NGNP Fuel Design Special Study

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AREVA NP Inc.

AREVA

Record of Revisions

Revision	Date	Comments
000	April 3, 2008	Initial Issue
001	April 15, 2008	Added discussion of the impacts of compact density variation on calculated operational fuel temperature on page 14. Corrected the fast neutron energy cutoff for reported fluence values (from 0.9 MeV to 0.27 MeV) on pages 10, 23, and 24.

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1.0 Introduction

The NGNP Fuel Design Special Study was conducted to define a fuel particle and compact design envelope which provides adequate design flexibility for future core designs. The resulting design envelope considers the range of design parameters including enrichment, packing fraction, and use of burnable absorbers, which are required to support a design which allows operation of a nominal 18 month cycle and complies with fuel-related requirements of plant safety case

This information will be used to help define an ongoing NGNP fuel testing and qualification program through specification of a suggested testing envelope. Definition of this testing envelope will be a balance between the needs of the core designer, as reflected in the above design envelope, and the needs of the fuel designer to have a reasonable chance of successfully qualifying the resulting particle and compact system.

The NGNP reactor system configuration and operating conditions described in the AREVA NGNP PCDSR will be the basis for this study. The fuel particle is assumed to be a silicon-carbide TRISO particle with a UCO kernel.

2.0 Fuel Particle Design

A proposed set of fuel particle physical attributes, including kernel diameter and coating layer thicknesses, and corresponding material properties, was selected as a starting point for subsequent analyses. This selection was based on a review of relevant past experience with the goal of reducing the iterations required to achieve an acceptable fuel design. For this selection, it was assumed that all fuel operating within the core will be of one physical particle design. That is, any required core design reactivity variation and power shaping will be achieved through changes in enrichment, compact packing fraction, and through the use of burnable absorbers.

The final set of parameters specified were based largely on the set of parameters that define the fuel fabricated for the initial AGR test irradiations. These parameters are provided in Table 2.1. It was decided to use this set of values in order to, if subsequent analysis results indicated it possible, limit the impact of this study on the ongoing AGR program. A review of additional relevant industry data (References 1-6, 8) indicated this parameter set to be a reasonable starting point. That is, the fuel design described is well within the experience band for TRISO fuel operated under similar conditions.

In addition to the fuel design parameters, an initial set of fuel operating conditions and limits was established to provide a framework for the core design and safety analyses. These would be updated based on the analysis results and factored into future iterations. A listing of these values is provided in Table 2.2.

Table 2.1 - Initial Nominal Design Parameters

Parameter	Value	Reference
Kernel		
Kernel Material	UCO	(7)
Kernel O/U Ratio	0.50	(7)
Kernel C/U Ratio	1.50	(7)
Kernel Material RT Density	10.5 Mg/m^3	(7)
Kernel Diameter	425 μm	(8)
Kernel Enrichment	14 w/o	(8)
Buffer		
Buffer Material	Graphite	(7)
Buffer Thickness	100 μm	(10)
Buffer RT Density	0.95 Mg/m^3	(7)
IPyC		
IPyC Material	Graphite	(7)
IPyC Thickness	40 μm	(7)
IPyC RT Density	1.90 Mg/m^3	(7)
SiC		
SiC Material	Silicon Carbide	(7)
SiC Thickness	35 μm	(7)
SiC RT Density	3.20 Mg/m^3	(7)
OPyC		
OPyC Material	Graphite	(7)
OPyC Thickness	40 μm	(7)
OPyC RT Density	1.90 Mg/m^3	(7)
Compact		
Compact Matrix Material	Graphite	(10)
Compact Diameter	12.5 mm	(13)
Compact Height	50 mm	
Fuel Particle Packing Fraction	≤ 35%	(10)
Compact Matrix Density	1.1995 g/cm^3	(9)
Fuel Element		
Fuel Element Material	H-451 Graphite	(10)
Fuel Element-to-Compact Radial Gap	1 mm	(13)
Fuel Element Material Density	1.74 g/cm^3	(9)

Table 2.2 – Initial Fuel Operating Conditions and Limits

Parameter	Value	Reference
Core Thermal Power	565 MWTh	(11)
Nominal Cycle Length	18 months	
Refueling Duration	25 days	(12)
Core Management Scheme	2 Batch Core	
Operational Temperature	≤ 1350 °C	(8)
Time-Averaged Temperature	≤ 1250 °C	(8)
Accident Temperature	~ 1600 °C	(8)
Discharge Burnup	≤ 17% FIMA	(8)
Discharge Fast Fluence	$\leq 5x10^{25}$ n/m^2	(8)

3.0 Core Design and Operational Performance

A series of core design analyses was completed (Reference 14) to provide an initial estimated range for the NGNP fuel characteristics that can meet the fuel performance requirements of a nominal 18 month during cycle operation, assuming a two-batch core fuel management scheme.

The calculations were performed using the simplified neutronic methodology implemented as part of the NEPHTIS calculational scheme (References 15 and 16). The fuel particle considered in this study was assumed to be a silicon-carbide TRISO particle with a $UC_{0.5}O_{1.5}$ kernel of 425 μm in diameter. The study focused on the impact of the particle packing fraction in the fuel compacts, fuel enrichment, and burnable absorber (BA) rods boron content on the cycle length and main fuel performance parameters (burnup, power peaking, fuel particle temperatures, fuel particle fast fluence, etc).

3.1 Methodology

The methodology used in the core design study was based on a suite of neutronic codes coupled in the NEPHTIS scheme, specially designed to treat the HTR prismatic reactor.

NEPHTIS scheme is comprised of the lattice code APOLLO2 that is used for generating the tablesets (homogenized neutronic cross sections) for the core nodal flux calculation that is performed with the finite element diffusion code CRONOS2.

APOLLO2 is a transport code that solves the multigroup Boltzmann transport equation using either the Integral Equation (collision probability 1D or 2D) or the Integral/Differential Equation (discrete ordinates and nodal methods in 2D geometry). This code allows the treatment of the double heterogeneity which is a characteristic of the

HTR fuel: in addition to the heterogeneity due to the position of the fuel compact inside the graphite block, there is a second heterogeneity resulting from the repartition of the fuel particles inside the compact. Cross section libraries are generated with APOLLO2 at different burn-up and temperatures to be used in CRONOS2 calculation.

The NEPHTIS scheme contains a number of APOLLO2 models for each type of fuel element (standard fuel elements or fuel elements with a control rod hole) and for all reflector assemblies, including those intended for the control rods.

CRONOS2 is a modular computational system for neutronic core calculations which reads cross section libraries issued from APOLLO2 (cell calculation). It solves the diffusion equation or the transport equation. Different geometries are available such as 1, 2 or 3 dimensions Cartesian geometries, 2 or 3D hexagonal geometries and cylindrical geometries. The numerical methods are based on the finite element method or the finite differences. The code is based on the diffusion theory with 8 energy groups (energy cutoff values: 1.6E-7, 0.625E-06, 1.67E-06, 2.7679E-06, 5.0045E-03, 2.7324e-01, 9.0718E-01 and 19.64 MeV). Flux calculation is carried out using a finite element method.

The HTR core model represents $1/6^{th}$ of the entire core for cyclic symmetry reasons. The radial modelling consists of 11 rings of hexagonal elements divided in three regions: the internal reflector, the fuel zone and the external reflector. Axially, the fuel region is represented by a stack of 10 blocks with a thickness of reflector on both sides.

CRONOS2 results are performed at different pre-assigned burn-up values between 0 (beginning of cycle-BOC) and the end of cycle (EOC) burnup value (which is an input to the code). At the beginning of the equilibrium cycle, the core is loaded with two categories of fuels blocks (fresh and single burnup irradiated blocks) according to a specific shuffling pattern. The subsequent transitional cycles toward equilibrium are using the same shuffle pattern but the once burned fuel assemblies (blocks) are retrieved from the ones available at the end of the preceding cycle.

The most important limitations identified so far for the version of NEPHTIS scheme used in this study are:

- No thermal feedback is currently included in the CRONOS2 calculations; the only calculations involving thermal feedback were previously performed on an experimental basis, using an external coupling with a thermal-hydraulic code (see References 15 and 17). Appendix 1 of Reference 14 discusses some of the results available from the calculation performed in Reference 17.

- The CRONOS2 code allows only axial shuffling patterns.

- The NEPHTIS scheme includes only partial burnup feedback (macroscopic) in the depletion calculation.

- The scheme does not have pin power reconstruction capability, so only assembly (block) peaking is calculated. Previous studies on pin power radial distribution have been used to assess local peaking factors (Reference 18).

- The current scheme allows as input only a limited number of fuel element (assembly) types. This feature limits considerably the flexibility of the tool in optimizing the cycle design (investigate zoned enrichment cores, zoned packing fractions, etc.)

3.2 Preliminary Neutronic Assessment

A preliminary study to estimate the cycle length for the standard HTR fuel with a kernel particle diameter of 425 μm was performed using the lattice code APOLLO2 assuming a single-batch fuel management scheme. The cycle length and corresponding fuel burnup were assumed reached when the assembly k_{inf} value equaled 1.05 (i.e., assuming a 5% neutron leakage at the core boundary). The fuel parameters varied in this study were packing fraction, enrichment and BA rods boron content. The BA rods were assumed to be compacts containing 200μm particles in compacts similar to fuel compacts. There was no assumed co-mingling of fuel and absorber particles in the compacts. The goal was to identify combinations of the above mentioned fuel parameters that result in a targeted cycle length above 1.5 years (18 months).

The results of this preliminary study were subsequently used as basis for defining the initial parameters of the fuel for the two-batch fuel management scheme presented in the next sections. An initial range of parameters that were expected to fulfill the requirements of a cycle length of 18 months, and up to 24, months was derived from the above calculations. The results suggested that fuel enrichment above 10 wt% and a compact packing fraction between 0.15 and 0.35 have the potential to sustain a cycle length above 18 months.

3.3 Fuel Cycle Design: Equilibrium Core

A set of cycle design calculations, using a two-batch fuel management scheme, were performed with different fuel parameters to confirm the trends identified in the preliminary study. Starting from a base case, variations of the fuel parameters (enrichment, fuel packing fraction and BA rods boron content) were investigated. A shuffling pattern that involves placement of once burned fuel in alternation with fresh fuel in each fuel column of the core was used.

The characteristics of the fuel for the equilibrium cycle were: enrichment 12 wt%, fuel packing fraction = 0.25 and BA rods content of B_4C particles (packing fraction) = 0.006. The NEPHTIS v1.2.4 runs were performed considering all rods out (ARO conditions) and average temperatures for fuel and moderator.

3.4 Summary of Results for Equilibrium Cycle:

The calculated equilibrium cycle had the following characteristics:

- Power density: 115.2 W/gU
- U235 mass/standard fuel element= 649.2 g
- Estimated cycle length (equilibrium): 522.5 EFPD
- Average discharge burnup = 120.4 GWd/MTU (12.7 %FIMA)
- Average burnup for once burned fuel = 66.2 GWd/MTU (7.0 % FIMA)
- Maximum discharge burnup = 146.4 GWd/MTU (15.4% FIMA)
- Estimated maximum fuel particle burnup: 152.7 GWd/MTU (16.1% FIMA)
- Fast fluence (E>0.27 MeV) for average burnup particle: 4.9×10^{21} n/cm^2
- Estimated Fast Fluence (E>0.27 MeV) for fuel particle with maximum burnup: 6.5×10^{21} n/cm^2

The maximum fuel particle temperature and fuel particle temperature distribution have been estimated by using an external thermal-hydraulic model that uses as input equivalent power distributions to those derived from the neutronic calculations. The results show that the base cycle design needs to be refined using various solutions (axial and radial zoned enrichment, BA rods placement and boron content, packing fraction zoning, etc) in order to meet the requirement of keeping the fuel particle operational temperature below 1350 ºC. The suggested solutions on axial and radial enrichment zoning are an effective mean of reducing power peaking from the beginning of cycle (BOC) to the end of cycle (EOC). A range of acceptable limits and shapes for the radial and axial power distributions have been investigated. A specific feature that is characteristic to this reactor is operation with an axial offset (skewed power peak distribution to the upper half of the reactor) observed in preliminary neutronic analyses that included thermal feedback. This shape is beneficial for keeping the maximum fuel temperature below the operational limit, and additional measures to preserve it during the whole cycle length are necessary.

It is estimated that an axial enrichment gradient of 2-4 wt% combined with a higher enrichment (up to 1wt%) in the middle ring of the fuel assemblies at each axial location are necessary to keep the shape of the radial and axial power within the calculated preliminary limits. Local enrichment and packing fraction zoning at fuel element periphery facing the reflector are also recommended to control local peaking. Use of BA rods is also beneficial in providing the necessary hold down for BOC core reactivity and also in reducing radial power peaking at BOC. The solutions investigated point out the complexity of the required final neutronic design and the preliminary nature of the calculations performed in this document.

The calculated distribution of fuel temperatures in the core at end of the cycle assuming the suggested distribution of axial and radial power is presented in Figure 3.1.

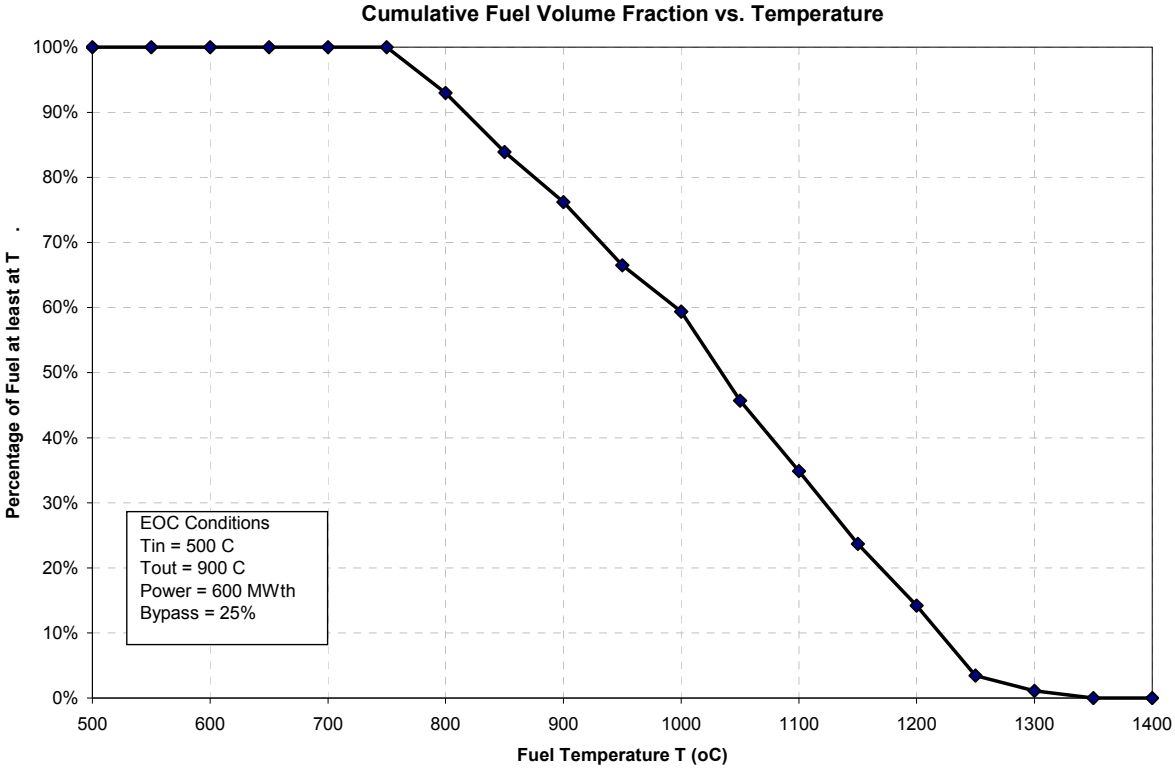

Figure 3.1 - Distribution of Fuel Particle Temperatures at EOC (Equilibrium core)

3.5 Fuel Cycle Design: First Core

A separate set of calculations were performed for the first transitional core for the two-batch fuel management scheme. In designing this core, one needed to assume that the fuel elements with 12 wt% enriched fuel with a packing fraction of 0.25 would be present in the subsequent equilibrium cores. Lower enrichments were found for fractional loading of the first core that preserves the length of the first and subsequent cycles. At the end of the first cycle, the burned lower enrichment fuel is replaced with standard fresh 12 wt% enriched fuel, allowing a smooth transition in the subsequent cycles toward equilibrium cycle.

Different alternatives were investigated and, for the first transitional core of the base case, a core half loaded with fresh 5 wt% enriched fuel and the other half with fresh 12 wt% enriched fuel have been identified as acceptable. The 12 wt% enriched fuel has a packing fraction of 0.25 and the 5 wt% fuel has been tested with different packing fractions. A packing fraction close to 0.35 for the low enrichment fraction of fuel results in assuring the required length of the first cycle and acceptable power peaking.

3.6 Summary of Results for First Core:

The calculated first core has the following characteristics:

- Power density: 99.9 W/gU
- U235 mass/standard fuel element (12 wt% enrichment and PF of 0.25) = 649.2 g
- U235 mass/standard fuel element (5 wt% enrichment and PF of 0.34) = 367.9 g
- Estimated cycle length (equilibrium): 548.0 EFPD
- Estimated core burnup = 58.9 GWd/MTU (6.2% FIMA)

Using the same approach as that presented for the equilibrium core calculations, the maximum fuel particle temperature and temperature distribution have been estimated by using an external thermal-hydraulic model that uses as input equivalent power distributions to those derived from the neutronic calculations. The initial results show that the first cycle design needs also to be refined using similar solutions as those discussed above (axial and radial zoned enrichment, BA rods placement and boron content, packing fraction zoning, etc) in order to meet the requirement of keeping the fuel particle operational temperature below 1350 ºC. Besides the enrichment zoning for the 12 wt% enriched fuel, additional enrichment zoning is also required for the low enriched (5 wt%) fuel fraction. Both an axial gradient and radial gradient for enrichment is necessary to be present for meeting the requirements of maximum fuel particle temperature. The range of enrichment suggested for the low enriched fuel fraction starts around 2 wt% (fuel element periphery) and can go to 10 wt% in the central ring of fuel elements placed in the upper half of the reactor. BA rods presence is also necessary for the initial hold down of reactivity in the core and to flatten the initial radial peaking.

3.7 Fuel and Compact Design Envelope

The findings regarding fuel enrichment and compact packing fractions required to preserve the parameters of the core within the accepted operational limits are summarized in the following.

A cycle length of minimum 18 month is achievable for an average enrichment for the equilibrium cycle between 12 and 13 wt%. A longer cycle of up to 24 month can be obtained using the same management scheme (two–batch cycle) for average enrichments between 14 and 15 wt%. The requirements for keeping the fuel particle operational temperatures below 1350 ºC impose utilization of both axial and radial enrichment zoning. Enrichment zoning is necessary at fuel element level (axially and radially) to appropriately shape the overall power distribution in the core (axial offset, radial power flattening during cycle) and also at local level (fuel element periphery at interface with reflector) to control local peaking. It is estimated that a range of ±2 wt% minimum, centered on the average enrichment value for the equilibrium and initial cycle, are

necessary to provide this flexibility. If the uncertainty of the methodology is factored in this estimation, the envelope for the enrichment spans from approximately 2 wt% (low enrichment fuel fraction for the initial core) to 18 wt% (upper enrichment value for a 24 month equilibrium core).

The compacts with a fuel packing fraction around an optimum value of 0.25 are providing the required cycle length for the selected average fuel enrichment. The specific needs of the initial core require increasing this value for the low enriched fuel to 0.35. On the other hand, local packing fraction zoning in the periphery of the fuel elements facing the reflector is an alternate solution for reducing local peaking. The reduction in power peaking can be obtained by lowering the packing fraction to as much as 0.15. Summarizing the findings, an envelope for the fuel packing fraction that is expected to cover the fuel cycle design needs spans from 0.15 to 0.35 (current manufacturing upper limit). Figure 3.2 summarizes the description of the estimated envelope.

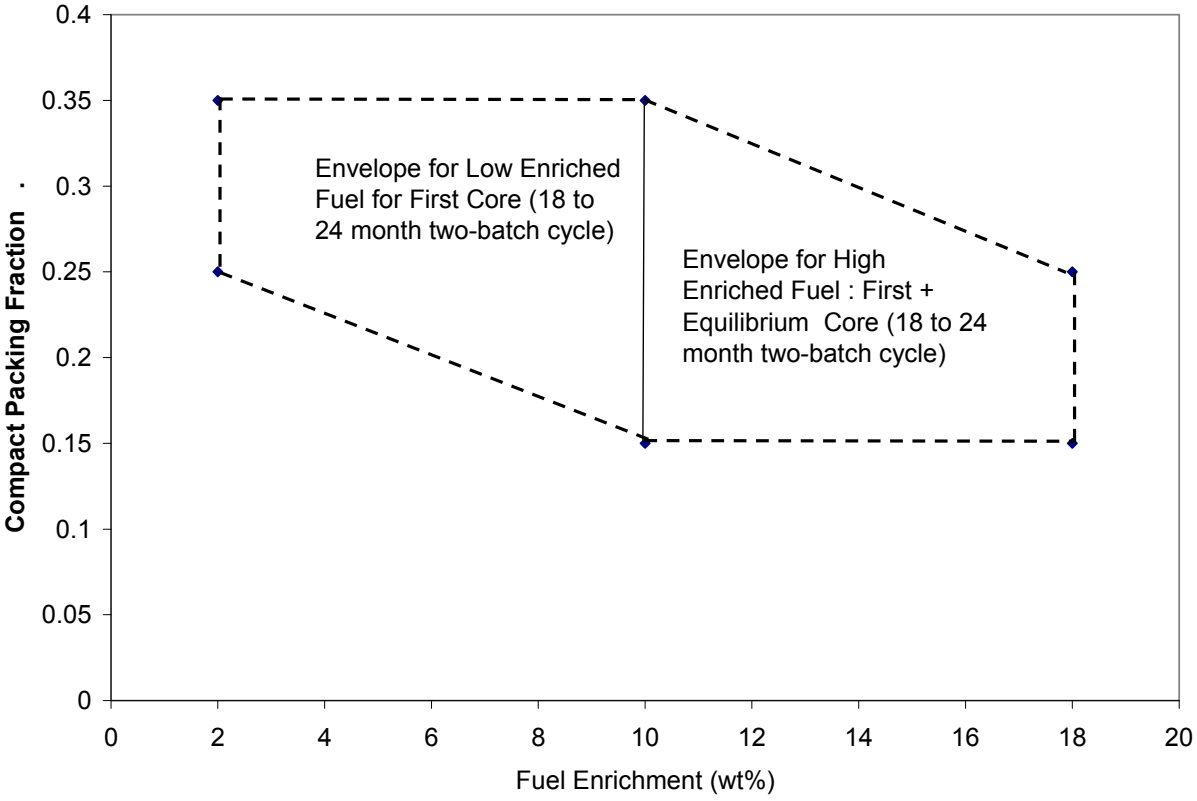

Figure 3.2 - Estimated Envelope for Packing Fraction and Fuel Enrichment for NGNP fuel

3.8 Impact of Compact Density Variation

During the course of review of this report, a question was posed, namely, what would be the expected impact of increasing the compact matrix density from 1.1995 to around 1.5, including neutronic and temperature impacts? This question is explored below.

The neutronic impact is insignificant. Parallel runs of the lattice code APOLLO with the two values for compact graphite density produced almost identical results.

The thermal impact is more difficult to evaluate because the change in graphite density affects the "equivalent thermal conductivity" of the compact (graphite + TRISO particles). The equivalent thermal conductivity estimation was performed in a previous AREVA study using a detailed finite element model of the compact (graphite matrix +TRISO particles).

The results showed that an increase of 50% of the thermal conductivity of the compact graphite resulted in an increase in the effective thermal conductivity of the compact of approx 7 %. This finally translated in a decrease of the central temperature in the hottest fuel particle of less than 10 ºC. The thermal conductivity increase due to graphite density increase from approximately 1.2 to 1.5 g/cc will be lower than the above estimation. Therefore, the estimated temperature impact will be a decrease of the temperature of few degrees in the center of the fuel particles.

4.0 Accident Performance

A series of representative, conservative depressurized conduction cooldown calculations were performed to assess the expected impact of the key fuel performance parameters on the plant safety case. The results of these analyses will be used to determine the accident temperature range requirements (time at temperature) that the fuel must meet. From these will be developed a bounding profile for the average and peak fuel temperatures applicable to the post-irradiation annealing tests during the qualification process..

4.1 The Model

Calculations were performed using STAR-CD, a general-purpose finite-volume heat-transfer and computational fluid dynamics (CFD) code. This code is capable of modeling heat transfer by conduction, convection, and radiation in arbitrary geometries. The system that is modeled here consists of physical phenomena that occur on a wide variety of temporal and spatial scales—more than are typically modeled by modern CFD software and a traditional CFD approach. Thus, the STAR-CD code was enhanced by a set of additional subroutines to model the hydraulic resistance and heat transfer in the coolant channels, the heat transfer across the reactor cavity to the RCCS, and the thermal output of the reactor core. These subroutines also provide the temperature-specific properties of the reactor materials.

The model itself is a three-dimensional representation of the reactor vessel and its internal components. Only a 30° section of the reactor is explicitly modeled and the 12-fold symmetry of the core in the circumferential direction is used to represent the rest of the core through symmetric boundary conditions. Obviously, features that cannot be represented by this symmetry, such as the exit to the cross-duct, are necessarily excluded from being modeled explicitly.

The graphite fuel elements and reflector blocks are assumed to be old, and thus, their thermal conductivity has been reduced by the effects of irradiation from the core. The reactor vessel and internal metallic components are constructed from Mod9Cr steel.

The conduction cooldown scenario was numerically simulated by first determining the steady-state solution that describes the conditions during normal operation. Then, the thermal field of this solution was used as the initial condition for the conduction cooldown calculation, which was performed for 500 hours of simulated time.

4.2 The Cases

For the conduction cooldown cases considered, the parameter that has the greatest impact on the calculated temperatures is the axial power profile. In order to understand the variation in calculated temperatures as a function of this parameter, three cases were considered.

Case 1 - This case uses the polynomial curve that was used for previous parametric studies conducted as part on the NGNP project.

Case 2 - This case uses the axial power distribution taken from the results of a calculation using an axial profile skewed to the top of the core using the AREVA CABERNET code, including temperature feedback effects. This case represents a beginning of cycle condition, that is, no burnup.

Case 3 - This case uses the axial power distribution taken from the results of the CABERNET analysis with a burnup of 60,000 MWd/ton.

The axial profiles for these three cases are depicted in Figure 4.1.

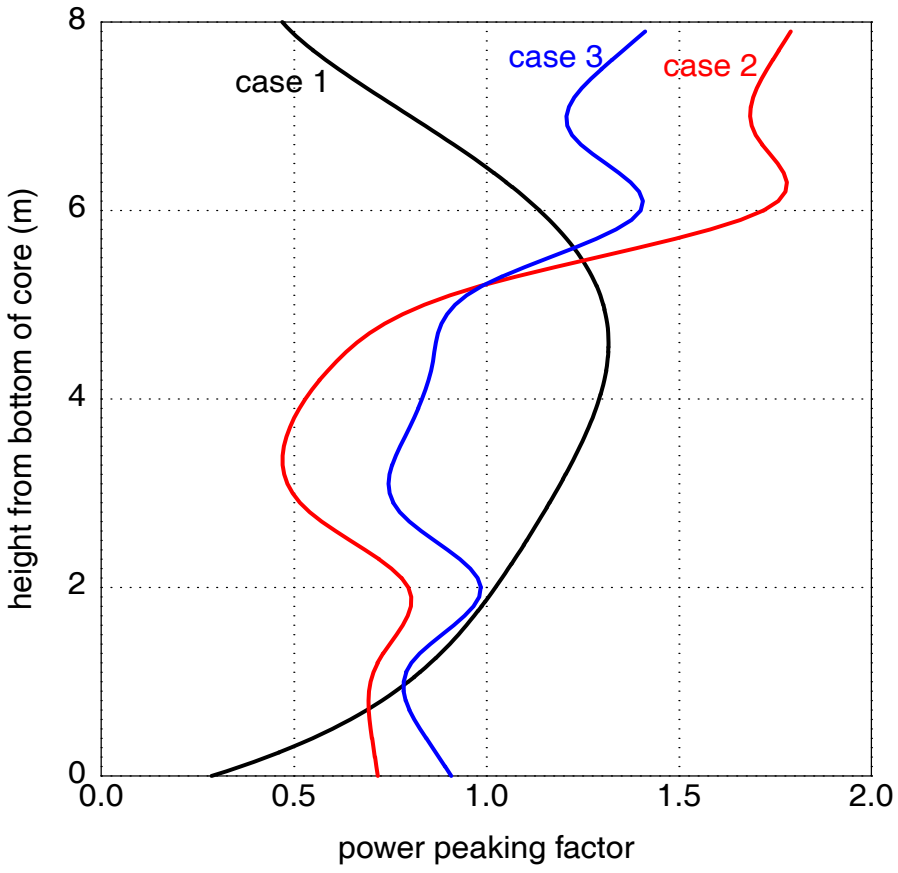

Figure 4.1 - Axial Power Distribution for the Three DCC Cases

The important parameters that make these cases conservative for the fuel are:

- Power level 565 MW + 2.66% = 580 MW
- Graphite reflectors irradiated with k 17% lower
- Residual power 6.6% higher
- Steel emissivity reduced to 0.633

In addition, other key parameters used in these analyses are:

- Pressure 5 MPa
- Inlet 500 degC
- Outlet 900 degC
- Fuel blocks irradiated
- Graphite emissivity 0.8
- RPV Mod9Cr steel, 15 cm thick
- Outer RPV emissivity 0.8
- RCCS emissivity 0.8
- RCCS temperature 65 degC
- Bypass flow 5%

4.3 Case 1 Summary of Results

Key results for the DCC base case are as follows and presented on Figures 4.2 and 4.3.

Normal Operation

Peak temperatures (°C)

Fuel blocks	1029
Fuel compacts	1229
Reactor pressure vessel	408
Core barrel	535

Conduction Cooldown

Peak temperatures (°C)

Fuel blocks	1654
Reactor pressure vessel	456
Core barrel	757

Time of peak temperature (h)

Fuel blocks	93
Reactor pressure vessel	111
Core barrel	107

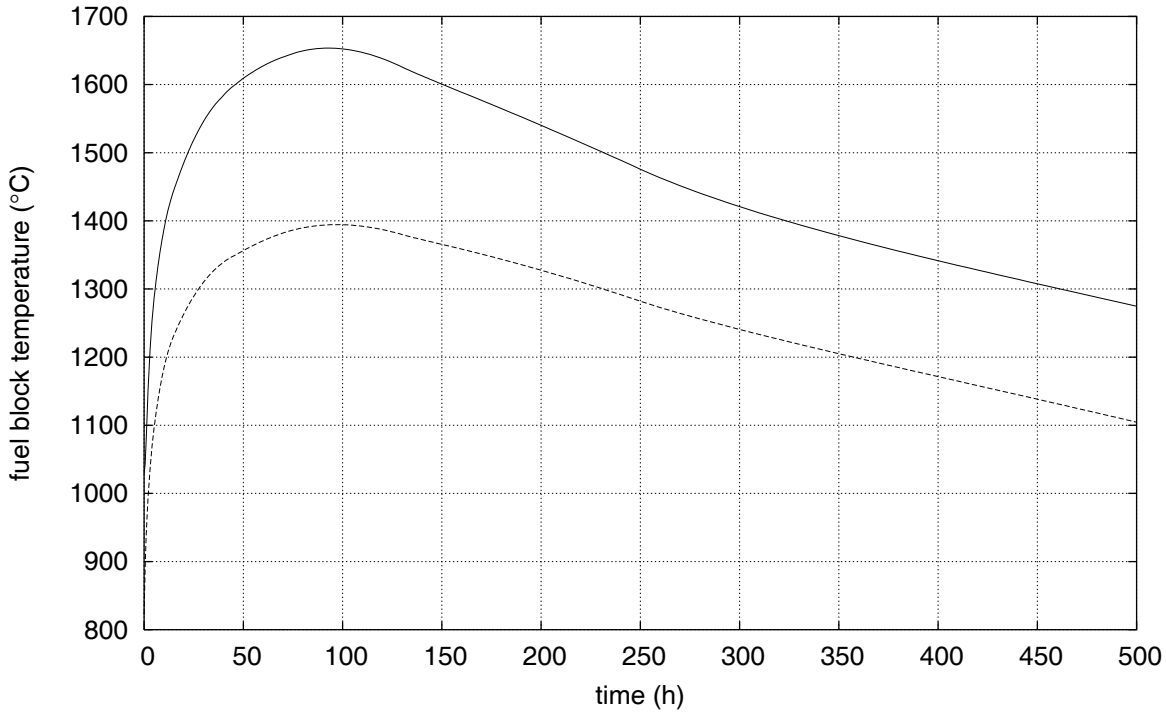

Figure 4.2 - Evolution of the maximum (solid line) and average (dashed line) temperatures of the fuel blocks during the transient for case 1

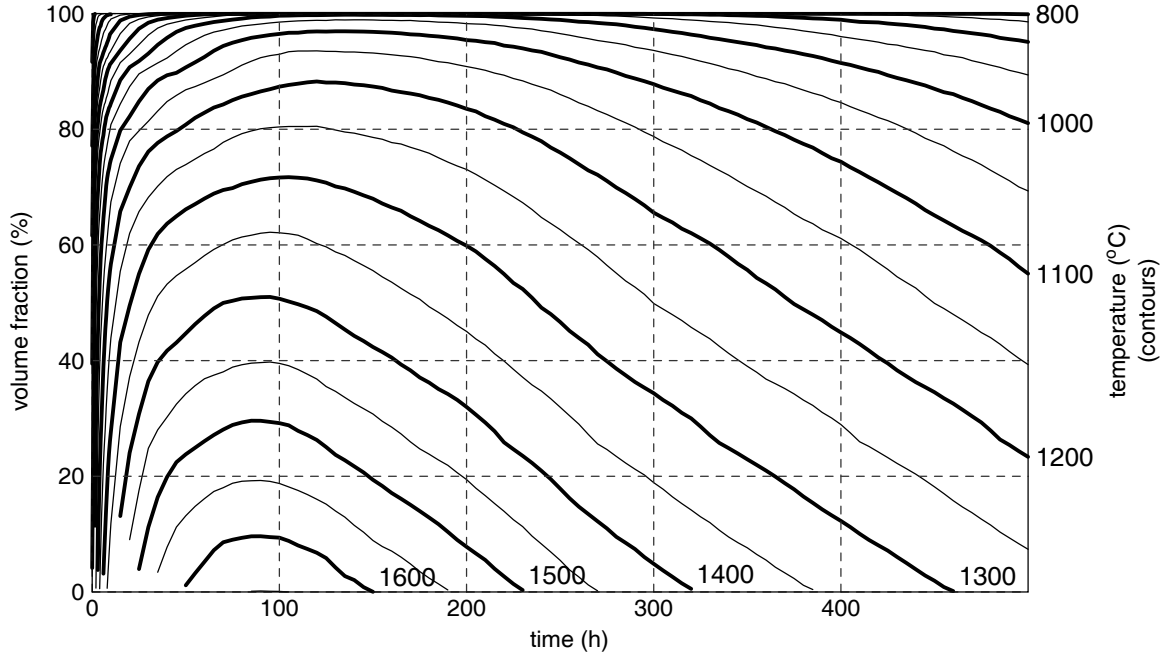

Figure 4.3 - Time evolution of the volume distribution of the temperatures of graphite during the transient for case 1

4.4 Case 2 Summary of Results

Key results for the CABERNET axial profile, zero burnup case are as follows and presented on Figures 4.4 and 4.5.

Normal Operation

Peak temperatures (°C)	
Fuel blocks	1043
Fuel compacts	1245
Reactor pressure vessel	411
Core barrel	537

Conduction Cooldown

Peak temperatures (°C)	
Fuel blocks	1662
Reactor pressure vessel	448
Core barrel	749
Time of peak temperature (h)	
Fuel blocks	95
Reactor pressure vessel	122
Core barrel	117

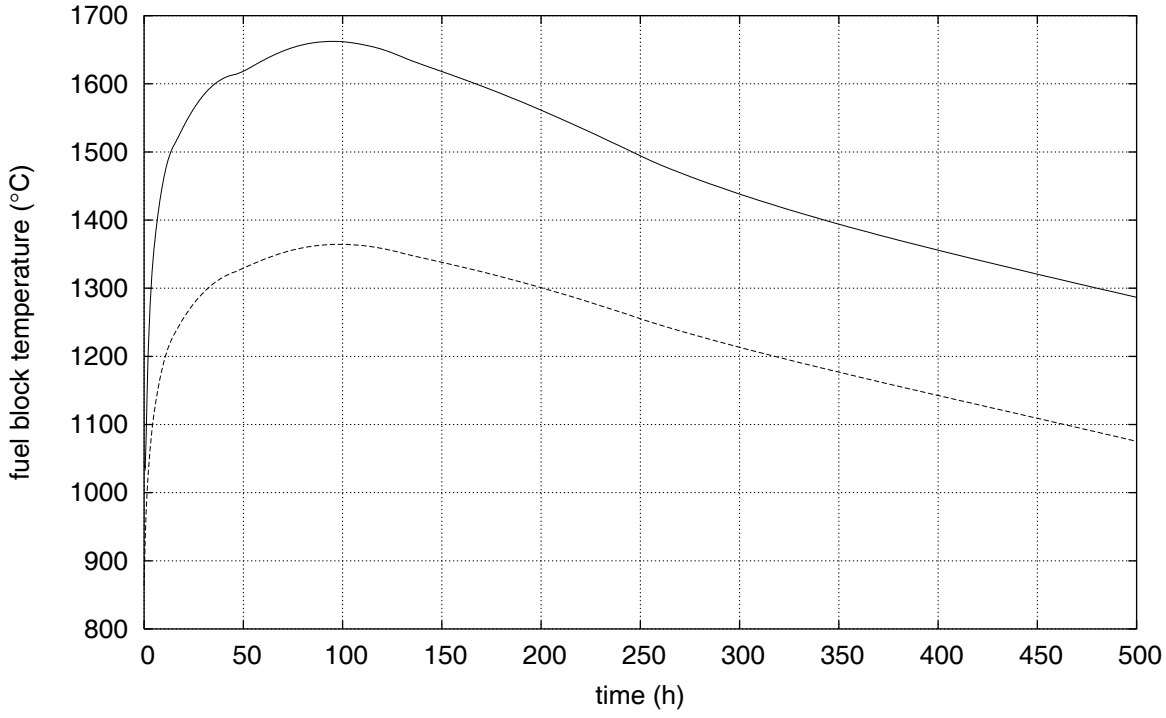

Figure 4.4 - Evolution of the maximum (solid line) and average (dashed line) temperatures of the fuel blocks during the transient for case 2

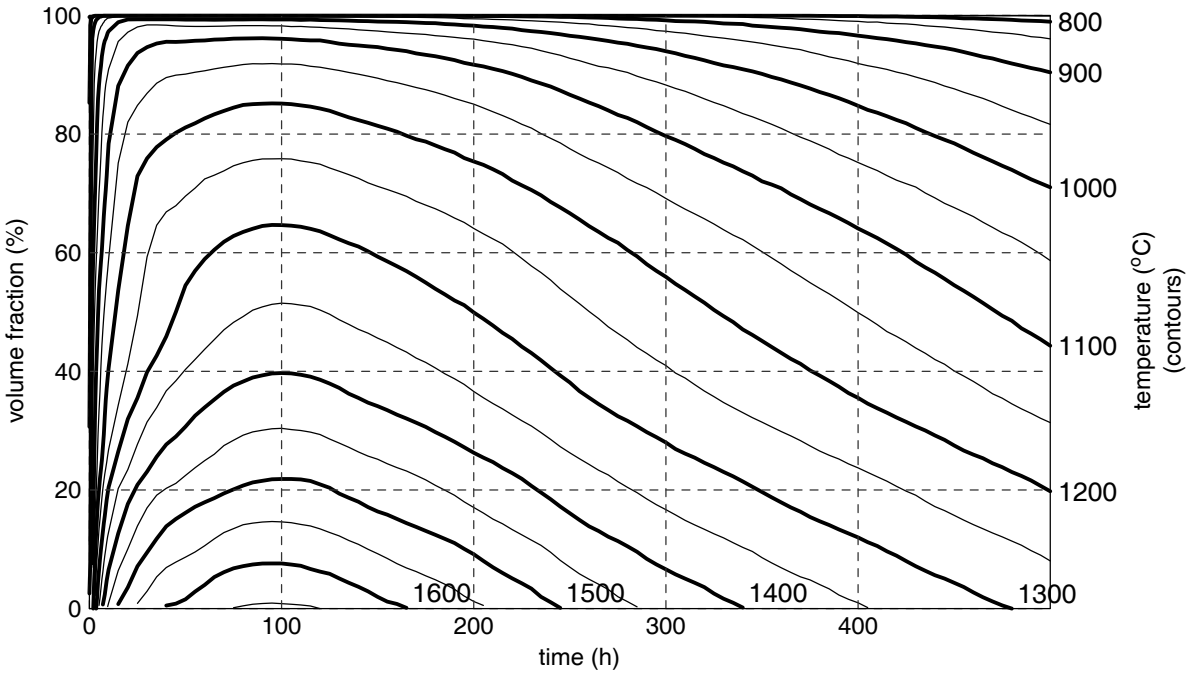

Figure 4.5 - Time evolution of the volume distribution of the temperatures of graphite during the transient for case 2

4.5 Case 3 Summary of Results

Key results for the CABERNET axial profile, 60,000 MWd/ton burnup case are as follows and presented on Figures 4.6 and 4.7.

Normal Operation

Peak temperatures (°C)	
Fuel blocks	1067
Fuel compacts	1272
Reactor pressure vessel	410
Core barrel	536

Conduction Cooldown

Peak temperatures (°C)	
Fuel blocks	1560
Reactor pressure vessel	434
Core barrel	727
Time of peak temperature (h)	
Fuel blocks	103
Reactor pressure vessel	122
Core barrel	119

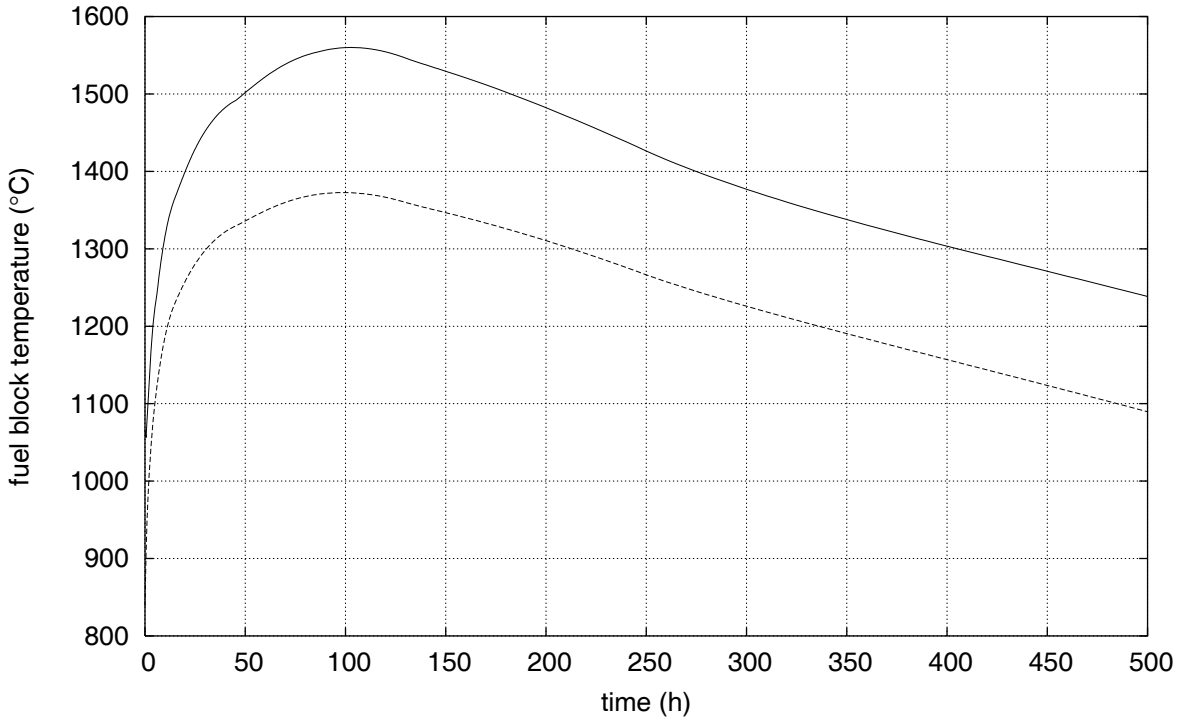

Figure 4.6 - Evolution of the maximum (solid line) and average (dashed line) temperatures of the fuel blocks during the transient for case 3

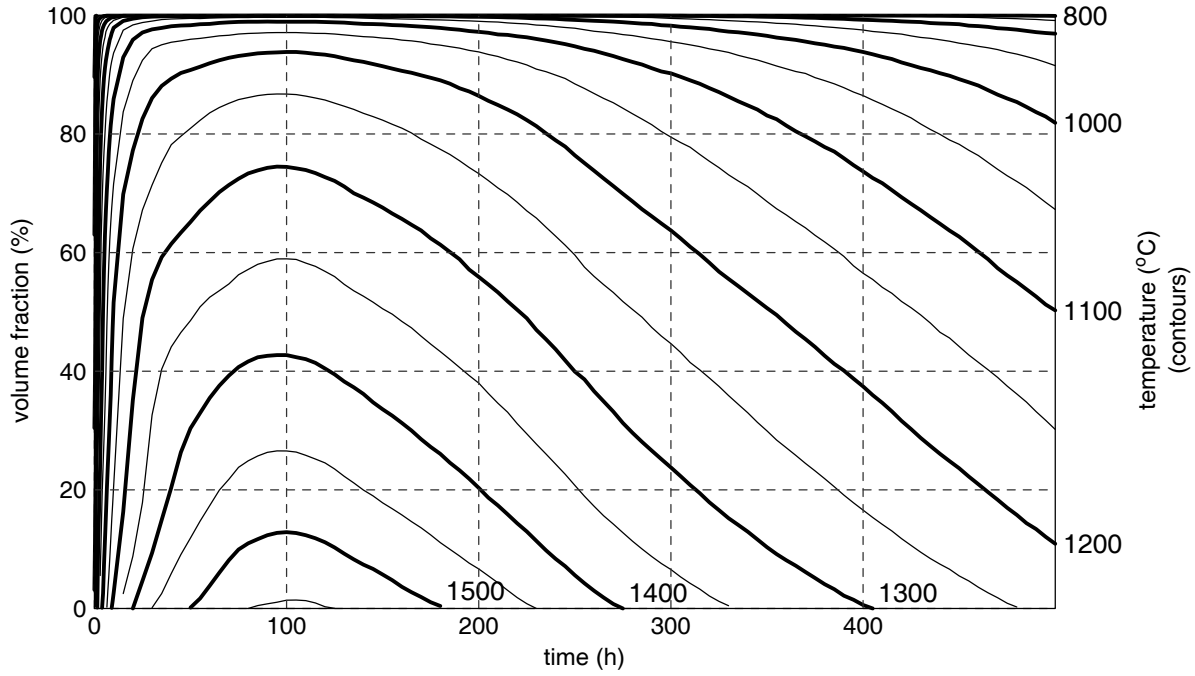

Figure 4.7 - Time evolution of the volume distribution of the temperatures of graphite during the transient for case 3

4.6 Overall Results

Figure 4.8 presents the maximum temperatures for the three cases as a function of temperature. It can be seen that the limiting case is the zero burnup CABERNET case, closely followed by the DCC base case. All of the cases reach a maximum temperature at around 100 hours.

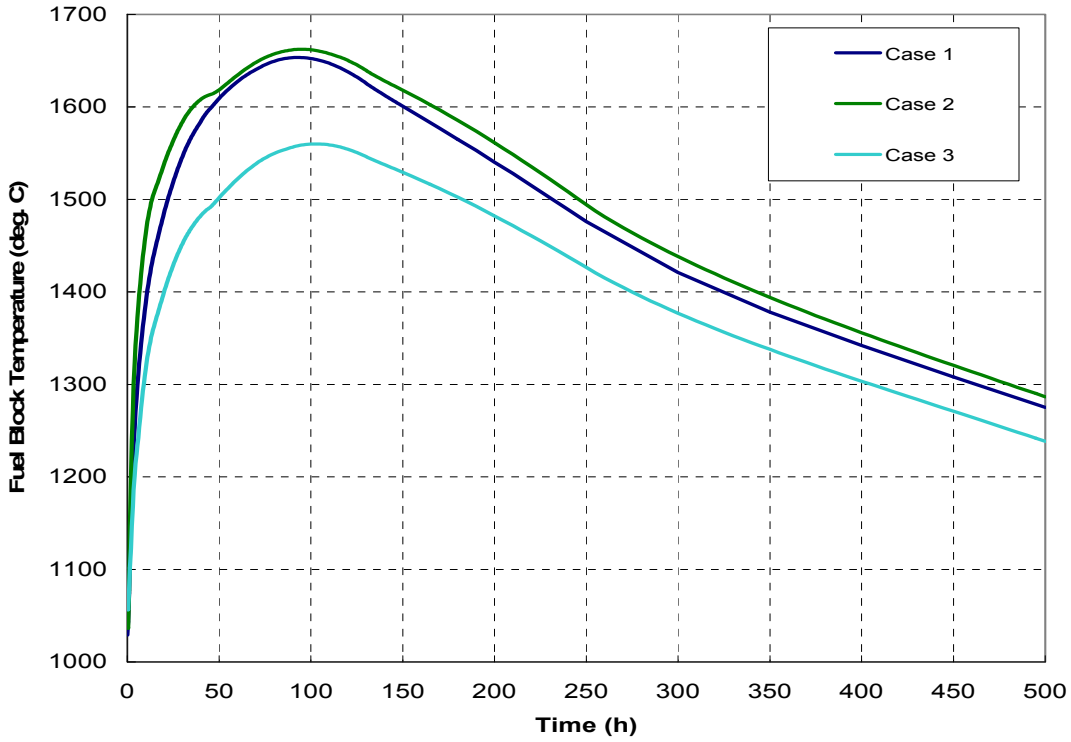

Figure 4.8 - Evolution of the maximum temperatures of the fuel blocks during the transient for the three cases

Figure 4.9 presents the temperature distribution, by volume, within the reactor active core at the time of the peak temperature for each case. It can be seen that under all of the Cases considered, the vast majority of the fuel within the active core remains below a temperature of 1600°C.

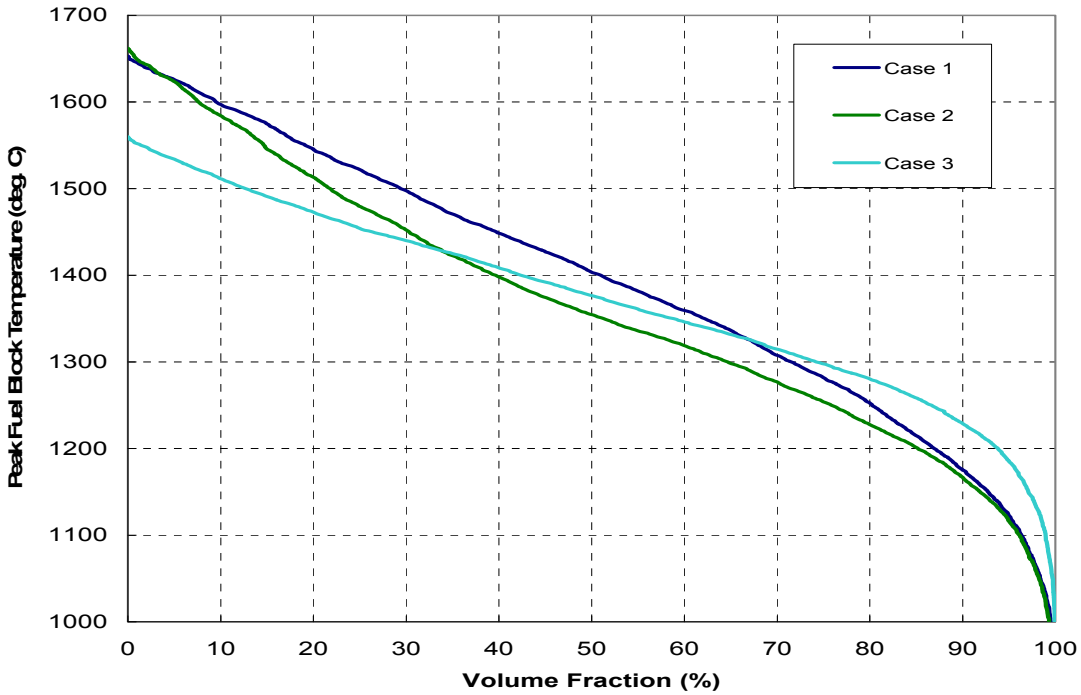

Figure 4.9 - Volume distribution of peak temperatures of graphite in the active core at the time during the transient when peak temperature occurs

5.0 Conclusions and Recommended Test Parameters

A study of the impact of changing fuel enrichment, fuel packing fraction and boron content of BA rods on cycle parameters has been performed on the NGNP prismatic HTR reactor with the purpose of identifying a fuel design envelope for the fuel qualification process.

The results show that a range of fuel enrichments between 2 to 18 wt%, a packing fraction between 0.15 and 0.35, and a B_4C particle packing fraction between 0.006 and 0.12 are necessary to address the foreseeable fuel design needs for the NGNP prismatic reactor using a two-batch fuel management scheme. The suggested envelope for fuel enrichment and packing fraction is depicted in Figure 5.1.

The following operational limits have been conservatively estimated for the fuel particles by evaluating a simplified two-batch 18 month long fuel cycles without optimization:

- Equilibrium Cycle (average fresh fuel enrichment 12 wt%, PF=0.25):
- Average kernel particle burnup: 120.4 GWd/MTU (12.7% FIMA)
- Maximum kernel particle burnup: 152.7 GWd/MTU (16.1% FIMA)
- Average kernel particle fast fluence (E> 0.27 MeV): 4.9×10^{21} n/cm^2

- Maximum kernel particle f ast fluence: (E> 0.27 MeV): 6.5×10^{21} n/cm^2
- Maximum fuel kernel temperature: 1349 ºC

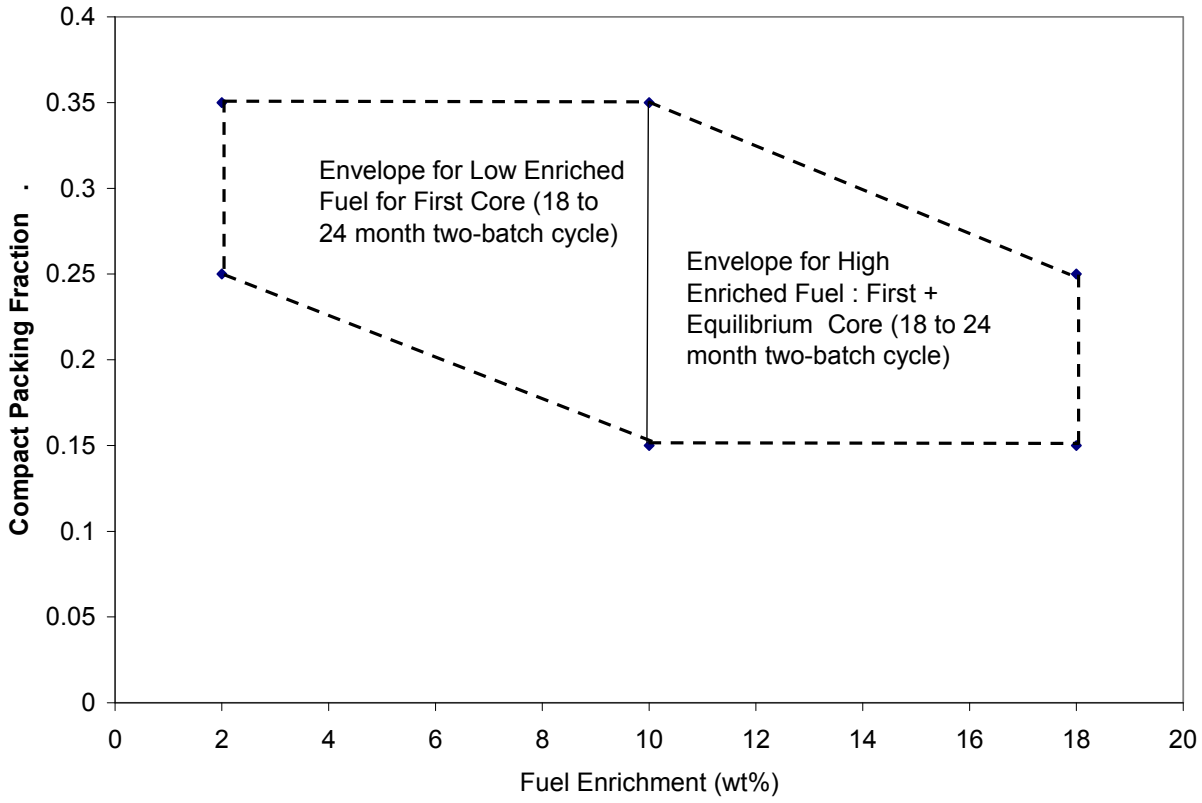

Figure 5.1 - Recommended Envelope for Packing Fraction and Fuel Enrichment for NGNP fuel

5.1 Fuel enrichment

The results of the present study suggest that an equilibrium cycle length above 18 months and up to 24 months can be obtained for the fuel with 425 μm diameter UCO kernel by using a range of average core enrichments above 12 wt% and up to 16 wt%. The need for axial and radial enrichment zoning increases this range to 10% for lower end and 18wt% for upper end. If the needs for the initial core and for the transitional cores are added in this equation, the enrichment range needs to be extended to lower enrichments (with a range between 2 to 10% wt% to accommodate the need for zoned axial and radial enrichment in both transitional cores and possibly equilibrium cores). The calculations show that core optimization can be achieved in a more predictable way by using fuel enrichment coupled with the use of BA rods, recommending enrichment as the parameter of choice for cycle optimization. Taking into account the uncertainties identified, the enrichment range to be used for this type of fuel should be larger than that already

mentioned. A suggested range between 2 to 18wt % should respond to all foreseeable needs of both equilibrium and transitional cores. Figure 5.1 shows a schematic representation of the estimated envelope.

5.2 Packing fraction

The calculations performed for this fuel (425 μm $UC_{0.5}O_{1.5}$ kernel) show that the core is extremely sensitive to this parameter and its impact is less straightforward to assess. The results suggest that an optimum cycle length that responds to the requirement of minimum 18 months for cycle length is obtained for packing fractions close to 0.25. Two-batch cycle management scheme calculations show that the optimum utilization can be reached at slightly higher values of the packing fraction allowing also an increase in cycle length. On the other hand, packing fraction proved also to be an effective solution for reducing radial peaking in inner and outer fuel rings by placing compacts with lower packing fractions in the rows of fuel rods placed closer to the reflector region. The range for reduced packing fractions needs to be extended to values as low as 0.15 to include a significant effect on peaking from BOC to EOC.

A special case is raised by the first core, where using lower enriched fuel with a higher packing fraction improves power peaking distribution and slightly increases the cycle length.

Taking into account the identified uncertainties, a packing fraction range between 0.15 and 0.35 (see Figure 5.1) seems to be sufficient to address fuel zoning needs necessary to respond to the current cycle requirements.

5.3 Boron Content in BA rods

Varying packing fraction in BA rods proved to be a very effective solution for holding down the initial reactivity of the core without significantly affecting the cycle length. At relative low concentrations BA rods tend to deplete before the end of cycle. Further enhancement of the geometric placement of the BA rods or increasing their number can be investigated to further improve local pin peaking, if necessary. A range of the B_4C particles packing fraction, based on 200μm particles, of up to 40 times that of the base case (0.006 to 0.24) seems to provide sufficient space for accommodating the existing requirements.

5.4 Post-Irradiation Annealing Temperature

The recommended temperature-time profile for the post-irradiation annealing test is based on the results of the DCC calculations, and is selected to be bounding of all of the results with some margin. Based on the conservatisms built into the various calculations presented here, including core neutronics and depressurized conduction cooldown, such

margin does not need to be significant in order to provide reasonable assurance that future core designs will be successfully bound by the qualification activities based on this envelope. The enveloping line, presented in Figure 5.2, provides a margin of approximately 13°C at the peak temperature of 1675°C.

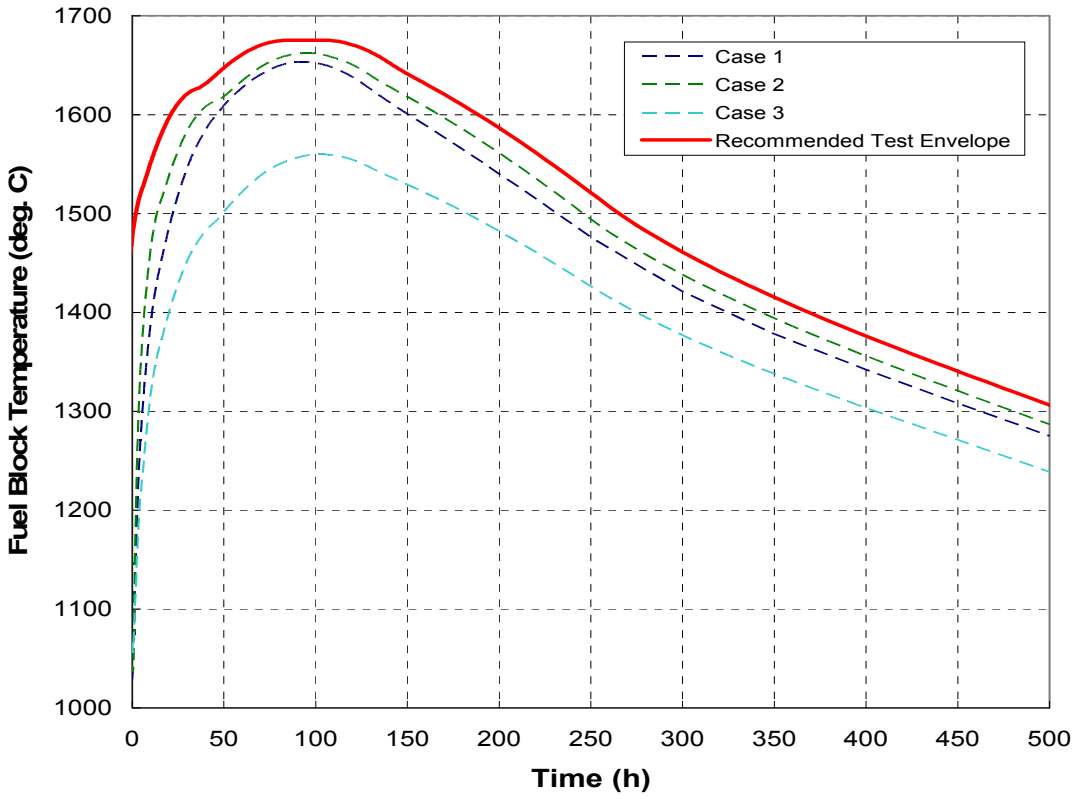

Figure 5.2 - Recommended Post-Irradiation Anneal Temperature Profile

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