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THE FAST-MIXED SPECTRUM REACTOR PROGRESS REPORT FOR 1979

Prepared by
G.J. FISCHER AND R.J. CERBONE

MASTER

May 1980

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THE FAST-MIXED SPECTRUM REACTOR PROGRESS REPORT FOR 1979

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EXECUTIVE SUMMARY

This report presents a general summary of the work performed on the Fast Mixed Spectrum Reactor (FMSR) during 1979. In particular, the report focuses on work completed since the publication of the Initial Feasibility Study⁽¹⁾ (January 1979). In the Initial Feasibility Study, a Reference Design was selected which employed helium cooling and a specific fuel cycle in which natural uranium was fed into the outermost part of a moderated region and then was successively shuffled toward the central fast region, from which it was discharged after a long residence there. The results of that study indicated that an FMSR operating on a once-through fuel cycle was feasible provided the reactor constituent materials could withstand peak fast fluences of approximately 8.5×10^{23} (nvt > 0.1 MeV). A resource utilization advantage factor of about four relative to the LWR was shown.

The initial studies have been extended to include several design variations. These include a centrally-moderated designed (CM)FMSR, a variation of the Reference Design, and an all-fast FMSR. Increased emphasis has been placed on the sodium-cooled versions of these designs. Studies are under way to investigate an FMSR which would utilize limited fuel reprocessing.

The centrally-moderated FMSR has been shown to have several advantages over the Reference Design. These include a reduction in reactivity swing over the fuel cycle, elimination of the fuel bearing control rods, extension in fuel cycle length, and reduction in power density swings during fuel residence in any given position. An all-fast FMSR was studied in order to test the benefit which might be gained through removal of the moderator near the radial outer perimeter of the FMSR. As expected, the neutronic performance was improved

somewhat, but the falloff of the radial power profile was considerably worse without the moderator. This could have serious deleterious effects on the thermodynamic efficiency.

In addition to the design studies, improvements have been made in the methods development areas, including reactor physics methods such as implementation of hexagonal geometry in the diffusion theory calculations, improved fission product libraries, and improved thermal-hydraulic methods. Comparisons have been made between BNL, ANL, ORNL and MIT methods and data; the results are reported in a topical report being issued at the same time as this report. Comparisons of thermal-hydraulic results have been made, where possible, with analyses performed elsewhere. As in the physics analyses, the comparisons show overall consistency of the BNL methods. Preliminary design studies were begun for an FMSR concept operating under very limited, controlled, fuel reprocessing. This design shows excellent promise with regard to performance within achievable material limits, low fuel cycle costs, and safety.

Little additional work has been done in the important fuels and materials area. Liaison between BNL, ANL, HEDL, GA, and MIT was maintained.

1. INTRODUCTION

1.1 OBJECTIVE OF FMSR PROGRAM

This report summarizes the progress of the Fast Mixed Spectrum Reactor (FMSR) since the publication of the Interim Report⁽¹⁾ in January 1979. The FMSR program was initiated to determine the feasibility of a breeder reactor concept which operated on a once-through-and-store fuel cycle and for which the only feed would be natural uranium. A first or startup core enriched to a maximum of about eleven percent in uranium-235 would be required. The concept has excellent antiproliferation advantages. In the once-through and store mode, the FMSR has a resource utilization which is a factor of four higher than a light water reactor.

1.2 INITIAL STUDY

The Initial Feasibility Study⁽¹⁾ concentrated on reactor physics, fuel cycle and thermal-hydraulics evaluations of the FMSR concept. All aspects of the engineering design employed conventional technology and materials, as far as possible, in order to reduce the need for new research and development. However, the FMSR depends on evolving technologies in the areas of steels for cladding and fuel ducts and metal fuel. It appears that ferritic steels currently under irradiation may be suitable for the FMSR. Mark-II fuel experience in EBR-II is also directly applicable to FMSR design.

A discussion of the Reference Design which evolved from the initial feasibility studies is presented in the following section. This design was compatible with the design objectives of reactor physics, thermal hydraulics and fuels and materials disciplines. Metal fuel of the Mark-II design employed in the

EBR-II reactor was adopted because of its excellent breeding potential. Limited amounts of moderator were employed to achieve radial power flattening. Features of this first design which required improvement and further study were the high fluence damage to the steel on discharge, the 185 full power days fuel cycle was shorter than desired, and for a small number of subassemblies the swing in power density over the full residence time was larger than desired. The design studies reported here were directed toward these problems.

Use of sodium as a coolant was emphasized during this study. Verification of FMSR physics, partly by way of an intercomparison study of FMSR analysis methods,⁽²⁾ indicated that the fission products treatment should be improved beyond the capability which existed at the time of the initial study.

1.3 CONTENT OF PRESENT STUDY

Following a brief review of the Reference FMSR Design, this report presents the basis for a considerably improved FMSR design, namely, the centrally-moderated FMSR. Results of design studies on this concept are given in Section 3 for sodium-cooled as well as helium-cooled design versions. Thermal-hydraulic analysis capabilities have been improved to include hot spot and hot channel analyses. Improved fission product treatments are still undergoing development and testing and will be reported separately. Unfortunately, little fuel design or fabrication before irradiation testing was possible due to lack of funding. Similarly, safety analyses were not begun for the same reason.

Many gains were achieved through adoption of the centrally-moderated FMSR. The fuel cycle is nearly one year in length; the local power density swing was considerably reduced, and the reactivity swing over the fuel cycle was reduced by almost a factor of two. Reactivity control and maintenance of power density distributions over the fuel cycle have been simplified.

A limited evaluation was made of the engineering potential of an all-fast FMSR reactor concept in which all of the moderator is removed. This work is described in Section 4. It was found that the breeding performance increased, as predicted, when the moderator was removed. However, it was also found, again as expected, that removal of this strategically located moderator considerably increased the falloff of radial power distribution. The hot channel cooling requirements for these outer subassemblies resulted in considerable undercooling for the total subassembly. This could reduce the overall mixed mean outlet temperature and the thermodynamic efficiency of the reactor.

2. DESIGN STRATEGIES AND OPTIONS

2.1 REFERENCE DESIGN FOR INITIAL FEASIBILITY STUDY

The results of an initial feasibility study of the Fast-Mixed Spectrum Reactor concept were published in the earlier Interim Report.⁽¹⁾ The results showed that the concept was feasible with respect to reactor physics and thermal hydraulics. The reactor design for the study was not optimized, but was adopted merely to demonstrate the feasibility of the concept. In this section, changes in the computed performance of this design since the publication of the Interim Report are reported in order to provide a basis of comparison to later calculations. The changes are due to improvements in the cross-section sets and the treatment of fission products. For the purpose of comparison with alternate reactor configurations and fuel management schemes, the above design can be treated as the Reference Design. Fuel management strategies modified to reduce the peak fluence and burnup are outlined in Section 2.3.

Figure 2.1 is a simplified cross-sectional diagram of the reference helium-cooled FMSR core. The core consists of a large central fast region surrounded by a moderated region. Most of the power, and hence reactivity, is provided by the fast region. Although the selected moderator for this study was beryllium, graphite could be used as an alternative. Table 2.1 compares some of the main core design parameters with those of the Gas-Cooled Fast Breeder Reactor Demonstration Plant. The gas-cooled FMSR uses metal fuel with a volume fraction of 0.39. The use of metal fuel and the relatively high fuel volume fraction are essential in order to maximize the breeding ratio of the reactor.

Figure 2.2 shows the schematic R-Z model of the reactor used in the reactor physics calculations. The reactor has been divided into 34 radial subzones

Figure 2.1. Cross-Sectional View of the Reference FMSR

Table 2.1

General Design Parameters of Gas-Cooled FMSR Reference Design
and Comparison to GCFR Demo Design

<u>Parameter</u>	<u>FMSR</u>	<u>GCFR Demo</u>
Reactor Power MW(e)	1000	300
Active Core Height, cm	160	100
Cladding O.D., cm	0.8804	0.7400
Cladding Thickness, cm	0.0432	0.0483
Fuel Pellet O.D., cm	0.7940	0.6333
Fuel-Cladding Gap, cm	0.0	0.0051
Duct Wall Thickness, cm	0.254	0.254
Hexcan Size (Inner) cm	18.69	16.55
Fuel Volume Fraction	0.39	0.277
Flowing Coolant Volume Fraction	0.40	0.40
Pitch, cm	1.1354	1.0000
P/D	1.29	1.35
No. of Pins/Subassembly	271	271
No. of Fuel Subassemblies	408	118
Number of Spacer Grids	15	10

Figure 2.2. Schematic R-Z Representation of the Helium-Cooled Reference FMSR (1/4 core)

containing 12 fuel subassemblies each, in order to accurately assess the buildup of plutonium and fission products. Each subzone is represented by a ring in the R-Z geometry. In reactor physics calculations leading to the equilibrium burnup cycle, fuel shuffling is achieved by moving one subzone at a time from one radial position to another. Actual physical shuffling, as explained later in this section, employs only six broad zones so that a fuel subassembly would be moved only six times during its residence in the reactor. For the purpose of calculational convenience, each half of the reactor (above and below the midplane) is divided into three axial layers. In the first and second layers, the 34 radial subzones are explicitly represented. In the top axial layer, however, the top 34 radial subzones are grouped into six broad zones because of constraints in computer storage. The first two axial layers are considered to be part of the core while the third axial layer corresponds to an axial blanket. The terms core and axial blanket are strictly valid only for the startup reactor which contains enriched uranium in the first two layers and natural uranium in the top layer. In the equilibrium fuel cycles, there is no sharp distinction between core and axial blanket in the FMSR.

Most of the results reported in the Interim Report were for the equilibrium burnup cycle, where each burnup cycle essentially repeats the previous one. At the end of each cycle, fuel from radial subzone 34 is discharged from the reactor and replaced by fuel from subzone 33, subzone 33 by subzone 32, and so on, until subzone 1 is replaced by fresh fuel subassemblies containing natural or depleted uranium. It is obvious that an equilibrium cycle will only be achieved 34 cycles after reactor startup, i.e., near the very end of a commonly assumed 30-year life of the reactor. By the time equilibrium cycle is reached, all the enriched uranium subzones of the startup core will have been discharged from the

reactor. These will have been replaced by subzones which started with natural uranium fuel and built up their plutonium and fission products during their residence in the reactor. Preliminary calculations have indicated that a smooth transition from the first core to the equilibrium core is possible provided the fuel enrichments used in the startup core are carefully selected. However, additional detailed analyses are needed.

In achieving the equilibrium fuel cycle, all the radial subzones were shuffled at the end of each fuel cycle. In practice, this would be unacceptable. Hence, the proposed fuel management strategy shown in Figure 2.3 is adopted. The 34 subzones are grouped together under six broad zones. At the end of each cycle, one subzone from each zone is moved to a higher numbered zone. A fresh subzone is fed into Zone 1 and one subzone from Zone 6 is discharged after each cycle. Zone 1 has eight subzones. Hence, a fresh subzone fed into Zone 1 resides there for eight cycles before being moved to Zone 2. Zones 2, 3, 4, 5 and 6 have, respectively, six, three, two, six and nine subzones, making a total of thirty-four subzones. The fuel shuffling strategy and the number of subzones in each zone were determined only after a careful study of their impact on variations of radial flux and power shapes. In this fuel management strategy, only twelve subassemblies from each of the six zones need be moved at the end of each fuel cycle, but this also has the effect that each fuel cycle is slightly different from the previous cycle. The Interim Report also contained results of the above fuel management strategy.

By now, it is evident that the equilibrium cycle results are only of academic interest since such a cycle may not be achieved in actual practice. Even so, analysis of the equilibrium cycle provides relevant information regarding the rates of fissile isotope and fission fragment buildup in the different

Figure 2.3. Fuel Shuffling Strategy for the Helium Cooled, Centrally-Moderated FMSR

zones. Actual fuel shuffling strategies can then evolve from the above information. The equilibrium cycle fuel compositions in the various zones can also be used to assess the enrichments required for the startup of the reactor. The results of equilibrium fuel cycle analysis also provide a convenient way to compare the merits of different design and fuel management strategies.

2.2 IMPROVED REACTOR PHYSICS CALCULATIONS FOR THE REFERENCE DESIGN FMSR

Reactor physics calculations reported in the Interim Report were performed using 50-group cross sections obtained from the LIB-IV(3) data set modified using the 1DX computer code(4) to reflect the influence of local core compositions. Fission products were represented by a single lump, i.e., Non-Saturating Fission Products (NSFP) of Pu-239, with a yield of two fission products per fission because the reactor physics 2DB code was only capable of handling one fission product isotope. Fast reactor burnup calculations, in general, use three lumped fission products in addition to Xe-135 and Sm-149, with a total yield of two per fission. Additional fission product lumps are the Slowly Saturating Fission Products (SSFP) and Rapidly Saturating Fission Products (RSFP). NSFP constitutes almost three fourths of the total yield. In general, however, SSFP, RSFP, Sm-149 and Xe-135 have a higher absorption cross section compared to NSFP. Hence, calculations of the absorptions in fission products were underestimated in the previous calculations. The situation was mitigated to a certain extent in the present calculations by assuming a yield of 2.7 NSFP per fission, a number which was determined by comparing the cross sections and yields of the five fission products.

Updated sets of 50-group cross sections at the nominal fuel temperature of 975°K were generated and fuel burnup calculations were performed using 2DB(5)

until an equilibrium cycle was obtained; the results are shown in Table 2.2. The fuel cycle length is 220 days during which time the reactivity increases by 2.1%. The fuel cycle length is selected so that the reactor is critical at the end of the equilibrium cycle, which means the reactor is subcritical by 2.1% at the beginning of the cycle. Fueled control rods were to be used to provide the required reactivity. Preliminary calculations indicate that nineteen control rods with a fuel volume fraction of 25% and an enrichment of 17% U-235 are needed to provide 2.1% reactivity. To compensate for the loss of reactivity of the control rods during their residence in the reactor, it was estimated that the rods would need an enrichment of 20%. These control rods would be replaced by fresh rods twice during the life of the reactor.

Alternatively, the reactor can be controlled by poison control rods such as boron carbide or tantalum; however, the reactor would then have to be critical (with the rods withdrawn) at the beginning of the cycle, instead of at the end. This could be achieved by increasing the fuel cycle length with a corresponding increase in the peak fluence and burnup.

Figure 2.4 shows the average power, in the first axial layer, for the various radial subzones corresponding to the equilibrium cycle. It is seen that for positions close to the center of the reactor the power increases during the cycle. This is due to the buildup of plutonium in these zones during the cycle coupled to an increase in the flux level in the region. In zones farther away from the center, the power may remain the same or drop slightly during the cycle, since the total power of the reactor remains constant. Properly located control rods would greatly reduce this variation.

Figure 2.5 shows the buildup of Pu-239 as a function of exposure time for the two core layers. For the first fourteen cycles, the rate of plutonium buildup is relatively low because the fuel resides in low flux regions. Once

Table 2.2
FMSR Performance
Helium-Cooled Reference FMSR

Number of Fuel Subassemblies	408
Number of Subassemblies Shuffled Per Cycle	72
Cycle Duration (Full Power Days)	220
Reactivity Swing Over Cycle, %	2.1
Breeding Ratio (BOEC)	1.66
Fuel Inventory	
Reactor, BOEC, kg Pu	6823
Discharge/Yr, kg Pu	507
Core Discharge Enrichment (% Pu-239)	6.9
Peak Burnup (MWD/T)	
30-Yr, Fuel Reintroduction	~165,000
Equilibrium Fuel Cycle	212,000
Peak Fluence (E > 0.1 MeV)	
30-Yr, Fuel Reintroduction	~ 7.5 x 10 ²³
Equilibrium Fuel Cycle	9.9 x 10 ²³

RADIAL POWER DISTRIBUTION
REFERENCE FMSR, HELIUM COOLED
B.O.E.C. - SOLID AND E.O.E.C. - DASHED

Figure 2.4.

PU-239 BUILDUP
REFERENCE FMSR, HELIUM COOLED
LAYER 1 - SOLID LAYER 2 - DASHED

Figure 2.5.

the fuel moves into the fast region where the flux and power levels are higher, the plutonium buildup rate increases. The plutonium buildup levels off to about 7.2% and may decrease slightly thereafter. Buildup in the second core layer exhibits the same trend as in the first layer, but lags behind because of the lower flux level compared to the first layer.

Figure 2.6 shows the buildup of fission products as a function of exposure time in the two core layers. It is seen that the burnup in layer 1 proceeds considerably faster than in layer 2 because of the higher flux levels and fissile percentages in layer 1 compared to layer 2. After 34 cycles of exposure in the reactor, the fission product content at discharge is 21.2% in layer 1 and 11.4% in layer 2.

It is worthwhile to briefly compare the main reactor physics results, of the helium-cooled FMSR published in the Interim Report, with the results shown in Table 2.2. An important difference is that the fuel cycle duration has increased from 185 days to 220 days. This is a direct result of improvements made in the treatment of fission products and cross-section sets. As a consequence of the longer cycle length, the peak burnup at discharge increased from 183,000 MWD/T (Interim Report) to the present value of 212,000 MWD/T. The peak fast fluence exposure ($E > 0.1$ MeV) correspondingly increased from 8.5×10^{23} n/cm² to the present 9.9×10^{23} n/cm².

2.3 STRATEGIES TO REDUCE FAST FLUENCE AND BURNUP

Table 2.2 shows two sets of peak fast fluence and burnup. The first set corresponds to the equilibrium cycle. These are obtained by adding the burnup and fluence attained during the equilibrium cycle by the thirty-four radial subzones in the first axial layer and applying the appropriate axial peak to average factors.

FISSION PRODUCT BUILDUP
REFERENCE FMSR, HELIUM COOLED
LAYER 1 - SOLID. LAYER 2 - DASHED

Figure 2.6.

An assumption inherent in the above procedure is that a fresh fuel subassembly fed into subzone 1 will reside in various radial positions in the reactor for thirty-four cycles before being discharged. However, it is possible to discharge a subzone at any desired time before the thirty-four cycles have expired. The burnup and fluence damage at the time of discharge will be correspondingly lower. This scheme can be used to match the peak fast fluence and burnup at the time of discharge to predetermined design limits set by material properties of clad and fuel. For example, if it is known that the clad can perform satisfactorily only up to a fast fluence of $6 \times 10^{23} \text{ n/cm}^2$ (as opposed to the equilibrium cycle fluence of $9.9 \times 10^{23} \text{ n/cm}^2$), it would be possible to derive a scheme in which fuel subassemblies are discharged prior to reaching this limit. Such early discharge of a subzone would leave positions in the reactor which would have to be filled by a new subzone of equivalent reactivity worth brought in from outside the reactor.

This new subzone could either be fresh partially enriched uranium or fuel subzones discharged from the startup reactor before reaching the preset limits of burnup and fluence. In either case, an economic penalty would have to be paid in the form of increased uranium resource requirements. As will be shown in Section 3, the natural uranium required over the 30-year life of a FMSR is lower by a factor of approximately four compared to a LWR with a once-through fuel cycle operating to 30,000 MWD/T. This resource advantage factor will be partially reduced if the above scheme is used to reduce the fuel discharge burnup and fast fluence damage.

The second set of fluence and burnup values in Table 2.2 are calculated for a specific fuel management plan covering the 30-year life of the reactor. The reactor starts with fresh enriched uranium fuel. The fuel management strategy

used in this case is designed to reduce the fuel exposure in the fast core by five cycles, i.e., by approximately 25%. It can be seen from Figure 2.6 that the bulk of the fuel burnup, and hence fluence damage, occurs during the twenty cycles of residence in the fast region of the reactor. Hence, reducing the exposure period in the fast region by five cycles would mean a reduction of the burnup and fluence by approximately 25%. The fuel management scheme is described below. The actual reactor physics calculations employing the scheme have not yet been completed, but preliminary calculations indicate that successful employment of this fuel management scheme may be feasible.

Figure 2.7 is a schematic representation of the fuel management scheme which has been designed to reduce the peak discharge fluence to about 7.5×10^{23} n/cm² ($E > 0.1$ MeV) from the equilibrium cycle value of 9.9×10^{23} n/cm². In Figure 2.7, the nine subzones of the shuffle Zone 6 (see Figure 2.3) used in the fuel shuffling strategy are explicitly shown. At the beginning of the first cycle, the reactor contains 34 subzones of fresh fuel containing enriched or natural uranium. At the end of the first cycle, one of the subzones of shuffle Zone 6 is discharged from the reactor and replaced by a subzone from shuffle Zone 5. This transaction is indicated by an x mark in Figure 2.7. The discharged fuel subzone goes into external storage and can be reused in the reactor for fifteen more cycles before reaching the limiting value of 7.5×10^{23} n/cm² ($E > 0.1$ MeV). This subzone in storage is marked R15. Similarly, the fuel subzone discharged at the end of the second cycle R14 can be reused for fourteen more cycles, and so on. After cycle 15, all subzones discharged from the reactor have experienced their maximum allowable exposure. Also, at the end of the sixteenth cycle, six subzones which have either reached or are close to reaching the fluence limit are discharged from the reactor. One of them is replaced by the subzone

E.O.C. Number	Shuffle Zone 6 of the FMSR									External Storage	
	Subzones										
	26	27	28	29	30	31	32	33	34		
1									x	R15	
2								x		R14	
3							x			R13	
4						x				R12	
5									x	R11	
6								x		R10	
7							x			R 9	
8						x				R 8	
9									x	R 7	
10								x		R 6	
11							x			R 5	
12						x				R 4	
13									x	R 3	
14								x		R 2	
15							x			R 1	
16	R15	R14	R13	R12	R11	x				6(R0)	
17									x	R0	
18								x		R0	
19							x			R0	
20						x				R0	
21									x	R0	
22								x		R0	
23							x			R0	
24						x				R0	
25									x	R0	
26								x		R0	
27					R10		x			2(R0)	
28				R 9		x				2(R0)	
29			R 8						x	2(R0)	
30		R 7						x		2(R0)	
31	R 6						x			2(R0)	
32						x				R0	
33									x	R0	
34								x		R0	
35							x			R0	
36						x				R0	
37					END OF LIFE						

Figure 2.7. Schematic Representation of a Fuel Management Scheme Over 30-Year Life of the FMSR

coming in from shuffle Zone 5 (marked x). The other five subzones are replaced by R15, R14, R13, R12 and R11, respectively, from the external storage. R11 will be replaced after 11 more cycles, i.e., at the end of the 26th cycle, by R10. Similarly, R12 is replaced by R9 at the end of the 27th cycle, and so on. This scheme ensures that no subzone is exposed for more than 16 cycles in the fast region alone. Also, no subzone is exposed for more than 15 cycles in the fast region if, in addition, it has received exposure in the moderated region. The effect is to reduce the peak fluence and burnup by approximately 25%.

It should be noted that this fuel management scheme is only one of many possible ways to reduce the discharge fluence damage. The peak discharge fluence can be reduced further, if needed, by further reducing the number of cycles of exposure in the fast region. However, in order to do this, fresh subzones of enriched uranium containing approximately 11.1% U-235 would have to be introduced into the reactor at certain times, in addition to recycling the fuel subzones from the first reactor. To a first order of approximation, the peak discharge fluence can be reduced to about $6.5 \times 10^{23} \text{ n/cm}^2$ if five subzones (i.e., 60 subassemblies) of 11.1% enriched uranium are introduced midway through the life of the reactor. Similarly, the discharge fluence can be reduced to about $6.0 \times 10^{23} \text{ n/cm}^2$ with ten subzones of enriched uranium and to about $5.5 \times 10^{23} \text{ n/cm}^2$ with 20 subzones of enriched uranium being available. This would require a substantial increase in uranium requirements. As an alternative to enriched uranium, subzones containing recycled plutonium could also be used in the reactor; this would significantly improve the FMSR resource advantage.

From the above discussion, it can be deduced that design limits on the peak clad fluence and fuel burnup can be accommodated, but at the expense of reducing the uranium resource advantage of the FMSR over competing reactor concepts. If,

on the other hand, recycling of plutonium on a limited scale is allowed, then the attractiveness of the FMSR from the point of view of resource utilization will be greatly enhanced. A detailed economic analysis of the FMSR should take these factors into account, i.e., design limits on fast fluence exposure and burnup, resource utilization and amount of reprocessing required (if allowed).

2.4 ALTERNATE DESIGN STRATEGIES

Many alternatives to the Reference Design are possible and a few of the more promising alternatives are being pursued. It should be pointed out, however, that design strategies are continuously being evolved and the options described in the report may be modified considerably for future analysis. A design optimized from the point of view of reactor physics may not provide optimal thermal-hydraulic performance and vice versa. Trade-off studies and parametric studies may lead to improved designs and fuel management schemes.

Various ways of controlling the reactor are available, e.g., fueled control rods, poison control rods, moderator control rods or a combination of the above. The method of control adopted will have a significant impact on various aspects of reactor design, operation, safety and economics. These factors should be studied before the most promising design strategy is selected.

2.5 CENTRALLY-MODERATED FMSR [(CM)FMSR]

The amount of fast fluence damage sustained per unit energy extracted is lower in the moderated region, by about a factor of two, compared to the fast region. The study of the (CM)FMSR was undertaken to see if the overall fluence damage can be reduced by shifting more power into the moderated region.

Figure 2.8 shows a schematic layout of the (CM)FMSR. Compared to the Reference Design, the amount of moderator in the outer core region has been greatly

Figure 2.8. A Schematic Layout of the Centrally-Moderated FMSR

reduced and may even be eliminated. This brings the fuel in the outer region closer to the fast region, thus increasing the flux level and also the rate of plutonium buildup in this region. In the central moderated region, each moderator subassembly is surrounded by fuel subassemblies.

Fresh fuel is fed into the outer region, where it resides for many cycles before being moved into the central moderated region. At the time of the move, the amount of bred plutonium is 4.5%. The fission product buildup, however, is still relatively low; thus, the loss of neutrons to absorptions in fission products in the moderated spectrum is low. After residence of several cycles in the moderated region, the fuel is moved into the fast region where it remains until discharged from the reactor.

The (CM)FMSR succeeds only modestly in reducing the overall equilibrium cycle fast fluence damage. Replacing fuel subassemblies by moderator subassemblies in the high reactivity worth central region leads to a loss of reactivity. To compensate for this loss, longer cycle lengths need to be adopted. This increases the equilibrium cycle peak fluence to a value nearly equal to that of the reference case.

Preliminary calculations indicate that the (CM)FMSR has some attractive features:

1. The reactivity swing over the equilibrium cycle is about 40% lower than for the reference case. This would make control of the reactor easier.
2. It appears from calculations so far, that the reactivity swing over a cycle can be managed by means of the central moderator subassemblies. The withdrawal of the moderator subassemblies from the reactor introduces a positive reactivity and vice versa. If practical, this would avoid the need for fueled control rods. Alternatively, a combination

of moderator, fueled and poison rods can be devised to enhance the safety and maximize the neutron economy of the reactor.

3. In the case of the sodium-cooled version, the presence of moderator would reduce the positive sodium void reactivity at the center of the reactor.

Reactor physics calculations for the (CM)FMSR are discussed in more detail in Section 3.

2.6 FMSR WITH LIMITED FUEL REPROCESSING

A design study was initiated to evaluate the potential of a fallback FMSR design which could operate at lower fuel and material irradiation limits. The objective was to determine the performance of an FMSR operating under very limited and controlled reprocessing. Since this work is still at a very preliminary stage, only a brief review of the progress to date will be given.

The basic principle of the FMSR with restricted fuel reprocessing, the (REP)FMSR, again stems from the employment of metal fuel and moderator. The moderator, however, is distributed throughout the entire core. There are two zones of enrichment. The presence of the moderator allows the use of relatively low fissile enrichment. The conversion ratio of the core zones, with the moderator present, is adjusted to be sufficient so that the internal generation of fissile material more than compensates for reactivity losses due to fission product generation at the beginning of the fuel cycle. Eventually, the reactivity loss due to fission product generation will exceed the reactivity gain due to fissile generation and the end of the fuel cycle will be reached. This simple description of the fuel cycle must be augmented by the need to preserve the desired radial power distribution by selective use of control rods over the fuel cycle.

The preliminary design work has shown that the (REP)FMSR can operate for the full plant lifetime on two core loadings. Table 2.3 shows the variation of reactivity of a core over 4400 full power days, which is equivalent to approximately 16 years of operation at 75% duty factor. The table also indicates the degree to which the moderator reduces the fluence damage per fission; the peak fluence damage to the steel at the end of this time is surprisingly low. The peak burnup is less than that obtained in current metal fuel irradiation in EBR-II. Preliminary estimates of the fuel cycle cost for this concept are 3.4 mills/kwhr, for Pu-239 at \$27/gram and oxide-fueled LMFBFR fabrication, etc. charges.

Due to the softness of the neutron energy spectrum, the Doppler feedback coefficient is -0.027. The sodium void coefficient must be evaluated with great care for a design of this type, and it is quite sensitive to the level of fuel burnup. Very preliminary 2D diffusion calculations using 50 groups indicate that full core voiding would add about only 30¢ at the beginning of the fuel cycle and about \$5 for the worst case, which occurs at the end of the 16-year fuel cycle.

The (REP)FMSR appears to be a concept of exceptional potential, quite beyond its immediate design objective of offering antiproliferation advantages. The fuel cycle cost appears to be unusually low. The potentially very long fuel cycle should add further economic interest in the concept. The attractive safety coefficients in a design which does not have problems associated with serious local power gradients and variations in power density over the fuel cycle, coupled to the absence of long-term power tilting, justify further efforts in the areas of design evaluation and development.

Table 2.3
Operating Data of the (REP)FMSR

No. Full Power Days	k _{eff}	Conv. Ratio Zone I Zone II	Power Fractions ZI ZII RB AB	Peak Burnup % HM	Peak Fluence *10 ²³
0	1.012	1.283 1.170	.563 .412 .008 .017	0.0	0.0
1100	1.009	1.169 1.116	.582 .383 .010 .025	3.12	0.80
2200	1.008	1.093 1.076	.598 .359 .011 .032	6.30	1.59
3300	1.004	1.040 1.045	.608 .340 .012 .040	9.50	2.37
4400	0.998	-----	.608 .330 .014 .048	12.76	3.15

3. CENTRALLY-MODERATED FMSR [(CM)FMSR]

3.1 PRINCIPLES OF OPERATION

A brief description of the (CM)FMSR was given in Section 2.3. Both sodium-cooled and helium-cooled versions are presently being studied. Results obtained so far indicate that while there are some important differences between them, the overall characteristics of the sodium- and helium-cooled versions are quite similar.

The fuel volume fraction for the sodium-cooled version is 45% as compared to 39% for the helium-cooled version. This, together with the differences in the moderating characteristics of the two coolants, leads to a shorter neutron diffusion length in the sodium-cooled version. Hence, the problem of power peaking, axially and radially, is more pronounced in the sodium-cooled FMSR. However, the superior heat transfer characteristic of sodium effectively overcomes this disadvantage.

Figures 3.1 and 3.2 show the schematic layout in the x-y plane of the sodium- and helium-cooled versions, respectively. The centrally-moderated regions are modified to suit each version. The sodium-cooled FMSR contains thirty-three fuel subzones, each containing twelve hexagonal fuel subassemblies. The helium-cooled FMSR has thirty-four subzones. The subassembly at the very center of the sodium-cooled FMSR contains fuel to help raise the power level in the surrounding subassemblies. Since the central fuel subassembly is not a part of any subzone, it must be treated separately. In the helium-cooled FMSR, the central subassembly is a moderator assembly. Another difference stems from the improved radial power flattening achieved by relocating some of the central moderating subassemblies radially outward.

- Moderator
- Reflector
- Fuel, Outer Region
- Fuel, Central Region
- Fuel, Fast Region

Figure 3.1. Schematic View of the Sodium-Cooled, Centrally-Moderated FMSR in the x-y Plane

Figure 3.2. Schematic View of the Helium-Cooled, Centrally-Moderated FMSR in the x-y Plane

The (CM)FMSR can be broadly divided into three regions, i.e., the outer region, the central moderated region and the fast region. Fresh fuel subassemblies are first fed into the outer region where they build up sufficient levels of plutonium to perform effectively when they are moved to the central moderated region. At the time of transfer from the outer region into the central moderated region, the amount of fission products in the fuel will be relatively low. This reduces neutron losses by capture in the fission products which exhibit relatively large cross sections in softer spectra. After many cycles of residence in the central moderated region, the fuel is moved to the fast region. The fast region provides most of the power and reactivity of the reactor. At the end of its residence period in the fast region, the fuel subzones are discharged from the reactor. During its residence in each of the three regions, the fuel may be shuffled once or twice within the region depending on the fuel management scheme employed.

In the moderated region, the amount of fast-fluence exposure ($E > 0.1$ MeV) per unit energy produced is significantly lower than in the fast region. One of the incentives for studying the centrally-moderated FMSR was to determine whether the overall fluence could be reduced significantly by increasing the power produced in the moderated region. The (CM)FMSR succeeds only modestly in achieving this objective because introduction of moderator subassemblies in place of power-producing fuel subassemblies in the high reactivity worth central region leads to a loss of reactivity. To compensate for this loss, a longer burnup cycle is required, bringing the fluence damage at the time of discharge almost back to the level corresponding to the reference FMSR. (A longer fuel cycle length increases the average plutonium concentration of those subassemblies

which have not yet achieved the equilibrium concentrations, and thereby adds reactivity.)

It is found that compared to the reference FMSR, the reactivity swing during a burnup cycle of the centrally-moderated FMSR is lower by about 40%. This reduces the demand on the control system. An important potential advantage of the (CM)FMSR is the possibility of employing some of the moderator subassemblies in the central region to control the reactor. Withdrawal of the moderator from the core leads to a hardening of the neutron spectrum in the central region. This effect, along with the changes in the shapes of neutron flux and adjoint function, leads to an increase in the reactivity. Inserting the moderator into the reactor reduces the reactivity. Preliminary calculations in R-Z geometry indicate that the reactivity change associated with the movement of the moderator is more than sufficient to control the reactivity swing over a burnup cycle.

However, more accurate and detailed calculations are needed before this can be verified; if feasible, the need for fueled control rods would be eliminated. The question of choosing an appropriate control system for the FMSR requires more intensive study.

In the sodium-cooled version of the (CM)FMSR, presence of moderator subassemblies in the central region would lead to a reduction of the local sodium void reactivity effect.

3.2 (CM)FMSR REACTOR PHYSICS CALCULATIONS

Reactor physics calculations were performed using 50-group cross-section sets obtained from LIB-IV data and the 2DB diffusion theory code.⁽⁵⁾ Burnup calculations were performed in R-Z geometry to obtain the equilibrium cycle. Calculations were also performed in hexagonal geometry in order to provide

comparisons with R-Z geometry calculations and more realistically represent the reactor.

The 2DB code was modified to handle multiple fission product isotopes with appropriate yields. In the present calculations, four fission products were used, namely, Sm-149 and three lumped fission products of Pu-239, i.e., Rapidly Saturating (RSFP), Slowly Saturating (SSFP) and Non-Saturating (NSFP) Fission Products. Fission product isotopes, in general, transmute to other isotopes of lower absorption cross sections upon neutron absorption. In the present calculations, these transmutations were approximated by the following scheme. RSFP were assumed to transmute to SSFP, SSFP to NSFP, and Sm-149 was assumed to transmute to NSFP. Only the fission products of Pu-239 were employed, even though U-238 fissions constituted about 25% of all fissions.

Figure 3.3 shows the 50-group neutron flux spectra in the sodium-cooled (CM)FMSR. Neutron spectra typical of the moderated region and the fast region are shown. It can be seen that even though the spectrum in the moderated region is softer than in the fast region, it is still "hard", in that a considerable fraction of neutrons is above 1 keV. The flux depression due to the sodium resonance at 2.85 keV can be clearly seen. Figure 3.4 shows the neutron spectra for the helium-cooled (CM)FMSR.

Low energy neutrons entering the fuel from moderator subassemblies can lead to hot spots at the interface of fuel and moderator. This problem can be especially acute in the central moderated region where the flux level is relatively high. Since "true" hot spot factors are not obtainable from R-Z calculations where the fuel and moderator subzones are represented by radial rings, 2DB hexagonal geometry calculations were performed; twenty-four mesh points per hexagonal subassembly were employed. Equilibrium cycle nuclide number densities of the

NEUTRON ENERGY SPECTRA
CENTRALLY-MODERATED FMSR, SODIUM-COOLED

Figure 3.3.

NEUTRON ENERGY SPECTRA
CENTRALLY MODERATED FMSR, HELIUM-COOLED

Figure 3.4.

first axial layer obtained from the R-Z calculations were used in the hexagonal calculations. The 2DB code was modified to treat a 30° sector of the reactor. Since there are twelve fuel subassemblies per subzone, the reactor exhibits a 30 degree symmetry. In a 30 degree sector, each fuel subzone is represented by one hexagon. Calculations performed so far indicate that the power peaks at the moderator-fuel interface can be reduced by reducing the moderator number density. The moderator density was adjusted wherever necessary to reduce the power peaks to acceptable levels. More detailed calculations using transport theory and a more detailed geometric representation of fuel and moderator are needed to make a more accurate assessment of power peaking problems.

3.3 RESULTS OF REACTOR PHYSICS CALCULATIONS

3.3.1 Sodium-Cooled (CM)FMSR

Figure 3.5 shows the details of the reactor layout in the x-y plane. A 60 degree sector of the reactor is shown. Moderator subassemblies have been marked with the letter M. The thirty-three fuel subzones are marked by their respective numbers. In the 60 degree sector, each fuel subzone contains two fuel subassemblies. The central subassembly also contains fuel, but is not a part of any fuel subzone. The nuclide number densities in the central subassembly are assumed to be the same as those in subzone 33.

In the R-Z geometry representation shown in Figure 3.6, the fuel subzones are represented by rings of fuel. Moderator subassemblies are grouped together and represented by moderator rings. The R-Z representation of the reactor necessarily leads to geometrical approximations which can have an effect on reactor physics calculations. Hexagonal calculations can treat the radial geometry more exactly, but cannot treat the axial direction. Hence, burnup calculations in

Figure 3.5. Reactor Layout in the x-y Plane for the Sodium-Cooled, Centrally-Moderated FMSR

Figure 3.6. Schematic R-Z Representation of the Sodium-Cooled, Centrally-Moderated FMSR

hexagonal geometry do not provide as much information about the whole reactor as the R-Z geometry calculations. More accurate results will require 3D calculations or coupled R-Z - hexagonal calculations.

Table 3.1 compares the sodium-cooled (CM)FMSR and the Clinch River Breeder Reactor design parameters. In order to take maximum advantage of current fast reactor development, the FMSR design parameters were selected to be as reasonably close as possible to the CRBR design parameters. The larger fuel pin diameter shown for the FMSR reflects the advantages to be gained from the higher thermal conductivity of metal fuel of the FMSR compared to the oxide fuel of the CRBR. The fuel volume fraction is 0.45. The smear density of 75% of the theoretical value along with sodium bonding is an attempt to follow the successful design principle of the metallic fuel irradiated to high burnup in the EBR-II.

The engineering design of the moderator subassemblies has not yet been finalized. This can be properly addressed only after heat generation rates in the moderator subassemblies from gamma heating and neutron energy deposition have been determined. For the present analysis, it is assumed that the moderator subassemblies have beryllium, steel and coolant volume fractions of 0.8, 0.1 and 0.1, respectively. To avoid hot spots in the fuel, the beryllium density, where necessary, was reduced in the reactor physics calculations.

Figure 3.7 shows the radial total flux distribution in the fuel subzones of the sodium-cooled (CM)FMSR corresponding to the reactor midplane at the beginning of equilibrium cycle. It is seen from Figure 3.7 that the total flux is relatively flat in the central moderated region and the fast region, but drops off rapidly in the outer or breeder region.

Figure 3.8 shows the Pu-239 buildup in the fuel subzones in the first and second axial layers of the reactor. Most of the buildup occurs in the outer and

Table 3.1
Thermohydraulic Parameters, FMSR and CRBR

<u>REACTOR CORE</u>	<u>FMSR (REFERENCE CASE)</u>	<u>CRBR (300 MW(e))</u>
Thermal Power, MW	2780	975
Inlet Temperature, °C	380	388
Outlet Temperature, °C	530	535
Total Pressure Drop, MPa	0.76	0.77
Average Flow, kg/hr/cm ²	2.36 x 10 ³	1.72 x 10 ³
<u>FUEL ASSEMBLIES</u>		
Rods Per Assembly	271	217
Volume Fraction		
Fuel	0.45	0.325
Coolant	0.35	0.419
Steel	0.20	0.234
Rod Outside Diameter, cm	0.861	0.5842
Fuel Diameter, cm	0.774 (metal)	0.4915 (oxide)
Clad Thickness, cm	0.04310	0.0381
Wire-Wrap O.D., cm	0.1422	0.142
Pitch	30.42	30.23
Rod Triangular Pitch, cm	1.0182	0.7308
P/D Ratio	1.183	1.251
Rod Length, cm	270	290.6
Fuel-Clad Gap, cm	0	0.0082
Duct Flat Distance, cm	18.692	11.011
Duct Wall Thickness	0.254	0.3048

RADIAL DISTRIBUTION OF TOTAL FLUX CENTRALLY-MODERATED FMSR, SODIUM-COOLED

Figure 3.7.

PU-239 BUILDUP
CENTRALLY MODERATED FMSR, SODIUM COOLED
LAYER 1 - SOLID LAYER 2 - DASHED

Figure 3.8.

central moderated regions. At the time of transfer from the outer region to the central moderated region, the average Pu-239 concentration is about 4.3% in the first axial layer and about 3% in the second axial layer. The plutonium concentration continues to build up in the moderated region and attains the peak value of about 7.9% in the fast region.

Figure 3.9 shows the average power density in the central axial layer of the fuel zones at the beginning and end of the equilibrium cycle. It is seen that the power densities in the central moderated and outer regions are higher at the end of the cycle than at the beginning of the cycle. This is principally due to an increase in the fissile content of these regions during the burnup cycle. For the same reason, the power levels in all the subzones of the second axial layer and the axial blanket are higher at the end of the cycle than at the beginning. On the other hand, as seen from Figure 3.8, in the fast region of the first axial layer, the plutonium concentration is approximately at a saturation level. Since the total power of the reactor is maintained constant, the power densities in this region are lower at the end of the cycle compared to the beginning of the cycle. Control rods and their effect were not considered.

Figure 3.10 shows the buildup of fission product concentrations in the first and second axial layers of the reactor. It is seen that most of the fission product buildup, or fuel burnup, takes place in the fast region.

Table 3.2 summarizes the results of the fuel cycle calculations. During an equilibrium burnup cycle of 260 full power days, the reactivity increases by 1.27%. The reactor is critical at the end of the cycle. The relatively high breeding ratio of 1.66 is consistent with the use of metal fuel and the high fuel volume fraction used in the reactor. The equilibrium cycle peak discharge burnup is about 237,000 MWD/T.H.M. However, as explained in Section 2.2, if the

RADIAL POWER DISTRIBUTION
CENTRALLY MODERATED FMSR, SODIUM COOLED
B.O.E.C. - SOLID AND E.O.E.C. - DASHED

Figure 3.9.

FISSION PRODUCT BUILDUP
CENTRALLY MODERATED FMSR, SODIUM COOLED
LAYER 1 - SOLID LAYER 2 - DASHED

Figure 3.10.

Table 3.2

FMSR PerformanceSodium-Cooled, Centrally-Moderated FMSR

Number of Fuel Subassemblies	397
Number of Subassemblies Shuffled Per Cycle	84
Cycle Duration (Full Power Days)	260
Reactivity Swing Over Cycle, %	+1.27
Breeding Ratio (BOEC)	1.66
(EOEC)	1.58
Fuel Inventory	
Reactor, BOEC, kg Pu	8930
Discharge/Yr, kg Pu	530
Core Discharge Enrichment (% Pu-239)	7.6
Peak Burnup (MWD/T)	
30-Yr, Fuel Reintroduction	~184,000
*Equilibrium Fuel Cycle	237,000
Peak Fluence (E> 0.1 MeV)	
30-Yr, Fuel Reintroduction	~7.3 x 10 ²³
*Equilibrium Fuel Cycle	9.9 x 10 ²³

*This number applies only if the fuel stays in the reactor for 33 burnup cycles.

subzones are discharged before the full thirty-three cycles of residence in the reactor, the peak discharge burnups would be correspondingly lower. If a fuel management scheme similar to that of Figure 2.7 is adopted over the 30-year life of the reactor, the peak discharge burnup can be reduced to approximately 184,000 MWD/T.H.M. At the same time, the peak fluence exposure at the time of discharge would be reduced from around $9.9 \times 10^{23} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) to around $7.3 \times 10^{23} \text{ n/cm}^2$.

Figure 3.11 shows the power densities, in selected fuel subzones, obtained from a calculation in the hexagonal geometry. The main purpose of the calculation was to obtain the radial power peaking factors within subassemblies for use in the thermal-hydraulic calculations. In order to more accurately estimate the hot spots at the fuel-moderator interfaces, twenty-four mesh points per hexagon were used. Preliminary calculations indicated that in order to reduce the unacceptably high power density of hot spots in the central moderated region, the beryllium density in this region would have to be reduced. In the present calculations, both in the R-Z and hexagonal geometry, the beryllium number density in the central moderated region has been reduced to half of the theoretical value. The isotope concentrations in the various fuel zones used in the hexagonal calculations were taken from equilibrium cycle R-Z geometry calculations. The number densities correspond to those in the first axial layer of the R-Z model. A group- and zone-independent axial buckling was used. The value of the axial buckling was chosen so that the k_{eff} values calculated by the R-Z and the hexagonal model were equal. It is seen from Figure 3.11, that in the central moderated region and the fast region, the power density within a given subassembly is relatively uniform, but in the outer region, because of the rapid decrease of flux as we move radially outwards, the power density within a subassembly exhibits a high peak/average ratio. However, since the average power density in the

Figure 3.11. Power Densities (MWT/L) in Typical Subzones, Sodium-Cooled, Centrally Moderated FMSR

outer region is relatively low, the problems posed by the high peak/average power ratio are found to be manageable. This is discussed in greater detail in the sections dealing with thermal-hydraulics calculations.

3.3.2 Reactivity Coefficients

The Doppler reactivity coefficient was calculated using effective multiplication factors at two temperatures. The Doppler coefficient is defined as

$$\alpha_D = T \frac{dK}{dT} \quad (T = \text{temperature in } ^\circ\text{K})$$

After integrating and rearranging,

$$\alpha_D = \frac{K(T_1) - K(T_2)}{\ln(T_1/T_2)}$$

where $K(T_1)$ and $K(T_2)$ are effective multiplication factors at temperatures T_1 and T_2 .

The calculated Doppler coefficient is $\alpha_D = -0.0062$ over the range 300°K to 975°K.

A preliminary value of the sodium void reactivity for complete voiding of sodium from the reactor is 3.16%. The calculation was performed in R-Z geometry where the heterogeneous geometry of the moderated central region cannot be adequately presented. Calculations in 3D geometry or coupled R-Z and x-y calculations are needed for more accurate evaluations of the sodium void reactivity.

Sodium void reactivity can be reduced by increasing the size of the moderated central region and by reducing the height of the reactor. Future calculations will incorporate the above in order to reduce the sodium void effect.

3.3.3 Helium-Cooled (CM)FMSR

Figure 3.12 represents the reactor layout in the x-y plane. The thirty-four fuel subzones are marked by their respective numbers. The reactor model in R-Z geometry used in burnup calculations is shown in Figure 3.13.

Table 3.3 compares the design parameters of the helium-cooled (CM)FMSR to those of the GCFR Demo plant. The FMSR uses a fuel volume fraction of 0.39 with a smear density of 75% of the theoretical value.

Figure 3.14 shows the radial distribution of the total flux in the fuel subzones of the helium-cooled (CM)FMSR. It is seen that the total flux again peaks in the fast region.

Figure 3.15 shows the Pu-239 buildup in the fuel subzones in the first and second axial layers. As in the sodium-cooled case, most of the plutonium buildup takes place in the outer and central moderated regions. At the time of transfer from the outer region to the central moderated region, the average Pu-239 concentration is about 5% in the first axial layer and about 3.4% in the second axial layer. The peak Pu-239 concentration of about 7.6% is reached in the fast region. The strong buildup of Pu-239 in the outer regions of the core, in position of low reactivity worth, helps to reduce the reactivity gain over the fuel cycle.

Figure 3.16 shows the average power density in the central axial layer of the fuel zones at the beginning and the end of the equilibrium cycle. The characteristics are similar to those of the sodium-cooled version. Figure 3.17 shows the buildup of fission product concentrations in the first and second axial layers of the reactor.

Table 3.4 summarizes the results of fuel burnup calculations for the helium-cooled, centrally-moderated FMSR. The equilibrium cycle is 250 full power

Figure 3.12. Reactor Layout in x-y Geometry for the Helium-Cooled, Centrally-Moderated FMSR

Figure 3.13. Schematic R-Z Representation of the Helium-Cooled, Centrally-Moderated FMSR

Table 3.3
Thermohydraulic Parameters, Helium-Cooled (CM) FMSR
and Gas-Cooled Demo

	FMSR (REFERENCE CASE)	DEMO
<u>REACTOR CORE</u>		
Thermal Power, MW	2780	826
Inlet Temperature, °C	300	312
Outlet Temperature, °C	512	540
Total Pressure Drop, psi	96	40
<u>FUEL ASSEMBLIES</u>		
Rods Per Assembly	331	271
Volume Fraction		
Fuel	0.39	0.28
Coolant	0.45	0.56
Steel	0.16	0.16
Rod Outside Diameter, cm	0.807	0.740
Fuel Diameter, cm	0.721	0.633
Clad Thickness, cm	0.0432	0.0483
Rod Triangular Pitch, cm	1.027	1.00
P/D Ratio	1.273	1.35
Rod Length, cm	270	300
Fuel-Clad Gap, cm	0	0.005
Duct Flat Distance, cm	18.692	16.55
Duct Wall Thickness	0.254	0.254

RADIAL DISTRIBUTION OF TOTAL FLUX CENTRALLY MODERATED FMSR, HELIUM COOLED

Figure 3.14. Radial Distribution of Total Flux, Centrally-Moderated FMSR, Helium Cooled

PU-239 BUILDUP
CENTRALLY MODERATED FMSR, HELIUM COOLED
LAYER 1 - SOLID LAYER 2 - DASHED

Figure 3.15.

RADIAL POWER DISTRIBUTION
CENTRALLY MODERATED FMSR, HELIUM COOLED
B.O.E.C. - SOLID AND E.O.E.C. - DASHED

Figure 3-16.

FISSION PRODUCT BUILDUP
CENTRALLY MODERATED FMSR, HELIUM COOLED
LAYER 1 - SOLID LAYER 2 - DASHED

Figure 3.17.

Table 3.4

FMSR PerformanceHelium-Cooled, Centrally-Moderated FMSR

Number of Fuel Subassemblies	408
Number of Subassemblies Shuffled Per Cycle	84
Cycle Duration (Full Power Days)	250
Reactivity Swing Over Cycle, %	+1.1
Breeding Ratio (BOEC) (EOEC)	1.61 1.54
Fuel Inventory	
Reactor, BOEC, kg Pu	8394
Discharge/Yr, kg Pu	493
Core Discharge Enrichment (% Pu-239)	7.3
Peak Burnup (MWD/T)	
30-Yr, Fuel Reintroduction	~186,000
*Equilibrium Fuel Cycle	238,000
Peak Fluence (E > 0.1 MeV)	
30-Yr, Fuel Reintroduction	~7.2 x 10 ²³
*Equilibrium Fuel Cycle	9.7 x 10 ²³

*This number applies only if the fuel stays in the reactor for 34 burnup cycles.

days in length. During this period, the reactivity increases by 1.1%, the reactor being critical at the end of the cycle. The equilibrium cycle discharge burnup is about 238,000 MWD/T.H.M. However, this may be reduced to about 186,000 MWD/T by adoption of a fuel management scheme similar to the one outlined in Figure 2.7. This would also reduce the peak discharge fast fluence ($E > 0.1$ MeV) from 9.7×10^{23} n/cm² to about 7.2×10^{23} n/cm².

The calculated Doppler reactivity coefficient is $\alpha_D = -0.0061$ over the range 300°K to 975°K.

3.3.4 Fuel Shuffling Strategy

Figure 3.18 is a schematic representation of the fuel management strategy applicable to the helium-cooled (CM)FMSR. A similar fuel management strategy may be used for the sodium-cooled version. The 34 fuel subzones are grouped under seven broad zones. In Figure 3.18, the numbers of the subzones grouped under each broad zone are marked. At the end of each burnup cycle, one fresh fuel subzone is fed into Zone I. One subzone from Zone I is moved to Zone II, one from Zone II to Zone III and so on. One subzone from Zone VII is discharged. The residence time (in cycles) of a fuel subassembly in any zone is equal to the number of subzones in that zone. For example, a fresh fuel subassembly fed into Zone I resides in that zone without being shuffled for seven cycles. Then it is moved to Zone II where it resides for four cycles. With this fuel management scheme, only 84 fuel subassemblies (seven subzones) are moved at the end of a burnup cycle.

The fuel shuffling strategy and the number of subzones in each zone were arrived at after a careful study of their impact on the variations of radial flux and power shapes. For example, a fuel subassembly moved from Zone III to

Figure 3.18. Fuel Shuffling Strategy for the Helium-Cooled, Centrally-Moderated FMSR

Zone IV resides in a fixed position in Zone IV for three cycles. During this time it continues to build up plutonium rapidly. This leads to an increase in the power density in the subassembly. From the point of view of thermal-hydraulics, it is advantageous to keep the power swing as small as possible. The problem is even more pronounced near the center of the reactor. A too high an increase in the fissile concentration in subassemblies near the center may lead to flux peaking at the center of the reactor. This could lead to unacceptably high power densities in these subassemblies. Hence, in Zones IV and V, the residence times are relatively short.

Figure 3.19 shows the changes in the plutonium concentrations in the first axial layer of the various fuel subzones for the fuel shuffling strategy described above. Only Zones IV, V, VI and VII are shown. The plutonium concentration increases rapidly during the three cycles of residence in Zone IV. The effect of the differences in the local flux levels and energy spectra on the rate of plutonium buildup can be seen clearly. In Zone V, the plutonium buildup is considerably smaller than in Zone IV. In Zones VI and VII, there is a net reduction in the plutonium concentration.

Figure 3.20 shows the changes in the power density in the fuel subzones during their residence periods in Zones IV, V, VI and VII. The variations in the power density are a result of changes in local fissile concentrations and flux levels. The biggest change, an increase of about 25%, takes place during three cycles of residence in a subzone of Zone IV. To accommodate the increases in the power densities, the coolant orifices of subassemblies of Zones IV and V may have to be adjusted during their residence time in these zones.

Figure 3.21 shows the changes of total fission product concentrations as a function of shuffling and residence time. The pattern of fission product buildup

PU-239 CONCENTRATION VARIATION
OVER FUEL RESIDENCE IN SUB-ZONES
CENTRALLY MODERATED FMSR, HE-COOLED
B.O.RES.-SOLID AND E.O.RES.-DASHED

Figure 3.19.

POWER DENSITY VARIATION
OVER FUEL RESIDENCE IN SUB-ZONES,
CENTRALLY MODERATED FMSR, HE-COOLED

Figure 3.20.

FISSION PRODUCT CONCENTRATION VARIATION
OVER FUEL RESIDENCE IN SUB- ZONES
CENTRALLY MODERATED FMSR, HE-COOLED
B.O.RES.-SOLID AND E.O.RES.-DASHED

Figure 3.21.

is clearly evident. The final peak concentrations at the time of discharge are reasonably uniform.

It should be pointed out that the above shuffling scheme is only one of the many possible alternatives. With further study, it may be possible to devise a better scheme.

3.3.5 Control System for the (CM)FMSR

During an equilibrium burnup cycle of 260 full power days, the sodium-cooled (CM)FMSR experiences a reactivity gain of 1.27%. The helium-cooled (CM)FMSR experiences a reactivity gain of 1.1% over a 250 full power days.

Results of preliminary calculations in R-Z geometry suggest that the central moderator subassemblies (30 in the sodium-cooled case and 31 subassemblies in the helium-cooled case) may be used for reactor control. Withdrawing the moderator subassemblies two thirds of the way from the reactor resulted in a reactivity increase of 1.4% in the sodium-cooled FMSR and 2.4% in the helium-cooled FMSR. More detailed calculations need to be performed.

At the beginning of a burnup cycle, the reactor would be critical with the moderator subassemblies in the out position. The reactivity gain during burnup would be compensated by progressively inserting the moderator subassemblies into the reactor. By mixing an appropriate neutron poison with the moderator, the worth of the moderator subassemblies may be increased. Thus, the number of moderator subassemblies needed to control the reactor may be decreased.

Two other options for the control of the FMSR are fueled control rods and conventional neutron poison control rods. The implications of selecting any one of the three systems for the reactor design and safety needs to be studied in greater detail. In practice, a combination of the three options may provide an optimum system for reactor control.

3.4 URANIUM RESOURCE REQUIREMENTS FOR THE (CM)FMSR

Uranium resource requirements for the FMSR can be grouped under three categories: 1) requirement for the enriched uranium startup core, 2) requirement for the natural uranium feed over 30 years, and 3) requirement for enriched uranium-fueled control rods if used.

Since a decision regarding a control system for the (CM)FMSR has not yet been made, the requirement for fueled control rods will not be analyzed. The enriched uranium startup core requires the larger portion of the uranium resource requirement. Natural uranium feed over the life of the reactor accounts for only about 10% of the total uranium requirement. If, however, plutonium can be used as fissile fuel and depleted uranium from enrichment plant tails can be used as the fertile material for the startup reactor, the total uranium requirement will be small and limited to the natural uranium feed.

The startup FMSR core is composed of enriched uranium of varying enrichments in the 160 cm core region. Fuel subzones 1 to 7 and the axial blankets (upper and lower 40 cm of the reactor) contain only natural uranium. An initial estimate of the enrichments in the various subzones was made by inspecting their fissile contents during the equilibrium cycle. The enrichments were then adjusted to obtain a critical reactor.

Uranium requirements for the startup core and for the feed fuel over 30 years are tabulated in Table 3.5. The startup core for the sodium-cooled (CM) FMSR requires 1850 T of natural uranium. It is assumed that the uranium tailings of the enrichment plant contain 0.2% U-235. The helium-cooled version requires 1665 T of natural uranium for the startup reactor. The natural uranium feed fuel requirements are 200 T and 179 T, respectively. It is assumed that the sodium-cooled FMSR goes through 32 burnup cycles and the helium-cooled FMSR

Table 3.5

Uranium Requirements for the Startup Core and Makeup Feed

	<u>Sodium-Cooled (CM)FMSR</u>	<u>Helium-Cooled (CM)FMSR</u>
Total Heavy Metal (kgs)	213,100	189,800
Natural Uranium (Fuel Subzones 1-7 and the Axial Blankets) (kgs)	101,100	89,338
Highest U-235 Enrichment	11.2%	10.7%
Amount of U-235 in Enriched Fuel	9144	8238
(1) Natural Uranium Required for the Startup Reactor (T)	1850	1665
30-Yr Requirement for Natural Uranium Makeup Fuel (T)	200	179

(1) Assuming enrichment plant tails content of 0.2% U-235.

goes through 33 cycles in 30 years of reactor life. It is possible to replace natural uranium feed with depleted uranium, but at the price of slightly higher burnup and fluence damage at the time of discharge. If depleted uranium is used as the feed material, the total uranium requirement for the reactor will be correspondingly smaller.

The total uranium requirement for the FMSR depends on the actual fuel management scheme adopted and the accounting procedure. Table 3.6 summarizes the total natural uranium requirement for the FMSR calculated by different methods. In Case 1 two startup reactors are coupled, i.e., enriched uranium fuel discharged from the first startup reactor before reaching the full burnup is used in a second startup reactor. Hence, the extra uranium requirement for the second startup reactor is quite small. In the present calculation, it has been assumed that subzones 1 to 14 of the second startup reactor contain fresh fuel. The rest of the subzones contain fuel brought in from the first startup reactor. All fuel discharged from the second reactor will have reached the full peak burnup of about 237,000 MWD/T. In Table 3.6, the average uranium requirement for the two coupled startup reactors is given. When they are discharged after full burnup, the fuel subzones which began with enriched uranium fuel still contain significant quantities of U-235. Table 3.6 also shows the approximate credits that can be taken into account for the discharged U-235. For Case 1 the credit is to be averaged for the two coupled reactors.

In Case 2, fuel discharged from the startup reactor before reaching burnup limits is recycled back into the reactor using a scheme similar to the one described in Section 2. The peak burnup at discharge in this case is about 185,000 MWD/T. The uranium requirement for the startup reactor in this case is substantially higher than in Case 1, since a second coupled startup reactor does

Table 3.6

Estimates of Total Uranium Requirement for the (CM) FMSR

	<u>Sodium-Cooled (CM) FMSR</u>	<u>Helium-Cooled (CM) FMSR</u>
<u>Case 1:</u> Two startup reactors are coupled. Peak Burnup 237,500 MWD/T		
Average Startup Reactor (T)	1045	949
30-Yr Makeup Fuel (T)	200	179
Total Requirement (T)	1245	1128
*Average Credit for Discharged U-235 (T)	120	100
*Total Requirement with the Credit (T)	1125	1028
<u>Case 2:</u> 30-Yr Fuel Reintroduction Scheme. Peak Burnup 185,000 MWD/T		
Startup Reactor (T)	1850	1665
30-Yr Makeup Fuel (T)	200	179
Total Requirement (T)	2050	1844
*Credit for Discharged U-235 (T)	400	330
*Total Requirement with the Credit (T)	1650	1514

*Approximate numbers

not enter the picture. Approximate credits for the discharged U-235 are also shown. The credits are higher for Case 2 because of the lower burnup and because they are not averaged for two reactors as in Case 1. The maximum fuel burnup and fast fluence exposure for the steel are lower for Case 2 compared to Case 1. For this, a price is paid in terms of higher uranium requirements for Case 2. Further reduction of fluence damage and peak burnup can be achieved by introducing enriched uranium subassemblies ($\approx 11\%$ U-235) during the 30-year life of the reactor. This again would mean a further increase in the uranium requirement.

It is cautioned that the above numbers should be taken as approximate, especially for the startup requirements for the second reactor for Case 1 and the credits for discharged uranium. Actual reactor physics calculations are under way to obtain more accurate results.

4. THE ALL-FAST FMSR

4.1 INTRODUCTION

A short study was made of an all-fast FMSR to examine the performance of the FMSR when the moderator is completely removed from the Reference FMSR Design. No major changes were expected since the beryllium moderator employed in the Reference FMSR is always in regions of low neutron flux and importance. This study of the significance of the moderator was made against the Reference FMSR rather than the centrally-moderated FMSR because that reactor concept was not yet well enough defined at the start of this comparison. The gain to be anticipated for the all-fast system would derive from the advantages resulting from a somewhat improved breeding ratio and probably a lower fuel burnup level on discharge. The disadvantages which were foreseen were in the direction of a somewhat smaller Doppler feedback coefficient, a somewhat larger sodium void reactivity gain, and higher radial power peaking. The most important practical change would be in the loss of power flattening and correspondingly increased thermal-hydraulic problems.

4.2 ALL-FAST CORE ANALYSIS PROCEDURE

The layout of the core which represented the all-fast FMSR was adapted very simply from that of the Reference FMSR by removing the beryllium moderator from between Zone 6 and Zone 2 as well as from between Zone 2 and Zone 1 of the Reference Design. These changes are shown in Figure 4.1. Graphite was selected for the radial reflector material of the all-fast FMSR because it would help to flatten the radial power distribution in the outermost row of subassemblies. It also avoids the need to treat the radial blanket in the fuel management procedure. Possible problems of local power peaking such as might occur with a more efficient moderator such as beryllium are also avoided.

Figure 4.1. All-Fast FMSR Core Layout Compared to Reference FMSR Layout

There were two principal differences between the Reference Design and the all-fast calculations. One concerned the treatment of fission products and the other concerned the use of sodium rather than helium as the coolant. At the time the Reference Design FMSR calculations were made, only a single "effective" fission product was allowed by the programmed capability of the 2DB fuel cycle code available at Brookhaven. By inter-calibration with results from ORNL calculations for the same FMSR system, this single fission product was adjusted to be 2.7 times as absorptive as that of the non-saturating Pu-239 fission product which is available on the LIB-IV library. By the time of the all-fast FMSR study, the 2DB code was able to use the three types of lumped fission products available in the LIB-IV library, as well as samarium. In spite of small differences, this capability was employed for that study, in order to maximize the consistency and applicability of the all-fast work for future comparisons. The other important difference is that the all-fast system was studied as a sodium-cooled version in order to reflect the growing interest in the sodium-cooled versions of this reactor concept. It was felt that the differences between the helium- and sodium-cooled FMSR had already been examined and were reasonably well understood.

This design study was not exhaustive because it was felt that the combination of a high sodium void coefficient (about 40% higher than for the centrally-moderated FMSR) and a correspondingly small Doppler feedback coefficient would make other FMSR designs much more attractive from a safety and operational point of view. The predicted degraded radial power distribution would also adversely impact the thermal-hydraulics design problem.

A core height of 160 cm was retained even though other studies have shown that a shorter core height may be more optimum in terms of such factors as resource utilization and cost. The volume fractions employed were those

previously optimized between thermal-hydraulic and reactor physics design objectives for the Reference sodium-cooled FMSR. Fuel cycle calculations were initiated using the same fuel shuffle pattern as in the Reference Design; only the equilibrium fuel cycle was studied. Studies relating the first core design and evolution toward the equilibrium fuel cycle were not made because this basic design turned out to be the least attractive of the designs studied to date.

4.3 RESULTS

The more prominent results of this study are shown in tabular form in Table 4.1. As predicted, the differences between the two reactor designs are small. The impact on thermal-hydraulic considerations is shown in Figure 4.2. The differences seen in Table 4.1 for burnup and inventories result in part from differences in fuel volume fractions (0.45 for the sodium-cooled FMSR and of 0.39 for the helium-cooled FMSR). For this reason alone, the burnup per atom for the sodium-cooled FMSR, relative to the corresponding helium-cooled FMSR, is lower; a higher material inventory is required for the sodium-cooled version. The all-fast FMSR breeding ratio is somewhat higher, as expected. The slightly smaller ΔK over the fuel cycle is most likely due to the higher plutonium concentration in the fuel moving to the center of the core in the all-fast design. The greater concentration of the discharged higher plutonium isotopes of the Reference FMSR reflects the degraded effective α (i.e., ratio of capture to fission) of plutonium-239 for fuel near the moderator, and also, the cumulative effects of the longer irradiation time per cycle for the Reference Design.

The important advantage of using moderator at the edge of the FMSR core can be seen from an examination of Figure 4.2. The vertical line in this figure represents the interface at a position corresponding to the beryllium moderator

Table 4.1

All-Fast Performance, Relative to Reference FMSR

	<u>All-Fast FMSR</u>	<u>Reference FMSR</u>
Number of Subassemblies	408	408
Number of Subassemblies Shuffled Per Cycle	72	72
Full Power Days Per Cycle	200	220
Overall Breeding Ratio	1.83	1.67
ΔK Over Cycle, %	+ 2.0	+ 2.1
Peak Fluence, nvt > 0.1 MeV (30-Yr ~70% of this)	9.6×10^{23}	9.9×10^{23}
Peak Burnup, MWD/T (30-Yr ~75% of this)	200,500	212,000
Average Burnup, MWD/T	134,500	163,000
Core Inventory (BOEC, TONNES)		
Pu-239	7.62	5.5
Heavy Metals (Total)	219	189
Discharge/Year, Pu-239	.66	0.5
Average Discharge Enrichment, Pu-239	7.2	7.0
Discharge Pu Composition, % (% 239/240/241/242)	87.2/11.9/0.8/0.1	80.2/17.4/1.9/0.5

RADIAL POWER DISTRIBUTION
 ALL-FAST FMSR, SODIUM COOLED - SOLID
 REFERENCE FMSR, HELIUM COOLED - DASHED
 BOTH CASES ARE E.O.E.C.

Figure 4.2.

placement in the Reference FMSR (see Figure 4.1). It can be seen from this figure that the effect on power-flattening influence from the moderator persists over about seven of the outermost subzones of the twenty subzones in the fast core. This power gradient reduction can have a significant influence on the thermal-hydraulic performance of these seven subzones. Since any good reactor design represents an effectively optimized compromise between physics, thermal hydraulics and material design objectives, an increased power gradient in any subassembly would imply a corresponding reduction in effective outlet temperature for those subassemblies. The decrease in thermodynamic efficiency would require a larger core volume to produce more thermal energy for the same electrical energy generation; higher burnup and power densities would be required if the core volume remained constant.

For the same subassembly movement pattern for the outer fourteen subassemblies, Figure 4.2 shows that the moderated subassemblies always produce more power than their corresponding equivalent of the all-fast design. From later work, it was shown that the power density in the outermost regions of the fast region of the Reference FMSR is considerably reduced upon inclusion of more accurate fission product treatments.

Figure 4.3 compares the buildup of Pu-239 in the two FMSRs. Layer 1 represents the axial core length from core mid-height to halfway to the core-blanket interface. Layer 2 represents the other half of the core axially. The initial Pu-239 buildup rate is more rapid in the all-fast FMSR. This is due in part to the higher radial leakage from the fast core which provides more neutrons for capture in the outer region. It is also partly due to the higher power generation rate in that region for the (moderated) Reference FMSR, as seen from Figure 4.2, whereby more of the Pu-239 generated in those subzones is consumed relative

PU-239 BUILDUP
 ALL-FAST FMSR, SODIUM COOLED
 REFERENCE FMSR, HELIUM COOLED (X)
 LAYER 1 - SOLID LAYER 2 - DASHED

Figure 4.3.

to the all-fast FMSR. Since the all-fast FMSR is sodium-cooled, it has a higher volume fraction of fuel, a higher fuel-to-steel ratio, and a lower fraction of fission products for the same power generated to that irradiation time. These factors tend to increase the effective equilibrium Pu-239 concentration for any sodium-cooled FMSR relative to that for any gas-cooled FMSR.

Figure 4.4 compares the buildup of fission products for the two FMSR designs. Again, it is seen that the higher power density in the outer, moderated region of the Reference FMSR results in a more rapid buildup of fission products. It can also be seen that the peak fission product buildup for the gas-cooled FMSR is greater than that for the sodium-cooled FMSR, largely for reasons which are analogous to those given for Pu-239 buildup. The somewhat greater rate of fission product buildup for "layer 2", the region halfway between the core center and the axial blanket, is another characteristic of the difference between gas-cooled and sodium-cooled FMSR behavior. A greater axial leakage of neutrons for the gas-cooled FMSR results from the difference of fuel volume fractions of 0.45 and 0.39 and the lower axial leakage inherent in the sodium-cooled version. This results in both a faster buildup of plutonium-239 in the upper layer of the gas-cooled FMSR, relative to the sodium-cooled FMSR, and in a greater power density with a correspondingly greater fission product buildup.

It is important to note that these differences are attributable to differences inherent in the choice of coolant and not to differences between the two FMSR designs.

No calculations have been made of the radial shielding requirements for the two designs. Since the all-fast FMSR has a larger radial gradient of power (flux), there will be a larger radial streaming current for this concept. Also, the absence of moderator may imply a harder and therefore more penetrating

FISSION PRODUCT BUILDUP
 ALL-FAST FMSR, SODIUM COOLED
 REFERENCE FMSR, HELIUM COOLED (X)
 LAYER 1 - SOLID LAYER 2 - DASHED

Figure 4.4.

neutron energy spectrum. The design of the radial shield will require detailed analysis and a specification of the balance of plant.

In summary, both FMSR designs appear to be feasible. The small neutronic differences were anticipated because the moderator of the Reference FMSR was in a position of low neutron flux and adjoint flux. The thermal-hydraulic problems of the all-fast FMSR design, as anticipated, are more severe than those of the Reference FMSR, particularly in the outer fast core regions. This problem could lead to difficult design problems for the helium-cooled, all-fast FMSR, especially with regard to achieving reasonable thermodynamic efficiencies. For the sodium-cooled, all-fast FMSR, the thermal-hydraulic difficulties are more manageable.

5. THERMAL-HYDRAULIC STUDIES

5.1 INTRODUCTION

In this section the results of the thermal-hydraulic analyses are presented for the sodium-cooled and helium-cooled systems. In both cases, the analyses were closely interfaced with physics and material requirements. Detailed studies were performed to assess the impact of hot-spot uncertainties and subassembly power peaking factors on the core thermal-hydraulic performance. In addition, the design study considered the requirements of overall plant balance such as pumping power and mixed mean outlet temperature. The preliminary study has shown that for both sodium-cooled and helium-cooled systems, the core behavior is compatible with all design limits and requirements.

5.2 SODIUM-COOLED DESIGN

5.2.1 General Design Considerations

The thermal-hydraulic design of the sodium-cooled FMSR core follows that of the conventional LMFBR. The sodium enters the core region from the lower plenum, flows through the core inlet section, including the orifice, and then flows upward along the fuel rod bundles. The fuel rods are assumed to be helically wire wrapped along their entire length. The hot sodium leaves the core region through the assembly outlet section located on the top of the core. The present study focuses on thermal-hydraulic computations for the core region only. Heat transport processes in the intermediate loop and steam-water loop are not included.

Metallurgical considerations set the maximum fuel centerline and fuel-clad contact temperatures at 850°C and 625°C, respectively. Thus, one objective is to achieve a thermal-hydraulic design consistent with the metallurgical limitations. In addition, the mixed mean coolant outlet temperature is

also a major factor in the thermal-hydraulic design; the coolant inlet temperature and the temperature rise in the core region are closely related to the thermodynamic efficiency of the power plant. For the present design, the coolant inlet temperature was 380°C and the mean temperature rise 160°C. These temperatures are comparable with those of CRBR, Super-Phenix and other large LMFBR designs. A preliminary study of the overall plant balance has shown that these temperatures yield good performance of the IHX and the steam generator, with an achievable thermodynamic efficiency of 36%. The sodium-cooled FMSR is a low pressure system. The inlet pressure selected for the present study is 140 psi, which is more than sufficient to cover the total pressure drop in the core region. For the sodium-cooled core, the ambient coolant pressure has no significant effect on pumping power and heat transfer performance. The design goals for this initial study are shown in Table 5.1.

The general design parameters for the sodium-cooled FMSR, such as the dimensions of the fuel pin, its pitch, duct, etc. have been kept as conventional as possible with respect to standard LMFBR designs. The parameters must be consistent with reactor physics, materials, mechanics, and thermal-hydraulics requirements; cost and other optimization objectives must also be considered. Selected design parameters are compared with the CRBR Demo design parameters in Table 5.2. The sodium-cooled FMSR has a large fuel volume fraction. The higher thermal conductivity of metal fuel relative to oxide fuel is the primary consideration for the larger FMSR pin diameter. The number of fuel pins is increased to 331 per assembly, primarily as a result of the larger subassembly size. Preliminary parameterization studies have included cases with 271 pins per assembly. However, by comparing the pressure drop and fuel temperature distribution, it was concluded that 331 pins per assembly is more

Table 5.1

Design Goals for the Na-Cooled FMSR

<u>Parameter</u>	<u>Design Objective</u>
Max. Fuel Centerline Temperature	850°C
Max. Clad Temperature	625°C
Coolant Inlet Temperature	380°C
Coolant Outlet Temperature (Mixed Mean)	540°C
Inlet Pressure	140 psi

Table 5.2

General Design Parameters of the Na-Cooled FMSR
and Comparison to CRBR Demo Design

<u>Parameter</u>	<u>FMSR</u>	<u>CRBR DEMO</u>
Reactor Power, MW(e)	1000	300
Core Inlet Temperature, °C	380	388
Core Outlet Temperature, °C	540	535
Active Core Height, cm	160	91.4
Fuel Pin Length, cm	270	290.6
Cladding OD, cm	0.8693	0.5842
Cladding Thickness, cm	0.0432	0.0381
Fuel Pellet OD, cm	0.7829	0.4915
Fuel-Cladding Gap, cm	0.0	0.0082
Duct Wall Thickness, cm	0.254	0.3048
Hexcan Size (Inner), cm	18.69	11.011
Fuel Volume Fraction	0.46	0.325
Flowing Coolant Volume Fraction	0.29	0.419
Pitch of Fuel Pins, cm	1.018	0.731
P/D	1.171	1.251
No. of Pins/Subassembly	331	217
Wire-Wrap OD, cm	0.149	0.142
Wire-Wrap Pitch, cm	30.0	30.23

suitable to the present application. (More detailed discussions of thermal-hydraulic performances are given in Section 5.2.3.) The fuel pin design used in the sodium-cooled FMSR is assumed to employ the same operating principle as the EBR-II Mark-II fuel pin, which is a sodium-bonded metal fuel. According to the operating experiences of the EBR-II, the fuel-cladding gap is practically closed after about 3 atom percent burnup due to the radiation damage and thermal expansion of the metal fuel. A closed fuel-cladding gap is assumed in the present design calculations, as indicated in Table 5.2.

The total core is divided into four axial sections: the lower axial blanket (40 cm), the active core (160 cm), the upper axial blanket (40 cm) and the upper vented plenum (30 cm). A comparison of the axial dimensions of the sodium-cooled FMSR fuel and CRBR fuel is given in Figure 5.1.

5.2.2 Methods

The thermal-hydraulic design evolved from several iterations which were interfaced with physics and materials requirements. Thermal-hydraulic computations were performed with the SATURN code which is a flexible, one-dimensional, multichannel, steady-state code for thermal-hydraulic design analysis of sodium-cooled fast reactors developed at BNL for the FMSR Program. The code uses the same radial and axial zoning for heat generation as those used in the fuel cycle calculations. Thus, the thermal hydraulics and physics can be matched to show the entire performance for a given FMSR design. The SATURN code first establishes the initial conditions by computing fuel and clad dimensions; all hydraulic parameters based on the input design requirements are determined. With these initial conditions established, the code then computes individual channel coolant flow rates to achieve a specified temperature rise and the same pressure drop for all radial zones. The total pressure drop

Figure 5.1. Fuel Subassembly Axial Dimensions

consists of pressure drops due to friction, gravity, acceleration and flow area changes. Detailed pressure drops in five axial regions are computed for each channel. The five regions are the core support module, inlet nozzle, orifice plate and shield, rod bundle, and assembly outlet. (Axial pressure distribution of the reference case is discussed in Section 5.2.3.) To achieve a consistent pressure drop for all radial zones, an iteration procedure is used in the code to adjust the loss coefficient of the orifice plate in each zone.

The sodium-cooled FMSR fuel is to employ the design principle of the Mark-II metal fuel which has had a long history of successful operation in the EBR-II reactor. ANL has shown that the thermal conductivity of the metal fuel depends on the fuel temperature, fission product content, and porosity.^(6,7) Empirical correlations of the thermal conductivity, which were developed by ANL, were used in this design study. The physical properties of sodium, such as enthalpy, density, viscosity and thermal conductivity, are functions of temperature, as given in Reference 8. The correlation of frictional pressure drop for turbulent flow in triangular rod bundles developed by Novendstern,⁽⁹⁾ and the correlation of heat transfer recommended by Kazimi⁽¹⁰⁾ are adopted in the SATURN code. These correlations are well developed for LMFBRs and are given in Appendix A for reference.

The detailed temperature distributions in fuel and clad, as well as the pressure and temperature of the coolant in each axial node, are computed in SATURN for all radial channels. The computations are for the average pin in each channel under the nominal operating conditions. Hot-spot and hot-channel calculations are also performed.

The impact of uncertainties on the thermal-hydraulic performance is represented by the application of hot spot factors. The hot spot factors were defined according to the conventional LMFBR practice.⁽¹¹⁾ These factors represent the statistical and other uncertainties of theoretical and experimental analyses, instrumentation accuracy, manufacturing tolerances, physical properties and correlations. Descriptions of the hot spot factors are given in Appendix B. The hot spot analysis was applied to the peak power channel only. It is referred to as the hot channel. The hot channel represents the hot pin in the peak power assembly affected by the simultaneous occurrence of all uncertainties. Preliminary results for this calculation are presented at the end of the following section.

5.2.3 Thermal-Hydraulic Performance

A reference case based on the centrally-moderated design at the beginning of the equilibrium cycle was constructed to demonstrate the thermal-hydraulic performance of the sodium-cooled FMSR. Parameters used for this reference case are given below:

Number of radial subzones	33
Subassemblies per subzone	12
Volume fraction	
Fuel	0.46
Flowing coolant	0.290
Fluid between ducts	0.078
Steel	0.172
Fuel porosity, %	25
Fission product content in fuel, %	10

To perform the multichannel thermal-hydraulic analysis, the 33 radial subzones, for which two-dimensional power density distributions were determined by physics computations, were used in this study. Each channel was represented by the average fuel pin of twelve subassemblies. The computed results are summarized in Table 5.3. The sodium-cooled FMSR is designed to operate at a rated power level of 2776 MW(t) which yields 1000 MW(e) for a 36% thermodynamic efficiency. The pressure drop across the core region under the rated design conditions is 113 psi, which is comparable to the pressure drops of the CRBR and FFTF. Using sodium as the coolant, the pumping power is expected to be a small portion of the power generated. The computed result shows that the pumping power for the core region is about 13.4 MW, less than 2% of the electric power generated. It is noted in Table 5.3 that Zone 22 (marked by an *) is the peak power channel. The average linear power and maximum power density are 7.11 kw/ft and 330.7 W/cm^3 , respectively. Corresponding to this power level, the maximum centerline temperature of the fuel is 685°C and of cladding 555°C . These temperatures are considerably lower than the metallurgical limit, which is 850°C for the fuel and 625°C for cladding.

Detailed thermal-hydraulics results for Zone 28, a high power zone under nominal conditions, are given in Table 5.4. The coolant temperature and pressure distributions, fuel and clad temperatures, heat transfer rate, friction factor and Reynolds numbers are listed for all axial nodes. The coolant enters the core region at 380°C ; its temperature rapidly increases in the active core region reaching 542°C at the core exit. The inlet temperature and the temperature rise are input parameters for this reference case. The clad outer surface temperature is slightly higher than the coolant temperature along the entire length of the fuel pin. This small temperature difference is due to

Table 5.3

Summary of Core Fuel Thermal-Hydraulics for
Average Pins in Na-Cooled FMSR

TOTAL FLOW = .1452E+08 GM/S
TOTAL POWER = .2776E+04 MW
TOTAL PUMPING POWER = .1346E+02 MW
TOTAL PRESSURE DROP = 113.23 PSI
MEAN COOLANT TEMPERATURE = 530.87 C

--AVERAGE PIN-- (NOMINAL)							
ZONE	NO OF ASSEMBLY	MAX POWER W/CM3	LINEAR POWER KW/FT	POWER FRACTION	FLOW FRACTION	MAX CLAD T (C)	MAX FUFL T (C)
20	12	279.6	5.70	.0451	.0396	553.2	659.4
19	12	262.6	5.47	.0433	.0415	539.7	642.1
18	12	249.3	5.20	.0412	.0390	540.9	636.7
17	12	230.9	4.80	.0382	.0356	542.8	628.9
16	12	220.6	4.58	.0364	.0352	536.7	620.1
15	12	225.8	4.67	.0373	.0352	539.0	624.2
14	12	199.4	4.14	.0328	.0311	539.6	611.5
21	12	225.1	6.30	.0500	.0457	547.5	665.0
24	12	322.8	6.98	.0552	.0518	544.6	674.2
23	12	324.5	7.01	.0555	.0495	552.9	680.7
* 22	12	330.7	7.11	.0564	.0494	555.4	685.3
25	12	250.3	5.54	.0441	.0514	512.1	618.0
26	12	288.5	6.49	.0517	.0498	540.0	654.6
29	12	291.9	6.73	.0537	.0508	543.0	657.1
28	12	303.1	6.99	.0559	.0520	545.8	664.7
27	12	304.0	6.98	.0558	.0519	547.9	666.8
33	12	222.1	5.30	.0426	.0438	528.9	613.0
32	12	235.4	5.61	.0452	.0442	536.4	624.8
31	12	229.7	5.46	.0440	.0437	534.0	620.4
30	12	210.7	5.01	.0404	.0408	531.0	608.9
13	12	78.7	1.72	.0138	.0204	482.1	507.0
12	12	89.7	1.97	.0159	.0207	496.0	523.6
11	12	82.2	1.80	.0147	.0199	491.0	515.0
7	12	16.9	.43	.0035	.0056	473.4	474.0
8	12	29.0	.64	.0052	.0107	452.7	456.3
9	12	36.1	.79	.0065	.0121	459.3	463.9
10	12	40.9	.89	.0073	.0140	459.0	466.7
6	12	5.6	.13	.0011	.0021	456.3	456.5
3	12	5.9	.14	.0011	.0015	495.4	495.6
5	12	5.8	.13	.0011	.0023	451.3	451.5
2	12	5.5	.20	.0017	.0032	460.1	460.5
4	12	7.2	.16	.0014	.0026	460.3	460.5
1	12	8.2	.20	.0017	.0031	463.0	463.4

Table 5.4

Thermal-Hydraulic Results for Average Pin at Nominal Conditions in Zone 26 of the Na-Cooled FMSR

S A T U R N												
ZONE 28--AVERAGE PIN AT NOMINAL CONDITION												
COOLANT AREA PER FUEL PIN = .3032 CM**2						HYDRAULIC DIAMETER = .3573 (CM)						
CLAD O D = .8693 CM						FUEL O D = .7829 CM						
COOLANT FLOW RATE = 627.40 GPM/CM**2/SEC						P/D RATIO = 1.171						
						MASS FLOW PER ASSEMBLY= 62978.70 GMS/SEC						
(11) POWER DENSITY OF EACH NODE W/CM3												
6.96	13.23	18.07	23.67	30.28	38.43	48.61	95.77	113.67	131.40	149.22	175.77	210.55
250.00	275.79	293.85	303.10	303.10	293.85	275.79	250.00	210.55	175.77	149.22	131.40	113.67
95.77	48.61	38.43	30.28	23.67	18.07	13.23	6.96					
LINEAR HEATING RATE = 6.99 KW/FT VOLUME HEATING RATE = 219.26 WATTS/CM3												
AXIAL NODE	HEIGHT (CM)	PRESSURE PSI	SATURATION	COOLANT	TEMPERATURE (C)		FUEL PELLET SURFACE	FUEL INNER ZONE	FUEL CENTERLINE	HTR (BTU/HR /FT**2/F	REYNOLDS NUMBER	FRICTION FACTOR
					CLAD OUTER SURFACE	CLAD INNER SURFACE						
LOWER PLENUM		140.00		380.00								
CORE INLET		139.02		380.00								
CORE SUPPORT MOD		128.48		380.00								
INLET NOZZLE		108.97		380.00								
SHIELD/ORIFICE		85.09		380.00								
1	5.00	84.08	1112.34	380.34	380.50	380.97	381.00	385.41	385.87	29100.39	77357.57	.0206
2	12.50	82.55	1109.45	380.63	380.54	381.84	381.90	390.24	391.10	29095.87	77401.86	.0206
3	17.50	81.53	1107.50	380.97	381.59	382.62	382.70	394.04	395.22	29090.66	77452.91	.0206
4	22.50	80.52	1105.53	381.42	381.57	383.59	383.69	398.47	400.00	29083.71	77520.98	.0206
5	27.50	79.50	1103.54	382.01	382.71	384.78	384.91	403.71	405.65	29074.72	77608.95	.0206
6	32.50	78.48	1101.53	382.75	383.65	386.27	386.43	410.11	412.56	29063.29	77720.96	.0206
7	37.50	77.46	1099.50	383.70	384.83	388.14	388.35	418.04	421.11	29048.81	77862.80	.0205
8	42.50	76.45	1097.45	385.27	387.59	394.03	394.44	430.85	436.67	29024.81	78097.99	.0205
9	47.50	75.43	1095.37	387.54	390.19	397.94	398.43	434.39	471.18	28990.04	78438.92	.0205
10	52.50	74.41	1093.28	390.20	393.26	402.23	402.79	477.92	485.65	28949.40	78837.48	.0205
11	57.50	73.39	1091.16	393.25	396.74	406.92	407.55	491.61	500.25	28902.96	79293.35	.0205
12	65.00	71.86	1087.93	398.70	402.81	414.81	415.56	512.39	522.32	28820.27	80105.64	.0204
13	75.00	69.82	1083.55	407.11	412.06	426.43	427.32	534.98	551.52	28693.23	81355.48	.0203
14	85.00	57.78	1079.07	417.15	423.06	440.12	441.18	578.71	583.96	28542.62	82840.05	.0202
15	95.00	55.74	1074.48	428.62	435.18	454.01	455.18	594.56	598.81	28371.82	84527.50	.0201
16	105.00	53.70	1069.78	441.08	448.11	468.17	469.42	614.86	629.72	28188.15	86346.62	.0200
17	115.00	51.67	1064.97	454.15	461.55	482.14	483.43	638.99	646.06	27997.23	88242.58	.0199
18	125.00	49.63	1060.04	467.43	474.79	495.48	496.77	642.59	657.49	27805.00	90156.81	.0198
19	135.00	47.61	1054.99	480.54	487.72	507.77	509.02	649.41	663.76	27617.34	92030.71	.0197
20	145.00	45.58	1049.81	493.06	499.84	518.66	519.83	651.28	664.73	27439.79	93808.39	.0196
21	155.00	43.56	1044.48	504.62	510.21	527.87	528.93	648.50	660.75	27277.25	95440.01	.0196
22	165.00	41.54	1039.01	514.76	520.09	534.37	535.26	637.08	647.53	27135.94	96861.85	.0195
23	175.00	39.53	1033.38	523.27	527.65	539.66	540.40	626.34	635.17	27018.17	98049.20	.0194
24	182.50	38.02	1029.05	528.79	532.52	542.71	543.34	617.05	624.64	26942.22	98816.10	.0194
25	187.50	37.02	1026.11	531.88	535.13	544.14	544.70	616.06	616.79	26899.78	99244.98	.0194
26	192.50	36.02	1023.11	534.58	537.44	545.19	545.68	612.65	608.52	26862.80	99619.01	.0194
27	197.50	35.01	1020.07	536.89	539.59	545.83	546.24	608.64	599.63	26831.24	99938.27	.0193
28	202.50	34.01	1016.99	538.48	539.71	543.02	543.23	604.47	571.09	26809.53	100158.14	.0193
29	207.50	33.02	1013.85	539.44	540.11	543.03	543.19	600.25	565.33	26796.44	100290.59	.0193
30	212.50	32.02	1010.65	540.20	540.96	543.03	543.16	595.03	560.68	26786.12	100395.12	.0193
31	217.50	31.02	1007.39	540.79	541.39	543.01	543.11	590.56	556.85	26778.02	100477.17	.0193
32	222.50	30.02	1004.07	541.25	541.71	542.94	543.02	585.55	553.54	26771.76	100540.62	.0193
33	227.50	29.03	1000.69	541.60	541.93	542.84	542.89	580.90	550.52	26767.06	100588.20	.0193
34	235.00	27.53	995.49	541.90	542.07	542.55	542.58	576.27	546.56	26762.99	100629.46	.0193
CORE TOP		26.53		542.05								
UPPER PLENUM OUTLET		30.56		542.05								
ASSEMBLY OUTLET		26.77		542.05								

ORIFICE LOSS COEFFICIENT

7.20

the high heat transfer rate possible with sodium. Inspection of the temperatures at the clad inner surface and in the fuel region shows that the radial temperature variation of the fuel pin is relatively small; this is due to the high thermal conductivity of the metal fuel and the low contact resistance at the sodium-bonded fuel-clad interface. Using the cosine-type power profile obtained from the physics calculations, the maximum fuel centerline temperature (664°C) occurs 25 cm above the mid-core position. The maximum clad temperature (545°C) at the clad inner surface occurs near the top of the active core region, 197.5 cm. The heat transfer rates and friction factors computed from correlations given in Appendix A, and the Reynolds numbers are also listed in Table 5.4 for reference. The Reynolds numbers, which range from 7.7×10^4 to 10×10^4 , are within the experimental range from which the empirical correlations were developed. The coolant saturation temperatures based on the local pressure were computed in SATURN and are also included in Table 5.4. Comparison of the saturation temperature with the coolant temperature indicates that the sodium is highly subcooled. The highly subcooled sodium should provide a large temperature margin for safety considerations. The axial pressure distribution in Table 5.4 is illustrated in Figure 5.2. It is seen that the pressure drops for the various regions are comparable to those of the CRBR.

The above discussion and the results shown in Tables 5.3 and 5.4 correspond to the average pin under nominal conditions. To account for the design uncertainties, the hot spot factors described in Appendix B were applied to the peak power channel, i.e., Zone 22. In general, a reasonable estimate of hot spot factors requires a complete knowledge of the reactor design and the interaction of the heat transport systems. Since these data are not presently

Figure 5.2. Pressure Drop in Fuel Assembly

available for the FMSR, the hot spot factors used in this preliminary study were largely based on a modification of CRBRP hot channel factors. The computed maximum clad and fuel temperatures are given below:

	Nominal Condition, No Hot Channel Effect	With Hot Spot Factors Applied
Clad	555°C	625°C
Fuel	685°C	765°C

These results show that the maximum hot pin temperatures in the peak power assembly satisfy metallurgical limits even when all the uncertainty factors are applied.

5.2.4 Conclusions

The following conclusions can be drawn from this initial study:

- 1) Fuel and clad temperatures can be maintained below the current maximum temperature limits even when hot spot factors computing clad and fuel temperatures are included.
- 2) The total core pressure drop corresponds to current LMFBR designs.
- 3) Total pumping power is small.

Future thermal-hydraulic related analyses will include:

- 1) Improved orificing scheme for both high-pressure and low-pressure regions, optimized on a pressure drop basis;
- 2) Evaluation of the consequences of duct dilation;
- 3) Evaluation of effects of including radial power and flow distributions within the subassembly;

- 4) Improved hot-channel analysis;
- 5) Modeling of fuel thermal behavior during the entire irradiation lifetime of the fuel.

5.3 HELIUM-COOLED DESIGN

Thermal-hydraulic design studies reported here utilized the data base assembled for the General Atomic GCFR program, with one important difference; the gas-cooled FMSR operates at 2000 psi ambient pressure as opposed to 1320 psi for the oxide-fueled GCFR. This was done in order to exploit the superior heat transfer characteristics of the metal fuel as well as to reduce pumping power requirements.

As discussed in the Initial Feasibility Study for the FMSR,⁽¹⁾ the heat transport system of the helium-cooled FMSR design closely follows that of the 300-MW(e) GCFR Demo design.⁽¹²⁾ Significant differences between the GCFR and the sodium-cooled fuel pin designs stem from the fact that in the GCFR the fuel pins may be partially roughened; in addition, spacer grids are used in the GCFR fuel pin subassemblies. Both metal-bonded and nonmetal-bonded fuel pins are considered in this study. It is found that fuel pin roughening could possibly be eliminated if a metal-bonded fuel pin design is possible for a gas-cooled system.

The centrally-moderated FMSR design is the basis of the present study. The physics data correspond to the beginning of equilibrium cycle. Only the reactor core thermal hydraulics are discussed here; balance of plant considerations will be deferred to later studies. This report also contains results for hot spot and radial power peaking effects which were not included in our previous report.

5.3.1 Design Goals and Design Parameters

As discussed in the earlier report,⁽¹⁾ metallurgical and conventional GCFR design requirements dictate many of the constraints imposed on the FMSR design. The main objective is to achieve a design consistent with an 850°C maximum hot spot fuel centerline temperature, 625°C maximum hot spot cladding inner surface temperature, and good thermodynamic efficiency at reasonable pumping powers. The design goals used in this study are shown in Table 5.5.

From the Initial Feasibility Study, it was concluded that a 331-pin fuel subassembly design was more suitable than a 271-pin fuel design.⁽¹³⁾ The reduction in fuel centerline temperature achieved with the 331-pin design outweighed the adverse effect of increased subassembly pressure drop.

The general design parameters for the FMSR, such as the fuel pin diameter, pitch, hexcan size, etc., are derived from the information obtained from reactor physics calculations. These are given in Table 5.6; a comparison with the 300-MW(e) GCFR Demo design is also shown. The active core height is 160 cm; the upper and lower axial blankets are each 40 cm long, and the total length of the upper and lower plenums is 30 cm. The GA venting and pressure balancing is also assumed for the FMSR. There is no central hole in the fuel or fuel-cladding gap in this design. Volume fractions for fuel, coolant and steel are 0.39, 0.46 and 0.15, respectively. The hot spot factors used in this design study are shown in Table 5.7.

5.3.2 Description of Methods

5.3.2.1 JUPITER

Heat transfer, pressure drop and pumping power calculations for the FMSR design were performed with the JUPITER code⁽¹⁴⁾ developed at BNL for this purpose. The FMSR design evolved from several iterations interfacing JUPITER calculations with physics and materials requirements.

Table 5.5

Design Goals for the He-Cooled FMSR

<u>Parameter</u>	<u>Design Objective</u>
Maximum Hot Spot Fuel Centerline Temperature	850 ⁰ C
Maximum Hot Spot Clad Inner Surface Temperature	625 ⁰ C
Coolant Temperature Rise	210 ⁰ C
Mixed Mean Coolant Outlet Temperature	510 ⁰ C
Inlet Pressure	2000 psi
Maximum Reynolds Number	2 x 10 ⁵
Pumping Power	170 MW

Table 5.6

General Design Parameters of He-Cooled FMSR
Reference Design

<u>Parameter</u>	<u>FMSR Values</u>	<u>GCFR</u>
Reactor Power, MW(e)	1000	300
Active Core Height, cm	160	100
Number of Fuel Subassemblies	408	118
Number of Pins/Subassembly	331	271
Hexcan Pitch, cm	20.0	17.51
Duct Wall Thickness, cm	0.2540	0.254
Hexcan Size (Inner), cm	18.6920	16.35
Cladding O.D., cm	0.8073	0.7400
Cladding Thickness, cm	0.0432	0.0483
Fuel Pellet O.D., cm	0.7209	0.6333
Fuel Cladding Gap, cm	0.0	0.0051
Fuel Volume Fraction	0.3900	0.277
Coolant Volume Fraction Inside Hexcan	0.3844	0.40
Stagnant Coolant Volume Fraction	0.0784	
Pitch, Fuel Pins, cm	1.0274	1.0000
P/D, Fuel Pins	1.2727	1.35
Number of Spacer Grids		10
Smooth Zone	10	
Roughened Zone	2	

Table 5.7
Hot Spot Factors Used for the
He-Cooled FMSR Design

<u>Parameter</u>	
Channel Hot Spot Factor	1.111
Film Hot Spot Factor	1.248
Cladding Hot Spot Factor	1.135
Gap Hot Spot Factor	1.100
Fuel Hot Spot Factor	1.300

JUPITER is a flexible, one-dimensional, multi-channel, steady-state code for thermal-hydraulic analysis of gas-cooled fast reactors. A JUPITER channel represents one or more subassemblies by the thermal-hydraulic performance of a single equivalent fuel pin. There is no interchannel thermal communication in JUPITER. In gas-cooled reactors, the fuel pins are surface-roughened to enhance heat transfer between the cladding and coolant. In JUPITER, the degree of cladding roughness that is available can be tailored to the cladding temperature requirements. To determine the amount of roughening required, the core is divided into five axial segments so that five different degrees of cladding roughness may be used. In addition to the active core region, a lower and upper plenum and a lower and upper axial blanket can also be modeled. The code can compute coolant flow rates required by each channel to achieve a desired temperature rise across all channels. The orifice pressure drops are also computed for each of the channels such that the pressure drops across all the fuel elements are constant. The user can also specify flow rates and orifice loss coefficients as input for each channel. Figure 5.3 gives a schematic description of a JUPITER channel. Each channel is divided into several axial nodes. Power generated at each node (obtained from 2DB reactor physics results) can be given in either kw/ft or W/cm³.

JUPITER employs two radial zones in the fuel which can have different physical properties. The radii of these zones can vary with axial height. The fuel can also have a central hole. For the FMSR design, there is no central hole. Although JUPITER allows a fuel-cladding gap, there is no gap in the present gas-cooled FMSR design. The temperature- and pressure-dependent thermophysical properties of the coolant have been taken into account. Details of the thermophysical properties used in this design study are given in the previous report.⁽¹⁾

Figure 5.3. Schematic Description of a JUPITER Channel

Several options are available in JUPITER to model the heat transfer and friction factor at the cladding and coolant interface. As in the GCFR, the roughness multiplication factors for the friction and heat transfer coefficients, which multiply the corresponding coefficients for smooth lattice rods, have been incorporated into JUPITER. The roughness multipliers can be specified as input or calculated by the code from built-in correlations. Another option available in JUPITER is the use of the subroutine TOMSON, which explicitly treats the cladding roughness design and determines the heat transfer and friction factor for each axial node from the corresponding temperature and velocity profiles. This has been obtained from the BASIC subroutine of CALIOP⁽¹⁵⁾ developed at General Atomic and made available to BNL.

The present version of JUPITER has been improved to include effects not previously modeled in order to refine the calculational results. Major improvements were made in the subassembly pin geometry specification and in the calculation of hot spot effects. The subassembly pin geometry calculation starts with a hexcan pitch which is consistent with the reactor physics models. The code needs one of three inputs; namely, coolant volume fraction, cladding O.D. or the fuel-cladding gap; the other two parameters are internally calculated. Improvements were also made in the pressure drop calculation procedures. Hot spot calculations are discussed in the next section.

JUPITER also initializes and does a consistency check of the subassembly fuel-pin design from the given set of input data. Since edge-pin effects are important, it is designed to perform one-dimensional parametric studies with different coolant channel geometries within a subassembly. The flow areas and the hydraulic diameters for these channels are given in Table 5.8. The flow distribution among the interior, wall, and corner channels is determined such

Table 5.8

Flow Area and Hydraulic Diameter Relations
for Various Channels in JUPITER

Type of Channel	Flow Area/Pin CAREA	Hydraulic Diameter
1 Infinite Lattice with Fuel & Flowing Coolant Volume Frac- tions VF & VCF	Fuel Area Per Pin * $\frac{VCF}{VF}$	$\frac{4.0 * CAREA}{\pi * clad OD}$
2 Hexcan Average Pin	Total Coolant Area in Hexcan/NPIN	$4.0 * CAREA / \{ \pi * Clad OD + 2\sqrt{3} * DFLAT/NPIN \}$
3 Interior Wall Corner (240)	$CAREAI = \frac{\sqrt{3}}{2} (Pitch)^2$ $- \pi (Clad OD)^2 / 4.0$ $CAREAW = 2.0 \{ Pitch * DWPIN$ $- \pi (Clad OD)^2 / 8 \}$ $CAREAC3 = 1.5 DWPIN * (Pitch + DWPIN / \sqrt{3})$	$\frac{4.0 * CAREAI}{\pi * Clad OD}$ $2.0 * CAREAW / \{ Pitch + \pi * Clad OD / 2 \}$ $4.0 * CAREAC3 / \{ \pi * Clad OD + DWPIN * \sqrt{3} + 1.5 * Pitch \}$
4 Interior Wall Corner (60)	<p align="center">Same as in 3</p> <p align="center">Same as in 3</p> $CAREAC4 = 6.0 \{ (DWPIN)^2 / \sqrt{3} - \pi (Clad OD)^2 / 24.0 \}$	<p align="center">Same as in 3</p> <p align="center">Same as in 3</p> $CAREAC4 / 1.5 \{ 2 * DWPIN / \sqrt{3} + \pi * Clad OD / 6.0 \}$

*NPIN = Number of pins per subassembly

DFLAT = Distance across inside hexcan flats

DWPIN = Distance of edge pin center from hexcan wall

that the pressure drop across these channels is the same. This is achieved reasonably well by an iterative procedure developed in JUPITER.

Pressure drop calculations include contributions due to friction, momentum change, orifices, and inlet and outlet pressure losses, in addition to spacer grid pressure drops. Spacer grid pressure drop calculations have been improved by providing a new option which uses two separate Rehme's correlations for spacer grids in the smooth and in the rough regions. Details of the procedure are given in Appendix C. The spacer grid pressure drop calculation assumes a design with 17.9% of the flow area blocked by the grid. These values are representative of the 300-MW(e) Demo GCFR design.⁽¹²⁾ The orifice loss for the peak channel is taken to be zero. The orifice pressure drops for the other channels are adjusted such as to produce the same pressure drop across the full subassembly as in the peak channel. The pumping power is calculated separately for each channel and then summed for the reactor.

An average pin was used for the thermal-hydraulic studies required for this study. Corner pin and wall pin effects are currently being studied. In the present calculations, core flow rates were obtained by first specifying the highest power channel requirements, as obtained from the reactor physics design. Resultant flow rates and the corresponding orifice loss coefficients were then used to determine the average thermal-hydraulic characteristics for the core.

5.3.2.2 Hot Spot Factors for FMSR

Inclusion of hot spot factors in the analysis of the FMSR core is one of the most important refinements implemented since the completion of the Initial Feasibility Study. The present thermal-hydraulic analyses can treat random and non-random uncertainties such as deviations in geometrical parameters from nominal dimensions or uncertainties in the thermal-hydraulic correlations. Five hot

spot factors are employed for this purpose. These are channel hot spot factor (FCH), film hot spot factor (FLM), cladding hot spot factor (FCL), gap hot spot factor (FGP) and fuel hot spot factor (FFL). These factors include the important uncertainties that affect the nominal temperatures.

Since the FMSR metal fuel is markedly different from the GCFR oxide fuel, and no metal manufacturing experience exists for the proposed sintered metal fuel, engineering judgments must be used to estimate which factors relate to the fuel. At present, the hot spot factors for the coolant and the clad, namely, FCH, FLM and FCL, are taken from General Atomic GCFR studies⁽¹⁶⁾ and are shown in Table 5.9. The estimates for the hot spot factors in Table 5.9 are accepted for the FMSR core design. The hot spot factors, FGP and FFL, will have to be determined separately for the FMSR fuel. The hot spot factor for the gap arises from changes in gap width and variations of gap/contact conductance. Since there is no gap in the FMSR fuel, the major uncertainty of gap conductance will be negligible here. However, the uncertainty in maintaining good fuel-cladding contact during the fuel's lifetime must be determined. If a liquid-metal bond is used for the helium-cooled FMSR fuel, the uncertainties arising from variations in dimensions of fuel and cladding will also be largely reduced. For the FMSR, a value of 1.10 will be used for FGP until better data are available. Major contribution to FFL comes from the uncertainty in the thermal conductivity of fuel. The FFTF oxide fuel assigns⁽¹⁷⁾ a subfactor of 1.15 for fuel thermal conductivity. Data available⁽¹⁸⁾ on some metal fuel reactors are shown in Table 5.10. Reasonable FMSR values for these factors are also shown. Since fuel for the FMSR is not yet completely defined, a full discussion on this subject is not possible. Table 5.10 indicates that a value of 1.3 for FFL will be adequate for our present analysis.

Table 5.9
Hot Spot Factors for GCFR Fuel Assembly

Parameter	Channel Factor FCH	Film Factor FLM	Cladding Factor FCL
Statistical Factors			
Tolerance	1.056	1.055	1.059
Correlation and Geometry	1.002	1.112	1.1
Total (3 σ)	1.056	1.125	1.116
(2 σ)	1.037	1.083	1.077
Cumulative Factors			
Analysis Uncertainty	1.07	1.054	1.054
Other	1.007	1.142	1.10
Total	1.071	1.152	1.154
Overall Hot Spot Factor (3 σ)	1.131	1.296	1.195
Overall Hot Spot Factor (2 σ)	1.111	1.248	1.135

Table 5.10
Fuel Hot Spot Factors for
Several Metal-Fueled Breeder Reactors

Parameter	Hot Spot Factors (f_i)			
	FERMI	EBR-II	1000 MW(e) ⁽⁷⁾	FMSR (Projected)
Neutron & Gamma Flux Distribution	1.05+	1.10	1.10	1.10
Fissile Fuel Maldistribution	1.02		1.035	1.03
Burnup in Core Fuel Pin	1.10			*
Pin Geometry Variation	1.02	1.02	1.01	1.01
Fuel Thermal Conductivity Variation	1.20	1.15	1.20	1.20 ^{†*}
Possible Transient Overload		1.05		
Total ^{\$} f_{sa}				1.23
Power Measurement & Control	1.08	1.02	1.05	1.05
Analysis Uncertainty				1.05
Total ^{\$} f_{ca}				1.07
Overall Hot Spot Factor ($f_{sa} * f_{ca}$)				1.32

*The subfactor could be much higher for FMSR.

†Since FMSR fuel is not yet completely defined, this number may change.

^{\$} f is defined as $1 + \sqrt{\sum_i (f_i - 1)^2}$.

Details of the hot spot calculations as currently done in JUPITER are given in Appendix D.

5.3.2.3 Verification of JUPITER

An initial verification of JUPITER has been performed by modeling the BR-2 mockup experiment (CE-III) performed at Karlsruhe.⁽¹⁹⁾ These experiments were carried out to verify the 12-rod fuel bundle design to be irradiated in the BR-2 reactor in Belgium. The specifications of these tests are similar to the GCFR fuel assembly design and conditions, e.g., roughened fuel rods with spacer grids in high pressure He coolant. The test tubes had a volumetric O.D. of 0.786 cm, a pitch of 1.11 cm and a roughened length of 59 cm. Further details are given in Reference 19. JUPITER predictions for the pressure drop are shown in Table 5.11 along with the experimental data and the prediction of COBRA*GCFR.⁽²⁰⁾ As observed in Reference 19, the computed pressure drop is slightly lower than the experimental result. This has been conjectured⁽¹⁹⁾ to be due to local pressure losses at the outlet. The computed tube outer surface temperatures in the interior channel more closely follow the test data (within approximately 4%). These results lend confidence to the adequacy of the JUPITER code.

5.3.3 JUPITER Model for the Centrally-Moderated FMSR Core

A 26-channel core model was constructed from the reactor physics data for the centrally-moderated FMSR design. Twenty-two channels were used for the fast and centrally-moderated zones. The outer two rings containing 168 subassemblies were divided into four channels. The number of subassemblies and the power produced in each channel are shown in Table 5.12. The table also shows the radial location of these channels (channel numbers increasing from the center to the outer zones) and their identification to reactor physics zones. Figure 5.4

Table 5.11

JUPITER Pressure Drop Results for BR-2 Calibration Experiment III

	JUPITER Pressure Drop (psi) (Interior Channel(#) ^f)		COBRA * GCFR Results (Ref. 20) psi	CE-III Experiment (Ref.19) psi
	TOMSON Subroutine	Friction Multiplication Factor 2.8(\$)		
Inlet Loss	0.9	0.9	1.0	1.0
Upper Plenum and Upper Axial (28.1 cm)	0.5	0.5	0.6	0.9
Active Core Region Smooth (1 cm)	0.0	0.0	5.5 - δ_1	6.0 - δ_2
Rough (59 cm)	4.1	3.9		
Lower Plenum and Lower Axial (1.4 cm)	0.0	0.0		
Momentum Change	0.9	0.9	δ_1	δ_2
Spacer Grids Smooth (2)	1.3	1.3	1.2	0.7
Rough (3)	2.8	2.8	2.8	2.2
Outlet Loss	0.0	0.0		
TOTALS	10.5	10.3	11.2	11.3

Inlet He-coolant temperature (pressure) = 210.8°C (570.2 psi).
 Heating rate = 9.375 kw/FL
 Temperature rise across active region = 295°C.
 Reynolds number = 7.3×10^4 to 5.3×10^4 .

^f Heat transfer by TOMSON subroutine used for both calculations.

^{\$} Obtained from Figure 10, Reference 19.

Table 5.12

Power Distribution in Different Channels of the
Centrally-Moderated He-Cooled FMSR Core

Radial Zone		Number of Subassemblies	Mid-Core Power Density		160-cm Core Average Axial Power, kw/ft	Axial Peak/Average Power Ratio	Radial Peak/Average Power Ratio	Total Power Produced MW(t)
JUPITER	Physics		kw/ft	W/cm ³				
8	22	12	8.98	320.4	6.67	1.35	1.06	152
11	24	12	9.14	326.0	6.95	1.32	1.02	159
1	19	6	4.01	142.9	2.97	1.35	1.29	34
2	19	6	4.57	162.9	3.41	1.34	1.23	39
3	21	12	6.04	215.5	4.48	1.35	1.22	104
4	20	12	6.88	245.3	5.07	1.36	1.14	117
5	16	6	5.90	210.4	4.29	1.38	1.10	49
6	16	6	6.29	224.4	4.56	1.38	1.10	53
7	17	12	6.87	245.1	4.98	1.38	1.10	114
9	23	12	8.87	316.2	6.63	1.34	1.04	151
10	18	12	7.88	281.1	5.81	1.36	1.07	132
12	25	12	8.92	318.1	6.89	1.30	1.02	160
13	26	12	8.63	307.9	6.72	1.29	1.02	156
14	27	12	7.26	258.8	5.72	1.27	1.11	133
15	28	12	7.11	253.5	5.65	1.26	1.12	132
16	29	12	7.47	266.3	5.98	1.25	1.10	140
17	30	12	7.65	272.9	6.15	1.24	1.09	145
18	31	12	5.78	206.0	4.47	1.29	1.21	110
19	32	12	5.12	182.5	4.14	1.24	1.17	98
20	33	12	5.68	202.6	4.60	1.23	1.21	109
21	15	12	3.61	128.7	2.69	1.34	1.13	63
22	34	12	6.29	224.4	5.10	1.23	1.30	121
23	12,13,14	36	3.31	118.0	2.44	1.35	1.30	168
24	9,10	24	1.31	46.8	0.97	1.35	1.70	46
25	8,11	24	1.33	47.5	0.99	1.34	1.54	49
26	1,2,...7	84	0.37	13.3	0.28	1.32	1.82	46
TOTAL		408						2780

Figure 5.4. JUPITER Channels for the Centrally-Moderated FMSR Core

shows JUPITER channel distributions for the core in a 30° sector. The total power produced in the core plus the axial blankets is 2780 MWt, 89% of which is produced in the fast plus moderated zones. The average power in the 160 cm active core zone for each channel, as well as the radial peaking factors for each channel, are also shown in Table 5.12. The axial power profile for each channel as obtained from the R-Z reactor physics calculations was used in the JUPITER model. The detailed hexagonal physics calculations were used to normalize the power profile at the core mid-plane. The radial peaking factors were taken from the hexagonal reactor physics data. Design data used for the core model are shown in Tables 5.6 and 5.7. Twelve spacer grids were used in the present design, of which only two of these were located in the roughened fuel pin region.

5.3.4 FMSR Thermal-Hydraulic Calculations

The primary objective of this study was to optimize the performance of the plant, i.e., to maximize core outlet temperature and fuel volume fraction while minimizing core pressure drop and pumping power. The operating conditions also had to be consistent with the fuels and materials requirements. Using these criteria as a basis, a series of parametric calculations were made, including trade-off studies, to determine the length and degree of fuel pin roughening required in order to meet fuel and cladding temperature requirements as well as the number of pins per subassembly. These were discussed in detail in our initial report (see Table 5.4 of Reference 1), and hence, will not be repeated here. For the present design, it is found that 50 cm of the fuel pin should be roughened for the regular fuel pin design (helium-bonded). If a metal-bonded fuel pin can be used, roughening may be eliminated.

A study was made to investigate the effect of inlet pressure on the thermo-hydraulic performance of the FMSR core. It is found that the inlet pressure

cannot be reduced significantly from the present 2000 psi design because the subassembly pressure drop and the pumping power would increase as the inlet pressure is decreased. This is discussed further in Appendix E.

A great deal of attention has been directed toward determining the effects of the hot spot factors and radial power peaking on the performance of the FMSR. Application of these factors presents a severe constraint on the design of the He-cooled FMSR and results in a lower mixed mean coolant outlet temperature. However, it should be noted that the present calculations are based on one-dimensional models, and the results will probably improve when two-dimensional effects are taken into account.

The thermal-hydraulic analysis of the FMSR core was done using detailed friction factor and heat transfer coefficients available in the TOMSON subroutine option in JUPITER. This option explicitly treats the cladding roughness design and determines the heat transfer and friction factor coefficients for each axial node from the corresponding temperature and velocity profiles. In the present design, a rib-height, HR, of 0.01 cm was used.

The spacer grid pressure drop calculations were done using Rehme's separate correlations for the grids in the smooth and in the rough regions. Details of these formulas are given in Appendix D. These improved correlations and the reduction in the number of spacer grids from 15 to 12 have resulted in a lower spacer grid pressure loss in these designs compared to the initial design.

Since JUPITER performs only one-dimensional calculations, coolant flow rates were first determined for the hottest pin in each channel such that the fuel and cladding materials constraints were satisfied. In these calculations, the powers in each channel were multiplied by the corresponding radial peaking factors; the hot spot factors were also taken into account. These flow rates

were than fixed to obtain the average channel results without the radial power peaking factors.

5.3.4.1 Results for the Helium-Bonded Fuel Pin Design

This fuel pin design has no fuel-cladding gap. A metal barrier of vanadium is used to prevent interdiffusion of fuel-cladding materials during normal operation and to prevent eutectic formation of metals even under severe transient conditions. The temperature drop across the fuel-cladding gap varies from approximately seventy to a few degrees centigrade, depending on the amount of heat being transferred at the particular location under the assumed contact conduction. The thermal-hydraulic design is determined so that the maximum hot spot inner cladding surface temperature is always kept below 625°C . This imposes severe constraints on the low power subassemblies, since these subassemblies usually have rather large radial power gradients. The average fuel pin in these subassemblies usually operates at a lower fuel temperature. The mixed mean coolant outlet temperature for this design is 512°C , which is 212°C higher than the inlet gas temperature. Figure 5.5 shows the axial temperature profiles for fuel centerline, inner and outer cladding surfaces and the bulk coolant for the hottest pin.

The pressure drop developed across the fuel subassembly was calculated from the sum of friction and acceleration pressure drops plus pressure drops due to spacer grids and orifices. The contributions for the increased friction factors resulting from fuel pin roughening were included in the pressure drop calculations. Details of the pressure drop results are shown in Table 5.13 for the peak power channel. The total pressure drop across a fuel subassembly is 96 psi, of which 52 psi is frictional pressure drop in the active core region and

Figure 5.5. JUPITER Prediction of Axial Temperature Profile

Table 5.13

Pressure Drop Description for the Peak Channel
of the He-Cooled FMSR

	<u>Pressure Drop</u> <u>(psi)</u>
Inlet Loss	5
Upper Plenum & Upper Axial	7
Active Core Region (160 cm) (50 cm roughened)	52
Lower Plenum & Lower Axial	9
Momentum Change	4
Spacer Grids	
Smooth Zone	12
Roughened Zone	4
Outlet Loss	3
Total	96

Orifice Loss Coefficient = 0.00

Core Pumping Power Needed for 12 Subassemblies = 9.45 MW

16 psi for the twelve spacer grids used in this study. There is no orifice pressure drop for the peak channel. The orifices for the other channels are adjusted to maintain a uniform pressure drop across the reactor core. The pumping power required for the core is 174 MW.

The essential thermal-hydraulic characteristics of the He-cooled FMSR core are summarized in Table 5.14 and compared to those of the 300-MW(e) GCFR design. The FMSR has certain distinct characteristics. It has a higher operating pressure and a higher pressure drop across the core. However, the $\Delta p/p$ for the FMSR (4.4%) is reasonable compared to that of the GCFR (3.4%). The coolant flow rates per unit coolant area are higher for the FMSR compared to those of the GCFR, resulting in a higher pumping power requirement. The Reynolds numbers in the fast zones of the FMSR vary from 8×10^4 to 1.6×10^5 . The average heat transfer coefficient used for the peak channel was $\sim 1900 \text{ Btu/hr/ft}^2 \text{ } ^\circ\text{F}$ for the smooth part of the pin and $\sim 3500 \text{ Btu/hr/ft}^2 \text{ } ^\circ\text{F}$ for the rough regions. Although the numbers are slightly different, the essential characteristics of the FMSR core design have not changed from the reference except for the mixed mean coolant outlet temperature. The reduction in mixed mean coolant outlet temperature is mainly due to the large radial power peaking in the outer zones and the constraint of the hot spot cladding inner surface temperature. If the inner cladding surface temperature can be increased, to say 640°C , the mixed mean coolant outlet temperature will increase to about 525°C . It should be remembered that the present design represents only one stage in the iterative process of achieving a final FMSR design; this work only indicates that further iterations with updated physics calculations are warranted.

Table 5.14

He-Cooled FMSR Thermal-Hydraulic Characteristics
and Comparison to GCFR Values

	<u>FMSR</u>	<u>GCFR</u>
Heat Output from Core, MW(t)	2780	826
System Pressure, (psi)	2000	1230
Reactor Pressure Drop, psi	96	40
Inlet Coolant Temperature, °C	300	312
Mixed Mean Core Outlet Temperature, °C	512	540
Maximum Axial Average Heating Rate, kw/ft	6.95	10.41
Coolant Flow Through Fuel Element, kg/sec		
Central Zone (Fast and Moderated)	2187	
Outer Zone	330	
Total	2517	~650
Maximum Inner Cladding Temperature, °C	625	~800
Maximum Fuel Centerline Temperature, °C	850	~2100
Core Heat Transfer Surface Area, m ²		
Central Zone (Fast and Moderated)	2878	
Outer Zone	2015	
Total	4893	
Average Peak Channel Surface Heat Transfer Coefficient, Btu/hr-ft ² - °F		
Rough Surface	3500	2400 (Avg.)
Smooth Surface	2000	
Maximum Reynolds Number	1.65×10^5	$\sim 1 \times 10^5$

5.3.4.2 FMSR Design with Liquid-Metal-Bonded Fuel Pins

If a suitable liquid-metal bond can be used for the helium-cooled FMSR fuel pins, the fuel centerline temperature limit can be more readily satisfied. However, the 625°C cladding inner surface temperature limit will still restrict the coolant outlet temperature. A design can be accomplished with either smooth or partially roughened fuel pins. Each of these two designs will have some advantages relative to the other. The smooth fuel pin design will have a lower subassembly pressure drop, and correspondingly, a lower pumping power requirement, but relatively higher fuel centerline temperatures. The roughened fuel pin design will have a subassembly pressure drop and pumping powers similar to the nonliquid-metal-bonded design, but with substantially lower fuel centerline temperatures. The maximum fuel centerline hot spot temperature is 830°C for the smooth pin design and only 790°C for the roughened pin design. The maximum cladding inner surface temperature is 625°C for both of these designs. Subassembly pressure drops for the two designs are 74 and 94 psi, respectively, and are shown in Table 5.15. The essential thermal-hydraulic characteristics of these two designs are given in Table 5.16. The pumping power requirements for the two designs are 137 and 172 MW, respectively. The conclusion to be drawn from these comparisons is that a metal-bonded fuel can significantly reduce the fuel centerline temperature. Additional iterations with reactor physics calculations should be made to achieve an optimum thermal-hydraulic design.

Table 5.15

Pressure Drop Description for the Peak Channel
of the Liquid Metal-Bonded He-Cooled FMSR

	<u>Pressure Drop (psi)</u>	
	<u>Smooth</u>	<u>Rough</u>
Inlet Loss	5	5
Upper Plenum & Upper Axial	8	7
Active Core Region (160 cm)	27	51
Lower Plenum & Lower Axial	10	9
Momentum Change	4	3
Spacer Grids		
Smooth Zone	13	12
Roughened Zone	4	4
Outlet Loss	<u>3</u>	<u>3</u>
Total	74	94

Orifice Loss Coefficient = 0.00

Core Pumping Power Needed for 12 Subassemblies = 7 MW Smooth Pins

9 MW Roughened Pins

Table 5.16

He-Cooled FMSR Thermal-Hydraulic Characteristics
of Metal-Bonded Designs

	<u>FMSR Design</u>	
	<u>Smooth Pin</u>	<u>Roughened Pin</u>
Heat Output from Core, MW(t)	2780	2780
System Pressure, psi	2000	2000
Reactor Pressure Drop, psi	74	94
Inlet Coolant Temperature, °C	300	300
Mixed Mean Core Outlet Temperature, °C	508	512
Maximum Axial Average Heating Rate, kw/ft	6.95	6.95
Coolant Flow Through Fuel Element, kg/sec		
Fast Zone	2241	2186
Outer Zone	336	330
Total	2576	2516
Maximum Inner Cladding Temperature, °C	625	625
Maximum Fuel Centerline Temperature, °C	850	790
Average Peak Channel Surface Heat Transfer Coefficient, Btu/hr-ft ² -°F		
Rough Surface	----	3500
Smooth Surface	2050	2000
Maximum Reynolds Number	1.7×10^5	1.6×10^5

6. FUELS AND MATERIALS

One of the objectives of the FMSR design study reported in Section 3 was to reduce the fluence damage to the steel and the fuel burnup. While a number of important gains were achieved as a result of that design study, the fuel burnup level and the integrated fluence damage both remained high. This was in large part due to inclusion of updated fission product cross sections which had the unfortunate effect of cancelling most other design gains in neutronic performance. Studies are now planned on determining the effect of discharging FMSR fuel whenever the burnup or fluence damage reaches some specified level.

One important trend in the FMSR development, as it affects fuels and materials, has been an increased emphasis on the sodium-cooled version of the reactor. Because the fuel for the sodium-cooled FMSR is much closer to that of the well-proven Mark-II fuel of the EBR-II reactor in contemplated fabrication procedure, operating characteristics, etc., the confidence level for success in design and operation of this fuel is proportionately greater than for the proposed sintered fuel for the gas-cooled FMSR.

Since a fuel and materials program for the FMSR would be relatively expensive, particularly when fabrication and irradiation activities begin, little work has been done in this area since the publication of the Initial Feasibility Study. The development of a successful fuel, cladding, and subassembly duct and the demonstration of their performance through fabrication and irradiation remains one of the most critical parts of the proof of feasibility of the FMSR concept. The long lead time between a decision to fabricate and test a particular fuel and the completion of the irradiation testing enhances the importance of early initiation of work in this area.

Some valuable services related to FMSR fuels and materials have been provided by Argonne National Laboratory and the General Atomic Company. These have often consisted of the planning of programs and schedules for work in this area and, also, of providing available data on such items as the change of thermal conductivity of metal fuel with degree of burnup. The Hanford Engineering Development Laboratory has provided information on ferritic steels proposed for FMSR clad and duct. Another important development for the FMSR metal fuel is the renewed interest in metal fuel performance required for the planned safety programs in the EBR-II reactor. Analytical methods for evaluation of both steady-state performance and for operation under off-normal conditions are being developed and will be immediately applicable to the FMSR. The planned experiments in EBR-II and the supporting tests in the TREAT transient test reactor will be particularly useful to the FMSR program.

The use of the EBR-II reactor as a tool for early irradiation testing experiments of proposed FMSR fuel is considered to be a particularly effective procedure for fast, flexible, and relatively inexpensive irradiation of early FMSR fuel designs. This testing will accelerate the availability and enhance the probability of success of more prototypical FMSR fuel which would then be proposed for irradiation in FFTF.

Little additional work on duct dilation under irradiation and hydraulic pressure differentials and mechanical analysis of whole core bowing has been done since the publication of the Interim Report. There is some indication from work at Argonne that HT-9 steel might perform considerably more successfully as duct material than the D-57 ferritic steel previously evaluated. Since reactor physics, thermal hydraulics, and cost studies are tending to converge to a FMSR fuel design, the validity of current assumptions concerning

dimensions of the duct and the clearances between the ducts are becoming critical items of uncertainty which will impede further progress unless they are resolved or reduced.

The nature and the specification of the moderator for the FMSR has not yet been determined. The moderator at the center of the centrally-moderated FMSR may in the future serve both control and moderator functions. For this reason, the design may result in large diameter rods, perhaps of sintered beryllium metal. Beryllium oxide may be the most desirable moderator material for the FMSR in terms of fabricability and long-term irradiation stability. Beryllium oxide would be compatible with neutron slowing-down requirements. Graphite has been under consideration primarily because of its low cost and fabricability. For the sodium-cooled FMSR, however, there is always the concern for the consequences for interaction between sodium and graphite, in which the sodium lodges between planes of the graphite crystal and induces severe swelling.

7. SAFETY OF THE FMSR

7.1 INTRODUCTION

Due to funding limitations, very little new work has been performed in the area of FMSR safety during this period, for which the primary emphasis was on evaluation of the potential of a major new FMSR design (the centrally-moderated FMSR), the switch in design emphasis from helium cooling to sodium cooling, and careful evaluation of the thermal-hydraulic performance potential of the design. It is most productive to have a reasonably stable base design before a major effort is launched in the area of FMSR safety. A similar argument would apply for the potential usefulness of a zero power critical experiment program.

The previously stated general safety characteristics of the sodium-cooled and helium-cooled FMSR concepts are summarized in the following sections.

7.2 FMSR SAFETY CHARACTERISTICS COMMON FOR BOTH FMSR COOLANT TYPES

1. The potential for nuclear supercriticality and recriticality due to fuel slumping or compaction is considerably reduced because of the low fissile fraction of the FMSR fuel and because of the tendency of molten metal fuel to form an inseparable metal eutectic with the steel of the clad, wrapper wire and duct. The axial distribution of the plutonium (peaked at core mid-height in the FMSR) is optimum from a criticality point of view in steady state, as it was built up during operation; any slumping which moves the more enriched fuel onto or past lesser enriched fuel is likely to lead to a loss in reactivity.
2. The Doppler effect feedback coefficient tends to be sufficiently large.

3. Metal fuel is often attributed to have excellent dispersive, shutdown characteristics for severe transients. This is due to inclusions in the pores of the low density metal fuel of pockets of fission product gases, and sodium as well as cesium, which can generate dispersive pressures at low fuel temperatures. Low fuel temperatures and gas blanketing are believed to inhibit strong fuel-coolant interactions as well for the sodium-cooled FMSR. These properties must be proven by carefully defined tests, and particularly for FMSR fuel designs, but these potentials for safety advantage are very attractive.
4. The "Post-Accident Heat Removal" (PAHR) characterization of a FMSR reactor, if the reactor were to suffer a hypothetical accident, could be particularly advantageous. The relatively modest power density of the FMSR fuel would ensure a correspondingly low decay heat volumetric rate since it is primarily recent power density and not the degree of burnup which controls decay heat power density. The decay power density is further reduced as the molten fuel integrates all of the steel of the core region to form U_6Fe , with a melting point of $815^{\circ}C$ and to tend to solidify as it is further diluted with steel as it tends to form the compound UF_{e2} with a higher melting point of $1080^{\circ}C$ as it might melt into the below-core shielding which protects the grid plate. The recriticality potential is correspondingly reduced by this significant dilution of the fuel with steel, as already stated.
5. The time required for neutron slowing down in the moderated region of the centrally-moderated FMSR could increase the prompt neutron lifetime of the whole core. This increased lifetime would have an

automatic consequence of reducing the rate of power generation at the peak of a severe transient, if such a transient should somehow ever occur. There would be, therefore, a built-in automatic safety feature which does not depend on any instrument, valve, operator, etc., for its activation. The existence of this lifetime and its consequences must be demonstrated by analyses.

7.3 SODIUM-COOLED FMSR SAFETY CHARACTERISTICS

1. In addition to the nuclear shutdown potential of metal fuel, which has already been described, a potential advantage is claimed for the molten fuel if injected into the coolant stream. Blanketing of this fuel by the essentially free fission gases from the large pores in the fuel and by vapor of sodium and cesium inclusions should inhibit any rapid or violent fuel-coolant interaction. This claim should also be tested by simple and relatively easily performed experiments.
2. There is a possibility in some fast reactor designs of going prompt critical during a hypothetical loss of flow accident due to reactivity addition which results if molten cladding is carried out of the core after the onset of "flooding" (the development of a high-friction wave surface) on the molten steel. For a metal-fueled reactor, the molten cladding is quite likely to interact very rapidly with the metal fuel to form an inseparable eutectic. From that point on, any achievement of the flooding condition should result in a loss of reactivity as the U₆Fe eutectic is swept out of the core region. Again, a well planned experiment would be essential. (The eutectic forms rapidly at temperatures above 1040°C if a diffusion barrier of vanadium is used between the fuel and clad, and at significantly

lower temperatures if the vanadium barrier is not present. The melting point of stainless steel is approximately 1400°C, and that of the uranium-plutonium fuel is approximately 1050°C and less with fission product buildup.)

3. Because of the differences in power densities in the various shuffle zones of the FMSR and because of changes in some power densities over the fuel cycle, while the orificing remains fixed, any sodium-voiding accident in a FMSR is very likely to be strongly incoherent in its behavior. This degree of incoherence could be important, and perhaps more important than the magnitude of the total sodium void worth of the reactor, in determining the rate of approach to prompt critical during a hypothetical accident. A modest rate of approach to prompt critical for a FMSR reactor, which has a strong dispersive mechanism as well as a strong Doppler feedback coefficient and a poor conversion to mechanical work, could result in a particularly mild loss of flow accident.
4. The change in reactivity on sodium voiding is often described in terms of a spectral component and a leakage component. The FMSR appears to have an added safety advantage in that, for the moderated zones of this reactor, there is a new leakage component which strongly reduces the local sodium void reactivity change. This is that there appears to be an enhanced local radial leakage of neutrons to the moderator upon local sodium voiding, followed by an enhanced return of lower energy neutrons. Both processes tend to lower the local sodium void reactivity contribution. A similar process occurs for breeder reactors which have well-designed heterogeneous cores.

7.4 GAS-COOLED FMSR SAFETY CONSIDERATIONS

1. The gas-cooled FMSR shares with the GCFR the generic characteristic that even a very severe disassembly accident will ordinarily result in almost no work energy damage to the reactor vessel (PCRVR) or to the plug. Care must be taken that pressures developed in the core should not be transmitted to the reactor plug via control rod guide tubes, instrument guide tubes, etc., but this should not be difficult.
2. A second generic characteristic is the need for particular care in post-shutdown heat removal and in post-accident heat removal. A gas-cooled FMSR would employ the same up-flow cooling design now envisioned by GA for the GCFR. GA designs are now directed toward assurance of convective heat removal with existence of no pumping capability whatever. Since the FMSR employs a higher fuel volume fraction than the GCFR and a correspondingly lower coolant volume fraction, full cooling of a shutdown FMSR by convective cooling may be more difficult to achieve. The potential advantages of FMSR metal fuel for the PAHR regime have been discussed previously. The potentially low temperature of the molten fuel, and its higher thermal conductivity, could lead to a simpler and cheaper core retention design.
3. For a FMSR such as the centrally-moderated FMSR, the moderated and the nonmoderated fuel zones are strongly coupled neutronically and have no decoupling regions between them. If the FMSR continues to possess geometrical configurations similar to those of the present design, there should be very little reason to evaluate reactor kinetics problems of strongly decoupled fast and moderated regions.

8. CONCLUSIONS

The work reported in this progress report has led to the following series of conclusions with respect to the FMSR concept:

- The centrally-moderated FMSR design offers significantly improved performance over that of the previous Reference FMSR Design. It should now be considered as the new Reference Design from which further improvements will evolve.
- The FMSR design for which limited and controlled fuel reprocessing is allowed appears to have excellent performance, cost, thermal hydraulic, materials requirements and safety potential. It should be pursued further.
- An all-fast FMSR reactor design was shown to have somewhat better breeding performance than the Reference FMSR, as expected. The thermal-hydraulic performance, however, was considerably degraded at the outer core radius. This could have a significant negative impact on the thermodynamic efficiency of the plant.
- The methods used in the thermal-hydraulic evaluation of both the sodium-cooled and helium-cooled (CM)FMSR design were considerably improved in 1979. New analyses which include hot spot, hot pin, edge pin, corner pin and sensitivity studies continue to support the validity of the FMSR design.
- Progress in the reactor physics area provided physics designs for coupling to thermal-hydraulic studies. Intercomparison studies with other laboratories provided confidence in the analyses. Work on fission

product cross sections has been undertaken as a result of these studies.

These data will be of general use to basic LMFBR technology.

- Only limited work was performed in the important areas of safety and fuels and materials during this year because of funding limitations.
- Future work will include more detailed analyses of the resource advantages of the FMSR. Work will include a definition of the startup cores and the approach to the equilibrium core. The FMSR design which allows restricted reprocessing has exceptional potential and will be pursued further. The fuels and materials aspects of the concept will receive more attention. Critical items such as fabrication to FMSR specifications and irradiation to FMSR conditions will receive high priority. A critical experiment program will be initiated with the initial emphasis on planning.

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APPENDIX A

Thermal-Hydraulic Correlations for the Na-Cooled FMSR

The required thermophysical properties for fuel and sodium, and the correlations of friction coefficient and heat transfer are provided in the SATURN code. The approach is based on providing the best available data represented in the convenient form of empirical equations.

For the Mark-II type metal fuel, ANL⁽¹⁾ has shown that the thermal conductivity of the fuel is reduced by the generation of fission products. For a given composition of the uranium-fissium alloy, the thermal conductivity increases with temperature. The general empirical form of the thermal conductivity is

$$K = a_0 + a_1 T + a_2 T^2$$

where K is in cal/S-cm-C and T in °C. The coefficients a's depend on the fission product content as given below:

Fission Product Content, %	a_0	a_1	a_2
0	0.0679	9.215×10^{-7}	7.543×10^{-8}
3	0.0535	1.764×10^{-5}	6.571×10^{-8}
5	0.0435	3.048×10^{-5}	5.867×10^{-8}
8	0.0375	3.230×10^{-5}	5.091×10^{-8}
10	0.0331	4.673×10^{-5}	1.953×10^{-8}

In addition, the thermal conductivity of the fuel is reduced by the presence of fuel porosity. According to Reference (2), experimental measurements have indicated that the reduction factor for the fuel conductivity is independent of temperature and depends on the porosity only. The effect of porosity on thermal conductivity follows the relationship:

$$\frac{K_p}{K_{100}} = \frac{1-P}{1+1.7 P} \quad (2)$$

where P is the porosity and K_{100} is the thermal conductivity of the solid fuel.

All of the required thermophysical properties for liquid sodium are taken from Golden and Tokar.(3) These are noted in the following.

Thermal Conductivity (W/mK)

The thermal conductivity of liquid sodium is given by the following equation:

$$k(T) = k_0 + K_1 T + k_2 T^2 \quad (3)$$

here

$$\begin{aligned} k_0 &= 109.7, \\ k_1 &= -6.4499 \times 10^{-2}, \\ k_2 &= 1.1728 \times 10^{-5} . \end{aligned}$$

Specific Heat Capacity (J/kg K)

The specific heat capacity at constant pressure for liquid sodium is given by

$$C_p(T) = c_0 + c_1 T + c_2 T^2 , \quad (4)$$

where

$$\begin{aligned} c_0 &= 1630.22 \\ c_1 &= -0.83354 \\ c_2 &= 4.62838 \times 10^{-4} \end{aligned}$$

Enthalpy (J/kg)

The enthalpy of saturated liquid sodium is given by

$$h_s(T) = h_0 + h_1T + h_2T^2 + h_3T^3, \quad (5)$$

where

$$h_0 = -6.7511 \times 10^4,$$

$$h_1 = 1630.22,$$

$$h_2 = -0.41674,$$

$$h_3 = 1.54279 \times 10^{-4}.$$

In solving the energy equation, it is necessary to compute the temperature of the sodium from its enthalpy value. It is done by inverting Equation (5); an approximate correlation was obtained:

$$T = c_0 + c_1h_s + c_2h_s^2 + c_3h_s^3 + c_4h_s^4, \quad (6)$$

where

$$c_0 = 55.5057,$$

$$c_1 = 5.56961 \times 10^{-4},$$

$$c_2 = 2.17341 \times 10^{-10},$$

$$c_3 = -7.27069 \times 10^{-17},$$

and

$$c_4 = 4.41118 \times 10^{-24}.$$

The saturation temperature of liquid sodium as a function of pressure is given by the following equation:

$$T_s = \frac{c_0}{\ln(9.869 \times 10^{-6}p) - c_1} \quad (7)$$

where

$$c_0 = -12130.0,$$

$$c_1 = 10.51,$$

and p is the pressure in N/m^2 .

Density (kg/m³)

The density of saturated liquid sodium is given by the following relation:

$$\rho_s(T) = \rho_0 + \rho_1 T + \rho_2 T^2 + \rho_3 T^3 \quad \text{for } T_1 \leq T \leq T_2 \quad (8)$$

where,

$$\rho_0 = 1011.597 ,$$

$$\rho_1 = -0.22051 ,$$

$$\rho_2 = -1.92243 \times 10^{-5} ,$$

$$\rho_3 = 5.63769 \times 10^{-9} ,$$

$$T_1 = 370.9 \text{ } ^\circ\text{K} ,$$

and

$$T_2 = 1644.2 \text{ } ^\circ\text{K} .$$

Dynamic Viscosity (μ , i.e., N s/m²)

The dynamic viscosity of liquid sodium is represented as

$$\log_{10} \eta = c_1 + \frac{c_2}{T} + c_3 \log_{10} T , \quad (9)$$

where

$$c_1 = -2.4892 ,$$

$$c_2 = 220.65 ,$$

and

$$c_3 = -0.4925 .$$

The liquid metal heat transfer correlations for forced convection in a rod bundle have been developed by West, Schad, Graber and Rieger, and Borishanskii. These correlations have been compared with the experimental data by Kazimi.⁽⁴⁾ In view of the comparison, the modified Schad correlation is selected. The following correlation is valid for $Pe > 10$, i.e., for turbulent flow regimes:

$$Nu = [h_1 + h_2(P/D) + h_3(P/D)^2] \cdot (a + b Pe^c) , \quad (10)$$

where

$$h_1 = -16.15 ,$$

$$h_2 = 24.96 ,$$

$$h_3 = -8.55 ,$$

and

$$a = 0 \quad b = 1, \quad c = 0.3 \quad \text{for } Pe \geq 150 ,$$

$$a = 4.496, \quad b = 0, \quad c = 0.3 \quad \text{for } Pe < 150 .$$

The friction coefficient is computed from the following correlation developed by Novendstern⁽⁵⁾ for wire-wrapped fuel rod bundles in the turbulent flow regime:

$$f = \left[\frac{1.034}{(P/D)^{0.124}} + \frac{29.7 (P/D)^{6.94} Re^{0.086}}{(PW/D)^{2.239}} \right]^{0.885} (0.316 Re^{-0.25}) \quad (11)$$

where P/D is the pitch-to-diameter ratio of the bundle and PW/D is the ratio of the wire-wrap pitch to rod diameter.

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APPENDIX B

Hot Channel Modeling of the Sodium-Cooled FMSR

There are six hot spot factors representing the statistical and other uncertainties of theoretical and experimental analyses, instrumentation accuracy, manufacturing tolerances, physical properties and correlations.⁽¹⁾ They are defined as:

- FFUEL = Fuel hot spot factor. It indicates the uncertainty of fuel thermal conductivity due to burnup, variation of composition, porosity, etc.
- FQ = Heat flux hot spot factor. It represents the uncertainties of power level measurement, nuclear data and fissile fuel maldistribution.
- FGAP = Gap hot spot factor. It indicates the uncertainty of gap conductance.
- FCLAD = Clad hot spot factor. It includes the uncertainties of clad thickness and thermal conductivity.
- FFILM = Film hot spot factor. It represents the uncertainties of subchannel flow area, film heat transfer coefficient and assembly flow maldistribution. This factor is applied to the local heat transfer coefficient to increase the film temperature drop.
- FCOOL = Coolant hot spot factor. This factor represents the direct uncertainties of inlet flow maldistribution and assembly flow maldistribution, and the statistical uncertainties of wire-wrap orientation, subchannel flow area and coolant properties.

In addition, there are two more factors which take into account the nonuniform distributions of power and flow. They are defined as:

FRPIN = Fuel pin radial power factor. It represents the radial power gradient in fuel subassembly.

FMDFL = Assembly flow maldistribution factor. This factor indicates the flow maldistribution in the interior and edge channels of the subassembly which is caused by the turbulent mixing, wire-wrap orientation and variation of flow area, etc.

Following the computational procedures given in Reference 2, the hot channel factors are included in the basic equations. For the hot channel, the equations become:

a) Heat conduction in fuel pin:

$$\left(\frac{K}{FFUEL} \right) \frac{1}{r} \frac{d}{dr} \left(r \frac{dt}{dr} \right) + (q \cdot FQ \cdot FRPIN) = 0$$

where k and q are the thermal conductivity and power density of fuel pin; $FFUEL$, FQ and $FRPIN$ are the fuel hot spot factor, heat flux factor and radial power factor, respectively.

b) Heat transfer at fuel-clad interface:

$$(q \cdot FQ \cdot FRPIN) = \frac{h_g}{FGAP} (t_{f,g} - t_{c,g}) / R_f$$

where $t_{f,g}$ and $t_{c,g}$ are fuel and clad temperature at the interface, h_g is the gap conductance, $FGAP$ the gap hot spot factor, and R_f the fuel radius.

c) Heat conduction in clad:

$$\left(\frac{K}{FCLAD} \right) \frac{1}{r} \frac{d}{dr} \left(r \frac{dt}{dr} \right) = 0$$

where K and $FCLAD$ are the thermal conductivity of clad and the clad hot spot factor, respectively.

d) Heat transfer at clad-coolant interface:

$$(q \cdot FQ \cdot FRPIN) = \frac{h}{FFILM} (t_{cd} - t_{cl}) / R_c$$

where t_{cd} , t_{cl} and R are clad outside surface temperature, coolant temperature and clad outside radius, respectively; $FFILM$ is the film hot spot factor.

e) Coolant enthalpy rise:

$$\frac{dE}{dZ} = \frac{(q \cdot FQ \cdot FRPIN)(1 + FRACQ) A_{fuel}}{G A_{cool}} FC00L$$

where A_{fuel} and A_{cool} are the cross-sectional area of fuel pin and coolant, respectively; G is the average flow rate, and $FC00L$ the coolant hot spot factor; the term $FRACQ$ represents the fraction of structure to coolant heating rate.

For the reference case in the preliminary design study, the following are the values of these factors used in the computations:

FC00L = 1.25	FFUEL = 1.10
FFILM = 1.16	FQ = 1.11
FCLAD = 1.15	FRPIN = 1.15
FGAP = 1.10	FMDFL = 1.08

In general, a reasonable estimate of hot channel factors requires the complete knowledge of the reactor design and the interaction of the loop system. Since such knowledge is not available for the FMSR, the hot channel factors in this preliminary study are based largely on those used for the CRBRP.⁽¹⁾ They are for non-vented oxide fuels with stainless clad and for a burnup to about 60,000 MWD/T. The modification of these values to the present FMSR fuel, vented metal fuel, is a rough approximation.

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APPENDIX C
Spacer Grid Pressure Drop Calculation

Pressure drop calculations for the spacer grids have been refined because of their importance. This appendix will summarize the correlations used for the spacer grid pressure drop calculations and discuss FMSR spacer grid locations.

JUPITER has two options for spacer grid pressure drop computations. The first option, obtained by setting MGR=0, uses the old BNL formula⁽¹⁾ and will be referred to as the Thompson correlation.⁽²⁾ In this case, the pressure drop across a grid is given by

$$\Delta P = \lambda * \frac{s}{(1-s)^2} * \frac{G^2}{2\rho} ,$$

where

λ = empirical loss coefficient = 1.373,

G = coolant flow rate (gm/cm²/sec),

$\bar{\rho}$ = average coolant density (gm/cm³), and

s = fraction of free flow area blocked by the spacer grid.

This formula has the correct boundary conditions; namely, $\Delta P = 0$ for no blockage and $\Delta P = \infty$ for complete blockage.

The second option uses Rehme's formula.⁽³⁾ MGR is 1 for this case. In this method, the loss coefficient depends on Reynolds number, and also, on the location of the spacer grids:

$$\Delta P = c_v s^2 \frac{G^2}{2\rho}$$

where

$$c_v = 5.14 + 828.0/(\text{Re})^{0.5} \quad \text{in the smooth region}$$

$$= 8.82 + 761.0/(\text{Re})^{0.5} \quad \text{in the rough region.}$$

These correlations,⁽⁴⁾ obtained from GA, are valid for $Re > 1000$ and for $0.1 < s < 0.45$.

The implications of these two correlations are shown in Table 1C which gives the spacer grid pressure drops for an earlier FMSR design. It is found that the Thompson correlation produces pressure drops which are 5% higher than Rehme's correlation results for the spacer grids in the roughened zone. Thus, all our previous work effectively treated all spacer grids as if these are located in the roughened region, and hence, predicted higher pressure drops. The pressure drops predicted by Rehme's correlation for the smooth region are about 36% lower than those predicted by the Thompson correlation. Thus, it is necessary for the FMSR design to precisely define the number of spacer grids in the smooth and in the roughened regions.

Our present ideas on the number of spacer grids to be used for the FMSR and their approximate locations are shown in Figure 1C. The design of the spacer grids has not been changed; the fraction of coolant area blocked is 0.179. The 300-MW(e) GCFR Demo uses spacer grids at an approximate interval of 22 cm⁽⁵⁾ along the active part of the fuel element. The 12-rod bundle BR-2 experiment for thermal-hydraulic studies of GCFR fuel elements uses spacer grids at 20 cm intervals.⁽⁶⁾ A similar spacing for grids has been used in the FMSR design. As shown in Figure 1C, the Reference Design will have only two spacers in the roughened section of the fuel element, which is about 60 cm long, and ten additional spacers in the smooth sections.

The present investigation has thus reduced the spacer grid pressure drop for the FMSR by reducing the total number of spacer grids from 15 to 12, and by refining the calculations. Further pressure drop reduction is possible by improving the design of the spacer grids.

Table 1C

Spacer Grid Pressure Drop Parametric Results
for an Earlier FMSR Design*

JUPITER Option (MGR)	Number of Spacers			Pressure Drop (psi)		
	Smooth	Rough	Total	Nominal [†]	Hot Spot	Revised ^{\$}
1 Rehme's Correlation	10	2	12	15.01 (11.58 + 3.42)	17.51 (13.50 + 4.01)	
2 Thompson's Correlation			12 15	21.57 26.96	25.59 31.86	

*331 pins/subassembly, He inlet pressure (temperature) 2000 psi (300°C),
 50 cm fuel pin roughened.

[†]Peak channel coolant outlet temperature 545.0°C.

^{\$}Peak channel coolant outlet temperature 524.7°C.

Figure 1C. Approximate Locations for the FMSR Spacer Grids .

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APPENDIX D

Hot Spot Calculations in JUPITER

To consider the effect of random and nonrandom uncertainties such as a change in geometry from nominal dimensions, or uncertainties in the thermal-hydraulic correlations, hot spot factors have been included in JUPITER. Five hot spot factors are defined. These are channel hot spot factor (FCH), film hot spot factor (FLM), cladding hot spot factor (FCL), gap hot spot factor (FGP), and fuel hot spot factor (FFL). These hot spot factors should include all possible uncertainties that affect the nominal temperatures. It does not include new physical phenomena not presently considered in nominal calculations such as the effect of radial power profile in the subassembly or the edge pin (flow maldistribution) effects. FCH, FLM and FCL hot spot factors for the GCFR fuel assembly have been obtained from GA⁽¹⁾ and are shown in the text in Table 5.9. GCFR design is mostly concerned with the mid-clad hot spot temperature, whereas a major concern for the FMSR is the fuel centerline temperature. Thus, FGP and FFL are more important in FMSR than for the GCFR. Now, FMSR fuel proposed for the gas-cooled FMSR is rather different from the GCFR fuel, as well as from the metal fuel proposed for the sodium-cooled FMSR, and no manufacturing experience exists for this fuel. Thus, engineering judgments have to be used in determining these factors.

The mid-clad and the clad inner surface hot spot temperatures are calculated by first determining the locations and the maximum nominal temperatures for these two quantities. For each of these two, the following formula is used:

$$\text{Hot Spot Temperature} = T_{IN} + (\Delta T)_{ch} * FCH + (\Delta T)_{film} * FLM + (\Delta T)_{cl} * FCL$$

where

T_{IN} = coolant inlet temperature,

$(\Delta T)_{ch}$ = nominal coolant temperature rise at the location of hot spot,

$(\Delta T)_{film}$ = nominal film temperature drop at the location of hot spot,

and

$(\Delta T)_{cl}$ = nominal temperature drop from surface to cladding midwall/inner surface at the location of hot spot.

The fuel centerline hot spot temperature is determined at present as an average of linear and mean square root hot spot temperatures. The hot spot temperatures, according to the linear and mean square root formulas, are also printed. Thus,

$$\text{Fuel centerline hot spot temperature} = (TFHOT1 + TFHOT2)/2.0,$$

where

$$TFHOT1 = TFHOT + (\Delta T)_{gap} * FGP + (\Delta T)_{fuel} * FFL$$

$$TFHOT2 = TFHOT + (\Delta T)_{gc} + \sqrt{\left\{ (\Delta T)_{gap}^* (FGP-1.0) \right\}^2 + \left\{ (\Delta T)_{fuel}^* (FFL-1.0) \right\}^2}$$

$$TFHOT = TIN + (\Delta T)_{ch} * FCH + (\Delta T)_{film} * FLM + (\Delta T)_{cl} * FCL.$$

The temperature differentials $(\Delta T)_{ch}$, $(\Delta T)_{film}$ and $(\Delta T)_{cl}$ are calculated at the location of nominal maximum fuel centerline temperature. Further,

$(\Delta T)_{gap}$ = temperature drop across the fuel-cladding gap at the location of maximum nominal fuel centerline temperature,

$(\Delta T)_{fuel}$ = temperature drop from fuel centerline to fuel pellet surface at the location of hot spot,

and

$(\Delta T)_{gc}$ = temperature drop from fuel centerline to clad inner surface at the location of hot spot.

The hot spot locations (height and axial node) and the nominal and hot spot temperatures are printed in the JUPITER output.

After the hot spot temperatures are calculated, the code checks if the hot spot cladding inner surface temperature exceeds the user-given maximum allowed cladding inner surface temperature (TCDMAX). If TCDMAX is exceeded, the code automatically goes through an iteration procedure of increasing the coolant flow rate (everything else, e.g., power, geometry, etc. remaining the same) to achieve this allowed hot spot temperature. The code then checks if the fuel centerline temperature criterion is met; if the maximum allowed fuel centerline hot spot temperature (TCLMAX) is exceeded, the code will further adjust the coolant flow rate to satisfy the maximum allowed fuel centerline temperature. The corresponding pumping power and temperature calculations are also done for the hot spot adjusted case.

Finally, it should be emphasized that the present hot spot analysis neglects other hot channel factors, namely, inlet flow maldistribution, intra-subassembly flow maldistribution, interchannel mixing, heat transfer and flow uncertainties near spacer grids, etc.

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APPENDIX E

Reduction of Inlet Pressure for the He-Cooled FMSR

An analysis was carried out to determine if the inlet pressure for the He-cooled FMSR can be reduced from the present value of 2000 psi. It is found that if the core inlet and outlet coolant temperatures are kept fixed, the pressure drop across the fuel pin subassembly will increase as the inlet pressure is decreased. The reason for this is that the density of helium decreases linearly with pressure (perfect gas law assumed), and hence, helium at higher velocity must be pumped through the core in order to remove the same amount of heat. Thus, the values for $\Delta P/P$ go up as the inlet pressure is reduced. These variations are displayed in Figure 1E. The analyses performed in this appendix utilize the data from an earlier FMSR design. It should be noted that the thermal characteristics of the FMSR core are not changed in these studies. Figure 2E shows the variation of pumping power and coolant velocity as a function of inlet pressure. Note that the gas velocity and the pumping power increase rather rapidly as the inlet pressure is decreased.

The $\Delta P/P$ values for the 300-MW(e) GCFR Demo are about 0.035. This is a controlling factor for GCFRs because it determines the circulator demand for the reactor. The maximum $\Delta P/P$ that can be used for the FMSR is not known at present.

The present study indicates that the inlet pressure can be reduced, but perhaps not lower than 1750 psi. It is suggested that the next reference design should use 1800 psi as the inlet pressure. This will, then, somewhat increase the core pressure drop and pumping power, but the ambient pressure would then be very close to that used in current European GCFR design.

Figure 1E. Subassembly Pressure Drop Vs Inlet Pressure.

Figure 2E. Core Pumping Power Vs. Inlet Pressure