OPTIMISATION OF THE GAS COOLED FAST REACTOR FOR PLUTONIUM AND MINOR ACTINIDE MANAGEMENT

T D Newton and P J Smith
AEA Technology plc
Winfrith, Dorset, England
tim.newton@aeat.co.uk ; peter.j.smith@aeat.co.uk

S J Crossley
BNFL plc
Springfields, Lancashire, England
steven.j.crossley@bnfl.com

R E Sunderland
NNC Ltd
Booths Hall, Cheshire, England
richard.sunderland@nnc.co.uk

ABSTRACT

Previous studies, based on the extensive UK experience with AGR technology, have demonstrated the feasibility of the gas cooled fast reactor (GCFR) to satisfy the design requirements envisaged for a large commercial liquid metal fast breeder reactor (EFR). Current LMFBR design studies, as given by the frame work of the CAPRA/CADRA project, are targeted to the effective management of plutonium and minor actinide stockpiles. The CAPRA reference oxide core design has shown that this objective can be readily achieved in a commercial scale sodium cooled fast reactor. However, an important limitation exists in this type of system due to the potential degeneration of the core safety parameters, and in particular the sodium void coefficient, when a wide variety of plutonium and minor actinide fuels are utilised.

The GCFR offers significantly improved opportunities for both plutonium and minor actinide consumption due to its low density gas coolant which provides a harder neutron spectrum. An additional benefit is the increased flexibility afforded by the negligible coolant void coefficient and thus the possibility to use a wider variety of plutonium and minor actinide fuels. To demonstrate the feasibility of the GCFR concept to satisfy current fast reactor design requirements studies have been undertaken to develop a viable GCFR fuel sub-assembly and core design. Thermal hydraulic and core performance calculations have been carried out and the results are presented and discussed in this paper. An illustration is also provided of the increased flexibility associated with this reactor concept.

1. INTRODUCTION

The studies presented in this paper demonstrate the possibility of using a gas cooled fast reactor to effectively manage plutonium and minor actinide stockpiles, while also showing the increased flexibility associated with this type of core concept when compared to existing sodium cooled fast reactor core designs, such as those developed within the framework of the CAPRA/CADRA[1] project. The origins of this work are based on a review[2] of the extensive UK experience with Advanced Gas Reactors (AGR's) which summarised the relevance of the different innovative concepts of AGR technology which could be adapted to meet the aims of more conventional sodium cooled fast reactor development. More recently the GCFR concept has been reviewed against the requirements of the EFR[3,9] project. One of the principal advantages of the gas cooled fast reactor concept has been identified to be the improved opportunity for the irradiation of poor quality plutonium and minor actinide fuels. This point relates to a major constraint imposed on the utilisation of sodium cooled fast reactors associated with the degeneration of the core safety parameters, and in particular the sodium void coefficient, when degraded plutonium fuels are utilised. The negligible coolant void coefficient in the gas cooled fast reactor ensures that the adverse effect on the core safety characteristics will be less severe. Another advantage of the low density coolant in gas cooled fast reactors, in this case CO_2 , is the absence of neutron scattering in sodium and the consequent softening of the neutron spectrum. The harder neutron spectrum means that the gas cooled fast reactor is more able to satisfy the current demands for the effective transmutation of the higher plutonium and minor actinide isotopes as well as certain long lived fission products. In addition there are economic and safety advantages of using a coolant such as CO_2 which is commonly available and has fewer coolant handling problems than the more reactive sodium which requires specialist handling and disposal.

The preliminary review of AGR technology concluded that the gas cooled fast reactor fuel pin and sub-assembly concept is very similar to that of existing sodium cooled cores, for example as given by the EFR design. However, as CO_2 is a poorer heat transfer medium than sodium it was recommended that a larger pin spacing, and consequently a lower fuel rating, hence requiring a considerably increased core volume, would be necessary in order to meet existing sodium cooled fast reactor design requirements. Hence an initial commercially sized gas cooled fast reactor core design was proposed, based on EFR design criteria, which demonstrated the feasibility of the gas cooled fast reactor concept.

Since these initial studies fast reactor options have evolved considerably from those typified by the EFR project, where the aim was to breed plutonium, to those today where the goal is to use fast reactors in an effective manner to reduce plutonium and minor actinide stockpiles. Hence this paper presents work that has been carried out to study a feasible gas cooled fast reactor design that satisfies the current requirements for the flexible management of plutonium stockpiles whilst remaining within the constraints imposed by existing conventional technology. The thermal hydraulic and neutronics performance of the proposed core design are presented and discussed. In addition, a demonstration of the flexibility of the core design to utilise a wide range of plutonium fuels is also provided.

2. BASIC CORE DESCRIPTION

The basis of the present work is the demonstration of a feasible design for a gas cooled fast reactor as a generator of electricity having a similar power output to that of a conventional commercially sized sodium cooled fast reactor while also meeting the same limitations on core design criteria and safety requirements. CO_2 is less effective than sodium as a heat transfer medium and so a larger pin spacing is required in order to satisfy the same design limiting criteria, which in turn results in a reduced fuel rating. Thus the basic core design was sized to accommodate as closely as possible the anticipated number of sub-assemblies required to satisfy the required power output while remaining within the CO_2 based fuel rating criteria. The proposed GCFR boiler design and the limitations imposed by existing fast reactor fuel technology have resulted in a core inlet and mixed mean core outlet temperature of 252C and 525C respectively with a coolant inlet pressure of 42 bar.

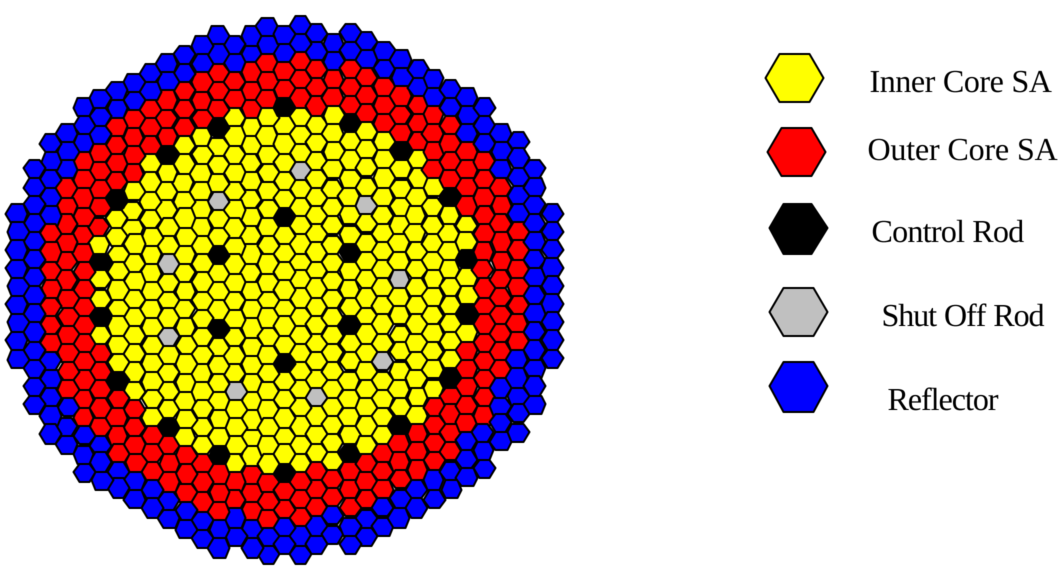

Figure 1. Basic Layout for the Gas Cooled Fast Reactor Core Design

In common with comparable sodium cooled fast reactor core designs, the gas cooled fast reactor core has a thermal output of 3600 MWth with a load factor of 80%. The reactor is assumed to have a net thermal efficiency of ~40% which therefore provides an electrical output of ~1400 MWe. As the core height is not constrained by limitations imposed by the coolant void coefficient, as in the case in a sodium cooled fast reactor, a core height of 1.5m has been chosen. Based on these assumptions, the gas cooled fast reactor core layout assumed for all of the core configurations examined during the current studies is shown in Figure 1.

The core consists of 550 fuelled sub-assemblies on a pitch of 180.61 mm divided into two enrichment zones, 334 sub-assemblies in the inner core zone and 216 sub-assemblies in the outer core zone. The reactor contains no axial or radial breeding regions and the reflector material adjacent to the core fuel, as well as the outer shielding sub-assemblies, are as currently assumed in comparable sodium cooled core designs. As no detailed data has yet been established for the axial regions above and below the core they have been represented in a simplified manner as regions of gas and steel. The control and shut off rods are the same as those of equivalent sodium cooled core designs and comprise 24 primary control rods and 6 secondary shut off rods. The rods are made up of two separate regions, an upper absorber section containing boron carbide pins 90% enriched in ^{10}B, and a lower follower region.

These basic core design parameters are common to both the preliminary gas cooled fast reactor core design based on EFR criteria, as presented in section 4 of this paper, as well as the gas cooled fast reactor core design that has been optimised for effective plutonium and minor actinide management that forms the main object of this paper, as presented in section 5.

3. MODELLING AND METHOD APPROXIMATIONS

3.1 THERMAL HYDRAULICS

In order to arrive at a feasible fuel pin and sub-assembly design which satisfies the constraints imposed by the current design criteria of the European Fast Reactor project while taking into consideration the relevant aspects of AGR engineering technology, a thermal hydraulic analysis has been initially undertaken. The gas cooled fast reactor design is based on the AGRs at Hinkley Point B and Heysham 2/Torness in the UK which use a concrete pressure vessel to house the core structure, the 12 steam generators and the gas circulators. The temperature of the coolant entering the boiler has been used as the link parameter between the AGR engineering and the fast reactor fuel technology. On this basis a core inlet temperature of 252C and a CO_2 gas pressure of 42 bar are assumed for all of the configurations considered during these studies.

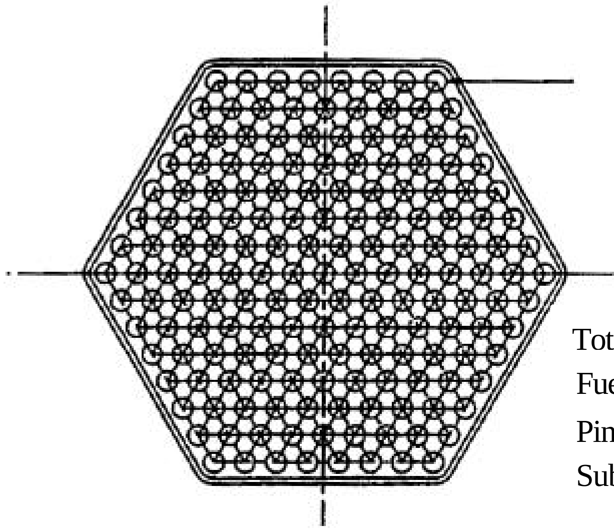

Figure 2. Basic Gas Cooled Fast Reactor Fuel Sub-assembly Design

The thermal hydraulic performance of the different sub-assembly designs considered during these studies has been carried out using the general purpose Computational Fluid Dynamics code CFX-4[4]. The basic gas cooled fast reactor sub-assembly design is shown in Figure 2. The objective of the thermal hydraulics studies has been to determine the magnitude of the gas flow rate that, for a given pressure drop fixed by the basic design constraints, would occur for the different sub-assembly geometries that have been considered. Consequently the efficiency of the heat transfer from the fuel pins to the coolant and the adequacy of the mixing of the gas have been evaluated. The basic mesh used in these calculations represents a 30° sector around a single pin and contains 100 cells in the axial direction and 10,000 cells in total. The flow is driven by the imposed pressure difference between the top and the bottom of the model and a uniform heat flux has been assumed from the surface of the fuel pin. The standard k-ε model has been used to determine the turbulence and the CO_2 coolant was modelled as having constant properties.

3.2 NEUTRONICS

All of the core configurations evaluated during these studies have been modelled using 120° sector geometry, as shown in Figure 3, with version 1.2 of the European Fast Reactor code scheme ERANOS[5] along with the ERALIB1[6] nuclear cross section data library. The deterministic ERANOS code system consists of a new generation of neutronic and gamma modules which have been developed within the framework of the European collaboration on fast reactors. The adjusted nuclear data library ERALIB1 is based on a comprehensive set of integral measurements performed in reactors and critical experimental facilities, and has been validated for a wide variety of fast reactor cores.

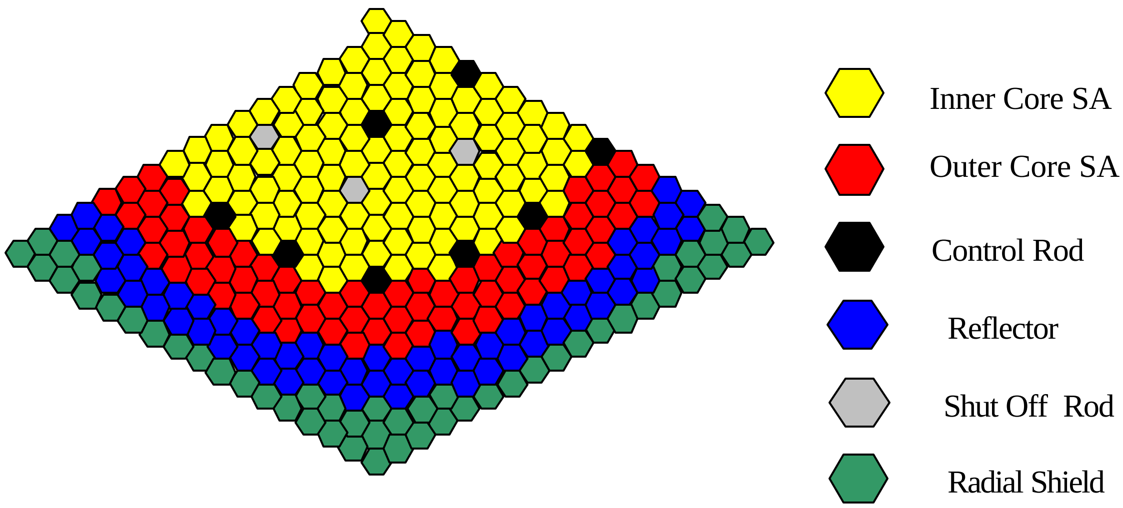

Figure 3. 120° Sector Core Model for the Gas Cooled Fast Reactor Core

Within the ERANOS code scheme the cell code ECCO[7] uses the subgroup method to treat resonance self shielding effects to prepare broad group self shielded cross sections and matrices for each material in the core model. A fine group slowing down treatment is combined with the subgroup method within each fine group to provide an accurate description of the reaction thresholds and resonances for the exact heterogeneous geometry of each type of critical and sub-critical sub-assembly. The self shielding formula for a standard effective cross section σ_x , where x denotes the reaction type, for a particular group g, and for a region i with neighbouring regions j is expressed in the following way :

$$\sigma^{g}_{xi} = \frac{\sum_j S^g_j \sum_k \alpha^g_k \sigma^g_{x_k} p_{ij}\left(\Sigma^g_{t_k}\right)}{\sum_j S^g_j \sum_k \alpha^g_k p_{ij}\left(\Sigma^g_{t_k}\right)} \tag{1}$$

where S^g_j is the source in group g and region j, α^g_k is the probability in group g to find the partial cross section $\sigma^g_{x_k}$ in subgroup k corresponding to the total cross section $\sigma^g_{t_x}$ used to calculate the macroscopic cross section $\Sigma^g_{t_k}$, and $p_{ij}\left(\Sigma^g_{t_k}\right)$ is the reduced collision probability for subgroup k within the group g. The self shielding of the total Legendre order one cross section, as well as order one of the elastic cross section, has an alternative formulation as this type of cross section is current weighted :

$$\sigma_{t_i} = \frac{\sum_j S_j \sum_k \alpha_k \sigma_{t_k} p^2_{ij}\left(\Sigma_{t_k}\right)}{\sum_j S_j \sum_k \alpha_k p^2_{ij}\left(\Sigma_{t_k}\right)} \tag{2}$$

The fluxes and currents are then calculated using the P1 consistent method so that the flux Φ and current J in group g and region i are expressed as follows :

$$\Phi_i^g = \sum_j \left(-B^g J_j^g + S_{fj}^g + \sum_{g'} \Sigma_{S_{0,j}}^{g'\to g} \Phi_j^{g'} \right) p_{ji}\left(\Sigma_t^g\right) \tag{3}$$

$$J_i^g = \sum_j \left(\frac{B^g}{3\Phi_j^g} + \sum_{g'} \Sigma_{S_{1,j}}^{g'\to g} J_j^{g'} \right) p_{ji}\left(\Sigma_t^g\right) \tag{4}$$

where B^g is the buckling , S_{fj}^g is the fission source and $\Sigma_s^{g'\to g}$ denotes the order of the scatter cross section for a neutron scattering from the group g' to the group g.

For conventional applications, such as the sodium cooled fast reactor, the cross sections and matrices are then condensed and homogenised in the required broad group scheme to provide effective cross sections that correctly treat the spatial heterogeneity of the sub-assembly structure. However, for situations involving low density regions and strongly anisotropic neutron streaming previous work has shown that these conventional formulations are insufficient. These method limitations are especially apparent in the gas cooled fast reactor where low density regions are present in both the fuel and control rod follower sub-assemblies. In addition, as the coolant channels are longer in the axial direction than in the radial direction the neutron streaming in this type of system is strongly anisotropic. Hence additional modifications[8] have been introduced into the ECCO cell code to treat these specific characteristics correctly. Thus the strong anisotropy of the streaming is taken into account by the use of directional collision probabilities $p_{ij}^k(\Sigma_t)$, where k denotes the direction, so that the flux is expanded in the following way :

$$\Phi(\vec{r},\vec{\Omega}) = \left(\Phi_0(\vec{r},\vec{\Omega}) + i\sum_k B_k \Phi_{l,k}(\vec{r},\vec{\Omega}) \right) e^{i\vec{B}\vec{r}} \tag{5}$$

Hence the current J in group g, previously given by equation (4), now becomes :

$$J_i^{g^k} = \sum_j \left(\frac{B^{g^k}}{3\Phi_j^g} + \sum_{g'} \Sigma_{S_{l,j}}^{g'\to g} J_j^{g'k} \right) p_{ji}^k\left(\Sigma_t^g\right) \tag{6}$$

where $J_i^{g^k}$ is the current in the direction k, group g and region i defined in the following way :

$$J_i^{g^k} = \int_i \int_{\vec{\Omega}} \Phi_0(\vec{r},\vec{\Omega}) \vec{\Omega}_k d\vec{\Omega} \tag{7}$$

and B^{g^k} is the buckling in the direction k.

This formulation requires that additional consideration is given to the buckling dependence of the diffusion coefficients, an effect which is usually considered to be negligible. However it becomes important in situations involving low density regions. If a region extends infinitely in two directions and the cross section in this region becomes small then the diffusion coefficient D depends strongly on the buckling. If the buckling is neglected and the diffusion coefficient is calculated by the Benoist formula in the standard way :

$$D = \frac{\sum_i \sum_j V_i \Phi_i P_{ij}(\Sigma) / \Sigma_j}{3 \sum_i V_i \Phi_i} \tag{8}$$

Then, as the cross section tends to zero, D tends to infinity. By considering the equivalence between the limiting behaviour of the diffusion coefficient as the buckling and the cross section tend to zero an alternative derivation of equivalent buckling dependent cross sections has been introduced for low density regions. The reduced collision probabilities $p_{ij} = P_{ij}(\Sigma)/\Sigma$ calculated with the modified transport cross sections are used to weight the unmodified transport cross section. In diffusion theory, which has been used throughout all of these studies, this corrected formulation is employed in the form of directionally dependent axial and radial diffusion coefficients D_z and D_r.

When using diffusion theory, previous work has shown that consideration should be given to the treatment of the control rod follower regions, which in a gas cooled fast reactor become large low density channels which can extend a considerable distance into the core. In this situation, using the standard definition of the diffusion coefficient $(1/3\Sigma_t)$ results in an overestimation of neutron leakage, which in turn leads to large errors in the prediction of control rod reactivity worth.

An alternative method has been developed to provide a modified diffusion theory solution that correctly treats the low density control rod follower channels the gas cooled fast reactor core. An approach has been adopted where the directionally dependent diffusion coefficients in the follower channel have been modified according to a Buckling Modified formulation so that :

$$D_z = \frac{1}{3\Sigma_1} \left(\frac{\left(2 - \frac{\Sigma_0}{\Sigma_1}\right) + 2a^* \Sigma_1}{1 + 2a^* \Sigma_0} \right) \tag{9}$$

and,

$$D_r = \frac{1}{3\Sigma_1}\left(\frac{4a\Sigma_1 + \left(3 - \frac{\Sigma_0}{\Sigma_1}\right)\left(1 + \frac{\Sigma_0}{\Sigma_1}\right)}{4a\Sigma_0 + \left(1 + \frac{\Sigma_0}{\Sigma_1}\right)}\right) \tag{10}$$

where Σ_0 and Σ_1 are the transport cross sections for the follower channel and the surrounding fuel medium, and a is the radius of the follower channel where $a^* = a(1 - ba)$ and,

$b = \frac{3pB_z}{8}$ for a core region (assuming a sinusoidal flux variation) and,

$b = \frac{3}{4}\frac{L}{H^2}$ for a reflector region (assuming an exponential flux variation),

here L is the diffusion length, H is the core height and B_z is the axial buckling.

In this way homogenised cross sections have been produced for each core material in 33 neutron energy groups. Whole core flux and depletion calculations have then been performed in 3-dimensional diffusion theory for each core configuration studied. The control rods have been modelled with a reduced ^{10}B content in order to allow for the transport, heterogeneity and mesh effects that are not included in the simplified homogeneous control rod representation employed during these studies.

Consistent flux, isotopic compositions and macroscopic cross section data corresponding to each equilibrium core burnup state have been produced in a series of iterations. The flux of one iteration has been used to determine the isotopic concentrations and macroscopic cross section data for the next iteration, this process being continued until convergence has been achieved. This average irradiation condition has then been used to produce equilibrium core models at start of cycle (SOC), middle of cycle (MOC) and end of cycle (EOC).

The fuel cycle lengths are determined by the criteria of attaining a peak clad damage over the lifetime of the fuel within the core close to 180 dpa NRT Fe. The inner and outer core feed fuel enrichments for each core configuration have been chosen on the basis of achieving a calculated EOC reactivity close to unity with all control rods withdrawn. The control rod insertions at SOC and MOC are then adjusted to obtain k-effective values close to unity. The core performance and safety parameters have then been calculated for the SOC, MOC and EOC conditions.

4. PRELIMINARY CORE DESIGN

A preliminary gas cooled fast reactor core design has been developed to demonstrate compatibility with the design objectives of the European Fast Reactor project. At this stage no specific design features have been introduced to enhance plutonium or minor actinide management and this core design serves purely to provide physics, performance and safety parameters which establish the fundamental feasibility of the gas cooled fast reactor concept. The EFR design limitations on fuel temperature, target burnup, fuel pin rating and clad damage, as well as the EFR basic safety requirements have been assumed for the purposes of these studies. In addition the isotopic composition of the feed plutonium, a conventional MOX fuel, is that specified for EFR. Thus a complete set of design parameters has been derived as summarised in a comparison with those of EFR in Table I.

Table I. Design Parameters for the Preliminary Gas Cooled Fast Reactor Core Design

	Gas Cooled Fast Reactor	EFR Reference Design
Reactor Thermal Output (MWth)	3600	3600
Electrical Output (MWe)	1400	1550
CO_2 Gas Pressure (bar)	42	Not Applicable
Core Inlet Temperature (°C)	252	395
Core Outlet Temperature (°C)	525	545
Active Core Height (mm)	1500	1000
Fuel Sub-assembly Pitch (mm)	180.61	188.00
Fuel Pin Pitch/Diameter	1.55	1.14
Fuel Pin Pitch (mm)	12.71	9.34
Number of Fuelled Pins per Sub-assembly	169	331
Fuel Pin (clad) Outer Diameter (mm)	8.20	8.20
Fuel Pin Inner Diameter (mm)	2.00	2.00
Clad Thickness (mm)	0.42	0.52
Number of Fuelled Sub-assemblies	550	387
Target Peak Fuel Pin Linear Rating (W/cm)	460	520
Target Burnup (GWd/t - at%)	190 - 20	190 - 20

Based on the gas cooled fast reactor design parameters outlined above a number of core performance calculations were subsequently carried out. The fuel dwell time has been determined on the basis of achieving a peak heavy atom burnup in the core close to 20%. By considering the capability of the control rods to compensate for the reactivity swing over the cycle, and with due allowance for uncertainties and margins to cold shutdown, it was found that a five batch refuelling scheme was required. A dwell time of 1720 effective full power days (efpd), with a cycle length of 344 efpd, was found to be acceptable for the purposes of the current study. Feed fuel enrichments were chosen so as to obtain an EOC reactivity close to unity with all rods withdrawn, and differential enrichments were used to balance the peak linear ratings between the inner and outer zones averaged over the fuel cycle. Thus acceptable feed fuel enrichments for these purposes were found to be 16.05 and 27.29 mass% for the inner and outer core respectively. A summary of the main performance and safety parameters is given in Table II.

Table II. Summary of Results for the Preliminary Gas Cooled Fast Reactor Design

Cycle Length (efpd)	344
Number of Cycles	5
Total Fuel Residence Time (efpd)	1720
Plutonium Enrichment (Inner/Outer - mass%)	16.05/27.29
Peak Linear Rating (Inner/Outer - W/cm) : SOC MOC EOC	347/370 347/385 343/367
Peak Burnup (Inner/Outer - %ha)	19.89/19.07
Peak Damage (Inner/Outer - dpa NRT Fe)	245/216
Doppler Constant (pcm) : SOC MOC EOC	-489 -498 -516
Reactivity Loss over the Cycle (pcm)	3083
Total β-effective (pcm)	357
Prompt Neutron Lifetime (s)	5.35E-07
Plutonium Consumption Rate during Operation (kg/TWeh)	22.1

The results for this preliminary gas cooled fast reactor design illustrate the viability of the concept and show no major difficulties with respect to core performance or safety. The peak pin burnup has been determined from the percentage change in heavy atom concentrations over the residence time of the fuel within the core and is close to the target value of 20% ha. The peak linear ratings, Doppler constant and delayed neutron fraction are similar to equivalent values obtained for EFR. The distribution and characteristics of the control rods, extrapolated from those of EFR, appear adequate as the necessary shutdown margins have been satisfied. It can be noted that although no specific optimisation measures have yet been implemented a reasonable plutonium consumption rate of 22.1 kg/TWeh has been achieved due to the increased plutonium content in the core and the absence of axial and radial breeder regions.

The large difference in inner and outer core plutonium enrichments is due to neutron economy effects caused by the low density coolant, the consequent increase in the fuel pin separation and the harder neutron spectrum, all of which result in a particular increase in neutron leakage in the outer core regions. The clad damage has been determined from the neutron spectrum, the damage cross sections used by ERANOS and the fuel residence time. The calculated peak pin clad damage of 245 dpa NRT Fe is considerably greater than the limit of 180 dpa NRT Fe currently assumed for fast reactor studies. This is not an unexpected result as the core has been primarily optimised to achieve the target burnup of 20% ha, and this high value reflects the harder neutron spectrum present in the gas cooled fast reactor. However, this preliminary design shows that fast reactor design concepts can be achieved using gas technology. This limited initial evaluation requires further optimisation to fully explore the potential of the gas cooled fast reactor, and some further studies are presented in the following three sections of this paper. Work has been carried out to determine the potential in relation to plutonium and minor actinide consumption as well as the capability to handle a large variety of plutonium fuels. In addition, an investigation has been performed to determine the reactivity effects of CO_2 loss and water ingress into the core due to steam leaks from the boilers.

5. MODIFICATION FOR EFFECTIVE PLUTONIUM AND MINOR ACTINIDE MANAGEMENT

5.1 THERMAL HYDRAULICS

Following the development of a preliminary gas cooled fast reactor design which demonstrated the viability of this concept the potential for the effective management of plutonium stockpiles has been addressed. In order to determine the scope for plutonium management the feasibility of a gas cooled fast reactor design optimised for high levels of plutonium consumption has been investigated. This reference option is based on a mixed oxide (MOX) fuel arising from a fuel cycle scenario assuming a balanced fleet of three differently fuelled types of PWR. The MOX feed fuel is the result of UOX output that has been twice recycled in MOX reactors to arrive at a degraded plutonium content. Using the preliminary gas cooled fast reactor design as a basis, and assuming the same overall core design constraints, the aim has been to increase the plutonium content as much as possible while remaining within the 45% by mass limit imposed by current reprocessing experience using the PUREX technique. For the same core volume the use of a high plutonium content fuel is only made possible by a significant reduction of the fuel inventory. This has been achieved by the introduction of dilution in the form of empty pin positions within the fuelled sub-assemblies. Although these pins are taken to be empty, or void, for the purposes of the current studies, this approach allows the possibility of further improving the core characteristics by introducing inert or moderator materials, or even minor actinide fuelled pins.

When applying the dilution concept it is essential to ensure that the heat transfer properties of the core are not compromised and so a thermal hydraulic analysis has been undertaken to determine the performance of the different sub-assembly designs that have been considered as potential candidates. To apply the dilution concept the approach taken has been to increase the number of pin positions while at the same time reducing the pin size. Starting from the preliminary 169 pin design, supplementary rings of pins have been added to the fuel sub-assembly and the effect on the heat transfer from the fuelled pins and the flow and mixing of the coolant gas has consequently been evaluated. During this process it is assumed that the pin pitch to diameter ratio as well as the core inlet temperature are unchanged. In addition, the core pressure differential and the heat flux per unit pin surface area are assumed to remain constant. The consequences of changing from an 8 ring (169 pin) sub-assembly design to a 12 ring (397 pin) design are shown in Figure 4.

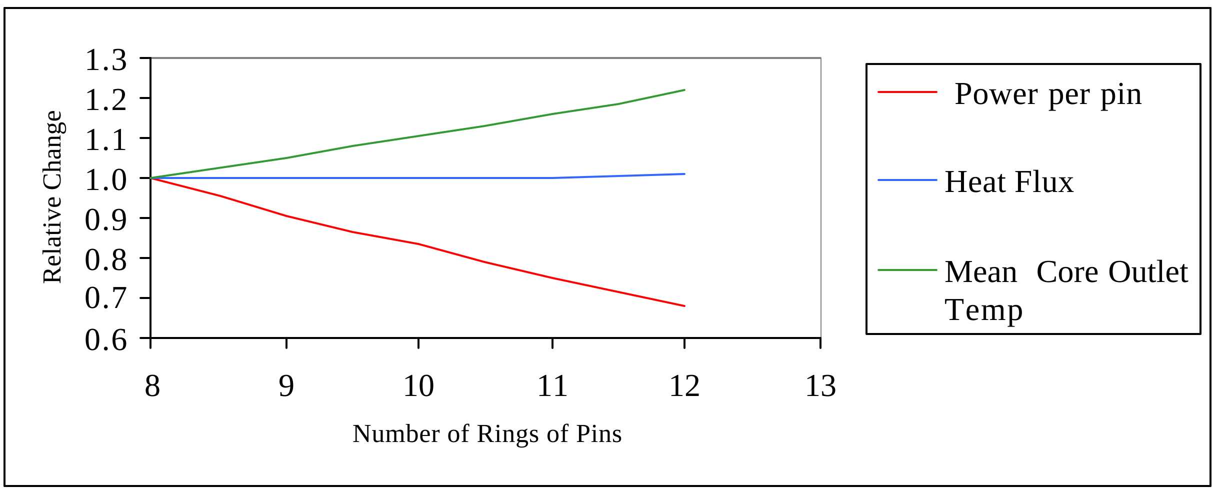

Figure 4. Thermal Properties for the Gas Cooled Fast Reactor Design

For each configuration the ratio of fuel to empty pins has been chosen on the basis of maintaining a constant heat flux. The reduction in the total power in each fuelled pin is balanced by a modest increase in the core mean outlet temperature caused by the reduced pin separation and the resulting change in the mixing of the coolant gas. On the basis of these results the 11 ring (331 pin) sub-assembly design, of which 260 pins are fuelled, and thus 71 are empty, has been chosen for the reference option of the gas cooled fast reactor core design. The fuel pin has a reduced clad outer diameter of 5.82 mm, a clad thickness of 0.42 mm, and an inner diameter of 2.16 mm. By comparison with the preliminary core design, the mixed mean core outlet temperature, increased by about 12%, is still considered to be satisfactory and to be well within the limiting design criteria.

5.2 NEUTRONICS

Using the sub-assembly design outlined above, core performance calculations have been subsequently carried out using the methods employed during the preliminary design studies. The fuel dwell time has been determined on the basis of achieving a peak clad damage close to 180 dpa NRT Fe. By considering the ability of the primary control rods to compensate for the cycle reactivity swing it was found that a four batch refuelling scheme was required. A total fuel dwell time of 1560 efpd, with a cycle length of 390 efpd, was found to be satisfactory for the purposes of these studies. Acceptable feed fuel enrichments were determined to be 31.12 and 44.94 mass % for the inner and outer core respectively. A summary of the main core performance and safety parameters is given in Table III.

Table III. Performance Parameters for the Burner Gas Cooled Fast Reactor Core Design

Cycle Length (efpd)	390
Number of Cycles	4
Total Fuel Residence Time (efpd)	1560
Plutonium Enrichment (Inner/Outer – mass %)	31.12/44.94
Peak Linear Rating (Inner/Outer – W/cm) : SOC MOC EOC	230/228 225/223 219/215
Peak Burnup (Inner/Outer - %ha)	18.12/18.52
Peak Damage (Inner/Outer – dpa NRT Fe)	180/159
Doppler Constant (pcm) : SOC MOC EOC	-310 -328 -346
Reactivity Loss over the Cycle (pcm)	6364
Total β-effective	337
Prompt Neutron Lifetime (s)	7.52E-07
Plutonium Consumption Rate during Operation (kg/TWeh)	52.4

The changes in the design have not led to any significant deterioration in the core performance or the safety parameters. A reduction in the fuel dwell time has been necessary to satisfy the limitation on peak clad damage and it can be noted that as a result of this the peak heavy atom burnup is slightly under the target value of 20% ha. In addition, the peak linear ratings are reduced by comparison with those of the preliminary core design as a consequence of the constraints imposed during the thermal hydraulic optimisation studies. However the plutonium consumption rate of 52.4 kg/TWeh, more than double that of the preliminary core design, demonstrates that conventional gas cooled fast reactor technology can be effectively employed to achieve high rates of plutonium consumption.

Additional measures could be undertaken to reduce the significant difference still apparent in the plutonium content of the inner and outer core zones which would serve to further increase the plutonium consumption rate. A further factor to be considered is the use of moderator or inert materials, made possible by the empty pin positions in the fuelled sub-assemblies, to make further improvements in the core performance. However, for this initial evaluation, this core design is considered to be a suitable basis to demonstrate the flexibility and versatility of the gas cooled fast reactor concept, and a review of this aspect is presented in the next section of this paper.

6. FLEXIBILITY

Previous work has shown that a significant constraint exists in the operation of sodium cooled fast reactors due to the degeneration of the core safety parameters when poor quality plutonium fuels, or fuels with a significant proportion of minor actinide isotopes, are utilised. The worsening of the core safety parameters is mainly attributed to the sodium void coefficient, whereby the accumulation of higher plutonium and minor actinide isotopes results in the sodium void coefficient becoming increasingly positive and consequently unacceptably large in magnitude. This constraint imposes an important limitation on the type of fuel that can be utilised in sodium cooled fast reactors. However, the negligible void coefficient in the gas cooled fast reactor ensures that the consequences for the core safety parameters of using degraded plutonium and minor actinide fuels will be less severe. In order to evaluate the increased flexibility afforded by the gas cooled fast reactor concept three different plutonium qualities have been considered for use in the optimised gas cooled fast reactor design, as shown in Table IV.

Table IV. Assumed Plutonium Isotopic Composition for the Gas Cooled Fast Reactor (mass %)

	Case 1	Case 2	Case 3
Pu238	1.9	5.6	2.2
Pu239	53.3	39.1	31.2
Pu240	25.6	26.7	37.3
Pu241	9.9	13.0	6.9
Pu242	7.9	14.3	19.6
Am241	1.3	1.3	2.7

Case 1 corresponds to plutonium fuel originating from a once through irradiation of UO_2 fuel. Case 2, representing the feed fuel assumed for the optimised reference design, is the result of twice recycling the case 1 UO_2 output in MOX reactors. Finally, case 3 corresponds to the multi-recycling of case 2 fuel in the optimised gas cooled fast reactor and is intended to represent the situation after the long term operation of gas cooled fast reactors whereby an equilibrium has been achieved with the rest of the reactor fleet. This equilibrium cycle fuel composition has been obtained by considering the difference between the initial and discharge heavy nuclide mass for subsequent cycles of the optimised gas cooled fast reactor. For each cycle the difference between the initial and discharge mass has been made up by the case 2 initial fuel feed in order to conserve the charge mass. This process has been repeated until an equilibrium cycle fuel composition has been obtained.

These three cases are intended to cover the range of possible fuel cycle hypotheses that may occur during the operation of a balanced fleet of UOX and MOX PWR's combined with gas cooled fast reactors corresponding to early, middle and late situations in the next century. On the basis of the optimised gas cooled reactor design described in the previous section of this paper, core performance calculations have been performed for the three different cases of plutonium quality. In each case the fuel dwell time has been determined on the basis of achieving a peak clad damage close to 180 dpa NRT Fe. The batch refuelling scheme has been fixed by considering the ability of the control rods to compensate for the reactivity loss over the cycle. A summary of the main core performance and safety parameters is given in Table V.

Table V. Performance Parameters for Different Plutonium Isotopic Qualities

	Case 1	Case 2	Case 3
Cycle Length (efpd)	390	390	430
Number of Cycles	4	4	3
Total Fuel Residence Time (efpd)	1560	1560	1290
Plutonium Enrichment (Inner/Outer – mass %)	28.92/40.54	31.12/44.94	33.12/44.94
Peak Linear Rating (Inner/Outer – W/cm) : SOC MOC EOC	237/223 225/221 211/212	230/228 225/223 219/215	224/222 224/223 214/211
Peak Burnup (Inner/Outer - %ha)	15.90/15.18	18.12/18.52	18.69/18.94
Peak Damage (Inner/Outer – dpa NRT Fe)	195/156	180/159	185/130
Doppler Constant (pcm) : SOC MOC EOC	-314 -331 -336	-310 -328 -346	-276 -292 -315
Reactivity Loss over the Cycle (pcm)	6432	6364	5987
Total β-effective	327	337	320
Prompt Neutron Lifetime (s)	7.60E-07	7.52E-07	7.45E-07
Pu Consumption Rate during Operation (kg/TWeh)	47.7	52.4	57.4

The performance results obtained for the different qualities of plutonium composition illustrate the significant amount of flexibility associated with the gas cooled fast reactor concept. It can be seen that no significant modifications are required for the utilisation of a wide range of plutonium isotopics, apart from a re-evaluation of the batch refuelling scheme to take into account the changes in reactivity loss with fuel burnup associated with the different plutonium vectors. All of the main performance parameters follow similar consistent trends and demonstrate no significant deterioration with changing plutonium quality. It can be particularly noted that high levels of plutonium consumption are achievable for all of the cases considered during these studies. Indeed the utilisation of even more varied plutonium isotopics may be feasible if moderator or inert materials are employed in the core design. A review of the Doppler and coolant void coefficients, as well as the reactivity effects that may arise with water ingress into the core regions, is provided in the next section of this paper. In addition, safety and control issues specific to the gas cooled fast reactor are discussed.

7. GCFR CONTROL AND SAFETY

It is essential for the verification of the viability of the gas cooled fast reactor concept that the safety parameters and control features are adequate. The Doppler constant, presented in the previous tables in this paper for each of the core configurations evaluated during these studies, has been determined from the changes in reactivity obtained by reducing the mean core fuel temperature from that of normal operation (1500K) to that of cold shutdown (453K). It can be noted from these results that the Doppler constant, at between 250-500 pcm, is considerably lower than that associated with a sodium cooled fast reactor such as EFR due to the harder neutron spectrum in the gas cooled fast reactor. The most significant reduction in the Doppler effect appears in the optimised gas cooled fast reactor design due to the increase in the plutonium enrichment and the subsequent reduction in the U238 capture contribution. An additional reduction in the Doppler constant can be observed during the transition to a more degraded plutonium quality due to a further hardening in the neutron spectrum.

In order to make an initial evaluation as to whether the magnitude of the Doppler constant is acceptable in the proposed gas cooled fast reactor design it has been considered in conjunction with the coolant void coefficient. The coolant reactivity effect has been determined by the difference between flux calculations with and without the CO_2 coolant present in the core fuel sub-assemblies, and the results are given in Table VI.

Table VI. Coolant Void Reactivity Effect for the Gas Cooled Fast Reactor (pcm)

	Preliminary Design	Optimised Design		
		Case 1	Case 2	Case 3
SOC	+52.8	-7.3	+17.9	+27.5
MOC	+57.9	-7.7	+16.1	+26.8
EOC	+63.1	-8.1	+15.4	+25.1

It can be seen that, as expected, the coolant void reactivity coefficient for the gas cooled fast reactor is very much smaller than that associated with sodium cooled fast reactor cores and as such can be considered to be a negligible effect. It is thus considered that the reduced magnitude of the Doppler constant is probably acceptable although a full safety accident analysis of the proposed gas cooled fast reactor design is required in order to confirm this.

A fault condition of potential concern in a gas cooled fast reactor is that of water ingress into the core due to leaks in the steam generators. For an initial evaluation of the importance of this effect it has been assumed that the water, at least during the early stages of an incident, will be in the form of steam at the same temperature and pressure as the coolant. The reactivity change between the situation with CO_2 coolant, and the situation with the coolant uniformly replaced by steam at the same temperature and pressure, is given in Table VII.

Table VII. Steam Reactivity Effect for the Gas Cooled Fast Reactor

	Preliminary Design	Modified Design		
		Case 1	Case 2	Case 3
SOC	-553.1	+135.4	+116.9	+77.8
MOC	-545.0	+144.4	+125.7	+84.5
EOC	-536.8	+153.2	+134.3	+91.2

It can be seen from these results that the effect of water ingress for the proposed gas cooled reactor design is relatively small in magnitude. It can be noted however that this reactivity effect can be either positive or negative depending on the characteristics of the core being considered. The water ingress effect is dominated by the moderation component which is in turn dependent on the shape of the adjoint flux as a function of energy. The modifications to the gas cooled fast reactor design to increase the plutonium burning rate, and the consequent increase in plutonium enrichment, has resulted in the high energy adjoint being considerably reduced due to the reduction in U238 threshold fissions. The result is a net positive increase in the steam ingress reactivity coefficient. Although this reactivity effect does not initially appear to be significant a more detailed analysis of the conditions of water ingress is required before a more definitive conclusion can be drawn. It is clear however that a wide variety of plutonium fuels can be utilised without any significant impact on the gas cooled fast reactor core safety parameters.

CONCLUSIONS

A preliminary evaluation has been undertaken to investigate the core performance of the gas cooled fast reactor concept. Thermal hydraulics and neutronics studies have shown that a feasible design for a reactor combining existing UK gas reactor experience with current European fast reactor design objectives can be achieved within the limitations of conventional technology. An initial sub-assembly and core design has been established and it has been demonstrated that the performance and safety parameters for a core configuration capable of 20% peak fuel burnup are acceptable.

This preliminary design has been further developed to explore its potential and flexibility for the effective management of plutonium stockpiles. This study, considering both thermal hydraulic and neutronic performance, has indicated that plutonium consumption rates of up to 55 kg/TWeh can be readily achieved. An investigation of the core safety parameters, including the reactivity effects associated with coolant voiding and water ingress, has demonstrated that the proposed core design is capable of safe operation with a wide range of fuels, and consequently has the potential flexibility to meet the needs of the many fuel cycle and reprocessing scenarios which may occur during the next century.

During the investigation of the gas cooled fast reactor design it has become clear that several areas require further examination in order to more completely confirm the potential and viability of this concept. Among the areas for additional consideration is the use of moderator or inert materials to further improve core performance. It has been shown in the current studies that considerable flexibility exists with respect to the variety of plutonium fuels that can be utilised and it is suggested that further investigations could be carried on a similar basis to assess the potential for the recycling of minor actinide based fuels. Finally, although acceptable core safety parameters have been established a complete transient analysis of the gas cooled fast reactor core should be undertaken to ensure that the proposed control and safety measures are sufficient.

ACKNOWLEDGEMENTS

Discussions with colleagues in the CAPRA/CADRA project were extremely useful during the course of this work and are acknowledged. This study has been sponsored by BNFL.

REFERENCES

1. A. Languille, J. C. Garnier, P. Lo Pinto, B. C. Na, D. Verrier, J. Depliax, P. Allen, R. E. Sunderland, E. Kiefhaber, W. Maschek, D. Struwe, "CAPRA Core Studies, The Oxide Reference Option," Proceedings of Global'95, Versailles, Paris, France (September 1995).

2. W. Kemmish, "Gas Cooled Fast Reactors," *Nuclear Energy,* Volume 21, Nr 1 (1982).

3. H. Sztark, D. Barnes, U. Wehmann, "Core Optimisation of the European Fast Reactor EFR," Proceedings of International Conference on Fast Reactors and Related Fuel Cycles, Kyoto, Japan (October 1991).

4. *CFX 4.2 : Solver*, CFX International, Harwell, England (1997)

5. J. Y. Doriath, C. W McCallien, E. Kiefhaber, U. Wehmann, J. M. Rieunier, "ERANOS The Advanced European System of Codes for Reactor Physics Calculations," Proceedings of International Conference on Mathematical Methods and Super Computing in Nuclear Computations, Karlsruhe, Germany (April 1993).

6. E. Fort, W. Assal, G. Rimpault, J. Rowlands, P. Smith, R. Soule, "Realisation and Performance of the Adjusted Nuclear Data Library ERALIB1 for Calculating Fast Reactor Neutronics," Proceedings of PHYSOR'96, Mito, Japan (September 1996).

7. M. J. Grimstone, J. D. Tullett, G. Rimpault, "Accurate Treatment of Fast Reactor Fuel Assembly Heterogeneity with the ECCO Cell Code," Proceedings of PHYSOR'90, Marseille, France (April 1990).

8. G. Rimpault, P. Smith, T. D. Newton, "Advanced Methods for Treating Heterogeneity and Streaming Effects in Gas Cooled Fast Reactors," Proceedings of Mathematics and Computing '99, Madrid, Spain (September 1999).

9. T. A. Lennox, D. M. Banks, J. E. Gilroy, R. E. Sunderland, "Gas Cooled Fast Reactors," Proceedings of ENC'98, Nice, France (October 1998).