AN EVALUATION OF NEUTRONIC SAFETY CHARACTERISTICS OF PROSPECTIVE FAST REACTOR METAL FUELS

By

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SYNOPSIS

Due to the increasing demand for electricity in India, fast breeder reactors to be set up beyond 2020 would be based on metal fuels as they have shorter doubling time. This work was motivated from the rising concern, that as fast reactor cores are not always in a most reactive state, there is always vulnerability with metal fuel regarding the potential of positive temperature reactivity feedback in accident scenarios. The major temperature reactivity feedbacks are associated with changes in temperature of the fuel (Doppler), structural material (Core radial expansion) and coolant (coolant expansion and voiding). In metal fuel reactors with U-Pu type, Doppler feedback is lower and sodium void coefficient is positive and large. It is always preferable to have a large negative Doppler reactivity effect and a negative or near zero sodium void reactivity. Most of the methods, which reduce sodium, void reactivity and increase negative Doppler coefficient have penalty on breeding ratio and capital cost. Open literature is scarce on comparative studies of the nuclear performance of advanced reactor fuels. In the earlier studies, the cross-section data are associated with large uncertainties, particularly for thorium isotope and detailed modeling of the core was not used. The present work tries to identify a fuel type with superior safety characteristics without much penalty on breeding ratio. Transport calculations are carried out taking the exact geometry of the core and using recent nuclear data. The need for such detailed modeling is brought out in the thesis. It has emerged from this study that among the hybrid combinations of Th-^{233}U fuel type, only carbide combination provides breeding and other combinations are not suited for

breeding in medium sized reactors. Comparative assessment of different candidate fuels is made and arrived at fuel compositions that have superior safety characteristics.

The principle reason that causes the initiation of any transient in FBR's is due to the imbalance in heat generation to heat removal along with the failure of Plant Protection System (PPS) to terminate the neutronic excursion.The inherent safety of fast reactors is studied through unprotected transients. Safety development in fast reactor program always looks for reactor designs to avoid coolant boiling and fuel element failure during unprotected ULOF accidents. The major accident sequences considered for this type of analysis are Unprotected Transient Over Power Accident (UTOPA), Unprotected Loss Of Flow Accident and Unprotected Loss of Heat Sink Accident (ULOHA). For metal fuelled reactors, ULOFA analysis provides information on starting and propagation of sodium void, initiation of melting of fuel, time availability for corrective actions such as opening the damper to initiate SGHDR system ensuring the passive shutdown capability of the reactor. For the first time, quantitative assessment of inherent safety characteristics of Th-Pu metal fuel is done by ULOFA analysis. It emerges from these studies, Th-Pu metal shows the best safety characteristics.

The thesis also presents results regarding criticality safety of metal fuels. Whenever there are operations with fissile materials during fuel fabrication, there are always chances for inadvertent nuclear chain reactions or criticality. To avoid the risk of criticality events in demonstration metal fuel fabrication facility, safety studies has been carried out to establish rigorous control of fissile materials. This study eliminates the possibility of inadvertent criticality in the plant thus ensuring public safety.

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List of Abbreviations

Initials	Full form
AERB	Atomic Energy Regulatory Board
ADS	Accelerator Driven System
B.R	Breeding Ratio
BARC	Bhabha Atomic Research Centre
CSR	Control Safety Rod
DAE	Department of Atomic Energy
DSR	Diverse Safety Rod
EBR	Experimental Breeder Reactor
FBTR	Fast Breeder Test Reactor
IGCAR	Indira Gandhi Centre for Atomic Research
IAEA	International Atomic Energy Agency
IFR	Integrated Fast Reactor
IPCC	Intergovernmental Panel on Climate Change
LFR	Lead fuelled Fast Reactor
MOX	Mixed Oxide Fuel
PHWR	Pressurized Heavy Water Reactor
PFBR	Prototype Fast Breeder Reactor
PPS	Plant Protection System
PUMP-F	Plutonium Uranium Metal Pin Fabrication
SGDHRS	Safety Grade Decay Heat Removal System
ULOHA	Unprotected Loss of Heat sink Accident
ULOFA	Unprotected Loss of Flow Accident
UTOPA	Unprotected Transient Over Power Accident

Chapter 1

Introduction

1.1Background and Motivation

The increase in global energy demand has led to the release of over 1100 GTe of Carbon dioxide (a green house gas that is contributing mainly to global warming) into atmosphere, which is mainly from the combustion of fossil fuels (Sims et al., 2007). In order to reduce this green house gas emission one has to look for more realistic options such as increased use of renewable sources of energy and nuclear power (Peter, 2013). There are regional variations in the demand for energy and according to the 4th Intergovernmental Panel for Climate Change (IPCC) report; highest per capita demand for energy is coming from developing countries (Sims et al., 2007). There are numerous projections made by several national and international agencies regarding the future demands of energy in India (TRS and TRAE, 1999; CEA, 2000; TERI, 2001; Grover and Chandra, 2006). From these studies it has evolved that even with the most effective use of fossil fuel reserves, non conventional resources and hydroelectric schemes; in future India has to rely on nuclear fuel resources to meet the gaps in demand and supply of energy (Grover and Chandra, 2006; Banerjee, 2010). The main objective of the nuclear energy program in India is to utilize, effectively, the scarce resources of uranium and vast potential of thorium. This is planned to be achieved through a three stage program which is outlined based on the domestic resource position of uranium and thorium. In the first stage, uranium is utilized in Pressurized Heavy Water Reactors (PHWR) and plutonium will be recovered from the spent fuel through reprocessing. In the second stage plutonium will be used with uranium in Fast Breeder Reactors (FBR) for increasing the fissile

material base and power generation. Thorium is envisaged to be used in the blanket region to generate ^{233}U. In the third stage, Thorium based fuels along with ^{233}U will be deployed in reactors (thermal/fast) or subcritical systems such as Accelerator Driven System (ADS) for sustainable power generation (Grover and Chandra, 2006).

Currently fast reactor development is actively pursued in India. Fast Breeder Test Reactor (FBTR), which is a test reactor, has been in operation since 1985 (Srinivasan et al., 2006) in Indira Gandhi Center for Atomic Research (IGCAR). It is a loop type reactor of 40 MWt capacity fuelled with mixed carbide fuel (Ganguly et al., 1988) and sodium is used as the coolant. Construction of Prototype Fast Breeder Reactor (PFBR) of 500 MWe capacity using Mixed Oxide (MOX) fuel and sodium as the coolant (Chetal et al., 2006) is in advanced stages of completion. Due to the increasing demand for electricity, FBRs to be set up beyond 2020 would be based on metal fuels as they have shorter doubling time (Grover and Chandra, 2006).

To deploy metal fuel cycle successfully in commercial FBRs, it is essential to gain experience in the fabrication of metal fuel as well as understanding about its in-pile neutronic performance. Test metal pins of natural uranium, enriched uranium and U-19%Pu, fabricated in Bhabha Atomic Research Center (BARC) are currently undergoing irradiation in FBTR. A demonstration facility to gain expertise in metal fuel fabrication of the above ternary alloy (Ganesan et al., 2011) is also nearing completion in IGCAR. The facility will handle pure plutonium metal, enriched uranium and their mixtures as feedstock.

In addition to its breeding potential, one of the vital roles of FBRs is to demonstrate safety and stability during normal and off-normal conditions (Benjamin, 1968; Lamarsh and Baratta, 2001). Safety requires that a reactor be fabricated and operated to the highest standards (Farmer and Gilby, 1967; Lamarsh and Baratta, 2001). During reactor operation, substantial changes in the operational states can change the reactivity of the system, which in turn will change power, temperature and geometry of the core (Tentner et al., 2010). The reactivity response of the core to such operating conditions (especially with variation in temperature) will determine the safety and stability in fast reactors (Benjamin, 1968). The major temperature reactivity feedbacks are associated with changes in temperature of the fuel (Doppler), structural material (Core radial expansion) and coolant (coolant expansion and voiding). Doppler reactivity and core radial expansion are usually negative feedbacks in large fast reactors, which will try to terminate any inadvertent power rise where as coolant voiding in the central part of the fast reactor core introduces positive reactivity feedback. The positive reactivity insertion in medium and large sized fast reactors during coolant voiding is maximum for the fuel type which generate hard neutron spectrum, e.g. metal fuels with U-Pu type (Riyas and Mohanakrishnan, 2008; Sathiyasheela et al., 2011). Studies are required to identify fuel types that have optimum breeding and safety characteristics.

Even if fast reactors are very sensitive to reactivity feedbacks, their operation and physical characteristics are amenable for heat removal in accident scenarios (Qvist, 2013). This will come under the inherent safety characteristics of the fast reactors. Usually inherent safety of a reactor design is analyzed through unprotected accident sequences where the response of the reactivity feedbacks and time availability for decay

heat removal is noticed. Most of the unprotected transient analyses are limited to metal and oxide fuels of U-Pu type (Cahalan et al., 1990; Yokoo and Ohta, 2002; Harish et al., 2009; Sathiyasheela et al., 2011). So far no attempt has been made to analyze the inherent safety characteristics of thorium based fuels and such studies are much needed for the effective utilization of country's vast thorium reserves.

India's current interest in metal fuel cycle, to achieve the targeted nuclear growth, demands the use of FBTR as test bed in irradiating large number of metallic fuel sub assemblies in the coming years (Ganesan et al., 2011). Fuel with very high fissile content has to be fabricated in the demonstration facility for this purpose. Nuclear criticality safety plays an important role in fuel fabrication stage, as there is a potential for accidental criticality. To ensure the safety of operating personnel, criticality safety has to be established by limiting the quantity of material at every stage of fuel fabrication.

1.2 Objectives

The scientific issues addressed in this study include the following:

(I) Breeding capability and safety related neutronic parameters of advanced fuels with detailed modeling of the reactor core and using recent cross-section set:

(i) Evaluation of breeding performance of advanced fast reactor fuels using recent cross-section set ENDF/B-VI.7 and JEFF-3.1 along with the detailed modeling of the core using 3D transport code.

(ii) *Evaluation of safety neutronic parameters such as delayed neutron fraction, fuel Doppler coefficient and sodium void worth for all possible fissile-fertile combination of advanced fuels.*

(II) Inherent safety characteristics of Thorium-Plutonium metal fuel:

(i) *Evaluation of static reactivity coefficient for Th-Pu metal fuel*.

(ii) *Analysis of the dynamic behavior of the core during ULOFA to understand the passive shutdown capability of the reactor*.

(III) Criticality safety studies of Plutonium Uranium Metal Fabrication facility:

(i) *Safe amount of metal ingots that can be loaded into induction melting furnace without a risk of criticality.*

(ii) *Quantity of metal fuel slugs and pins, which can be safely handled during injection casting and fuel pin fabrication*.

Several geometric configurations of fissile material were considered in these studies.

1.3 Organization Of The report

The thesis is organized into 6 chapters as follows:

A brief introduction on fast reactors and its role in Indian nuclear energy program, the current status of the fast reactor program in the country and the studies required in the safety neutronic characteristics of fast reactor fuels has been presented in the first chapter. Specific objectives of present work are also given in this chapter.

Chapter 2 presents a detailed literature review on the safety concerns in metal fuelled fast reactors and in the metal fuel fabrication plant. This chapter also covers the physics of nuclear reactors and the simulation tools used for the present study.

Chapter 3 presents analysis carried out on breeding ratio and safety neutronic parameters (effective delayed neutron fraction, Doppler coefficient and sodium void coefficient) of advanced fast reactor fuels comprising of all possible fissile- fertile combination of metal, oxide, carbide and nitride using recent cross-section set ENDF/B-VI.7. The core used in this study is a sodium cooled fast reactor similar to 500 MWe PFBR. A comparative study on the breeding performance of the above fuel types has been done with another standard cross-section set JEFF-3.1.

Chapter 4 contains analysis on static reactivity coefficient and unprotected loss of flow accident in a reactor core fuelled with Th-Pu metal fuel. This study is done to identify the passive shut down capability of the reactor under unprotected transient under

cooling event. An uncertainty analysis on the thermo physical properties of Th-Pu metal fuel, which will reflect in the reactivity feedback, is also presented in this chapter. Sensitivity analysis has been carried out to take care of uncertainties in the thermo physical properties of Th-Pu metal fuel and to verify, if the final conclusions on the reactivity feedbacks change.

In Chapter 5, methodology and calculations are presented to estimate safe amount of maximum fissile mass that can be handled during different stages of metal fuel pin fabrication in **P**lutonium-**U**ranium **M**etallic **P**in **F**acility (PUMP) Facility. Before commercial deployment of metal fast reactors, it is essential to proceed in steps to understand the irradiation behavior of the fuel as well as to get experience in fuel fabrication, reprocessing and fabrication of fuel using reprocessed fuel. In India, as a first step, it is planned to use FBTR as a test bed in irradiating large number of metallic fuel pins. In order to fabricate fuel for test program, PUMP facility is planned in Indira Gandhi Center for Atomic Research (IGCAR). Since criticality depends not only on mass of the fissile material but also on geometry, several geometric configurations of fissile material were considered.

The salient conclusions of the present research are given in Chapter 6, along with possible future work, in light of the current results.

Chapter 2

Safety And Physics Considerations For Fast Reactors

2.1 Introduction

An important characteristic of fast reactors is that they produce more fuel than they consume, along with their ability for energy production. The world's first nuclear plant which generated electricity was Experimental Breeder Reactor 1 (EBR-1), a fast breeder reactor designed by Argonne National Laboratory (ANL) in 1951 (Michal, 2001). This reactor, fuelled with uranium metal pins, validated the breeding concept and gave impetus to the idea of long term dependence on electricity from atomic energy.

The principal goals in the development of fast reactor design are safe operation, high breeding ratio and low cost (Walter and Reynolds, 1981). These requirements are in conflict when one goes to optimize the nuclear design of medium and large fast power reactor systems (Benjamin, 1968; Driscoll and Hejzlar, 2005). Safe operations require reliable components and providing sufficient safety margin for potential accidental situations. Reactor designs, which enhance nuclear safety sometimes, cause penalty on breeding gain and cost. Higher breeding ratio implies lower doubling time and lower fissile specific inventory. Low cost to a larger extent can be achieved through optimum utilization of high fissile content in the fuel, i.e. through a fuel type allowing high burnup.

Metallic fuels are neutronically ideal for breeding, in fast reactor system, as they produce extremely hard neutron spectrum (Hofman et al., 1997). The use of metallic fuel in commercial reactors during late 1970's, specifically with binary U-Pu alloy, were hindered by the low burnup (due to excessive swelling) and the low melting temperature

of the fuel alloy. But the advantages of metallic fuel are high thermal conductivity, which can enhance safety features of the system (Bauer et al., 1990) and the economic recycling techniques such as electro refining, which provides non-proliferation and reduction of the quantity of high level waste, as all the actinides are recycled back to the fuel and fissioned for energy (Laidler et al., 1997). The Integral Fast Reactor (IFR) concept (Till and Chang, 1988; Chang, 1989) which took birth in ANL during 1980's, found metallic fuel with U-Pu-Zr system, the most suitable candidate over other metallic fuel system. Central to the concept is to utilize plutonium as its principal fuel, as it has the potential to simultaneously produce plutonium by irradiation of ^{238}U in blankets. By incorporating space for swelling, very high burnup was achieved with the fuel type (Hofman et al., 1997). Also, the Zirconium addition raises the fuel melting temperature and enhances compatibility between fuel and clad (Porter et al., 1990). Thus the IFR concept restored the interest in utilizing metal fuel cycle in commercial fast reactors.

The combination of U-Pu metal fuel type and sodium coolant provides harder neutron spectrum, which can have some adverse effect on safety and stability in large commercial fast breeder reactors. The present chapter surveys safety concerns during the normal and off-normal conditions in medium sized commercial fast breeder reactors. Safety considerations in the fabrication of metal fuels are also studied. Physics of nuclear reactors and simulation tools used for the thesis are also presented in this chapter.

2.2 Safety Considerations In Fast Reactors

The focus of safety considerations in fast reactors is very different from that of thermal reactors. Fast reactor cores are not always in a most reactive state as in the case of thermal reactors (Walter and Reynolds, 1981). A thermal reactor core becomes critical

only for a particular moderator to fuel ratio. Decrease in coolant volume fraction in a thermal reactor will always results in the reactor's subcritical state (Walter and Reynolds, 1981; Rineski, 2011). But in a fast reactor, the effect of decrease in volume fraction of coolant can be a major safety concern, especially with metallic fuels of U-Pu type, as they have the hardest neutron spectrum. Coolant voiding can give rise to positive reactivity feedback in large reactors. Core compaction can also happen with the resulting temperature rise, which again introduces positive reactivity to the system.

Even though large margin of temperature is available between peak sodium temperature and sodium boiling temperature, as demonstrated during various safety tests (Planchon et al., 1986), there is always a finite probability that a limited amount of sodium can undergo boiling in a metal fuelled FBR (Chang et al., 1991). Some of the postulated mechanisms that can lead to local coolant boiling are flow blockages, fuel pin failure with the release of fission gas which can blanket the fuel pin, gas entrainment which can lead to cavitations in coolant etc (Kuesters, 1976). Therefore in the design of metal fuelled fast reactors, care should be taken to ensure that negative reactivity feedbacks from fuel and structural expansion will override positive coefficient of the coolant or it is always preferable to have a large negative Doppler reactivity effect and a negative or near zero sodium void reactivity (Benjamin, 1968; Wade et al., 1997; Tucek et al., 2006).

2.2.1 Doppler Effect

In fast reactors, since the neutron spectrum is harder when compared to thermal reactors, it includes the resonance regions of both fissile (^{235}U, ^{233}U, ^{239}Pu, ^{241}Pu) and fertile (^{232}Th, ^{238}U, ^{240}Pu) fuel isotopes. As the temperature of a material is raised, the thermal motion of material nuclei is increased thus modifying the relative motion and effective cross-section. This results in a large variation in the effective fission and capture cross-section of the material. The reactivity change associated with this phenomenon is described as Doppler reactivity or Doppler effect. The resulting change in reactivity can be positive or negative, depending on the exact composition of the core (Stacey, 2001). During the operation of the reactor, the Doppler reactivity can undergo significant changes, as there is alteration in the composition with burnup (Kusters and Ganesan, 1978; Walter and Reynolds, 1981; Wallenius, 2009). A positive Doppler effect is a concern for reactors with a high concentration of fissile materials, especially with ^{239}Pu (Yang, 2011). The neutron energy, which is very important for Doppler effect, starts below 25 KeV (Stacey, 2001).

The temperature coefficient of reactivity with respect to fuel can be estimated from the equation

$$\frac{\partial k}{\partial T_F} = \int N_F \left[\varphi_f^+ \frac{\partial \sigma_f}{\partial T_F} - \phi^+(E)\left(\frac{\partial \sigma_\gamma}{\partial T_F} + \frac{\partial \sigma_f}{\partial T_F}\right) \right] \phi(E)dE \rightarrow (2.1)$$

$$\cong N_F \int \frac{1}{\upsilon} \frac{\partial \sigma_f}{\partial T_F} (\upsilon - 1 - \alpha)\phi(E)dE \rightarrow (2.2)$$

Here N_F is the density of the fuel nuclei, $\alpha \equiv \frac{\sigma_c}{\sigma_f}$ is the capture to fission ratio, $\phi^+(E)$ and φ_f^+ is the importance of a neutron at energy E and of a fission neutron. For critical system, each neutron on an average produces $1/\upsilon$ fissions, $\phi^+ \sim \varphi_f^+ \sim 1/\upsilon$ is used in the second form of the estimate. Since α generally decreases with increasing neutron energy, the reactivity change will tend to be more positive or negative depending on the hardness of the neutron spectrum in the core. For large power reactors, the density and volume effects of the core diminish and the role of prompt negative Doppler effect becomes vital for reactor safety and stability. Those reactor cores, which have moderating atoms such as carbon or oxygen with the fuel type, the neutron spectrum will be softer, so that a larger part of the energy spectrum will lie within resolved resonances of fertile nuclei. This will enhance negative Doppler feedback.

2.2.2 Sodium Voiding Effect

The temperature rise can lead to boiling or loss of the sodium in the core and the reactivity change associated with change in density of sodium is the sodium void effect (Nims and Zweifel, 1959; Walter and Reynolds, 1981; Tentner et al., 2010). All the effective cross-sections of fuel and structure will change when the scattering effect of sodium is removed. The reactivity change associated with sodium voiding in a fast reactor core can be attributed from leakage, absorption and spectral components. The change in reactivity associated with perturbation (first order) in coolant density can be represented as follows (Stacey, 2001)

$$\frac{\Delta k}{k} = \frac{\sum_{g=1}^{G} \int dr \left[\phi_g^+ \Delta \bullet (\nabla D_g \nabla \phi_g) - \phi_g^+ \Delta \Sigma_{rg} \phi_g + \phi_g^+ \sum_{g' \neq g}^{G} \Delta \Sigma_{g' \to g} \phi_{g'} + \phi_g^+ \chi_g \sum_{g'=1}^{G} \Delta (\upsilon \Sigma_{fg'}) \phi_{g'} \right]}{\sum_{g=1}^{G} \int dr (\phi_g^+ \chi_g \sum_{g'=1}^{G} (\upsilon \Sigma_{fg'}) \phi_{g'})}$$

$\rightarrow$ (2.3)

The leakage and spectral components corresponds to the first term (ΔD_G) and third term ($\Delta \Sigma_{g' \to g}$) in the numerator of equation (2.3) respectively. The absorption corresponds to the second ($\Delta \Sigma_{rg}$) and fourth ($\Delta \upsilon \Sigma_f$) terms in equation (2.3). Since the change in fission cross-section is usually small and neglected, this component is referred to as the capture component. Sodium voiding has strong spatial dependence i.e. if the voiding is in the central part of the core, the coefficient is positive, where as if it is in the periphery of the core, it is negative. The spectral and capture components are normally high in the central part of the core where the neutron flux and importance function are largest, whereas in the outer part of the core, leakage component is high because of the large flux gradient.

Sodium void coefficient is a function of the ratio of the number of sodium atoms removed to the number of fuel atoms present. The spectral component of void coefficient is generally positive, but is more positive for ^{239}Pu than ^{235}U and ^{233}U. The negative leakage component is generally smaller than the capture and spectral components, but can be enhanced by the choice of geometric configuration (Stacey, 2001). Therefore one of the main safety concerns in large sized FBRs is the possibility for positive sodium void coefficient and this could be offset only by proper nuclear design, which will ensure that other negative reactivity coefficients are dominant.

2.2.3 Structural Effect

Temperature variation can result in change in the number of fuel atoms per unit volume of the reactor as well as in the expansion of fuel and structural materials. The expansion can be in axial/radial direction and distortion in the fuel sub assembly due to core restraints. This produces structural expansion reactivity effect. The reactivity effect due to fuel and structural expansion is highly dependent on the details of the design. For small FBRs, this will give negative feedback (McCarthy et al., 1958; Walter and Reynolds, 1981; Stacey, 2001), but as the size of the core increases the structural expansion effect is likely to be positive.

Doppler coefficient and sodium void reactivity varies for different fuel types. Fuel type which generates hardest neutron spectrum, e.g. U-Pu metal fuel, will have enhanced sodium void coefficient and reduced Doppler feedback (Hamid and Ott, 1993; Tsujimoto et al., 2001; Riyas and Mohanakrishnan, 2008; Harish et al., 2009). There are many studies for U-Pu metal fuel on reduction in sodium void worth a) by decreasing height to diameter ratio of the core (Khalil and Hill, 1991; Hamid and Ott, 1993) b) introducing moderating materials inside the core (Jevremovi et al., 1993; Hamid and Ott, 1993; Tsujimoto et al., 1994; Macdonald, 1996; Tsujimoto et al., 2001) c) introducing sodium plenum replacing axial blanket and d) varying smear density of the fuel (Yu et al., 1986; Matveev et al., 1990; Riyas and Mohanakrishnan, 2008). Most of these designs proved to have penalty on breeding ratio and increase in burnup reactivity (Tsujimoto et al., 2001). The unconventional neutronic designs, such as introducing moderating material,enhances Doppler coefficient, but raises the cost to a greater extent. Some of the moderating materials e.g. ZrH, if not used as fuel pins, can cause the hydrogen dissociation at high

temperatures (Tsujimoto et al., 2001). There has been a concern regarding local power peaking in fuel closer to moderator (Rachi et al., 1997) and was proved in the later experiments (Tsujimoto et al., 2001).

Open literature is scarce on comparative studies of the nuclear performance, specifically breeding ratio, delayed neutron fraction, fuel Doppler feedback and sodium void reactivity, of advanced reactor fuels comprising of all possible fissile (^{235}U, ^{233}U, ^{239}PU)–fertile (^{232}Th, ^{238}U) combination for metal, carbide, nitride and oxide fuel combinations. Such studies will provide very useful comparative data relating to the breeding performance and safety related parameters of various types of advanced fuels. They help in identifying fuels, which have optimum breeding potential and safety characteristics. In the studies that were carried out earlier (Reddy et al., 1977; INFCE-IAEA, 1980), simplistic assumptions like a spherical reactor (Reddy et al., 1977) or a homogenous one of large capacity were made and also a great deal of uncertainty was associated to the data for thorium fuel (INFCE-IAEA, 1980). In the study of Reddy et al. (1977), computational limitations restricted to 2D simulations and the steel volume fractions were kept very low (~15%). Some of the recent studies (Robert, 2007; Riyas and Mohanakrishnan, 2008) mainly focused on the metal, carbide and oxide combination of uranium and plutonium. The nature of reactivity feedback, especially fuel Doppler feedback and sodium void reactivity is an important issue in fast reactors and must be clearly understood in evaluating the safety and stability as well as the design criteria of fast sodium cooled breeder reactor systems. Therefore studies, which will identify fuels with optimum breeding performance and safety characteristics, with more realistic design inputs and detailed modeling, also provide useful data towards establishing feasibility of alternate fuel cycles.

Safety development in fast reactor program always looks for reactor designs to avoid coolant boiling and fuel element failure during unprotected accidents (Cahalan, 1990). This can be translated as introducing inherent safety features in the reactor design. Inherent safety implies that without any human intervention, triggering signals and supply of external energy, by exploiting the laws of nature such as thermal expansion, buoyancy driven flow, and gravity, the system remain in safe shutdown. Inherent safety features are very much important when engineered system such as SCRAM is not working (Dam, 1999). Even though there are sufficient design provisions in fast reactors to preclude the transients (Walter and Reynolds, 1981), usually inherent safety of any reactor design, from neutronic side, is analyzed through accident sequences such as Unprotected Transient Over Power Accident (UTOPA), Unprotected Loss Of Flow Accident (ULOFA) and Unprotected Loss of Heat Sink Accident (ULOHA). The term 'unprotected' implies the unavailability of control rods that represents the absence of SCRAM.

Usually control rod worth requirements are low in metal fuelled reactors due to the reason that high thermal conductivity of the metal keeps the operating temperature of the fuel low. This results in low Doppler reactivity to be over come upon start up, yielding a reduced control-reactivity requirement. Since the fertile to fissile conversion ratio is high in metal fuelled cores, reactivity loss associated with burnup is also low, so that less reactivity has to be vested on control rods. Several analyses of UTOPA, ULOFA and ULOHA sequences has been carried out, earlier, in large sodium pool type reactors fuelled with metal fuel (U-Pu type) and oxide fuels (Cahalan, 1986; Cahalan et al., 1990; Royl et al., 1990; Yokoo and Ohta, 2002; Harish et al., 2009; Sathiyasheela et al., 2011). From these studies, it has evolved that, depending on the fuel type used in reactors,

certain accidents will be more significant in projecting the safety characteristics of the core. For example in metal fuelled reactors, since the control rod worth requirements are low compared to ceramic fuelled ones, UTOPA incidents are less severe than ULOFA (Cahalan et al., 1990).

So far, no studies have been done quantitatively on thorium based metal fuel, to look on the favorable reactivity feedbacks and the time available for the decay heat to be removed through passive cooling system.

2.3 Safety Consideration During Fuel Fabrication

Whenever there are operations with fissile materials during fuel fabrication, there are always chances of inadvertent nuclear chain reactions or criticality. The primary objective, in such a situation is to prevent criticality or establishing criticality safety, as neutrons and the associated harmful radiations are a serious threat to personnel and equipments. Criticality safety is an important factor in fuel cycle operations which comprises of mining, extraction and conversion of fuel to the required chemical form, fuel fabrication, reactor operation, spent fuel storage, reprocessing, radioactive waste management and disposal. In nuclear reactors energy production is through controlled neutron chain reactions. Fuel matrix with clad will contain the radioactive fission fragments while shielding will prevent harmful radiations to personnel. Other than reactor operation, rest of the fuel cycle operations will not have any provision for controlling chain reaction and there will not be any shielding and containment. When the fuel to be fabricated is for an experimental reactor, the fissile content requirement may sometimes be higher than that of conventional power reactor fuels. Therefore the risk of criticality is a greater concern in such fuel fabrication plant.

2.3.1 Nuclear Criticality Safety

The primary objective of nuclear criticality safety is to prevent or terminate nuclear chain reaction i.e. criticality and super criticality events thus assuring subcritical operations with fissionable materials outside nuclear reactor a subcritical one (Knief, 1985). The existing criticality safety practice has evolved from several previous accidental criticality events as well as from accident free operations. The statistics of the criticality accidents shows that 20% of these accidents happened in production plants, 10% in working reactors and rest happened in critical facilities (Brandy-Raap et al., 1999).

From 1960 to 1980, large number of experiments was conducted to establish a knowledge base for the safe handling of fissile materials. They served the purpose of generating data to provide general guidelines for nuclear criticality safety. Those general purpose experiments such as Godiva, Jezebel, and Comet (Profio, 1976) were focused on problems associated with highly enriched uranium and plutonium. Experiments were conducted to identify the criticality heights of water reflected nitrate and oxide solutions of plutonium and uranium and their mixtures in different geometries for different concentrations (Fruchard et al., 1965; Paxton, 1973; Paxton, 1978; Lloyd, 1982). Through these experiments, the importance of including the presence of containers, piping and control rods in calculation models has been well understood.

There were also experiments to correlate numbers and dimension for critical arrays of units (Mihalczo, 1963; Johnson and Cronin, 1964; Johnson, 1965). Even if a unit is in a favorable geometry with fissile elements, such unit can go critical in an array due to the strong interaction among the units. The benchmark experiments with well defined

composition, dimension and regular geometry served as the backbone for code development as they can be modeled accurately and hence can be validated.

But every situation of practical interest cannot be validated with experiments. All through these years there has been enormous expertise in code development and with the advent of rigorous computational tools which are validated, one no longer has to depend on expensive experiments for data. With the advent of advanced computational tools which are validated internationally, one can simulate the actual problem and obtain the results. From computation side there are two ways of approaching a criticality problem i.e. through diffusion theory and transport theory. For criticality safety calculations, transport theory is used extensively through discrete ordinates or Monte Carlo formulation. The earlier codes, which are based on discrete ordinate method, were ANISIN (Engle, 1967) and DTF-IV (Lathrop, 1965). These are 1D code which is well suited for simple geometry of spheres, infinite cylinders, infinite slabs and certain infinite arrays of these shapes. Codes such as MONK (Rushton, 1978) and KENO (Petrie and Cross, 1975) were based on Monte Carlo method and the advantage of these codes is that they can handle combinations of regular and irregular shapes.

Even though these calculations are expensive (i.e. when going to additional mesh points, directions, energy and cross-section groups, cost were increasing as there is a rise in computational time) 3D Monte Carlo codes are comparable in cost to 2D versions based on discrete ordinate method. One of the earlier reports discussing Monte Carlo applications in criticality safety was by Whitesides (Whitesides, 1970).

Criticality safety studies in the fuel cycle facilities have to follow requirements and guidelines given by licensing authorities. These are specific to the type of the fuel, its

fabrication and reprocessing. This is to ensure safety and prevent any possibility for criticality hazards. In the Indian fast reactor programme, current interest is development of metal fuel cycle (Baldev Raj et al., 2005; Chetal, 2009; Devan et al., 2011) and the proposed fuel type for this program is a ternary alloy U-Pu-6Zr with two different weight fraction of plutonium. Criticality safety studies related to metal fuel fabrications are limited. In the study which is carried by Reynolds et al., (1990), the fuel was U-9.3Zr binary alloy. Each problem encountered in the criticality safety of process plants is very unique and therefore one cannot rely on the data available in the literature directly.

2.4 Physics Of Nuclear Reactors

The transport of neutrons and their interaction with matter within a reactor determines the behavior of nuclear reactor. The fundamental task of a reactor physicist is to calculate this rate of neutron reactions i.e. accurate prediction of neutron loses and production at various part of reactor (Stacey, 2001). This calculation requires knowledge of nuclear cross-sections and their energy dependence and of the distribution of neutrons in space and energy throughout the reactor. The simplest and most widely used mathematical formulation of neutron distribution in a reactor is provided by neutron diffusion theory, which is a neutron conservation equation. In this formulation, neutrons are treated as if they are all of one effective speed, the medium is assumed to be uniform, scattering is isotropic and the cross-sections are averaged over appropriate neutron energy distribution. Neutron diffusion equation removes the neutron direction of motion from consideration and is based on an approximate relationship between the neutron flux and the neutron current i.e., the dependent variable is the total flux at each energy (Case et al., 1953; Bell and Glasstone, 1970; Duderstadt and Hamilton, 1976; Duderstadt and

Martin, 1979; Ragheb, 1982; Lamarsh, 1983). The diffusion theory expression for neutron current on a differential volume element is given as

$$\frac{\partial N}{\partial T} = S + v\Sigma_f\phi - \Sigma_a\phi - \Delta \bullet J = S + v\Sigma_f\phi - \Sigma_a\phi - \Delta \bullet D\Delta\phi \rightarrow (2.4)$$

Σ_a = Macroscopic absorption cross-section

Σ_f =Macroscopic fission cross-section

D= Diffusion constant

J= Neutron current

The above equation states that the time rate of change of the neutron density within a volume is equal to the rate at which neutrons from external source (*S)* are produced and from fission ($v\Sigma_f$) minus the rate at which neutrons are lost by absorption ($\Sigma_a\phi$) and leakage of neutrons out of the volume ($\Delta \bullet J$). Neutron diffusion equation fails especially in highly absorbing media, when the scattering is highly anisotropic and near the boundary interfaces.

2.4.1 Boltzmann Neutron Transport Equation

The mathematical description of the neutron distribution in space and angle in a reactor is provided by Boltzmann neutron Transport Equation (BTE), which is again a neutron conservation equation (Case, 1953; Bell and Glasstone, 1970; Duderstadt and Hamilton, 1976; Duderstadt and Martin, 1979; Lamarsh, 1983). Certain assumptions are made to derive BTE such as a) neglect of decay of neutrons, b) neutrons are treated as a classical interacting particle, c) the interactions between neutrons are neglected and d)

neutrons travel in straight line after collision. The reliable prediction of neutron interaction rate comes from the solution of BTE. The most common formulation of BTE is the integro-differential form which is given below.

$$\frac{\partial N(r,\Omega,t)}{\partial t}drd\Omega = v(N(r,\Omega,t) - N(r+\Omega dl,\Omega,t))dAd\Omega + \int_0^{4\Pi} d\Omega'\Sigma_s(r,\Omega'\to\Omega)vN(r,\Omega',t)drd\Omega + \frac{1}{4\pi}\int_0^{4\Pi} d\Omega'\upsilon\Sigma_f(r)vN(r,\Omega',t)drd\Omega + S_{ex}(r,\Omega)drd\Omega - (\Sigma_a(r) + \Sigma_{s(r)}))vN(r,\Omega,t)drd\Omega \to (2.5)$$

$N(r,\Omega,t)drd\Omega$ = Number of neutrons in the volume element dr at position r, moving in the cone of direction $d\Omega$ about direction Ω

Σ_a = Macroscopic absorption cross-section

Σ_f =Macroscopic fission cross-section

Σ_s = Macroscopic scattering cross-section

υ =Average no of neutrons produced in fission

dl = vdt, where v is the neutron speed

S_{ex} = external source of neutrons

dA = cross-section area surrounding the direction of neutron motion

The rate of change of $N(r,\Omega,t)$ within the differential volume is equal to the rate at which neutrons with direction Ω are flowing into the volume element less the rate at

which they are flowing out of the volume element, plus the rate at which neutrons traveling in direction Ω are being introduced into the volume by scattering of neutrons within the volume element from different directions Ω' and by fission, plus the rate at which neutrons are being introduced into the volume element by an external source S_{ex}, minus the rate at which neutrons within the volume element traveling in direction Ω are being absorbed or being scattered into a different direction Ω'.

2.4.2 Neutron Transport Calculation

Neutron transport problems can be either fixed source problems or criticality. In fixed source problems, a known fixed source is imposed on the system and determines the resulting neutron distribution through out the system. This type of approach is very useful in shielding calculations. In criticality calculation, steady state multiplying media is analyzed. The loss terms (like scattering, absorption etc.) and source terms (fission and in-scattering) are proportional to neutron flux in criticality problems where as in fixed source problem; source is independent of the flux. Criticality can be attained with very fine tuning of the geometry as well as composition; one introduces a variable parameter into the equation which can be adjusted until steady state solution is found. Mathematically, this transforms the problem into an eigenvalue problem. The dominant eigenvalue actually corresponds to the effective multiplication factor k_{eff} which is defined as the ratio of number of neutron in two successive generations. Therefore criticality problems are also referred to as k_{eff} eigen value problems.

2.4.3 Solution Methodology

The numerical solution of neutron transport equation can be based on either deterministic (Lathrop, 1972; Carlson and Lathrop, 1968; Engle, 1967; Rhoades and Childs, 1988; Rhoades and Childs, 1991) or stochastic methods. In deterministic methods, phase space parameters are first discretized and numerical methods are then employed to solve the problem. Stochastic method, commonly known as Monte Carlo approach (Breismeister, 1994; Halbleib et al., 1992), is based on direct simulation of the interaction of neutrons with matter by probabilistic laws and employs statistical tools to obtain the expected values of the quantities of interest. Both approaches introduce distortions in the physical problem. In deterministic methods, it is the loss of information about the parts of phase space, neglected when discretizing by applying boundary conditions (Bell and Glasstone, 1970; Duderstadt and Hamilton, 1976; Duderstadt and Martin, 1979; Lewis and Miller, 1984), while in Monte Carlo method, it is the statistical uncertainty arising from insufficient number of simulated particles. Reduction of errors in both methods takes more computation time. Even though deterministic method is capable of producing reasonable results in a much less time when compared to stochastic method, the latter is considered more reliable for irregular geometries when sufficiently low statistical uncertainty is achieved. In the present thesis, MCNP code based on Monte Carlo computational methodology is used for the evaluation of neutronic parameters such as k_{eff}, rate of reactions, Doppler and sodium void reactivity.

2.5 Nuclear Data

A complete and validated set of nuclear data is required for estimating various reactor parameters as close to the actual situation as possible. The evaluated nuclear data file (ENDF-B/VI.7) system is used.

2.6 The Dynamics Of Fast Reactor

It is of importance to obtain an understanding about the temporal variation of neutron population (or power) and reactivity, towards a planned change in reactor conditions or to unplanned and abnormal conditions (Hetrick and Weaver, 1966; Weaver, 1968). Delayed neutrons play a very important role in the dynamics of a reactor and are determined primarily by the nature of emission of delayed neutrons (Keepin, 1965). Total delayed neutron emission, ν_d, depends on the type of isotope undergoing fission and the energy of the neutrons causing fission. There are six effective groups of delayed neutron precursor fission products, where each group is characterized by decay constant λ_i and a relative yield fraction β_i / β. The fraction of the total fission neutrons that are delayed is ν_d / ν. The dynamics of thermal and fast reactors are governed by the kinetic equations which are identical for both the systems. The delayed neutron precursors satisfy the following equation (Stacey, 2001).

$$\frac{\partial \hat{C}_i}{\partial t}(r,t) = \beta_i \upsilon \Sigma_f(r,t)\phi(r,t) - \lambda_i \hat{C}_i(r,t),\ i = 1,....,6 \rightarrow (2.6)$$

The one speed neutron diffusion equation (2.4) can be written as

$$\frac{1}{v}\frac{\partial \phi}{\partial t}(r,t) - D(r,t)\Delta^2\phi(r,t) + \Sigma_a(r,t)\phi(r,t) = (1-\beta)\upsilon\Sigma_f(r,t)\phi(r,t) + \sum_{i=1}^{6}\lambda_i \hat{C}_i(r,t) \rightarrow (2.7)$$

By taking in to account that a fraction, β, of the fission neutrons are delayed and there is a source of neutrons due to the decay of the delayed neutron precursors.

Assuming separate variable solution,

i.e. $\phi(r,t) = vn(t)\psi_1(r)\,;\, \hat{C}_i(r,t) = C_i(t)\,\psi_1(r) \rightarrow (2.8)$

Where $\psi_1(r)$ is the fundamental mode solution of $\Delta^2\psi_n + B_g^2\psi_n = 0$

B_g is the geometric buckling appropriate for the reactor geometry. Substituting equation (2.8) in equation (2.7) leads to point kinetic equations.

$$\frac{dn(t)}{dt} = \frac{\rho(t)-\beta}{\Lambda}n(t) + \sum_{i=1}^{6}\lambda_i C_i(t) \rightarrow (2.9)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda}n(t) - \lambda_i C_i(t)\,, \qquad i = 1,....,6 \rightarrow (2.10)$$

Where Λ is the mean generation time between the birth and absorption leading to another fission and is defined as $\Lambda \equiv (v\upsilon\Sigma_f)^{-1}$

$$\rho(t) \equiv \frac{\upsilon\Sigma_f - \Sigma_a(1+L^2B_g{}^2)}{\upsilon\Sigma_F} \equiv \frac{k(t)-1}{k(t)}$$

ρ is the reactivity and k is the effective multiplication constant, given as

$$k \equiv \frac{\upsilon\Sigma_F/\Sigma_a}{1+L^2B^2}$$

The point kinetic equation, based on the assumption of constant spatial shape is valid for reactors where the dimensions are only few diffusion lengths (L). Point kinetic approximations are well valid for fast reactors, where the core is tightly coupled, so that neutron flux is separable in space and time. Therefore much of the fast reactor codes are employing point kinetics to determine the dynamics of the reactor.

2.7 Simulation Tools

The major part of the numerical simulations performed within the scope of this thesis is with the Monte Carlo code MCNP.4B (Briesmeister, 1994). Accidental analysis has been carried out with an in-house developed code PREDIS (Harish et al., 1999), which deals with Pre-Disassembly phase analysis.

2.7.1 The Monte Carlo Method

The Monte Carlo method approximates solutions to neutron transport equation via statistical sampling experiments on a computer. One of the main advantages of Monte Carlo method is that, particle transport can be simulated in any arbitrary geometry. It can take continuous energy nuclear data. Deterministic codes solve the neutron transport equation for an average particle, where as Monte Carlo obtains a solution by simulating individual particles and then inferring their average behavior. It is particularly useful for complex problems that cannot be modeled by deterministic codes. Monte Carlo solves the integral transport equation, where the probability distributions governing these events turn out to be the same as the integral transport equation. One of the disadvantages of Monte Carlo method is that, since it is statistical in nature one cannot obtain exact solution to the problem. All results have associated statistical uncertainties. These can be

reduced by increasing the number of histories, which determines the precision of the result. Accuracy of the solution also depends on the code (physics features and mathematical model employed), exact geometrical representation of the system and the user.

When a neutron traverses a material, it gets scattered or absorbed depending on the process cross-sections of the material. The life of a neutron can begin from an external neutron source or from a fission event, and ends with absorption or leaking out of the system. The event from birth to the point where it is lost from the geometry becomes the history of the particle. This will be simulated using random numbers, which produce a random distribution for the quantity to be computed. Since a single particle cannot represent a whole system, a number of histories must be evaluated for the accurate prediction of the solution. Therefore the process is to run for a large number of source particles to obtain a statistically reliable result and the program records the average behavior of the simulated particles.

Particle interactions with matter are simulated using nuclear data libraries, e.g., JEFF, JENDL, ENDF/B, containing cross-section information for all relevant isotopes and processes. This data is then processed into a format appropriate for MCNP with a directory file 'XSDIR' which is directly used by MCNP.

2.7.2 MCNP

MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent Monte Carlo transport code (Briesmeister, 1994). It can be used in several transport modes: neutron only, photon only, electron only or combined neutron/photon transport.The neutron energy regime is from 10^{-11} MeV to 20 MeV for all isotopes and up

to 150 MeV for some isotopes. Capability to calculate k_{eff} eigen values for fissile systems is also a standard feature. The user creates an input file which contains information on the geometry, source neutrons, materials, selection of nuclear evaluations and tallies to be computed. To improve the efficiency of tally, variance reduction techniques can also be used. The reliability of the results can be verified using 10 statistical tests which are related to Tally mean ($\bar{x}$), Relative error (R), Variance Of the Variance (VOV) of the relative error R and Figure Of Merit (FOM).

a. Tally mean, $\bar{x} = \frac{1}{N}\sum_{i=1}^{n} x_i \rightarrow (2.11)$

b. Relative error, R = $\frac{\sigma_x}{\bar{x}} \rightarrow (2.12)$

σ_x is the estimated standard deviation of the mean $\bar{x}$

c. Variance of the variance, VOV = $\frac{\sigma^2 \sigma_{\bar{x}}^{\ 2}}{\sigma_{\bar{x}}^4}$, $\rightarrow (2.13)$

$\sigma_{\bar{x}}^{\ 2}$ is the estimated standard deviation of the mean $\bar{x}$ and σ^2 is the estimated variance in $\sigma_{\bar{x}}^{\ 2}$.

d. Figure of merit, FOM = $\frac{1}{R^2 T} \rightarrow (2.14)$

R is the relative error of the mean and T is the computer time used in running a Monte Carlo simulation.

The Ten Statistical Tests are summarized below.

1. The mean, $\bar{x}$, must exhibit, for the last half of the problem, only random fluctuations as number of particles "N" increases. No up or down trends must be exhibited.

2. The relative error R must be less than 0.1 (0.05 for point/ring detectors).

3. R must decrease monotonically with N for the last half of the problem.

4. R must decrease as 1/√N for the last half of the problem.

5. The magnitude of the VOV must be less than 0.1 for all types of tallies.

6. VOV must decrease monotonically for the last half of the problem.

7. VOV must decrease as 1/N for the last half of the problem.

8. FOM must remain statistically constant for the last half of the problem.

9. FOM must exhibit no monotonic up or down trends in the last half of the problem.

10. The SLOPE determined from the largest scoring events must be greater than 3.

2.7.3 PREDIS

PREDIS is a neutron point kinetic-thermal hydraulic accident analysis code which is used for evaluating power, reactivity feedback and temperature distribution in a reactor core in the pre- disassembly phase during the progression of an unprotected transient. PREDIS has been validated against the European LOFA benchmark problem (Dharmadurai and Singh, 1983) and up to onset of sodium boiling in the BN800

benchmark exercise (IAEA-TECDOC-1139, 2000). The code assumes 2D cylindrical single pin model. The point kinetics model is assumed to be adequate to calculate the neutronics because of tight neutronic coupling in medium sized FBRs. The code uses point kinetics model for calculation of reactor power, where point kinetic equations are solved numerically similar to Runge-Kutta procedure (Cohen, 1958). The predictions of the point kinetics module have been checked against analytical predictions for step change in reactivity (Sharada and Singh, 1990).

The entire reactor is divided into several radial cylindrical rings depending on the flow zoning of the reactor. Different rings have different number of subassemblies and each subassembly has fixed number of pins. All calculations are done for a representative pin in the ring. There is considerable variation of linear heating rate in the axial direction. Hence, the axial height of the core is divided into several axial meshes. The calculations are done for the product of the number of radial rings (meshes) and the number of axial meshes in the reactor.

Steady state linear power and temperature calculations are done for two powers: the nominal power and of a higher power. The temperature differences corresponding to these two powers are converted to temperature and power coefficients. In the steady state condition, fuel, clad and coolant temperatures are calculated using the following equations (Ozisik, 1993)

$$T_f(r,z) = T_{fs}(z) + q(r,z)\,(1 - r^2/a^2)/h_l \rightarrow (2.15)$$

$$T_{fs}(z) = T_{si}(z) + q(r,z)/h_z \rightarrow (2.16)$$

$T_s(r,z) = T_{so}(z) + q(r,z)/h_s \rightarrow (2.17)$

$T_{so}(z) = T_c(z) + q/ h_4 \rightarrow (2.18)$

Where

$T_f(r,z)$ = temperature of the fuel pin along the radial and axial directions

$T_{fs}(z)$ = temperature of the fuel surface at z

$T_{si}(z)$ = clad inner surface temperature at z.

$T_{so}(z)$ = temperature of the clad outer surface temperature at z.

q(r,z) = linear power at mesh (r,z) (W/cm)

$h_l = 4nK_f$ (W/cm.C)

$h_2 = 2nbh_b$ (W/cm.C)

$h_3 = 2nK_s / \ln(c/b)$ (W/cm.C)

$h_4 = 2\pi ch_c$ (W/cm.C)

K_f = fuel thermal conductivity (W/cm.C)

K_s= steel thermal conductivity (W/cm.C)

c= fuel pellet radius (cm)

V= coolant flow velocity (cm/sec)

T_{fc}= fuel centre line temperature (at r=O) (C)

C= clad outer surface radius (cm)

h_b = gap conductance between fuel and clad (W/cm2.C)

C_c=coolant specific heat (J/cm.C)

T_m = melting temperature of fuel (C)

T_{fuel}=average fuel temperature (C)

T_{sat}=sodium saturation temperature (C)

The heat transfer calculations from fuel pin to the coolant is done using lumped heat transfer model (Singh, 1987). It gives one representative temperature each for the fuel, clad and coolant in each mesh. The temperatures in each mesh are calculated by solving the mass-momentum and energy balance equations by finite difference method. The time step chosen is determined by the requirement of the point kinetics calculations (see above). The equations are solved in one phase or two phases, depending on the temperature of the coolant. The lumped model is found to be adequate for accident analysis. Energy balance equations for each channel at a given axial position and for unit length of the fuel pin are given below. Coolant is assumed to be in single phase.

$$C_f\,(dT_f/dt) = q - h_{fs}(T_f - T_s) \rightarrow (2.19)$$

$$C_s\,(dT_s/dt) = h_{fs}\,(Tf - T_s) - h_{sc}\,(T_s - T_c) \rightarrow (2.20)$$

$$C_c\,(dT_s/dt) = h_{sc}(T_s - T_c) - C_c\,.\,v(dT_C/dz) \rightarrow (2.21)$$

The heat transfer coefficients used in the equations (2.15- 2.18) are given by

$l/ h_{fs} = 1/2h_1 + l/h_2 + l/2h_3$

$l/ h_{sc,}. = l/2h_3 + l/h_4$

$h_C = K_C(5 + 0.025 (Re.Pr)^{0.8})/de$

$de = 4A/2\pi c$ = hydraulic diameter= 4*flow area/perimeter

$Re.Pr = (\rho_c * v * de/\mu).(\mu C_p/K_c)$

T_f = average fuel temperature (C)

T_s = average clad temperature (C)

T_C = average coolant temperature (C)

C_f = specific heat of fuel per unit length (W/cm,C)

C_s = specific heat of clad per unit length (W/cm.C)

A= cross sectional area of the flow (cm2)

L= thermal conductivity of the coolant (W/cm.C)

μ = viscosity of the coolant (g/cm.s)

ρ_c = density of the coolant (g/cm3)

The equations (2.19 – 2.21) are solved by the finite difference method. In equation (2.21), the term dT/dZ is the gradient of coolant temperature in the axial direction calculated by taking the difference of coolant temperatures in the successive meshes. The equations are applied to calculate the temperatures in all the meshes in the reactor. Since lumped model

is used for the calculations in the transient, the initial conditions are obtained by averaging the fuel temperature in the pellet, and in the clad. The initial condition for the coolant is the bulk temperature calculated in the steady state.

The removal worth (per/mesh) of fuel, steel (structural material + clad) and coolant along with Doppler coefficients are calculated by first order perturbation method. These removal worth are then converted into the temperature reactivity coefficients and used for calculating the feedback reactivity in point kinetic equation. The net reactivity is a sum of the input reactivity and the feedback reactivity. Feedback reactivity which are of important in the present context are the axial fuel expansion, radial expansion of the core, clad and coolant expansion, Doppler feedback due to broadening of resonances, spacer pad expansion and coolant voiding.

2.8 Summary

Safety considerations of prospective fast reactor metal fuels have been reviewed and identified areas which requires more understanding, particularly, on the feasibility of alternate fuel cycle, evaluation of their safety parameters such as fuel Doppler coefficient and sodium void reactivity for all possible fissile fertile combinations of the fuels, along with their breeding potential. To explore the inherent safety characteristics of thorium based metal fuel, analysis of unprotected transient are required to quantify the various reactivity feedbacks and the time availability for the passive shutdown of the reactor. Safety in the fuel fabrication plant is another very important issue for the operating personnel and equipments. A study is taken up to establish nuclear criticality safety in the various stages of fuel fabrication in the PUMP facility coming up in Kalpakkam. Evaluation of the neutronic parameters such as k_{eff}, breeding ratio, delayed neutron

fraction, Doppler coefficient and sodium void reactivity will be carried out using, MCNP.4B code. The dynamics of neutronic parameters such as power, reactivity feedback and temperature distribution in a reactor is computed with the code PREDIS, which is a point-kinetics-thermal hydraulic accident analysis code.

Chapter 3

Breeding Capability And Safety Related Parameters Of Advanced Fast Reactor Fuels

3.1 Introduction

Current designs of fast reactors in India uses Mixed Oxide (MOX) fuel because of its large scale operational experience, economic competitiveness and well established reprocessing technology. On the other hand metal fuelled fast reactors particularly with U-Pu combination have superior breeding performance, but this fuel type has lower Doppler coefficient and enhanced sodium void reactivity as reviewed in Chapter 2. The main focus of the work presented in this chapter aims to identify fuel types with superior safety characteristics without much penalty on the breeding ratio. Towards this task, fast reactor fuels with all possible fissile (^{233}U, ^{239}Pu) – fertile (^{232}Th, ^{238}U) combination are used to evaluate the breeding ratio, delayed neutron fraction, fuel Doppler coefficient and sodium void reactivity. Since ^{235}U is not a fissile material of interest in breeding scenarios, it is not used in the study.

In FBRs, the core is driven by a seed region of fuel assemblies loaded with 10% - 30% of fissile material (^{235}U/^{239}Pu). Blanket sub assemblies containing fertile isotopes (^{238}U/^{232}Th) are loaded in the radial periphery (Homogenous core) or interspersed in the core (Heterogeneous core). As the fertile isotopes in the core and the blanket region absorb neutrons leaking out of the seed fuel, fissile isotopes are produced through nuclear transmutation given below.

$$^{238}U(n,\gamma)^{239}U \xrightarrow[23.5m]{\beta_-} {}^{239}Np \xrightarrow[2.35d]{\beta_-} {}^{239}Pu \rightarrow (3.1)$$

$$^{232}Th(n,\gamma)^{233}Th \xrightarrow[23.4m]{\beta_-} {}^{233}Pa \xrightarrow[27d]{\beta_-} {}^{233}U \rightarrow (3.2)$$

The fuel assemblies have to be taken out regularly for reprocessing where ^{239}Pu and ^{233}U are separated from fission products. A fast reactor can breed over a broad neutron energy spectrum, but good breeding ratios can be achieved only by selecting appropriate fissile isotope for a given energy spectrum. High breeding gain can be obtained with hard neutron spectrum, as breeding ratio is a function of 'η', which is defined as the number of neutrons produced per neutron absorbed.

An important factor in the safety and stability of fast and thermal reactors during normal operation is the effect of delayed neutrons, even though they constitute a small fraction of the total number of neutrons generated from fission (Keepin, 1965). Reactivity, which is defined as change in k_{eff} from the critical state, introduced into the system due to any perturbation of geometry and physical properties of various components constituting the core can be represented in units based on delayed neutron fraction, β_{eff}. The unit of reactivity 'ρ' is expressed as

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \rightarrow (3.3)$$

Delayed neutron fraction, β_{eff}, in unit of reactivity is

$$\beta_{eff}(pcm) \equiv \left[k_{eff} - k_{eff}(prompt)\right] \times 10^5 \rightarrow (3.4)$$

When the reactivity of the system is same as $\beta_{eff,}$ the unit of reactivity is defined as 1$. This is having profound importance in core safety characteristics. If the reactivity, ρ, exceeds 1$, reactor will be in super prompt critical state. Effective delayed neutron fraction, β_{eff}, is one of the most important kinetic parameters which are used in period reactivity relation to derive the transient response of the system. Due to the increased experimental difficulties in determining the delayed neutron fraction especially for mixed isotope system, this value is often determined from theoretical calculations (Svetozar et al., 2008).

The most important cause of transient changes in an operating reactor which tends to increase or decrease $k_{eff,}$ is due to the variation in the temperature of the system (Lewis, 2008). Negative temperature coefficient of reactivity is desirable and fuel temperature coefficient is very significant in this respect. The neutrons which are being absorbed in the broadened resonances of the fertile isotopes are translated into fuel Doppler feedback. Its shorter time constant makes fuel temperature coefficient an important parameter in reactor operation. The alteration in the composition of the fuel material will also affect the temperature coefficient in a number of ways (Walter and Reynolds, 1981). The reactivity change that occurs in fast reactors with temperature rise of sodium, which manifests as sodium void effect and sodium temperature coefficient, has received intensive study. The reactivity effects occurring during normal operation with sodium temperature coefficient is very small and cannot be counted as a source of instability in fast reactor operation, where as sodium voiding from a reactor core is significant during accident situations (Hummel and Okrent, 1970).

Table 3.1: Description of reference core design parameters.

Fuel pin diameter(cm)	0.66
Fuel column height (cm)	100
Total axial blanket height (cm)	60
Assembly pitch (cm)	13.5
Fuel pins per subassembly	217
Number of Fuel subassemblies in core1/core2/blanket regions	85/96/120
Volume fraction in the core for metal, (fuel /steel /sodium- %)	35/24/41
Volume fraction in the core for oxide, carbide and nitride fuel (Fuel/steel/sodium- %)	33.1/25.9/41
PHWR grade Plutonium composition (%) (^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu)	68.79/24.6/5.26/1.35
Depleted Uranium (%)(^{235}U, ^{238}U)	0.25/99.75
Volume fraction in the radial blanket region (%) (fuel/steel/sodium)	52.3/19.3/28.4
Number of CSR/DSR	9/3

3.2 Reactor Core Configuration

The analysis is done for a reactor of fixed core volume similar to 500 MWe (~1250 MWt) sodium cooled PFBR. The core consists of 181 fuel subassemblies with two enrichment zones. Top view and sectional view of reactor is given in Figures 3.1 and 3.2. The main characteristics of the core are listed in Table 3.1

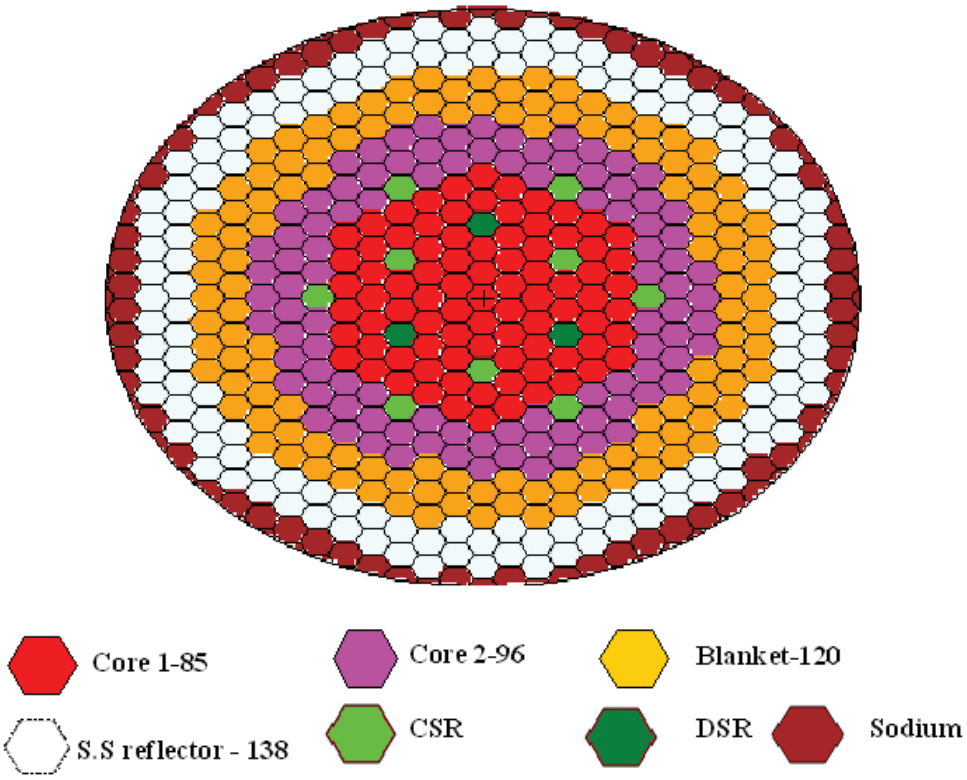


Figure 3.1: Top view of the 500 MWe reactor.

Based on the safety criteria, the reactor is designed to have two independent and diverse reactor shut down systems. First system consists of a set of 9 rods called Control Safety Rods (CSR) and second system consists of a set of 3 rods called Diverse Safety Rods (DSR). In the present study, position of control rods is considered to be in the fully withdrawn condition. The fuel types considered are Metal with and without zirconium addition (6 wt% of Zirconium), Oxide, Carbide and Nitride. The fissile isotopes used in this study comprise of PHWR grade Pu and ^{233}U, and the fertile isotopes include ^{232}Th and depleted Uranium. In the blanket, depleted Uranium -Thorium is considered. The blanket is presumed to always have the same fertile material of the same chemical type as present in the core. Smear densities (Matthews, 1993; Riyas et al., 2008; Srinivasan et al.,

2006) and theoretical density of the fuel in core and blanket regions used in the present study are given in Table 3.2 and 3.3.

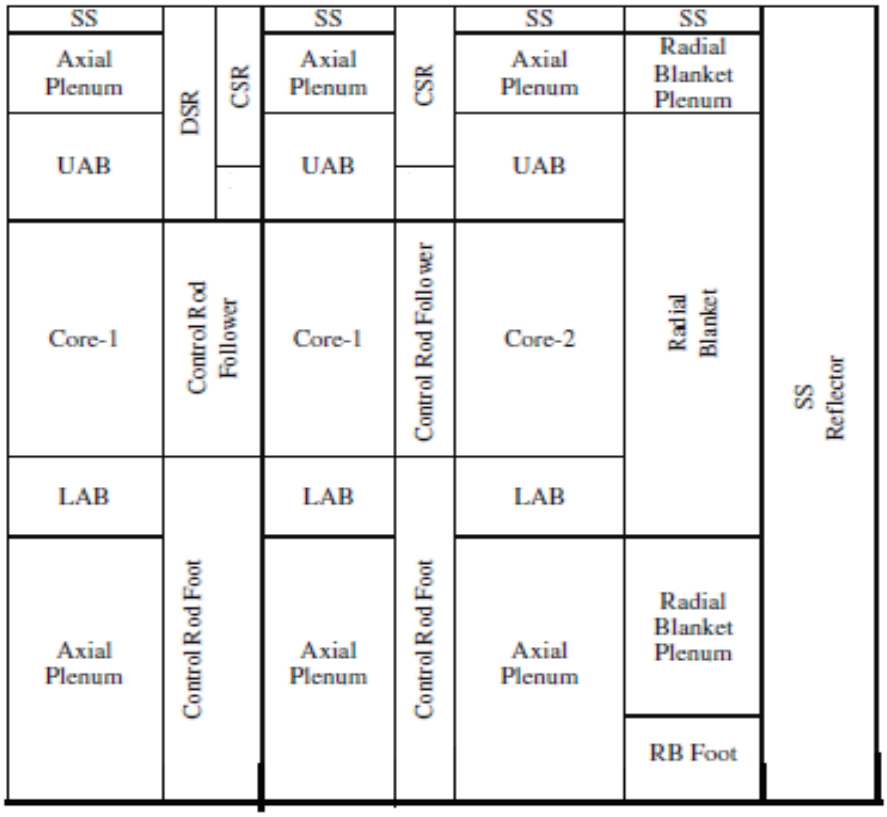


Figure 3.2: Sectional view of the 500 MWe reactor.

The study is restricted to a purely neutronics evaluation of the inherent breeding capability and safety parameters for different fuels for a fresh core. Actual breeding ratio evaluation requires the discharge burnup, losses in reprocessing & re-fabrication of fuel and the type of reprocessing employed for each fuel type. One does not have sufficient data and knowledge of reprocessing and fabrication aspects of these fuels. At this stage,

relative comparison of different fuels is more important. For purposes of comparison, we the study is confined to fresh core. It is expected that the final conclusions would remain the same even if the full fuel cycle aspects are considered. Once a choice, which may also depend on reprocessing and fabrication aspects, is made on the type of fuel, more realistic values of breeding ratios will have to be evaluated. Since the smear density of metal fuel is less than that of MOX, MC and MN, fuel volume fraction is greater for metal fuel. The excess reactivity for all the fuel compositions used in the calculations is approximately 1.047.

Table 3.2: Design parameters - Fuel

Properties	**Fuel type**			
	Metal	**Oxide**	**Carbide**	**Nitride**
Bonding	sodium	Sodium	sodium	sodium
Fraction of theoretical density	0.75	0.83	0.83	0.83
Smear density of the blanket	0.8	0.9	0.9	0.9

The excess reactivity provided in the core will take care of isothermal temperature coefficient, power coefficient and burnup loss of the reactivity. For all fuel types, linear

power is fixed to 450 W/cm. This value is true in the case of oxides as well as metals however for carbides and nitrides this value will vary. Since the thermo physical data of carbides and nitrides are not available to the same degree of accuracy as oxides and metals, their linear power is also kept as 450 W/cm. The enrichments are fixed on the basis of the matching linear power over core 1 and core 2 regions. Enrichments used in the core region for different fuel compositions are given in Table 3. 4.

Table 3.3: Theoretical densities of Fuel.

Fuel Type	Theoretical densities (gm/cm^3)			
	Metal	**Oxide**	**Carbide**	**Nitride**
Uranium	19.05	10.37	13.60	14.30
Plutonium	19.68	11.50	13.58	14.40
Thorium	11.70	10.00	10.60	11.60

3.3 Core Simulations

The detailed modeling of the core is done with MCNP.4B code using recent nuclear cross-section data, ENDF-B/VI.7. Another standard cross-section set JEFF-3.1 is also used to study the effect of different cross-section sets on the breeding potential of the different fuel types. These are also the cross-section sets available with us. Criticality calculations to find the effective multiplication factor (k_{eff}) were done with KCODE card of MCNP. Rate of reactions in the different region of core and blanket are calculated using F4 tally card. To find the delayed neutron fraction, prompt method is used which

requires two k_{eff} calculations, one with TOTNU NO card and the other without this card (Bretscher, 1997; Svetozar et al, 2008). Source uses 10^5 neutrons for each cycle and code ran for 65 cycles skipping first 15 cycles.

Table 3.4: Core design properties - fuel.

Fuel type	Fissile Fertile combination	Enrichment Core1/Core2 (%)	Inpile fissile Inventory (Te)
Metal	Pu*-^{238}U	11.8 / 15	2.259
	Pu*-^{232}Th	18.8 / 24	2.467
	^{233}U-^{238}U	7.4 / 12	1.653
	^{233}U-^{232}Th	13.2 / 19.2	1.760
Metal (6 % Zr alloy)	Pu*-^{238}U	11.4 / 16.2	2.110
	Pu*-^{232}Th	18.2/25	2.302
	^{233}U-^{238}U	7.6 / 12	1.498
	^{233}U-^{232}Th	12.4 / 19.2	1.645
Oxide	Pu*-^{238}U	19.6 / 27.7	2.409
	Pu*-^{232}Th	24.2 / 30	2.730
	^{233}U-^{238}U	12.2 / 16	1.503
	^{233}U-^{232}Th	20.4 / 28	2.422
Carbide	Pu*-^{238}U	16.1 / 20.9	2.432
	Pu*-^{232}Th	23 / 26	2.570
	^{233}U-^{238}U	9.3 / 14.95	1.620
	^{233}U-^{232}Th	13.9 / 21.7	1.757
Nitride	Pu*-^{238}U	15.5 / 20.9	2.563
	Pu*-^{232}Th	21.8 / 26	2.830
	^{233}U-^{238}U	9.9 / 14.95	1.747
	^{233}U-^{232}Th	13.9 / 20.2	1.989

*PHWR grade plutonium

Relative error in each simulation is less than 25 pcm. Modeling framework is given in Figure 3.3.

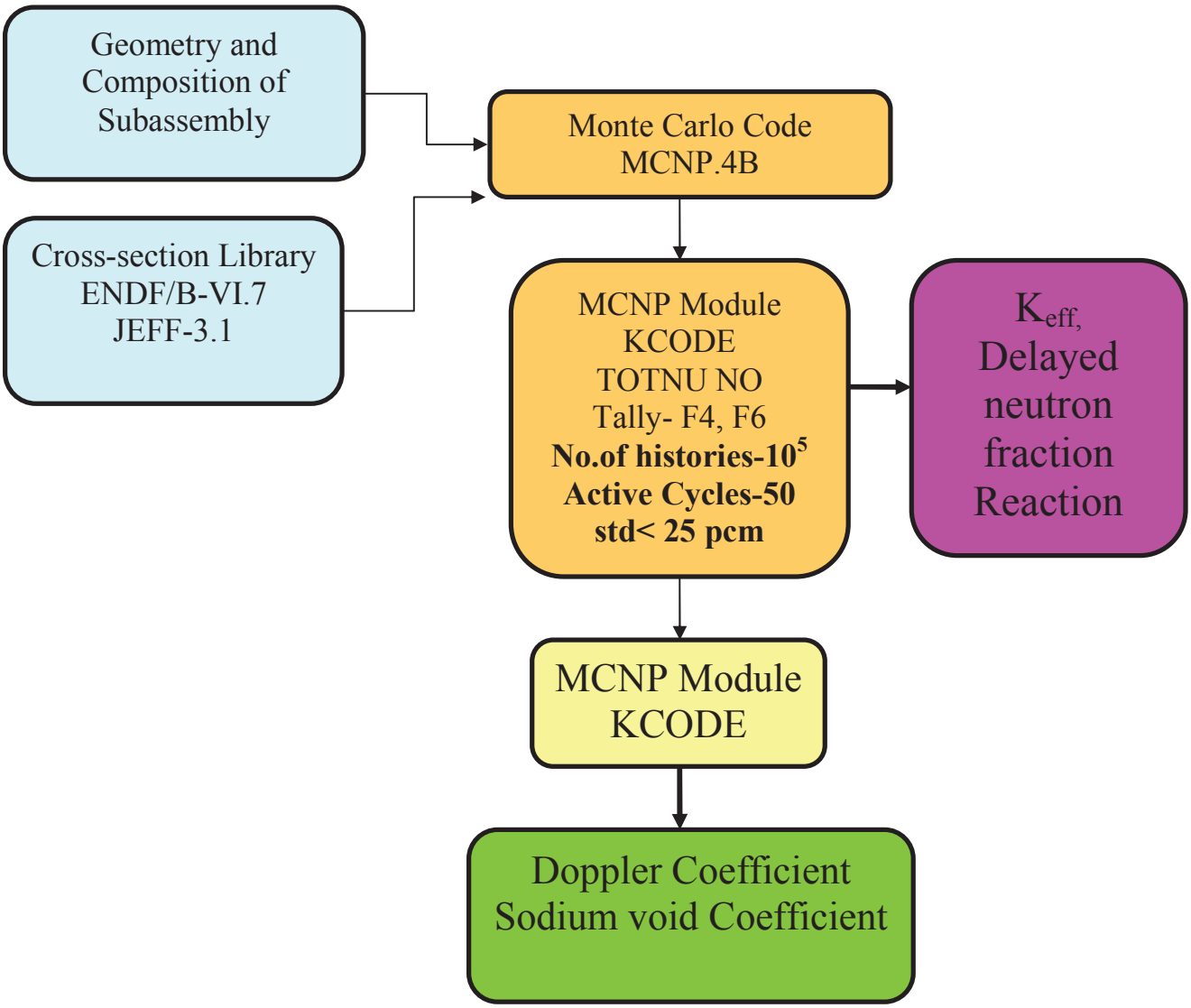


Figure 3.3: Modeling Framework in MCNP code.

3.4 Breeding Ratio

Breeding ratio (B.R) is defined as the ratio of production rate of the fissile isotopes (^{233}U, ^{239}Pu and ^{241}Pu) to their rate of destruction. For the fresh fuel, if the core and blanket regions are divided into a number of cells, B.R can be evaluated for each cell as follows

$$B.R = \frac{\Sigma_{fertile}\overline{\Phi}_V}{\Sigma_{fissile}\overline{\Phi}_V} \rightarrow (3.5)$$

Where $\overline{\Phi}_V = \frac{1}{V}\int dE \int dV \int \Psi(r, \hat{\Omega}, E) d\Omega$ is the flux averaged over a cell volume (Briesmeister, 1994).

Reactor dependent worth of fissile isotopes should be taken into account for calculating breeding ratio, especially when the core has undergone sufficient burnup, to get a more realistic value (Andrianov, 2008). The relative worth of these isotopes depends on the type of the reactor as well as its operating history from which they were discharged. In the present study, since fresh core is used, no attempt is made to differentiate the worth of fissile isotopes. For the B.R evaluation, apart from using ENDF/B-VI.7, European cross-section library JEFF - 3.1 (JEFF-3.1, 2005) is also used for a comparison with respect to cross- section.

The breeding ratio values obtained with ENDF/B-VI.7 and JEFF-3.1 cross-section sets are given in Table 3.5. The values obtained with both cross-section sets are in good agreement. The trends in B.R values are similar to the previous studies where metal fuel shows the highest breeding ratio followed by carbides, nitrides and oxides, but magnitude wise lower values are obtained in the present study. The highest breeding ratio for

metallic fuel is due to the hardest neutron spectrum generated in the core, since the fuel density is high as there is no lighter atom present in the fuel type. This can be seen from the internal breeding ratio given in Table 3.5.

Table 3.5: Comparison of breeding characteristics of advanced fast reactor fuels.

Fuel type		Breeding ratio ENDF/B-VI.7			Breeding ratio JEFF-3.1		
		Internal B.R	External B.R	Total B.R	Internal B.R	External B.R	Total B.R
Metal	U-Pu	1.21	0.359	1.57	1.23	0.35	1.58
	Th-Pu	.890	0.330	1.22	0.91	0.33	1.24
	U-U	1.150	0.320	1.47	1.15	0.33	1.48
	Th-U	0.685	0.268	0.953	0.670	0.302	0.972
Oxide	U-Pu	0.807	0.283	1.09	0.821	0.290	1.11
	Th-Pu	.0644	0.209	0.853	0.661	0.212	0.873
	U-U	0.764	0.256	1.02	0.785	0.275	1.05
	Th-U	0.631	0.295	0.946	0.679	0.259	0.938
Carbide	U-Pu	0.95	0.35	1.30	0.961	0.349	1.31
	Th-Pu	0.709	0.331	1.04	0.723	0.327	1.05
	U-U	0.822	0.298	1.12	0.848	0.302	1.15
	Th-U	0.675	0.415	1.09	0.692	0.438	1.13
Nitride	U-Pu	0.932	0.298	1.23	0.941	0.299	1.24
	Th-Pu	0.691	0.292	0.983	0.702	0.296	0.998
	U-U	0.794	0.266	1.06	0.815	0.265	1.08
	Th-U	0.673	0.248	0.921	0.694	0.253	0.947
Metal-Zr (6%)	U-Pu	1.17	0.323	1.49	1.13	0.38	1.51
	Th-Pu	0.822	0.348	1.17	0.834	0.356	1.19
	U-U	0.862	0.318	1.18	0.886	0.314	1.2
	Th-U	0.646	0.218	0.847	0.475	0.391	0.866

For all fuel types other than carbide, Th-^{233}U combination is giving a B.R value which is lower than one. From the Table 3.5, breeding in the blanket regions is more for Th-^{233}U carbide fuel when compared to other fuel combinations of Th-^{233}U. The reduction in the breeding ratio values in the core region may be due to the increased fast fission threshold of ^{232}Th and low η value of ^{233}U in the fast spectrum. Also the fast fission yield of ^{232}Th is less when compared to ^{238}U (Belle and Berman, 1984). Since η value of ^{233}U is only 2.5 in fast energy region, large amount of ^{233}U is required, as seen from Table 3.4, to achieve the excess reactivity of 1.047. This is also a factor for lower breeding ratio.

Can a larger reactor core with a proper fuel combination of ^{238}U-^{233}U -^{232}Th, give rise to higher breeding ratios has to be studied further taking into consideration not only harder neutron spectrum, but also pin diameter. This is due to the fact that along with harder neutron spectrum, pin diameter is one of the factors which determine the breeding ratio. Harder neutron spectrum is determined mainly by the fuel type. For fast reactors, if fuel inventory has to be minimized, smaller pin diameter is preferred. This will reduce the breeding ratio. So usually an optimum pin diameter is chosen taking care of fuel inventory and breeding ratio. For large power reactors (e.g. 1000MWe) pin diameter is greater (~8mm) than that of 500 MWe reactors (~6.6 mm). This will increase the fertile fuel fraction and the internal breeding gain, therefore larger breeding ratio. These systems will have shorter doubling times.

Th-Pu oxide and Th –Pu Nitride combinations also give lower breeding ratio. Apart from the smaller fast fission cross-section of ^{232}Th, softer neutron spectrum in the oxide core and the absorption of neutrons by ^{14}N in the nitride fuel may be the reason for

this reduction in the breeding. The low internal breeding ratio of fuel combination also causes the excess reactivity requirements for such system to be high. ^{233}U-^{238}U system gives good breeding ratio values for metal (with and with out zirconium alloy), carbide, nitride and oxide, even though it is lower than ^{238}U- Pu combination. These two results favor the use of thorium directly as a fast reactor fuel or as a means of supplying ^{233}U to thermal as well as fast reactors in the third phase of the Indian nuclear energy program.

3.5 Effective Delayed Neutron Fraction

Traditionally β_{eff} is calculated by taking the ratio of the adjoint and spectrum weighted delayed neutron production rate to the adjoint- and spectrum-weighted total neutron production rate (Keepin, 1965). The estimation of β_{eff} is done in MCNP code using two eigen values (Bretscher, 1997). This method eliminates the requirement of adjoint flux which is used to weight the neutron spectrum to estimate the delayed neutron production rate.

To evaluate the effect of delayed neutrons on reactor transient behavior (β_{eff}), the quantity to be computed is the delayed neutron fraction which is effective in leading to fission. As a first step, one should determine how many delayed neutrons are generated. Without an external source, total neutron production rate by fission is

$$P = \int \upsilon(E)\Sigma_f(r', E, \Omega)\Phi(r', E, \Omega)dEd\Omega dr' \rightarrow (3.6)$$

Here E, r' and Ω are the energy, position and solid angle of neutrons, Φ is the neutron flux, Σ_f is the macroscopic fission cross-section of the material at position r' and υ the average neutron multiplicity per fission. Delayed neutron production rate, P_d, can be obtained by replacing the factor $\upsilon(E)$ by $\upsilon_d(E)$, the average delayed neutron

multiplicity per fission. The ratio of P_d, to P is the delayed neutron fraction β_0. Next step is the evaluation of the number of fissions induced by delayed neutrons, as well as by all neutrons.

One can calculate the same quantity for delayed neutrons only ($P_{d,\text{eff}}$), by replacing $\chi(E')$ by $\chi_d(E')$ and $\nu(E)$ by $\nu_d(E)$.

$$P_{d,eff} = \int \Phi'(r',E',\Omega')\chi_d(E')\upsilon_d(E)\Sigma_f(r',E,\Omega)\Phi(r',E,\Omega)dEd\Omega dr' \rightarrow (3.7)$$

When one takes the ratio of $\frac{P_{d,eff}}{P_{eff}}$, one arrives at the Keepin definition of β_{eff}.

Denoting the integral in equation (3.6) and (3.7) as follows

$$\beta_{eff} = \frac{\langle \chi_d \nu_d \rangle}{\langle \chi \nu \rangle} \sim 1 - \frac{\langle \chi_p \nu_p \rangle}{\langle \chi \nu \rangle} \rightarrow (3.8)$$

$$= 1 - \frac{k_p}{k} \rightarrow (3.9)$$

In Monte Carlo scheme, the neutrons are simulated by generating them with a probability that is proportional to *P*. Assessing the effectiveness of these neutrons in generating the 'next' fission, is then straightforward in a Monte Carlo scheme. All neutrons are labeled either 'prompt' or 'delayed' at birth, and subsequently they are tracked through the reactor until they are 'removed' from it either by an interaction such as fission or capture or by escape to the surroundings. In those cases where the removal is due to fission, one needs to check whether the incident neutron is a delayed one or not.

One can then calculate the average number of fissions generated by delayed neutrons, divided by the average number of fissions generated by all neutrons.

Table 3.6: Effective delayed neutron fraction of fast reactor fuels.

	Fuel type	Effective delayed Neutron fraction (β_{eff})
Metal	U-Pu	0.00481 ± 0.00039
	Th-Pu	0.00332 ± 0.00035
	U-U	0.00496 ± 0.00037
	Th-U	0.00348 ± 0.00042
Oxide	U-Pu	0.00381 ± 0.00039
	Th-Pu	0.00324 ± 0.00041
	U-U	0.00434 ± 0.00039
	Th-U	0.00311 ± 0.00048
Carbide	U-Pu	0.00456 ± 0.00044
	Th-Pu	0.00291 ± 0.00038
	U-U	0.00468 ± 0.00043
	Th-U	0.00325 ± 0.00046
Nitride	U-Pu	0.00292 ± 0.00048
	Th-Pu	0.00293 ± 0.00039
	U-U	0.00328 ± 0.00041
	Th-U	0.00298 ± 0.00052
Metal-Zr (6%)	U-Pu	0.00501 ± 0.00038
	Th-Pu	0.00284 ± 0.00048
	U-U	0.00517 ± 0.00046
	Th-U	0.00340 ± 0.00053

Effective delayed neutron fractions which depend upon the fuel type and composition are generated for all possible fissile fertile combinations and are given in Table 3.6. The variation of delayed neutron fraction for various fuel cycle options is not

very significant. Among the metal and other hybrid combinations, nitride fuels have lowest β_{eff} values. The low β_{eff} value observed for a few cases is due to the high enrichment of fissile isotopes in core 1 and core 2 regions, with plutonium (since plutonium has lower β_{eff} value). Even though the Plutonium enrichment is almost same for carbide and nitride fuel, the absorption cross-section of ^{14}N is significant especially for ^{14}N (n, p) ^{14}C (around 1 barn for 300-400 keV) reaction which is affecting the neutron economy and hence a lower delayed neutron fraction for nitride fuels.

3.6 Fuel Doppler Reactivity

The Doppler coefficient of reactivity is a crucial parameter in the evaluation of transients in FBRs. The broadening of reaction cross-sections by an increase in temperature is called the Doppler effect. The Doppler broadening yields an increase in effective cross-sections due to increase in cross-section in the wings. The effect is more pronounced for neutron energies less than 25 keV, because cross-section resonances are predominantly in the low energy region (Hummel and Okrent, 1970). Therefore Doppler effect is lower in cores which has harder neutron spectrum. Even though Doppler effect mainly occurs in fuel, the effective absorption and scattering cross-sections of the coolant and structural materials in the core also increase with temperature and this is referred to as non-fuel Doppler effect. The non-fuel Doppler effect, specifically structural Doppler effect, amounts up to ~ 30% of Doppler effect of the fuel, but is dependent on the temperature profile of the structure with different time constants (Qvist, 2013; Yang, 2011). The characteristics of the feedback due to Doppler effect are determined by neutron energy spectrum and isotopic composition of the materials in the core. While any reactor with predominantly fertile (^{238}U or ^{232}Th) based fuel will have a negative overall coefficient, this is not the case for cores fueled with high fissile content.

In the present study, the main focus is to evaluate fuel Doppler contribution, i.e. the reactivity associated with the resonances of fuel isotopes alone. The aim is to quantify the impact of the neutron spectrum, generated with different fuel types in the core, on reactivity. Doppler defect, $\Delta\rho_{Dop}$ is calculated as the reactivity difference between the normal condition, i.e. Hot Zero Power condition (HZP) and Hot Full Power condition (HFP).

$$\Delta\rho_{Dop}(pcm)= \frac{k_{HFP} - k_{HZP}}{k_{HFP} * k_{HZP}} \rightarrow (3.10)$$

and Doppler coefficient is defined as the change in reactivity per degree change in fuel temperature and is expressed in pcm/ K° (Mosteller, 2007).

$$\frac{dk}{dT} \text{ (pcm/ } K^{\circ}) = \Delta\rho/\Delta T \rightarrow (3.11)$$

Doppler defect and Doppler coefficient evaluation will be more realistic, if the temperatures during operating condition are being used. Since the operating temperature for all fuel combination (with the particular enrichments) are not available in literature, from international experimental experience (Davey and Redman, 1970), temperatures corresponding to zero power and full power condition is taken as 473 K° and 1100 K° for all fuel types. The Doppler defect is computed by simulating successively two criticality calculations (Walter and Reynolds, 1981).

Table 3.7: Doppler defect and Doppler coefficient of advanced fast reactor fuels

		Doppler defect on going from zero to full power, Δρ (pcm)	Doppler Coefficient, $\frac{d\rho}{dT}$ (pcm/ K°)
Metal	U-Pu	-502.57	-0.802
	Th-Pu	-913.89	-1.458
	U-U	-946.30	-1.509
	Th-U	-1145.19	-1.826
Oxide	U-Pu	-893.24	-1.425
	Th-Pu	-1388.11	-2.214
	U-U	-1509.61	-2.408
	Th-U	-2187.36	-3.489
Carbide	U-Pu	-804.73	-1.283
	Th-Pu	-808.66	-1.290
	U-U	-1438.70	-2.295
	Th-U	-2009.89	-3.206
Nitride	U-Pu	-949.24	-1.514
	Th-Pu	-1246.78	-1.988
	U-U	-1334.57	-2.128
	Th-U	-1826.99	-2.914
Metal-Zr	U-Pu	-408.40	-0.651
	Th-Pu	-885.39	-1.412
	U-U	-935.50	-1.492
	Th-U	-1180.39	-1.883

Doppler defect from zero to full power condition is given in Table 3.7. From zero to full power condition, Th-^{233}U fuel for all metal and hybrid combination gives larger negative Doppler defect and the minimum is from ^{238}U-Pu fuel. When compared to ^{238}U-Pu system, the contribution to Doppler defect from Th-Pu system is also higher. The large negative Doppler feedback of Thorium based fuels indicate that the system would be safer and stable during accidental conditions, which has to be estimated with further studies. Larger the Doppler feedback, better would be the stability. But of course, during the normal operating condition the negative reactivity introduced by Doppler feedback must eventually be fully compensated by other systems (generally, engineered systems).

3.7 Sodium Void Reactivity

The reactivity change that occurs when sodium is voided can be computed by simulating successively two criticality calculation, i.e. with and without sodium in the core region (Walter and Reynolds, 1981). The main components contributing towards the voiding effect are spectral, capture and leakage components. To study sodium void reactivity, coolant voiding in core region and control rod region is considered. Sodium void reactivity is computed as follows,

$$\Delta\rho_{Void}(pcm) = \frac{k_{void} - k_{no\ void}}{k_{void} * k_{no\ void}} \rightarrow (3.12)$$

The reactivity contribution of various fuel types during the voiding of sodium from core 1 and core 2 regions is given in Table 3.8.

Among all possible fissile fertile combination, ^{238}U-Pu fuel gives largest positive reactivity during sodium voiding for metal and other hybrid combinations. Th-^{233}U fuel gives the lowest values. The major positive reactivity contribution towards

sodium voiding is due to spectral hardening. The large positive value of sodium void reactivity for U-Pu carbide fuel when compared to U-Pu metal fuel is due to the higher enrichment of Plutonium in the core employing carbide fuel. Spectral component of the sodium void effect is more positive for ^{239}Pu than ^{235}U (Hummel an Okrent, 1970). The variation of fission and capture cross-section of Th, ^{239}Pu and ^{233}U is given in Figures 3.4 and 3.5.

Table 3.8: Sodium void reactivity of advanced fast reactor fuels

	Fuel type	Sodium void reactivity ($)
Metal	U-Pu	5.56
	Th-Pu	2.87
	U-U	1.27
	Th-U	-6.85
Oxide	U-Pu	4.86
	Th-Pu	5.07
	U-U	-3.37
	Th-U	-12.42
Carbide	U-Pu	6.60
	Th-Pu	3.36
	U-U	-1.60
	Th-U	-11.75
Nitride	U-Pu	4.75
	Th-Pu	1.52
	U-U	-2.57
	Th-U	-8.84
Metal-Zr	U-Pu	5.75
	Th-Pu	4.50
	U-U	0.15
	Th-U	-6.98

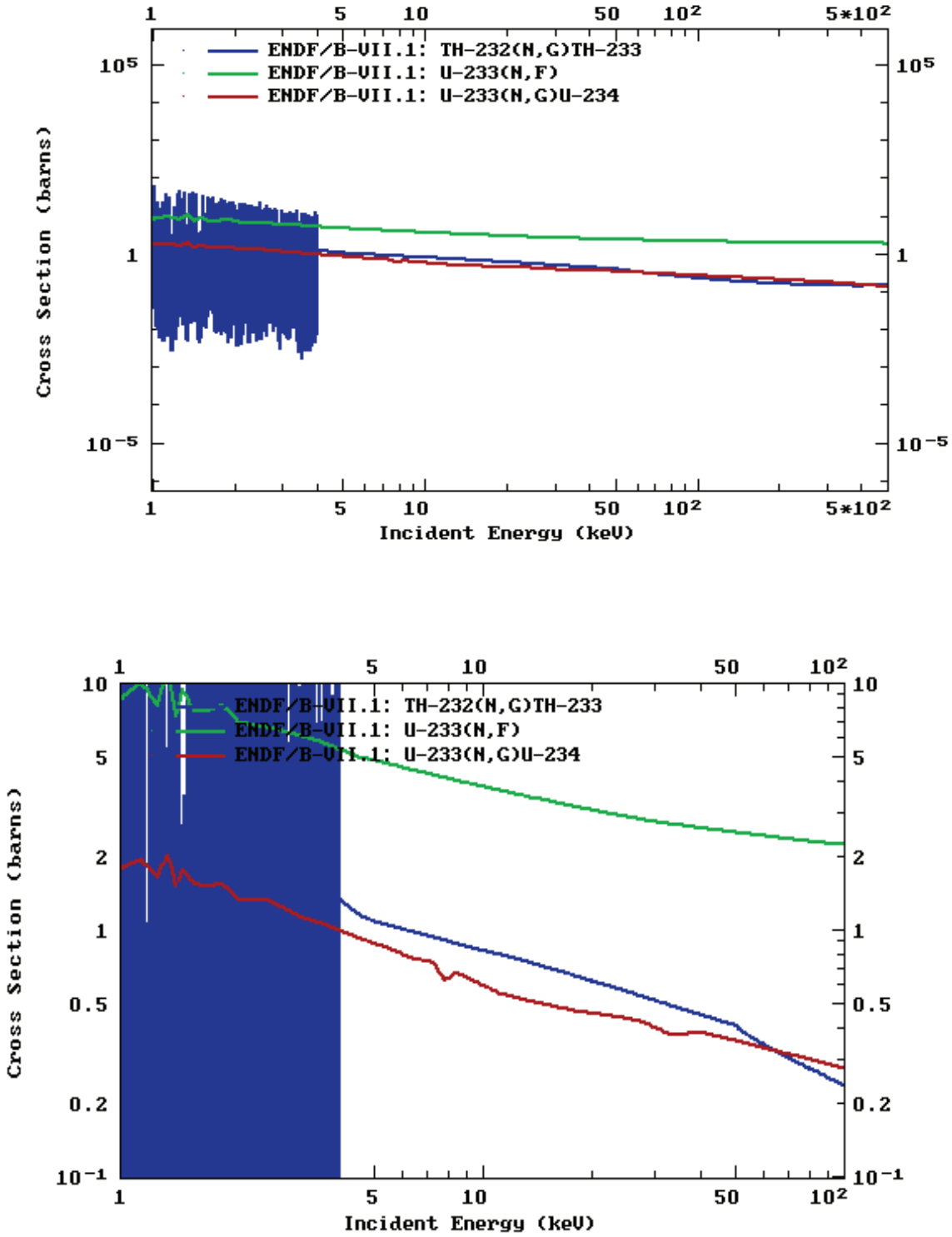


Figure 3.4: Fission and capture cross-section variation of Th and ^{233}U isotopes with energy (https://www-nds.iaea.org/exfor/endf.htm)

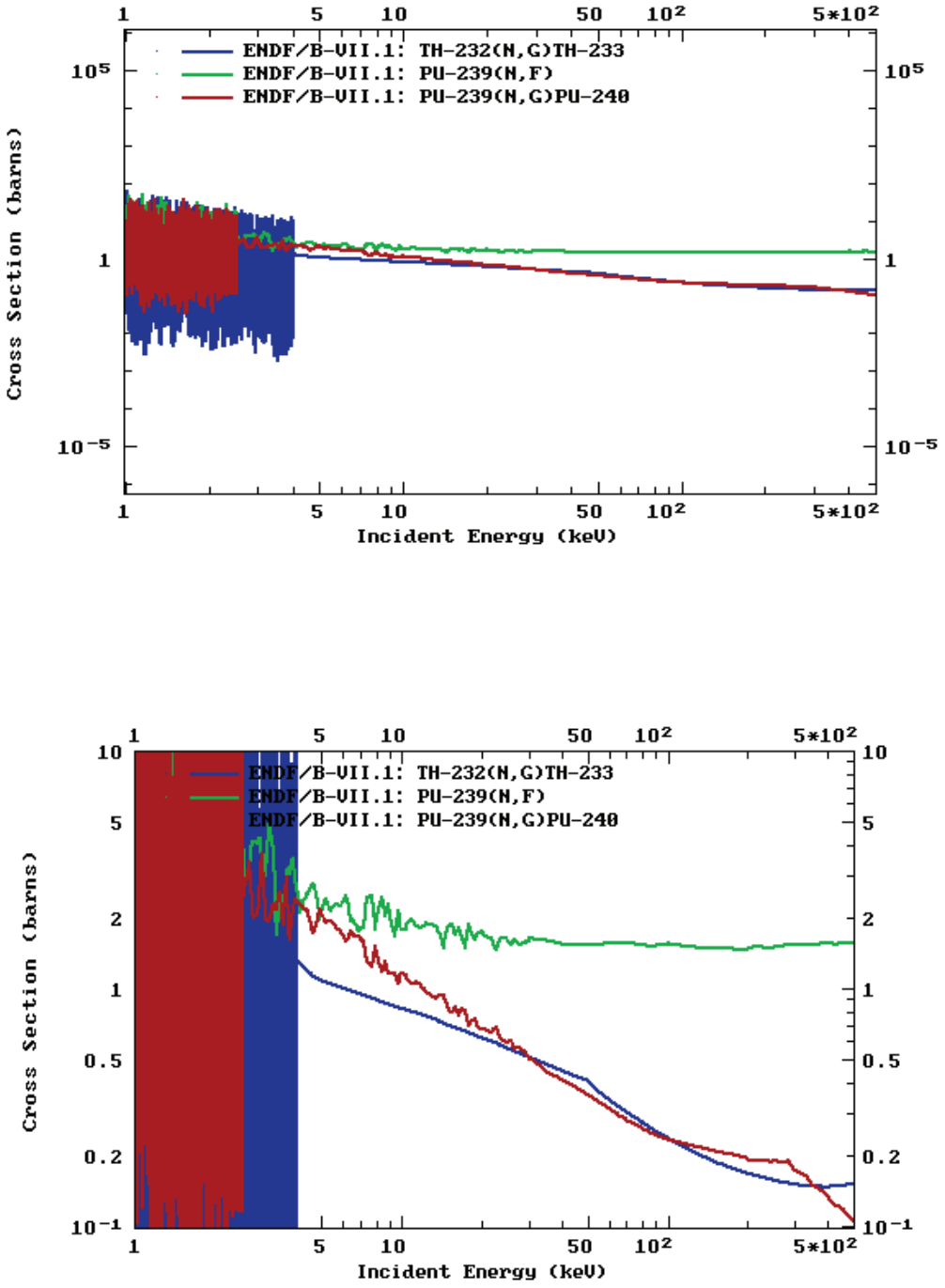


Figure 3.5: Fission and capture cross-section variation of Th and Pu isotopes with energy (https://www-nds.iaea.org/exfor/endf.htm)

For Th-^{233}U fuel types, sodium voiding is giving negative reactivity feedback even for metal combination. The Th-Pu metal combination also exhibits a lower sodium void reactivity compared to ^{238}U-Pu system. The high threshold for fast fission in thorium (than ^{238}U) and the relative flatter η behavior (shown in Figure 3.6) in ^{233}U results in the

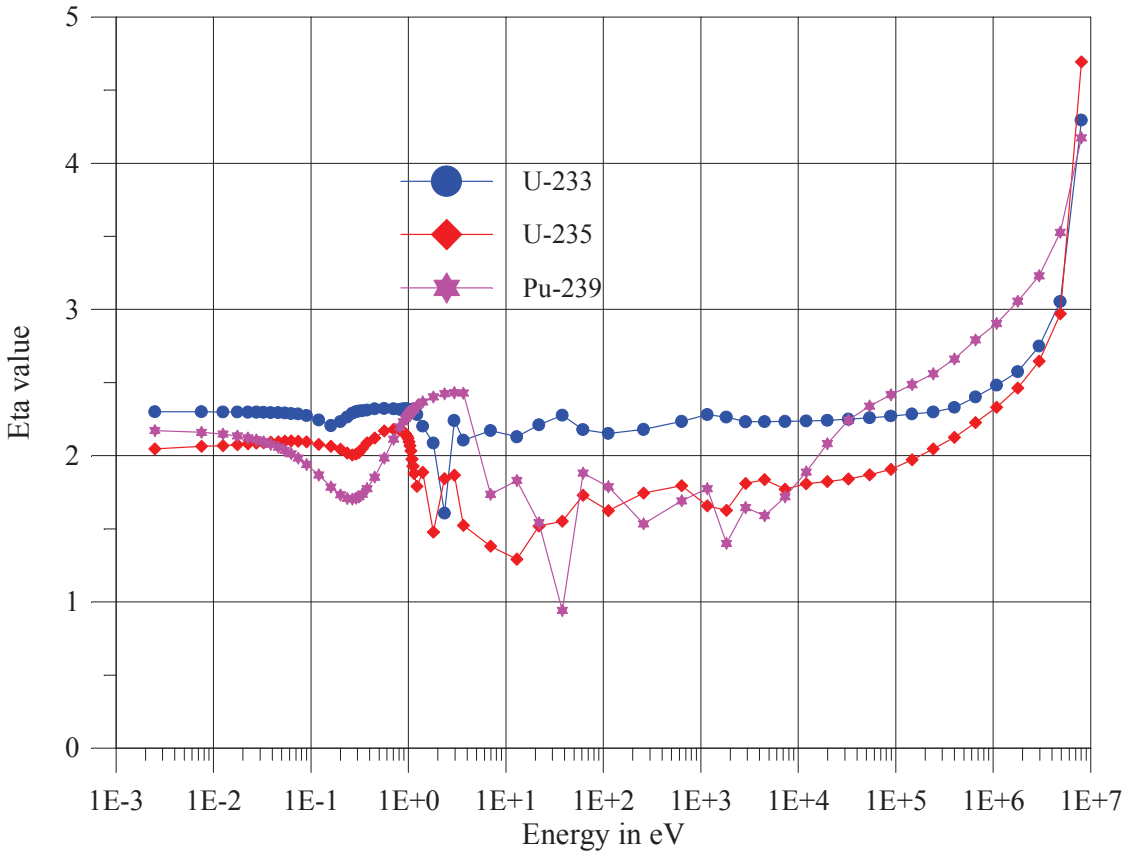


Figure 3.6: Variation of η with energy for ^{233}U, ^{235}U & ^{239}Pu fissile isotopes

lower value of sodium void reactivity in Th- ^{233}U system.

This result shows the big advantage of going towards Thorium based fuels with respect to sodium voiding. ^{233}U- ^{238}U hybrid combination also gives a negative coolant void reactivity. Results are given in Table 3.8. Together with these studies, the contribution of additional reactivity from the absorber material movement and the

changes which arise on the core reaching the equilibrium state must be accounted to get an exact picture of safety.

3.8 Summary

An analysis has been performed on the breeding capability and safety parameters of advanced fast reactor fuels with all possible fertile-fissile combinations using recent cross-section set ENDF/B-VI.7. A comparative study is made on the breeding ratio of advanced fast reactor fuels using JEFF-3.1 cross-section set. The breeding ratios generated using ENDF/B-VI.7 and JEFF- 3.1 cross-section set are in close agreement. In the case of Th-^{233}U system, the core is able to breed for carbide composition where as in the case of Th-Pu combination metal and carbide fuel types breed. Studies have to be done on larger reactor cores to establish further breeding potential of Th-U system.

Safety related nuclear parameters such as effective delayed neutron fraction, Doppler defect and sodium void reactivity were also studied for all possible fissile fertile combination of metal, oxide, carbide and nitride fuel. There is not much variation in the delayed neutron fraction among the different fuel types; the nitride fuels have a lower delayed neutron fraction when compared to carbide fuels, even if the Pu enrichment is almost same in both fuel types. This is due to the significant absorption cross-section of nitrogen for ^{14}N(n, p) ^{14}C reaction.

From zero to full power condition Th-^{233}U system for all metal and hybrid combinations has larger Doppler defect and the least contribution is from ^{238}U-Pu system. In the case of sodium void reactivity, among all possible fissile fertile combinations, ^{238}U-Pu system, gives largest positive reactivity for metal and other hybrid combinations (especially the carbide fuel type) and Th-^{233}U system gives the lowest one. In the case of

Th-^{233}U system, sodium voiding gives negative reactivity feedback even for metal combination. For this fuel system, the high threshold of thorium for fast fission along flat variation of η of ^{233}U in the high neutron energy range lessens the spectral hardening effect and contributes to such a low sodium void reactivity. Among the different metal fuel types, ^{233}U- ^{238}U combination has superior safety characteristics with respect to Doppler coefficient and sodium void reactivity when compared to U-Pu metal fuel.

This study has brought out the impact of fuel cycle on resource utilization with special regard to the use of thorium either as a fast reactor fuel or as a means of supplying ^{233}U to thermal reactors operating in symbiosis. On the basis of breeding potential and safety related performance, it is found that there are better prospects of utilization of thorium resources with hybrid fuel cycles employing Pu-^{232}Th and Th-^{233}U in fast reactors. Once enough ^{233}U have been bred from ^{232}Th, a shift towards ^{233}U-^{238}U cycle would be a preferred choice. The studies show that ^{233}U-^{238}U fuel cycle has the best potential for thorium resource utilization from a physics perspective. It is true that one has to look into technological feasibility, which is beyond the scope of the present thesis. The evaluation of breeding ratio and safety related neutronic parameters for an equilibrium core as well as the management of fissile inventory in a burnt core will be taken up in future studies.

Chapter 4

Unprotected Loss Of Flow Accident In Th-Pu Metal Fuelled 500 MWe Fast Reactor

4.1 Introduction

As discussed in chapter 2, future course of FBRs depends critically on the choice of the fuel that has potential for high breeding so that doubling time is shorter. Therefore the development of metal fuel reactors has assumed importance. Metal fuels have high breeding ratios and therefore the development of metal fuel reactors has been a priority (Baldev Raj et al., 2005; Chetal, 2009). Metal FBR cores having various sizes have been designed to study their neutronics characteristics (Riyas and Mohanakrishnan, 2008; Devan et al., 2011).

As mentioned in chapter 2, section 2.2, in medium and large sized metal fast reactors, especially with U-Pu fuel type; the principal concern is its large sodium void coefficient and reduced Doppler effect (Yokoyama et al., 2005; Riyas and Mohanakrishnan, 2008). Therefore an alternative fissile/fertile combination of metal fuel which has superior safety characteristics of large negative Doppler coefficient and low positive sodium void coefficient without much penalty on breeding ratio is desirable. The studies described in chapter 3, where the breeding potential and safety related parameters of a wide range of advanced fuels with different fertile-fissile combinations were compared, it was found that Th-Pu metal fuel has better safety characteristics related to Doppler coefficient and sodium void reactivity than U-Pu metal fuel. This can be seen in

Table 4.1. The breeding potential of Th-Pu metal fuel is also better than that of U-Pu oxide fuel.

Table 4.1: Comparison of breeding potential and safety parameters of metal and oxide fuels

Fuel type	Breeding ratio (Stephen and Reddy, 2013)	Doppler coefficient, K_D (pcm/ ^{0}K) (Stephen and Reddy, 2013)	Sodium Void Reactivity ($) (Stephen and Reddy, 2013)
U-11.5Pu	1.57	-0.802	+5.56
U-19.6Pu-O	1.09	-1.425	+4.86
Th-18.8Pu	1.22	-1.458	+2.87

These neutronic parameters by themselves do not suffice to explain the inherent safety characteristics of the Th-Pu fuel type. Analysis of unprotected transients in metal reactors is also necessary, as reviewed in Chapter 2. The principle reason that causes the initiation of any transient in FBR's is due to the imbalance in heat generation to heat removal along with the failure of Plant Protection System (PPS) to terminate the neutronic excursion. The imbalance in heat generation to heat removal can come from either Unprotected Transient Over Power Accident (UTOPA) in which uncontrolled reactivity insertion causes the rise of power or from the Unprotected Loss Of Flow Accident (ULOFA) in the core where primary coolant flow is lost and as a result the system can go to super critical state. The frequencies of such accidents are very low (less than 1.0×10^{-6} per reactor year) and analyzing such accidents will provide an insight towards the passive safety characteristics of the reactor. It will be also useful in obtaining

inputs for design measures and source term necessary for planning emergency preparedness.

The assumed initiator for TOPA events are the uncontrolled withdrawal of a single, maximum-worth control rod. In medium and large sized metal reactors, the primary safety concern is the large sodium void coefficient and the reduced Doppler effect resulting from the coolant voiding. For metal reactors with U-Pu fuel, control rod worth requirements are low as reviewed in chapter 2. Usually control rod worth requirements are low in metal fuelled reactors due to the reason that high thermal conductivity of the metal keeps the operating temperature of the fuel low. This results in low Doppler reactivity to be over come upon start up, yielding a reduced control-reactivity requirement. Since the fertile to fissile conversion ratio is high in metal fuelled cores, reactivity loss associated with burnup is also low, so that less reactivity has to be vested on control rods. Several analyses of UTOPA, ULOFA and ULOHA sequences has been carried out, earlier, in large sodium pool type reactors fuelled with metal fuel (U-Pu type) and oxide fuels (Cahalan, 1986; Cahalan et al., 1990; Royl et al., 1990; Yokoo and Ohta, 2002; Harish et al., 2009; Sathiyasheela et al., 2011). From these studies, it has evolved that, depending on the fuel type used in reactors, certain accidents will be more significant in projecting the safety characteristics of the core. For example in metal fuelled reactors, since the control rod worth requirements are low compared to ceramic fuelled ones, UTOPA incidents are less severe than ULOFA (Cahalan et al., 1990).

In this context, it would be valuable to get an estimate of the excess reactivity requirements of the Th-Pu metal core and how the system will behave in a transient scenario especially in Unprotected Loss of Flow Accident (ULOFA). ULOFA analysis provides information on starting and propagation of sodium void, initiation of melting of

fuel, time availability for corrective actions such as opening the damper to initiate SGDHR system ensuring the passive shutdown capability of the reactor. An attempt is also done to identify the differences in the behavior during ULOFA in Th-Pu system and U-Pu-6Zr system of similar capacity.

4.2 Inputs For Design

The analysis is carried out for a fresh reactor core with sodium as coolant and Th-19.3Pu metal fuel as the driver fuel. Core design parameters are same as the design used previously to study breeding characteristics, except for the enrichment of core 1 region and is given in Table 4.1. In the previous study of Stephen and Reddy, 2013, enrichment in core 1 region was 18.8% and in the present study it is assumed as 19.3%. Since the previous study considered wide range of fuel types, the excess reactivity limit was k_{excess}~ 1.047. For the sake of comparison with U-Pu-6Zr metal fuel core of 500 MWe capacity, the excess reactivity in the present study is taken as 1.05 which also happens to be the excess reactivity of the reference core (Riyas and Mohanakrishnan, 2008, Sathiyasheela et al., 2011). Top view and sectional view of the core is same as given in Figures 3.1 and 3.2. The reactivity worth distribution and power densities prior to the start of transient calculation are available from diffusion theory code ALCIALMI which uses ABBN (Devan, 2003; Manturov, 1997) cross-section set. ALCIALMI (Byard, 1965; Giacometti, 1969; Narayanan and John, 2000) is a code developed by CEA Cadarache, to solve the two dimensional neutron diffusion equation. The code has been extensively modified to suit the core neutronic calculations at IGCAR.

4.2.1 Thermo Physical Properties

Thermo physical properties of the fuel, with the enrichments used for the studies are not available in the literature, other than solidus temperature (Peterson, 1990). Physical and mechanical properties of the alloy in most cases are expected to lie between those of the pure components fabricated similarly (Peterson et al., 1965). Wherever thermo physical data of an alloy (which is a solid solution) is not available; estimates are made by linear weighting of data of pure components as this method is often used in the literature (Carbajo et al., 2001), for computation of alloy densities. The solidus temperature of the Th-Pu alloy is about 1630 K and usually the operating temperature will be less than half of the solidus temperature. At this temperature, the Th-Pu alloy forms solid solution for the enrichments considered in the study and the thermo physical property of individual elements, i.e. thorium and plutonium are taken from literature IAEA-THPH, 2008 and is given in Table 4.3.

Table 4.2: Core design properties-fuel

Fuel pin diameter(cm)	0.66
Fuel column height (cm)	100
Total axial blanket height (cm)	60
Assembly pitch (cm)	13.5
Fuel pins per subassembly	217
Number of Fuel subassemblies in core1/core2/blanket regions	85/96/120
Volume fraction in the core for metal, (fuel /steel /sodium- %)	35/24/41
Pu enrichment core-1/core-2 (%)	19.3/24
PHWR grade Plutonium composition (%) (^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu)	68.79/24.6/5.26/1.35
Depleted Uranium (%)(^{235}U, ^{238}U)	0.25/99.75
Volume fraction in the radial blanket region (%) (fuel/steel/sodium)	52.3/19.3/28.4
Number of CSR/DSR	9/3
Total coolant flow rate (kg/sec)	13,670
Type of primary system	Pool type

However it should be mentioned that thermal conductivity is found to be lower in the alloy form when compared to the thermal conductivity of pure elements (Kingery, 1959; Callster and Rethwisch, 2009).

Table 4.3: Thermo physical properties of Th-Pu metal

Fuel density (gm/cc)	12.358
Smeared density (%)	75
Linear expansion coefficient (k^{-1})	13.161
Thermal conductivity (W/mk)	15.675
Specific heat (j/gm/°c)	0.1203
Melting point (k)	1603
Gap conductance ($W/cm^2/k$)	27.02
Boiling point (k)	4619
Latent heat of fusion (J/g)	76.05
Latent heat of vaporization (J/g)	1591

4.2.2 Core Neutronic Parameters

Delayed neutron fraction is computed with the code PERT (ABBN) based on first order perturbation theory, which is a modified version of NEWPERT (John, 1984) and is given in Table 4.4.

Table 4.4: Kinetic parameters for Th-Pu metal core.

j	*1*	*2*	*3*	*4*	*5*	*6*	β
β_j *(pcm)*	7.135	64.094	48.917	81.098	28.477	7.354	237
$\lambda_{j(s^{-1})}$	0.01289	0.030943	0.13259	0.33527	1.36297	3.30454	

To obtain the reactivity worth distribution and power densities prior to the start of transient calculation, reactor core is divided into number of meshes and the removal worth is evaluated by noting the reactivity change when fuel, steel and sodium are removed from each mesh. Doppler worth in each mesh is the reactivity change arising due to the temperature change going from zero power to nominal power (473 K° - 1100 K°). Integral value of Doppler worth along with voiding worth of fuel, steel and sodium is given in Table 4.5.

When compared to U-Pu-6Zr system there is a large difference in the sodium voiding and Doppler worth. For fuel types containing thorium, spectral hardening does not play an important role as thorium has higher energy threshold for fast fission than ^{238}U. Therefore sodium void worth is smaller in Th-Pu system than U-Pu metal core. Doppler feedback is mainly from neutron captures in fertile isotopes occurring below 100 keV. The lower Doppler worth in Th-Pu system when compared to U-Pu-6Zr system is due to the difference in the extent of resonances in fertile isotopes. In the case of thorium, resonances are up to ~4 keV where as in the case of ^{238}U it is extending up to ~20 keV (https://www-nds.iaea.org/exfor/endf.htm) .

Table 4.5: Perturbation reactivity worth in Th-Pu and U-Pu-6ZR 500 MWe metal core.

	Th-Pu Present study (unit in pcm)	U-Pu-6Zr (Sathiyasheela et al., 2011), (unit in pcm)
Fuel worth – core only	-42,848	-37,729
Steel worth– core only	1838	4965
Coolant worth– core only	953.88	2228
Doppler worth– core only	-281.9	-426
Fuel worth – whole reactor	-42921	-37,231
Steel worth– whole reactor	392.04	4190
Coolant worth– whole reactor	434.15	2050
Doppler worth– whole reactor	-325.26	-470

4.3 Core Simulations

The ULOF transient analysis has been carried out using the pre-disassembly phase analysis code PREDIS (Harish et al., 1999). The code uses point kinetic model for evaluation of reactor power. The modeling framework is given in Figure 4.1.

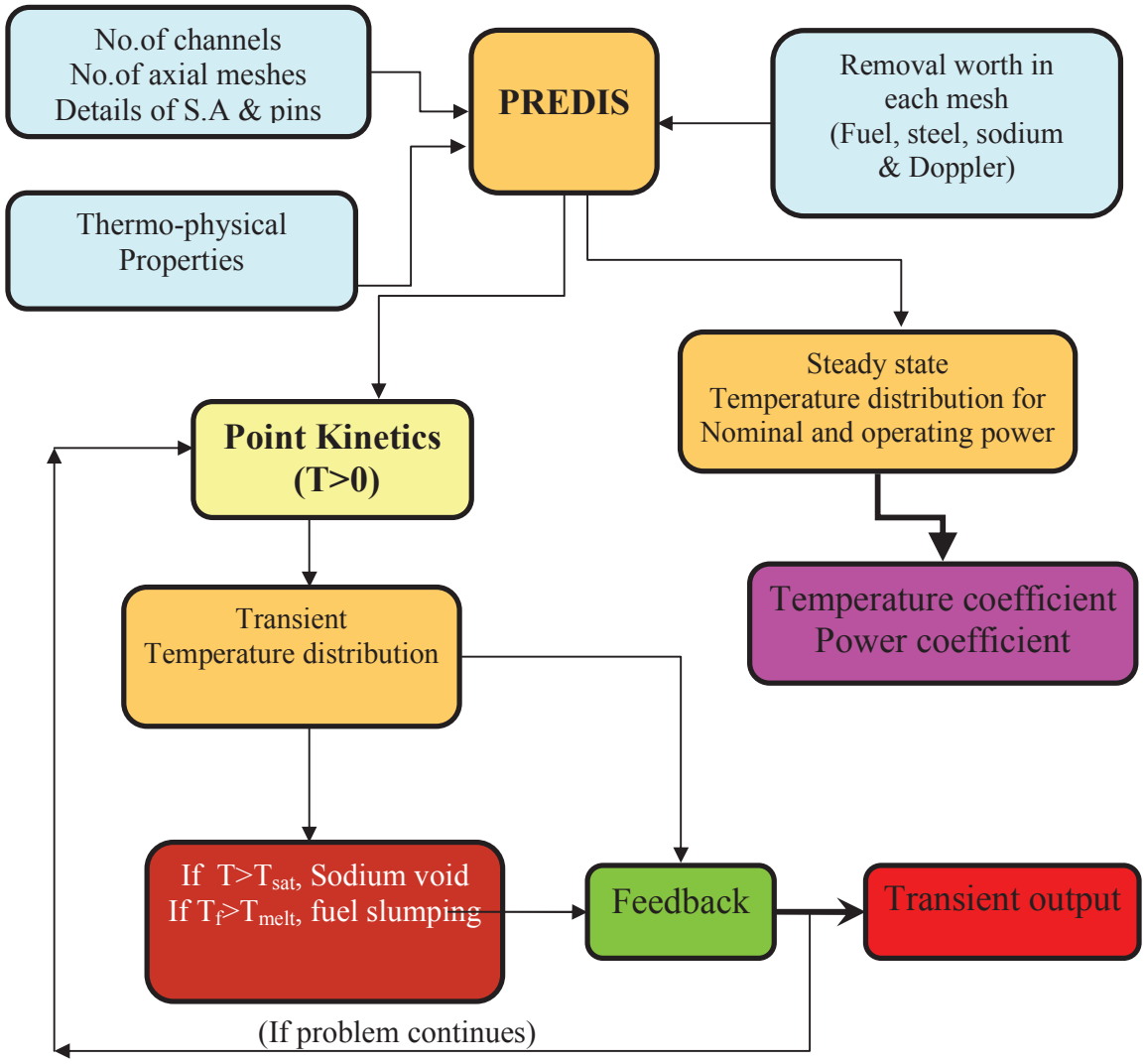


Figure 4.1: Modeling Framework in PREDIS code.

4.3.1 Steady State Analysis

This analysis is carried out to compute isothermal temperature coefficient and static power coefficient. Their magnitudes give indication of excess reactivity requirements in the Th-Pu metal core. The need for large excess reactivity imply larger number of control rods and hence an increased cost of the core as they occupy good locations of the system (Lewis, 2008). Isothermal temperature coefficient is defined as the reactivity change per unit change in temperature when brought from cold critical coolant temperature to the operating coolant inlet temperature.

Static power coefficient is defined as the change in reactivity for a unit change in power keeping all the other reactor parameters like coolant flow and inlet temperature constant between zero and nominal power.

For steady state and transient analyses, core is divided into four radial zones in the inner core, three radial zones in the outer core and three radial zones in the radial blanket based on flow zoning which is similar to the flow zoning in 500 MWe U-Pu-6Zr metal reactor. In the axial direction, the core is divided into 14 zones. The lower and upper axial blankets are divided into two zones each. The magnitude as well as the sign of temperature and power coefficients strongly affects the reactivity control in a power reactor. The factors which affect temperature coefficient and power coefficient other than Doppler coefficient and coolant expansion are fuel axial expansion, clad & sheath axial expansion and spacer pad expansion. Axial expansion coefficient is the reactivity change due to small change in dimension which results in increased dimension and the corresponding reduction in material density.

The enlarged dimension itself tends to decrease the leakage, but the reduced material densities increase the radial leakage. Therefore the net effect is increased leakage, yielding a negative reactivity feedback. The computed isothermal temperature coefficient and static power coefficient are given in Tables 4.6 and 4.7.

The overall isothermal temperature coefficient of Th-Pu metal core is calculated to be about -1.696 pcm/ °C and the power coefficient is -0.183 pcm/ MWt. For U-Pu-6Zr 500 MWe metal reactor, isothermal temperature coefficient is -1.5 pcm/ °C and power coefficient is -0.21 pcm/ MWt (Sathiyasheela et al., 2011). This slightly more negative value of isothermal temperature coefficient in Th-Pu system as compared to U-Pu system is mainly due to the difference in voiding worth, boundary movement worth, Doppler constant, spacer pad expansion and thermal expansion coefficient.

In a fast reactor core, the hexagonal subassemblies, each holding 217 fuel pins, are set in to lower grid plate. There are buttons impressed on wrapper tube through which each subassembly contacts the six surrounding assemblies. The temperature rise of the coolant can result in different temperatures at the six walls of hexagonal fuel assembly duct structure which causes the subassembly to deflect from its original shape and spacer pad starts to touch each other. As the inlet coolant temperature increases, there reaches a particular point where spacer pad cannot expand anymore causing the subassembly to bend so that diameter of the outer core is increasing. This enhances the leakage and contributes to negative reactivity. Fuel worth in Th-Pu system is higher and hence it is expected to give more negative axial expansion feedback.

Table 4.6: Isothermal temperature coefficient of 500 MWe core.

Reactivity component	Reactivity coefficient of Th-Pu core (pcm/^{0}C)	Reactivity coefficient of U-Pu-6Zr core (pcm/^{0}C) (Sathiyasheela et al., 2011)
Doppler	-0.438	-0.624
Fuel Axial expansion	-0.058	-0.504
Clad & sheath axial expansion	0.008	0.084
Coolant expansion	0.124	0.584
Spacer pad expansion	-1.303	-1.039
Total	-1.696	-1.499

But the large boundary movement worth, which is positive in Th-Pu system, overrides the axial fuel expansion feedback and hence it is smaller in Th-Pu system. Coolant void worth of Th-Pu core is about 3-4 times lower than that of the reference case. Steel expansion feedback is also smaller by 10 times than U-Pu system due to the smaller steel removal worth.

In the case of power coefficient, it is less negative than that of U-Pu-6Zr system. This is due to the lower contribution of negative reactivity from fuel axial expansion. Since thermal conductivity of Th-Pu metal fuel is less than that of U-Pu-6Zr fuel type, the temperature seen by the Th-Pu fuel is higher when reactor is taken from zero to nominal power. But the higher negative fuel worth and the positive boundary movement worth makes the overall axial expansion coefficient smaller.

Table 4.7: Static Power coefficient of 500 MWe core

Reactivity Component	Reactivity coefficient of Th-Pu core (pcm/MWt)	Reactivity coefficient of U- Pu-6Zr core (Sathiyasheela et al.,2011) (pcm/MWt)
Doppler	-0.071	-0.093
Fuel Axial expansion	-0.017	-0.079
Clad & sheath axial expansion	0.003	0.008
Coolant expansion	0.015	0.042
Spacer pad expansion	-0.114	-0.092
Total	-0.183	-0.214

Thus these results show that the control rod worth and excess reactivity requirement of Th-Pu metal core will be similar to that of U-Pu-6Zr metal core.

4.3.2 Transient Analysis

The unprotected loss of flow (ULOF) accident, which is assumed as a worst case, is initiated by total loss of offsite power with all the control rods unavailable. This cuts the electrical power to all primary pumps, intermediate-loop pumps, and feed water pumps. The ULOFA analysis has been carried out to quantify the passive safety margin for a representative flow halving time of 8 s. Coolant flow decrement is represented by the standard expression as follows $V(t) = V(0) / (1+ t/\tau)$, where V (0) is the initial flow, t is the time at which flow having is taking place and τ is the flow halving time. At this

stage, response from reactivity feedbacks controls the variation in power.

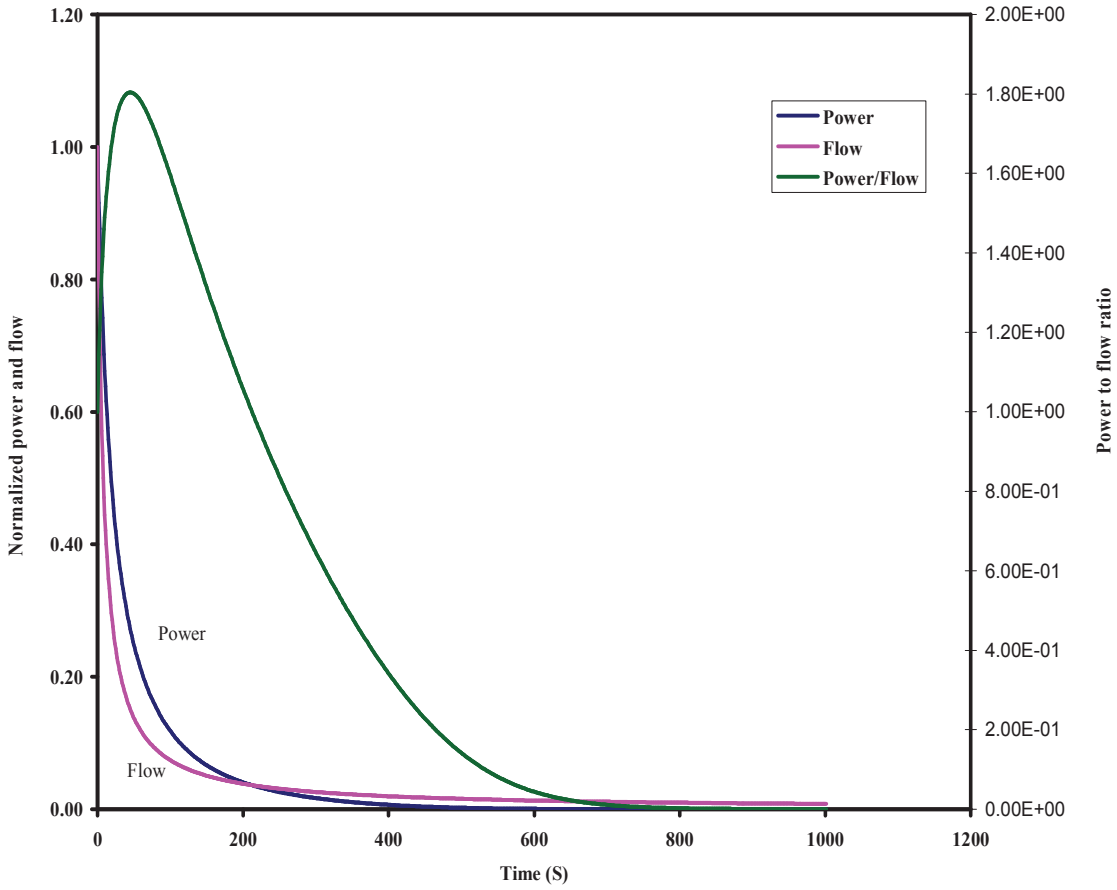


Figure 4.2: Power, Flow and Power to flow ratio during ULOFA in 500 MWe Th-19.3Pu metal reactor.

The variation of power, flow and power to flow ratio with time during the flow coast down is shown in Figure 4.2.

The contribution of reactivity feedbacks are shown in Figure 4.3. The temperature profile of fuel, clad and coolant is shown in Figure 4.4. As the flow decreases, core outlet temperature increases, which provides core radial expansion through expansion of core load pads. This introduces negative feedback reactivity and hence reduction in power. In

all the unprotected transients one of the key rules to avoid core disruptive accident is to maintain coolant outlet temperature below its boiling point (Cahalan, 1986). Since the boiling point of sodium is around 1160 K, to keep the core average sodium temperature below this value, normalized power-to-flow ratio should be below ~ 4 (Cahalan, 1986).

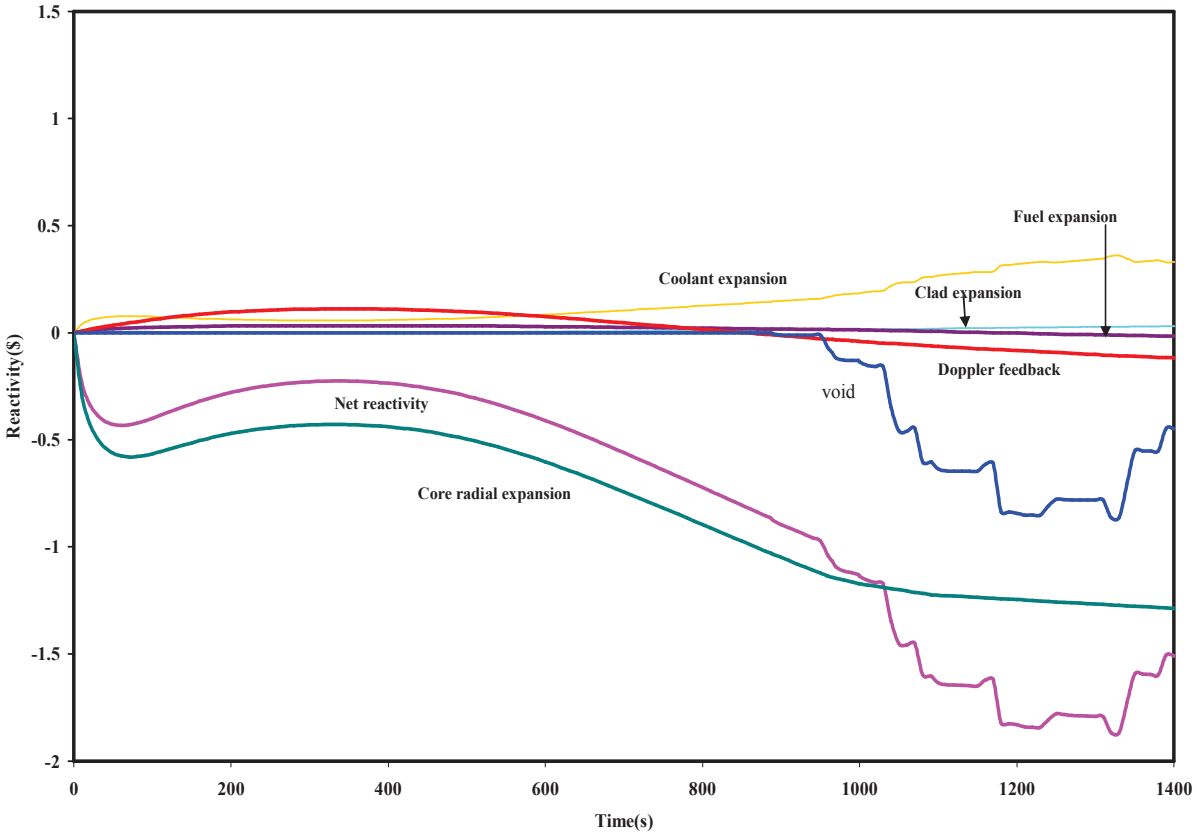


Figure 4.3: Feedback reactivity during ULOFA in 500 MWe Th-19.3Pu metal reactor.

In Figure 4.2, it can be seen that power to flow ratio increases up to 53 s, but it never goes beyond 2 and then decreases continuously. In the case of U-Pu-6Zr fuelled reactors, both power and flow decrease (Harish et al., 2009). However power reduction rate is slower and hence power to flow ratio increases to the value 2.4 in 70 seconds. Thereafter the ratio starts decreasing up to 390 s followed by a continuous increase. The ratio is about 4 at 800 seconds (Harish et al., 2009). This is in marked contrast to the Th-Pu case.

In the present study it is also found that the major negative feedback comes from core radial expansion. The core radial expansion reactivity feedback consists of two seperate effects: the grid plate expansion and the bowing reactivity. The reactivity change from an increase in the core radius has a positive as well as a negative component. The enlarged dimension decreases the radial geometric buckling and tends to decrease the leakage, but the reduced material densities increase the leakage. The combined effect is an increased leakage, yielding a negative reactivity feedback. Power drops due to this and hence temperature also drops between 53 seconds to 340 seconds and then starts rising due to reduction in flow.

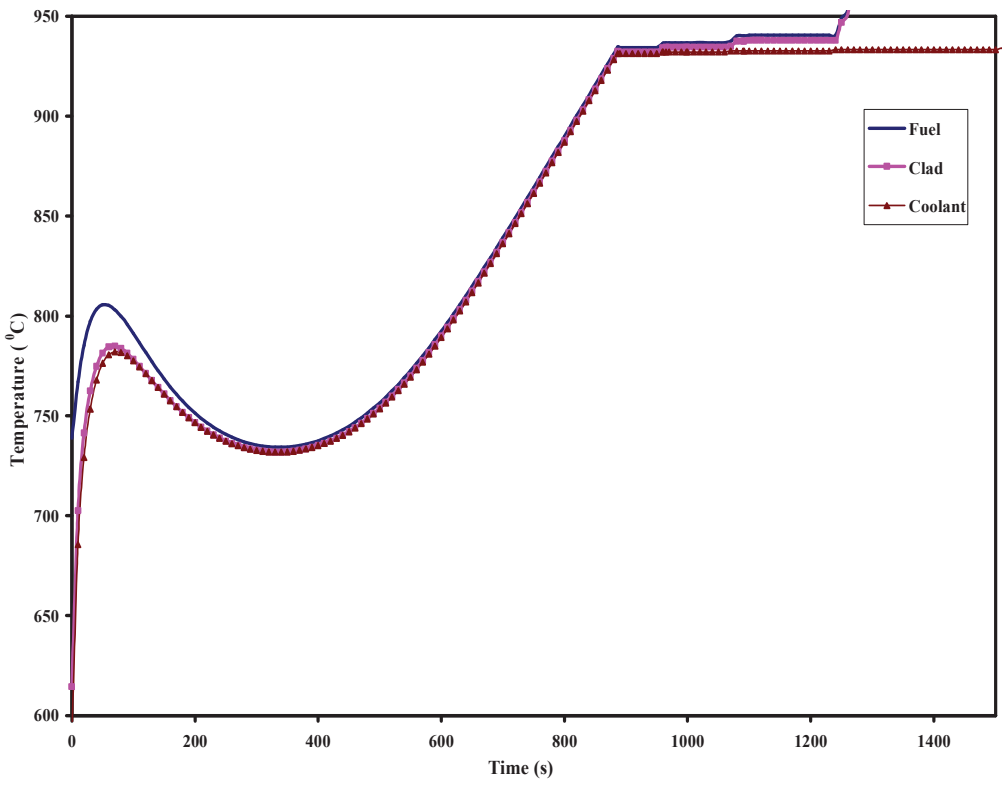


Figure 4.4: Temperature profile of fuel, clad and coolant during ULOFA in 500 MWe Th-19.3Pu metal reactor.

Doppler feedback becomes positive and stays positive up to 886 seconds. Temperature profile of the fuel is given in Figure 4.4. In the case of reference core, Doppler

feedback is negative throughout the transient. The difference in the sign of Doppler feedback in the two systems can be attributed to the enrichments, thermal conductivity and radial expansion feedback. Enrichment with plutonium in Th-Pu core is higher than that of U-Pu-6Zr core. Even though a rise in fuel temperature is observed for first 53 seconds, Positive Doppler feedback during this time period may be due to the broadening of fission resonances of plutonium.

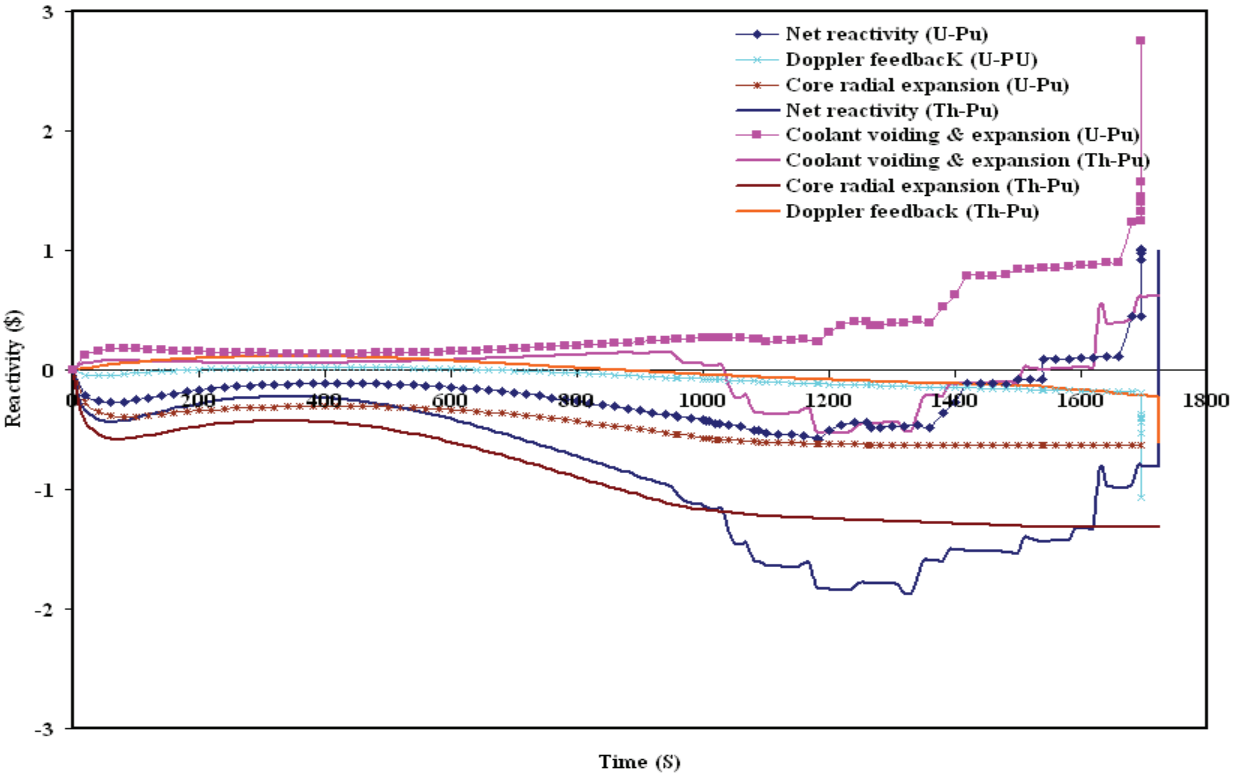


Figure 4.5: **Feedback reactivity in Th-Pu and U-Pu-6Zr metal core (Lines with dots are representing Th-Pu system)**

Also during the transient, large negative core radial expansion results in rapid power drop and hence there is temperature drop in fuel. Due to this, there will not be significant broadening in the capture resonances of thorium and this reduces the absorption of neutrons in the capture resonances. Therefore between 53 seconds and 880 seconds,

positive Doppler feedback results from the reduced absorption of neutrons in thorium capture resonances.

It can be seen in Figure 4.4 that coolant expansion and fuel expansion is contributing positive reactivity, even though their magnitude is lower as compared to Doppler feedback. After 900 seconds coolant expansion alone is contributing significantly to positive reactivity and hence net reactivity is more negative after 1000 seconds. A plot comparing the feedback reactivity of two systems is given in Figure 4.5. The negative reactivity feedback from core radial expansion is greater for Th-Pu metal fuel due to the increased fuel removal worth (enrichment is high) and the reduced positive coolant expansion contribution than that of U-Pu-6Zr.

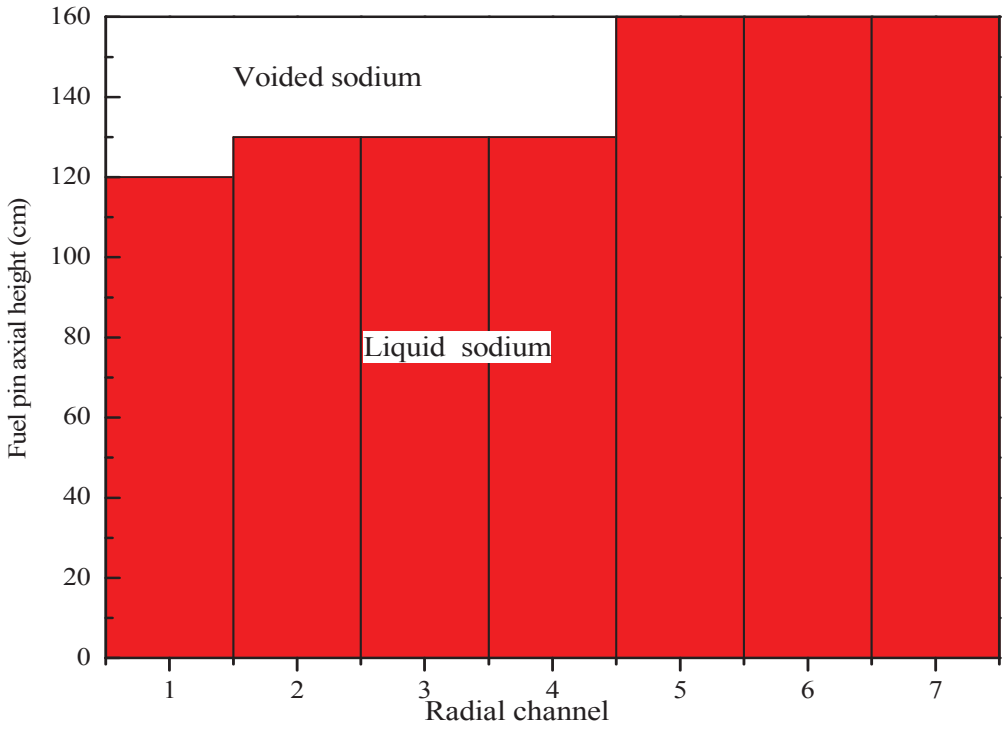


Figure 4.6: Sodium void distribution at 1000s in Th-19.3Pu 500 MWe metal reactor during ULOFA.

The pattern of sodium void propagation is shown in Figure 4.6. Sodium voiding starts at 886 seconds in the first channel across the upper axial blanket and is contributing negative reactivity due to the leakage of neutrons. The voiding is propagating to channels 2, 3 and 4 at 959, 968 and 998 seconds mainly in the upper axial blanket regions and is not propagating to the core center. In the reference core, the coolant expansion and void reactivity feedback is positive even up to the onset of boiling. In Th-Pu core, sodium expansion feedback is initially positive but after the onset of voiding it is dropping to -0.5 $. After 1000 seconds net negative reactivity in U-Pu-6Zr metal core is lower when compared to Th-Pu metal core. This is due to the positive reactivity feedback arising from coolant voiding and coolant expansion.

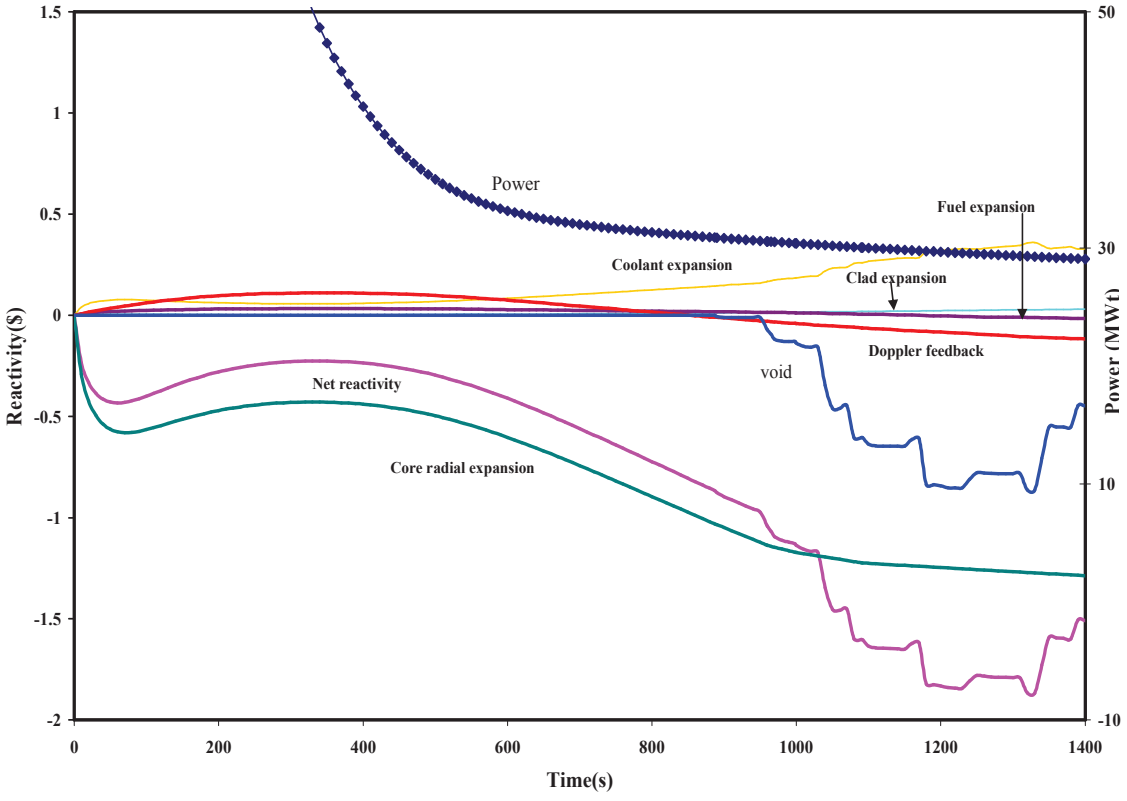


Figure 4.7: Feedback reactivity with conservative decay heat estimation Th-19.3Pu metal reactor

In Th-Pu metal core, power drop is so rapid that within 8.3 minutes, observed power is mainly due to decay heat. Within 11.5 minutes power approaches 32 MWt as seen in Figure 4.7 and 4.8. This is the upper limit of power which SGDHRS can handle. As the reactor remains in subcritical state till 1750 seconds, ample time is available for operator action and safe shutdown is possible with SGDHRS.

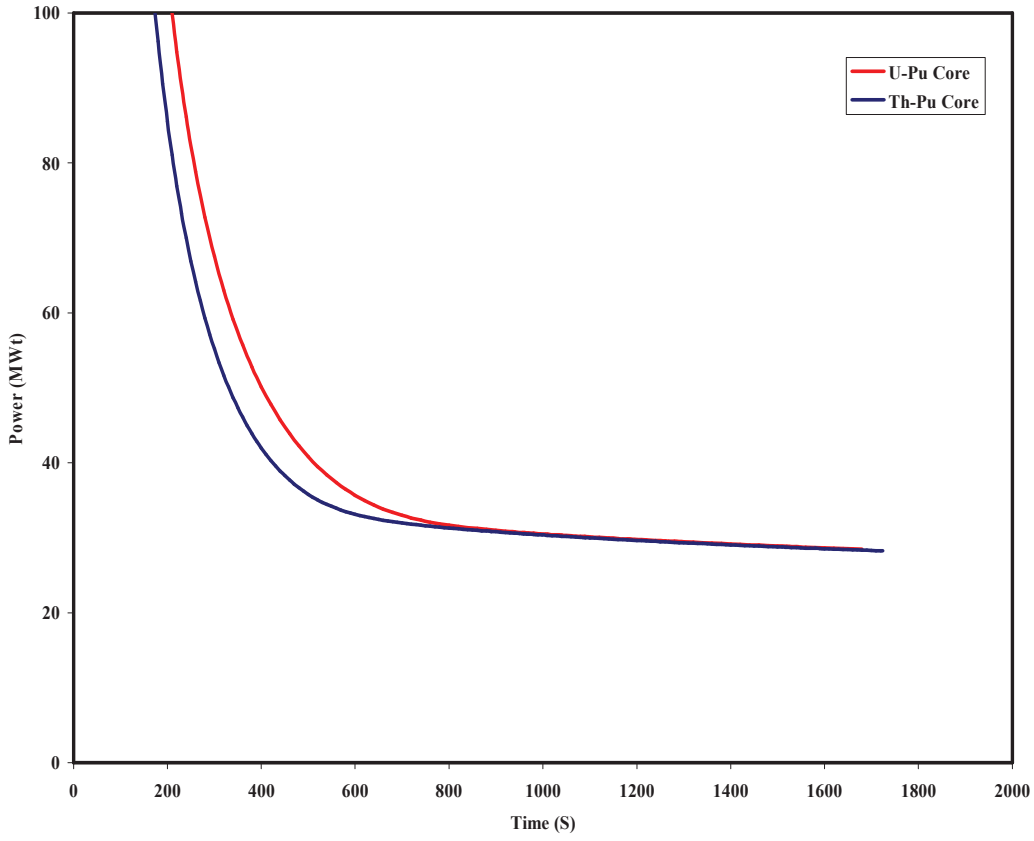


Figure 4.8: Power variation approaching SGDHR capacity in Th-19.3Pu and U-Pu-6Zr 500 MWe metal core.

In the case of U-Pu-6Zr system, power drop is at a slower rate and decay heat is the dominant contribution to the observed power after 12 minutes. The system is

approaching SGDHRS capacity in about 13 minutes as seen in Figure 4.9. From the net negative reactivity insertion rate and the extended time period available for corrective action, it can be concluded that Th-Pu system is safer than the reference U-Pu-6Zr core.

4.4 Sensitivity Analysis

The thermo physical properties of Th-Pu system, other than melting point, are not available in the literature. Therefore a sensitivity analysis has been carried out to get an estimate of the influence of uncertainty in thermo physical properties on the reactivity feedback during transients.

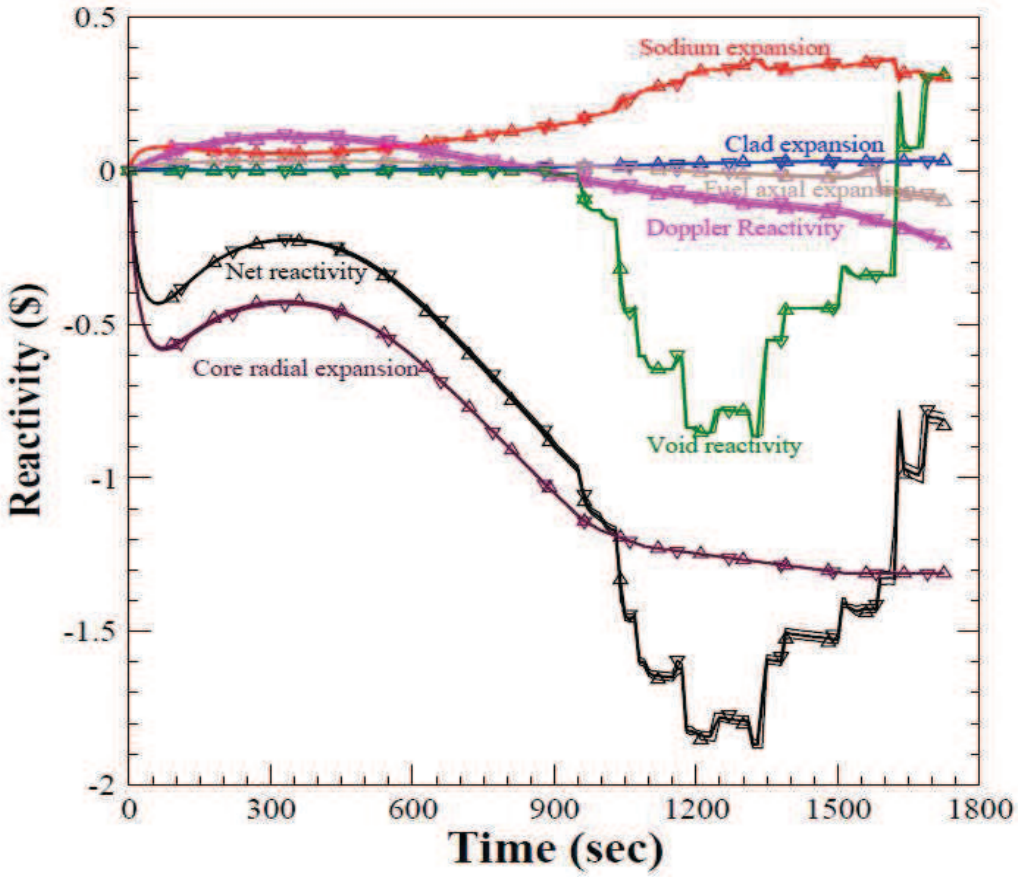


Figure 4.9:.Sensitivity analysis on the feedback reactivity with ±10% uncertainty in thermo physical data of Th-19.3Pu alloy. Upper triangle represents +10% uncertainty, lower triangle represents -10% and the reference case is the thick line.

An uncertainty margin of ±10% is considered in the study. The result of the sensitivity analysis is given in Figure 4.9. It can be seen that the trends in reactivity feedback behavior, with ±10% uncertainty marginin thermo physical data, does not vary much with that of reference case. This has indicated that uncertainties in thermo physical properties do not have much effect on the reactivity feedbacks of Th-Pu system and the conclusion regarding safe shutdown of the reactor without coolant boiling of entire reactor core remains valid.

4.5 Summary

The ULOFA analysis has been carried out for Th-19.3Pu 500 MWe metal fuelled reactor core to understand the passive shutdown capability of the reactor. Through steady state analysis, isothermal temperature coefficient and power coefficients has been evaluated and found to be -1.696 pcm/ °C and -0.183 pcm/MWt. Control rod worth as well as excess reactivity requirement of this system are similar to that of U-Pu-6Zr metal core. The unprotected loss of flow (ULOF) accident is assumed to be initiated by a total loss of offsite power with all the control rods unavailable. It is found that power to flow ratio is increasing initially, but within 53 seconds it is decreasing and no further increase is observed. Power to flow ratio never goes beyond 4, thus ensuring the absence of coolant boiling in the entire core. This reduction in the power mainly comes from core radial expansion which is the major contributor towards the observed net negative reactivity.

The observed net negative reactivity is higher in Th-Pu system when compared to U-Pu-6Zr system. Sodium voiding which initiated during 886 seconds in the upper axial blanket is also contributing towards negative reactivity. Voiding is not propagating

towards the central part of the core. It is also found that power drop is so rapid that within 11.5 minutes power approaches 32 MWt which is the upper limit of power removal capacity of SGDHRS. Therefore an operator action for safe shutdown is possible with SGDHRS and the reactor will not go towards a core disassembly phase. A sensitivity analysis has also been carried out to get an estimate of the dependence of uncertainty in thermo physical data on reactivity feedback contribution by considering a plausible uncertainty margin of ±10%. This analysis shows that system will undergo safe shutdown even with considered uncertainty in thermo physical data of Th-Pu system. To sum up, the performance of the Th-Pu metal fuel reactor during ULOFA shows that the inherent safety of the fuel type is superior to that of reactor core employing U-Pu-6Zr fuel.

Chapter 5

Criticality safety studies for Plutonium Uranium Metal Fuel Pin Fabrication Facility

5.1 Introduction

Metallic fuels are superior in terms of breeding as reviewed in Chapter 2. Fuel cycle technology of metal fuel is quite different from that of oxide and carbide fuels. Before commercial deployment of metal fast reactors, it is essential to proceed in steps to understand the irradiation behavior of the fuel as well as to get experience in fuel fabrication, reprocessing and fabrication of fuel using reprocessed fuel. In India, as a first step, it is planned to use FBTR as a test bed in irradiating large number of metallic fuel pins/ subassemblies in the next few years (Ganesan et al., 2011). The following steps are envisaged in the metal fuel development:

- Fabrication of metal pins and their irradiation followed by Post Irradiation Examination (PIE). In this step, each type of fuel will be irradiated in three special subassemblies, each sub assembly carrying three pins. PIE will be carried out subsequently with fuel undergoing burnup of 25, 50 and100 GWd/te.

- In the second phase, fuel will be tested at sub assembly level.

- In the third step either part of the FBTR core will be loaded with metal fuel, keeping mixed carbide fuel as the main driver fuel or full core will be loaded with metal fuel subassemblies.

The above steps give the confidence to embark on commercial metal fuel reactor

programme. In order to fabricate fuel for test program, a small scale **P**lutonium-**U**ranium **M**etallic **P**in **F**acility (PUMP-F) is planned (Ganesan et al., 2011) to be constructed in FBTR complex in Indira Gandhi Center for Atomic Research (IGCAR).

Table 5.1: Fuel requirements for different options of FBTR metal core.

Core Size	Material	Sodium Bonded U-19Pu-6Zr	Mechanically bonded U-15Pu
64 SA (37 Pins/SA)	Pu (metal)	48 kg (20.3 g/pin)	43 kg (18.2 g/pin)
	U Enrichment (%)	46.3	40
	U metal	188 kg (79.4 g/pin)	245 kg (104 g/pin)
84 SA (37 Pins/SA)	Pu (metal)	62.6 kg (20.3g/pin)	56.5kg(18.2 g/pin)
	U Enrichment (%)	42.6	36.8
	U metal	247 kg (79.4 g/pin)	319.8kg(104g/pin)

In the initial phase of metal fuel program, the driver fuel used in EBR II was Uranium-Fissium, U-5Fs, (5 %) binary alloy (Waiters et al., 1984). Fissium is an alloy that approximates the equilibrium mixture of metallic fission product elements left by the pyrometallurgical recycling designed for the EBR-II; it consists of 2.5 wt% molybdenum, 1.9 wt% ruthenium, 0.3 wt% rhodium, 0.2 wt% palladium, 0.1 wt% zirconium, and 0.01 wt% niobium (Hofman et al., 1997). The U-Pu binary fuels have very low melting point and their anisotropic swelling characteristics leads to lower linear power and burnup (Walter et al., 1975). To mitigate this problem, Zirconium (~10%) is added to the fuel and lot of irradiation tests were carried with binary and ternary alloys such as U-10Zr, U-

8Pu-10Zr, U-19Pu-10Zr (Pahl et al., 1990 a,b). The addition of Zirconium reduces the breeding ratio (causing an increase in the doubling time) and increases the critical mass leading to increase in capital cost (Riyas and Mohanakrishnan, 2008). Since India needs shorter doubling time fast reactor systems, it is also desirable to reduce the Zirconium content in the fuel. Though there is good amount of irradiation experience with 10% Zirconium fuel, the irradiation data with lower Zirconium content is not very extensive. The enrichment used in the fuel will be based on design linear heat rate and the core height of the reactor in which the fuel will be irradiated. If FBTR is fully converted in to metal fuelled core, it has various fuel and size options. These options are given in Table 5.1. This study considers both binary and ternary fuel types with enriched Uranium. There are also plans for testing U-Pu binary fuel which is mechanically bonded. The clad in this case will be lined with 125μ thick Zirconium.

The main focus of the work presented in this chapter is to study criticality safety in PUMP-F, during the preparation and fabrication of metal fuel pins comprising the fuel type U-15Pu, U-19Pu and U-19Pu-6Zr. This study has to set limit on permissible fuel mass during the metal feedstock preparation and the permissible number of the fuel slugs and pins during injection casting and fuel pin fabrication.

5.2 PUMP-F Facility

The proposed plant consists of three facilities where the first two are related to plutonium and uranium reduction facility followed by fabrication facility. Plutonium and uranium reduction facilities have glove boxes and fume hoods to handle and reduce the metal oxides. Fabrication facility consists of containment boxes of size 10m x 2.5m x 3m, out of which one is an air cell and other two are argon cells (Ganesan et al., 2011). The

main fuel fabrication steps such as fuel feedstock preparation, induction melting, injection casting, slug de-molding, swaging and shearing, quality control, slug loading in the clad, end plug welding and sodium bonding are carried out in the first two argon cells which are at negative pressure. The details of the fabrication scheme are given in Figure 5.1, and also show the stages where criticality studies are done.

The preparation of master alloy, which is also termed as feedstock preparation, will be carried out in induction melting furnace. The alloying elements are obtained from cold fuel inventories or from reprocessed fuels (i.e. from spent oxide or metallic fuels). The ingots of uranium, plutonium and zirconium which are to be alloyed will be loaded into Yittria coated graphite crucible and melted in a high-frequency-powered pressure/vacuum induction furnace at approximately 1500^{o}C (Jelinek and Iverson, 1962; Wilkes et al., 1987). This is referred as metal feedstock preparation. Placement of feedstock into the crucible is important to aid the alloying process.

In the injection casting stage, cassette of preheated quartz moulds is lowered in to the alloy melt followed by a rapid pressurization which will allow the melt to rise to the pre-determined height in the mould. Then the quartz moulds are withdrawn from the melt to facilitate the separation of fuel slug and the quartz mold, followed by swaging and shearing to obtain the required dimensional tolerances (Burkes et al., 2009). After the fuel slugs had passed the inspections, they will be fabricated. A fuel jacket is then fabricated, loaded with sodium to facilitate bonding, followed by the insertion of the fuel slug and finally closure welded. The finished fuel rod is dimensionally characterized and He-leak checked and transferred to fuel storage area.

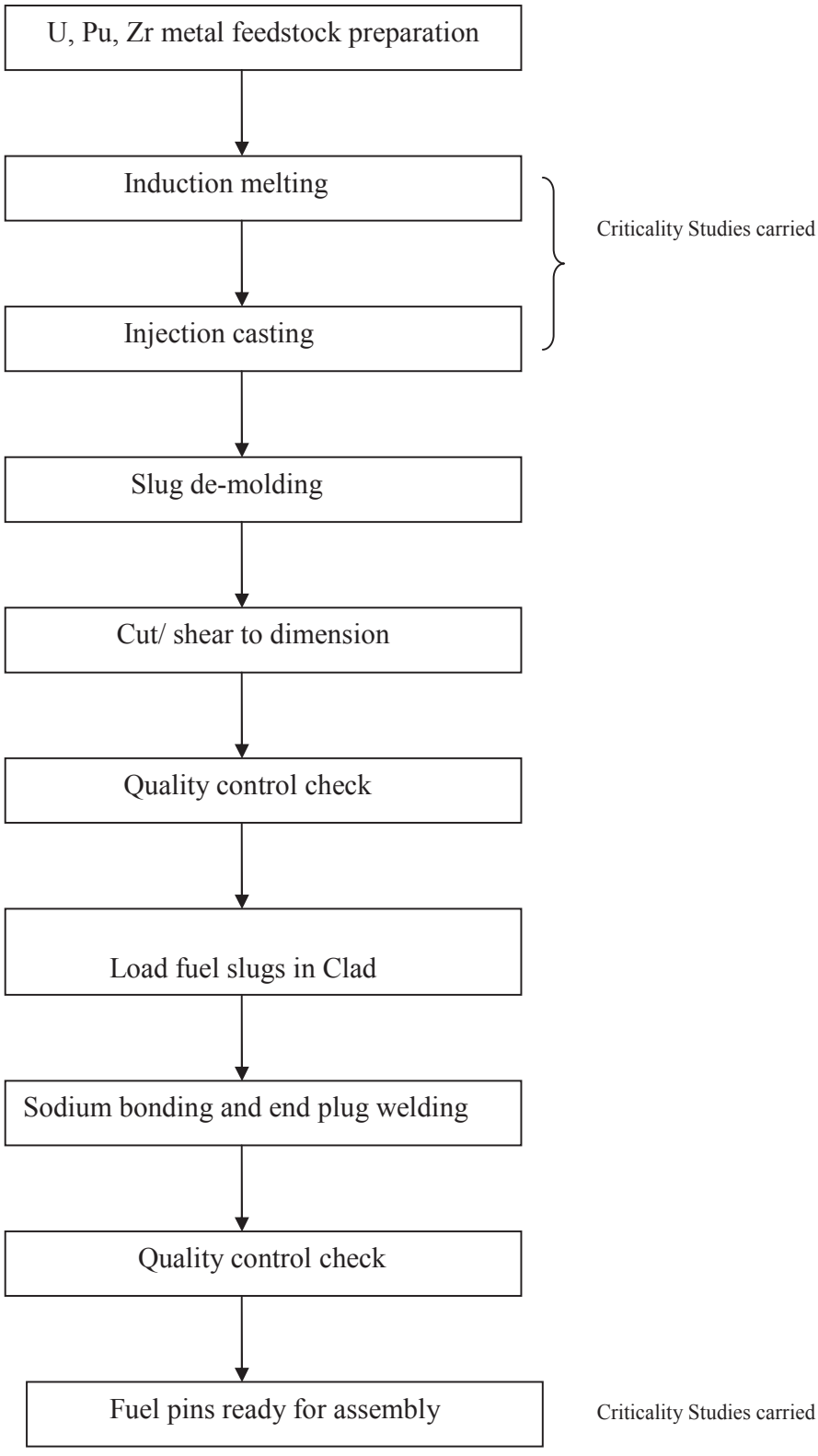


Figure 5.1: Fabrication stages of metal fuel pins.

Plutonium- uranium metal form as well as fabricated pins will be stored in Thick aluminum containers having stainless steel double envelope (Ganesan et al., 2011).

5.3 Method Of Analysis

The most important parameter which specifies the safe handling of fissile material is criticality. Criticality accidents such as Tokai-Mura accident (NRC review report, 1999) resulted in high doses to operating personnel and large spreading of contamination. Safety requirements specific to the type of the fuel and the fabrication, are essential to ensure safety and prevents any possibility of criticality hazards during its preparation and fabrication. Therefore an evaluation of the criticality safety of the system together with the factors which significantly affects them is necessary. Metal fuel fabrication follows dry route, but the large amount of Plutonium in fast reactors fuels, is the reason for concern. Criticality hazards should be avoided by means of proper design and IAEA guidelines insists that criticality calculation and evaluation should be done on conservative assumptions (NS-R-5, IAEA).

Criticality is a function of mass, geometry, moderation, reflection, neutron absorption and concentration (Knief, 1985; IAEA, 1966). The presence of unknown reflector in the form of equipment, containers or non-controlled materials which may be close to the fissionable material in the glove box, increase the reactivity of the system in the fabrication facility. Care is to be taken to avoid the presence of moderating materials inside glove box. But water can not be totally avoided due to a) atmosphere cooling system which contains a mixture of water and glycol and b) fire protection sprinkler system (Ganesan et al., 2011). A 50 cm water reflector which is equivalent of an infinite thick reflector as well as a moderator has been used in calculation to take care of the

above concerns. A measure of the increase or decrease in neutron production in an infinite (or finite) multiplication system is defined by infinite multiplication factor k_∞ (or effective multiplication factor, k_{eff}).

Infinite multiplication factor, $k_\infty = \dfrac{\text{neutron production from fission in one generation}}{\text{neutron absorption in the preceding generation}}$

There are four factors that are completely independentof the size and shape of the envelop containing fissionable isotopes, that give the inherent multiplication ability of the fuel and moderator materials, without regard to leakage. This four factor formula represents the infinite multiplication factor as shown in the equation below

$K_{\inf} = \varepsilon p f \eta \rightarrow (5.1)$

ε = Fast fission factor

P = Resonance escape probability

f= Fuel utilization factor

η = reproduction factor

Since commercial reactor can have a maximum of 19% Plutonium in U-Pu fuel type (Crawford et al., 1993; Hofman et al., 1997; Wigeland and Cahalan, 2009), the content of plutonium is restricted to 19% in the ternary fuels and 15% in the binary fuels. The enrichment in the uranium is adjusted to attain the required criticality. The enrichment of uranium required for various fuel options is given in Table 5.1. It can be seen that 46% is the maximum enrichment required, but to make the study more conservative, an enrichment of 50% is used in the calculation. Pin design parameters are given in Table 5.2. Pin and fuel diameters are similar to commercial reactors, where as

the height is decided by the FBTR core height. The feed plutonium for the Indian fast breeder program is expected to come from Pressurized Heavy Water Reactor (PHWR) whose typical isotopic vector is given in Table 5.2. However in this study pure ^{239}Pu is also considered as it is most conservative and will give the flexibility of using plutonium coming from any other reactor which will be having different isotopic vector.

Table 5.2: Fuel pin design parameters used in the study.

Pin diameter (mm)	5.7
Clad thickness (mm)	0.45
Theoretical density of U-15Pu/ U-19Pu/ U-19Pu-6Zr (gm/cc)	19.20/19.23/17.15
Smear density (%)	85
Height of the pin without cladding (mm)	360
Height of the pin with cladding (mm)	320
Uranium isotope ratios: U235/U238	50/50
PHWR grade Plutonium:Pu239/Pu240/Pu241/Pu242	68.8/24.6/5.3/1.3

In order to take care of errors in the calculations that can arise from sources such as nuclear cross-section data and physical parameters of the fuel, the k_{eff} is limited to maximum value of 0.9.

5.4 Simulation

Criticality calculations are carried out with Neutron Transport Monte Carlo code MCNP.4B. Steps in the calculation are given in Figure 5.2. Continuous energy cross-sections in ACE format (ENDF/B-VI.7) at room temperature (300K) were used.

Criticality run is made with 1000 neutrons for 150 minutes with the skipping of first 100 cycles.

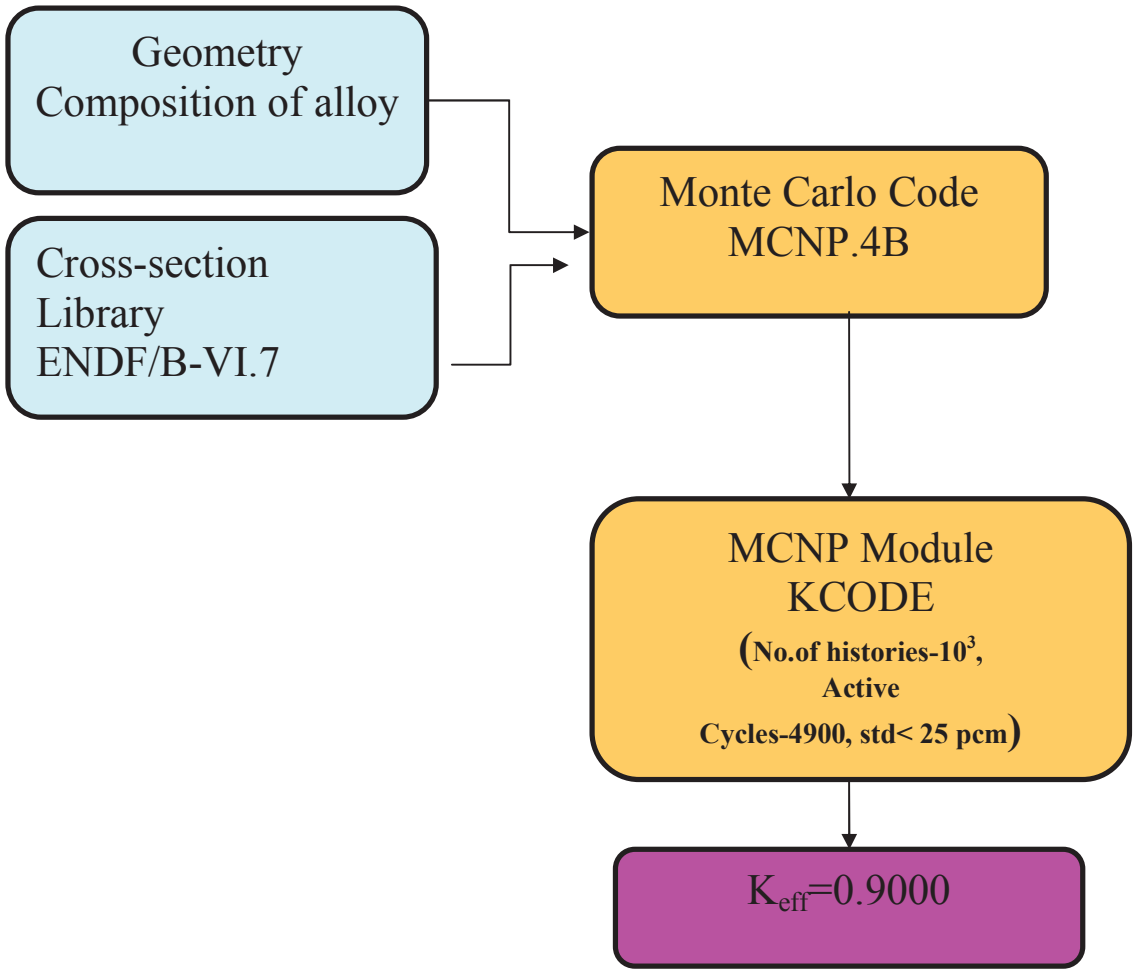
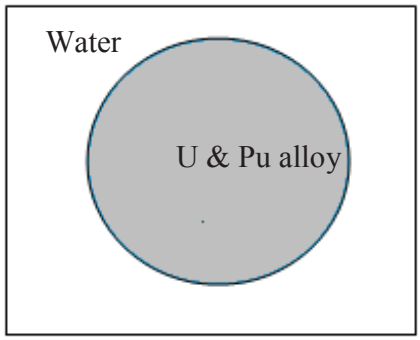


Figure 5.2: Steps in the calculation.

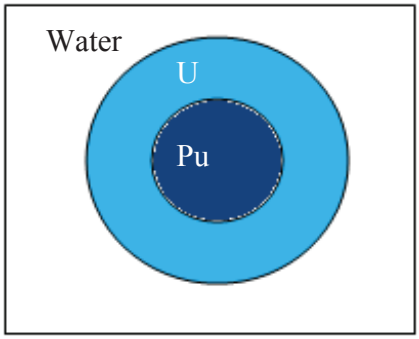
5.5 Criticality Safety In The Induction Melting Stage

At this stage, the aim is to estimate the mass of the metal ingots that can be loaded into the induction melting furnace without making the system critical. To do this, fuel of specified composition, given in Table 5.2, is assumed conservatively; in the form of

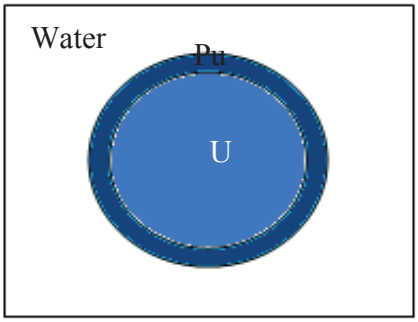
sphere and the criterion adopted is to ensure a $k_{inf} < 0.9$ under water flooded condition. Since the placement of the alloying material inside the crucible is very important to aid alloying process, the lower melting temperature material is allowed to consume the other. Actual geometry of fissile material is a matter conjecture.



Homogenous mixture of uranium and plutonium



Plutonium sphere surrounded by molten uranium shell



Uranium sphere surrounded by molten plutonium shell

Figure 5.3: Conservative geometry configurations of the fuel material in crucible.

Therefore three conservative cases are considered for each fuel composition and are given in Figure 5.2. A homogenous mixture is considered first which results from proper mixing of alloy materials.

The other two cases can happen during an accidental condition such as station black out which prevents uniform mixing of fuel material. To study this, firstly plutonium sphere surrounded by a shell of enriched uranium is considered for all fuel compositions. The reverse case of enriched uranium sphere surrounded by a shell of molten plutonium is also studied. Both the cases studied takes into account a water reflector of thickness 50 cm. For these two cases, Zirconium is not considered as it is expected to reduce the k_{eff} and will only add the safety margin. The calculations are carried out with MCNP code using KCODE module. The masses of various fuel compositions in spherical geometry under flooded condition that will ensure safety during the metal feedstock preparation are given in Table 5.3. As seen from the Table 5.3, the variation in the safe mass for different fuel types is from 10.5 to 24.25 kg. The lowest mass is for plutonium sphere surrounded by a shell of uranium. This is because, immediately after fission most of the neutrons will have energy around1 MeV and η value of ^{239}Pu is higher than that of ^{235}U in fast spectrum. Since the facility has to be a general purpose one, the limiting value of the mass is determined by selecting the most conservative fuel type and condition, i.e. plutonium sphere inside uranium sphere with U-19Pu fuel type. Therefore the safer mass during induction melting stage can be set to 10.5 kg.

Table 5.3: Masses of various fuel compositions that are safe during feedstock preparation.

Homogenous sphere of fissile isotopes				
Fuel Type (Enriched Uranium)	**Mass of Plutonium (Kg)**	**Mass of Uranium (Kg)**	**Total fissile mass (kg)**	**k_{eff}**
U-15Pu	3.4	19.6	23.1	0.89037±0.00020
U-15Pu*	3.6	20.5	24.1	0.89956±0.00020
U-19Pu	3.4	14.8	18.2	0.89848±0.00019
U-19Pu*	4.01	17.09	21	0.89309±0.00019
U-19Pu-6Zr	4.1	16.3	20.4	0.89089±0.00019
U-19Pu-6Zr*	4.9	19.35	24.25	0.89137±0.00019
Uranium sphere inside Plutonium sphere				
U-19Pu	4.22	18	22.22	0.89521±0.00020
U-19Pu*	5.56	23.7	29.26	0.89523±0.00020
U-15Pu	3.529	20	23.529	0.89775±0.00020
U-15Pu*	4.46	25.3	29.76	0.89865±0.00021
Plutonium sphere inside Uranium sphere				
U-19Pu	2.0	8.5	10.5	0.89074±0.00020
U-19Pu*	2.4	10.23	12.63	0.89777±0.00020
U-15Pu	1.85	10.48	12.33	0.89332±0.00018
U-15Pu*	2.9	16.43	19.33	0.89261±0.00021

*PHWR grade plutonium

5.6 Criticality Safety During Injection Casting And Fuel Pin Fabrication

In this phase, the issue is to obtain the number of fuel slugs as well as fabricated pins that can be arranged in lattices without criticality accident, even in the flooded condition. Therefore, in addition to the 50 cm thick water reflector, ingression of water in to the lattice is considered. For a particular fuel composition, the study starts with pin cell configuration with minimum spacing between them, extending to arrays which will give K_{eff}< 0.900. The arrangement of fuel slugs and pins in square and hexagonal lattices are shown in Figure 5.3. Now with the obtained number of pins in the array, spacing between the fuel slugs is varied.

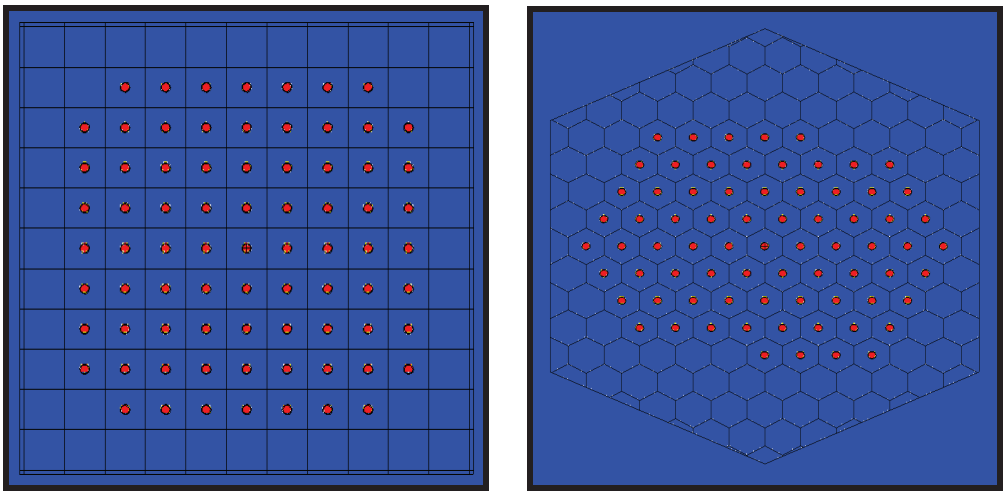


Figure 5.4: Arrangement of fuel pins in square and hexagonal lattices.

To obtain the safer number of fuel slugs/pins, the water gap between fuel slugs/pins is varied in square and hexagonal lattices and the corresponding k_{inf} value is noted. K_{inf}

increases with pin spacing reaching a maximum for 2.5 cm of water gap and then starts to decrease. The variations of k_{inf} with water gap in square and hexagonal lattices are given in Figure 5.4 and 5.5. This variation of k_{inf} with water in the gaps is due to the combined effect of fuel utilization factor and resonance escape probability. When water in the gap is increased, moderator to fuel ratio increases.

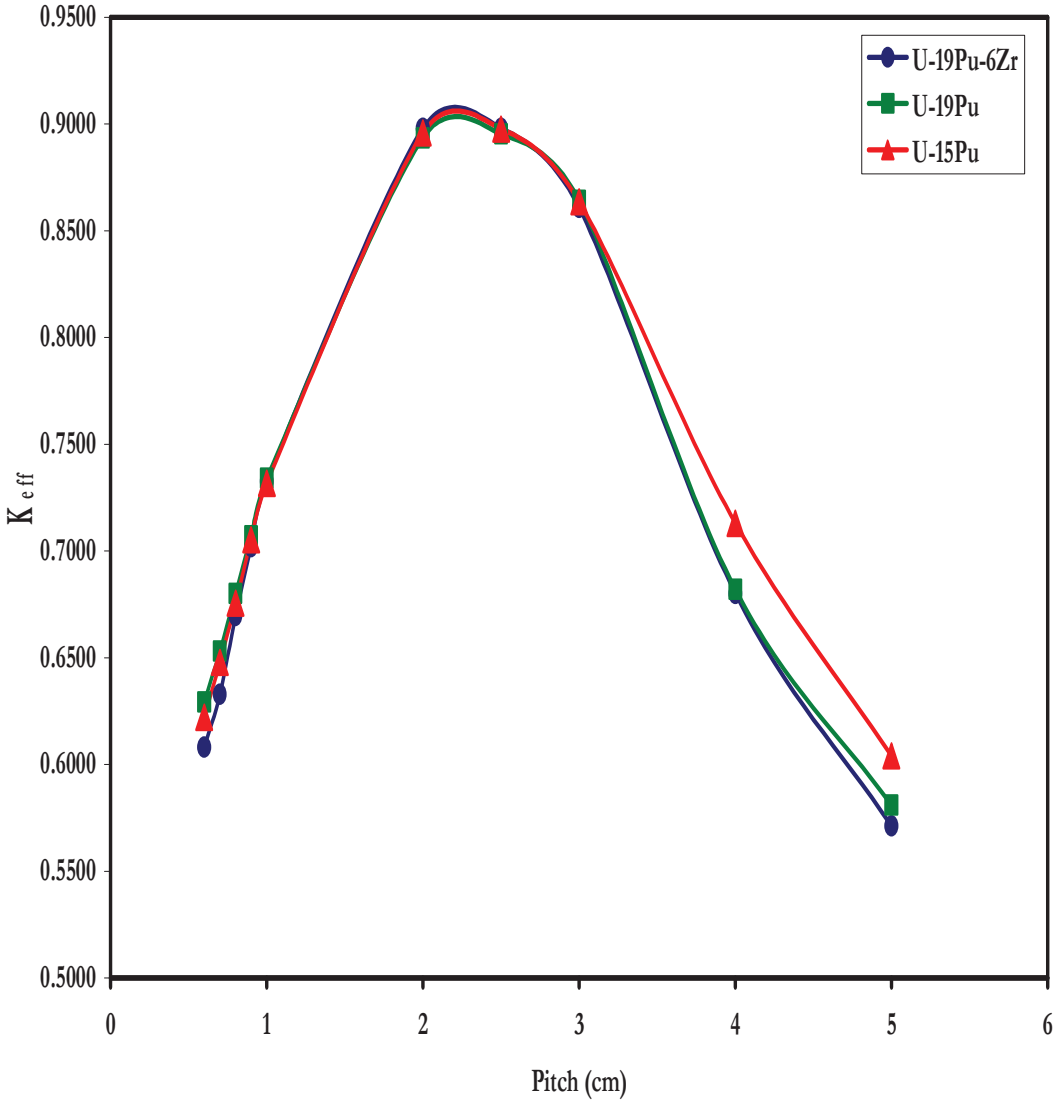


Figure 5.5: Variation of k_{inf} with spacing of fuel slugs in square lattice

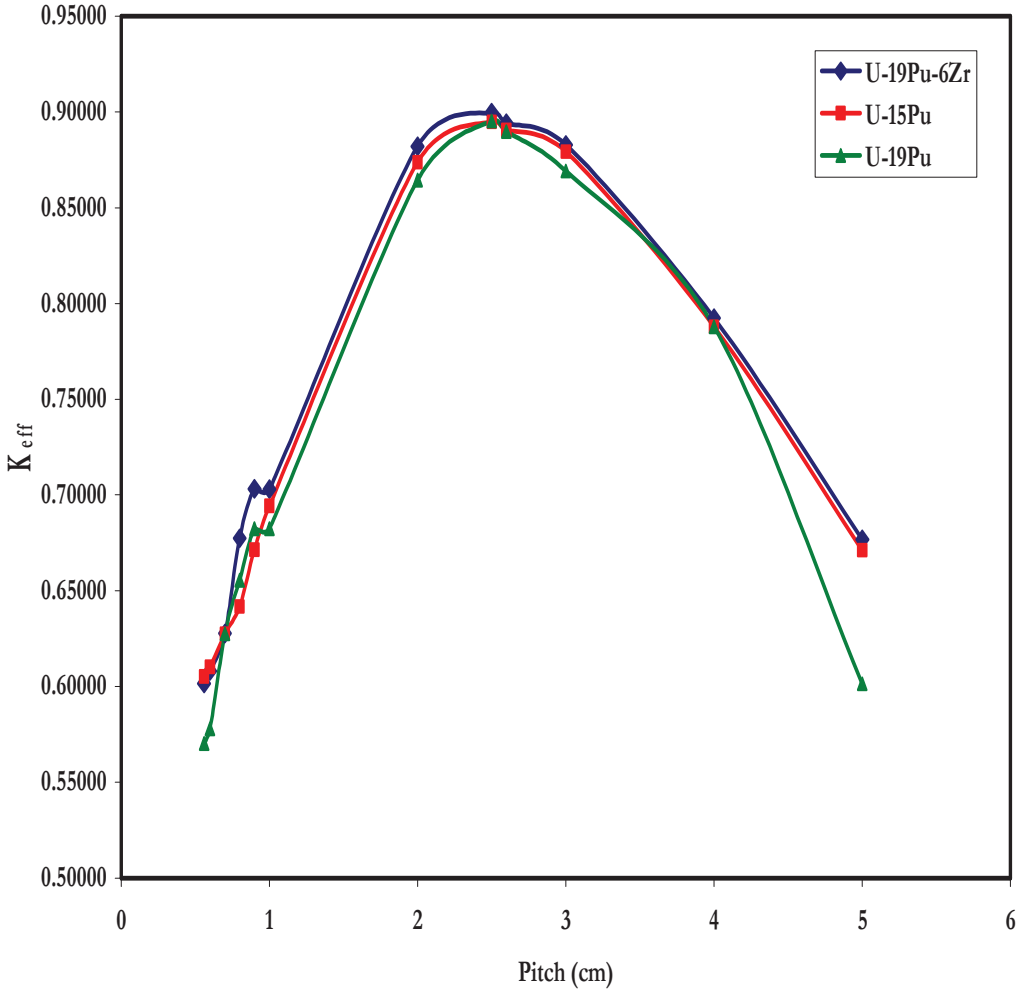


Figure 5.6: Variation of k_{inf} with spacing of fuel slugs in hexagonal lattice.

This will increase neutron moderation causing an increased absorption of neutrons in moderator. At the same time, thermal utilization factor decreases. Increased neutron moderation causes larger number of neutrons to escape resonances and thus an increase in resonance escape probability. Thus k_{inf} increases initially and reaches a maximum value at 2.5 cm and then starts to decrease as seen in Figure 5.5 and 5.6. The overall

variation in k_{inf} is due to the competing effects of reduced fuel utilization factor and increased resonance escape probability. The number of fuel slugs and fuel pins that can be arranged without criticality event, for different fuel compositions are given in Table 5.4 and 5.5.

Table 5.4: Safe number of fuel slugs after injection casting.

Fuel type	Hexagonal lattice	K_{eff}	Square lattice	K_{eff}
U-15Pu	54	0.89774±.00023	57	0.89768±0.00022
U-15Pu*	63	0.89511±.00022	66	0.89810±0.00021
U-19Pu	53	0.89542±.00022	56	0.89536±0.00022
U-19Pu*	63	0.89584±.00022	66	0.89583±0.00021
U-19Pu-6Zr	55	0.89632±.00023	58	0.89820±0.00022
U-19Pu*-6Zr	66	0.89656±.00022	68	0.89555±0.00020

*PHWR grade plutonium

Table 5.5: Permissible number of fuel slugs after fabrication.

Fuel type	Hexagonal lattice	K_{eff}	Square lattice	K_{eff}
U-15Pu	72	0.89193±.00021	75	0.89673±0.00020
U-15Pu*	85	0.89335±.00020	88	0.89274±0.00020
U-19Pu	71	0.89118±.00021	74	0.89722±0.00020
U-19Pu*	87	0.89689±.00020	90	0.89400±0.00019
U-19Pu-6Zr	73	0.89700±±.00020	76	0.89804±0.00022
U-19Pu*-6Zr	92	0.89430±.00021	96	0.89511±0.00019

*PHWR grade plutonium

It can be seen that hexagonal lattice configuration is safe for lower number of fuel slugs/pins for the same fuel type. This is due to the fact that hexagonal lattice allows higher fuel volume fraction than square lattice. Therefore this configuration can be considered for limiting the fuel slugs/pins for criticality safety.

5.7 Summary

Criticality safety of U-15Pu, U-19Pu, and U-19Pu-6Zr metal fuel compositions has been studied during their feedstock preparation, injection casting and fabrication in the presence of water which acts as moderator and reflector. The masses of various fuel compositions under water flooded condition that will ensure safety during the metal feedstock preparation have been methodically estimated using MCNP.4B code for several conservative configurations.The configuration which gives large value of k_{eff} is the one where plutonium sphere is covered by a shell of enriched uranium, safe amount of fuel mass for U-15Pu, U-19Pu, and U-19Pu-6Zr during feedstock preparation can be limited to be 10.5 kg. Among the binary and ternary fuel composition, U-19Pu is the most reactive one because of the increased weight fraction of plutonium and therefore the number of fuel slugs which can be handled safely after injection casting and fuel fabrication can be fixed keeping this as the reference fuel. The number of fuel pin/slugs that can be handled safely after injection casting and fuel fabrication for the three different fuel compositionsare estimated by varying thickness of water filled gaps between fuel pins.The conclusions of the study are

- Hexagonal lattice gives higher k_{eff} value as compared to square lattice, for the same number of fuel pins.

- k_{eff} increases with pin spacing. It reaches a maximum for 2.5 cm of water filled gap and then starts decreasing.
- During injection casting, cassette can safely handle 53 fuel slugs.
- Following fuel fabrication, the safe number of pins that can be stored is 71.

Chapter 6

Conclusions And Scope For Future Work

The present work has dwelt with the investigation of safety neutronic parameters of advanced reactor fuels that can be deployed in fast breeder reactors. A study of metal fuels types with particular emphasis both on safety in reactors and fabrication process is carried out for the first time. The significance of this study is recognizing the feasibility of an alternate fuel cycle in Indian scenario from neutronics point of view. The results provide a fuel composition with superior safety characteristics and optimum breeding characteristics. The results show that among the hybrid combinations of Th-U fuel type, only carbide fuel can provide suitable breeding. The following sections summarize the contributions and salient results that were presented in this thesis. The scope for the future research work is also discussed in this chapter.

6.1 Analysis On The Breeding Capability And Safety Related Neutronic Parameters

A comprehensive analysis has been performed for the first time on the breeding capability and safety parameters of advanced fast reactor fuels with all possible fertile-fissile combinations using recent cross-section set ENDF/B-VI.7 and detailed modeling of reactor core. Another advanced cross-section set, JEFF-3.1, was also used to study effect of cross-sections on the breeding ratio of advanced fast reactor fuels. Safety related nuclear parameters such as effective delayed neutron fraction, Doppler defect and sodium void reactivity were also estimated for all possible fissile fertile combination of metal, oxide, carbide and nitride fuels.

Results of the above analysis show that:

- On the basis of breeding potential and safety related performance, it is found that there are better prospects of utilization of thorium resources with hybrid fuel cycles employing Th-Pu.
- From this study it has evolved that among the hybrid combinations of Th-^{233}U fuel type, only carbide fuel can provide breeding, which is in marked contrast to earlier studies.
- Sodium voiding from the central part of the reactor core, with Th-^{233}U fuel for all combinations of metal, oxide, carbide and nitride fuel, provide negative reactivity.
- The negative fuel Doppler reactivity contribution is also larger for hybrid combination of Th-^{233}U fuel.
- Once enough ^{233}U have been bred from ^{232}Th, a shift towards ^{233}U- ^{238}U cycle would be more beneficial.

6.2 Unprotected Loss Of Flow Accident In Th-Pu 500MWe Metal Reactor

The ULOFA analysis has been carried out for Th-19.3Pu 500 MWe metal fuelled reactor core to understand the passive shutdown capability of the reactor. Accident analysis to get an overview of the inherent safety of Th-Pu fuel type has been carried out for the first time. Through steady state analysis, isothermal temperature coefficient and power coefficients have been evaluated. ULOFA analysis is done to identify the contribution of different reactivity feedbacks towards arresting the transient, i.e. their influence on starting and propagating sodium void, initiation of melting of fuel and time

availability for corrective actions such as opening the damper to initiate SGHDR system ensuring the passive shutdown capability of the reactor.

Results of ULOFA analysis show that:

- The net reactivity feedback is negative and magnitude wise larger than that of U-Pu-6Zr metal reactor.
- During the ULOFA, sodium voiding starts in the upper axial blanket and do not propagate to the core center thereby contributing to negative reactivity.
- Core radial expansion is the major contributor towards the observed net negative reactivity.
- The resulting power drop in Th-19.3Pu metal core is so rapid, that an operator action for safe shutdown is possible with SGDHRS.

6.3 Criticality Safety Studies In Metal Fuel Fabrication Facility

Criticality safety of binary and ternary Uranium- plutonium metal fuel alloys comprising of U-15Pu, U-19Pu, and U-19Pu-6Zr have been studied for the first time during their feedstock preparation, injection casting and fabrication in the presence of water which acts as moderator and reflector. For conservatism, various spherical geometry configurations have been considered. The safe masses of various fuel compositions in spherical geometry under flooded condition that will ensure safety during the metal fuel fabrication were determined. Among the binary and ternary fuel composition of U, Pu and Zr, U-19Pu is found to give highest k_{eff} due to large weight fraction of plutonium. Therefore the number of fuel slugs which can be handled safely after injection casting and fuel fabrication can be fixed keeping this as the reference fuel.

Results of this analysis show that:

- During feedstock preparation, safe mass of plutonium and uranium is limited to 2 kg and 8.5 kg respectively, i.e. total fissile mass is 10.5 kg.
- The number of fuel pin slugs that can be handled safely after injection casting is 53.
- The number of fuel pins that can be handled safely after fuel fabrication is 71.
- Safe limit obtained on fissile mass in the present facility can also take care of the fabrication needs of future experimental metal reactors.

6.4 Future Research Directions

The directions of future research are focused on the following aspects:

- Analysis of neutronic parameters is planned to be extended to burnt reactor cores. This will give an insight on the influence of minor actinides, specifically ^{241}Am, on sodium void reactivity and Doppler constant.
- Optimization studies including production of ^{232}U in the core and blanket regions of metal reactor fuelled with Th–Pu and Th-^{233}U fuels. ^{232}U is a bugbear in terms of shielding as well as remote handling required during reprocessing of the fuel. This will also increase the cost of fuel cycle.
- Safety studies on lead/lead-bismuth as the coolant, as this coolants appear to eliminate the concerns of sodium, since the chemical reactivity of these coolants with respect to air and water is very low.

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