

**OPERATIONAL REACTOR PHYSICS ANALYSIS CODES (ORPAC)**

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GOVERNMENT OF INDIA  
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MUMBAI, INDIA  
2007

**BIBLIOGRAPHIC DESCRIPTION SHEET FOR TECHNICAL REPORT**  
(as per IS : 9400 - 1980)

01	<i>Security classification :</i>	Unclassified
02	<i>Distribution :</i>	External
03	<i>Report status :</i>	New
04	<i>Series :</i>	BARC External
05	<i>Report type :</i>	Technical Report
06	<i>Report No. :</i>	BARC/2007/E/008
07	<i>Part No. or Volume No. :</i>	
08	<i>Contract No. :</i>	
10	<i>Title and subtitle :</i>	Operational Reactor Physics Analysis Codes (ORPAC)
11	<i>Collation :</i>	34p., 2 tabs., ill.
13	<i>Project No. :</i>	
20	<i>Personal author(s) :</i>	Jainendra Kumar; K.P. Singh; Kanchhi Singh
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22	<i>Corporate author(s) :</i>	Bhabha Atomic Research Centre, Mumbai-400 085
23	<i>Originating unit :</i>	Research Reactor Services Division, BARC, Mumbai
24	<i>Sponsor(s) Name :</i>	Department of Atomic Energy
	<i>Type :</i>	Government

Contd...

30	<i>Date of submission :</i>	June 2007
31	<i>Publication/Issue date :</i>	July 2007
40	<i>Publisher/Distributor :</i>	Associate Director, Knowledge Management Group and Head, Scientific Information Resource Division, Bhabha Atomic Research Centre, Mumbai
42	<i>Form of distribution :</i>	Hard copy
50	<i>Language of text :</i>	English
51	<i>Language of summary :</i>	English, Hindi
52	<i>No. of references :</i>	4 refs.
53	<i>Gives data on :</i>	
60	<i>Abstract :</i>	For efficient, smooth and safe operation of a nuclear research reactor, many reactor physics evaluations are regularly required. As part of reactor core management the important activities are maintaining core reactivity status, core power distribution, xenon estimations, safety evaluation of in-pile irradiation samples and experimental assemblies and assessment of nuclear safety in fuel handling/storage. In-pile irradiation of samples requires a prior estimation of the reactivity load due to the sample, the heating rate and the activity developed in it during irradiation. For the safety of personnel handling irradiated samples the dose rate at the surface of shielded flask housing the irradiated sample should be less than 200 mR/Hr. Therefore, a proper shielding and radioactive cooling of the irradiated sample are required to meet the said requirement. Knowledge of xenon load variation with time (Startup-curve) helps in estimating 'Xenon override time'. Monitoring of power in individual fuel channels during reactor operation is essential to know any abnormal power distribution to avoid unsafe situations. Complexities in the estimation of above mentioned reactor parameters and their frequent requirement compel one to use computer codes to avoid possible human errors. For efficient and quick evaluation of parameters related to reactor operations such as xenon load, critical moderator height and nuclear heating and reactivity load of isotope samples/experimental assembly, a computer code ORPAC (Operational Reactor Physics Analysis Codes) has been developed. This code is being used for regular assessment of reactor physics parameters in Dhruva and Cirus. The code ORPAC written in Visual Basic 6.0 environment incorporates several important operational reactor physics aspects on a single platform with graphical user interfaces (GUI) to make it more user-friendly and presentable
70	<i>Keywords/Descriptors :</i>	XENON; NEUTRON FLUX; S CODES; O CODES; DOSE RATES; SPENT FUELS; CIRUS REACTOR; REACTOR OPERATION; REACTOR START-UP; SAFETY ANALYSIS; DHRUVA REACTOR
71	<i>INIS Subject Category :</i>	S21
99	<i>Supplementary elements :</i>	

## सारांश

नाभिकीय अनुसंधान रिएक्टर के कुशल, सुचारू एवं संरक्षित प्रचालन हेतु अनेक रिएक्टर भौतिकी मूल्यांकन किए जाने की नियमित आवश्यकता है। रिएक्टर क्रोड प्रबंधन के भाग के रूप में मुख्य गतिविधियां हैं, क्रोड अभिक्रियता स्थिति का अनुरक्षण, क्रोड विद्युत वितरण, जेनान पूर्वानुमान, इन-पाइल किरणन प्रतिदर्शों एवं प्रयोगात्मक समुच्चयों का संरक्षा मूल्यांकन एवं ईंधन हस्तन। भंडारण में नाभिकीय संरक्षा का मूल्यांकन। प्रतिदर्शों के इन-फाइल किरणन हेतु प्रतिदर्श उष्मा दर, एवं किरणन के दौरान उत्पन्न सक्रियता के कारण अभिक्रियता भार के पूर्व परिकलन की आवश्यकता होती है। किरणित प्रतिदर्शों के हस्तन करने वाले कार्मिकों की संरक्षा हेतु किरणित प्रतिदर्श युक्त शील्डेड फ्लास्क की सतह पर डोज रेट **200 mR/Hr** से कम होना चाहिये। अतः उक्त आवश्यकता को पूरा करने के लिये किरणित प्रतिदर्शों की सही शील्डिंग एवं विकिरण शीतलन होना चाहिये। समय के साथ जेनान भार में भिन्नता की जानकारी (स्टार्ट-अप कर्व) से “जेनान ओवरराइड टाइम” के परिकलन में सहायता होती है। रिएक्टर प्रचालन के दौरान प्रत्येक ईंधन वाहक में विद्युत का मानीटरन आवश्यक हैं ताकि असुरक्षित स्थिति से बचने के लिये किसी असामान्य विद्युत वितरण का पता चल सके।

ऊपर उल्लिखित रिएक्टर प्राचलों के परिकलन की जटिलता एवं उनकी लगातार आवश्यकता के कारण संभाव्य मानवीय त्रुटि से बचने के लिये कंप्यूटर कोडों का प्रयोग अनिवार्य हो जाता है। जेनान लोड, क्रांतिक मंदक की उच्चता एवं आइसोटोप प्रतिदर्शों/प्रयोगात्मक समुच्चयों के

नाभिकीय ऊष्मीकरण जैसे रिएक्टर प्रचालन से संबंधित प्राचलों के कुशल एवं शीघ्र मूल्यांकन के लिये एक कंप्यूटर कोड ORPAC (आपरेशनल रिएक्टर फिजिक्स एनालिसिस कोड्स) का विकास किया गया । ध्रुवा एवं सायरस में रिएक्टर भौतिकी प्राचलों के नियमित मूल्यांकन हेतु इस कोड का प्रयोग किया जा रहा है । विजुअल बेसिक 6.0 एन्वायरमेंट में लिखित यह कोड ORPAC में एक ही प्लेटफार्म में अनेक महत्वपूर्ण प्रचालन रिएक्टर भौतिकी पहलुओं को ग्रेफिकल यूजर इंटरफेस (GUI) के साथ सम्मिलित किया गया है ताकि यह प्रयोक्ता के लिये उपयोगी एवं प्रदर्शनीय हो ।

## Abstract

For efficient, smooth and safe operation of a nuclear research reactor, many reactor physics evaluations are regularly required. As part of reactor core management the important activities are maintaining core reactivity status, core power distribution, xenon estimations, safety evaluation of in-pile irradiation samples & experimental assemblies and assessment of nuclear safety in fuel handling/storage. In-pile irradiation of samples requires a prior estimation of the reactivity load due to the sample, the heating rate and the activity developed in it during irradiation. For the safety of personnel handling irradiated samples the dose rate at the surface of shielded flask housing the irradiated sample should be less than **200 mR/Hr**. Therefore, a proper shielding and radioactive cooling of the irradiated sample are required to meet the said requirement. Knowledge of xenon load variation with time (Startup-Up curve) helps in estimating “**Xenon override time**”. Monitoring of power in individual fuel channels during reactor operation is essential to know any abnormal power distribution to avoid unsafe situations.

Complexities in the estimation of above mentioned reactor parameters and their frequent requirement compel one to use computer codes to avoid possible human errors. For efficient and quick evaluation of parameters related to reactor operations such as xenon load, critical moderator height and nuclear heating and reactivity load of isotope samples/experimental assembly, a computer code ORPAC (Operational Reactor Physics Analysis Codes) has been developed. This code is being used for regular assessment of reactor physics parameters in Dhruva and Cirus. The code ORPAC written in Visual Basic 6.0 environment incorporates several important operational reactor physics aspects on a single platform with graphical user interfaces (GUI) to make it more user-friendly and presentable.

**Keywords:** XENON, ORPAC, SHARDA, OPF, STARTUP, DECAY HEAT, DOSE RATE, FLUX DEPRESSION FACTOR, CRITICAL HEIGHT, ISOTOPE SAMPLE

## Introduction

Research reactors have been centers of innovation and productivity for nuclear science and technology for several years. Their multi dimensional roles can be seen in the areas of nuclear fuel cycle programme, radio-isotope production, neutron beam research, nuclear medicine, material development and computer code validation. Once the research reactor plant is commissioned and put under regular operation, management of safety acquires the prominent role. Operational reactor physics essentially deals with systematized application of physics safety principles to ensure that the reactor is operated within design limits with adequate safety margins. As part of reactor core management the important activities are core reactivity status, core power distribution, xenon estimations, safety evaluation of in-pile irradiation samples / experimental assemblies and nuclear safety in fuel handling / storage.

Operational reactor physics plays an important role in efficient, smooth and safe operation of nuclear reactor. In research reactors like Dhruva and Cirus reactor physics calculations are required for activities related to reactor **operation** and in-pile **irradiations**. Xenon estimation in equilibrium and non-equilibrium state of the reactor, startup curve, output power fractions, shielding and cooling requirements for irradiated isotope samples are some of the reactor physics calculations required frequently. Complexity of the reactor physics estimations and their frequency, make it necessary to use computer codes which play very prominent role in simplifying the task and complete it in minimum time. In this regard a new software **ORPAC (Operational Reactor Physics Analysis Codes)** which is a system of many computer codes on a single platform, has been developed to address the issues related to reactor physics aspects of Dhruva and Cirus. The enhancement of computer base for reactor physics calculation is an ongoing process and in future also many such computer programmes will be incorporated under this developed system.

In Dhruva and Cirus computer codes have been used for reactor physics calculations pertaining to reactor operation. Codes like SHARDA, OPF have been used very frequently. These computer codes were developed long back and need for modification and extension in the existing codes was always felt. With the sustained development in computer technology, the possibility for enhancement of computer code capability always exists. The pre-existing computer codes were rewritten to make them more user friendly and wherever modification and extension were required, the same were incorporated.

The following sections give brief descriptions of various reactor physics estimations needed on regular basis and methodologies used to carry out such tasks with help of computer codes.

## Description of important reactor physics estimations

### 1. PIR (Pile Irradiation Request) evaluation

Sample irradiation for isotope production has been a major activity in research reactors like Dhruva and Cirus. Isotope production, is playing a crucial role in fulfilling the country's requirement for radio nuclides which are used as nuclear medicine and also as radioactive source for research activities. A sample when put inside a reactor for irradiation, perturbs the steady operation of the reactor due to parasitic absorption of neutrons in the sample. So, putting a sample inside a reactor is equivalent to adding a negative reactivity load and removing the sample from the reactor is equivalent to adding a positive reactivity to the reactor. A large perturbation has undesirable effects on smooth and safe operation of the reactor. Therefore, reactivity load on the reactor due to a sample is an important parameter which requires watchful eyes. Further, when the sample is inside the reactor it absorbs  $\alpha$ ,  $\beta$  and  $\gamma$ -radiations produced inside the reactor and in the sample itself, which cause heat generation in the sample. The rate of heat generation must be within the range allowed by the engineering design of the reactor.

A sample after irradiation becomes radioactive and emits  $\alpha$ ,  $\beta$  and  $\gamma$ -radiations. Without proper protection these radiations can cause tissue damage to the personnel handling and carrying the irradiated sample. Exposure to these radiations has been quantified in terms of radioactive **dose rates**. For the safety of the concerned personnel it is required that the dose rate at the surface of the sample should be less than **200 mR/Hr**. Therefore, a proper shielding and cooling of the irradiated sample are required to meet the above criterion.

### 2. Xenon evaluation and startup curve

There are many factors which can affect the criticality of the reactor. Some important ones are Xenon, temperature, power of the reactor, and other reactivity loads. These have direct bearings on the critical height of the reactor. For reactor startup the critical height should be known in advance. For estimation of critical height, Xenon load estimation in equilibrium and non-equilibrium state of reactor is an important requirement. Xenon load variation with time (startup curve) helps in estimating "**Xenon override time**". In addition to Xenon load, reactivity loads due to temperature, power etc. are also required to be known in advance before starting the reactor.

### 3. Output Power Fractions (OPF)

Contribution of each fuel channel to the total power produced in the reactor requires a high level of attention. Any abnormal trend of power production in even a single fuel channel can lead to serious problems. Moreover power distribution is used in monitoring the burn-up status of fuel assemblies and deciding their discharge time. Therefore, OPF (Output Power Fraction) chart representing power fraction of each individual channel is required regularly, especially after the substantial loading changes in the core. Though theoretical estimates of power distribution for a specific core configuration are always

available, the experimental power distribution in the fuel channel is estimated with the help of coolant flow and outlet temperatures of each individual channel.

#### 4. Core Reactivity Status and Critical Height

Dhruva and Cirus being research reactors, their core loading/configurations undergo frequent changes. Core configuration is changed due to replacement of fuel rods by some experimental assembly. As burn-up levels of discharge fuel assemblies are low, replacement of a burnt fuel assembly by a fresh one does not lead to significant change in reactivity status of the core. Occasionally the loading may be changed due to adoption of new fuel design. In case of changed core loading there may be extra reactivity load or reactivity gain for the reactor. Hence, from time to time reactivity status of reactor is maintained. Owing to change in core configuration, new critical height estimation is very often required.

#### 5. Decay Heat in Spent Fuel

Once a fuel assembly has completed its tenure in the reactor core (based on discharge burn-up, MCP hrs etc), it is taken out from the core and kept for cooling in SFSB / WSB. Spent fuel generates heat (decay heat) due to radioactive decay of fission products in it. Decay heat depends on reactor power, operating time and cooling time after its removal from the core and is required to be calculated for safe handling of spent fuel.

In sum OPF chart, startup curve, critical height and decay heat of discharged fuel rod are required for smooth and safe operation of the reactor. Reactivity load, heating rate, activity, dose rate, cooling time and shielding requirements are important parameters to be evaluated and provided to Pile Irradiation Requests (PIR) form for its safe loading in reactor and safe transportation to users. To make these calculational task handy and error proof, computer codes named 'SHARDA+', 'STARTUP' and 'OPF' etc. in FORTRAN-77 were developed. The newly developed code (ORPAC) is a computer code system for comprehensive simulation of reactor physics aspects of research reactor Dhruva. It has all the features of pre-existing codes and many extensions and modifications have also been included. ORPAC encompasses many computer codes like SHRADA, OPF, STARTUP, etc. We are extending the scope of ORPAC code for Cirus reactor also.

The present code ORPAC written in Visual Basic 6.0 environment incorporates all the above mentioned codes on a single platform with graphical user interfaces (GUI) to make it more user-friendly and presentable along with printing options. Some more features like '**multiple capsules in a single flask**' case and irradiation in **Pneumatic Carrier Facility** have been added. For **critical height** and **decay heat** estimation, computer programs have also been added to the current code. For a given moderator height the extra reactivity load/gain available to the reactor can also be estimated using same program. Any concerned personnel can run this code with small training. There is a plan to enhance the scope of this code in future. Following gives a description of reactor physics theory and related system of computer codes.

## Method of calculation

### 1. PIR (Pile Irradiation Request) evaluation code SHARDA+

#### 1.1. Flux Depression Factor

For estimation of activity and heat produced in a sample kept for irradiation, it is necessary to know the flux in the sample. For this flux depression factor is an important parameter to be calculated. It decides the extent up to which neutron flux has been changed in the sample due to absorption. It is defined as the ratio of average flux in the sample to the flux at surface of the sample. i.e.

$$f = \frac{\bar{\phi}}{\phi_0} \quad (1.1.1)$$

Where,  $f$  = flux depression factor  
 $\bar{\phi}$  = average flux in the sample,  
 $\phi_0$  = flux at outer surface of the sample

In integral form  $f$  is defined as

$$f = \frac{\int_V \phi(r) dr}{\int_V \phi_0 dr} \quad (1.1.2)$$

Where,  $\phi(r)$  = neutron flux at position  $r$ .

Flux depression factor ( $f$ ) depends upon geometry of the sample. It is approximately determined by solving the Neutron Diffusion Equation for a given geometry with appropriate boundary conditions. One group diffusion equation in non fissile absorbing medium is

$$D\nabla^2\phi - \Sigma_A\phi = 0 \quad (1.1.3)$$

Where,  $D$  = diffusion coefficient,  
 $\Sigma_A$  = macroscopic absorption cross section,  
 $\nabla^2$  = Laplacian operator for given geometry,  
and,  $\phi$  = neutron flux.

Equation (1.1.3) can be written as:

$$\nabla^2\phi - k^2\phi = 0 \quad (1.1.4)$$

Where,  $k = \sqrt{\frac{\Sigma_A}{D}}$

$\phi(r)$  the solution of equation (1.1.4) for a given geometry, when substituted in (1.1.2) and integrated gives the expression for flux depression factor for that particular geometry of the sample.

Two types of geometries have been considered here

### 1.1.1. Spherical Geometry

The neutron diffusion equation for spherical geometry is

$$\frac{d^2\phi}{dr^2} + \frac{2}{r} \frac{d\phi}{dr} - k^2\phi = 0 \quad (1.1.5)$$

Solution of equation (1.1.5) is given by

$$\phi(r) = \frac{A e^{kr} + B e^{-kr}}{r} \quad (1.1.6)$$

Where A and B are arbitrary constants which are determined using the following boundary conditions:

$$(1) \text{ At } r=0 ; \quad r^2 \frac{d\phi}{dr} = 0$$

$$\text{and } (2) \text{ At } r=R ; \quad \phi = \phi_0 \quad (1.1.7)$$

Constants A and B are determined as

$$A = -B = \frac{R\phi_0}{e^{kR} - e^{-kR}} \quad (1.1.8)$$

Substituting the expression for  $\phi(r)$  in (1.1.2) and after completing the integration the expression for flux depression factor ( $f$ ) for spherical geometry is obtained as

$$\boxed{f = \frac{3}{k^2 R^2} [kR \coth(kR) - 1]} \quad (1.1.9)$$

### 1.1.2. Cylindrical Geometry

For cylindrical geometry neutron diffusion equation is given by

$$\frac{d^2\phi}{dr^2} + \frac{1}{r} \frac{d\phi}{dr} - k^2\phi = 0 \quad (1.1.10)$$

Solution of equation (1.1.10) can be written as

$$\phi(r) = AI_0(kr) + BK_0(kr) \quad (1.1.11)$$

Where, A and B are arbitrary constants determined by boundary conditions and,  $I_0(kr)$  and  $K_0(kr)$  are Spherical Bessel functions of zeroth order. The function  $K_0(kr)$  diverges at  $r=0$ . Therefore,  $B=0$ .

Applying boundary conditions at  $r=R$ ;  $\phi(R) = \phi_0$

$$\text{i.e.} \quad \phi_0 = AI_0(kR)$$

Therefore,

$$A = \frac{\phi_0}{I_0(kR)}$$

$$\text{or,} \quad \phi(r) = \frac{\phi_0 I_0(kr)}{I_0(kR)} \quad (1.1.12)$$

Substituting  $\phi(r)$  from (1.1.12) in (1.1.2) we get

$$f = \frac{2 \int_0^R r I_0(kr) dr}{R^2 I_0(kR)} \quad (1.1.13)$$

Solving the integral in (1.1.13) the flux depression factor ( $f$ ) for a cylindrical sample is obtained as

$$\boxed{f = \frac{2I_1(kR)}{kRI_0(kR)}} \quad (1.1.14)$$

## 1.2. Heating Rate in the Sample

A sample inside a reactor absorbs radiations produced inside the reactor and in the sample itself. Radiations like alpha, beta, gamma and neutrons when absorbed by nuclei present in the sample, it undergoes nuclear transitions. Many complex nuclear reactions are involved in the process. The excess energy of the excited nuclei supplied by radiations absorbed is lost in collision and scattering and finally converted to heat energy. This causes heating of the sample. The heat developed in the sample due to alphas and betas has been ignored in the analysis.

### 1.2.1. Capture gamma heating

Energy deposition due to the process of gamma capture is expressed in terms of Build-Up factor, average energy of photons, number of photons, energy deposition coefficient and linear attenuation coefficient. Build-Up factor is function of mean free path and the energy of the photons. For gamma photons having energy  $> 2$  MeV and for materials of intermediate atomic mass number an approximate form for buildup factor is given by a linear relationship formulae

$$B.F = 1 + a_1 \mu R \quad (1.2.1)$$

For a sample having spherical geometry of radius R the average rate of energy deposition per cc considering Build-Up factor of form given by (15) is obtained as [ref.(4)]

$$\bar{H} = \frac{C_0 E_0 S v \mu_A (\psi_0 + a_1 \psi_1)}{\mu} \quad (1.2.2)$$

Where,

$C_0 = 1.6 \times 10^{-13}$  (WS/MeV) a conversion factor

$a_1 = 0.65$

$E_0$  = average energy of photons (MeV)

$Sv$  = no. of gamma photons / cc. of energy  $E_0$

$\mu_A$  = energy deposition coefficient for energy  $E_0$  ( $\text{cm}^{-1}$ )  
 $\mu$  = total linear attenuation coefficient ( $\text{cm}^{-1}$ )

For spherical sample of radius  $R$ ,  $\psi_0$  and  $\psi_1$  are given as

$$\psi_0 = 1 - \frac{6\mu^2 R^2 - 3 + (3 + 6\mu R)e^{-2\mu R}}{8\mu^3 R^3}$$

$$\psi_1 = 1 - \frac{3\mu^2 R^2 - 3 + (3 + 6\mu R + 3\mu^2 R^2)e^{-2\mu R}}{2\mu^3 R^3}$$

For cylindrical sample of radius  $R$ ,  $\psi_0$  and  $\psi_1$  are given as

$$\psi_0 = 1 - \frac{2}{3}\mu R \left[ 2\mu R K_1(\mu R) I_1(\mu R) + K_0(\mu R) I_0(\mu R) - 2 + \frac{1}{\mu R} K_1(\mu R) I_1(\mu R) - K_0(\mu R) I_1(\mu R) + K_1(\mu R) I_0(\mu R) \right]$$

$$\psi_1 = 1 - 2I_1(\mu R) K_1(\mu R)$$

Cylindrical shell for practical purpose, can be approximated to be a solid cylinder with same cross sectional area with an effective radius  $R$  given as

$$R = \sqrt{R_{\text{out}}^2 - R_{\text{in}}^2}$$

For a sample with mass  $M$  and density  $\rho$  the total heating rate is

$$H = C_0 E_0 S v (\psi_0 + a_1 \psi_1) \frac{\mu_A}{\mu} \frac{M}{\rho} \quad (1.2.3)$$

To simplify calculations and to minimize the requirements for huge number of data it is assumed that for every neutron captured in the sample a single 8-MeV gamma photon is emitted. The values of parameters which are characteristics of materials in the sample has been taken as representative of medium atomic weight elements. So that errors involved in the estimations for lighter and heavier nuclei do not become too high to ignore. These assumed values has been taken to be

$$E_0 S v = 8 \Sigma_a \phi f \text{ MeV/cm}^3$$

$$\frac{\mu_A}{\rho} = 0.025 \text{ cm}^2/\text{gm}$$

and,  $\frac{\mu}{\rho} = 0.042 \text{ cm}^2/\text{gm}$

These values although a rough approximation give a reasonably good estimation for wide range of materials.

### 1.2.2. Core gamma heating

Heating in structural materials due to absorption of gamma radiations produced inside the core of Dhruva reactor has been found to be 400 mw/gm at thermal neutron flux  $\Phi = 1.5E+14 \text{ n/cm}^2/\text{sec}$ . For Cirus it is 160 mw/gm at  $\Phi = 6.5E+13 \text{ n/cm}^2/\text{sec}$ .

### 1.3. Reactivity Load

Reactivity load due to a sample is determined by solving neutron diffusion equation. For introduction of small reactivity change, perturbation theory is best suitable to solve the neutron diffusion equation. Neutron diffusion equation for steady state in operator form can be written as

$$H \phi = \lambda_0 F \phi \quad (1.3.1)$$

Where,  $H = \Sigma_A - D \nabla^2$

$$F = \nu \Sigma_f$$

And,  $\lambda_0 = \frac{1}{K_0}$

Adjoint equation of (1.3.1) is

$$H^+ \phi^+ = \lambda_0 F^+ \phi^+ \quad (1.3.2)$$

For small perturbation (1.3.1) can be written as

$$(H + \delta H) \phi_1 = (\lambda_0 + \delta \lambda) (F + \delta F) \phi_1 \quad (1.3.3)$$

Where,  $\phi_1$  is the perturbed flux.

Equation (1.3.3) along with its adjoint equation can be rearranged after neglecting second order terms as

$$\frac{\delta K}{K} = \frac{-\delta \lambda}{\lambda_0} = - \frac{\langle \phi^+, (\delta H - \lambda_0 \delta F) \phi_1 \rangle}{\lambda_0 \langle \phi^+, (F + \delta F) \phi_1 \rangle} \quad (1.3.4)$$

Here only non fissile material is considered for which  $\delta F=0$ . Hence equation (1.3.4) is reduced as

$$\frac{\delta K}{K} = - \frac{\langle \phi^+, \delta H \phi_1 \rangle}{\lambda_0 \langle \phi^+, F \phi_1 \rangle} \quad (1.3.5)$$

In one group model the adjoint flux is same as real neutron flux (i.e.  $\phi^+ = \phi$ )

Writing the above equation in integral form

$$\frac{\delta K}{K} = \frac{-\int_{V_s} \phi \delta \Sigma_A \phi_1 dV - \int_{V_s} \Delta \phi \delta D \Delta \phi dV}{\lambda_0 \int_{V_R} \phi \nu \Sigma_f \phi_1 dV} \quad (1.3.6)$$

Where,  $V_s$  = Volume of the sample,

and,  $V_R$  = Volume of the reactor

The range of integration for numerator has been taken to be within the sample volume because there is no perturbation in other part of the reactor. Further the second term of

the numerator contains second order term ( $\Delta\phi\Delta\phi$ ) which can be neglected compared to the first term.

$$\frac{\delta K}{K} = - \frac{K_0 \int_{V_s} \phi \delta \Sigma_A \phi_1 dV}{\int_{V_R} \phi v \Sigma_f \phi_1 dV} \quad (1.3.7)$$

Since, there is no perturbation in other part of the reactor  $\phi_1 = \phi$ . The flux in the sample can be replaced by average flux i.e.  $\phi_1 = f\phi$ . Therefore equation (1.3.7) can be rewritten as

$$\frac{\delta K}{K} = - \frac{K_0 f \delta \Sigma_A \int_{V_s} \phi^2 dV}{\int_{V_R} \phi^2 v \Sigma_f dV} \quad (1.3.8)$$

If at the sample location  $\phi_h$  is the flux then integral in the numerator can be replaced by  $\phi_h^2 V_s$ . Therefore

$$\frac{\delta K}{K} = - \frac{K_0 f \delta \Sigma_A \phi_h^2 V_s}{\int_{V_R} \phi^2 v \Sigma_f dV} \quad (1.3.9)$$

For a white absorber i.e.  $f=1.0$  having  $\delta \Sigma_A V_s = 1.0 \text{ cm}^2$ , values of  $\frac{\delta K}{K}$  at maximum flux  $\phi_m$  have been found to be  $0.0258 \times 10^{-3}$  and  $0.0299 \times 10^{-3}$  for Dhruva and Cirus respectively. Substituting these values in equation (1.3.9) the final expression for reactivity load in case of Dhruva is given as

$$\boxed{\frac{\delta K}{K} = -0.0258 f \delta \Sigma_A V_s \frac{\phi_h^2}{\phi_m^2} \times 10^{-3}} \quad (1.3.10)$$

and for Cirus it is given as

$$\boxed{\frac{\delta K}{K} = -0.0299 f \delta \Sigma_A V_s \frac{\phi_h^2}{\phi_m^2} \times 10^{-3}} \quad (1.3.11)$$

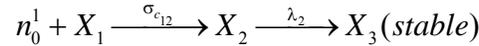
#### 1.4. Activity

A sample under irradiation absorbs neutrons and a series of nuclear reactions start and new isotopes are produced in the process. These newly generated isotopes are usually radioactive. The rate of production of these isotopes is proportional to the neutron flux and the capture cross section of the parent nuclide. As soon as they are generated they also start decaying to their daughter nuclide. Rate of decay of an isotope is proportional to its decay constant, its population density. There are many nuclear reactions possible. A daughter nuclide produced due to the decay may also be radioactive and may give rise to

a chain of decay reactions till stable isotope is formed. An isotope may absorb a neutron and give rise to a new isotope along with an alpha particle.

#### 1.4.1. First order case

In this case the daughter nuclide of radioactive neutron capture product is not radioactive. i.e.



If  $N_1$  and  $N_2$  are number density of  $X_1$  and  $X_2$  equations governing first order case are:

$$\frac{dN_1}{dt} = -\sigma_{A_1} \phi N_1 \quad (1.4.1)$$

$$\frac{dN_2}{dt} = \sigma_{c12} \phi N_1 - (\lambda_2 + \sigma_{A_2} \phi) N_2 \quad (1.4.2)$$

Here,  $\sigma_{A_1}, \sigma_{A_2}$  are absorption cross section of nuclides  $X_1$  and  $X_2$  respectively and  $\sigma_{c12}$  is activation cross section of nuclide  $X_1$  to form  $X_2$ .

Solution of (1.4.1) with initial boundary conditions  $N_1 = N_1^0$  and  $N_2 = 0$  at  $t = 0$ ;

$$N_1 = N_1^0 e^{-\sigma_{A_1} \phi t} \quad (1.4.3)$$

Substituting (1.4.3) in (1.4.2) and when solved

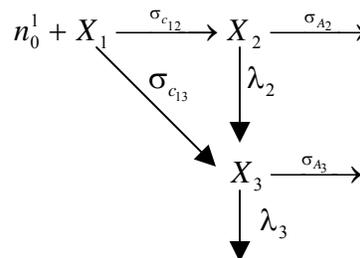
$$N_2 = N_1^0 \sigma_{c12} \phi \left[ \frac{e^{-\sigma_{A_1} \phi t} - e^{-(\lambda_2 + \sigma_{A_2} \phi) t}}{\lambda_2 + (\sigma_{A_2} - \sigma_{A_1}) \phi} \right] \quad (1.4.4)$$

Activity due to the daughter nuclei is the product of its number density and its decay constant. i.e.

$$A_2 = \lambda_2 N_2 \quad (1.4.5)$$

#### 1.4.2. Second order case:

In this case isotope  $X_2$  may decay to another radioactive isotope  $X_3$  or may be another isomeric state of the same isotope  $X_2$ .  $X_3$  may also be produced from  $X_1$  due to neutron capture.



The kinetics of reaction is governed by three differential equations

$$\frac{dN_1}{dt} = -\sigma_{A_1} \phi N_1 \quad (1.4.6)$$

$$\frac{dN_2}{dt} = \sigma_{C_{12}} \phi N_1 - (\lambda_2 + \sigma_{A_2} \phi) N_2 \quad (1.4.7)$$

$$\frac{dN_3}{dt} = \sigma_{C_{13}} \phi N_1 + \lambda_2 N_2 - (\lambda_3 + \sigma_{A_3} \phi) N_3 \quad (1.4.8)$$

Using initial boundary conditions  $N_1 = N_1^0$ ,  $N_2 = 0$  and  $N_3 = 0$  at  $t = 0$ , solutions of above equations are-

$$N_1 = N_1^0 e^{-\sigma_{A_1} \phi t} \quad (1.4.9)$$

$$N_2 = N_1^0 \sigma_{C_{12}} \phi \left[ \frac{e^{-\sigma_{A_1} \phi t} - e^{-(\lambda_2 + \sigma_{A_2} \phi) t}}{\lambda_2 + (\sigma_{A_2} - \sigma_{A_1}) \phi} \right] \quad (1.4.10)$$

$$N_3 = B \left[ \frac{e^{-\sigma_{A_1} \phi t} - e^{-(\lambda_3 + \sigma_{A_3} \phi) t}}{\lambda_3 + (\sigma_{A_3} - \sigma_{A_1}) \phi} \right] + D \left[ \frac{e^{-(\lambda_2 + \sigma_{A_2} \phi) t} - e^{-(\lambda_3 + \sigma_{A_3} \phi) t}}{\lambda_3 + \lambda_2 + (\sigma_{A_3} - \sigma_{A_2}) \phi} \right] \quad (1.4.11)$$

Here,  $\sigma_{A_1}$ ,  $\sigma_{A_2}$ ,  $\sigma_{A_3}$  are absorption cross section of nuclides  $X_1$ ,  $X_2$  and  $X_3$  respectively.  $\sigma_{C_{12}}$  and  $\sigma_{C_{13}}$  are activation cross section of nuclide  $X_1$  to form nuclide  $X_2$  and  $X_3$  respectively. B and D are constants given below.

$$B = \phi N_1^0 \left[ \sigma_{C_{13}} + \frac{\lambda_2 \sigma_{C_{12}}}{\lambda_2 + (\sigma_{A_2} - \sigma_{A_1}) \phi} \right] \quad (1.4.12)$$

And

$$D = \frac{\phi N_1^0 \lambda_2 \sigma_{C_{12}}}{\lambda_2 + (\sigma_{A_2} - \sigma_{A_1}) \phi} \quad (1.4.13)$$

Total activity immediately after irradiation:  $A = \lambda_2 N_2 + \lambda_3 N_3$

And total activity after cooling time  $t_c$ :

$$A(t_c) = \lambda_2 N_2 e^{-\lambda_2 t_c} + \lambda_3 N_3 e^{-\lambda_3 t_c} \quad (1.4.14)$$

## 1.5. Dose Rate

An irradiated sample contains radioactive isotopes which emit  $\alpha$ ,  $\beta$  and  $\gamma$  radiations. Alpha and beta radiations do not require special shielding. The shielding needed to stop  $\gamma$  radiations is also adequate for these ( $\alpha$ ,  $\beta$ ) radiations. Bare dose rate  $D_0$  (mR/hr) at a distance of 1 foot from the irradiated sample is given as,

$$D_0 = 6C \sum_{\nu} E_{\nu} f_{\nu} \quad (1.5.1)$$

Where,  $C$  = activity in Curie,

$E_v$  = photon energy of gamma ray (MeV),

$f_v$  = fractional yield of the gamma ray having photon energy  $E_v$

The bare dose rate may be too high to permit handling the irradiated sample. To reduce the dose rate to acceptable limit, a suitable shielding material is used. The shielded dose rate  $D$  (mR/hr) at a distance of 1 foot is given as,

$$D = 6 C \sum_v E_v f_v e^{-\mu x} B(E_v, \mu x) \quad (1.5.2)$$

Where,  $\mu$  = attenuation coefficient for gamma radiation

$x$  = shield thickness

And  $B(E_v, \mu x)$  = Build-up factor

Gamma- Ray Exposure Buildup factors  $B(E_v, \mu x)$  for a point isotropic source in case of lead as shielding material are given in Table-1

**Table-1**

Energy (MeV)	$\mu x$						
	1.0	2.0	4.0	7.0	10	15	20
0.5	1.24	1.42	1.69	2.00	2.27	2.67	2.73
1.0	1.37	1.69	2.26	3.02	3.74	4.81	5.86
2.0	1.39	1.76	2.51	3.66	4.84	6.87	9.00
3.0	1.34	1.68	2.43	3.75	5.3	8.44	12.3
4.0	1.27	1.56	2.25	3.61	5.44	9.80	16.3
6.0	1.18	1.40	1.97	3.34	5.69	13.8	32.7
10.0	1.11	1.23	1.58	2.52	4.34	12.5	39.2

### 1.5.1. Dose rate due to capsule activity

In the process of irradiation of samples the capsule which is made of Aluminium also gets irradiated and some activity is developed in the capsule. Irradiation of Aluminium ( $Al^{27}$ ) with neutron flux generates  $Na^{24}$  by  $(n, \alpha)$  reaction\*.  $Na^{24}$  being a hard gamma emitter it also contributes to the total dose rate. The bare Dose rate at one foot due to capsule activity is given by the equation

$$D = D_0 \frac{\phi}{\phi_0} (1 - e^{-\lambda t_i}) e^{-\lambda t_c} \quad (1.5.3)$$

Where,  $\lambda$  = decay constant of  $Na^{24}$ ,

$\phi$  = neutron flux at which sample is being irradiated,

$t_i$  = irradiation time,

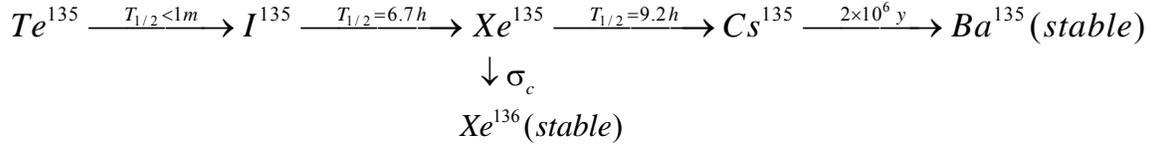
$t_c$  = cooling time and

$D_0$  = saturation dose rate at 1 foot due to irradiation at flux  $\phi_0$ .

\*In its irradiation,  $Al^{28}$  is also produced. As half life of  $Al^{28}$  is only 2.8 min., for all practical purposes its contribution to dose rate may be neglected. An irradiated sample sees more than 30 min. of radioactive cooling before being loaded into the lead flask. Thus, the activity from  $Al^{28}$  at the time of loading in the flask will be negligible.

## 2. Xenon estimation and Startup computer code

In nuclear reactors estimation of Xenon poisoning is very important.  $Xe^{135}$  is produced by  $\beta$ - decay of  $I^{135}$  which is again a product of  $\beta$ - decay of a fission product  $Te^{135}$ . Reactions leading to Xenon production and decay are as follow.



For a reactor in operation, the concentrations of  $I^{135}$  and  $Xe^{135}$  are governed by two differential equations. Here symbols have their usual meanings

$$\frac{dI}{dt} = -\lambda_I I - \sigma_I \phi I + \gamma_I \Sigma_f \phi \quad (2.1)$$

$$\frac{dX}{dt} = -\lambda_X X - \sigma_X \phi X + \lambda_I I + \gamma_X \Sigma_f \phi \quad (2.2)$$

In equation (2.1), Iodine has been considered as a direct fission product. After some time when equilibrium has reached and the rate of loss and rate of production become equal, then  $\frac{dI}{dt} = 0$ . At equilibrium, if  $I_0$  is the iodine concentration then

$$0 = -\lambda_I I_0 - \sigma_I \phi I_0 + \gamma_I \Sigma_f \phi \quad (2.3)$$

Rearranging (2.3), the iodine concentration at equilibrium is obtained as

$$I_0 = \frac{\gamma_I \Sigma_f \phi}{\lambda_I + \sigma_I \phi_0} \cong \frac{\gamma_I \Sigma_f \phi}{\lambda_I} ; \quad (\sigma_I \phi_0 \ll \lambda_I) \quad (2.4)$$

Similarly, Xenon concentration at equilibrium

$$X_0 = \frac{\lambda_I I_0 + \gamma_X \Sigma_f \phi}{\lambda_X + \sigma_X \phi} \quad (2.5)$$

Now, reactivity load due to Xenon poisoning at equilibrium

$$\Delta \rho_0 \sim -\frac{X_0 \sigma_X}{\Sigma_a} = -\frac{(\lambda_I I_0 + \gamma_X \Sigma_f \phi) \sigma_X}{(\lambda_X + \sigma_X \phi) \Sigma_a} \quad (2.6)$$

Where, the values of parameters are

$$\gamma_I = 0.061 \text{ ( fission yield of } I^{135} \text{ ) ; } \quad \gamma_X = 0.003 \text{ (fission yield of } Xe^{135} \text{)}$$

$$\lambda_I = 0.1035 hr^{-1} \text{ (decay constant of } I^{135} \text{ ) ; } \quad \lambda_X = 0.0753 hr^{-1} \text{ (decay constant of } Xe^{135} \text{)}$$

$$\sigma_X = 3.0 \times 10^6 b \text{ (absorption cross section of } Xe^{135} \text{)}$$

After shutdown,  $\phi = 0$ , production of  $I^{135}$  and loss of  $Xe^{135}$  due to neutron capture stop but radioactive decays of Iodine and Xenon continue. Since  $\lambda_I > \lambda_X$ , Xenon concentration will increase after S/D due to faster decay of Iodine. After certain time Xenon concentration will reach to a maximum value and then it will keep on decreasing with time.  $I^{135}$  and  $X^{135}$  concentrations after reactor S/D are governed by following equations:

$$\frac{dI}{dt} = -\lambda_1 I \quad (2.7)$$

$$\frac{dX}{dt} = -\lambda_X X + \lambda_1 I \quad (2.8)$$

Solutions of (2.7) and (2.8) along with boundary condition give the variation of Iodine and Xenon concentration with time after shut-down.

$$I(t_s) = I_0 e^{-\lambda_1 t_s} \quad (2.9)$$

And,

$$X(t_s) = \frac{\lambda_1}{\lambda_X - \lambda_1} I_0 (e^{-\lambda_1 t_s} - e^{-\lambda_X t_s}) + X_0 e^{-\lambda_X t_s} \quad (2.10)$$

Where,  $t_s$  is time after shut-down.

The  $Xe^{135}$  concentration becomes maximum at time  $t_{\max}$  after S/D.

$$t_{\max} = \frac{\lambda_1}{\lambda_X - \lambda_1} \ln \frac{\lambda_X}{\lambda_1} \left( 1 - \frac{\lambda_X - \lambda_1}{\lambda_1} \cdot \frac{X_0}{I_0} \right) \quad (2.11)$$

Reactivity load due to  $Xe^{135}$  at time  $t_s$  after S/D

$$\Delta\rho(t_s) \sim -\frac{X(t_s)\sigma_X}{\Sigma_a} \quad (2.12)$$

$$\Delta\rho(t_s) \sim -\frac{\sigma_X}{\Sigma_a} \cdot \left[ \frac{\lambda_1}{\lambda_X - \lambda_1} I_0 (e^{-\lambda_1 t_s} - e^{-\lambda_X t_s}) + X_0 e^{-\lambda_X t_s} \right] \quad (2.13)$$

For Dhruva Xenon Load variation with time following changes in reactor power from  $P_0$  (MW) to  $P$  (MW) is given as,

$$\Delta\rho(t_s) = \frac{a.P}{\lambda_X + A.P} (1 - e^{-(\lambda_X + A.P)t_s}) + \frac{\lambda_1 \rho_I^0 - a.P}{\lambda_X - \lambda_1 + A.P} (e^{-\lambda_1 t_s} - e^{-(\lambda_X + A.P)t_s}) + \rho_X^0 \cdot e^{-(\lambda_X + A.P)t_s} \quad (2.14)$$

Where,  $\rho_X^0$  = equilibrium  $Xe^{135}$  load (mk) at  $P_0$  (MW),

$\rho_I^0$  = equilibrium equivalent  $I^{135}$  load (mk) at  $P_0$  (MW)

Coefficients  $a$  and  $A$  depend upon core loading e.g number of fuel rod assemblies etc.

$$\text{i.e. } a = \frac{a_0}{NH} \quad \text{and} \quad A = \frac{A_0}{NH}$$

Where,  $N$  = number of fuel rods,

$H$  = extrapolated equilibrium moderator height (m),

$a_0$  and  $A_0$  are reactor specific constants. Their values for Dhruva and Cirus reactors are given in Table-2 .

**Table-2**

	Dhruva	Cirus
$a_0$	75.875	92.0
$A_0$	2.42	3.92

With above values of  $a_0$  and  $A_0$ ,  $\Delta\rho(t_s)$  is in units of mk.

In addition to Xenon Load there are additional loads due to power and temperature. Therefore total reactivity load is given as,

$$\Delta\rho_{total}(t_s) = \Delta\rho(t_s) + PowerCoef \times P_0 + TempCoef \times T_0 \quad (2.15)$$

For Dhruva reactor,  $PowerCoef = -0.042mk / MW$

And,  $TempCoef = -0.198mk / ^\circ C$

For Dhruva reactor, the critical height  $H(t_s)$  corresponding to the load  $\Delta\rho_{total}(t_s)$  at time  $t_s$  after S/D is

$$H(t_s) = \sqrt{\frac{33.05\pi^2 H_{ref}^2}{33.05\pi^2 - \Delta\rho_{total}(t_s) H_{ref}^2}} \quad (2.16)$$

For Dhruva,  $H_{ref} = 1.812$  m.

### 3. Output Power Fraction

Power production in a fuel channel is estimated by simple calorimetric formula. The power of ith channel is

$$P(i) = m_i c \Delta T_i \quad (3.1)$$

Where,  $\Delta T_i$  = temperature rise across the ith fuel channel,

$c$  = specific heat of the coolant,

and,  $m_i$  = coolant mass transfer rate in the ith channel

Putting  $\Delta T_i = T_{out}(i) - T_{in}$  and  $m_i = \rho \cdot flow(i)$  in (3.1)

$$P(i) = \rho c \cdot [T_{out}(i) - T_{in}] \cdot flow(i) \quad (3.2)$$

$T_{out}(i)$  = channel outlet temperature ( $^\circ C$ )

$T_{in}$  = inlet plenum temperature ( $^\circ C$ )

$\rho$  = density of the coolant

$flow(i)$  = coolant flow rate (lpm) in the ith channel

Substituting the value of  $\rho$  and  $c$  for heavy water in (3.2), one gets power (kW) as

$$P(i) = 0.07722 [T_{out}(i) - T_{in}] \cdot flow(i) \quad (3.3)$$

Output Power Fraction of a fuel channel is defined as the ratio of the power produced in the channel to the total power through all channels.

$$OPF(i) = \frac{P(i)}{P_{total}}$$

or,

$$OPF(i) = \frac{P(i)}{\sum_i P(i)} \quad (3.4)$$

#### 4. Critical Height estimation

If core loading has changed and  $\Delta\rho$  is the reactivity load/gain in changed configuration then, the critical height for Dhruva reactor will be:

$$H_c = \sqrt{\frac{33.05\pi^2 H_{ref}^2}{33.05\pi^2 - \Delta\rho H_{ref}^2}} \quad (4.1)$$

Where,  $H_c$  = critical height,  
 $H_{ref}$  = critical height of a reference core,  
 and,  $\Delta\rho$  = reactivity load/gain

#### 5. Reactivity Load

Reactivity load for a given moderator height in Dhruva reactor is governed by the equation.

$$\Delta\rho(mk) = 33.05 \left( \frac{\pi^2}{H_{ref}^2} - \frac{\pi^2}{H^2} \right) \quad (5.1)$$

Where,  $H$  = Moderator height in meter,  
 and,  $H_{ref}$  = Critical moderator height in meter for a reference core,

#### 6. Decay Heat

A fuel rod when discharged from the reactor generates heat due to the decay of the fission products. Decay heat following the discharge from the reactor is given as,

$$P = 0.005.P_0.a \left[ T_c^{-b} - (T_c + T_{op})^{-b} \right] \quad (6.1)$$

Where,  $P$  = decay heat,  
 $P_0$  = operating Power,  
 $T_{op}$  = operating time (days),  
 $T_c$  = cooling time (days),  
 $a = 27.43$   
 and,  $b = 0.2962$   
 $a$  and  $b$  are parameters.

## ORPAC Code Description

In present form the code ORPAC includes four parts.

### A. SHARDA+

The first part **SHARDA+** performs evaluations required in PIR (Pile Irradiation Request). For a sample (non fissile) this part calculates heat rate, reactivity load, activity, dose rate, cooling time and shielding requirements. This part of the code includes three form modules named as 'frmSharda.frm', 'Result.frm' and 'Detail.frm' to provide graphical user interface(GUI) and one code-module named as 'modSharda.bas'. The form named 'frmSharda.frm' is the main interface. Inputs regarding sample e.g. composition, mass, density, geometry and location in tray rod etc are to be provided in this form. When invoked to run, the output results viz. dose rate, cooling time, shielding, heating rate and reactivity load are displayed on this form. Variation of dose rate with cooling time for a given shielding and also variation of dose rate with shielding for a given cooling time according to the users' choice is displayed in graph on this form. The form 'Result.frm' provides a summary of results regarding isotopes produced in the sample after irradiation, their half lives and their contribution to the total dose rate. Variation of dose Rate with cooling time and shielding can also be seen on this form. In 'Detail.frm' there are detailed informations about each individual isotopes regarding their activities, gamma rays energy, half lives and their dose rate contribution at every cooling time and shielding. The code module 'modSharda.bas' contains subroutines used to calculate activity, dose rates and shielding requirements etc.

### B. OPF

The **OPF** code calculates the fractional power output of each individual fuel channel and print the output in a chart form. This part includes three form modules named as 'frmOPF.frm', 'frmRodInput.frm', 'frmShowOPF.frm'. 'frmOPF.frm' has input boxes corresponding to reactor power, inlet temperature, outlet temperature and coolant flow for each loop, shield and vault. 'frmRodInput.frm' is designed to input channel outlet temperature and channel flow rates. 'frmShowOPF.frm' when called shows the OPF.

### C. STARTUP

In this part xenon load and its variation with time after S/D and corresponding critical moderator height are estimated. The **STARTUP** part includes two form module 'frmStartup.frm' and 'frmSdetail.frm' and a code-module 'modStartup.bas'. 'frmStartup.frm' is the main interface and all relevant input parameters are required to provide in this form and when invoked it display the result in graphical form (startup curve). 'frmSdetail.frm' provides detailed data regarding startup curve.

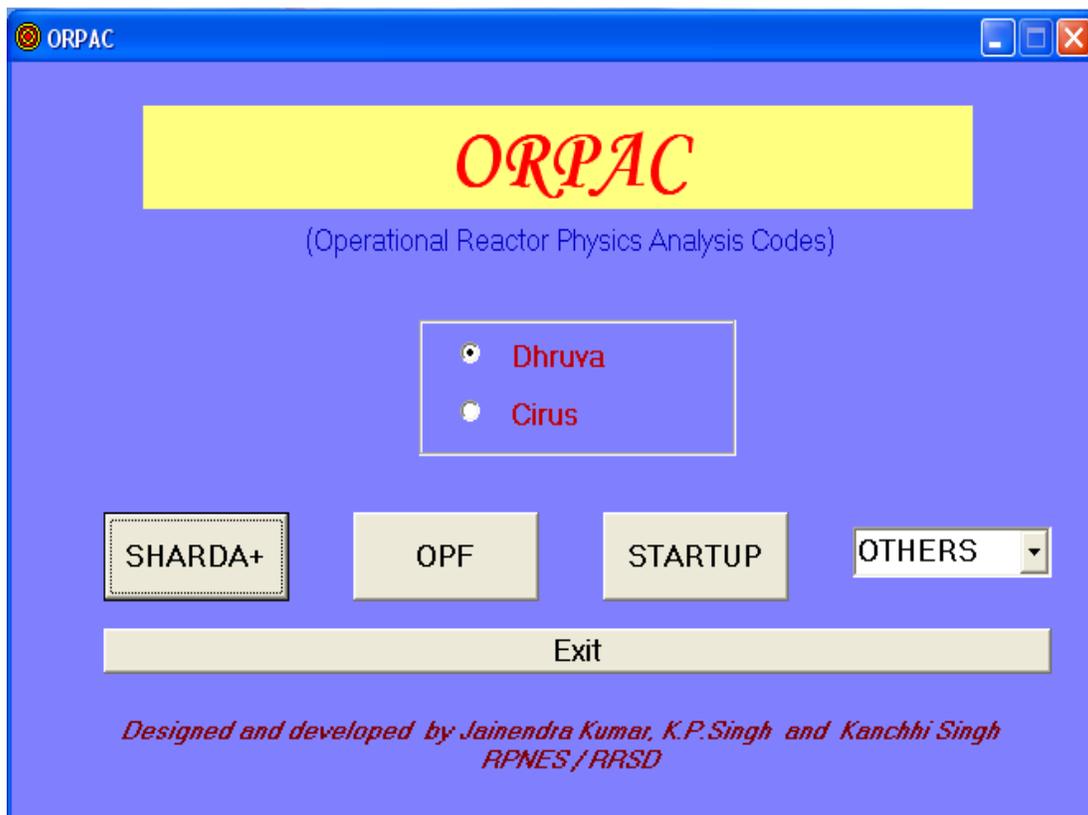
## D. OTHERS

This part is open for further addition of new codes in future. Presently it includes two programs

- (a) **Criticality**: for estimation of critical height and reactivity load/gain.
- (b) **Decay heat**: for estimation of decay heat production in a spent fuel

## Running the Code

The opening Graphical User Interface (GUI) of ORPAC is shown below



## 1. Running SHARDA+

**Inputs required:** sample composition, sample mass, density and geometry description and irradiation required (e.g. details about flux, irradiation time and location inside the reactor core viz. tray rod location, circle number and PCR etc.)

### Steps:

(1) Selecting sample: Select a sample and its composition from the list included in the boxes named as “Sample Name” and “Composition”. If the list does not include the sample of your interest, simply type the name of the sample and its composition in respective boxes. The composition of the sample should be in conventionally accepted format. Some examples are:

- (a) For pure elements: write the chemical symbol of the element e.g. Mg, Na, Al, Mo, P etc.
- (b) For compounds: Write the chemical formula of the sample e.g. MoO<sub>3</sub>, MgBr<sub>2</sub>, SeO<sub>2</sub>, Al<sub>2</sub>O<sub>3</sub> etc. In case of complex compounds write their chemical formula in simplified form.
- (c) For mixtures: Write Chemical composition of the sample with wt% of each constituent. e.g. for scandium glass, the composition is SiO<sub>2</sub>(65.00%) + CaO(16.00%) + Na<sub>2</sub>O(18.00%) + Sc<sub>2</sub>O<sub>3</sub>(1.00%).

(2) Click to the button “Select” after selecting/typing the sample name and composition in their respective boxes.

(3) In case of new sample a message box will ask for addition to the list. Select Yes/No/cancel according to your choice.

(4) Enter sample mass, density in respective input boxes.

(5) Choose geometrical shape option appropriate to your sample and provide relevant details regarding inner/outer radius and height in case of cylindrical sample. Default option will be for spherical geometry. No detail about radius is required in case of spherical sample.

(6) Choose sample location option from Tray Rod, Self Serve and PCR and respective position and circle number from the nearby boxes. In case of PCR option Tray Rod position and circle number are not required.

(7) Enter irradiation time in terms of either MWD or in units of Hour in one of the respective input box.

(8) Input box for flux is required to be filled when flux is known in advance. Flux should be entered in multiple of  $1.0E+14$  n/cm<sup>2</sup>/sec only. For example a flux  $9.7E+13$  n/cm<sup>2</sup>/sec should be written as 0.97.

(9) Select pile factor corresponding to the operational power. Default value is taken to be one.

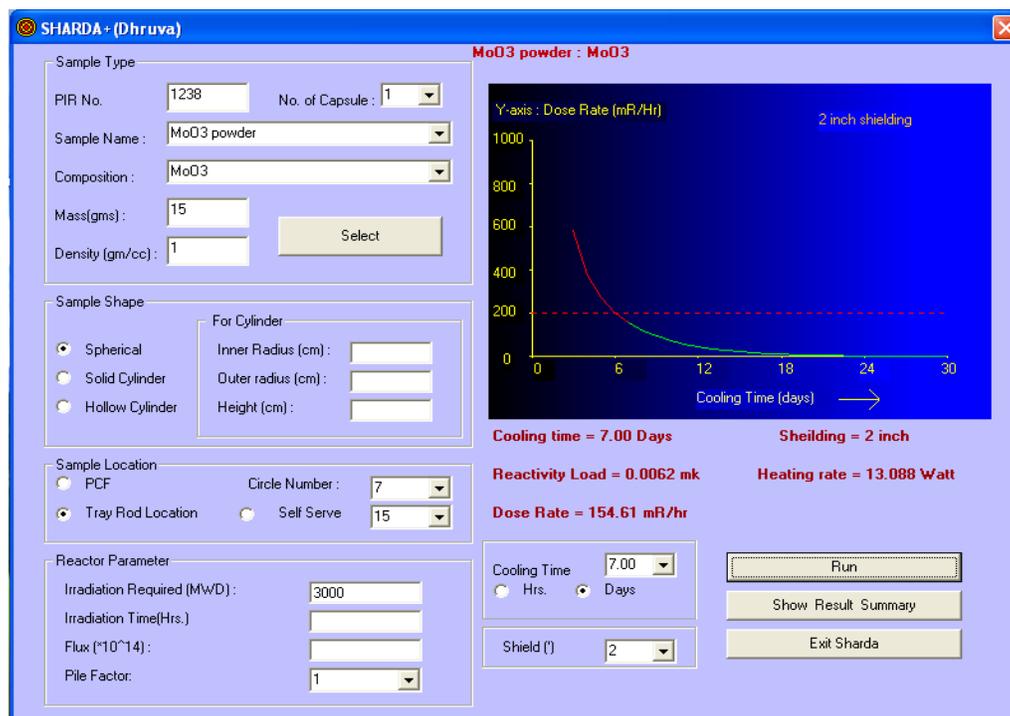
(10) After checking all input and options click to the “Run” button.

(11) Results for reactivity load, heating rate, cooling time and shielding requirement and dose rate at the surface corresponding to the shown cooling time and lead shielding will be displayed. A graph showing the variation of dose rate for given lead shielding with time will also be displayed on this form. Dose rate (mR/hr) will be displayed on the Y-axis of the graph.

(12) If cooling time is greater than 30 days then a message box will appear to ask for higher shielding. In this case select more shielding from the shielding box. Selection of new shielding time will invoke the program to run. The value of cooling time and graph will be updated accordingly without clicking of the “Run” button.

(13) If for a given cooling time it is required to know the shielding, then select a cooling time from the cooling time list in Hours or days. Simple selection of the cooling time will invoke the program to run and the value of shielding will be updated automatically without clicking to the “Run” button. In this case the dose rate graph will show variation of dose rate (mR/hr) with shielding up to 10 inch for a chosen cooling time. A message box will ask for more cooling if shielding is greater than 9 inch.

An example displaying a sample MoO<sub>3</sub> (in powder form) with mass 15gm for irradiation up to 3000 MWD having spherical geometry is shown below. The sample is placed at tray rod ( H-07 ) and at location 15 in Dhruva.



The output result shows the reactivity load due to the sample is 0.0062 mk and heating rate in the sample is 13.088 watt. For 2 inch lead shielding the sample is required to cooled for one week. The dose rate at the surface of the 2 inch lead shield sample after one week of cooling is 154.61 mR/hr. The dose rate graph shows the variation of dose rate with time up to 30 days for 2 inch lead shielding. The broken line (red coloured) is the 200 mR/hr line (maximum permissible dose rate limit). The green coloured part of the curve represents the permissible/safe values of dose rate whereas red coloured part of the curve represents the unsafe values.

To see summary of result, click to the button “Show Result Summary”. You will see new window shown below. Here information about isotopes produced, their activity, half life and their contribution to the total dose rate at 1 foot are shown.

Isotope	Half life(hrs.)	Activity (Curie)	Doserate at 1 feet (mR/Hr)
Mo- 93	6.9	0.203	2949.369
Mo- 99	66.0	9.256	3278.328
Mo- 101	0.2	5.270	21428.175
Tc- 101	0.24	5.270	11059.051
O - 19	0.0075	0.000	1.881

Total(Sample+Capsule) Bare Dose rate(mR/hr) at 1 feet = 39608.061

Irradiation of  $\text{MoO}_3$  produces isotopes Mo-93, Mo-99, Mo-101, Tc-101 and O-19 with half lives 6.9 hrs, 66 hrs, 12 min, 14.4 min and 27 sec respectively.

Scrolling down will show the values of shielded dose rate at 1 foot and at the surface at each cooling time(hr) and shielding(inch).

CoolTime	Shield	Shielded Dose Rate at 1 feet	Shielded Dose Rate at surface
0.00	2.0	1201.50	43253.95
1.00	2.0	339.97	12238.81
2.00	2.0	291.16	10481.85
3.00	2.0	270.19	9727.01
4.00	2.0	251.72	9062.08
5.00	2.0	234.78	8452.19
6.00	2.0	219.21	7891.58
7.00	2.0	204.89	7375.87
8.00	2.0	191.70	6901.16
9.00	2.0	179.55	6463.87
10.00	2.0	168.35	6060.75
11.00	2.0	158.02	5688.86

To have a detailed look on the result, click to the button “Details”. New form (window) giving information about gamma radiations emitted by each isotope with their energy (MeV), yield (%), their activity and dose rate at every cooling time and shielding will be displayed.

CoolTime= 0    Shield = 2

For Element Mo

Mo- 93		Half life(hrs.) = 6.9	Activity =0.203	
Gamma Energy(Mev)	Yield (%)	Bare Dose Rate(mR/hr)	Shielded Dose Rate(mR/hr)	
0.26	100.00	316.8743	0.0000	
0.68	100.00	828.7483	4.9215	
1.48	100.00	1803.7462	184.7989	
Bare Dose Rate(mR/hr) at 1 foot =		2949.369		
Shielded Dose Rate(mR/hr) at 1 foot =		189.720		
Mo- 99		Half life(hrs.) = 66	Activity =9.256	
Gamma Energy(Mev)	Yield (%)	Bare Dose Rate(mR/hr)	Shielded Dose Rate(mR/hr)	
0.74	6.00	2465.8266	24.7605	
0.38	1.00	209.3731	0.0005	
0.18	6.00	603.1278	0.0000	
Bare Dose Rate(mR/hr) at 1 foot =		3278.328		
Shielded Dose Rate(mR/hr) at 1 foot =		24.761		
Mo- 101		Half life(hrs.) = 0.2	Activity =5.270	

<< Back to Result      Close

## 2. Running STARTUP

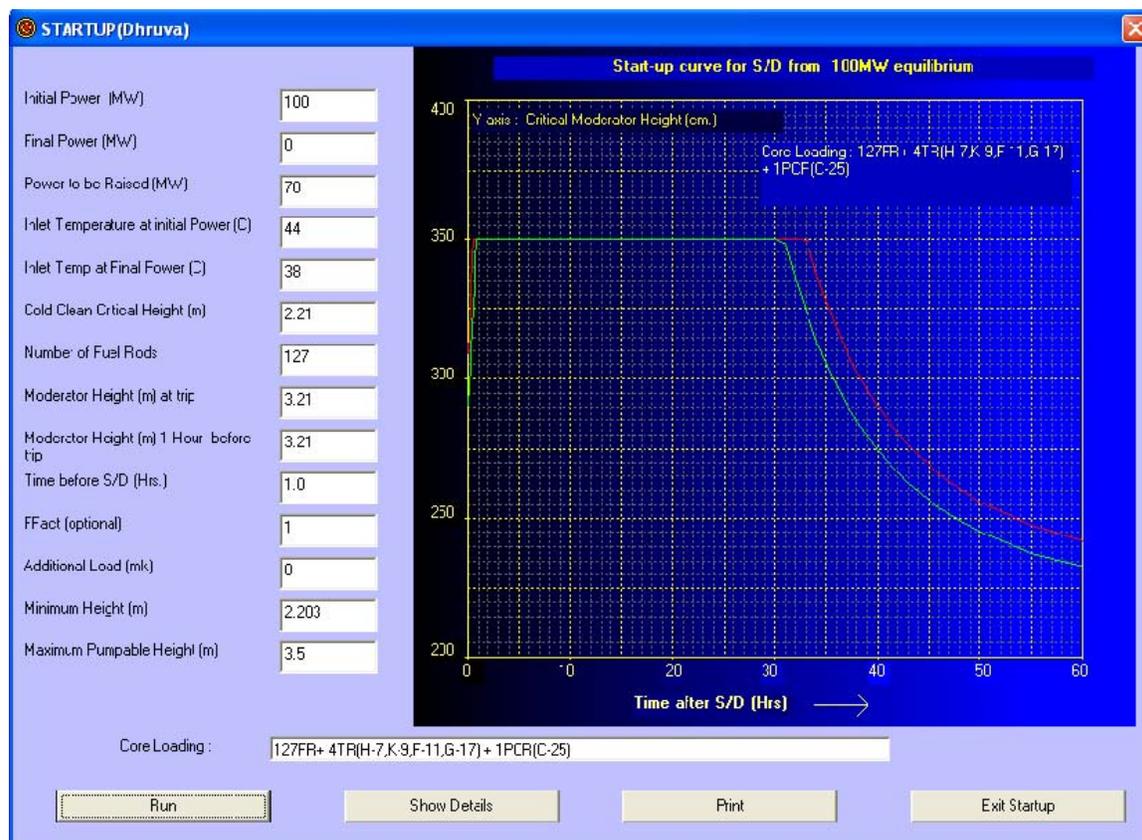
**Inputs required:** Initial power, final power, step power, inlet temperature at initial power and final power and clean core critical height and also equilibrium/non-equilibrium moderator heights when reactor is to be started after S/D.

### Steps:

- (1) Click to the startup button. The main interface for startup program will be displayed.
- (2) Enter initial power (MW) and final power (MW) of the reactor in respective input boxes. In case of initial startup (clean core) initial power will be zero and final power will be the power to be raised in steps. In case of S/D or trip the initial power will be the power at which the reactor was tripped and final power will be zero.
- (3) Enter the power step (MW) to which power is to be raised. Normally it is taken to be 70% of the initial power.
- (4) Enter inlet plenum temperature at initial power and at raised power.
- (5) Enter clean core critical height and number of fuel rods.

- (6) Enter moderator height at S/D and also before S/D and corresponding time interval (T) before S/D. In case of S/D from equilibrium both height will remain same. In case of initial startup (clean core) moderator height is not required.
- (7) In case of any additional load to the reactor enter this load in mk.
- (8) Enter core loading.
- (9) Click to the “Run” button. Startup curve will appear on this form. To see details click to “Show Details”. A form showing detailed information regarding Iodine and Xenon load, power load, temperature load and total reactivity load, will be displayed.
- (10) Click to “Print” to get a printed copy of the startup curve.

An illustration showing startup curve for S/D from 100 MW equilibrium is given below:



In the above illustration the reactor is restarted after S/D from 100 MW equilibrium power and reactor power raised to 70 MW (70% of reactor power at the time of S/D). The inlet moderator temperature at 100 MW is taken to be 44 °C and at 70 MW power it is taken to be 38 °C. The clean core critical height has been taken as 2.21 meter for 127 fuel rods with core loading: 127FR+ 4TR(H-7,K-9,F-11,G-17) + 1PCR(C-25). The equilibrium moderator height at 100 MW has been taken to be 3.21 meter. Maximum pumpable moderator height is assumed to be 3.5 meters.

Variations of xenon load and moderator height with time at low power and at 70 MW are shown in detail form.

Time(Hr)	Xenon(mK)	Total Reactivity at LP (mk)	Total reactivity at P1 (mk)	Height LP(cm)	Height P1(cm)
0.00	27.76	28.19	32.68	290.69	309.26
0.10	29.38	29.81	34.30	297.00	316.90
0.20	30.98	31.40	35.90	303.61	324.96
0.30	32.54	32.96	37.46	310.54	333.50
0.40	34.07	34.49	38.99	317.82	342.56
0.50	35.57	35.99	40.49	325.47	352.20
0.60	37.04	37.47	41.96	333.54	362.49
0.70	38.49	38.91	43.41	342.07	373.51
0.80	39.90	40.32	44.82	351.09	385.35
0.90	41.29	41.71	46.21	360.66	398.12
1.00	42.64	43.07	47.56	370.85	411.95

### 3. Running OPF

**Inputs required:** Inlet -outlet temperatures of fuel channels, calandria, shield, vault and pile block. D<sub>2</sub>O flow rates in fuel channels, calandria, shield and vault and air flow in pile block.

#### Steps:

- (1) Click to “OPF” command button. Form “OPF” will be displayed.
- (2) Fill up all relevant input boxes and click to the “Next” Button. A new inter face will be displayed for further input. To skip or to exit click to “Exit OPF”.

Core Loading: 127FR(114 Thick clad+13 thin clad) + 4TR(H-7,K-9,F-11, G-17) + 1PCR(C-25)

TED (MWD): 8445.3    Time: 1400    Date: 18.04.07    Average Inlet Plenum Temp.: 42.3

Set Power (%): 60    Indicated Digital Power (MW): 58.9    RV Level Manometer (cm): 314    RV Level digital (cm): 317.1

	Flow to Inlet Plenum (lpm)	Tube Sheet Flow (lpm)	Moderator Return Flow (lpm)	Inlet Plenum Temperature	Coolant Temperature	Moderator Return flow temp	D2O inlet Temp HW/PW HX.
Loop 1	22000	1860	1800	43.6	17.27	52.7	62.5
Loop 2	23000	770	1620	43.1	17.3	52.1	62
Loop 3	22000	1840	1500	42.6	15.4	53.2	59

Vault flow (lpm): 2280    Shield Flow (lpm): 95    Pile Block Air Flow (lpm): 7400

Vault inlet Temp. (C): 43.2    Shield Outlet Temp. (C): 39.6    Pile Block Inlet Temp. (C): 27

Vault Outlet Temp. (C): 46.5    Shield Outlet Temp. (C): 41.3    Pile Block Outlet Temp. (C): 46.5

Buttons: Next, Exit OPF

OPF(Dhruva)

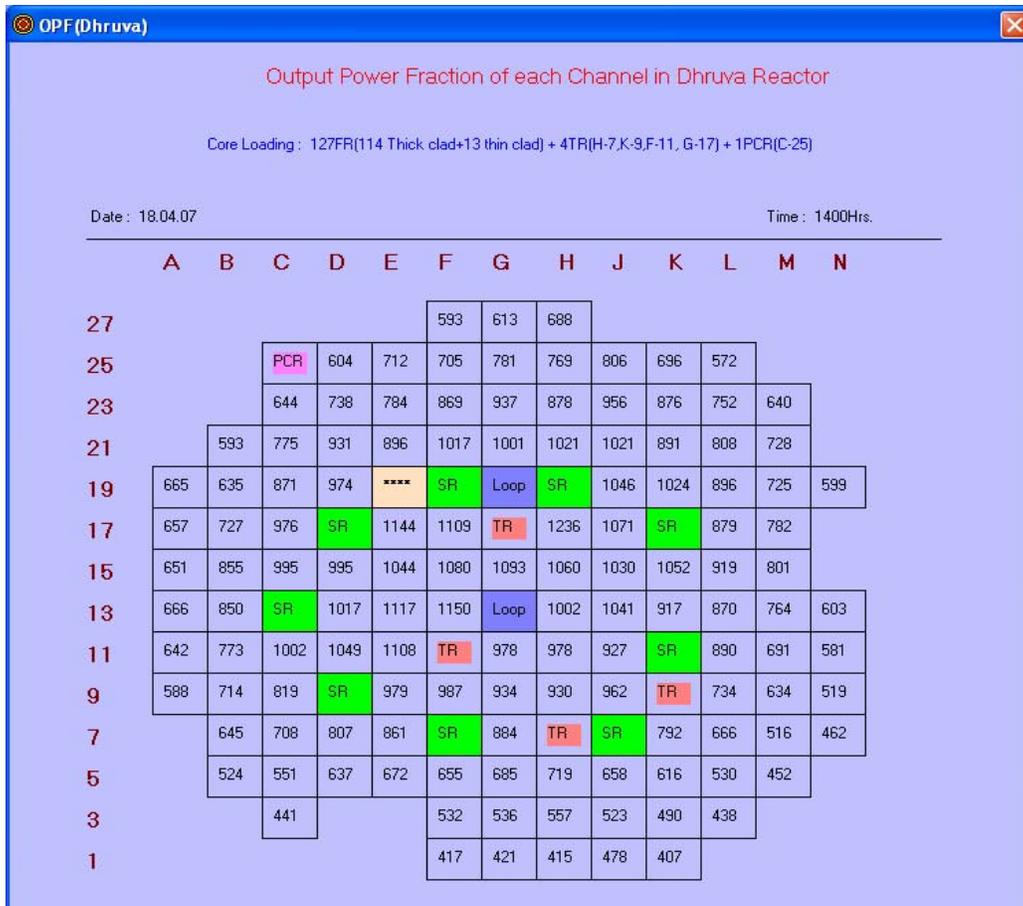
Enter Fuel Channel Flow (lpm) and Fuel Channel Outlet Temperature (C)

	Temp	Flow														
A-09	50.5	84.3	C-21	53.1	84.3	E-25	52.1	84.3	G-23	55.4	83.7	J-13	56.5	85.7	L-11	54.5
A-11	50.9	86.7	C-23	51	87	F-01	48.2	82	G-25	52.9	86	SP			L-13	54.3
A-13	51.5	84.3	D-05	51.2	84	F-03	49.8	83.3	G-27	50.6	87	J-17	57	85.3	L-15	54.9
A-15	51.3	83.7	D-07	53.5	85	F-05	51.6	82.7	H-01	48.1	82.7	J-19	56.7	85.3	L-17	54.7
A-17	51.2	85.3	D-09#	0	0	F-07#	0	0	H-03	49.8	87.3	J-21	55.8	89.3	L-19	54.9
A-19	51.4	84.7	D-11	56.7	85.7	F-09	56.1	84	H-05	52	87	J-23	55	88.3	L-21	53.7
B-05	49.5	84.3	D-13	56.1	87.3	F-11#	0	0	H-07#	0	0	J-25	53.4	84	L-23	52.3
B-07	51.2	85	D-15	56.3	83.7	F-13	58.1	85.7	H-09	55.3	84	K-01	48	82.7	L-25	50.3
B-09	51.9	86.3	D-17#	0	0	F-15	56.9	87	H-11	55.7	86	K-03	49.3	82	M-05	48.5
B-11	53.1	84.7	D-19	55.5	87	F-17	57.4	86	H-13	56.2	84.7	K-05	50.7	86	M-07	49.7
B-13	54	85.3	D-21	55.5	82.7	F-19#	0	0	H-15	56.7	87	K-07	53.1	86.3	M-09	51.5
B-15	54.3	83.7	D-23	52.7	83.3	F-21	56.3	85.3	H-17	59.2	86	K-09#	0	0	M-11	51.4
B-17	52.5	84	D-25	50.6	84.3	F-23	54.7	82.7	H-19#	0	0	K-11#	0	0	M-13	52.7
B-19	51.3	83.3	E-05	51.5	85.7	F-25	52.3	83.7	H-21	56	86.7	K-13	55.3	83.7	M-15	53.2
B-21	50.6	83.3	E-07	54.6	81.7	F-27	50.6	83	H-23	54.2	86.7	K-15	56.5	87.3	M-17	53
C-03	48.1	88	E-09	55.5	86.3	G-01	48	85.3	H-25	53	84.7	K-17#	0	0	M-19	52.3
C-05	49.8	86	E-11	57.7	84.3	G-03	49.9	83.3	H-27	51.8	85	K-19	56.6	84.3	M-21	52.2
C-07	51.9	86.7	E-13	57.7	85.3	G-05	51.7	85.3	J-01	48.8	85.3	K-21	54.7	84.3	M-23	51.2
C-09	53.7	84.3	E-15	56.4	86.7	G-07	54.8	83.3	J-03	49.4	86	K-23	54.2	86.3	N-07	48.8
C-11	55.3	89.7	E-17	57.7	87	G-09	54.8	87	J-05	51.4	84.7	K-25	52.2	82.3	N-09	49.6
C-13#	0	0	E-19#	0	0	G-11	56.3	82.3	J-07#	0	0	L-03	48.4	83	N-11	50.2
C-15	56	84	E-21	55.3	81.3	G-15#	56.9	88	J-09	55.4	86.3	L-05	49.6	85	N-13	50.5
C-17	55.8	84.7	E-23	53.2	84	G-17#	0	0	J-11	54.9	86	L-07	51.4	85.7	N-19	50.6
C-19	54.5	84	SP	0	0	G-21	55.8	87.3	J-13	54.9	86	L-09	52.3	86.3	J-15	56.5

Back Show OPF Print Temp. Print Flow Print OPF Save

- (3) Fill up Outlet Channel Temperature and Coolant flow rate (%) for each channel. (100% flow rate in Dhruva reactor is equal to 600 lpm.)
- (4) To see OPF click to the button “Show OPF “. A typical OPF (normalized to 100000) chart for Dhruva is displayed in the next page.
- (5) To print OPF click to the button “Print OPF”.
- (6) To print channel outlet temperature click to the button “Print Temp”.
- (7) To print channel flow click to the button “Print Flow”.
- (8) To save the input data click to the button “Save”.

In case of a particular fuel channel having OPF higher than 5% of the reference/standard OPF chart, the concerned fuel channel in the OPF chart will be displayed in RED colour for drawing attention to make corrective action.



\*\*\*\* Blocked position

#### 4. Running OTHERS programs

##### (a) Criticality :

Select option “Criticality” from “Others”. To find reactivity load enter the moderator height in the input box and click to the command button “Find Load”.

An illustration below shows reactivity load corresponding to moderator height 2.21 m is 32.561 mk.

Criticality

Critical Height

Load (mk)

Height (meter)

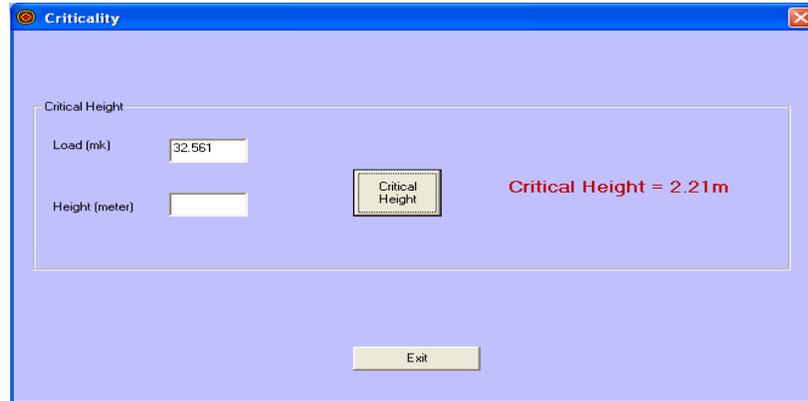
Find Load

Total reactivity Load = 32.561 mk

Exit

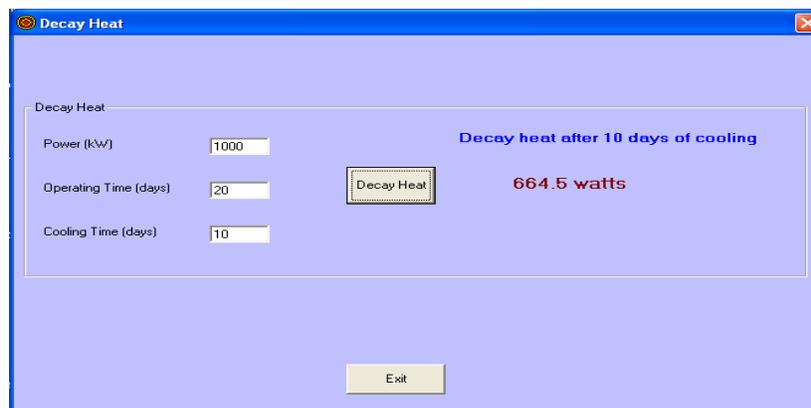
To find out critical height, enter reactivity load (mk) and click the command button “Critical Height”. The calculated critical height will be displayed.

The illustration below shows critical height for a load of 32.561 mk is 2.21m



### (b) Decay Heat

Enter reactor power (kW), operating time (in days) and cooling time (in days) and click “Decay Heat” button. The calculated decay heat will be displayed on the form.



## Conclusions

Although most of the calculations involved in this code are based on crude approximate techniques, the accuracy of results is good enough for practical purposes. The code presents handy tools for personnel involved in reactor operation and fuel / tray rod handling works. All necessary data are provided within the code. Because of its graphical user interface (GUI) features only little training to run this code is required.

## **Acknowledgements**

We are thankful to Shri V.K. Raina, Director Reactor Group and Shri S.K. Marik, Head RRSD for their keen interest in this work. We are also thankful to Shri Kunal Chakraborty and Shri Jainendra Tiwari, ROD for useful suggestions from time to time during the development of the present code. Thanks are due to Shri S.K. Singh, Shri Y.S. Rana and Ms. Archana Sharma, RRSD for their help and valuable suggestions.

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