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THE RANDOMLY PACKED SETTLED BED FAST REACTOR CONCEPT
[1000-MW(e) REACTOR DESIGN]

L. GREEN AND M.M. LEVINE, Editors

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

June 1, 1965

BROOKHAVEN NATIONAL LABORATORY

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THE RANDOMLY PACKED SETTLED BED FAST REACTOR CONCEPT

[1000-MW(e) REACTOR DESIGN]

Introduction

The Settled Bed Fast Reactor (SBFR) is a breeder embodying a new fuel handling concept which combines many of the advantages of both rigid and mobile fueled reactors. In the design concept discussed here, the core (as well as the blanket) consists of a bed of small randomly packed fuel spheres composed of uranium carbide-plutonium carbide. The bed is supported by a perforated plate which permits passage of the liquid sodium coolant. During reactor operation the coolant flow is downward so that the drag forces rigidly restrain the bed. While the reactor is shut down, however, the flow can be reversed, and for the fuel particle size considered here a moderate flow is sufficient to fluidize the bed. This fluidization makes it possible to transport fuel out of and into the reactor simply by operation of pumps and valves without opening the reactor vessel. Similarly, with fluidization the fuel can be stirred in order to increase the uniformity of burnup, or the fuel can be partially replaced or supplemented if necessary.

The solid fraction of a densely packed bed of hard spheres approaches 0.64. The interstices constituting the remaining 0.36 of the volume form a set of connected coolant channels adjacent to all fuel spheres, thus eliminating the need for internal structure in the form of rigid fuel boxes or rigid cladding which are required for maintaining the coolant channels in conventional rigid fuel reactors.

In addition to reduced fuel fabrication costs, the neutron economy is also greatly improved. The ratio of fuel to nonfuel volume is considerably higher in this reactor than in conventional designs, which leads to a much harder neutron spectrum with its attendant advantages of high fission-to-capture ratio in the fissile Pu and a high proportion of fissions in the fertile U. The low proportion of nonfuel material results in few parasitic captures in the structure and moderator. These conditions account for the high breeding ratio (1.7 to 1.8) that can be achieved in a SBFR. Moreover, it is possible to arrange the geometry so that the breeding ratio for the core alone is close to unity; this allows long burnup cycles with little or no reactivity changes requiring fuel addition or removal.

The major limitation is that of maintaining fuel integrity. The present design work is based on unclad fuel particles with a diameter of 0.120 in. The small particle size assures good heat transfer between fuel and coolant and a small temperature rise between the surface and the center of a fuel particle. While the possibility of fuel melting is small, the thermal stress due to the temperature gradient is not negligible. In the design presented here, the power density was limited primarily by this factor in order to assure the physical integrity of the monocarbide reference fuel spheres.

Although the use of unclad spheres improves the thermal contact between fuel and coolant, and at the same time substantially reduces the fuel fabrication cost, some fission products escape from the unclad fuel. Hence there is additional radioactivity in the primary coolant stream over that which would exist as a result of the normal activation of the sodium.

The discussion which follows describes the analysis of a conceptual design for a 1000-MW(e) reactor fueled with (U,Pu)C spheres and cooled by liquid sodium. Because of the importance of maintaining the physical integrity of the fuel, many of the final results will be seen to depend on the maximum burnup which can be achieved with this fuel. Another advantage of this fuel mobility is that a reduction in the refueling time will increase the plant availability factor. It has been estimated that this plant availability factor could be as high as 95%. In the design presented in this report, (U,Pu) was chosen as the reference fuel; however, other fuels can be considered. For example, use of cermet fuel may be sufficiently desirable from the point of view of obtaining large burnups to make the accompanying loss in breeding rates tolerable. A cermet would have the additional advantage of being a fully clad fuel.

Fuel

The composition of the reference fuel, (U,Pu)C, for this concept was selected on the basis of the state of the art of fuel technology. A hypostoichiometric composition (with respect to C) was selected for two reasons: (1) to avoid carbon transfer by the sodium coolant from the unclad carbide fuel to the

Figure 1. Macrophotographs of pre- and postirradiated fuel. Top: Dark beads of UC (18% U^{235}) prepared at the Vitro Corporation by the plasma-arc process; light beads of $U_{0.7}Zr_{0.3}C_{0.81}$ (18% U^{235}) prepared at BNL by arc-melting. Lower left: UC beads after irradiation (0.25% burnup) at 1400° F under sodium. Lower right: $U_{0.7}Zr_{0.3}C_{0.81}$ beads after irradiation (0.18% burnup) at 1400° under sodium.

stainless steel components, and (2) to take advantage of the plastic behavior of this fuel composition. The small fuel particle size (0.12 in. diam) reduces thermal stress and allows the bed to fluidize easily. However, internal stresses due to fission gas buildup in the particle could not be evaluated.

In the selection of fuel for the SBFR, the necessity for maintaining fuel integrity for the life of the core is of prime importance. Since the fuel is not clad, excessive thermal stress and excessive buildup of fission gas pressure must be avoided. Furthermore, the fuel spheres must not stick together.

In-pile capsule tests carried out in the Brookhaven Graphite Research Reactor (BGRR) have shown that sintering and cracking did not occur in a statically loaded column of hypostoichiometric UC spheres immersed in NaK at 1400° F up to burnups of 1800 to 2500 MWD/T at calculated thermal stresses of about 2500 psi. Figure 1 shows unirradiated fuel and fuel irradiated under the

above conditions. Capsule studies at Battelle Memorial Institute showed no cracking for burnups of 400 to 500 MWD/T and calculated thermal stresses of about 6000 psi. High burnup data for carbide fuels are available at present only at thermal stresses of the order of 70,000 psi, and most of the specimens subjected to these conditions cracked – some at burnups as low as 1 atom %.

In the present design the thermal stress in the fuel has been limited to 10,000 psi, and the economic analysis has been carried out for burnups from 25,000 to 100,000 MWD/T.

During the course of this design study, the experimental effort on fuel development was redirected to the study of mixed oxide cermet (UO_2 - PuO_2 /metal matrix) which were expected to exhibit better irradiation behavior at the higher burnups required for economically attractive breeder systems. This expectation has been reinforced by the work reported by Frost et al.¹ This indicates that properly fabricated (U,Pu) O_2 stainless steel cermet rods having from 30 to 50% by volume of the oxide experienced negligible dimensional change for burnups from 11.4 to 8.1 atom %, respectively. The present fuel irradiation studies include cermets with somewhat higher ceramic fractions with the aim of maintaining high breeding ratios typical of the present concept.

Settled Bed Physics

INTRODUCTION

From the reactor physics point of view, the SBFR concept is distinguished from current rigid fuel reactor concepts by its low volume fractions of structural material and coolant in proportion to its fuel volume fraction. This reduces both parasitic captures and energy degradation of the neutrons. The latter reduction is beneficial since a hard spectrum yields high average $\eta\epsilon$ values for fuel and fertile material, thereby giving a high breeding rate. This is borne out by the fact that, in the systems to be discussed, approximately one-fifth of the core fissions are in U^{238} , which amounts to a good bonus in breeding.

A large core volume is needed for 1000-MW(e) operation. As for other large fast power reactors, it is necessary in the present case to design for large neutron leakage out of the core. The reasons for this are threefold:

1. *The Sodium Temperature Coefficient Problem.* When sodium expands, the spectrum hardens because of loss of moderating material. This leads to higher fission-to-capture ratios in fuel and fertile materials, larger numbers of neutrons per fission, and smaller amounts of parasitic capture in the coolant. The consequent tendency to higher reactivity is counterbalanced by increased neutron leakage out of the core, which becomes less dense when the coolant expands. In order to secure a sodium coefficient that allows reactor control stability, it is necessary to enhance the leakage factor by going to core shapes that are thin in some dimension. This is accomplished in the axial flow reactor concept by using for the core a cylinder of small height-to-diameter ratio, and in the radial flow reactor concept by using a relatively thin annulus.

2. *Balance of Breeding Rates Between Core and Blanket.* It is desirable to have sustained operation without large reactivity changes due to isotopic changes in fuel and fertile material. This reduces the need for frequent refueling operations or for control devices which may absorb large numbers of neutrons. Hence the core breeding ratio should be near unity. To achieve this it is necessary to enhance leakage of neutrons out of the core, thereby shifting excess breeding to the blanket.

3. *Hydraulic Design.* Reduction of power required for pumping coolant through the reactor can be accomplished by having a reactor with large dimensions perpendicular to the flow directions and a correspondingly small dimension parallel to the flow.

CALCULATIONAL METHODS

The physics calculations have been carried out for the most part with one- and two-dimensional multigroup diffusion theory codes based on the 16-group Yiftah, Okrent, and Moldauer² cross-section library as amended to include the Pu²³⁹ alpha data of Diven and Hopkins.³

Most of the work reported here used the one-dimensional multigroup AIM-6 code.⁴ Some two-dimensional cases were examined with the TWENTY-GRAND code.⁵ These were run in six groups with cross sections produced by averaging the 16-group cross sections with flux weighting functions produced by AIM-6 calculations. It was found that accuracy could be improved in some cases by (a) calculating region- and group-dependent bucklings from the TWENTY-GRAND results, and by (b) using these bucklings in 16-group AIM-6 calculations to

Figure 2. Reactivity vs relative sodium density (ρ/ρ_0); blanketed cylindrical core; $H/D=0.2$; volume = 5-ft sphere.

produce improved 6-group cross sections for further TWENTY-GRAND calculations.

In all cases, however, the cross sections used in the two-dimensional calculations were region dependent. That is, the cross sections for a given material appearing in core and blanket are not taken to be the same in both regions but were averaged in such a way as to reflect spectrum differences in the various regions.

For the parameter surveys on the flattened reactors, most of the calculation was done with the buckling iteration method.⁶ This is an approximation to a two-dimensional calculation that can be carried out with a one-dimensional code. Here, estimated axial bucklings are used to make a radial calculation. The resultant flux is used to give estimates of radial core bucklings (for each energy group) for a subsequent axial calculation. This process can be continued, alternating between radial and axial calculations until the bucklings and multiplication factors converge to constant values. For a blanketed cylindrical settled bed reactor having a height-to-diameter ratio of 0.2 and a volume equal to that of a 5-ft-diam sphere, the buckling iteration method gave results for sodium temperature coefficient and breeding rate in close agreement with two-dimensional results. Indeed, the core sodium temperature coefficients as calculated by the two methods are no more than 10% apart, as shown by the slopes of the curves in Figure 2.

EFFECT OF FUEL MANAGEMENT ON REACTOR PROPERTIES

The SBFR concept allows for a number of important variations in fuel management. The core

and blanket regions may be periodically stirred. Compensation for reactivity change during burnup can be made simply by periodic addition of new material. In a reactor where the core breeding rate is greater than unity, the new material might be unenriched U^{238} -C spheres or even spheres containing poison for holding reactivity down.

Figure 3. Fraction ϵ of core which has a maximum burnup exceeding μ times the average burnup. N = number of fluidizations during burnup cycle.

Figure 4. Ratio of Pu^{239} loading in sphere to that in cylinder of equal volume as a function of cylinder height-to-diameter ratio H/D , for various volumes (spheres and cylinders with blankets).

All these possibilities interact with the isotope buildup processes and affect the course of the breeding rate changes, etc., during power operation.

Repeated stirring of the core tends to make the burnup more uniform. Although some small number of fuel spheres will find their way back to the peak power region every time the core is stirred, most will not. For the reactor designs considered in this report, the standard deviation in burnup for a single exposure period is of the order of half the average burnup. If the total exposure is divided into N subexposures with stirring of the core each time, then the standard deviation for the total exposure will be $1/\sqrt{N}$ times as large.

Thus, in order for all but a fraction ϵ of the core to have an exposure not exceeding μ times the average burnup, it is necessary to have N subperiods where N is given by

$$1 - \epsilon = \frac{1}{2} \{ 1 + \text{erf}[(\mu - 1) \sqrt{2N}] \}.$$

The error function used here is defined as usual by

$$\text{erf}(u) = \frac{2}{\sqrt{\pi}} \int_0^u e^{-x^2} dx.$$

This relationship is plotted in Figure 3.

As an example of the application of this result, we can state that for all but 1% of the core to have

Figure 5. Pu^{239} loading in blanketed spherical core vs core diameter for $k_{eff} = 1.01$.

a burnup not exceeding 1.3 times the average burnup it is necessary to have 15 subperiods; for all but 3% of the core to have a burnup not exceeding 1.2 times the average burnup it is necessary to have 22 subperiods, etc.

Whether or not the blanket is stirred may depend on which reactor design is being considered. For example, when the core breeding rate is less than unity it may be desirable to leave the blankets fixed for long periods, allowing preferential buildup of Pu^{239} near the core. This would tend to keep the reactivity from dropping during burnup. Since the full coolant flow goes through the outer blanket in the radial flow design, the increase in power generation rate in the radial blanket would not be disadvantageous. As an example of the improvement of behavior in reactivity vs exposure, an 825-MW(t) radial flow reactor which lost 3% in reactivity in 700 days of operation when the blanket was well stirred lost only 1% in reactivity for the same period when the blanket was not stirred. The preferential buildup of Pu^{239} adjacent to the core increases the effective reactor size and might impair the Na temperature coefficient to a small extent. This would limit the amount of time during which the blanket could be left unstirred.

For the present study the core and blanket regions have been assumed to be stirred frequently except as otherwise noted.

Figure 6. Sodium temperature coefficients for blanketed cylinders vs height-to-diameter ratio (H/D) and cylinder volume. (Volume expressed as diameter of sphere of equal volume.) Sodium temperature coefficients of spherical reactors indicated by points at $H/D = 1.0$.

PARAMETER STUDY OF FLATTENED CYLINDER CORES

While the dimensions of the SBFR concepts were being established, an investigation of critical loadings and sodium and Doppler temperature coefficients as a function of core shape and core volume was made. The reactors consisted of cylindrical cores of $\text{U}^{238}\text{-C}$, $\text{Pu}^{239}\text{-C}$ surrounded on all sides by 1.5-ft-thick blankets of $\text{U}^{238}\text{-C}$. The sodium volume fraction was 0.40 for this series. The calculations were made with the buckling iteration method.

Figures 4 and 5 indicate the Pu^{239} loading in the core necessary for $k_{eff} = 1.01$ as a function of core volume and height-to-diameter ratio. The quantity plotted in Figure 4 is the ratio of the loadings for a sphere and a cylinder of equal volumes. Figure 5 shows the required loading for spheres. The volumes in Figure 4 were chosen equal to the volumes of spheres measuring 4.5, 5, 6, and 7 ft in diameter.

Figure 6 shows the core sodium temperature coefficient as a function of reactor size and shape. The core sodium coefficients for spheres are shown in the same figure.

Figure 7 shows the Doppler reactivity effect for changes in temperature of core fuel. Doppler cross-section changes were obtained by means of a

Figure 7. Doppler effect in axial flow reactor design vs temperature, volume, and H/D of cylinder.

Table 1

Reactor Physics Properties

	Axial	Radial
Core height	11.25 in.	6.5 ft
Core thickness	11 in.	4 ft
Internal blanket diameter	3 ft	1 ft 11 in.
Core volume	94.25 ft ³	109.19 ft ³
N_{28}/N_{19} (for $k_{eff} = 1.01$)	8.33	8.79
Pu ²³⁹ loading (for $k_{eff} = 1.01$)	2264 kg	2500 kg
Breeding ratio in core	0.90	0.95
Breeding ratio in central blanket	0.12	0.27
Breeding ratio in external radial blanket	0.20	0.58
Breeding ratio in axial blanket	0.47	0.02
Na coefficient in core	$+0.7 \times 10^{-6}/^{\circ}\text{C}$	$1.6 \times 10^{-6}/^{\circ}\text{C}$
Doppler coefficient in core	$-1.7 \times 10^{-6}/^{\circ}\text{C}$	$-2.9 \times 10^{-6}/^{\circ}\text{C}$
Fraction of core fissions in U ²³⁸	0.208	0.212
Peaking factor, axial	1.10	1.37
Peaking factor, radial	1.32	1.12
Peaking factor, over-all	1.45	1.54
Prompt neutron generation time	22.3×10^{-8} sec.	19.0×10^{-8} sec
Effective delayed neutron fraction	0.00402	0.00391
Pu ²³⁹ capture/fissions in core	0.173	0.172
Pu ²³⁹ mean fission energy in core	0.23 MeV	0.23 MeV

Figure 8. Axial flow reactor design.

modification of the GAM-1 spectral code⁷ based on a method given by Greebler.⁸ These cross-section changes were inserted into AIM-6 multigroup one-dimensional calculations to obtain the reactivity changes with the correct flux and importance effects. The saturation of the Doppler effect as higher temperatures are approached is seen clearly in this figure. The negative contribution from U^{238} is from two to four times as large as the positive contribution from Pu^{239} .

Table 1 gives the significant results of the reactor physics calculations.

STABILITY OF REFERENCE REACTORS

Axial Flow Design

The core temperature coefficient for the SBFR concepts consists of two parts – the sodium expansion and the Doppler components. Because of the small size of the fuel spheres, the time lag for heat transfer from the fuel to the sodium is very small – of the order of a few hundredths of a second. Hence for practical purposes the sodium coefficient can be regarded as prompt (like the Doppler coefficient). Furthermore, the average temperature inside the fuel spheres is not very different from that of the surrounding coolant. In normal operation the difference in these two temperatures is $<10^\circ\text{C}$ for an average fuel sphere. This is important in considerations of reactor stability, since it means that at the operating point the Doppler temperature coefficient is applied to fuel temperatures only slightly higher than sodium temperatures.

As a consequence, it is very nearly accurate to require the algebraic sum of the Na and the Doppler coefficients ($\alpha_{Na} + \alpha_{Dopp}$) to be negative for reactor stability, at least as far as core temperatures are concerned. It is somewhat more accurate to require

$$1.04 \alpha_{Dopp} + \alpha_{Na} < 0,$$

where the factor 1.04 takes into account the greater temperature rise in the fuel spheres than in the sodium.

For axial flow reactors the parametric information in Figures 6 and 7 allows choice of a height-to-diameter ratio for a given core volume that will ensure a prompt negative power coefficient as expressed above.

The reference design reactor, Figure 8, differs in two important respects from the reactors described in the parameter study cited above. One of these is

the presence of the internal breeding blanket in the design reactor. This increases the leakage, and for temperature coefficient purposes it is equivalent to a decrease in height-to-diameter ratio and hence a more favorable sodium coefficient. Another and even more important difference is the presence in the design reactor of the sodium plenum regions above the core and the lower blanket regions. The reactivity coefficient for sodium expansion in either of these regions is negative and in magnitude it is about ten times as large as the corresponding core coefficient (Table 2). This introduces a strong delayed negative temperature coefficient with a delay corresponding to the time that it takes for core sodium to flow into the lower plenum and fill it.

The reactivity worth of a 25% reduction in sodium density in various positions in the core and plenum is shown in Figure 9.

Table 2

Na Coefficients in Units of $10^{-6}/^\circ\text{C}$

Region	Axial flow	Radial flow
Core	+0.7	+1.6
Plenum	-7.3 ^a	-4.8
Axial blanket	-0.3	~0
Outer radial blanket	~0	-0.7
Inner radial blanket	~0	~0

^aThis number applies to each plenum separately.

Figure 9. Sodium temperature coefficient vs position in axial flow reactor.

Radial Flow Design

In the radial flow design (Figure 10) as in the axial flow case, both the sodium and the Doppler coefficients can be regarded as prompt. The Doppler coefficient is negative and stronger than the positive core sodium temperature coefficient. The plenum sodium (in this case the zone just inside the core region) contributes a negative temperature coefficient, stronger than that of the core by a factor of about three. Again, since the sodium flow is inward, this gives rise to a delayed negative coefficient with a delay of the order of the time it takes for sodium to flow from the core and fill the plenum.

The reactivity worth of a 25% reduction in sodium density in various positions in the core and plenum regions is shown in Figure 11.

SODIUM LOSS

When a part of the reactor is emptied of sodium, the reactivity may rise or fall depending on the volume and location of the voided region. Figures 9 and 11 indicate this behavior. This possibility of positive or negative reactivity change applies to almost any of the large liquid-metal-cooled fast reactors. Thus, if sodium is voided from the central portion of the core the reactivity rises; if it is removed from the outer portion of the core the reactivity may fall. The prediction of the course of events during a reactor incident in which sodium is expelled has been attempted for some reactors.⁹ However, predictions of the course of such incidents are still highly speculative.

No attempts at such predictions have been carried out for the reactors considered here. However, because of the strong negative reactivity effect of voiding the plenum regions, and because of the high inertia of the bed and the difficulty of ex-

Figure 10. Radial flow reactor design.

PELLING sodium rapidly through the narrow passages of the bed, these reactors appear to be safe from the point of view of sodium voidage.

BED SETTLING DURING OPERATION

The physics calculations for the reference designs assume the volume fraction of sodium to be 0.35. If this volume fraction changes because of settling of the bed during reactor operation, the reactivity will change. The settling would probably be in the direction of coolant flow – downward in the axial flow reactor and inward in the radial flow reactor. This would mean an increase in reactivity in both reactors. The coefficient is $\approx 0.1\%$ for a sodium volume fraction change of 0.002 in 1% of the reactor volume. This number should be weighted by the relative importance of the region in which the settling occurs. For example, a change in sodium volume fraction from 0.350 to 0.349 in a volume comprising 10% of the core and located at a position of average importance would give a reactivity increase of

$$\delta\rho = 0.1 \times 0.001 / 0.002 \times 10 = 0.5\%$$

The changes that might be possible due to settling during actual reactor operation are indicated by current experimental work to be even smaller than this and are discussed in the section on bed-engineering.

Figure 11. Sodium temperature coefficient vs position in radial flow reactor.

COOLANT VARIATION

Change of Coolant Material and Coolant Volume Fraction

Substitution of lead or bismuth for the sodium coolant has been investigated for a number of spherical reactors containing settled bed fuel over a range of volume fractions. The actual design of a settled bed reactor with larger coolant volume fraction has not been examined, but it might be accomplished, for example, by the introduction of coolant channels through the bed.

For volume fractions near those of the reference design, substitution of lead or bismuth for sodium as coolant was found to have only a small effect on the reactor. This is simply because there is not much coolant. The critical loading decreases in going from sodium to lead to bismuth, the total fractional change being not more than about 2%. The breeding ratio drops in going from sodium to bismuth to lead, with a total fractional change not more than about 3%.

For larger coolant volume fractions, the differences become more pronounced but are still not

Figure 12. Pu^{239} loading for $k_{eff} = 1$ in blanketed spherical cores of diameter equal to 6 ft. Coolants are sodium, lead, and bismuth.

large. The larger coolant fractions give smaller critical loadings and breeding rates while the coolant temperature coefficient rises to a maximum and then decreases rapidly to negative values. The coolant volume fraction where the coolant temperature coefficient is at a maximum varies with core volume: it is about 0.6 for a 6-ft-diam spherical core and about 0.4 for a 4.5-ft core.

Figure 12 shows Pu^{239} loading as a function of coolant volume fraction for the three coolants in the case of a 6-ft-diam spherical core with a 1.5-ft-thick blanket. Figure 13 shows sodium temperature coefficient vs core size and coolant volume fraction.

Reactivity Control With Lithium

A possible control mechanism is based on the addition of lithium to the coolant. The attendant reactivity decrease is due to softening of the spectrum by the increased moderating power of the low atomic weight lithium plus the increased absorptions in the Li^6 isotope. As an example of this effect the control capabilities both of natural lithium and of enriched lithium-6 were calculated for a reactor with a 5-ft spherical core and a $1\frac{1}{2}$ -ft blanket. The results are shown in Figure 14. The

reactivity worth of substituting lithium for 1% of the sodium coolant is

$$\begin{aligned}\delta K &= 0.0013 \text{ for natural lithium} \\ &= 0.012 \text{ for } \text{Li}^6 \\ &= 0.054 \text{ for } \text{B}^{10}.\end{aligned}$$

The sacrifice in breeding is given by

$$\begin{aligned}\delta K/\delta (\text{breeding ratio}) &= 2.1 \text{ for natural lithium} \\ &= 0.75 \text{ for } \text{Li}^6 \\ &= 0.48 \text{ for } \text{B}^{10}.\end{aligned}$$

With lithium control, an expansion of the coolant would expel control material. This leads to a less negative temperature coefficient for the reactor, the change in temperature coefficient depending on the amount of lithium in the coolant.

CERMET FUELS

A number of cermet fuel materials have been examined in the settled bed physics studies. These cermets were taken to be 80% by volume of PuO_2 - UO_2 (UO_2 in the blanket) and 20% by volume of matrix material. Table 3 shows the effect on critical loading (for $k_{\text{eff}} = 1.01$) and breeding for a 6-ft

Figure 13. Sodium temperature coefficient vs sodium volume fraction and size of spherical core.

Figure 14. Reactivity and breeding loss vs fractional replacement of sodium atoms by lithium or boron atoms.

Table 3
Effects of Using Various Matrix Materials in Cermet Fuels

Fuel	Mass of Pu ²³⁹ (kg)	Breeding ratio	
		In core	In blanket
Reference PuC, UC	1569	1.56	0.44
Cermets with:			
Stainless steel	1256	1.17	0.45
Cr	1263	1.16	0.44
Zr	1230	1.19	0.46
Mo	1420	1.00	0.33
Nb	1464	0.93	0.30
Th ²³²	1471	0.92 in U ²³⁸ 0.40 in Th ²³²	0.33 in U ²³⁸ 0.12 in Th ²³²

spherical core and a 1.5-ft blanket. The volume fractions were 0.6 for fuel and 0.4 for coolant. The loss in core breeding due to introduction of cermet fuels is large. The advantages of these fuels lie, however, in improved thermal conductivity and burnup characteristics.

Bed Engineering

From the beginning of the engineering studies on the SBFR concept, the problem of stability of a randomly packed fuel bed has been considered of paramount importance. Since, in the case of randomly packed beds, the system is difficult to define, it is necessary to approach the problem of bed stability from a purely experimental standpoint. The point of reference for the settled bed reactor is a permissible abrupt consolidation of the fuel bed to the extent of an increase of 0.002 in the solid fraction. In this study, therefore, the requirements for determining and demonstrating stability of packed beds is much more stringent than has been the case heretofore as when dealing, for example, with the potential for settlement of foundation structures. It becomes, therefore, a matter of providing an adequate demonstration of bed stability, based on this requirement, with the necessary factor of safety.

The experimental program on bed settling has resulted in the development of a laboratory technique to measure changes in solid fraction as small as 0.0002, e.g., an increase in solid fraction from 0.6000 to 0.6002. The technique includes a rubber membrane, under slight differential pressure, to

conform to the top of the bed and to separate the bed from a measured volume of water contained in a special cap above. The water extends into a buret mounted on top of the cap, and settling of the bed is indicated by changes in the volume of water in the buret (see Figure 15). In addition, a reasonably conservative test has been adopted as a standard for measuring bed resistance to settling. A brass hammer of 3% of the combined weight of the bed material and vessel imparts a horizontal blow to the vessel wall after a free fall from a height of 5 in. (see Figure 16).

By utilizing the measurement technique and the hammer test, a bed preparation procedure has been developed which stabilizes the bed to a satisfactory degree of resistance to settling. Stabilization of the bed appears to be best accomplished hydraulically by pulsing the bed with short upward injections of flow at slightly greater than fluidization velocity (see Figure 17). The upward flow, which lifts the bed a fraction of an inch as a slug, is diverted before the bed is disrupted and the bed falls back in place with a slightly greater solid fraction. After ≈ 500 cycles on 6-in.-diam beds (which can be achieved in < 10 min), the effect of pulsing is greatly diminished, which indicates stabilization with respect to disturbances through the coolant. In addition, the hammer test indicates that beds prepared in this manner are satisfactorily stable, especially with downward coolant flow which induces drag forces tending to lock the bed particles in place (see Figure 18). Also shown in Figure 18 are the beneficial effects of both wall roughness and the injection of down-flow as the bed falls back in place following the

Figure 15. Membrane cap, showing water level in buret, attached to 6-in.-i.d. Lucite column containing $\frac{1}{8}$ -in. SS spheres. A partial vacuum in the bed vessel pulls the membrane down to conform to the top surface of the bed.

Figure 16. The 3-lb brass hammer test device with associated mechanism, pin pickup, and counter. As the hammer rebounds from striking the bed vessel, the pin in the rotating 10-in. wheel engages the arm attached to the pivoting shaft of the hammer and lifts the hammer to the dropping height until the pin slips past the end of the arm and the hammer again falls to strike the container. (Also shown is the transparent Lucite section above the bed which allows observation of the bed top during flow conditions.) Sound-proofing is required.

upward pulse. It should be noted that prepared beds have never exhibited abrupt changes even approaching a 0.002 increase in solid fraction. In addition, polished $\frac{1}{8}$ -in. stainless steel ball bearings were used as the bed material which should show more slippage than nonperfect fuel spheres. These studies were performed on 6-in.-diam beds and are being extended to 14-in.-diam beds in the 1000-gpm water-flow test loop (see Figure 19).

The effect of thermal expansion and contraction on randomly packed beds has been briefly examined. The solid fraction of 6-in.-diam randomly packed beds has been found to vary somewhat during thermal cycling between room temperature and 500°C when spheres with coefficients of thermal expansion ranging from 30 to 100% of that for the vessel are used.

Although it has a small effect on reactor reactivity, the solid fraction variation within a randomly packed bed is under study, because it represents nonuniformity of heat generation and coolant flow within the bed. It has not been deter-

Figure 17. The valving arrangement for supplying both up-pulse and down-pulse consists of a 3-way PVC ball valve with the stop removed. A motor turns the valve continuously to provide a short injection of up-flow in the bed and to abruptly divert the flow to the drain allowing down-flow in the bed.

Figure 18. Cumulative arithmetic change in solid fraction vs cumulative 3-lb hammer blows on bed container. The bed container holds ≈ 75 lb of $\frac{1}{8}$ -in. 316 SS beads in a 6.065-in.-i.d. 347 SS vessel.

Figure 19. Water flow test loop, 1000-gal/min capacity.

mined whether solid fraction variations will be present to an excessive extent. The spacing of the slots in the bed support plate has a strong influence on the internal structure of the packed bed, and the results of current work will indicate the nature of the support plate and vessel walls for minimizing the solid fraction variation.

In an effort to determine qualitatively the extent of solid fraction variations in beds prepared by the pulsing technique, several 6-in.-diam beds were prepared without special precautions to minimize the orderliness that might form at the support plate. Photographs were taken of the top of each bed as it was carefully dismantled. The only order observed occurred adjacent to the vessel wall for 3 to 4 ball diameters inward.

To characterize more quantitatively the solid fraction variation, a bed of $\frac{1}{8}$ -in. bronze spheres was prepared by the pulsing method in a 6-in.-diam vessel with a support plate which discouraged close ordered packing at the bottom. This bed was cast by using an epoxy resin to retain the pulsed-down structure. The casting was cut into 1-in. cubes, and the volume of the cubes was determined. The epoxy resin was then removed. The volume of the spheres in each cube is now being determined to give solid fraction. A radial and vertical plot of these values will quantitatively reveal the solid fraction variation.

Previous calculations indicated negligible effects of molecular and eddy diffusion to provide a uniform temperature distribution within a SBFR; however, these calculations assumed a uniform solid fraction and velocity profile for the packed bed. A similar conclusion is anticipated for a large cluster of dense packing in an otherwise randomly packed bed with an attendant difference in velocity between the two regions. This tentative conclusion for a large cluster of dense packing is supported by observations of a small-scale, two-dimensional dye dispersion test with liquid flow through a packed bed which contains contiguous regions of different solid fraction.

In a three-dimensional test, the difference in cooling rate for a hexagonal-packed cluster in an otherwise randomly packed bed has been briefly investigated. A hexagonal-type cluster measuring about 1 in. on a side was formed by cementing together layers of $\frac{1}{8}$ -in. spheres. This cluster was placed in a 6-in.-diam bed of less dense packing. The central sphere of this cluster had been suitably instrumented with a thermocouple on the

surface. Another sphere in the less dense bed was similarly instrumented. Air heated to 500°F was passed through the bed to burn off the cement and bring the bed to a uniform temperature. The air heater was abruptly bypassed to cause a rapid drop in inlet temperature of about 50°F/min. During this transient the temperature of the sphere in the center of the dense cluster lagged behind that of the random portion by as much as 80°F. During one run the central sphere of the dense cluster was found to be cooled at a rate of 27°F/min compared with a rate of 32°F/min for the sphere in the random part. In progress is a three-dimensional test in which an induction heated metal sphere placed in a bed of glass spheres is used. Measurement of the outlet temperature profile will assist in the evaluation of the effect of solid fraction variation within a packed bed.

Thermal and Hydraulic Characteristics

This section of the report surveys the thermal and hydraulic characteristics of the SBFR with random packing. Specifically included are (1) a parametric survey of selected variables showing the behavior of randomly packed bed systems with internal heat generation (axial and radial power peaking factors and other corrections for deviation from average operating conditions were neglected here); (2) a discussion on estimating the heat transfer coefficient for sodium flowing through a packed bed; and (3) the performance characteristics of the reference axial flow design.

The reference design is based on a system comprising a dense, randomly packed bed of spherical fuel particles in which heat is generated internally, because of fissioning of U and Pu, and removed by liquid Na coolant flowing downward through voids in the bed. As a reactor, this type of system combines a very high packing density of solids with a very large heat transfer surface to produce very high power densities.

PARAMETRIC SURVEY

The interaction of selected thermal and hydraulic characteristics of the reference design on each other was investigated. They included core height, area, volume, and power density, super-

ficial coolant velocity, and fuel sphere size. The core power rating and the sodium temperatures and flow rate were set simply to conform to the reference design; no attempt was made to optimize them here. The average bed solids packing density of 0.64 was chosen as the highest value obtained thus far in random bed studies¹⁰ at BNL. The fuel was assumed to be in the form of spheres and with an allowable thermal stress of 10,000 psi. It should be specifically noted that temperature variations, axial and radial power peaking factors, and other items causing deviations from the average assumed operating conditions were neglected here. They are, however, included in the section on Performance Characteristics of the Reference Design.

Additional constraints for this parametric study were (1) heat generation rate of 2500 MW, (2) UC fuel spheres, and (3) maximum pressure drop through the bed of 300 psi.

HYDRAULIC CONSIDERATIONS

The pressure drop of fluid in turbulent flow through a randomly packed bed of spheres may be expressed in Leva's¹¹ equation as

$$\Delta P = \frac{2fL\rho v^2(1-\epsilon)^{3-n}\lambda^{3-n}}{gD_p\epsilon^3}, \quad (1)$$

where

- ΔP = pressure drop, lb/ft²,
- f = friction factor,
- L = bed height, ft,
- ρ = density of fluid, lb/ft³,
- v = superficial fluid velocity, ft/sec,
- ϵ = bed voidage,
- λ = particle shape factor, 1.0 for spheres,
- g = gravitational constant, 32.2 ft/sec²,
- D = sphere diameter, ft, and
- n_p = Leva flow factor,

and for flow in the region at $Re > 300$, the friction factor varies with Reynolds number as

$$f = 1.75Re^{-0.1} = 1.75(D_p v \rho / \mu)^{-0.1} \quad (2)$$

where μ = viscosity of fluid, lb/sec-ft. Since the bed is packed with spheres, the particle shape factor λ is equal to unity, and the exponent n was assumed to be 2. The correlation generally predicts values of pressure drop which are slightly on the high side compared with other correlations. Conservative values were considered to be desirable here in view of the uncertainties involved in actual fuel particle uniformity, roughness, and sphericity.

By using values for the appropriate sodium properties (Table 4), a sodium flow rate of 43.4×10^6 lb/hr, and a bed voidage of 0.36, and by substituting Eq. (2) into Eq. (1), one obtains the relationship between coolant pressure drop, superficial coolant velocity, core height, and sphere size:

$$\frac{\Delta P}{L} = 0.154(v^{1.9}/D_p^{1.1}), \quad (3)$$

where ΔP is expressed in psi.

The value of $\Delta P/L$ at a voidage of 0.36 obtained in Eq. (3) can be multiplied by the following values to obtain the pressure drop at other voidages.

Voidage	Multiplier
0.36	1.000
0.37	0.908
0.38	0.823
0.39	0.749
0.40	0.682

The variation of coolant pressure drop with fuel sphere diameter and superficial velocity is shown in Figure 20. If the maximum allowable pressure drop through the core is arbitrarily set at 300 psi, the limiting velocities for various core heights are as shown in the figure.

Table 4

Selected Properties of Sodium¹²

	550°F	875°F	1200°F
Density, lb/ft ³	55.0	52.4	49.7
Viscosity, lb/sec-ft	2.39×10^{-4}	1.862×10^{-4}	1.333×10^{-4}
Specific heat, BTU/lb-°F	0.313	0.303	0.300
Thermal conductivity, BTU/hr-ft-°F	44.1	39.8	35.6

Each fluid velocity corresponds to a specific core cross-sectional area, since

$$A = w / (3600 \rho v) \quad (4)$$

where w = total fluid flow rate, lb/hr, and A = core area perpendicular to flow, ft². At a fixed sodium flow rate of 43.4×10^6 lb/hr, then,

$$A = 230 / v, \quad (5)$$

and these values are also shown in Figure 20.

These curves (which neglect thermal considerations) show that if the total pressure drop is fixed (i.e., at 300 psi), the operating range of the reactor is bounded by the core height curves at the top and the superficial coolant velocity (or core area) at the right. At a constant fuel size, an increase in core height requires a decrease in superficial velocity (or equivalent increase in core area). At a constant core height, a decrease in bed sphere size causes a decrease in superficial velocity.

The coolant pressure drop is much more sensitive to a change in coolant velocity than to the same change in either fuel size or core height. Therefore, factors tending to reduce the velocity (e.g., increased flow area) should be emphasized. It is also much more beneficial to increase the flow area than to reduce the bed height; the pancake shape for a reactor is preferable, therefore, on the basis of both pressure drop considerations and reactor physics.

THERMAL STRESS CONSIDERATIONS

The maximum thermal stress in an elastic sphere with uniform internal heat generation and surface temperature is determined from

$$\sigma_t = \frac{\alpha E D_p^2 q'''}{60(1-\nu)k}, \quad (6)$$

where

- σ_t = maximum thermal stress, psi,
- α = coefficient of linear thermal expansion, per °F,
- E = modulus of elasticity, psi,
- q''' = power density within the sphere, Btu/hr-ft³,
- ν = Poisson's ratio, and
- k = thermal conductivity, Btu/hr-°F-ft.

It is assumed that this relationship holds for each of the fuel spheres investigated - UC, UO₂, and 80% UO₂-20% SS cermet - and that the value of

Figure 20. Variation of pressure drop per foot of core with fuel size, core area, and superficial coolant velocity in core.

Figure 21. Variation of maximum core power density with fuel type and size, core height, and ratio of pumping power to total electric power.

Table 5
Selected Properties of Fuels

	UC ¹³	UO ₂ ¹⁵	80% UO ₂ -20% SS ¹⁵
Coefficient of linear thermal expansion, per °F	6.4×10^{-6} ¹⁴	5.62×10^{-6}	6.45×10^{-6}
Modulus of elasticity, psi	32×10^6	27.5×10^6	18×10^6
Poisson's ratio	0.3	0.3	0.3
Thermal conductivity, BTU/hr-ft-°F	14.5	1.94	6.92

allowable thermal stress in each case is 10,000 psi. The stress-limited maximum core power density was computed for each fuel by using the values in Table 5 and then plotted against fuel sphere diameter in Figure 21. There appeared to be very little difference in the maximum allowable power density between UC and the cermet, but one must bear in mind that considerable uncertainties exist in their measured data. UO₂, on the other hand, is considerably poorer, the maximum power density being lower by a factor of about five.

COMBINED THERMAL AND HYDRAULIC CONSIDERATIONS

The total heat output Q (in Btu/hr) of the core is

$$Q = 0.64ALq''' , \quad (7)$$

and the amount removed by the sodium coolant is

$$Q = wc_p \Delta T_c , \quad (8)$$

where

c_p = specific heat of fluid, Btu/lb-°F, and
 ΔT_c = temperature rise in the fluid, °F.

Since all the heat generated in the core has to be removed by the coolant,

$$0.64ALq''' = wc_p \Delta T_c . \quad (9)$$

The expression for A from Eq. (4) is then substituted in Eq. (9), and the values for the sodium properties in Table 4 are used to obtain

$$q''' = 5.81 \times 10^7 \frac{v}{L} . \quad (10)$$

The expression for v from Eq. (3) is substituted into Eq. (10) to obtain the relationship between core power density, coolant pressure drop, fuel size, and core height.

$$q''' = 1.554 \times 10^8 (\Delta P^{0.526} D_p^{0.579} / L^{1.526}) . \quad (11)$$

The value of q''' at a bed voidage of 0.36 obtained in Eq. (11) can be multiplied by the following values to obtain the power density at other voidages.

Voidage	Multiplier
0.36	1.000
0.37	0.951
0.38	0.903
0.39	0.859
0.40	0.818

The effect of core height on the maximum core power density was computed from Eq. (11) for various ratios of pumping power to total electric power produced by the reactor core and also plotted in Figure 21. The reactor system is defined by the core height curves at the top and the fuel curves calculated previously from Eq. (6) on the right. At a constant fuel ball size, an increasing core height causes a decrease in power density. At a constant core height, the power density varies inversely with the fuel size. The effect of a change in core height on power density, however, is far greater than the equivalent change in either coolant pressure drop or fuel size.

The relationship between the core power density and core volume may be obtained by transposing Eq. (9) and substituting the appropriate values,

$$q''' = 2.055 \times 10^7 (\Delta T_c / V) , \quad (12)$$

where V = core volume, ft³. This is plotted in Figure 22. The operating range lies between the coolant ΔT curves on the right and the various stress-limiting values of core power densities (for UC only) and their corresponding fuel sphere diameters at the top. At a fixed sodium flow (i.e., 43.4×10^6 lb/hr) and constant power density, the core volume varies directly as the coolant ΔT through the reactor. Similarly, at a constant core volume (and coolant flow rate), a decreasing coolant ΔT will cause a corresponding drop in the heat re-

moved – the same effect as a reduction in core power density. If the fuel spheres are already operating at the maximum stress-limiting power density, this will cause a corresponding decrease in total power output of the core (Figure 23). Figure 23 is based on the relationship

$$Q = 1.91 \times 10^4 (V/D_p^2), \quad (13)$$

which is obtained by substituting Eq. (6) into Eq. (7).

The various reactor parameters can also be expressed in terms of the fuel ball diameter (d_p) as follows:

- 1) superficial coolant velocity,
 $v \propto d_p^{-0.31}$;
- 2) core height,
 $L \propto d_p^{1.69}$;
- 3) core area,
 $A \propto d_p^{0.31}$;
- 4) core volume,
 $V \propto d_p^2$;
- 5) core power density,
 $q''' \propto d_p^{-2}$.

These quantities are plotted in Figure 24. It can be seen that a given decrease in (U-Pu)C fuel size produced a much smaller decrease in core area, along with an equivalent increase in superficial coolant velocity. The decrease in core volume was somewhat greater than that of core height, both decreasing considerably more than fuel size. The

Figure 22. Variation of maximum core power density with core volume and coolant (ΔT).

core power density increased by an amount equivalent to the decrease in core volume. It should also be noted that a change in core height has a much greater effect on the core power density (and core volume) than the same change in fuel size. The core area has a much smaller effect. The smallest cores will, therefore, be those with small fuel size and minimum core height (with only a small resultant increase in superficial coolant velocity), and the limiting reactor size appears to be set principally by reactor physics considerations (e.g., critical mass).

ESTIMATING THE SODIUM HEAT TRANSFER COEFFICIENT IN A PACKED BED

An attempt was made to estimate the film coefficients for transferring heat from the UC fuel spheres to liquid Na coolant flowing through a dense, randomly packed bed. The packed bed correlations which are available do not apply to

Figure 23. Variation of maximum core power output with core volume and fuel size.

liquid metals, while the liquid metal correlations are not specific to packed beds. The procedure used here was suggested by Zizza and Patti¹⁷ and involves: (1) calculating the heat transfer coefficient for fixed beds by using an available correlation; (2) considering liquid flow through tubes with a hydraulic diameter equivalent to that of the spheres, and calculating the heat transfer coefficient for these tubes by using correlations for both water and liquid metal; (3) multiplying the fixed bed coefficient obtained in (1) by the ratio of liquid metal to water coefficients for the equivalent tubes.

The correlation of Glaser and Thodos¹⁸

$$j_H = \frac{0.535}{\{[\sqrt{s_p} v \rho \lambda / \mu (1 - \epsilon)]^{0.3} - 1.6\}}, \quad (14)$$

where

j_H = Colburn factor, and

s_p = particle heat transfer surface, ft²,

was used to obtain the sodium heat transfer coefficient for flow through a packed bed of spheres. Since

Figure 24. Various reactor parameters vs fuel ball size. Peaking and hot channel factors not included.

$$j_H = (h / C_p v \rho) / (C_p \mu / k)^{2/3}, \quad (15)$$

the heat transfer coefficient, h , may be calculated using a voidage of 0.36, a sphere size of 1/8 in. diam, and the appropriate values for sodium properties given in Table 4.

The hydraulic diameter D_h of the equivalent tubes was calculated from

$$D_h = 4d = 4 \left[\frac{D_p}{6} \frac{\epsilon}{(1 - \epsilon)} \right] \quad (16)$$

and is 0.00390 ft for the bed in question.

The correlation¹⁹ for nonmetals flowing through tubes used was

$$h = 0.023 C_p v \rho / \text{Re}^{0.2} \text{Pr}^{2/3}, \quad (17)$$

while a modified Lyon equation recommended by Herrick²⁰ was used for liquid metal flow,

$$\text{Nu} = h D_h / k = 5.0 + 0.025 (\text{RePr})^{0.8}. \quad (18)$$

The values for the heat transfer coefficient were calculated with the following results:

Case	Correlation	Heat transfer coefficient, Btu/hr-ft ² -°F	
		550°F	1200°F
packed spheres	water	134,400	140,000
tubes	water	46,200	52,000
tubes	liquid metal	61,200	49,900

The coefficients for the packed bed were corrected as described before, and an average value in the reactor of 150,000 Btu/hr-ft²-°F was obtained. This is extremely large, and the resultant temperature drop in the sodium film will be only a few degrees. Therefore, even if the method for computing the heat transfer coefficient introduced considerable error, the net effect will be very minimal at best, and could probably be neglected altogether.

PERFORMANCE CHARACTERISTICS OF THE REFERENCE DESIGN

The objective of the reference design was the selection of a SBFR which produced 1000 MW(e) at maximum power density and, therefore, minimum core volume and inventory, commensurate with the limitations imposed by reactor physics, mechanical design, fuel properties, and thermal and hydraulic characteristics.

The criteria used in selecting the core and fuel characteristics were:

1. Total reactor heat output of 2500 MW(t), based on an over-all cycle efficiency of 40%.

2. Hypostoichiometric (U-Pu)C fuel in the form of spheres with an allowable thermal stress of 10,000 psi, on the basis of the best current experimental data.

3. Mixed average Na coolant temperatures of 1200°F and 550°F at the reactor outlet and inlet, respectively. The outlet temperature was set by the AEC as one of the design objectives of the Fast Reactor Design Studies^{21,22,23} in order to utilize the high thermal efficiency of modern steam plants, and the inlet temperature was the lowest that is compatible with commercial turbine condenser water conditions.

4. The maximum Na temperature anywhere in the reactor could not exceed the boiling point.

5. The total coolant pressure drop through the reactor was limited to 300 psi on the basis of reasonable extrapolation of current Na pump technology.

The thermal and hydraulic characteristics of the reference SBFR are shown in Table 6. A core volume of 94.2 ft³ is required to produce 2500 MW(t). This represents 60.3 ft³ (21,154 kg) of (U-Pu)C fuel, at an average bed voidage of 0.36.

The Na flow rate was calculated from Eq. (8) to be 43.4×10^6 lb/hr, since the coolant temperature rise was fixed at 650°F, and the total power at 2500 MW(t). The superficial coolant velocity in the core is 2.2 ft/sec, and the average heat flux is 2.46×10^5 Btu/hr-ft² over a surface of 34,700 ft². The average core power density is 938 kW/liter and the specific core power is 1100 kW/kg PuC.

Fuel

A fuel size of 0.120-in.-diam hypostoichiometric (U-Pu)C was selected to meet the requirements of reactor physics, allowable heat generation rate, and ease of fabrication. While UO₂-PuO₂ cermets may provide more attractive burnups, on the basis of the available measured properties, they will probably not alter the reactor dimensions significantly in the heat transfer and fluid flow area (Figure 21).

Average Temperature Drops

The average Na film temperature drop, ΔT_f , on the surface of the fuel spheres was calculated to be 1.7°F from

Table 6
Thermal and Hydraulic Characteristics
of the 1000-MW(e) Reference SBFR

1. Total heat output	2500 MW(t)
2. Fuel	UC-PuC
3. Coolant	Na
4. Total core volume	94.2 ft ³
Fuel volume	60.3 ft ³
Void volume (at $\epsilon = 0.36$)	33.9 ft ³
5. Total weight of fuel	21,154 kg
6. Fuel size	0.120 in.
7. Heat transfer surface	34,700 ft ²
8. Average heat flux	2.46×10^5 BTU/hr-ft ²
Volumetric core heat generation	
Rate	26.5 MW/ft ³ core
Power density	938 kW/liter core
Specific power	1100 kW/kg PuC
9. Maximum heat flux	5.26×10^5 BTU/hr-ft ²
Volumetric core heat generation	
Rate	56.2 MW/ft ³ core
Power density	2005 kW/liter core
Specific power	2360 kW/kg PuC
10. Coolant flow	43.4×10^6 lb/hr
11. Number of passes	One
12. Average coolant velocity	2.2 ft/scc
13. Average coolant temperatures	
Reactor inlet	550°F
Reactor outlet	1200°F
Rise	650°F
14. Maximum coolant temperature	1640°F
Coolant temperature rise	1090°F
15. Average film temperature drop	1.7°F
Fuel surface temperature	873°F
Fuel temperature drop	41°F
Fuel center temperature	914°F
16. Maximum film temperature drop	4°F
Fuel surface temperature	1643°F
Fuel temperature drop	87°F
Fuel center temperature	1709°F
17. Over-all hot channel factors	
$F_{\Delta T}$ - maximum temperatures	1.68
maximum ΔT	1.87
F_q - maximum temperatures	1.77
maximum ΔT	2.32
F_θ - maximum temperatures	1.61
maximum ΔT	2.11

$$\Delta T_f = QS/h, \quad (19)$$

where S = total heat transfer surface in the core, ft². The average temperature drop within the fuel sphere, ΔT_s , based on uniform heat generation and a limiting thermal stress of 10,000 psi, was calculated to be 45°F from

$$\Delta T_s = (q''' D_p^2) / 24k. \quad (20)$$

Table 7
Hot Channel Factors

	Maximum temperatures			Maximum ΔT 's		
	$F_{\Delta T}$	F_q	F_θ	$F_{\Delta T}$	F_q	F_θ
ENGINEERING FACTORS						
Maldistributions in local coolant flow	1.10	1.10	—	1.10	1.10	—
Deviations in size, composition, density, and enrichment of individual fuel spheres	1.05	1.05	1.05	1.05	1.05	1.05
Variations in heat flux around surfaces of individual fuel spheres	—	1.25	1.25	—	1.25	1.25
Total	1.16	1.44	1.31	1.16	1.44	1.31
NUCLEAR FACTORS						
Axial power peaking	1.00	0.85	0.85	1.11	1.11	1.11
Radial power peaking	1.32	1.32	1.32	1.32	1.32	1.32
Local power peaking	1.10	1.10	1.10	1.10	1.10	1.10
Total	1.45	1.23	1.23	1.61	1.61	1.61
Over-all hot channel factor	1.68	1.77	1.61	1.87	2.32	2.11

Hot Channel Factors

The hot channel factors used in the study are listed in Table 7 in two different sets, to determine the maximum temperatures and maximum temperature differences, respectively. The former occur at the core outlet, where the axial flux is $\approx 85\%$ of the average, while the latter occur at a point about two-thirds of the way down the core, where both axial and radial flux peaking are assumed to be still maximum (conservative estimate). In both cases, a value of 1.10 was assumed to correct for local power peaking because of control rod perturbations.

An engineering factor of 1.10 reflects the influence of maldistributions in local coolant flow through the bed on the bulk coolant and Na film temperature drops. A factor of 1.05 corrects for the effect of deviations in fuel sphere size, composition, density, and enrichment on the coolant flow, fuel, and Na film temperatures. These values are based on average values used in the most recent fast reactor studies.^{21,22,23} A heat transfer correction of 1.25 was applied to the fuel and film temperature drops to compensate for variations in thermal conductivity and temperature in any single sphere. The same value was used in the ORNL Pebble Bed Reactor study.²⁴

Maximum Temperatures at the Hottest Core Regions

The Na and fuel temperatures and temperature drops were calculated at the core outlet and also at a point about two-thirds of the axial distance down the core at its hottest region to determine the maximum values. The two-thirds point represents the value at which the coolant temperature has increased most of the way, while the core power densities are still close to their maximum.²⁴ The individual reactor characteristics determine at which point the maximum values occur.

In this design, the maximum Na temperature and temperature rise of 1640° and 1090°F , respectively, and the maximum fuel center temperature, 1709°F , occur at the core outlet of the hottest region, while the maximum temperature drops (4°F through the Na film and 87°F through the fuel sphere) occur at the two-thirds point, as shown in Table 8.

None of these values is excessive. The maximum Na temperature is 180°F below the boiling point of 1820°F ¹² at a system pressure of 35 psia. The fuel temperatures are well below the melting point of 4262°F and of 3650°F , the temperature at which UC vaporization has been noted.²⁵ The ΔT 's do not exceed the allowable thermal stress.

Figure 26. Flow diagram for primary and secondary coolant systems.

service building, and a warehouse. A switchyard, waste holdup tanks, fuel oil storage, auxiliary power, roads, and railroad spurs are also provided.

The reactor fuel is in the form of small spheres which are arrayed in beds. The core is annular in shape and is surrounded on all sides by blanket material. Reactor heat is removed by a multiloop primary coolant system, which in turn transfers it to the steam-generating equipment. Reactor inlet conditions are 550°F and 316 psig, while outlet values are 1200°F and 20 psig; steam conditions are 1050°F at 2450 psig. A composite diagram of all coolant systems is shown in Figure 26.

Reactor Building

The reactor building, shown in Figures 27 and 28, houses the reactor, primary coolant system, and fuel handling system. The building is a vertical steel cylinder with a hemispherical top head and ellipsoidal bottom head. The inside diameter is 115 ft, and the over-all height is 168 ft, a 65-ft portion of which is below grade. The building is a pressure vessel which is capable of containing all radioactive fission products or radioactive sodium released by a primary system failure or by a reactor incident. Preliminary analysis of the pressure

Figure 27. Equipment and piping arrangement, plan.

buildup in the containment vessel after an accidental release indicates a maximum internal pressure of 40 psig.

The building is divided into two main areas by the operating floor. The area above the floor is designated as the operating area, while that below the floor consists of shielded compartments which contain the reactor and radioactive coolant and fuel handling systems.

The operating area provides general access to the building and working space for the maintenance and replacement of the reactor system components. The area is served by an overhead crane for this purpose.

In the shielded area below the floor, the reactor is located in the center of the building with its six coolant loops radiating outward like the spokes of a wheel. System fill and drain tanks are located at

Figure 28. Equipment and piping arrangement, elevation.

loop low points, and the fuel handling equipment is arranged between the coolant loops.

Shielding for the radioactive systems consists of the concrete operating floor above the reactor, a concrete wall which surrounds the reactor and separates it from the coolant loops, a perimeter concrete wall which encloses the coolant loops, and a bottom concrete base. The concrete operating floor provides shielding for the operating area. The concrete wall which surrounds the reactor minimizes neutron activation of the coolant system components and piping. The perimeter wall is the outer shield for the coolant system. The concrete base which is the bottom shield also serves as the main support base for the entire building and its equipment. Supplementary shielding is also provided. This shielding consists of a neutron shield reflector in the reactor vessel which serves to protect the vessel against radiation damage from neutrons, and a borated graphite shield around the reactor for the moderation and capture of neutrons escaping from the vessel.

Control Room

The control room is located between the reactor building and the turbine generator building. This room functions as the control point for the reactor and the entire generating complex. The room also serves as the control station for certain high-level remote maintenance operations involving the radioactive systems. For this reason the control room is provided with the necessary shielding for these operations.

Steam Generator and Turbine Generator Buildings

These buildings are interconnected and are located directly adjacent to the reactor building. The steam generator building contains the steam generators, the secondary coolant system pumps, and the secondary sodium fill and drain tanks. The turbine generator building houses the turbine generators, main condensers, feedwater heating system, and all the associated systems and equipment of a conventional power plant. Both the steam generator and turbine generator buildings are provided with overhead bridge cranes for servicing equipment.

Sodium and Inert Gas Service Building

This building contains the sodium and inert gas receiving facilities, storage tanks and purification equipment. The building is connected to the reac-

tor building by an underground pipe tunnel. Because the sodium purification equipment maintains the purity of the primary sodium during reactor operation, the system is located in a shielded room for personnel protection against radioactive sodium.

Fuel Handling and Hot Shop

The fuel handling and hot shop building provides the facilities for the handling of irradiated fuel and radioactive materials and components removed from the reactor building. The building contains decay pits, hot cells, storage pools, and shielded work galleries and is equipped with cranes, transfer dollies, and remote handling devices. Tracks connect this building to the reactor building. These tracks are used by the transport dolly for the transfer of spent fuel casks and radioactive components from the reactor building. A railroad spur which enters the building provides for the direct removal of large and heavy units by railroad car.

Heating and Auxiliary Power Plant

A heating and auxiliary power plant is located alongside the steam generator building between the reactor and turbine generator buildings. This plant provides the necessary heat and power required by the reactor complex during periods of plant shutdown and during plant start-up.

Office and Service Building

This building is located adjacent to the reactor building and provides space for the plant's conventional office needs and support service facilities. The support facilities include machine shops, instrument repair shop, health physics and radioactive chemical laboratories, counting room, locker rooms, sanitary facilities, and contaminated laundry room.

Warehouse

The warehouse is a conventional structure located along the railroad spur entering the plant. This building is used for the receipt and storage of equipment, spare parts, and miscellaneous materials required by the plant.

REACTOR

Both the axial and radial type of reactor were investigated in this study. However, because the

core configuration is simpler greater emphasis was given to the axial type, and this reactor is used as the reference design in this report.

Axial Type (Figure 8)

The reactor vessel is 16 ft 10 in. i.d. by 18 ft high with a 5-in.-thick wall. It is designed for a pressure of 325 psig at 1200°F. The top of the vessel is designed as a flanged and removable closure head which may be removed for maintenance of an internal component or removal of the entire internal structure. During operation the head is seal welded to the vessel to prevent leaks. The vessel, as well as all internal components with the exception of the neutron shield, are of AISI type 316 stainless steel.

Contained within the reactor vessel are the core and blanket regions, coolant plenums, neutron shield, support structures, segmenting plates, and control rods. The sodium coolant flows axially downward during power operation. Fluidization of the fuel is accomplished by flowing sodium up through the bed after a reactor shutdown. This permits removal, addition, and mixing of fuel. The core and upper and lower blanket regions are provided with segmenting plates to aid fluidization and to improve bed settling.

Because of the unusual features of the axial flow design, particularly the large operating temperature rise and reverse coolant flow during fluidization, preliminary stress analyses were devoted to three main categories, namely, temperature effects, pressure distribution, and fatigue.

The relatively high temperature of 1200°F places the reactor vessel and its internal core components in the range of material creep. Therefore, the thickness requirements of the reactor vessel and internal structure were determined on the basis of an allowable membrane strength of 6800 psi, which represents 1% creep in 100,000 hr for AISI type 316 stainless steel.

The inlet pressure of 316 psig is relatively low; however, the internal pressure distribution is a complex and unique characteristic of the settled bed axial flow concept.

At the steady-state operating condition, the coolant flows vertically down through three parallel flow paths. These paths are through the central cylindrical blanket, the external blanket annulus, and the core annulus. The pressure drops through the center and exterior blankets are small relative to the pressure drop through the core annulus.

With the use of orifice plates set above the center and exterior blankets, these pressure drops can be increased, and in this manner the pressure differentials that exist across the core ring and center blanket ring can be both reduced and controlled.

Since fluidization with reverse coolant flow will take place at a fixed interval, under shutdown conditions, as part of normal operation of the reactor and will be utilized in core and blanket removal, an investigation of cyclic loading was made. The bottom region of the reactor is subjected to the largest fluctuations of temperature and pressure when the reactor undergoes a full cycle from normal to reverse flow. A thermal and pressure discontinuity analysis was performed at the junction of the cylindrical shell and semi-ellipsoidal head in the bottom region of the reactor vessel.

The results of this fatigue analysis indicate that there is no practical limitation to the allowed number of operating cycles for this reactor.

The core is a right cylinder 12 ft in diam by 11¼ in. high with a 4-ft-diam center blanket. Above and below the core are 1-ft-thick upper and lower blankets. Between the core and the blankets are coolant plenums which enable the particles in the core and blanket to expand during fluidization. Surrounding the entire core is a 1-ft-thick external blanket and surrounding the external blanket is a neutron shield reflector which consists of 14 in. of laminated iron separated by sodium coolant passages. This shielding protects the vessel against radiation damage from neutrons.

The fuel in the various areas is supported by slotted grid plates which enable the coolant to flow while preventing the fuel particles from passing through. Each grid is reinforced by a radial and circumferential rib construction of 1 ft depth.

The reactor vessel is also provided with coolant inlet and outlet nozzles and fuel loading and unloading nozzles.

The control rods are shown installed internally in the core ring. Six rods are shown, but the final number and exact location require more detailed physics calculations.

Radial Flow (Figure 10)

The reactor vessel is 13 ft 6 in. i.d. by 26 ft 6 in. high with a 3-in.-thick wall. It is designed for a pressure of 130 psig at 1200°F. The top head of the vessel is flanged to permit removal for maintenance of an internal component. Contained within the reactor vessel are the core and blanket

regions, coolant plenums, neutron shield, support structures, and control rods. The vessel as well as all internal components, except the neutron shield, are of AISI type 316 stainless steel.

Coolant enters the reactor vessel through six 20-in. pipes at the bottom of the reactor and then flows up along the external wall of the vessel in the inlet plenum. The coolant then flows radially inward through the external blanket, the core, and finally into the outlet plenum. The blanket and core rings have slotted perforations which allow coolant to flow while preventing the fuel particles from passing through the rings. The perforations are spaced so that there is a slight downward component of flow through the fuel bed serving to hold the bed in a settled condition. From the outlet plenum the coolant leaves the vessel through six 20-in. outlet pipes. A small amount of coolant for the center blanket flows into the upper part of the reactor from an 8-in. inlet pipe, flows downward through the center blanket and out the 8-in. coolant outlet pipe. This small flow removes the heat generated in the center blanket and also holds the blanket in a settled condition.

The core is an annulus, 6 ft 9 in. o.d. by 4 ft 9 in. i.d. by 10 ft 6 in. high. The fuel region is contained in the annulus to a height of 6 ft 6 in. The additional 4-ft height of the core vessel permits expansion of the fuel bed during fluidization. The fuel pellets rest on perforated plates which are supported by rib construction. A plenum below the supports is fed by six 8-in. fluidizing inlet lines. During fluidization the sodium flows up through the bed and out the six 8-in.-diam pipes at the top of the core region. Fuel loading and unloading ports consist of four 4-in. lines located at the bottom of the core.

The blanket area consists of the center and external blankets. These areas have fluidizing, loading, and unloading lines similar to the core. The neutron shield is located outside the inlet plenum and is similar to the one described for the axial flow reactor.

As with axial flow, the control rods are shown installed internally in the core ring. Six are shown, but the final number and exact location require more detailed physics calculations.

REACTOR COOLANT SYSTEMS

The reactor coolant systems are designed to remove the heat from the reactor and utilize it to

Table 9

Primary Coolant System

Coolant data	
Primary coolant	sodium
Flow rate per loop	7.23×10^6 lb/hr
Flow rate per loop	16,500 gpm at pump temp
Number of loops	6
Total flow rate	43.4×10^6 lb/hr
Total flow rate	99,000 gpm at pump temp
Temperature:	
Heat exchanger inlet	1200°F
Heat exchanger outlet	550°F
Total sodium volume	11,200 ft ³
Piping layout	
Piping material	type 316 stainless steel
Design pressure	350 psig
Design temperature	1400°F
Pipe size	20 in.
Intermediate heat exchanger	
Number of units	6
Heat transferred per unit	1410×10^6 BTU/hr
Pumping equipment	
Number of pumps	6
Operating data (each pump):	
Capacity	16,500 gpm
Temperature	550°F
Total head	840 ft
Efficiency	70%
Horsepower	4300 hp

generate steam for the turbine generator. This heat transfer is accomplished with two multiloop systems: the primary coolant system and the secondary coolant system. The primary coolant system removes the heat from the reactor and transfers it to the secondary coolant system in the intermediate heat exchangers; the secondary coolant system transfers the heat to the water and steam in the steam generators.

The reactor is rated at 2500 MW(t) (8533×10^6 Btu/hr) and uses sodium as the coolant for both primary and secondary coolant systems.

Primary Coolant System

The primary coolant system consists of six heat transfer loops connected in parallel to the reactor. Each loop contains a primary pump and an intermediate heat exchanger. The system also includes interconnecting piping, valves, fill and drain tanks, inert gas venting and blanketing connections, and necessary instrumentation. The system arrangement is shown in Figures 27 and 28. System data are presented in Table 9. Primary loop and bed handling schematics are shown in Figures 29 and 30.

Figure 29. Flow diagram for primary cooling loop and bed handling system, axial flow.

Figure 30. Flow diagram for primary cooling loop and bed handling system, radial flow.

During operation, the primary pumps circulate sodium through the reactor, the loop piping, and the intermediate heat exchangers. The sodium enters the top of the reactor vessel and flows down through the blanket and core regions where it removes heat generated by the fission process. The hot sodium emerges from the bottom of the reactor vessel and flows to the intermediate heat exchangers where the heat is transferred to the secondary coolant system. The sodium then returns to the suction side of the primary pumps to repeat the cycle.

The reactor vessel is provided with separate exit piping connections for the center and outer blanket coolant, which is $\approx 5\%$ of the total flow. These connections are equipped with valves for the purpose of controlling the sodium flow. Downstream from the valves, the blanket-coolant piping is combined with the main loop piping returning to the intermediate heat exchangers.

During fluidizing operations, flow through the reactor is reversed, i.e., sodium enters the bottom of the reactor, flows upward through the blanket and core beds, and exits from the top. Bypass piping and valves are provided in each primary coolant loop for this purpose.

Each loop is provided with a fill-and-drain tank with appropriate interconnecting piping. The tanks are sized to contain the entire primary system volume and to provide surge and expansion volume for the sodium. The loops are filled by pressurizing the tanks with an inert gas; draining is by gravity. Connections are provided at all high points of the coolant system for system venting during filling and draining operations. These vents are connected to the inert gas system to prevent entry of air into the system components and piping. All piping is pitched to insure complete and positive drainage and to eliminate localized stagnant gas pockets during system filling.

All outside surfaces of the system are equipped with electric heaters for preheating the system before filling, and to prevent sodium freeze-up during system shutdown. Insulation is provided throughout for 1200°F.

The primary pumps are electric motor driven, vertical shaft, centrifugal sump-type pumps having a mechanical shaft seal. Each pump has a flow capacity of 16,500 gpm against a head of 775 ft. The pump extends up through the operating floor with the pump drive located above the floor level to permit ready access to this unit for maintenance.

A stepped shield plug is provided to prevent radiation from streaming through the floor. The pump internals, including the shield plug, are designed as a unit which can be removed without disturbing the primary piping or draining the primary sodium.

An inert gas blanket maintains a positive inert atmosphere above the sodium level in the pump to prevent oxide contamination of the sodium. This inert gas chamber also provides a convenient high point in the primary loop for venting purposes and for purging fission products if necessary. The pump shaft seal prevents the inert gas from escaping into the building.

The intermediate heat exchangers are counter-flow shell-and-tube-type units with primary sodium on the shell side and secondary sodium in the tubes. The exchangers are vertical units with the shell extending through the operating floor to permit direct access to the exchanger for removal of the tube bundle. In this way, the tube bundle may be removed without draining the sodium or disturbing the system piping on the exchanger shell. Stepped shield plugs are provided to prevent radiation from streaming through the floor.

Secondary Coolant System

The secondary coolant system consists of three steam generator reheaters, six circulating pumps, and interconnecting piping. The system also includes fill-and-drain tanks, inert gas venting connections, and necessary instrumentation. Piping and component arrangement is shown in Figures 27 and 28. System data are presented in Table 10.

The six intermediate heat exchangers are connected in pairs to the three steam generators. Heated sodium flows from each bank of two exchangers to a steam generator where the sodium heat is transferred to the feedwater and steam. The cooled sodium then flows to the suction side of the two circulating pumps and is returned to the two exchangers for reheating by the primary sodium.

The secondary coolant system is kept at a higher pressure than the primary system to prevent leakage of radioactive primary coolant into unshielded areas.

Fill-and-drain tanks are provided with appropriate interconnecting piping. These tanks also furnish a surge-and-expansion volume for the system. The system is filled by pressurizing the tanks with inert gas; draining is by gravity. Other services, similar to those described for the primary

Table 10
Secondary Coolant System

Coolant data	
Secondary coolant	sodium
Flow rate per loop	14.6×10^6 lb/hr
Flow rate per loop	33,000 gpm
	at pump temp
Number of loops	3
Total flow rate	43.4×10^6 lb/hr
Total flow rate	99,000 gpm
	at pump temp
Temperature:	
IHX inlet	500°F
IHX outlet	1150°F
Total sodium volume	18,000 ft ³
Piping layout	
Piping material	type 316 stainless steel
Design pressure	50 psig
Design temperature	1350°F
Pipe size	20 and 30 in.
Steam generator and reheater	
Number of units	3
Heat transfer per unit	2820×10^6 BTU/hr
Pumping equipment	
Number of pumps	6
Operating data (each pump):	
Capacity	16,500 gpm
Temperature	500°F
Total head	130 ft
Efficiency	70%
Horsepower	700 hp

coolant system, are provided, namely, inert gas blanket, high-point vent connections, pitched piping, electric heaters, and insulation.

The secondary pumps are electric motor driven, vertically mounted units with mechanical shaft seals. Each pump has a capacity of 16,500 gpm against a head of 130 ft. An inert gas blanket is maintained under positive pressure above the pump sodium level to prevent oxide contamination.

The steam generators are vertical shell-and-tube-type units with secondary sodium on the shell side and water and steam on the tube side. Each unit consists of three sections: the boiler, superheater, and reheater. The steam generator is equipped with a pressure-relieving rupture disc to protect the unit against severe damage in the event of a tube leak and subsequent sodium-water reaction.

Steam System

The steam system consists of the steam piping, turbine generator, condenser, condensate and

feedwater pumps, feedwater heaters, and feedwater piping. System data are presented in Figure 26.

Steam, generated in the three steam generators, is combined into one main steam line and piped to the higher pressure unit of the turbine generator. Exhaust steam from the intermediate pressure unit is returned to the reheater section of the steam generators for reheat before being admitted to the turbine's low pressure section. Bleed steam is extracted from the turbine at several points for use in the feedwater heaters.

Turbine exhaust steam is condensed in the surface condenser and is returned to the steam generators through the feedwater system to complete the cycle.

The main steam line is provided with an emergency dump bypass which is connected to emergency dump condensers. This bypass system is normally on a standby basis and is used for the removal of steam from the steam generators in the event of a turbine trip.

The turbine generator is a double shaft machine designed for continuous operation and it includes the necessary surface condenser, auxiliaries, instrumentation, and controls. All of the shafts are directly connected to generators. The turbines consist of high pressure, intermediate pressure, and low pressure units, arranged to permit cross and tandem-compound multiple flow of steam with reheat provided between the intermediate pressure and low pressure stages. The over-all dimensions of the double shaft installation are 164 ft long by 60 ft wide by 23 ft high. The turbine units include speed control, pressure regulation, gland-sealing systems, complete lubrication oil systems, motor extraction devices where required, vacuum breakers, and motor-operated turning gear.

The generators are totally enclosed, self-ventilated, synchronous, 3-phase, 60-cycle, hydrogen-cooled units. They include shaft-driven main exciters, motor-driven spare exciters, gas-cooling and purge systems, gland-seal oil systems, and voltage regulators.

Feedwater heaters to regeneratively heat the feedwater from the temperature at the surface condenser hot well to the temperature required at the steam generator inlet are used in the steam cycle. Bleed steam from the turbine is the heating medium. The feedwater heaters are of the closed, shell-and-tube type with the steam on the shell side and the feedwater in the tubes. Condensate

drains are cascaded with high pressure heater drains going to the deaerating feedwater heater and low pressure heater drains to the surface condenser hot well.

The surface condenser(s) have 2-pass divided water boxes with a nondeaerating hot well and internal air cooling sections. The condenser(s) are rigidly supported on foundations with expansion joints between the turbine exhausts and the steam inlets. Auxiliaries include the following: steam jet air ejectors with surface-type inter-after-condensers, priming ejectors, loop seal trap or traps, and integral steam piping to a common connection. Condensate from the surface condenser is used as the cooling medium for the air ejector condensers.

Condensate from the surface condenser(s) hot wells is pumped by means of condensate pumps through the low pressure feedwater heaters to the deaerator. The condensate pumps are horizontal, centrifugal, volute-type units. Boiler feed pumps take suction from the elevated deaerator and discharge the feedwater through the high pressure feedwater heaters to the steam generators. Cooling water is pumped through the surface condenser(s) by means of circulating water pumps.

AUXILIARY SYSTEMS*

This section presents a brief description of some of the more important auxiliary systems which serve the plant. These systems include: containment ventilation, inert gas system, and sodium storage and purification system.

Containment Ventilation

For ventilation purposes, the containment vessel is divided into two areas: (1) above the operating floor, and (2) below the operating floor. A normal air atmosphere is maintained above the floor to facilitate personnel access to this area. A nitrogen atmosphere is maintained below the floor to inhibit sodium fires in this area in case of a leak. Seals are provided for the operating floor to separate the two areas.

Heating and cooling of the area above the floor is accomplished with fan-coil units located near the top of the containment vessel. These units recirculate the containment air and are equipped with electrical heaters for use when heating is required. Cooling is accomplished by supplying the

coils with a coolant from a system located outside the containment vessel.

Only cooling is provided for the nitrogen atmosphere below the floor because it is expected that there will be sufficient heat gain from the operating systems. The system consists of fans, heat exchangers, and supply and return ducts located outside the containment vessel, and distribution ducts located in the below-floor area. Nitrogen is circulated across the shield wall surfaces in the containment vessel, ducted out to the heat exchangers where it is cooled, and is then returned to the containment vessel for recirculation. The entire system is designed to provide containment integrity at least equal to that of the containment vessel.

Inert Gas System

An inert gas system is provided for supplying the sodium systems with blanket gas to prevent oxide contamination of the sodium. The system consists mainly of a receiving and storage station, purification equipment, pressure regulating station and distribution piping. Gas is supplied by either gas trailer tank or banks of manifolded gas cylinders. Purification is accomplished by bubbling the gas through NaK-filled tanks for removal of oxygen and water. The NaK is maintained at room temperature for this purpose. The purified gas is then distributed to the various sodium systems as required. System pressures are maintained by the pressure regulating station.

Sodium Storage and Purification System

This system provides for the receipt, storage, and purification of the plant's bulk sodium requirements. It is also capable of monitoring and maintaining the sodium purity of the operating coolant systems on a continuous basis. The system includes storage tanks, cold traps, plugging indicators, and samplers. The entire system is provided with electrical heaters and inert gas-blanketing connections.

System equipment is located in the sodium and inert gas service building which is connected to the reactor building by an underground pipe tunnel.

Bulk sodium is received in tank cars, is melted, and is then transferred to the storage tanks. The sodium is then circulated through NaK-cooled cold traps for the removal of oxides. Plugging indicators are used to check sodium purity; samples

for chemical analysis are obtained at the sampler. Upon completion of the purification cycle, the sodium is transferred to the various coolant systems as required. During plant operation, the purification system maintains sodium purity by processing a continuous stream of sodium extracted from the reactor coolant systems.

The storage tanks and purification equipment are located in concrete shielded cells for biological protection against the radioactive sodium which the system handles.

PLANT OPERATIONS AND PROCEDURES

In addition to normal power generation, certain supporting functions must be performed for the successful operation of the plant. For this conceptual study, the scope of work did not include a detailed investigation of all plant functions; therefore, only those considered to be of prime interest and of significant importance to proper plant operation are presented. These functions include start-up and shutdown, fuel handling, and maintenance procedures for radioactive equipment.

Start-up and Shutdown

Prior to start-up, the primary and secondary sodium coolant systems are evacuated and purged with inert gas. System components and piping are then preheated to $\approx 300^\circ\text{F}$ at the rate of about 50°F per hour. Purified sodium is transferred to the fill-and-drain tanks. The tanks are pressurized with inert gas, forcing the sodium into the coolant system. During filling, the system is vented to the inert gas system.

After the primary coolant system is completely filled, the primary pump(s) are started, and the primary system valves are positioned for reverse flow through the reactor vessel. During this period the fuel is inserted in the reactor. All control rods are "in" during fueling operations.

Upon completion of fueling, the system valves are repositioned for normal, downward flow through the reactor. The control rods are slowly withdrawn and the primary sodium temperature is raised to 550°F . The secondary coolant system is started and withdraws heat until the reactor inlet and outlet temperatures are stabilized at 550° and 1200°F .

An alternative method of bringing the system up to power is to preheat the sodium-filled system

to 550°F before inserting the fuel. However this method requires more external power for heating purposes.

Procedures for shutdown are basically the reverse of those for start-up. Shutdown is accomplished by inserting the control rods and continuing to circulate the sodium systems for decay heat removal. When sufficient decay heat has been removed, and it is safe to do so, the primary coolant valves are positioned for reverse flow through the reactor. The fuel handling system is employed, and the fuel is removed from the reactor. Upon completion of fuel removal, the sodium is allowed to cool to $\approx 300^\circ\text{F}$ and is then drained to the fill-and-drain tanks. Inert gas is vented into the system during sodium drain. It is recognized that the sodium need not be drained at every shutdown; the steps are described here merely to present a complete system shutdown procedure.

Fuel Handling

Reactor fuel insertion and removal are accomplished with the fuel handling system. The system consists of dump tanks, fill tanks, decay tanks, and interconnecting piping. All system components and piping are sized and arranged to insure subcritical geometry. A schematic of the system is shown in Figures 29 and 30.

Prior to fueling operations, the primary system valves are positioned for reverse flow through the reactor. The fuel pellets are placed in the fuel handling system fill tanks through an air-lock-type port which maintains the system's inert atmosphere. The fuel is then piped to the sump tank after which the fill tank is valved off. Valves in the bypass lines connecting the dump tank to the primary coolant loop and reactor vessel are opened. This allows a portion of the primary sodium to flow up through the dump tank and into the reactor with the sodium acting as a carrier for the fuel pellets. The reverse-flow condition of the reactor sodium assists in drawing the fuel pellets into the reactor vessel.

Upon completion of the fueling operations the valves in the dump tank bypass loop are closed, and the primary loop valves are repositioned for normal coolant flow through the reactor. Small valves which bypass the dump tank valves are opened. This allows a continuous trickle of sodium coolant to flow through the fuel pellets which are in the small portion of pipe between the reactor vessel and the closed feed valve. This

bleed stream of sodium is combined with the main coolant flow returning to the pump suction.

Fuel removal from the reactor is accomplished by first reversing the sodium flow through the reactor and fluidizing the fuel bed. While the fuel pellets are in this fluidized condition, the dump tank valves are opened and a stream of pellet-laden sodium is directed to the dump tank. The tank is equipped with an internal grid plate in the bottom head. This grid plate serves as a sieve to catch and retain the fuel pellets while permitting the sodium to flow through. The sodium is then piped to the main coolant return flow upstream of the intermediate heat exchanger. The exchanger removes any decay heat picked up from the fuel.

When filled with fuel, the dump tank is valved off from the reactor. The other dump tank valves are repositioned to permit upflow through dump tank and to the decay tank. The sodium flows up through the grid plate and carries the fuel pellets to the decay tank where it deposits them on a grid plate similar to the one described for the dump tank. The sodium is then returned to the main coolant stream.

Upon completion of fuel transfer, the decay tank is valved off from the fuel handling system and the fuel is allowed to decay. During the decay period, the fuel is cooled by a forced circulation sodium system provided for this purpose. After the decay period when forced cooling is no longer required, the decay tank sodium system is shut down and the fuel pellets are transferred to a shielded spent fuel cask and removed.

During the entire fuel removal cycle, irradiated fuel decay heat is removed by a continuous flow of sodium coolant.

Maintenance of Radioactive Equipment

As a means of providing a practical approach to the maintenance of the radioactive equipment in the reactor building, maintenance procedures are divided into three categories: routine maintenance, special maintenance, and major maintenance.

Routine maintenance is defined as a regularly scheduled program of inspection and repair or replacement of equipment above the operating floor and outside the sealed system. Typical items of equipment in this category are drive units for the primary pumps, valves, and control rods, and associated electrical gear, cables, and switches.

Special maintenance requires the breaching of the primary system seal or shielding. This proce-

dures requires strict health physics control and the use of a shielded coffin for the radioactive component. This procedure is confined to small components such as control rod guide tubes.

Major maintenance involves the removal and replacement of large components such as the primary pump internals, the intermediate heat exchanger tube bundles, or the reactor internals. It is expected that removals of this magnitude will occur only once or twice during the plant's lifetime; therefore, no special shielding equipment is provided. The radioactive component is removed from the system and placed in equipment decay tanks beneath the operating floor. During removal, the equipment is sealed in plastic bags to prevent the spread of contamination in the operating area. After removal, a new component is installed in the system and the plant is ready to resume operation. After sufficient decay time, and during plant shutdown, the inoperative component is removed from the equipment decay tank and is transported to the hot shop for repair or disposal. During those periods when the large radioactive components are unshielded, control of operations is performed from the shielded control room.

Economics

Because the fuel is an important part of the cost of any reactor system, the results of an economic study will depend strongly on the maximum burnup which can be achieved by the fuel. The earlier discussion in this report shows that the limiting burnup is not well established for the carbide fuel of the SBFR; hence the information which follows should be regarded more as a statement of economic possibilities than as a cataloging of well established costs.

FUEL COSTS

Assumptions

In general, the procedure outlined in reference 26 for estimating fuel costs was followed closely. Some items which were included to enable a more realistic investigation of costs for the SBFR are briefly discussed below.

The number of cores irradiated before discharging each blanket loading, hereafter referred to as CPB, was a variable in the cost study. It was assumed that the CPB core fuel cycles comprising

each blanket cycle are identical, thereby neglecting differences among them in the core/blanket power split, core irradiation time, and other items which vary with the amount of fissile material which builds up in the blanket. It was assumed that each of the CPB cores is operated at a power level averaged over the blanket irradiation time. The irradiation time was thus considered to be the same for each core batch and was determined by the average core power level. As a result, the fuel costs determined are for average conditions over each blanket cycle. At this point in the study, no fuel recycle effects have been considered; therefore, each batch of fuel charged to the core is fresh, unirradiated material, and all CPB cores have the same composition throughout irradiation.

Fuel costs include fuel cycle operating costs as well as fixed charges on fuel cycle working capital. The second item includes working capital requirements for fuel fabrication, for purchase of privately owned nuclear material, for delayed payment of postirradiation processing charges, and for delayed receipt of credit for bred Pu. A single interest rate is applied to all working capital required for the fuel cycle. It is assumed that the fuel fabricator owns the nuclear material in his possession when private ownership of fuel is in effect.

Processed Pu and U will eventually be recycled as nitrates to the fabricators. Conversion of $\text{Pu}(\text{NO}_3)_4$ to PuC and UNH to UC is included in the cost of fabrication. Any Pu returned to the AEC is acceptable in nitrate form.²⁷ Since only depleted U is used, there is no requirement for conversion of UNH to UF_6 for re-enrichment in a diffusion plant. Depleted U is assumed to be leased from the AEC when private fuel ownership is not in effect.

The separation plant throughput rate is a function of the fissile Pu content in the fuel batch discharged from the reactor and is determined according to the procedure given in reference 28. Batches discharged from the core and blanket are processed separately, since they can be discharged at different intervals.

The discussion in the reactor physics section shows that frequent mixing of the core fuel material tends to give a more uniform fuel exposure. To achieve a peak/average exposure of 1.0, it would be necessary to continuously mix the fuel material, which is obviously impossible. Any realistic interval of reactor operation between successive mixings would result in a nonuniform fuel

exposure, but for the purpose of estimating fuel costs, it was assumed that all fuel having the same residence time in the core also has the same exposure. This effect will be considered in more detail during subsequent studies.

Periodic Fuel Addition

To minimize the excess reactivity required for reactor operation and still enable the core fuel to reach its maximum allowable exposure before discharge, reactivity is periodically inserted by adding fuel of the initial composition to the core and fluidizing the bed to give a uniform mixture of fresh and irradiated fuel. Since no irradiated fuel is removed, the core height and volume increase with each fuel addition made. Consequently, fuel addition serves to decrease the axial leakage of neutrons from the core and to increase k_∞ slightly by effectively increasing the average enrichment of the irradiated fuel material. Enough fuel is added each time to provide the excess reactivity required to overcome fuel burnup effects between additions. Sufficient head room exists above the core bed for both the axial and radial flow designs to permit a reasonable increase in height without restricting the capability for thorough mixing of the fuel during the fluidization process. The effect of this bed height increase on sodium temperature coefficients has not been estimated as yet, but must be considered before accepting this method of reactivity adjustment in a final design.

The cumulative mass of fuel added during a core lifetime must be added to the original fuel loading to give the total mass of fuel irradiated per core. As irradiation proceeds, fuel added to the core will be exposed for smaller and smaller time periods; hence, only that amount of fuel which is in the core at time zero will suffer the maximum exposure. The average discharge exposure for all fuel charged to the reactor will obviously be less than the maximum. Approximate expressions are given below for the average fuel exposure ($\bar{E}$), the maximum exposure ($\bar{E}_{\text{max}}$), and the ratio $\bar{E}/\bar{E}_{\text{max}}$, assuming that the core thermal power output remains constant at P_0 over the core lifetime, that fresh and irradiated fuel spheres have the same power density despite differences in isotopic composition, that the time interval (Δt) between additions of fresh fuel is constant throughout core life, and that the mass of fuel in each addition (Δm) is the same. In the following equations, m_0 is the fuel

mass charged to the reactor at time zero and N is the total number of fuel additions made over the core lifetime. The total number of time intervals is $N+1$ since no fuel is added after the final interval.

$$\begin{aligned}\bar{E} &= P_0(N+1)\Delta t / (m_0 + N\Delta m), \\ E_{\max} &= \frac{1}{m_0} \left[P_0\Delta t + \frac{P_0\Delta t}{1+(\Delta m/m_0)} + \right. \\ &\quad \left. \frac{P_0\Delta t}{1+(2\Delta m/m_0)} + \dots + \frac{P_0\Delta t}{1+(N\Delta m/m_0)} \right] \\ &= \frac{P_0\Delta t}{m_0} \sum_{n=0}^N \frac{\Delta n}{1+(n\Delta m/m_0)}, \quad \text{since } \Delta n = 1 \\ &= \frac{P_0\Delta t}{\Delta m} \sum_{n=0}^N \frac{\Delta x}{1+x_n},\end{aligned}$$

where

$$x_n = n\Delta m/m_0 \quad \text{and} \quad \Delta x_n = \Delta n\Delta m/m_0 \equiv \Delta x.$$

If $\Delta m/m_0 \ll 1$, then $\Delta x \ll 1$. In this case, Δx can be replaced by dx and the summation can be approximated by an integral between $x=0$ and $x=N\Delta m/m_0$. The upper limit of integration is the ratio of the cumulative mass of fuel added over the core lifetime to the initial core loading. Hereafter this ratio will be referred to as f .

$$E_{\max} = \frac{P_0\Delta t}{\Delta m} \int_0^f \frac{dx}{1+x} = \frac{P_0\Delta t}{\Delta m} \ln(1+f).$$

By using the last approximation for $E_{\max}$, we get

$$\begin{aligned}\bar{E}/E_{\max} &= (N+1)\Delta m / (m_0 + N\Delta m) \ln(1+f) \\ &= (N+1)f / N(1+f) \ln(1+f),\end{aligned}$$

since $f = N\Delta m/m_0$. If $N \gg 1$,

$$\bar{E}/E_{\max} = f / (1+f) \ln(1+f).$$

A further approximation, which is commonly used for problems of this type, is

$$E/E_{\max} \approx 1/(1+f).$$

Although this expression can be relied upon only for small values of f , it was felt to be adequate for an analysis of the present nonoptimized reactor design. At this point in the study, detailed estimates of N and Δm have not been made, but subsequent optimization of the reactor design and core breeding ratio will result in a very small fuel addition requirement over the core lifetime. The validity of the various assumptions made above will be investigated when the reactor design is completed.

For this study, f values were estimated by setting $N\Delta m$ equal to the fuel mass which must be added

to the core to restore the initial reactivity after the original core fuel loading was irradiated to $E_{\max}$ during continuous irradiation at a power level of P_0 .

An alternative method of causing the reactivity to become inert would be to remove irradiated fuel material from the core when fresh fuel is added. If the mass removed equals the mass added, the core height will remain constant throughout the core lifetime. This procedure results in a very large fuel addition requirement, e.g., $f=0.3$ for the axial flow design when $E_{\max} = 100,000$ MWD/T. There are two main reasons for this: (1) since the core height does not change, the fuel added does not significantly decrease neutron leakage from the core, and (2) the fissile content of the discharged fuel is not very much lower than that for the fuel added because the core breeding ratio is above 0.9. Thus, this method was eliminated from further consideration. Addition of fuel with a higher fissile content than that of the fuel loaded at time zero would reduce the f value but would require fabrication of two types of fuel material and could also result in an excessive power density in the fresh fuel added.

A composite study which includes fuel handling, temperature coefficient, thermal stress, and breeding considerations will be necessary before an acceptable procedure is established for reactivity adjustment.

Range of Investigation

Certain parameters were varied in the study of the reactor fuel cycle economics. The ranges of variation considered for each of these items is discussed in the following paragraphs.

Unit Fabrication Costs.

Reference values for unit fuel fabrication costs were taken as \$40/kg for core material and \$30/kg for blanket material. These charges apply to unclad fuel spheres and include the cost of converting fuel to carbide form, but exclude fuel loss and shipping charges. The higher cost of fabricating core fuel as compared with blanket fuel is a result of the increased conversion costs which apply to $\text{Pu}(\text{NO}_3)_4$ as well as the increased cost of handling Pu material. The effect on fuel cycle economics of increased unit fabrication costs was determined by also considering \$120/kg for core material and \$90/kg for blanket material. It is felt, however, that unit fabrication costs will ultimately be nearer to the reference values.

Maximum Fuel Exposure

The reference value for maximum exposure was assumed to be 100,000 MWD/T, but, since an allowable value for the SBFR fuel has not been established as yet, maximum exposures of 75,000, 50,000, and 25,000 MWD/T were also considered. Other fast reactor design studies^{29,30} have assumed average exposures of 100,000 MWD/T for UC-PuC fuel, so the reference value chosen seems reasonable.

Number of Cores Irradiated per Blanket

For each maximum fuel exposure considered, the number of cores irradiated per blanket loading was varied until minimum fuel cost, averaged over the blanket cycle, was achieved. This optimum value for CPB obviously results in the most economic reactor operation.

Fuel Ownership

Two sets of general economic conditions were used to estimate the effect of a change in the fuel ownership rules. One set, termed the 1964 basis, considers fuel to be lease from the AEC, while the other set, the 1975 basis, features private ownership of all nuclear material. Other differences between the two cost bases are as follows:

(1) In 1964 the separations charge for the processing of irradiated fuel material in an AEC installation was \$18,000/day, whereas it is estimated that processing will be carried out in a privately owned plant in 1975 at a daily charge of \$23,500.

(2) The annual interest rate on working capital is taken as 6% for 1964 and 10% for 1975.

(3) The cost of Pu, which is \$10/g ($\text{Pu}^{239} + \text{Pu}^{241}$) and \$0/g ($\text{Pu}^{240} + \text{Pu}^{242}$) under 1964 conditions, is taken as \$6.60/g ($\text{Pu}^{239} + \text{Pu}^{241}$) and \$11.10/g ($\text{Pu}^{240} + \text{Pu}^{242}$) for 1975. The latter values are the result of estimates made by General Electric of the economic conditions which will exist in 1975.³¹

Reactor Design

Both the axial and radial flow concepts were considered in the economic study, but emphasis was placed on the axial flow reactor. The complete range of parameter variation discussed above was considered for the axial flow design. For the radial flow design, the only maximum fuel exposure considered was 100,000 MWD/T, but the other parameters were varied as previously described.

Table 11

Summary of Cost Parameters

Total thermal power output	2500 MW
Over-all thermal efficiency	40%
Estimated plant factor	0.8
Maximum fuel exposure	25;50;70;100 GWD/T
Number of cores irradiated per blanket	variable
Unit fabrication cost:	
Core fuel	40;120 \$/kg
Blanket fuel	30;90 \$/kg
Separation plant charge	18,000 \$/day (1964) ^a 23,500 \$/day (1975) ^b
Cost of $\text{Pu}^{239} + \text{Pu}^{241}$	10.0 \$/g (1964) ^a 6.60 \$/g (1975) ^b
Cost of $\text{Pu}^{240} + \text{Pu}^{242}$	0 \$/g (1964) ^a 1.10 \$/g (1975) ^b
Wt % U^{235} in depleted U	0.3
Cost of depleted U	3.0 \$/kg
Interest rate on working capital	6.0 %/year (1964) ^a 10.0 %/year (1975) ^b
Use charge rate for leased fuel	4.75 %/year
Fabrication plant throughput rate:	
Core	2000 kg/mo
Blanket fuel	4000 kg/mo
Separation plant throughput rate, kg/day	see Ref. 27
Fraction of total fabrication cost used to estimate fabrication working capital	0.5
Shipping charges:	
To fabricator	1.50 \$/kg
To reactor and scrap recycle	1.50 \$/kg
To processing	16.0 \$/kg
Return for credit	1.50 \$/kg
Shipping times:	
To fabricator	20 days
To reactor and scrap recycle	20 days
To processing	20 days
Turnaround time for separation plant	7 days
Lag times:	
In fabrication	60 days
In separations	30 days
Holdup at reactor before loading	30 days
Decay cooling time	120 days
Irrecoverable fuel losses:	
During fabrication	2.0 %
During separations	1.0 %
Scrap recycled from fabrication	10.0 %
Fraction of annual throughput maintained as spares	0.05%

^a1964 refers to present-day economic conditions.

^b1975 refers to assumed future conditions.

Cost Parameters and Mass Data

Table 11 summarizes the values assigned to various parameters used in the cost study. Where applicable, values for both the 1964 and 1975 bases are presented.

Table 12 includes information on isotopic masses for a number of the cases studied. In addition, f values are listed for each maximum fuel exposure, and the fraction of reactor power which is generated in the core, averaged over an entire blanket cycle, is given for each value of CPB. The average core exposure and the full power irradiated time for both core and blanket batches are also listed. The isotopic masses listed for the core include the effect of fuel additions over the core lifetime.

Discussion of Results

Axial Flow Design. Figures 31 through 34 show the variation of fuel cost with maximum fuel exposure, number of cores irradiated per blanket

(CPB), unit fabrication cost, and fuel ownership basis. The minimum attainable fuel costs shown in Figure 35 correspond to the optimum CPB value for any set of conditions selected. A detailed breakdown of fuel costs determined for the reference conditions is presented in Table 13 for both the 1964 and 1975 bases. The reference conditions include the discharge of each blanket loading after every second core cycle (CPB = 2), a procedure which minimizes fuel costs for 100,000 MWD/T operation.

Figures 31 through 34 indicate the existence of an optimum CPB value for each set of economic conditions considered. The minimum fuel cost corresponding to the optimum CPB results from opposing effects on the blanket fuel costs as CPB increases. Blanket fabrication, reprocessing, and shipping costs, in mills/kWh, tend to decrease as CPB increases. This results from the corresponding increase in average blanket fuel exposure. On the other hand, as CPB is increased, the credit for bred Pu decreases and nuclear material working

Table 12
Burnup Data for Various Cases Studied

Coolant flow direction	Radial		Axial		Axial		Axial		Axial	
Max core exposure, MWD/T	100,000		100,000		75,000		50,000		25,000	
f-Value	0.300		0.133		0.100		0.067		0.033	
Av core exposure, MWD/T	76,920		88,240		68,180		46,875		24,190	
Fuel charged to core, kg:										
U ²³⁸	28,562		21,373		20,744		20,115		19,486	
Pu ²³⁹	3,250		2,566		2,491		2,416		2,340	
U ²³⁸ charged to blanket, kg	143,000		107,127		107,127		107,127		107,127	
Fuel discharged from core, kg:										
U ²³⁸	25,046		18,784		18,852		18,891		18,886	
Pu ²³⁹	3,004		2,307		2,306		2,300		2,285	
Pu ²⁴⁰	335		272		211		145		75	
Pu ²⁴¹	23		18		11		6		2	
Pu ²⁴²	1		1		*		*		*	
Number of cores per blanket	2	3	2	3	3	4	4	6	8	13
Av core power/total power	0.906	0.886	0.937	0.920	0.933	0.920	0.937	0.920	0.937	0.916
Av core lifetime, F.P. days	1,081	1,105	902	918	679	689	450	459	226	230
Blanket residence, F.P. days	2,162	3,315	1,803	2,755	2,037	2,755	1,803	2,755	1,803	2,990
Fuel discharged from blanket, kg:										
U ²³⁸	138,716	136,623	103,960	102,412	103,571	102,412	103,960	102,412	103,960	102,028
Pu ²³⁹	3,660	5,170	2,800	3,967	3,106	3,967	2,800	3,967	2,800	4,237
Pu ²⁴⁰	84	181	69	148	86	148	69	148	69	171
Pu ²⁴¹	1	4	1	3	2	3	1	3	1	4
Pu ²⁴²	*	*	*	*	*	*	*	*	*	*

*Negligible.

Figure 31. Fuel costs for maximum core fuel exposure = 100,000 MWD/T.

Figure 32. Fuel costs for maximum core fuel exposure = 75,000 MWD/T.

Figure 33. Fuel costs for maximum core fuel exposure = 50,000 MWD/T.

Figure 34. Fuel costs for maximum core fuel exposure = 25,000 MWD/T.

Figure 35. Minimum attainable fuel cost as function of maximum core fuel exposure – axial flow design.

capital costs increase, when both items are expressed in mills/kWh. The Pu credit decreases because of the asymptotic behavior of the blanket Pu content with increasing irradiation time; however, as the blanket residence time increases, a larger working capital cost is incurred because of the deferred receipt of Pu credit.

Fuel costs determined for the 1975 economic basis are higher than those for the 1964 conditions mainly because of higher working capital costs and reduced credit for bred Pu. The extreme sensitivity of fuel costs to the unit fabrication charges used is obvious from the figures and provides a strong incentive for the development of minimum cost fabrication techniques for the fuel spheres.

For each maximum fuel exposure considered, a larger optimum CPB value is found for the higher fabrication charges than for the reference fabrication charges. This shift to a longer blanket residence time, hence to a higher blanket exposure, is required to minimize the effect of the increased cost of fabricating blanket fuel.

Figure 35 indicates the strong dependence of fuel costs on the maximum core fuel exposure. No minima exist in the cost vs exposure curves since the periodic addition of fuel to the core makes it unnecessary to increase the initial enrichment in order to achieve a higher core discharge exposure. The effect on fuel costs of adding an increased

Figure 36. Minimum attainable fuel cost as function of maximum core fuel exposure – radial flow design.

amount of fuel to achieve a higher exposure is overridden by the reductions in fabrication, shipping, and reprocessing costs (in mills/kWh) which result from the larger energy yield per unit mass of fuel.

Table 13 indicates that extremely low fuel cycle operating costs can be expected. Although working capital charges are higher, the over-all fuel costs are attractive for both the 1964 and 1975 conditions.

It is of interest to note from Table 12 that the f value for the axial flow design is 0.133 for a maximum core fuel exposure of 100,000 MWD/T and is proportionately smaller for lower exposures. Thus, the core bed height is increased by $\approx 13\%$ over the core lifetime. This value could be lowered by choosing a larger core size, hence a larger core breeding ratio.

Radial Flow Design. Figure 36 shows the variation of fuel cost with CPB, with the fuel ownership basis, and with unit fabrication costs, for a maximum exposure of 100,000 MWD/T. The cost results are higher than those for the radial case (Figure 31) at 100,000 MWD/T mainly because of the larger cumulative mass of fuel which must be added over the core lifetime. From Table 12, the f value for this case is 0.30, which results in an average fuel exposure of 76,920 MWD/T. The corresponding average exposure for the axial flow

Table 13

Breakdown of Fuel Costs for Reference Conditions

Maximum core fuel exposure = 100,000 MWD/T

2 cores irradiated per blanket loading

Unit fabrication charges = \$40/kg core fuel, \$30/kg blanket fuel

Note: Costs are given in mills/kWh

	1964 Basis			1975 Basis		
	Core	Blanket	Total	Core	Blanket	Total
Fuel cycle costs						
Fabrication	0.068	0.074	0.142	0.060	0.074	0.134
Fuel depletion	0.111	-0.647	-0.536	0.060	-0.429	-0.369
Shipping	0.021	0.051	0.072	0.021	0.051	0.072
Reprocessing	0.078	0.059	0.137	0.095	0.072	0.167
Inventory charges						
Prereactor	0.052	0.001	0.053	0.073	0.001	0.074
At reactor	0.190	0.002	0.192	—	—	—
Postreactor	0.017	0.000	0.017	—	—	—
Subtotal	0.537	-0.460	0.077	0.309	-0.231	0.078
Working capital costs						
Core fabrication	0.011	0.015	0.026	0.021	0.026	0.047
Nuclear material	-0.015	0.127	0.112	0.267	0.127	0.394
Subtotal	-0.004	0.142	0.138	0.288	0.153	0.441
Total fuel costs	0.533	-0.318	0.215	0.597	-0.078	0.519

case is 88,240 MWD/T. Consequently, all cost items which are affected directly by the core fuel exposure are larger for the radial flow case. A larger Pu credit was determined for the radial flow case but it does not balance the core fuel exposure effect when the results are compared with those for the axial flow case.

The difference in f values for the two designs is a direct result of their basic geometrical dissimilarity. A small increase in the height of the axial flow core can give a substantial positive reactivity effect since there is large axial leakage in a pancake configuration and relatively little leakage in the radial direction. For the radial flow design, leakage is large radially and small axially because of the tubular shape of the core; thus, reactivity is only slightly sensitive to small bed height variations.

At fuel exposures of 100,000 MWD/T, the f value for the radial flow design would be even larger had the reactivity effect of blanket Pu not been included. The substantial mass of Pu^{239} which builds up in the portion of the radial blanket near the core is coupled closely enough with the core to effect a significant reduction in the re-

activity swing during irradiation, provided that the radial blanket bed is not mixed during its residence in the reactor. This effect can reduce the reactivity loss to about one-third of the loss which results when the radial blanket is periodically fluidized and mixed. For no blanket mixing, the rate of reactivity change with core exposure is approximately $-1.1 \times 10^{-7} (\Delta k/k) / \Delta(\text{MWD/T})$. Even for this small reactivity loss rate, an f value of 0.30 is required to reach a maximum core exposure of 100,000 MWD/T.

On the other hand, no reactivity contribution was assumed from axial blanket Pu buildup in the axial flow case, although this buildup is substantial. The presence of sodium plena between the core and axial blanket regions results in poor neutron coupling between these regions and minimizes the blanket reactivity effect. The rate of reactivity change for this design is about $-4.8 \times 10^{-7} (\Delta k/k) / \Delta(\text{MWD/T})$. However, the f value is only 0.133 for operation to 100,000 MWD/T.

For both designs, sodium temperature coefficients will become less negative (or more positive) during operation. For the radial flow case, the effective core diameter will increase if the

radial blanket is not mixed periodically, thereby decreasing the over-all core leakage. Core leakage is also reduced for the axial flow design as fuel is added and the core height increases.

In general, the considerations above and their effect on fuel costs indicate that the axial flow design is the more promising of the two.

Table 14 lists fuel costs for the axial flow SBFR and those for three other proposed 1000-MW(e) reactors, namely, the General Electric²² and Westinghouse³² fast breeder concepts and the Westinghouse PWR.³² All costs were determined for economic conditions which correspond to the 1964 basis. The PWR results were included as a point

of interest and would not be expected to compare favorably with the other reactors in fuel costs. For the SBFR, a separate listing of the costs is given for each set of unit fabrication costs considered. The cost figures which should be compared among the breeder reactors are the "subtotal" results, rather than the final "fuel costs," because the working capital costs are complete only for the SBFR. No estimate was made of the working capital requirement for delayed receipt of Pu credit for any other fast reactors. The SBFR costs based on the smaller unit fabrication charges are easily the most favorable, despite the lower average fuel exposure chosen for the SBFR relative to the other fast

Table 14

Fuel Costs for Various 1000-MW(e) Reactor Concepts

Note: All costs are in mills/kWh and are based on 1964 (leased fuel) economic conditions

Fuel material: Average core exposure:	GE fast breeder UO ₂ -PuO ₂ 110,000 MWD/T			WAPD fast breeder UC-PuC 100,000 MWD/T			WAPD PWR UO ₂ 24,000 MWD/T
	Core	Blanket	Total	Core	Blanket	Total	Total
Fabrication	0.248	0.096	0.344	0.209	0.161	0.370	0.78
Depletion	0.322	-0.510	-0.188	0.312	-0.885	-0.573	0.58
Shipping	0.034	0.064	0.098	0.013	0.030	0.043	0.07
Reprocessing	0.036	0.060	0.096	0.116	0.082	0.198	0.24
Inventory	0.191	0.000	0.191	0.325	0.000	0.325	0.16
Fabrication working capital	0.019	0.010	0.029	0.007	0.007	0.014	0.19
Subtotal	0.850	-0.280	0.570	0.982	-0.605	0.377	2.02
Nuclear working capital	*	*	*	*	*	*	negligible
Fuel cost	*	*	*	*	*	*	2.02

Fuel material: Average core exposure:	SBFR (reference conditions) UC-PuC 88,250 MWD/T			SBFR (increased fabrication charges) UC-PuC 88,250 MWD/T		
	Core	Blanket	Total	Core	Blanket	Total
Fabrication	0.068	0.074	0.142	0.157	0.223	0.380
Depletion	0.111	-0.647	-0.536	0.111	-0.647	-0.536
Shipping	0.021	0.051	0.072	0.021	0.051	0.072
Reprocessing	0.078	0.059	0.137	0.078	0.059	0.137
Inventory	0.259	0.003	0.262	0.259	0.003	0.262
Fabrication working capital	0.011	0.015	0.026	0.020	0.043	0.063
Subtotal	0.548	-0.445	0.103	0.646	-0.268	0.378
Nuclear working capital	-0.015	0.127	0.112	-0.015	0.127	0.112
Fuel cost	0.533	-0.318	0.215	0.631	-0.141	0.490

*Nuclear material working capital charge not estimated so total fuel cost is incomplete.

reactors. Costs for the SBFR based on the larger unit fabrication charges are still highly competitive.

POWER GENERATION COSTS

Introduction

The estimated total power generation costs are presented herein for the plant design utilizing axial coolant flow, (U-Pu)C fuel, and with a net generating capacity of 1000 MW(e). The fuel cycle costs given are for both AEC and private fuel ownership, a maximum fuel exposure of 100,000 MW/tonne of U+Pu, and a scheme in which the blanket remains in the reactor for two core lifetimes. This procedure gives a blanket residence time which minimizes over-all fuel costs. Reference fabrication charges for carbide fuel spheres are \$40/kg for core fuel and \$30/kg for blanket fuel. The total costs of generating power were determined to be 3.84 mills/kg-hr for AEC leased fuel and 4.14 mills/kg-hr when fuel is privately owned. A summary of power costs is given in Table 15.

An annual load factor of 0.80 was assumed; however, the SBFR concept has the potential for significantly less refueling downtime than fixed

fuel reactors. This potential is due to the ability to refuel remotely by hydraulic transport and periodically mix the fuel, *in situ*, by fluidization, for more uniform burnup.

An increase in plant use factor to 0.805 from the conventional value of 0.80, an increase in plant availability of 2 days/year, is equivalent to a capital cost reduction of a \$150,000,000 plant by more than \$1,000,000.

Capital Costs

The estimated total capital cost for the reference design is \$156,062,000. This estimate, given in Tables 16 and 17, was prepared in accordance with the uniform system of accounts established by the AEC, Division of Finance, and in accordance with the ground rules and procedures specified in the AEC Nuclear Power Plant Cost Handbook.²⁶ Accordingly, an annual fixed charge rate of 14.5% is used for depreciating capital costs and 13% for nondepreciating items. Equipment costs were developed by Burns & Roe utilizing the plant layouts prepared for this study, and utilizing their experience in the design and construction of conventional power plants.

The cost of land and land rights was not included in Account 20, because most of the mark-

Table 15
Power Generation Costs

	Fuel leased from AEC		Fuel owned privately	
	Annual cost (\$1,000)	Unit cost (mills/kWh)	Annual cost (\$1,000)	Unit cost (mills/kWh)
Fixed charges				
Depreciating capital				
Total capital cost (less land and land rights)	22,600	3.22	22,600	3.22
Nondepreciating capital				
Land and land rights	52	0.01	52	0.01
Working capital				
(a) Plant operation and maintenance	29	0.01	29	0.01
(b) Fuel cycle operations	905	0.14	3,088	0.44
Annual nuclear liability insurance	360	0.05	360	0.05
Subtotal - fixed charges	24,026	3.43	26,129	3.73
Operating costs				
Operating and maintenance cost	2,309	0.33	2,309	0.33
Fuel cost	560	0.08	561	0.08
Subtotal - operating costs	2,869	0.41	2,870	0.41
Total power generation costs	26,895	3.84	28,999	4.14

Table 16

Settled Bed Fast Reactor, Summary of Capital Costs

Acct. No.	Description	Material	Labor	Total
20.	Land and land rights		Included below	
21.	Structures and improvements	\$ 5,648,000	\$ 2,741,000	\$ 8,389,000
22.	Reactor plant equipment	57,070,900	7,864,100	64,935,000
23.	Turbine generator units	26,077,600	3,666,400	29,744,000
24.	Accessory electric equipment	3,415,000	1,069,200	4,484,000
25.	Miscellaneous power plant equipment	409,000	207,000	616,000
	Total direct cost	\$92,620,000	\$15,547,500	\$108,168,000
	Indirect construction cost			
	General and administrative - 4%			4,327,000
	Subtotal			\$112,495,000
	Miscellaneous construction costs - 2%			2,250,000
	Subtotal			\$114,745,000
	Architectural and engineering services - 3.85%			4,417,000
	Subtotal			\$119,162,000
	Nuclear engineering - 3.6%			4,290,000
	Subtotal			\$123,452,000
	Start-up costs			600,000
	Subtotal			\$124,052,000
	Land and land rights			400,000
	Subtotal			\$124,452,000
	Contingency - 10%			12,445,000
				\$136,897,000
	Interest during construction - 14%			19,165,000
	Total capital costs			\$156,062,000

ups in the indirect cost portion of the estimate do not apply to the land. Since no specific site has been established, the cost of land was set at a reasonable value for a plant of this size. Since land cost is only a small portion of the total capital cost, any error has negligible effect on the over-all estimate.

In Account 21, the ground improvements costs were estimated on the basis of the site plan. This plan was also used to estimate floor areas for the buildings which were then priced on a square-foot basis. The unit prices for the buildings were based on the complexity and end use of the respective structures. The exception to this method of pricing structures was the reactor containment. The nominal wall thickness was calculated on the basis of the estimated internal pressure. This thickness was used to estimate the weight and the containment was priced on the basis of unit weight.

Primary and intermediate system piping for Account 22 was estimated from the layouts of the

plant. The intermediate heat exchanger, steam generators, and reheaters were priced on the basis of heat transfer surface. Pump prices were established from horsepower requirements. Many other items in this account were estimated on the basis of the plant layouts and flow diagrams. However, because of the lack of layouts, it was necessary to estimate the items in Accounts 225 to 229 on the basis of experience with similar systems. As an example, the price of steam and condensate piping in Account 228 represents an average of similar costs for conventional plants.

The turbine generator and condenser costs were estimated on the basis of capacity and steam conditions. The circulating water system cost is based on the limited information on the site plan and on the cost of similar large installations.

Accessory electrical equipment and miscellaneous power plant equipment prices are based on those for conventional power plants of similar size.

Table 17

Settled Bed Fast Reactor, Capital Cost Breakdown

Acct. No.	Description	Material	Labor	Total
21	Structures and improvements			
211	Ground improvements			
.1	Access roads - permanent use	\$ 59,000	\$ Included	\$ 59,000
.2	General yard improvements	160,000	120,000	280,000
.3	Railroad	55,000	Included	55,000
.5	Yard service and			
.9	miscellaneous	140,000	80,000	220,000
	Total cost item 21	414,000	200,000	614,000
212	Buildings			
A.	Sodium service building	275,000	Included	275,000
	Building services	75,000	50,000	125,000
B.	Offices and services building	384,000	Included	384,000
	Building services	150,000	135,000	285,000
C.	Fuel handling and hot shop	480,000	Included	480,000
	Building services	80,000	40,000	120,000
D.	Warehouse	120,000	Included	120,000
	Building services	20,000	18,000	38,000
E.	Turbine building	1,350,000	Included	1,350,000
	Building services	175,000	225,000	40,000
F.	Heating plant	31,000	Included	31,000
	Building services	8,000	9,000	17,000
G.	Water treatment building	105,000	Included	105,000
	Building services	5,000	7,000	12,000
H.	Steam generator building	480,000	Included	480,000
	Building services	160,000	140,000	300,000
I.	Auxiliary equipment building	300,000	Included	300,000
	Building services	120,000	140,000	260,000
J.	Reactor containment	916,000	1,777,000	2,693,000
	Total item 212	5,234,000	2,541,000	7,775,000
	Total cost account 21	\$ 5,648,000	\$ 2,741,000	\$ 8,389,000
22	Reactor plant equipment			
221	Reactor equipment			
.1	Reactor vessel	3,970,000	80,000	4,050,000
.2	Reactor control	1,500,000	200,000	1,700,000
.3	Reactor shielding	760,000	507,000	1,267,000
.4	Reactor auxiliary cooling and heating systems	551,000	332,000	883,000
.7	Reactor plant cranes and hoists	200,000	35,000	235,000
.8	Equipment foundations	20,000	30,000	50,000
.9	Painting	10,000	40,000	50,000
	Total cost item 221	7,011,000	1,224,000	8,235,000
222	Heat transfer systems			
.1	Reactor coolant system	8,253,000	278,000	8,531,000
.2	Intermediate coolant system	11,976,000	329,000	12,305,000
.3	Steam generators and reheaters	21,627,000	245,000	21,872,000
.4	Reactor coolant receiving, supply and treatment	339,000	135,000	474,000
.6	Coolant, initial charge	170,000	50,000	220,000
.8	Equipment foundations	45,000	60,000	105,000
.9	Painting	8,000	24,000	32,000
	Total cost item 222	42,418,000	1,121,000	43,539,000

Table 17 (Continued)

Settled Bed Fast Reactor, Capital Cost Breakdown

Acct. No.	Description	Material	Labor	Total
223	Nuclear fuel handling and storage equipment			
.1	Cranes and hoisting equipment	70,000	20,000	90,000
.2	Special tools and service equipment			
.3	Spent fuel storage, cooling, cleaning and inspection equipment	1,429,000	281,000	1,710,000
.4	Shipping casks and cars	190,000	136,000	326,000
	Total cost item 223	402,900	184,100	587,000
225	Radioactive waste treatment and disposal			
.1	Liquid waste	50,000	15,000	65,000
.2	Gaseous waste	90,000	30,000	120,000
.3	Solid waste	210,000	110,000	320,000
.5	Equipment foundations and shield walls	*85,000	120,000	205,000
	Total cost item 225	435,000	275,000	710,000
226	Instrumentation and controls			
.1	Reactor control including in-core instrumentation	1,350,000	220,000	1,570,000
.2	Reactor plant instrumentation for heat transfer, waste disposal and auxiliary systems	400,000	60,000	460,000
.3	Control panels	12,000	3,000	15,000
.4	Secondary plant	85,000	40,000	125,000
.5	Radiation monitoring	60,000	8,000	68,000
.6	Steam generator controls	280,000	150,000	430,000
.7	Control and instrument piping, tubing, wiring, etc.	325,000	240,000	565,000
	Total cost item 226	2,512,000	721,000	3,233,000
227	Feedwater supply and treatment	675,000	197,000	872,000
228	Steam, condensate and feedwater piping	3,326,000	4,064,000	7,390,000
229	Other reactor plant equipment	291,000	78,000	369,000
	Total cost account 22	\$57,070,900	\$ 7,864,100	\$64,935,000
23	Turbine generator units			
231	Turbine generators			
.1	Foundation	200,000	210,000	
.2	Turbine generator	20,700,000	1,000,000	
.3	Standby exciter	—	—	
.8	Foundations	—	—	
	Total cost item 231	20,900,000	1,210,000	22,110,000
232	Circulating water system			
.1	Pumping and regulating equipment	793,600	165,800	
.2	Circulating water lines	482,000	775,000	
.3	Intake and discharge structures	429,000	643,000	
.4	Fouling, corrosion, control, and water treatment systems	20,000	5,000	
.5	Cooling towers	—	—	
.8	Foundations	—	—	
	Total cost item 232	1,724,000	1,588,800	3,313,400
233	Condensers	2,250,000	275,000	2,525,000
234	Central lubricating system		In item 231	
235	Turbine plant boards, instruments and controls	529,000	210,000	739,000
236	Turbine plant piping	210,000	235,000	445,000
237	Auxiliary equipment for generators	344,000	43,000	387,000
238	Other turbine plant equipment	110,000	70,000	180,000
239	Miscellaneous	10,000	35,000	45,000
	Total cost account 23	\$ 26,077,600	\$ 3,666,800	\$ 29,744,400
	Turbine generator units			

Table 17 (Continued)

Settled Bed Fast Reactor, Capital Cost Breakdown

Acct. No.	Description	Material	Labor	Total
24	Accessory electric equipment			
241	Switchgear			
.1	Generator main and neutral circuits			
.2	Station service			
.3	Unit substations			
	Total cost item 241	1,420,000	38,000	1,458,000
242	Switchboards			
.1	Main control board			
.2	Auxiliary power, battery and signal boards			
.3	Motor control centers			
	Total cost item 242	530,000	22,000	552,000
243	Protective equipment	33,000	31,000	64,000
244	Electrical structures	75,000	110,000	185,000
245	Conduit	350,000	525,000	875,000
246	Power and control wiring	774,000	301,000	1,075,000
247	Station service equipment	225,000	30,000	255,000
249	Miscellaneous	8,000	12,000	20,000
	Total cost account 24	\$ 3,415,000	\$ 1,069,000	\$ 4,484,000
25	Miscellaneous power plant equipment			
251	Cranes and hoisting equipment	110,000	40,000	150,000
252	Compressed air and vacuum	55,000	60,000	115,000
253	Other power plant equipment	230,000	85,000	315,000
254	Foundations	12,000	14,000	26,000
255	Painting	2,000	8,000	10,000
	Total cost account 25	\$ 409,000	\$ 207,000	\$ 616,000

The indirect costs applied to the direct costs listed above are typical for a project of this size. These are generally lower than those suggested by the AEC in TID-7025.

As a check on the validity of the cost estimate, a comparison was made with two other cost estimates, the Westinghouse 1000-MW(e) Pressurized Water Reactor,³² and the General Electric 1000-MW(e) Liquid Metal Fast Oxide Breeder Reactor.²² Table 18 summarizes a comparison of costs for these three reactors. Within the accuracy of these estimates it may be noted that the total costs for these plants are quite similar. In comparing the SBFR with the PWR, it is evident that the major cost differences may be found in Account 22, Reactor Plant Equipment, which shows a cost difference of \$14,575,000 in favor of the PWR, and in Account 23, Turbine Generator Units,

which shows a cost difference of \$8,102,000 in favor of SBFR. These somewhat compensating costs account for the 3% difference in the total cost for these two plants. Because of the detailed nature of the Westinghouse study, it was possible to make a comparison utilizing a breakdown of the individual accounts. This breakdown is given in Table 19. Major items of difference are the reactor vessel, 2½ times higher for the PWR, a primary and secondary coolant system, almost twice as high for the SBFR, and the turbine generator, \$5.6 million higher for the PWR because of the relatively low pressure steam obtained in a pressurized water cycle as compared to the high temperature, high pressure steam from the SBFR. A more direct comparison with the General Electric study was not possible since a detailed breakdown of costs was not available.

Table 18
Comparison of Capital Costs

Acct. No.	Description	1000-MW settled bed reactor	G.E. 1000-MW fast oxide reactor	Westinghouse 1000-MW closed-cycle PWR
20.	Land and land rights			
21.	Structures and improvements	\$ 8,389,000	\$ 13,456,000	\$ 12,157,000
22.	Reactor plant equipment	64,935,000	35,227,000	50,340,700
23.	Turbine generator units	29,744,000	18,834,000	37,846,000
24.	Accessory electric equipment	4,484,000	6,103,000	3,786,900
25.	Miscellaneous power plant equipment	616,000	13,463,000	608,900
	Total direct cost	\$108,168,000	\$ 87,083,000	\$104,739,500
	Indirect construction cost			
	General and administrative 4%	4,327,000	3,483,000	4,189,500
	Subtotal	\$112,495,000	\$ 90,566,000	\$108,929,000
	Miscellaneous construction costs 2%	2,250,000	1,811,000	2,178,000
	Subtotal	\$114,745,000	\$ 92,377,000	\$111,107,000
	Architectural and engineering services 3.85%	4,417,000	3,556,000	4,278,000
	Subtotal	\$119,162,000	\$ 95,933,000	\$115,385,000
	Nuclear engineering 3.6%	4,290,000	3,454,000	4,154,000
	Subtotal	\$123,452,000	\$ 99,387,000	\$119,539,000
	Start-up costs	600,000	600,000	600,000
	Subtotal	\$124,052,000	\$ 99,987,000	\$120,139,000
	Land and land rights	400,000	220,000	360,000
	Subtotal	\$124,452,000	\$100,207,000	\$120,499,000
	Contingency 10%	12,445,000	10,021,000	12,050,000
		\$136,897,000	\$110,228,000	\$132,549,000
	Interest during construction 14%	19,165,000	15,432,000	18,557,000
	Total capital costs	\$156,062,000	\$125,660,000	\$151,106,000

Operation and Maintenance Costs

The annual operation and maintenance cost is estimated at \$2,309,000 per year or 0.33 mills/kWh. As shown in Table 20, the cost of plant supervision, engineering, and station labor is based on an estimated plant operating staff of 67 persons. Salaries and wage rates correspond to those given in the AEC Cost Evaluation Handbook. The cost of fringe benefits and payroll taxes for station personnel was taken as 20% of the total payroll as specified by the AEC. These costs are similar to those used in the Westinghouse study,³² and it must be assumed that sufficient operating experience must be obtained before these costs could be considered realistic for the SBFR.

To determine the annual cost for sodium replacement, it was assumed that when the system

is opened, oxygen is dissolved until a concentration of 0.5% or 5000 ppm is reached. It was also assumed that on start-up the dissolved oxygen is reduced to 10 ppm by the cold trap; therefore, the amount to be removed is about 5000 ppm. The initial sodium charge is 1,000,000 lb. Thus, about 5000 lb of oxygen must be removed. The amount of sodium loss is then $(22/16) \times 5000 = 6875$ lb. At \$0.17/lb, the cost of sodium is \$1170, say \$1200. If the system is opened once every three years this cost comes to about \$400/year. Putting this into the system would cost about another \$100/year.

Based on the Westinghouse study, the annual cost of maintenance, materials, and operating supplies is \$1,721,000. With the addition of the cost of sodium replacement, the approximate annual total cost is \$1,722,000.

Table 19

Cost Comparison Breakdown

Acct. No.	Description	Settled bed fast reactor			Closed cycle reactor - pressurized water		
		Material	Labor	Total	Material	Labor	Total
221							
.1	Reactor vessel	\$ 3,970,000	\$ 80,000	\$ 4,050,000	\$10,748,800	\$ 99,000	\$10,847,800
.2	Reactor controls	1,500,000	200,000	1,700,000	5,266,800	20,000	5,286,800
.3	Reactor shielding	760,000	507,000	1,267,000	18,000	15,000	33,000
.4	Reactor auxiliary cooling and heating systems	551,000	332,000	883,000	766,500	82,000	848,500
.7	Reactor plant cranes and hoists	200,000	35,000	235,000	324,000	33,000	357,000
.8	Equipment foundations	20,000	30,000	50,000	5,000	3,000	8,000
.9	Painting	10,000	40,000	50,000	8,000	42,000	50,000
	Totals -	\$ 7,011,000	\$1,224,000	\$ 8,235,000	\$17,137,100	\$ 294,000	\$17,431,100
222							
.1	Reactor coolant system	\$ 8,253,000	\$ 278,000	\$ 8,531,000	\$ 7,978,000	\$ 385,000	\$ 8,363,100
.2	Intermediate coolant system	11,976,000	329,000	12,305,000	—	—	—
.3	Steam generators and reheaters	21,627,000	245,000	21,872,000	11,975,200	80,000	12,055,000
.4	Reactor coolant receiving supply and treatment	339,000	135,000	474,000	—	—	—
.6	Coolant, initial charge	170,000	50,000	220,000	—	—	—
.8	Equipment foundations	45,000	60,000	105,000	62,000	33,000	95,000
.9	Painting	8,000	24,000	32,000	6,000	24,000	30,000
	Totals -	\$42,418,000	\$1,121,000	\$43,539,000	\$22,821,400	\$ 536,000	\$23,357,400
Note: Equipment covered by accounts 222.5 and 222.7 not used in these plants.							
223							
.1	Cranes and hoisting equipment	\$ 70,000	\$ 20,000	\$ 90,000	\$ 347,800	\$ 40,000	\$ 387,800
.2	Special tools and service equipment	—	—	—	573,000	29,000	602,000
.3	Spent fuel storage, cooling, cleaning, and inspection equipment	142,900	28,100	171,000	4,000	1,000	5,000
.4	Shipping casks and cars	190,000	136,000	326,000	—	—	—
	Totals -	\$ 402,900	\$ 184,100	\$ 587,000	\$ 924,800	\$ 70,000	\$ 994,800
225							
.1	Liquid waste	50,000	15,000	65,000	158,000	38,000	196,000
.2	Gaseous waste	90,000	30,000	120,000	54,000	5,000	59,000
.3	Solid wastes	210,000	110,000	320,000	3,000	1,000	4,000
.5	Equipment foundations and shield walls	85,000	120,000	205,000	60,000	52,000	112,000
	Totals -	\$ 435,000	\$ 275,000	\$ 710,000	\$ 275,000	\$ 96,000	\$ 371,000
226							
.1	Reactor control including in-core instrumentation	\$ 1,350,000	\$ 220,000	\$ 1,570,000	\$ 1,842,900	\$ 61,000	\$ 1,903,000
.2	Reactor plant instrumentation for heat transfer, waste disposal and auxiliary systems	400,000	60,000	460,000	416,500	13,000	429,500
.3	Control panels	12,000	3,000	15,000	2,000	14,000	16,000
.4	Secondary plant	85,000	40,000	125,000	190,000	96,000	286,000
.5	Radiation monitoring	60,000	8,000	68,000	77,800	9,000	86,800
.6	Steam generator controls	200,000	150,000	430,000	216,600	—	216,600
.7	Control and instrument piping, tubing, wiring etc.	325,000	240,000	565,000	105,300	110,000	215,300
	Totals -	\$ 2,512,000	\$ 721,000	\$ 3,233,000	\$ 2,851,000	\$ 303,000	\$ 3,154,000

Table 19 (Continued)

Cost Comparison Breakdown

Acct. No.	Description	Settled bed fast reactor			Closed cycle reactor - pressurized water		
		Material	Labor	Total	Material	Labor	Total
227							
	1 Raw water supply system				—	—	—
	2 Make-up water supply				\$ 1,000	\$ 3,000	\$ 4,000
	3 Steam generator feed-water purification and treatment system				32,000	11,000	43,000
	4 Feedwater heaters				53,700	31,000	84,700
	5 Feedwater pumps and drives				541,000	21,000	562,000
	6 Blowdown and blowoff equipment				2,000	—	2,000
	7 Miscellaneous tanks				19,000	4,000	23,000
	8 Equipment foundations				7,000	7,000	14,000
	9 Painting				12,000	58,000	70,000
	Totals -	\$ 675,000	\$ 197,000	\$ 872,000	\$ 667,000	\$ 135,000	\$ 802,700
228	Steam, condensate and feedwater piping	\$ 3,326,000	\$ 4,064,000	\$ 7,390,000	\$ 2,330,000	\$ 1,516,000	\$ 3,846,000
229	Other reactor plant equipment	291,000	78,000	369,000	306,600	77,000	383,600
	Total cost -	\$57,070,900	\$7,864,100	\$64,935,000	\$47,313,700	\$3,027,000	\$50,340,700
	Account 22 reactor plant equipment						
231							
	1 Foundation	\$ 200,000	\$ 210,000	\$ 410,000	\$ 210,000	\$ 225,000	\$ 435,000
	2 Turbine generator	20,700,000	1,000,000	21,700,000	26,550,000	737,000	27,287,000
	3 Standby exciter	—	—	—	2,000	2,000	4,000
	Totals -	\$20,900,000	\$1,210,000	\$22,110,000	\$26,762,000	\$ 964,000	\$27,726,000
232							
	1 Pumping and regulating equipment	\$ 793,600	\$ 165,800	\$ 959,400	\$ 1,159,000	\$ 34,000	\$ 1,193,000
	2 Circulating water lines	482,000	775,000	1,257,000	322,000	233,000	555,000
	3 Intake and discharge structures	429,000	643,000	1,072,000	31,000	9,000	40,000
	4 Fouling, corrosion, control and water treatment systems	20,000	5,000	25,000	19,000	3,000	22,000
	5 Cooling towers	—	—	—	3,000	7,000	10,000
	8 Foundations	—	—	—	—	—	—
	Totals -	\$ 1,724,600	\$1,588,800	\$ 3,313,400	\$ 1,534,000	\$ 280,000	\$ 1,820,000
233	Condensers	\$ 2,250,000	\$ 275,000	\$ 2,525,000	\$ 7,355,000	\$ 385,000	\$ 7,740,000
234	Central lubricating system	—	—	—	—	—	—
235	Turbine plant boards, instruments and controls	529,000	210,000	739,000	17,000	8,000	25,000
236	Turbine plant piping	210,000	235,000	445,000	216,000	156,000	372,000
237	Auxiliary equipment for generators	344,000	43,000	387,000	—	—	—
238	Other turbine plant equipment	110,000	70,000	180,000	106,000	7,000	113,000
239	Miscellaneous	10,000	35,000	45,000	5,000	45,000	50,000
	Total cost -	\$26,077,600	\$3,666,800	\$29,744,400	\$35,995,000	\$1,851,000	\$37,846,000
	Account 23 turbine generator units						

Table 20
Operating and Maintenance Costs^a

	Salary or wage rate	Personnel required	Annual expense
Plant management			
Station superintendent	\$15,000/yr	1	\$ 15,000
Assistant superintendent	12,000/yr	1	12,000
Clerk - storekeeper	2.50/hr	1	5,200
Secretary - typist	2.31/hr	1	4,800
Subtotal		4	\$ 37,000
Technical staff			
Nuclear engineer (L)	\$10,800/yr	1	\$ 10,800
Results engineer	8,000/yr	1	8,000
Health physics supervisor	8,400/yr	1	8,400
Laboratory technician	2.85/hr	3	17,900
Subtotal		6	\$ 45,100
Operating staff			
Shift supervisor (L)	\$10,800/yr	5	\$ 54,000
Senior control operator (L)	3.75/hr	5	39,000
Control operator	3.65/hr	5	38,000
Turbine operator	3.50/hr	5	36,500
Equipment attendant	3.00/hr	5	31,200
Special operator	3.50/hr	8	58,400
Relief control operator (L)	3.65/hr	2	15,200
Relief equipment operator	3.50/hr	2	14,600
Janitor	2.25/hr	2	9,400
Watchman	2.25/hr	5	23,500
Subtotal		44	\$ 319,800
Maintenance staff			
Maintenance superintendent	\$10,800/yr	1	\$ 10,800
Instrument mechanic	3.25/hr	3	20,400
Electrician	3.25/hr	3	20,400
Steam fitter - welder	3.25/hr	2	13,600
Mechanics helper	2.65/hr	4	22,000
Subtotal		13	\$ 87,200
Total labor		67	489,100
Fringe benefits, 20%			97,900
Total labor and fringe benefits			\$ 587,000
Maintenance materials and operating supplies			1,722,000
Total annual operating and maintenance cost			\$2,309,000

^aBased on TID-7025, Section S30. (L) denotes licensed reactor operator.

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