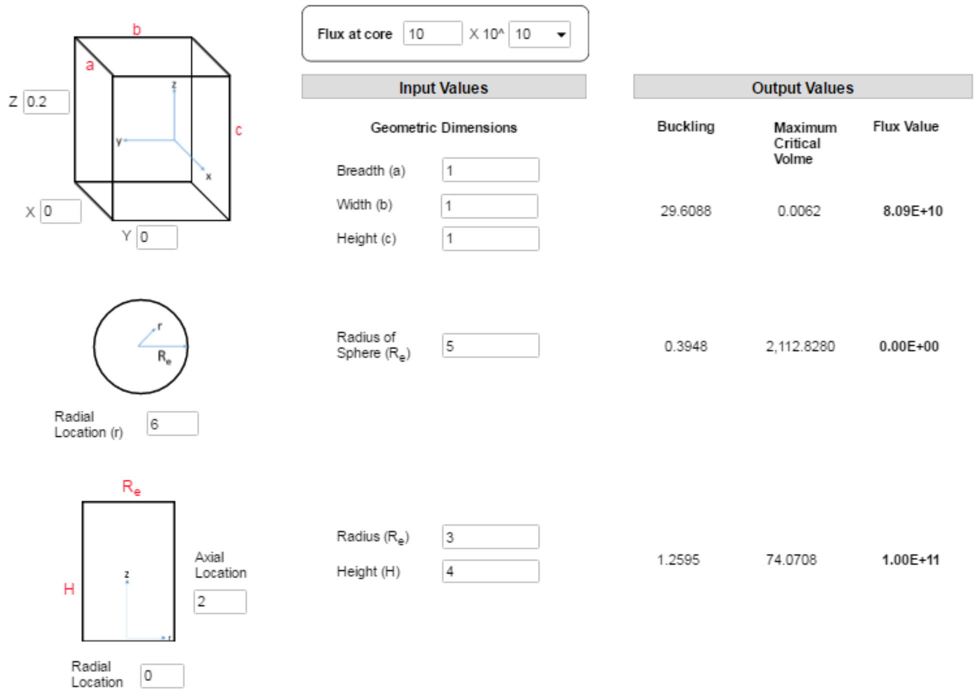

Figure 3.7 Nuclear Flux distribution

The module created uses reactor of parallelepiped, sphere and cylindrical shapes. The module allows user to enter the geometrical dimension of the reactor. User also enters the point of interest where the flux value has to be calculated. If the flux at the core is given then the module calculated the geometrical buckling, flux at the point of interest.

3.7 Heat generated at a location

When nuclear fission occurs, large amount of heat is generated in the reactor. It is very important to know how much heat is generated within a reactor in order to design the heat removal process efficiently so that the reactor works within safe limits. The heat generated (q''') is directly proportional to flux at the point of interest and the expression is given by

$$q''' = G_f \, N_{ff} \, \sigma \, \Phi$$

- G_f = Energy per fission reaction
- N_{ff} = Density of fissionable fuel
- σ = Effective fission microscopic cross section
- ϕ = Neutron flux at the point of interest

N_{ff} is given by $N_{ff} = \frac{A_v}{M_{ff}} r \rho_{ff} f i$

- A_v = Avogadro number
- M_{ff} = Molecular mass of fissionable fuel used.
- r = enrichment of the fuel
- f = mass fraction of the fuel in fuel material
- ρ_{ff} = Density of fissionable fuel used
- i = Number of fuel atoms per molecule of fuel

And neutron flux is given by $\varphi = \varphi_{co} \cos\frac{\pi z}{H_e} \; J_o\left(\frac{2.4048\, r}{R_e}\right)$

- z = axial location of point of interest
- r = radial location of point of interest

Heat generated q''' is in MeV/sec cm^3

- Multiply by 1.602×10^{-10} to convert to kW/m^3
- Multiply by 1.5477×10^{-8} to convert to Btu/hr ft

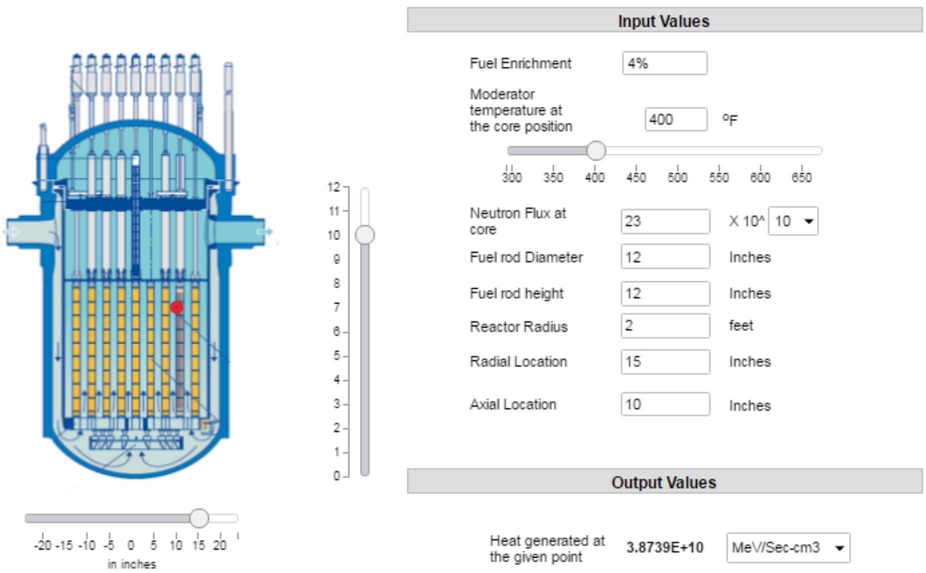

Figure 3.8 Heat generated at a location

The module developed will help user know the heat generated at any location. User will enter the required input values like enrichment, moderator temperature and dimensions of the reactor and the module calculates the heat generated at the desired location. User will have an option to move through the sliders and the value of the heat generated changes dynamically. This will help user get a better picture as to how the heat is distributed inside a reactor and user will be able to compare the values at different locations.

3.8 Heat generated in a single rod

The neutron flux in a single fuel element is not constant. A fuel element will be situated vertically such that the height of the fuel element (H) will be equal to the height of the reactor. The variation of flux is purely a cosine function of z and the maximum value of the flux and heat generated occur in a single fuel element at its centre. The total heat generated by one fuel element can be found out by the following expression.

$$q_t = \int_{-H/2}^{H/2} q'''(\mathrm{z}) A_s dz$$

Where A_s is the cross sectional area of the fuel element. After solving the equation we end up with the expression,

$$q_t = \frac{2}{\pi} q_c''' A_s H$$

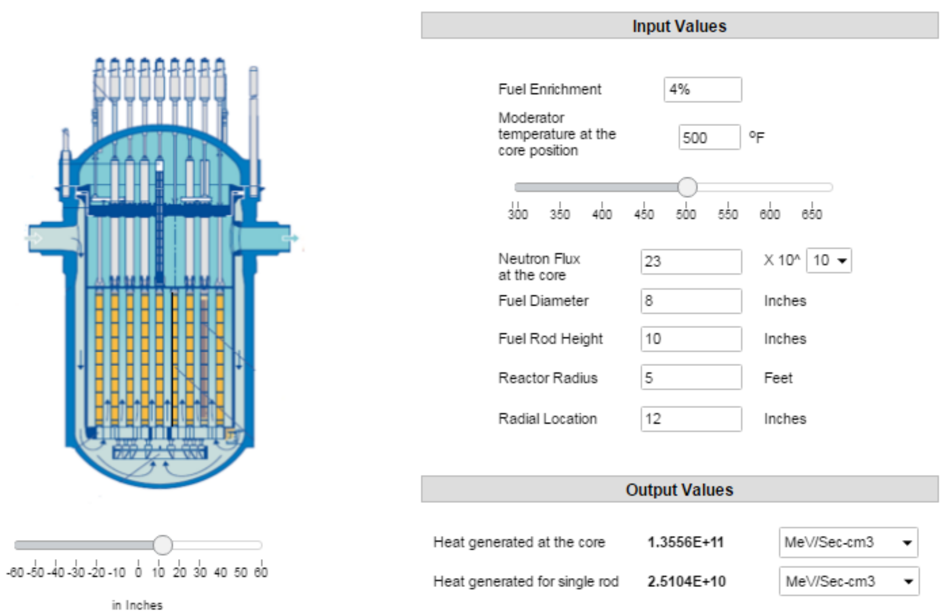

Figure 3.9 Heat generated in a single rod

The module developed lets user calculate how much heat is generated for a single fuel rod and also the heat generated at the core. The heat generated in a single fuel depends on the position of the fuel rod inside the reactor. So user has an option of selecting the fuel rod position inside the reactor with the help of slider and the value of the heat generated for that single rod at that particular radial location will be calculated.

3.9 Total heat generated inside a reactor

In a typical reactor there is no single fuel rod and it is always a combination of number of fuel rods assembled. The total heat generated inside the reactor is the cumulative sum of all the heat generated by individual fuel rods assembled inside it, heat generated by the structural components, coolant moderator and other components due to radiation. In order to evaluate the total heat generated, the flux distribution throughout the core should be known. The expression for total heat generated inside the reactor is given by

$$Q_t = 0.289\, n q_c''' A_s H$$

Where, n = total number of fuel rods assembled inside the reactor

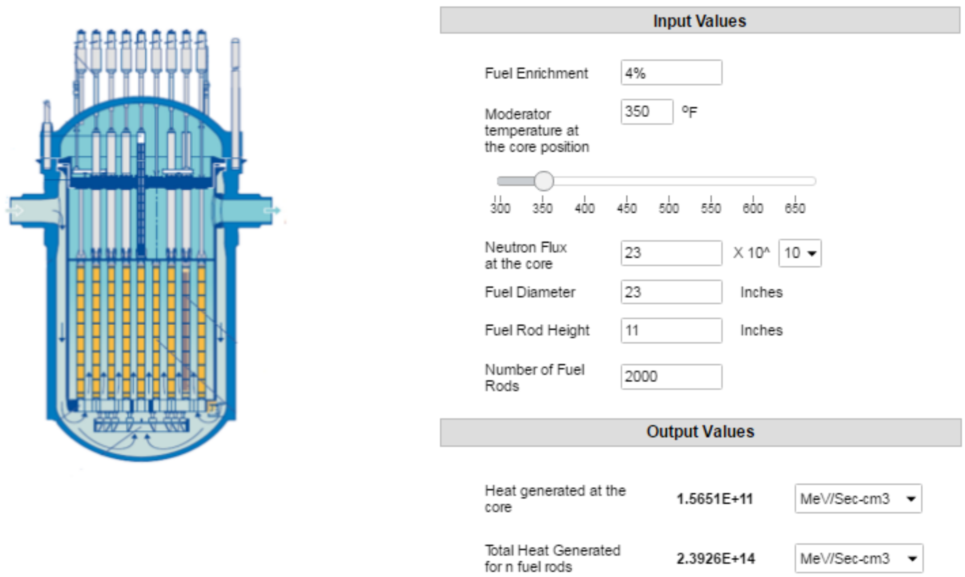

Figure 3.10 Total heat generated inside a reactor

The module developed allows user to calculate the total heat generated inside a core. User will enter some of the required input values like enrichment, dimension of the fuel rod and total number of fuel rods. The module calculates and outputs the value of heat generated at the core and total heat generated for the given fuel rods assembled.

3.10 Reactor shutdown

When the reactor is shutdown the reactor power does not immediately drop to zero but falls off rapidly. The fission fragments and fission products existing in the fuel continue to decay at negative rate for long periods of time. The reactor produces about 3% of its original power and considering the total heat generated, this number is significant. The generation of power is a function of time. So it is important to analyze the heat generated after the reactor is shut down and the heat removal process should be designed accordingly.

The ratio of volumetric thermal source strength after shutdown to that before shutdown is same as the ratio of respective powers i.e. $\frac{q_s'''}{q_o'''} = \frac{P_s}{P_o} = 0.095\ \theta_s^{-0.26}$. Total energy release as a function of time is given as $E_s = 0.128\ \theta_s^{0.74}$ where θ_s is the time elapsed time after shutdown of the reactor.

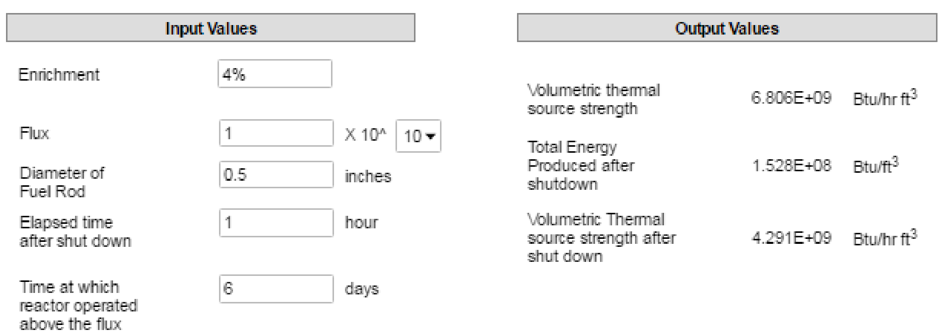

Figure 3.11 Reactor shutdown

The module calculates the volumetric thermal strength before and after the shutdown of reactor and total energy produced after shutdown.

3.11 Heat conduction through the solid plate fuel

After finding out about the heat generated in fuel elements. It is now time to find how the heat is transferred or distributed inside the fuel elements and also the components surrounding it. The nuclear fuel comes in various geometrical shapes and one such type which is commonly found are solid plate type fuel and cylindrical type fuel. Even though the solid plate type fuel are commonly found in the research reactors it is important to know how the temperature is distributed in it to start with.

The temperature distribution inside a plate type fuel is assumed to be a one dimensional heat conduction and the modelling will be done. We will assume the thermal conductivity of the fuel and cladding, physical properties of coolant is constant throughout. The heat transfer co-efficient between solid and coolant will be considered constant too. The fuel element is surrounded by cladding and coolant is flown upon that. Now the temperature distribution is given by the expression

$$t_m - t_f = \frac{q'''S^2}{2k_f} + q'''S^2\left[\frac{c}{k_c} + \frac{1}{h}\right]$$

Also can be written as $q_s = \frac{t_m - t_f}{\frac{S}{2K_f A} + \frac{c}{K_c A} + \frac{1}{hA}}$

Where,

$t_m - t_f$ = temperature difference between the fluid and core of fuel.

$\frac{q'''S^2}{2k_f}$ = temperature drop across the core.

$\frac{q'''s^2c}{k_c}$ = temperature drop across the cladding.

$\frac{q'''s^2}{h}$ = temperature drop across the fluid coolant.

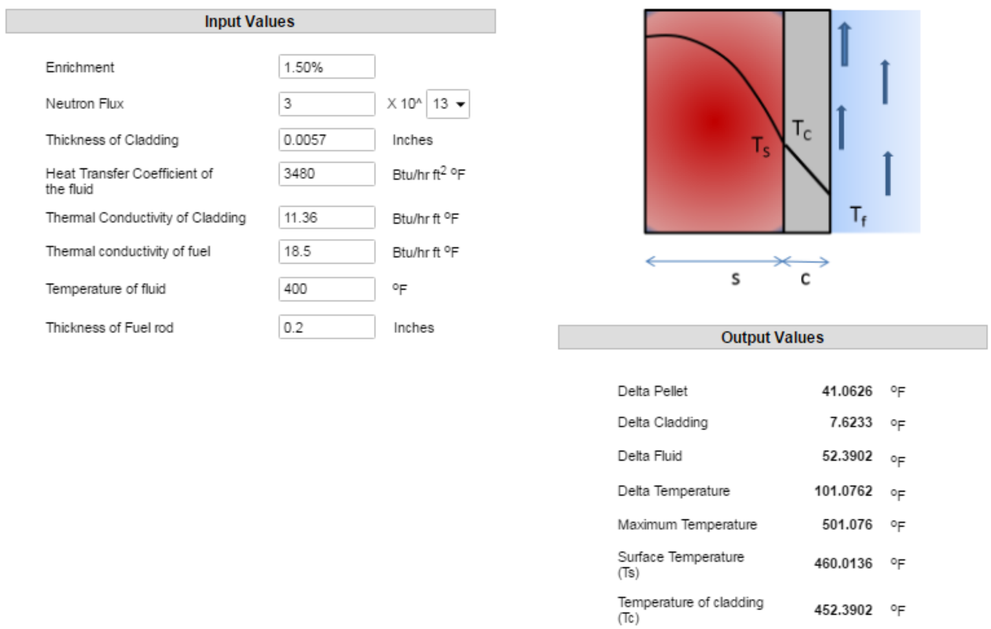

Figure 3.12 Heat conduction through the solid plate fuel

The module developed lets the user enter the thermal and fluid properties of fuel, clad and coolant. User should also enter the dimensional properties of fuel. The module calculates the individual temperature drops across fuel, cladding and fluid. It also calculates the surface temperature of fuel and cladding.

3.12 Heat flow out of solid cylindrical fuel elements

Heat flow out of the cylindrical fuel elements will substantially be radial direction and will be equal in all directions. It should be noted that in cylindrical type fuel, there is a small helium gap between the fuel element and cladding to permit better thermal contact. The helium gap is so small that there won't be any convection effect. The heat transfer can be totally assumed to be through conduction. The expression for the temperature difference between fuel and fluid is given as

$$t_m - t_f = \frac{q'''R^2}{4k_f} + \frac{q'''R^2}{2}\left[\frac{1}{k_{He}} ln \frac{R+\delta}{R} + \frac{1}{k_c} ln \frac{R+c+\delta}{R+\delta} + \frac{1}{h(R+c+\delta)}\right]$$

Where,

$t_m - t_f$ = temperature difference between the fluid and core of fuel.

$\frac{q'''R^2}{4k_f}$ = temperature drop across the core.

$\frac{q'''R^2}{2} \times \frac{1}{k_{He}} ln \frac{R+\delta}{R}$ = temperature drop across cladding.

$\frac{q'''R^2}{2} \times \frac{1}{k_c} ln \frac{R+c+\delta}{R+\delta}$ = temperature drop across helium gap.

$\frac{q'''R^2}{2} \times \frac{1}{h(R+c+\delta)}$ = temperature drop across at fluid.

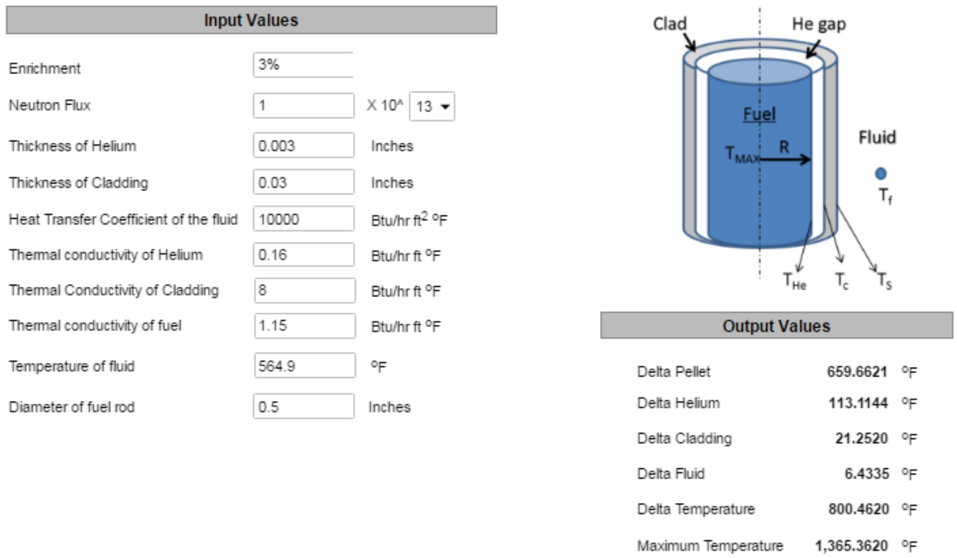

Figure 3.13 Heat flow out of solid cylindrical fuel elements

The module developed lets the user enter the thermal and fluid properties of fuel, clad, helium and coolant. User should also enter the diameter of the fuel. The module calculates the individual temperature drops across fuel, cladding, helium gap and fluid. The module also outputs the maximum temperature which at the core of fuel.

3.13 Heat conduction in shielding materials

Strong gamma radiations, neutron and other radiations emanate from active reactor cores. These radiations get absorbed by the surrounding materials like pressure vessels, shields and other structural materials. The radiation absorbed by these materials also plays an important part while designing the cooling system. The pressure vessels and shielding materials are cylindrical in nature. The diameter to thickness ratio is so small that the problem can be treated as flat slab.

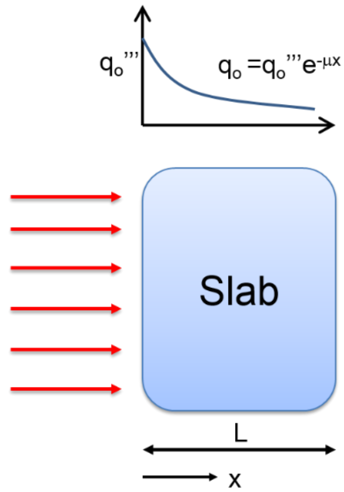

Figure 3.14 Heat transfer in a body subjected to radiation from one side

The expression to find the temperature at any point in the slab is given by

$$t(x) = t_i + (t_o - t_i)\frac{x}{L} + \frac{q_o'''}{\mu^2 k}\left[\frac{x}{L}(e^{-\mu L} - 1) - (e^{-\mu x} - 1)\right]$$

Where, μ = absorption co-efficient.

t_i = inner temperature of the slab

t_0 = outer temperature of the slab

k = thermal conductivity of the slab

The location at which the maximum temperature occurs is given by the expression

$$x_m = -\frac{1}{\mu}\ln\left[\frac{\mu k}{q_o'''L}(t_i - t_o) + \frac{1}{\mu L}(1 - e^{-\mu L})\right]$$

Heat generated at the inner surface and outer surface of the slab is given by the expressions

$$q_{x=0} = \frac{kA(t_i - t_o)}{L} - \frac{q_o'''A}{\mu}(1 + \frac{e^{-\mu L} - 1}{\mu L})$$

$$q_{x=L} = \frac{kA(t_i - t_o)}{L} - \frac{q_o'''A}{\mu}(e^{-\mu L} + \frac{e^{-\mu L} - 1}{\mu L})$$

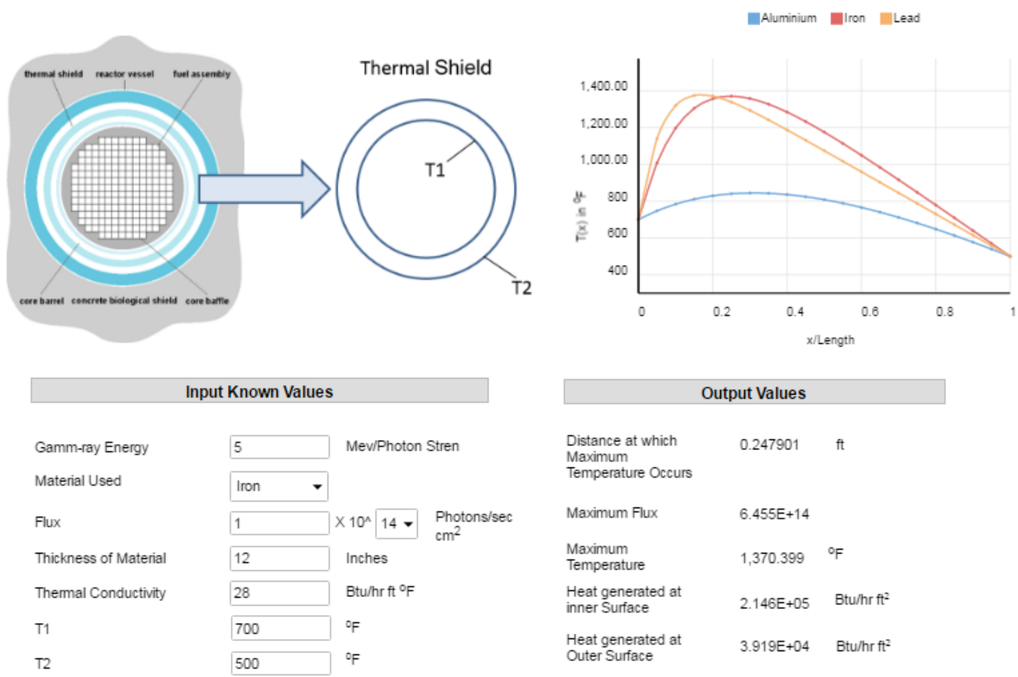

Figure 3.15 Heat conduction in shielding materials

The module created will solve the above expressions and gives output in the form of numbers. The user has an option of choosing a particular shielding material such as iron, lead and aluminium. The user should provide the gamma ray energy and the surface temperature on each side. The module also outputs the graph which shows the temperature distribution inside the slab by dividing the slab into ten equal parts. The temperature distributions for different materials are also plotted in the same graph. This will help user visualize and compare the temperature distributed for different shielding materials.

3.14 Time constant

The thermal capacity of the body is treated as a single or lumped parameter. The energy balance at time θ for the particular body would be given as

$$\frac{t_2 - t(\theta)}{t_2 - t_1} = e^{-(\frac{hA}{c\rho V})\theta}$$

Where, $\frac{c\rho V}{hA} = \tau$ is called time constant for the body. It is the product of its thermal capacitance $c\rho V$ and external thermal resistance $\frac{1}{hA}$.

The time constant is the measure of rapidity of the response of a body to environmental temperature changes. To avoid an unsafe condition the time constant of fuel and moderator must be short so that core temperature would follow power changes and temperature coefficient takes effect rapidly.

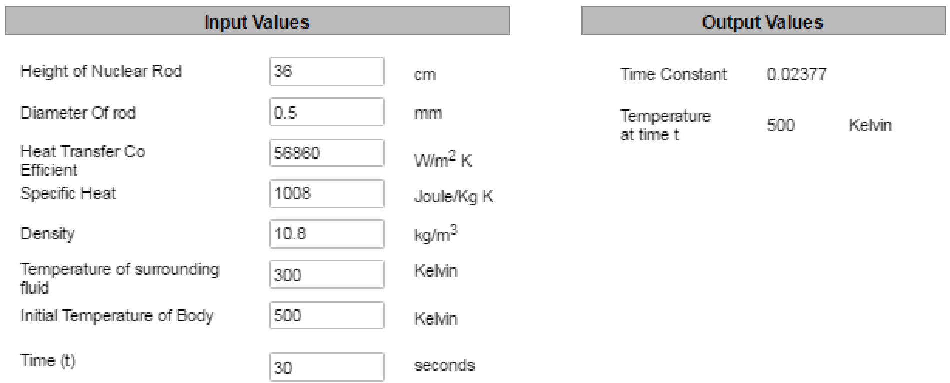

Figure 3.16 Time constant

The module calculates the time constant value and the temperature for the time desired when user provides the different parameters like density, heat transfer co-efficient. User should also provide the temperature of the surrounding fluid and initial temperature of the body.

3.15 Graphical solution for the fixed boundary condition

Numerical techniques are powerful tools which are used to calculate the temperature distribution when the temperature change inside the body is dependent on time and spatial co-ordinates. These results when presented in the form of charts will yield better realisation of results. Graphical methods use numerical techniques to generate a series of solutions giving a continuing picture of temperature profiles at preselected time intervals. The heat conduction within a body is considered to be symmetric about the middle axis hence only half of the volume is considered for the calculation. The given medium is divided into n number of equal parts and with Δx is the nodal distance and $\Delta\theta$ is the time step considered.

$$t_n^{\theta+\Delta\theta} = (1-2Fo)t_n^{\theta} + Fo\left(t_{n-\Delta x}^{\theta} + t_{n+\Delta x}^{\theta}\right) + \frac{q'''\Delta\theta}{\rho c}$$

Where,

n = n^{th} node

$\Delta\theta$ = Time interval

F_o = Fourier number $\left(\frac{\alpha\Delta\theta}{(\Delta x)^2}\right)$

Δx = Nodal distance

q''' = Heat generated at core

ρ = Density of material

c = Specific heat

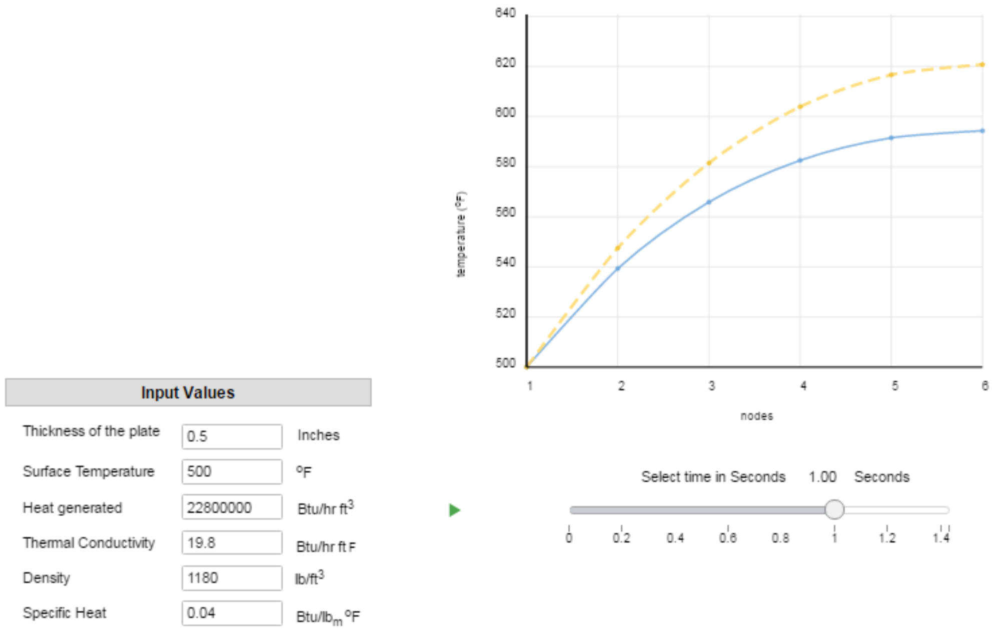

Figure 3.17 Graphical solution for the fixed boundary condition

The module developed will calculate temperature distribution inside a slab for first 25 time steps and produce the results in graphical form. One line in the graph is fixed for the maximum temperature it attains for the 25th time step. User can also animate how the temperature is distributed for the first 25 time steps in the graph.

3.16 Heat flow through circular channels

Cooling is the most important part of the nuclear reactor. Number of coolants such as light and heavy water, organic liquids and some gases are used to remove the heat generated by the fuel. Choosing a particular coolant for a reactor to remove the heat is also an important decision. Number of characteristics is considered while selecting the coolant.

Economic -

- Low initial cost
- Abundantly available

Physical –

- Low vapour pressure
- Low melting point
- Good thermal stability

Nuclear –

- Low neutron absorption cross – section
- Low induced radio activity
- Good radiation stability

One of the most important parameter is the heat transfer co-efficient of the coolant. The heat transfer co-efficient are calculated with the help of Nusselt number given by $Nu = \frac{\mathrm{h}D_e}{k}$.

The Nusselt number is a function of Reynolds number $Re = \frac{\rho V D_e}{\mu}$ and Prandtl number $\mathrm{Pr} = \frac{c_p \mu}{k}$

Depending on the type of flow there are several co-relations which are developed to find out the Nusselts number. Some of them are –

- The Dittus – Boelter equation -

$$Nu = 0.023\ Re^{0.8}Pr^{0.4}$$

- The Seider – Tate equation -

$$Nu = 0.023\ Re^{0.8}Pr^{0.4}\left(\frac{\mu_w}{\mu}\right)^{0.14}$$

Where, w and m are the properties evaluated at the wall or film temperatures respectively.

- For Organic coolants -

$$Nu = 0.015\ Re^{0.85}Pr^{0.3}$$

- For heat transfer to superheated steam at high pressures -

$$Nu = 0.0214 Re^{0.8}Pr^{\frac{1}{3}}\left(1 + \frac{2.3}{L/De}\right)$$

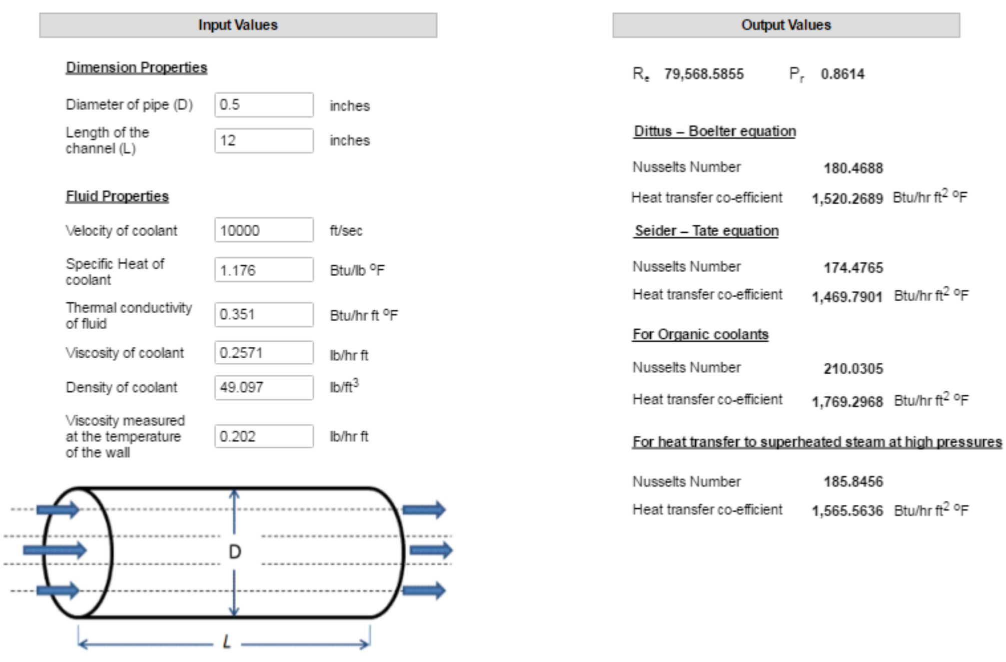

Figure 3.18 Heat flow through circular channels

For the module developed, user has to provide the geometric properties of the pipe and also the fluid properties of the coolant. The module calculates the Nusselts number and the heat transfer co-efficient for each of the co-relations mentioned above and the values are shown.

3.17 Heat flow through non circular channels

In practical purpose the flow of the coolant is never through the circular channels. The coolant always flows through the fuel rods whose cross section is always non-circular. When flow occurs in non-circular tubes the concept of equivalent diameter or effective diameter (De) comes into picture.

$$De = \frac{4\ X\ cross - sectional\ area}{wetted\ perimeter}$$

Some of the co-relations for the flow through non-circular channels and the calculations of equivalent diameter are given below

- Flow parallel to rod bundles

$$Nu = CRe^{0.8}Pr^{\frac{1}{3}}$$

C = $0.042\ \frac{S}{D}\ - 0.024$ For the square lattices

C = $0.026\ \frac{S}{D}\ - 0.006$ For triangular lattices

- Flow across rod bundles

$$Nu_D = 0.33CF\left(\frac{G_m D}{\mu}\right)^{0.6} Pr^{0.3}$$

- Flow between parallel plates

$$Nu = 0.023Re^{0.8}Pr^{0.4}$$

$$D_e = 4\ \frac{ab}{2a + 2b}$$

- Flow in an annuli

$$\frac{h}{C_p G} Pr^{2/3}\left(\frac{\mu_w}{\mu}\right)^{0.14} = \frac{0.021(1 + \frac{2.3D_e}{L})}{\left(\frac{D_e G}{\mu_m}\right)^{0.2}}$$

$$D_e = D_2 - D_1$$

Where, w and m are the properties evaluated at the wall or film temperatures respectively.

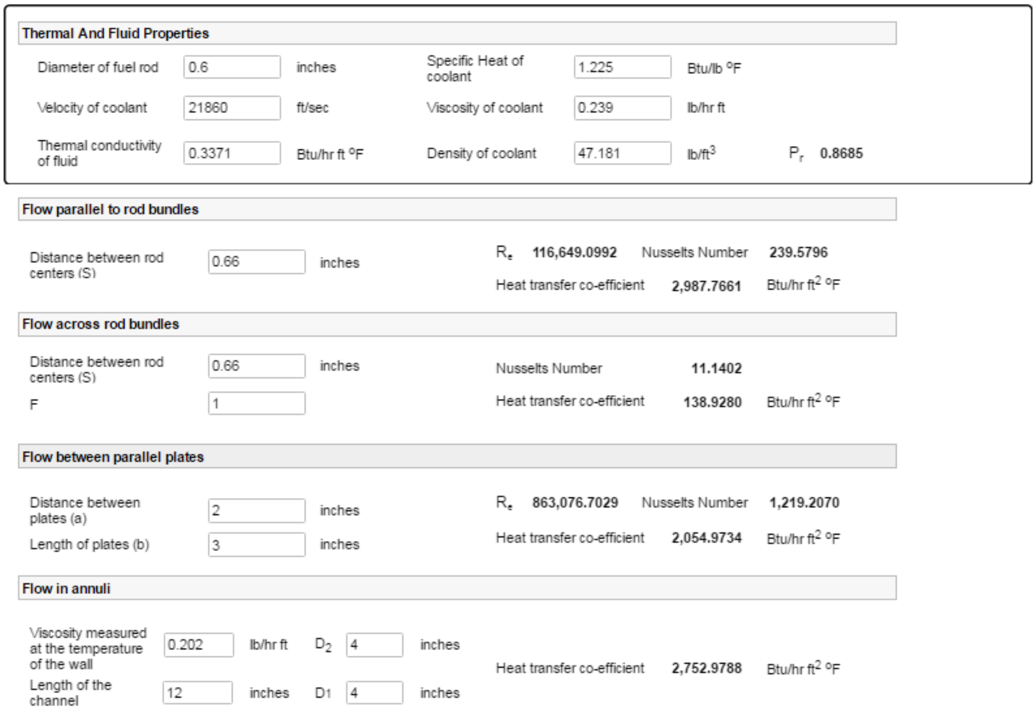

Figure 3.19 Heat flow through non circular channels

The module developed lets the user enter the input values which are used for the general calculation and also which are specific to a particular co-relation. The module calculates the equivalent diameter for the required cases and the heat transfer co-efficient using the Nusselt number for each of the co-relations defined above.

3.18 Critical heat flux

It is important to know that many coolants which are used in the reactors are subjected to change in phase because of the contact with high heat fluxes. Critical heat flux describes the thermal limit of a phenomenon where a phase change occurs during heating (such as bubbles forming on a metal surface used to remove heat to the coolant), which suddenly decreases the efficiency of heat transfer, thus causing localised overheating of the heating surface.

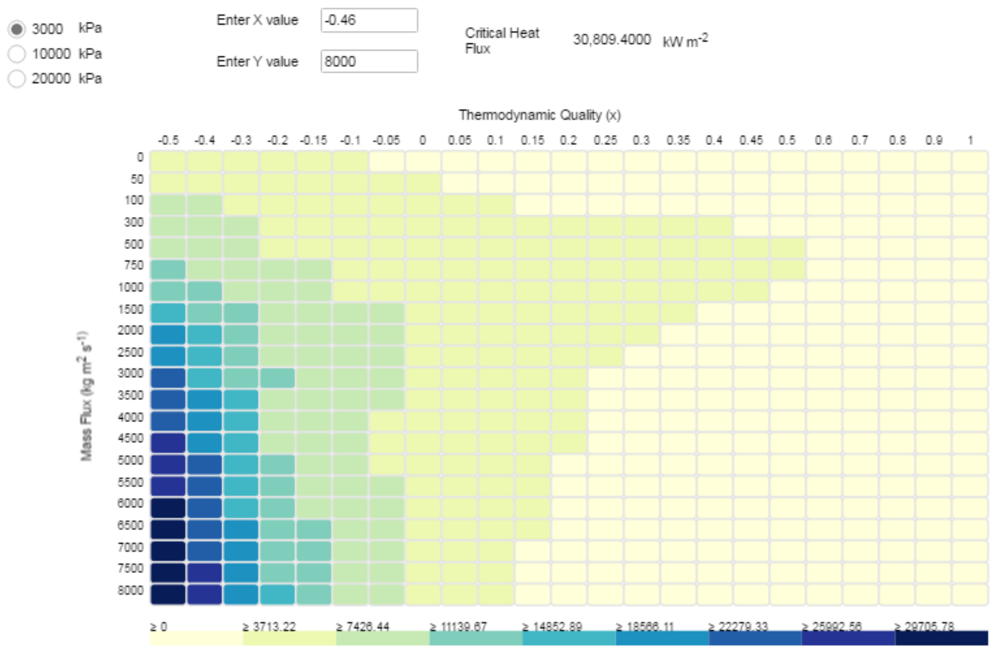

Figure 3.20 Critical heat flux

The module developed has some of the critical heat flux data collected for different pressure values, different mass fluxes values and for different thermodynamic qualities. The values are shown as the heat map and are distributed throughout the map. User can select a particular pressure, enter the mass flux and thermodynamic quality for which the value of critical heat flux is desired. The module then interpolates between the known data and critical heat flux value is shown as an output.

Chapter 4

Conclusions and future work

4.1 Conclusion

After a through literature review, different aspects of nuclear energy, nuclear heat generation, heat removal was studied. The need for e-learning in nuclear reactor heat transfer course was studied and accordingly different modules were developed picking some of the important sections from the textbook followed in the class room.

In each module, all the required formulas were formulated which are used to perform calculations and necessary output values were displayed. Various graphs like line graphs, bar graphs and pie charts are plotted whenever necessary to simplify user understanding. This will also provide user a continuing picture and see a trend in output values. Aesthetic aspects of the modules are also taken into consideration and several images are shown in the modules to make it as user friendly as possible.

A web page is developed where all the modules are integrated on a single platform for easy access. All the modules are divided into different parts for ease of access and user can traverse through the different modules very easily.

Master tables were developed whenever there was a need for the data to be fetched automatically. These master tables serve as a data repository that is used to automate the data-fetching process, minimizing the user dependency to input some of the general and constant values.

4.2 Future work

Although the modules developed are made as much user friendly and robust as possible, since development is a continuous process there is always scope for enhancing the current project to make the module more robust.

Some of the new modules can be developed such as graphical solution for the convective boundary condition, heat flow over circular channels and non-circular channels for Liquid metal coolant, Critical Heat Flux calculation and some more.

All the modules developed consider British thermal units for all the calculation. However in order to make the existing modules more robust there has to be a provision for user to enter from some of the standard units of his choice. So keeping this in mind, unit conversions for all the input values and output values can be done.

While developing any application on web, handling negative cases or error capturing is very important. Some of the unexpected values which show up in the calculation are better when not shown. Limiting users to enter the input values within a particular range should also be taken care of. Appropriate error messages should be shown whenever user enters undesired values.

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