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A STUDY ON BREEDING PERFORMANCE OF CARBIDE FUELED COMMERCIAL LIQUID METAL COOLED FAST BREEDER REACTOR

Nanami Kosaka and Shigehiro An(*)

ABSTRACT

Breeding and safety characteristics of large fast breeder carbide fueled reactor were evaluated with primary emphasis on minimizing doubling time.

Carbide fuel was considered as future potential LMFBR(**) fuel due to its high density and high thermal conductivity. Carbide fuel provides a reactor doublind time around 12 years. Since the homogeneous core configuration presents a high positive sodium void reactivity a heterogeneous core is recommended to ensure high breeding performance and low positive sodium void reactivity.

1 – INTRODUCTION

The world demand for energy grows as man-power is replaced more and more by machine power. It is known that the oil and natural gas are going to decline in the near future, hence to solve the energy problem in future assiduous efforts should be made to develop a new energy system. Among them the most promissing alternative energy is nuclear energy and various kind of thermal reactor system have been developed and are in operation now.

Even considering the vast amount of energy produced by fission of ^{235}U, uranium-235 atoms are used up, being converted to other atoms during the process. Since there is only a finite reserve of uranium in the earth, it is an exaustible, nonrenewable fuel source, just like the fossil fuel.

Nuclear energy strategy calculations have shown that the uranium consumption can be markedly reduced by the large future fast breeder reactors introduction. Present fast breeders under development utilize mixed uranium-plutonium oxide fuel and they present a doubling time ranging from 20 to 30 years.

In the case of uranium supply difficulties or higher energy demands, a reactor with short doubling time will be required. The fast breeder reactors with carbide fuel have a specially greater potential on this respect than that with oxide fuel. Therefore the carbide fuel introduction in the future goal of fast breeder reactor development.

Some researches concerned with carbide fuel were evaluated in Germany, France and USA. The applicability of the carbide fuel in large commercial reactors has been evaluated by Argonne National Laboratory[27] and Combustion Engineering, Inc.[26]. These works have concentrated on analytical study of nuclear properties for homogeneous core configurations. In order to evaluate the future potential of carbide fuel for LMFBR, the characteristics of several carbide fueled reactor configurations will be investigated in this work. Also heat transfer, nuclear and safety studies are performed and compared with those of the oxide fuel.

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(**) LMFBR: Liquid Metal Fast Breeder Reactor.

The objectives of this work include three itens:

1) - To perform several comparative studies around a 1200MWe reference reactor LMFBR configurations;

2) - To achieve an improved reactor core design that produces a short doubling time;

3) - To analyze the inherent safety characteristic of the cores considered.

2 – FAST BREEDER REACTOR

2.1 – FBR Development

Generally the development and evaluation of new major technologies is seen to follow a pattern that distinguishes three feasibility, as:

- scientific feasibility

- engineering feasibility

- commercial feasibility

In case of fast breeder development the technologies for liquid sodium as coolant and for mixed oxide as fuel are essentially in hand. It can, therefore, be considered that the phases of scientific and enginnering feasibility have almost passed, the commercial feasibility however has not yet passed.

The success and the early commercialization of the LMFBR will depend on the success Super-Phénix(*) operation and performance. Commercial feasibility of LMFBR including the remaining part of the fuel cycle such as fuel fabrication plant, fuel reprocessing plant and their waste disposal must be also fully developed.

The energy crisis has reemphasized the importance of achieving a short doubling time in a FBR to save world uranium resources. The next target will be to develop an advanced FBR fuel with thermal and neutronic properties superior to those of oxide. Carbide fuel can satisfy above conditions and will achieve a shorter doubling time than oxide fuel.

2.2 – Core Design Parameters Specification Procedure

2.2.1 – Core Design Procedure

The various parameters that contribute to core design generally are not independent but form an interrelated matrix in which the degree of influence of one parameter on another may vary considerably. The desired reactor core parameters are generated from trade-off envolving a lot of interdependent parameters.

The primary analytical areas are: nuclear design, thermal-hydraulic design, material design, safety design and economic related design, and the desired design specifications are determined by considering all the analytical areas. Fixed conditions, together with data needed for the analysis, such as cross sections, thermal and material properties, serve as input in each analytical areas. The required fuel loading, flux and power distribution are obtained from the nuclear design, and some feedback to both the nuclear and thermal-hidraulic design.

(*) Super-Phénix, the first commercial fast breeder reactor under construction.

The coolant temperature, flow and velocity and geometric arrangement evolve from the thermalhydraulic design. Thermal conditions also contribute with some feedback to the material specification. The thermal conditions is also taken as the limiting criteria for the core design. This representation emphasizes some important interdependences among the design parameters.

In this study only nuclear and thermal design are considered with safety design analysis. Since fast reactors are important primarily because of their potentially high breeding ratio, the design objetive is to obtain a high breeding ratio or specifically a short doubling time. So, the parameters associated with the production of new fissile fuel from fertile material are the interest. However the core breeding ratio tends to become a dependent variable rather than primary design objective since the design is also related with reactor size and configuration from the heat removal view point, coolant characteristics and perhaps various safety requirements.

First the homogeneous core design was taken into account and later the heterogeneous core was considered. Some design parameters are fixed and the fuel center temperature and cladding surface temperature were considered as limiting criteria.

2.2.2 – Nuclear Analysis

The design decision concerned with the selection of fuel element composition and diameter are critical to the fuel cycle economics. Design data on the thermal conductivity and limiting temperature of the fuel are used to compute the design limit linear power rating on the fuel element.

The steps for a core design approach considered are as follow:

- Fuel pellet diameter is specified;
- Cladding thickness and gap are specified;
- With wire diameter defined and space between pins fixed the pin pitch is determined;
- With core power and core heigh specified, the total number of pin fuel in the core is calculated;
- Number of subassembly is determined as well as the subassembly pitch;
- After definition of the assembly, the core material volume fraction is determined. The coolant volume fraction is a dependent parameter deduced from coolant velocity;
- Core sizing is determined;
- Reactor size is defined and geometric parameters are specified. The axial and radial blanket thicknesses are fixed;
- Neutronic calculations are evaluated. The fuel composition is determined taking into account the peaking factor and k_{eff} at EOEC (end of equilibrium cycle);
- Power density distribution is calculated and linear power is determined;

Nuclear characteristics calculated are:

- breeding ratio, doubling time, fuel inventory and power distribution;
- axial linear power profile is determined for the maximum power channel and thermal analysis is evaluated.

2.2.3 – Thermal Analysis

In this process, thermal analysis begins with the primary coolant inlet temperature and the mean outlet temperature specifications; the steps are as follow:

- The coolant temperature profile in the maximum power channel is determined;
- The axial profile of the cladding temperature is calculated using axial power profile and empirical correlations for the surface heat transfer coefficient;
- Cladding inner surface temperature is computed;
- With gap conductance specified the fuel surface temperature is computed;
- Using empirical correlation for thermal conductivity the fuel center temperature is calculated;
- Coolant mass flow rate necessary to maintain the diference between inlet and outlet temperatures equal to 150°C is calculated;
- From coolant flow area determined in design specification the coolant velocity is calculated.

The calculated maximum fuel center temperature and maximum cladding outer surface temperature are compared with allowable temperature specified from material properties. The coolant velocity is also compared with the permited value.

If the temperature at the fuel center or cladding surface is superior than the allowable temperature the new fuel pin diameter is considered and the calculations are performed again. If the coolant velocity is too high the coolant volume ratio is reconsidered and the calculations are repeated.

The limiting criteria adopted are:

- Maximum allowable temperature at fuel center: 1800°C
- Maximum permitted temperature at cladding surface: 620°C
- Coolant velocity recomended: 5–7 m/sec.

2.2.4 – Safety Analysis

In the safety analysis the remote possibility of a total voiding in the core (total loss of coolant) is accepted. Doppler coefficient and sodium void reactivity are calculated in this conditions.

The severity of the reactivity change perhaps can be reduced by providing enhanced Doppler coefficient or reduction of the void coefficient. The second option can be obtained by modifying the reactor core configuration, for example, to heterogeneous one.

2.3 – Breeder Principles and FBR Introduction

Of the fissionable isotopes, U-233, U-235 and Pu-239, only U-235 occurs in nature where it makes up about 0.71 per cent of the natural uranium. Although the other two fissionable isotopes do not exist in nature they can be manufactured.

In reactors such as LWR(*) the fuel used is enriched uranium with 3.0 to 3.5 per cent uranium-235 and less than 5 per cent of the total weigh of fuel in the core is fissioned before the fuels are removed.

Furthemore, a large amount of natural uranium has been used, whereas in fast breeder reactor, by its characteristics, a large fraction of natural U or Th ores can be fissioned multiplying, perhaps by more than 50 times, the usability of natural uranium resources.

The breeding principles consist of transforming the fertile material into new fissile material, consequently a fast breeder produces more new fissionable material then it consumes. It provides a way to use most of the natural uranium, that is, then non-fissionable U-238 or the thorium ore.

In the case of fission reactors, two kinds of reaction lead to breeding. The chain for the producing plutonium from uranium ($^{239}Pu/^{238}U$) and uranium from thorium ($^{233}U/^{232}Th$) are described by following chains.

$$_{92}U^{238} \xrightarrow{(n,\gamma)} {}_{92}U^{239} \xrightarrow[23.5\,min]{\beta^-} {}_{93}Np^{239} \xrightarrow[2.35\,d]{\beta^-} {}_{94}Pu^{239}$$

$$_{90}Th^{232} \xrightarrow{(n,\gamma)} {}_{90}Th^{233} \xrightarrow[23.4\,min]{\beta^-} {}_{91}Pa^{233} \xrightarrow[27.0\,d]{\beta^-} {}_{92}U^{233}$$

As shown in Table II.3.1 a number of fast breeder concepts could be considered with various types of fuel and fertile material, coolants and reactor arrangement. The main type of fast breeder presently under construction for power reactor is Liquid Metal Cooled Fast Breeder Reactor (LMFBR) type. It is a liquid (Na) metal cooled reactor with (UO_2-PuO_2) fuel in the core and depleted UO_2 in the blankets.

Table II.3.1

Design Variables for Fast Breeders

Fissile material	U-235, Pu Mixtures
Fertile material	U-238, depleted U, Th-232
Types of fuel	metal, oxide, carbide
Reactor coolant	liquid metal (Na, NaK)
	gas (He, CO_2)
Reactor arrangement	integrated concept (pool)
	loop design.

(*) LWR *Light Water Reactor.*

The LMFBR by its characteristics can produce energy and simultaneously produce fuel in the greater quantities than it is consumed. The plutonium produced by existing LWR will be used in LMFBR and thermal reactor and fast breeder will probabily coexist for a number of years till FBR-FBR self fuel cycle will become possible.

3 – CALCULATIONAL METHODOLOGY

3.1 – Physical Model

In the plutonium-uranium fueled fast breeder homogeneous reactor the core consists of plutonium-uranium mixed fuel pins and it is subdivided into two enrichment zones to minimize the radial power peaking factor. Surrounding the core is a blanket of depleted uranium wich absorbs neutron leaking from the core and produces additional plutonium. Blanket consists of two parts, axial and radial. Cooling is accomplished by means of liquid sodium in the case of LMFBR and the plutonium loaded in the core is supplied by a light water reactor.

For the calculations the reactor was modeled in R-Z-geometry with two core zones, a radial blanket zone, a radial reflector zone, an axial blanket zone and an axial reflector zone and each zone was homogenized into equivalent annular rings. The control rods were negleted. The heterogeneous configuration will be described in section 5.

3.2 – Nuclear Properties Calculations

3.2.1 – Cross Section Set

Two computer codes were used to determine the reactor physics parameter. APOLLO[17] a two-dimensional, multigroup diffusion theory code and ANDROMEDA[16] a one-dimensional diffusion code.

All calculations were performed with 3 or 6 neutron energy groups collapsed from 25-group data library, the JAERI-FAST-SET 2. The collapsed energy groups were corrected for resonance self-shielding and spacial flux weighting.

3.2.2 – Methodology

The iterative scheme represented in Figure 3.2.1 was used to determine the beginning of life (BOL) fuel composition and the physical parameters of the reactor. The calculational procedure aimed at minimizing the power peaking factor while keeping reactor criticality condition at end-of-equilibrium cycle (EOEC).

The scheme starts with a guess of BOL fuel composition and the minimum peaking factor survey is realized with one-dimensional calculation. With the minimum peaking factor determined burnup is proceeded till the equilibrium cycle is achieved and the EOEC effective multiplication factor (K_{eff}) is checked. If it is not in the range of 1.005 ± 0.002 the BOL composition is revised. The iterations are repeated till BOL fuel composition obtained results in EOEC reactor composition that satisfies both the criticality and the flat power distribution.

In the burnup calculations the following sequence was considered in order to achieve the equilibrium cycle. A reactor fuel at BOL was burned one cycle, then 1/3 of core and axial blanket as well as 1/5 of radial blanket were reloaded and the fuel was burned for another cycle. This operation is repeated till equilibrium cycle is obtained.

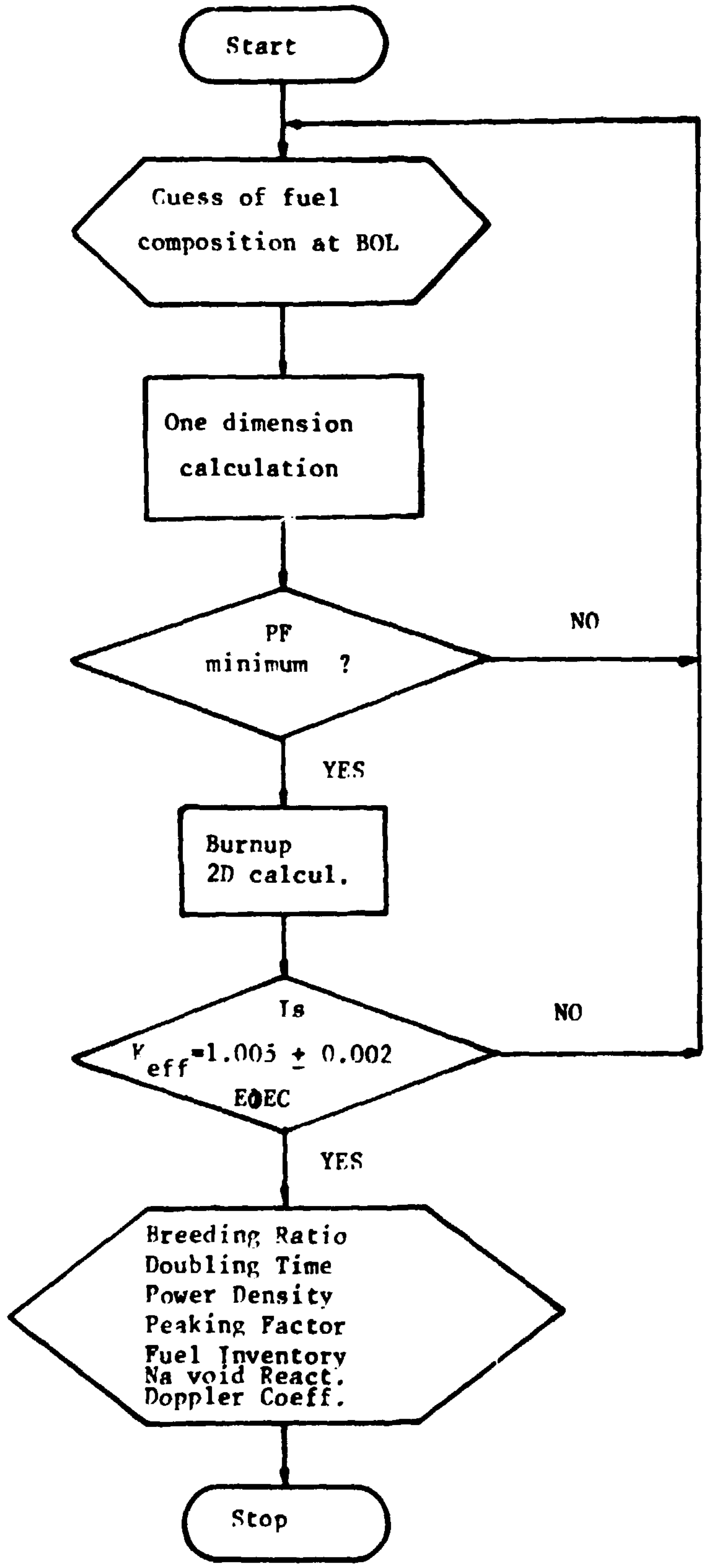

Figure 3.2.1 – Iterative Scheme for Physic Parameters Calculation.

3.2.3 – Breeding Ratio and Doubling Time

Breeding ratio and doubling time are important parameters to characterize breeder reactor. The breeding ratio is a measure of the production of fuel and doubling time is a measure of the time that it takes to produce fuel sufficient to start up another identical reactor.

The breeding ratio is derived from reaction rates integrated over the equilibrium cycle. For the region k breeding ratio is defined as

$$BR_k = \frac{\sum_{m2} \sum_{n} \int_{V_k} N_k^m \, \bar{\sigma}_{a,k}^{m,n} \, \phi^n(\bar{r}) \, d\bar{r}}{\sum_k \left(\sum_{m1} \sum_{n} \int_{V_k} N_k^m \, \bar{\sigma}_{a,k}^{m,n} \, \phi^n(\bar{r}) \, d\bar{r} \right)}$$

Where,

m1 : ^{235}U, ^{239}Pu, ^{241}Pu

m2 : ^{238}U, ^{240}Pu, ^{242}Pu

N_k^m : effective concentration of nuclide m in the region k

n : energy group

σ : microscopic cross section

ϕ : neutron flux

The total breeding ratio is the summation over all regions

$$BR = \sum_k BR_k$$

Doubling Time

A wide variation of expressions is currently in use in order to calculate the doubling time[14,2,32]. In this study the doubling time is calculated according to the definition presented in reference 2. The reactor doubling time (RDT) is defined according to

$$RDT(\text{year}) = \frac{\text{fissile BOEC}}{(\text{fissile gain}) \times (\text{fuel cycle/year})}$$

Where:

Fissile material considered are ^{235}U, ^{239}Pu, ^{241}Pu

Fissile BOEC : all fissile amount in the core and blankets at beginning of equilibrium cycle (kg);

Fissile gain : net increase in quantity of fissile during one fuel cycle (kg).

The reactor doubling time provides information about the breeding capacity of a particular reactor. As such it is useful for comparing various reactors breeding performance. On the other hand a full-cycle-inventory-doubling time (IDT) is defined to examine a reactors performance in terms of its overal fuel cycle. In addition it includes a term of fuel loss during fuel fabrication and reprocessing. Then the definition for IDT is

$$\mathrm{IDT(year)} = \frac{M_{in} + M_{ex}}{(G - L_p - L_d) \times \mathrm{cycles/year}}$$

With,

M_{in} : reactor fissile inventory at BOEC

M_{ex} : external cycle fissile inventory

G : fissile gain/cycle

L_p : fuel cycle loss (fabrication and reprocessing)

L_d : ^{241}Pu decay loss for the external cycle.

The components of above expressions are defined as:

External cycle fissile inventory

$$M_{ex} = M_{in} \times RF \times T_{ex} / T_{cycle}$$

With,

RF : refueling fraction

T_{ex} : external cycle time

T_{cycle} : cycle length

Fuel cycle loss

$$L_p = \text{fissile at MOEC} \times RF \times 0.02$$

^{241}Pu decay loss

$$L_d = {}^{241}Pu\,(\mathrm{MOEC}) \times RF \times 0.0462$$

With,

$$^{241}Pu\,(\mathrm{MOEC}): \frac{1}{2}\,(\mathrm{BOEC} + \mathrm{EOEC})\,^{241}Pu \text{ inventory}$$

In the case of a system containing a large number of breeders it would be convenient to define fuel-cycle-compound-system-doubling time (CSDT) related as

$$\mathrm{CSDT\ (year)} = 0.693 \times \mathrm{IDT}$$

3.2.4 – Sodium Void Reactivity and Doppler Coefficient

The transient characteristics of large plutonium fueled fast reactor core depend on two important parameters, ne Doppler Coefficient and Sodium void reactivity.

In a large plutonium fueled reactor the void effect usually adds positive reactivity whereas the Doppler effect is able to add prompt negative reactivity. A general problem of commercial LMFBR core design is to reduce the sodium void reactivity as low as possible.

Sodium Void Reactivity

In accident situations due to the sodium voiding the fuel temperature is drastically increased and the reactivity is increased due to the following effects:

a) Less neutrons are captured in sodium;

b) Neutron leakage from core is increased as a concequence of less scattering;

c) Spectrum hardening.

Effect (b) diminishes, effect (a) and (c) normally increase the reactivity.

A set of six neutron energy group cross section was used in two dimensional configuration and sodium void reactivity was determined from direct eigenvalue calculations. The effective multiplication factor (k_{eff}) was calcualted with sodium-in and with sodium-out conditions at end-off-equilibrium cycle.

sodium void reactivity = $\rho_1 - \rho_0$

Where,

ρ_0 : reactivity at normal condition

ρ_1 : reactivity at sodium voided condition

Doppler Coefficient

In the case of the Doppler coefficient, broadening of fission resonance in fissile material increases the reactivity whoreas broadening of capture resonance in both fissile and fertile materials decreases the reactivity. For large fast reactors, in which the ratio of ^{238}U to ^{239}Pu is high, the broadening with fuel temperature increasing results in a significant negative value for the Doppler coefficient.

For Doppler coefficient calculation the core temperature was changed and the variation in the reactivity was computed.

3.3 – Fuel Element Temperature Profile

The LMFBR has a higher inlet temperature and higher specific heat generation rate than a LWR. Both factors tend to yield higher fuel temperature. The coolant sodium, however, has a good heat transfer characteristics which permit to remove the heat produced without producing local boiling or significant changes in its physical properties at normal operation condition. Table III.3.1 lists the main thermo-hydraulic properties of sodium[24].

Table III.3.1

Thermo-Physical Properties of Sodium (Liquid)[24]

Melting Temperature	°C	97.82
Boiling Temperature	°C	881.0
Melting heat	cal/g	27.08
Boiling heat	cal/g	956.6
Density	g/cc	
at 97.98 °C		0.9275
at 400 °C		0.8563
*Specific heat	cal/g °C	0.312
*Heat conductivity	kcal/m h °C	66.0
*Viscosity	kg/m sec	3.4E - 04

* at 300 °c

The temperature profile in the maximum power channel is the prime importance in reactor design due to the fact that reactor power capacity is limited by its thermal transport capacity. A reactor core must be operated in such way that the temperature of the fuel and the cladding anywhere in the core must not exceed safe limits. The temperature calculations were performed using the program TASS (Thermohydraulic Analysis at Steady State)[19] and the following conditions were assumed:

- Reactor power fixed;
- The coolant is in the steady-state;
- The coolant does not suffer changes in phase.

3.3.1 – Axial Temperature Distribution

Energy is generated in the fuel volume at a rate depending on the fission rate. The variation of the power density in the axial direction is a pure cosine function. The axial coolant temperature distribution changes along the channel in response to both the change in energy source distribution along the fuel pin rod, and the rise in coolant temperature through the core

In a fuel the peak temperature tends to follow the heat generation pattern with the maximum occuring at the mid-height to the fuel pin rod. However the coolant flows from botton to upper core, so the coolant temperatures increase along the rod and its maximum occurs at the top of the channel. The axial variation of the power density is given by the equation.

$$q''' = q_0''' \cos(\pi z/H)$$

where q''' and q_o''' are the volumetric heat source at any point and at center of the fuel element, respectively.

The numerical solution for steady-state heat transfer in rod bundle was evaluated using a triangular lattice model and the lumped parameters method[25,33]. In the lumped parameters method, often referred to as subchannel analysis, the subassemly is divided in a number of subchannels whose boundaries are defined arbitrarily by surfaces of fuel element. Axial coolant temperature in each subchannel is evaluated by solving equations of continuity, conservation of energy and the entalphy definition, as follow:

a) Equation of Continuity

$$\frac{\partial(\rho v)}{\partial z} = 0$$

b) Equation of Conservation of Energy

$$\rho v \times \frac{\partial h}{\partial z} = q'''$$

c) Entalphy

$$\int_{100}^{T} C_p\,(T_0)\,dT_0 = h$$

Where,

ρ = density of coolant	(Kg/m^3)
v = velocity of coolant	(m/sec)
z = axial position	(m)
h = entalphy	(Kcal/Kg)
q''' = power density	($Kcal/m^3$ sec)
C_p = specific heat	(Kcal/Kg °C)

The power density was obtained from the neutronic calculations described in section 3.2.2. The coolant density and specific heat were considered dependent of temperature as are shown by following equations[12]

a) Density

$$\rho(T) = 949.0 - 0.223T - 1.76 \times 10^{-5}\,T^2 \quad (Kg/m^3) \qquad T \longrightarrow K$$

b) Specific Heat

$$C_p(T) = 0.3893 - 1.9908 \times 10^{-4}\,T - 11.0542 \times 10^{-8}\,T^2 \quad (Kcal/Kg\,^\circ C) \quad T \longrightarrow K$$

3.3.2 – Radial Temperature Distribution

Radial temperature distribution for a cylindrical fuel pin rod was calculated. According to the heat transport process, the heat flow path consists of the following two steps:

a) Heat transfer by conduction through heat-generation fuel, gap and cladding;

b) Heat transfer by convection to the coolant.

In the calculations, the thermal conductivity of the fuel (k_f) and the cladding (k_c), as well as the physical properties of the coolant (density, viscosity, specific heat) were assumed dependent of temperature. Such dependence are shown in the following equations[10].

· Thermal Conductivity

Oxide fuel

$$k_f = 1.10 \times 10^{-2} \times \frac{1}{t(0.4848 - 0.4465\,D)} \quad (\text{Kcal/m sec } ^\circ\text{C})$$

Where, D = actual fuel density / theoretical fuel density

· Carbide fuel[8]

$$k_f = \frac{56.2773}{t + 273} + 0.761 \times 10^{-4}\,(t + 273) \quad (\text{Kcal/m sec } ^\circ\text{C})$$

· Cladding material

$$k_c = 0.1 \times (0.03066 + 0.3584 \times 10^{-4}\,t - 0.0042 \times 10^{-6}\,t^2) \quad (\text{Kcal/m sec } ^\circ\text{C})$$

· Viscosity

$$\text{Log}\,\eta = 0.5108 + \frac{220.65}{t + 273} - 0.4925 \log (t + 273)$$

From the point of generation, the heat flows through fuel material, through a gap between fuel and cladding material, through cladding material to an interface with a coolant and finally through a portion of the coolant which will transport the heat from the core.

At a given power level, the temperature in the fuel depends both on the temperature gradient through the various materials and on the bulk temperature of the coolant at point considered, along the length of the fuel element under study.

Starting with the coolant temperature, which may be considered as a reference temperature, and using known temperature gradients it is possible to determine the fuel temperature profile.

The temperature gradient is represented by Fourier heat flow equation.

$$\frac{dT}{dx} = -\frac{1}{k}\,\frac{q}{A}$$

In the steady-state with no heat generated in cladding or coolant and with negligible resistance to heat flow at the fuel-cladding interface, the fuel element radial temperature distribution can be calculated using following expressions:

At cladding outer surface

$$t_{c1} = \frac{q''' \quad R^2}{2\,(R + c + g)H} + t_f$$

At cladding inner surface

$$T_{c2} = \frac{q''' \quad R^2}{2k_c} \; \ln \frac{(R + g + c)}{R} + t_{c1}$$

At fuel pellet outer surface

$$t_s = \frac{q''' \quad R^2}{2h_g} + t_{c2}$$

At fuel pellet center

$$t_p = \frac{q''' \quad R^2}{4k_f} + t_s$$

Where,

q''' : power density

R : fuel pellet radius

c : cladding thickness

g : gap thickness

h : heat transfer coefficient of coolant

k_f : thermal conductivity of fuel material

k_c : thermal conductivity of cladding material

4 – HOMOGENEOUS CONFIGURATION CORE

4.1 – Introduction

Present commercial fast breeder reactor developed or under development utilize the mixed oxide fuel $(Pu, U)O_2$. However such reactors present a long doubling time, ranging from 20 to 30 years, even with some design parameters changes introduction.

However the modern nuclear power strategy requires fast breeder with a short doubling time, around 10 years to supply the required future energy demand. Thus recent events have led to renewed interest in advanced fuels.

Generally advanced fuels refer to those which are different from oxide uranium or uranium-plutonium fuel and possess promising and better realization of fast reactor potential breeding capabilities. They include carbide, nitride, phosphide and, in principle, metallic fuel.

Main advantages of auvanced fuel compared to oxide ones are higher density, higher thermal conductivity and smaller content of neutron moderating and absorving material. An important feature of advanced fuel is the possibility to increase the breeding in the core.

Using carbide in commercial fast breeders following improvements on reactor performance can be expected:

- Higher rod power;
- Higher breeding ratio;
- Lower doubling time;
- Initial plutonium inventory is smaller.

In this section the design characteristics and performance of plutonium-uranium carbide fuel are compared with those of plutonium-uranium oxide fuel in a large fast liquid metal cooled reactor through analytical studies of nuclear and thermal performance.

In spite of early study of carbide utilization in fast reactor, its development and use has fallen a behind the development of oxide fuel utilized in water moderated reactors[21,10,15].

Recently FBR fueled with carbide became attractive subject in many countries and most of the national programs are in the process of evaluating the merits of these core concepts[13,22,31,30].

Two primary advantages for using carbide fuel in place of oxide are:

- The density carbide is approximately 30% higher than that of oxide. This higher density reduces the required concentration of fissionable material improving the breeding performance;
- The thermal conductivity of carbide is substantially higher than oxide, approximately five times, and slight variation with temperature is observed. It permits larger fuel rod diameter meaning lower fabrication cost.

A summary was made of fuel carbide properties available in the literature[21,15] to supply necessary information to the calculations. These data are listed in Table IV.1.1.

Works are carried out in the direction of improving advanced fuel fabrication technology. At present, the experience in advanced fuel investigation is limited to tests in experimental reactors[18,35,11,5,6].

Table IV.1.1

Properties of Carbide and Oxide Fuels[21]

Characteristics \ Fuel Type	OXIDE			CARBIDE		
	UO_2	PuO_2	$(Pu,U)O_2$	UC	PuC	(Pu,U)C
Melting point (°C)	2730	2300	2700	2400	1850	2270
100% theoretical density (g/cc)	10.96	11.46	10.87	13.63	13.62	1360
Percentage of elements as (O,N,C,...) (w/o)	11.8			4.80		
Thermal conductivity (W/cm)						
at 500 °C	0.047			0.16		
1500 °C	0.025			0.17		
1000 °C			0.027			0.177

4.2 – Fuel Pin Diameter Optimization

The performance of plutonium-uranium carbide fueled LMFBR was analyzed and compared with the results previously obtained from the study in the case of plutonium-uranium oxide fueled reactor[26]. The main parameters of comparison are breeding ratio, specific fissile inventory, doubling time, sodium void reactivity and Doppler coefficient.

Initially using doubling time as the parameter, an optimum pin diameter was determined for homogeneous core. The Table IV.2.1 lists the fixed design parameters for the cores analyzed. For each pin diameter burnup calculation was carried out and doubling time at equilibrium cycle was calculated. The doubling time, breeding ratio, core volume and material composition variation with fuel pellet diameter are illustred in Figure 4.2.1. From this figure one may observe that the minimum doubling time occurs at the fuel pellet diameter ranging from 0.85 to 0.95 cm whereas the optimim range for oxide fuel is about 0.60 to 0.70 cm[28].

4.3 – Nuclear Characteristics Analysis

4.3.1 – Reactor Core Geometry

The design data of carbide fueled homogeneous core considered in this study are listed in Table IV.3.1. The reactor core contains 313 hexagonal fuel assemblies arranged in two radial core zones of different fuel enrichment. Differente radial core enrichment is usually applied in order to flatten the radial flux profile.

Hexagonal fuel element contains 169 fuel pins, 10.5 mm outer diameter, with core fuel (Pu, U) C in the middle and fertile fuel (UC) at the both ends. The radial blanket assembly contains 61 pins and the design for fuel rods is basically the same as that for the oxide.

The primary sodium is pumped upwards through the fuel element and heated from 395 to 545°C. In the carbide reactor the coolant fraction must be high. This is necessary due to the high power density for this fuel. Otherwise, on account of high thermal conductivity a more compact core is possible generating the same total thermal power of that of oxide fuel.

In the ractor physics analysis, one and two dimensional diffusion calculations were carried out using 3 and 6 energy group cross section. For burnup calculation the following considerations are assumed

- The loaded core fuel is plutonium from a light water reactor discharged fuel;
- The fuel supplied for the blankets is depleted uranium and was assumed for this study to be 99.80 w/o U-238 and 0.20 w/o U-235;
- The reactor load factor is 0.82;
- The core fuel is irradiated at fixed locations. The core presents 3 refueling batchs and the cycle length is 300 days;
- 1/5 of radial blanket assemblies is changed each year.

The isotopic composition of the light water reactor plutonium is sensitive to the discharged exposure and is also influenced by the particular reactor spectrum.

For this study the isotopic composition of plutonium is follows:

ISOTOPE	WEIGHT PERCENT(w/o)
^{239}Pu	63.0
^{240}Pu	22.0
^{241}Pu	12.0
^{242}Pu	3.0

Table IV.2.1

Fixed Design Parameters

Total reactor power (MWth)	3000
Core height (cm)	100.0
Thickness of each axial blanket (cm)	30.0
Thickness of each axial reflector (cm)	20.0
Radial blanket thickness (row)	3
Fuel pellet density (%TD)	85/95
Maximum burnup (MWd/T)	1.0×10^5

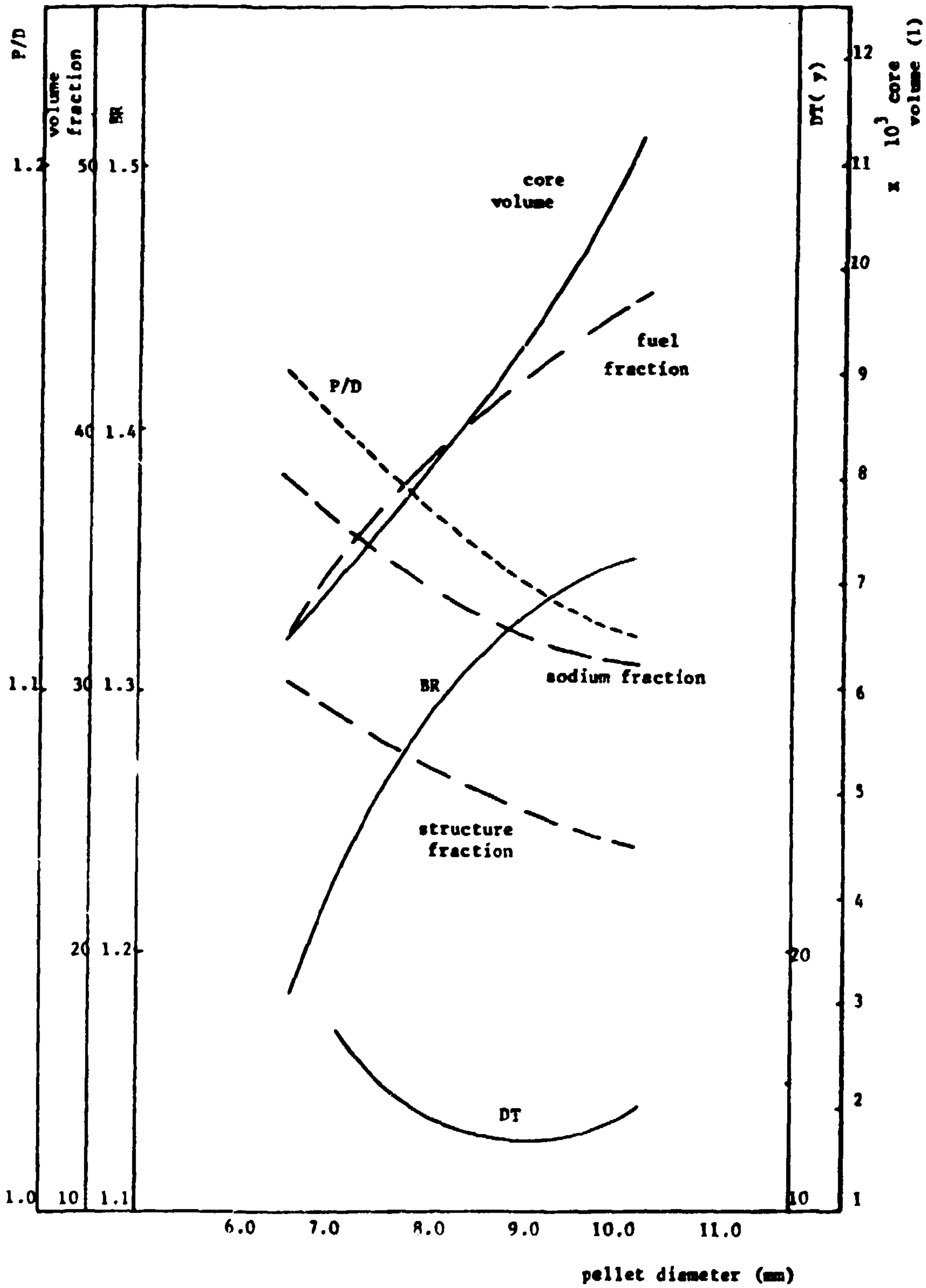

Figure 4.2.1 – RDT, BR and Material Composition as Function of Pellet Diameter (Carbide fuel).

Table IV.3.1

Reactor Characteristics
(carbide fuel)

Reactor thermal power	MWth	3.000
Reactor electrical power	MWe	1.200
Coolant		Na
core inlet temperature	°C	395
core outlet temperature	°C	545
Core height	cm	100.0
Axial blanket (upper/lower)	cm	30.0/30.0
Axial reflector	cm	20.0
Core diameter	cm	331.0
Radial blanket	row	3
Radial reflector	row	2
Number of assemblies		
inner core		169
outer core		144
radial blanket		216
reflector		176
Fuel cycle lenght	day	255
Load factor		0.70
Number of refueling batches (core/radial blanket)		3/5
Fuel assembly design		
fuel material		(Pu, U)C
density	% TD	85
pins per assembly		169
lattice pitch	cm	17 83
fuel pin OD	cm	1.05
fuel pellet OD	cm	0.902
cladding material		SS 316
cladding thickness	cm	0.07
wire diameter	cm	0.12
Radial blanket assembly design		
blanket material		(UC) dep.
pis per assembly		61
pin OD	cm	1.85
pellet OD	cm	1.72
cladding thickness	cm	0.065
fuel density	% TD	95
wire diameter	cm	0.095

Applying the iterative method described in section 3 the fuel composition at the beginning of life was determined and equilibrium cycle search was carried out. For this reactor the equilibrium cycle was achieved after 4 cycles. For doubling time calculations the expressions defined in section 3 was utilized considering also the specifications listed in Table IV.3.2.

Table IV.3.2

Input Data for Doubling Time Calculation

Refueling fraction	0.33
Processing loss fraction	0.02
External cycle time	1.0 year
Fuel cycles / year	1.0
Cycle length	300 day
Annual load factor	0.82
^{241}Pu half-life	14.7 year
Fissile material	^{235}U, ^{239}Pu, ^{241}Pu

4.3.2 – Nuclear Performance

On the basis of the results the following comparisons with oxide fuel[20] are important:

Higher breeding ratio	
oxide fuel	1.18
carbide fuel	1.42
Shorter compound system doubling time (year)	
oxide fuel	39.0
carbide fuel	15.7
Lower fissile inventory (ton)	
oxide fuel	4.607
carbide fuel	3.952
Higher linear power (W/cm)	
oxide fuel	439.3
carbide fuel	948.5
Smaller core volume (litre)	
oxide fuel	10,521
carbide fuel	8,615
Larger pin diameter (cm)	
oxide fuel	0.85
carbide fuel	1.05
Smaller burnup swing (% Δ k)	
oxide fuel	2.86
carbide fuel	0.16

4.3.3 – Cladding Thickness Effect on Breeding Ratio

It is recognized that there is a potential advantage in breeding by decreasing the cladding thickness and the benefits of using thin can are quantified in this section.

The cladding thickness of 0.45 mm was assumed and new core dimensions as well as the fuel composition were determined. An increase in the fuel volumetric fraction is verified in consequence of thinner cladding, thus a lower fuel concentration is required. A significant improvement on breeding performance was observed and a reactor doubling time around 10 years was achieved.

4.4 – Safety Characteristics

In this section the sodium void and Doppler effects in carbide fueled homogeneous reactor are analyzed and the differences in these two safety parameters for oxide and carbide reactors are discussed. Table IV.4.1 gives the sodium void reactivity and Doppler coefficient for the two reactors.

Table IV.4.I

Sodium Void Reactivities and Doppler Coeficients
for Oxide and Carbide Fueled Reactors

OXIDE FUEL[20]			
	BOL	BOEC	EOEC
Sodium void reactivity, total core, Δk	0.0199	0.0238	0.0261
Doppler coefficient total core, $T\frac{dk}{dt} \times 10^4$	- 92.0	- 68.0	- 68.0
CARBIDE FUEL			
	BOL	BOEC	EOEC
Sodium void reactivity total core, Δk	0.0224	0.0260	0.0306
Doppler coefficient total core, $T\frac{dk}{dt} \times 10^4$	- 64.0	- 59.0	- 42.0

Obs.: Isothermal Doppler for sodium in core:

$1000 \longrightarrow 2100$ K for oxide and $1000 \longrightarrow 1800$ K for carbide.

4.4.1 – Sodium Void Reactivity

Sodium void reactivities were calculated by eigenvalue differences assuming total reactor voiding. Carbide reactor presents higher sodium void reactivity than that of the oxide reactor for entire reactor life time. The increase in sodium void reactivity with burnup are also greather for carbide fuel.

The reasons for these differences are:

- Carbide reactor has a higher heavy-metal concentration;

- Carbide reactor presents a greater sodium volume fraction.

Sodium void effect increases with burnup due to the high plutonium isotope production. As can be observed at beginning of life (BOL) the sodium void reactivity is low for both fuels.

4.4.2 – Doppler Coefficient

The primary negative feedback needed for reactor stability is derived from the Doppler effect. This effect is influenced by neutron spectrum, fertile-to-fissile ratio and heavy metal concentration.

The carbide reactor considered has a higher fertile-to-fissile ratio and heavy metal concentration, but its spectrum is harder than oxide reactor. Consequently the Doppler coefficient of the carbide reactor is less negative.

Another difference between these two fuels is the Doppler coefficient behaviour with burnup. While the Doppler coefficient of the oxide reactor remains unchanged with burnup that of the carbide reactor is greatly reduced. This effect shows how the fertile-to-fissile ratio in the carbide reactor changes with burnup.

Doppler effect is defined as change in reactivity due to the rise in average reactor fuel temperature and it is determined by direct eigenvalue calculations.

4.5 – Thermal Characteristics

If fuel melting point is a temperature limit in reactor design, carbide will have a lower peak temperature than the oxide fuels. Melting point of the oxide carbide is approximately 2400°C. However since PuC, in particular, enters to the vapor phase before the melting point is reached, limitation of the fuel center temperature to a maximum of about 1800°C is feasible for technical reasons[21].

Despite of low permitted temperature for carbide fuel, the higher thermal conductivity allows one to design larger fuel pin diameter. To keep the fuel integrity, temperature distributions in a fuel pin were calculated using the expressions described in section 3. The sodium outlet temperature was set to 545°C and with power density obtained from nuclear determined. The fuel pin taken is that located at the point where the power generations maximum in the reactor core. From the results it was found that the temperature at center of fuel and cladding surface are 1309°C and 868°C respectively, then they do not exceed the limiting temperature.

4.6 – Results

Comparison of the plutonium-uranium carbide cores with the plutonium-uranium oxide cores shows that the breeding performance of the former is superior. With the carbide fueled homogeneous reactor a breeding ratio of 1.49 and a doubling time of 11.6 years result, while the best values for oxide fueled reactor are 1.27 and 19 years, respectively. Otherwise, the sodium void reactivity is larger and Doppler coefficient has a lower negative value for carbide fueled reactor.

To make good use of the carbide fuel breeding performance and to ensure low positive sodium void reactivity a carbide fueled heterogeneous configuration is recommended.

5 – HETEROGENEOUS CONFIGURATION CORE

5.1 – Introduction

One of the primary motivations of developing the heterogeneous core configuration of a LMFBR is its increased breeding ratio, reduced sodium void reactivity worth and reduced hypothetical disruptive accident energy.

It is known that there is a not just one heterogeneous reactor but a great variety of heterogeneous cores depending on the arrangement of the internal blankets assemblies into the core zone. We can mention the heterogeneous core with the axial internal blankets (axial heterogeneous core); the heterogeneous cores with the radial internal blankets (the radial heterogeneous core)[34,8,1] and the modular island core[29].

A systematic methods for designing heterogeneous configuration having a low value of sodium void reactivity is presented in the ref[34] and its conclusion is that among several core configurations the heterogeneous one that consits of sucessive radial core and blanket zones (radial heterogeneous core) are very promising.

The arrangement of the internal blanket assemblies in the core is of great significance to the performance of the reactor. A basic characteristics of heterogeneous cores is then the degree of neutronic coupling between different core zones. As the thickness of the internal zones increases the neutronic coupling among the core zones decreases and this increases the reactivity of the power distribution.

The comparative studies[1,29,39] between heterogeneous and homogeneous configurations have shown higher breeding performance and lower positive sodium void reactivity for to the former core. Several experiments for sodium void and physics properties measurements have been made in experimental reactors such as ZPPR and Masurca[23,7].

Bayley[4] suggests that the relative tightly coupled (no more than two-row blanket width) radial heterogeneous cores with optimum fuel pin design provide a better balance between the non-energetic core disruption safety feature and the economic performance.

The heterogeneous core concept has been widely investigated since last years and the applicability of this concept for large commercial reactors has been evaluated currently by various laboratories. In spite of the advantages presented by heterogeneous configuration there is not a definite plane to introduce into the planned reactors. The breeder program in the world has been concentrated on the mixed uranium-plutonium oxide fuel.

In this section in order to evaluate the carbide fueled radial heterogeneous core performance, the characteristics of three configurations were investigated, and the results compared with those of the oxide fueled ones.

5.2 – Carbide Fuel

5.2.1 – Reactor Configurations

The reactor configurations analyzed in this section are shown in Figure 5.2.1, where the case (2-1) and case (3-1) are tightly coupled configurations while case (2-2) is loosely coupled.

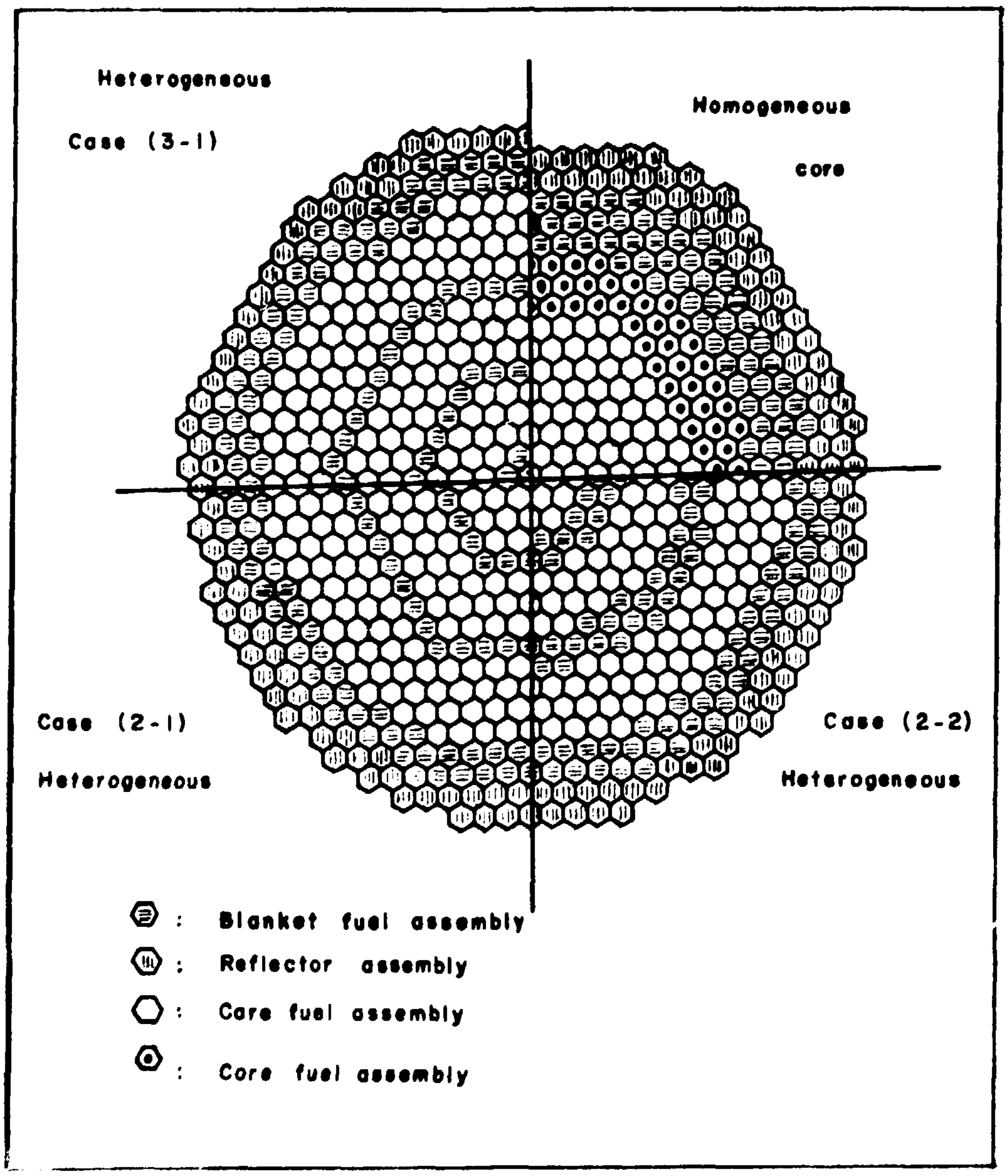

Figure 5.2.1 – Carbide Fueled LMFBR Core Layout Considered.

The optimum pin diameter determined for homogeneous reactor was utilized in all heterogeneous calculations. In addition the internal and radial blanket assemblies are identical in design. The core height and the fuel assemblies design were fixed for all the configurations considered but for case (2-2) two pin design were selected, one conventional with 0.70 mm cladding thickness and other with 0.40 mm.

Case (2-1) and (3-1) have same internal blanket volume ratio of 18% while case (2-2) has higher volume ratio, approximately 28%. In all the configurations the refueling of the core zones and internal blanket assemblies is performed each year with 1/3 of the assemblies replaced. The radial blanket has a fuel cycle length of 5 years.

5.2.2 – Nuclear Characteristics

The calculational method and the cross section data employed for the nuclear characteristics calculations were the same as those utilized in homogeneous core.

The generation of the equilibrium cycle was performed by the two dimensional R-Z geometry three energy group burnup calculations. The plutonium loads are those discharged from a LWR.

Comparisons are made with those of the oxide fueled heterogeneous reactor. The data of oxide fueled heterogeneous reactor used for comparative study were taken from the work developed by Konomura[19].

An analysis of the results obtained resulted in the following conclusions:

- Breeding Ratio

Small or no difference was seen in the breeding ratio for the three heterogeneous configurations considered but when compared with homogeneous configuration an increase of 3.5% was verified. This gain in breeding is primarily the result of the introduction of the greater fertile mass of the blanket assemblies and its effect on the neutron importance. The neutron spectrum in the heterogeneous design has higher average energy that tends to improve breeding performance.

On the other hand when compared with oxide fueled heterogeneous reactor a substantially increase in the breeding ratio is observed emphasizing the good nuclear properties of the carbide fuel.

- Reactor Doubling Time

In spite of the high breeding ratio, the reactor doubling time is larger in heterogeneous core because high fissile inventory required to makes reactor critical. But 16 years of the carbide fueled reactor compared with 32.7 years to the oxide fueled ones represent a significant reduction, nearly half of the doubling time.

- Burnup Swing

Generally the reactivity change with burnup is smaller in heterogeneous cores than homogeneous one for the same fuel pin diameter. The degree of neutronic coupling in a radial heterogeneous configuration affects the burnup swing. For the same pin diameter the burnup swing is smaller in tightly coupled core. For the same configuration using oxide fuel results in a higher burnup reactivity due to the smaller fuel pin diameter.

Changing fuel cladding thickness the breeding ratio will increase due to the fuel volume fraction increase. The configuration (2-2) was choosen by considering its low positive sodium void reactivity value, and assuming a cladding thickness of 0.40 mm the calculations were performed. A illustrated in Table V.2.1 a breeding ratio of 1.52 and reactor doubling time of 13 years were achieved for this case.

Table V.2.1

Reactor Nuclear Characteristics Case (2 - 2) — Results

(Cladding Thickness = 0.45 mm)

* Fuel enrichment	w/o	15.1/17.66/17.01
Fuel density	% TD	85
Cladding thickness	mm	0.45
Driver pin outer diameter	mm	10.0
Blanket pin outer diameter	mm	18.0
Internal blanket volume ratio		0.28
Fissile plutonium initial inventory	ton	5.017
Breeding ratio		1.53
Reactor doubling time	year	13.0
CSDT	year	14.5
Peaking factor		1.68
Maximum burnup	MWd/ton	9.103E 04
Core outer radius	cm	192.0
Fuel volume fraction		0.42
Na void reactivity (10^{-2})		1.09
Doppler coefficient ($T \frac{dk}{dt}$) x 10^4		- 71.0

Obs.: * means fuel concentration.

5.2.3 – Safety Characteristics

Sodium void reactivity and Doppler coefficient were calculated at EOEC as inherent reactor safety related parameters and listed in Table V.2.2.

Table V.2.2

Doppler Coefficients and Na Void Reactivities at EOEC

	CARBIDE				OXIDE	
	HETEROGENEOUS			HOMOG.	HETEROG.	HOMOG.
	(2 – 1)	(3 – 1)	(2 – 2)		(2 – 2)	
Doppler coefficient						
Core	-0.0058	-0.0062	-0.0061	-0.0073	-0.0087	-0.0096
Core + AB + IB	-0.0070	-0.0074	-0.0073	-0.0078	-0.0112	-0.0103
Na void reactivity						
Core + IB	0.0181	0.0167	0.0098		0.0093	
Core + AB + IB	0.0236	0.0224	0.0144	0.0306		0.0260

The sodium void reactivity is smaller in the heterogeneous core than in homogeneous core by roughly a factor of two. All the sodium void reactivities have been calculated from direct k_{eff} calculations for the voided and unvoided reactor. Both fuel and blanket (axial and internal) zones were considered voided.

The sodium void decrease for the radial heterogeneous core is mainly caused by the increment of the leakage neutron from the core into the internal blanket. Thus sodium void reactivity depends on the size of the core zones and the thickness of the internal blankets. For thick internal blanket the sodium void is reduced as observed for the case (2-2).

The Doppler coefficient at core zone is smaller for heterogeneous core due to its high fissile enrichment. For heterogeneous reactor an additional Doppler feedback is verified in internal blanket zones, thus considering the entire core small defference is observed between heterogeneous and homogeneous configurations.

5.2.4 – Results

Three heterogeneous core configuration have been analyzed. The nuclear characteristics are quite equal for all the cases but the sodium void reactivity is lower in the loosely coupled case.

The breeding ratio increased nearly 3.5% doubling time reduced approximately 12% relatively to the homogeneous core.

The fissile inventory of heterogeneous core is 30 to 50% higher than that of the homogeneous design and for the same residence time average and peak burnup are lower in heterogeneous design. Comparing the burnup swing for same fuel pin diameter, it was found that the burnup swing is smaller for heterogeneous case and this reduction depends on the neutronic coupling. For the tightly coupled configuration the burnup swing is smaller than that for loosely core. When compared with oxide fuel heterogeneous configuration a significant improvement on breeding performance was verified. For the same configuration an increase of 16% is attained for breeding ratio and it is near 16 years shorter.

For heterogeneous configuration the degree of neutronic coupling affects the sodium void reactivity and it is lower for loosely coupled core. The reduction in sodium void reactivity relatively to the homogeneous one is 30 to 60%.

Although the thick internal blanket zone induces a low sodium void reactivity, great decoupling reduces core performance. Consequently, to design heterogeneous core the compromise between sodium void effect and core performance must be analyzed.

From the results of the present study we conclude that the configuration (2-2) with thin cladding, 0.40 mm thickness, can offer a short doubling time of 13.0 years and low positive sodium void reactivity.

5.3 – Summary of Carbide Fuel Development Programs

In sections 4 and 5 of this work the good performance of carbide fuel was demonstrated. The high heavy metal density of the carbide fuel allows a better breeding ratio than with the oxide fuel, and in addition, the good thermal conductivity permits a higher linear power, in consequence a shorter doubling time is achieved.

However, in order to use carbide fuel to improve on the oxide fuel economics a number of difficulties must be overcome.

The main areas of requered development are:

1 - Development of fabrication techniques

For fabrication of mixed carbide fuel, various procedures exist[14,21,22] but a real fabrication route on an industrial scale is not yet developed. The fabrication process of carbide fuels requires special facilities for handling the fuel in low oxigen and low moisture atmosphere.

Therefore a comparison of fabrication between mixed oxide and mixed carbide is not possible today, taking into account the fact that the state of technology reached in mixed oxide fabrication is much more advanced.

2 - Cladding material development

As oposite to the oxide fuel, the compatibility of carbide fuel with cladding material like as stainless steel constitutes a serious problem in the design of a fuel element. A new material like vanadium alloy is considered[15].

3 - Demonstration of high exposure - high temperature irradiation

Demonstration of the satisfactory behaviour of (Pu, U)C fuel is required for exposure of approximately 100,000 MWD/t. The fuel swelling and creep must be investigated.

4 - Fuel reprocessing technology

Reprocessing for (Pu, U)C fuel has still to be developed.

The irradiation experiments performed for the carbide fuel in experimental reactors as Rapsodie and DFR and EBR-II, were planned to investigate the behaviour of fuel pin, the fuel swelling, fission-gas release, fuel/cladding compatibility. Several programs for the carbide fuel development are in evolution.

Since 1968 a research program has been performed in Germany[11]. The main objectives of this program are: development of a suitable fuel fabrication process; basic research to obtain specific material data for irradiation creep and swelling of (Pu, U)C; planning, design, construction and performance of irradiation experiments; post-irradiation examination and evaluation data. Introduction of carbide fuel into SNR[18] are also being considered.

The objective of French irradiation program[22] is to determine the satisfactory behaviour of carbide fuel under high exposure. They hope to substitute mixed carbide for mixed oxide as Rapsodie driver fuel.

Up to now He bonded and Na bonded fuel rod concepts are actively tested in experimental programs[22,30,18]. For carbide fuels, burnup up to 70,000 MWD/t have already be obtained without failure[30]. The experimental results showed also that the carbide fuels suffer more swelling than the oxide ones and that the degree of swelling depend very much on the fuel center temperature[22].

Irradiation testing of mixed carbide fuel however lags considerably behind that of the oxide fuels. The urgent need for an accelerated carbide fuel testing program is clear.

6 – CONCLUSIONS

In the proceding sections a variety of specific conclusions have been reached concerning the breeding performance of the several fast breeder core configurations.

A major conclusions of this work are summarized below:

1 - Homogeneous Reactor

Comparisons between oxide and carbide fuels show that the breeding performance is improved significantly with the carbide fuel. With this fuel a breeding ratio of 1.49 and reactor doubling time of 11.8 years are achieved while the best values for the oxide fuel are 1.27 and 19 years respectively.

Regarding the safety parameters, carbide fueled homogeneous reactor presents a high positive sodium void reactivity (0.0306). Thus to make good use of the carbide fuel breeding performance ensuring a low positive sodium void reactivity a heterogeneous configurations are favorably recommended.

2 - Heterogeneous Reactor

Among the heterogeneous configurations analyzed the loosely coupled core presents a lower sodium void reactivity (case (2-2)) without penalizing the breeding performance. For this case the doubling time achived was 13.0 years and maximum sodium void reactivity is approximately 0.0098.

The carbide fuel in fast breeder reactor offers the promise of significant improvement in the reactor performance. But in order to evaluate the future potential of carbide, the characteristics of this fuel must be investigated in depth and the several problems listed in section 5 must be overcome. Due to the lack of experience with carbide fuel an accelerated testing program is required.

RESUMO

Objetivando-se minimizar o tempo de duplicação de um reator rápido comercial introduz-se combustível na forma de carbeto misto de plutônio e urânio e as suas características de regeneração são analisadas, incluindo-se entre estas, aquelas referentes à segurança inerente.

Com referência à segurança inerente, deseja-se que o reator apresente um coeficiente de vazio de sódio ("sodium void coefficient") negativo e, se positivo, o menor possível. Normalmente o tamanho e a geometria do núcleo de um reator de potência são ditados pelas condições térmicas, assim são também realizadas análises de perfil de temperatura.

Com combustível na forma de óxido misto de plutônio e urânio obteve-se 19 anos como menor tempo de duplicação de reator. Todavia passando-se à forma de carbeto esse tempo reduz-se para 12 anos, pelo fato do carbeto apresentar melhores propriedades térmicas relativamente ao óxido.

Quanto à segurança inerente obteve-se alto valor positivo para coeficiente de vazio de sódio (≃0,026) para a configuração homogênea. Por outro lado adotando-se a configuração em conceito heterogêneo esse coeficiente é diminuído para 0,0098 mantendo-se o tempo de duplicação relativamente pequeno, cerca de 13 anos, para o caso de carbeto.

Assim sugere-se o desenvolvimento de reatores rápidos em conceito heterogêneo com o emprego de combustível na forma de carbeto misto de plutônio e urânio por ser esta a combinação mais adequada tendo em vista os aspectos de regeneração e segurança.

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