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REACTORS — POWER

**FAST OXIDE BREEDER - REACTOR PHYSICS
PART I. PARAMETRIC STUDY OF 300(e) MW
REACTOR CORE**

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November 15, 1959

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I Introduction.

Physics scoping studies on a 300 MW(e) $\text{PuO}_2\text{-UO}_2$ fueled fast breeder reactor were made under Task C (Reactor Design) of Atomic Energy Commission Contract No. AT(04-3)-189PA#10. Physics design parameters that affect fuel costs, fuel conservation, and reactor safety were evaluated to guide the selection of parameters for a reference design.

Fuel burnup costs (which are negative in a breeder reactor) depend on the plutonium breeding ratio. Part of the plutonium is produced in the core (internal breeding) and the remainder in a fertile blanket surrounding the core. The internal breeding ratio is important to the fuel cost, aside from its being a component of the total breeding ratio, since a plutonium atom produced in the core can be burned without undergoing additional reprocessing and fabrication steps.

Fuel inventory costs depend on the fuel specific power (kw/kg Pu), which for a fixed power rating is inversely proportional to the plutonium fuel loading. Several of the design parameters, such as steel clad thickness, affect breeding ratio and plutonium inventory in a consistent direction (e.g., both burnup and inventory costs rise with increasing steel content). Other parameters, such as fuel rod diameter and oxide fuel density, affect the breeding ratios and the plutonium inventory in opposite directions. Selection of the optimum parameters requires a complete fuel cost analysis, taking into account fabrication and reprocessing costs as well as factors based on physics data.

Composite effects of plutonium production (breeding ratio) and plutonium inventory (fuel specific power) are expressed in the fuel doubling time. The doubling time is the number of years that a breeder reactor must operate to produce sufficient excess plutonium to fuel another identical reactor, including both the in-pile, and out-of-pile plutonium inventory. Doubling time is an important figure of merit for evaluating the breeding performance of a breeder reactor. Short doubling time requires high breeding ratio, high fuel specific power, low fractional out-of-pile fuel inventory, and high plant utilization factor.

A fast reactor has some unique safety problems. These arise principally from (1) a very short prompt neutron generation time (of the order of one-tenth microsecond) and (2) lack of an overriding prompt negative temperature coefficient of reactivity. It is desirable that the reactor be designed to prevent a prompt critical condition under any conceivable circumstances. One method of insuring against a prompt critical excursion is to limit the excess operating reactivity (the reactivity held down by shim rods) to a value less than one dollar. This has the disadvantage of requiring a short refueling interval. It might be possible to permit operating reactivities greater than one dollar, and to design the control system to safely accommodate the required excess reactivity. The difficulty of designing such a safe control system will be directly related to the excess reactivity value which must be controlled. In these scoping studies, excess operating reactivity is evaluated as a function of the refueling interval for various combinations of reactor design parameters.

Coolant (sodium) temperature coefficient of reactivity is another important factor for reactor stability and safety. It is important that the sodium coefficient be small and negative. The coefficient is sensitive to the core size and the uranium-to-plutonium ratio in the core, reflecting the competition of the effect of sodium on (1) neutron leakage and (2) neutron spectrum. In this study, the sodium temperature coefficient is evaluated over the range of design parameters considered.

A 300 MW(e) plant rating, 812 MW(t) reactor, was selected for the initial scoping studies. This plant size is sufficiently large to exploit many of the advantages of a sodium cooled fast oxide breeder, yet not too large for a power plant in the near future. Considerable improvement of the fuel economics is expected for larger reactor power ratings.

Fuel irradiation exposure of 100,000 MWD/t was assumed. Ability to withstand high burnup is perhaps the outstanding potential advantage of oxide fuel over metallic and alloy fuels. The Knolls Atomic Power Laboratory successfully irradiated a $\text{PuO}_2\text{-UO}_2$ specimen, with low density oxide (65% theoretical) to 50,000 MWD/t.* Ability to use appreciable steel clad in

*See reference 1 at end of report

the fast neutron spectrum enhances the prospect of attaining high fuel burnup. For this reason the steel clad thickness, or more specifically the ratio of clad thickness to fuel rod diameter, is included as a design variable in this study.

Other design parameters considered as variable are (1) fuel rod diameter, (2) density of the $\text{PuO}_2\text{-UO}_2$ fuel, and (3) core and blanket configuration. With an assumed peak radial fuel temperature differential from surface to center, the power per unit volume of fuel is determined by the thermal conductivity k of the fuel, the core peak-to-average power density F , and the fuel rod diameter r . Assuming fixed k^* and F and a fixed core power rating (812 MW thermal), the fuel volume of the core is uniquely determined by the rod diameter r . Thus the fuel rod diameter is the most important of the design parameters considered for determining the core size. Small rod diameters are associated with small, high power density cores; large rods with large, low power density cores.

Neutron leakage from the core and parasitic capture in steel and structure are sensitive to the oxide fuel density. This is considered as a design variable to evaluate the physics incentives for going to high density oxide, which might require more difficult fabrication procedures than for low density oxide.

Most of the work for this study was done on a conventional, right cylindrical core completely surrounded by an oxide blanket. An alternate configuration covered in this report is an annular core surrounded by both central and external blankets. This configuration appeared to offer potential advantages which for the most part, failed to materialize after detailed calculation. For this reason, Summary and Conclusions (Section II) are restricted to the conventional, cylindrical core configurations.

Section III of this report lists the range of values for the variable design parameters and the conditions that were held fixed for these scoping studies.

*Variation of k , e.g. with temperature, fuel density, fuel-void distribution, and fuel composition (including its change with burnup), was not factored into this preliminary analysis, in view of the uncertainty as to these variations and the added complication involved.

II Summary and Conclusions.

Breeding Ratio*

Total breeding ratio varies from 1.1 to 1.5 over the range of parameters considered. Breeding ratio increases with increasing fuel rod diameter (core size), increasing oxide fuel density, and decreasing steel content.

Internal (core) breeding ratio is most sensitive to the core size, and, consequently, the fuel rod diameter. Internal breeding ratios range from about 0.5 to 1.0 from the smallest (125 mils) to the largest (280 mils) of the fuel rod diameters considered. Raising oxide fuel density (65 to 90 percent theoretical UO_2 density) and lowering steel content increases the internal breeding ratio.

Fuel Specific Power and Plutonium Inventory*

Plutonium core loading ranges from 500 to 1500 Kg. The low value is for the smallest fuel rod diameter and is relatively insensitive to the oxide density over the 65 to 90 percent theoretical range. The high value is for the largest rod diameter at high (90 percent) oxide density. Decreasing steel content lowers the plutonium loading, although the steel penalty is more pronounced in a reduced breeding ratio than in an increased plutonium mass.

Fuel specific powers corresponding to the above range of plutonium loadings are about 500 to 1500 kw (t)/kg Pu. This includes only the in-pile plutonium.

Doubling Time

Doubling time is appreciably reduced by going to high density fuel and low steel content. A 17-year doubling time is obtained with the following conditions:

* Breeding ratio and plutonium inventory are defined here on the basis of Pu-239 and Pu-241 only. The appendix lists values based on total plutonium. The total plutonium basis yields both higher breeding ratio and higher inventory, with a slightly increased doubling time.

- (1) 90% theoretical oxide density
- (2) steel clad thickness = 10 percent of fuel rod diameter and 10 percent steel over and above clad is allowed for miscellaneous structure
- (3) plant utilization factor = 0.7
- (4) fuel rod diameter is in the range 150 to 200 mils
- (5) out-of-pile inventory equals two fuel batches, where the fuel batch size is that fraction of the core load required to yield a 4-month refueling interval.

Condition (4) is in the intermediate range of rod sizes. Conditions (2) and (3) are conservative. Increasing the plant factor to 0.8 and removing half of the above steel results in a 10-year doubling time.

Excess Operating Reactivity Versus Refueling Interval and Sodium Temperature Coefficient

Except for the largest fuel rod diameters (largest cores) which have low specific power and high internal breeding, a limitation of one dollar on the operating reactivity imposes an extremely short refueling interval. On the basis of (1) a 4-month interval between shutdowns for partial core reloading, (2) 90 percent oxide density, and (3) a 0.1 ratio of clad thickness to fuel rod diameter; the excess operating reactivity is \$7.5 for a 125 mil fuel rod diameter, \$2.5 for 200 mils, and \$0.7 for 280 mils. Increasing oxide density and lowering steel content tend to decrease the required operating reactivity.

The sodium temperature coefficient of reactivity is negative except for the largest fuel rod diameters (core sizes) at high oxide density, where neutron leakage from the core is extremely small. In particular for the design parameters yielding the 17-year doubling time listed above, the coefficient is calculated to be $-6 \times 10^{-6} \Delta\rho/^\circ\text{C}$ (for the 200 mil rod). Lowering the steel content and allowing appreciable plutonium buildup in the blanket yields positive components to the temperature coefficient, but not of sufficient magnitude to reverse the negative sign. Lowering the oxide fuel density renders the sodium temperature coefficient more negative as a result of the increased neutron leakage.

There appears to be a conflict of safety considerations as between excess operating reactivity, which favors a large diameter fuel rod, and the sodium temperature coefficient which favors a small rod. The intermediate

range of rod sizes (200 mils or slightly less) which yields a \$2.5 operating reactivity and a small, negative sodium temperature coefficient appears as a good compromise. Further, this range appears to be near optimum from fuel cost considerations related to physics parameters, although a detailed evaluation of fabrication and reprocessing costs must be made before this can be established.

III Design Criteria for Scoping Studies.

A reactor power rating of 812 MW(t), corresponding to a 300 MW(e) plant for 37% efficiency was selected for the scoping study. A fuel burnup of 100,000 MWD/t, or about 12 atom percent of the fissionable and fertile material, was assumed based upon an extrapolation of the KAPL (1) irradiation tests for oxide fuel. These tests indicated a successful irradiation of about half of this burnup for a plutonium-uranium oxide mixture.

Using a thermal conductivity of 1.0 Btu/hr-ft-[°]F for the oxide fuel, a 3500[°]F temperature difference between the center and surface of the fuel at the position of peak power, a 2.5 peak-to-average power density, and a thermal power rating of 812 MW uniquely determines the fuel volume in the core for a chosen fuel rod diameter. The sodium flow area was obtained from an average inlet and outlet temperature of 750[°]F and 1000[°]F with a flow velocity of 23 feet per second. A two percent void volume was allowed for possible control rod passages and an additional ten percent of the clad stainless steel volume was allowed for structure. The core height-to-diameter was set equal to unity in the scoping studies.

The isotopic composition of the uranium portion of the mixed oxide fuel was assumed to be entirely depleted, that is, just U-238. The slight advantage of U-235 in natural or depleted uranium fuel was not included because in long term irradiations and fuel recycling its concentration would be negligible. The isotopic composition of the input plutonium was taken to be that characteristic of the output of a high burnup thermal reactor: 60% Pu-239, 25% Pu-240, 12% Pu-241 and 3% Pu-242. The blanket surrounding the cylindrical core was eighteen inches thick and consisted of fully depleted uranium oxide. The blanket volume composition was 50% fuel, 20% stainless steel and 30% sodium. No attempt was made to optimize the blanket in these scoping studies and hence the isotopic blanket compositions are those representing the clean condition (no plutonium), while the core isotopic composition is characterized by a 50,000 MWD/t burnup, half way between that of the clean fuel composition and that of the most depleted fuel in the core which is ready for discharge after a 100,000 MWD/t burnup. (The ton in this instance refers to 2,000 pounds of uranium and plutonium, excluding the oxygen contribution.)

In a preliminary analysis such as this, it will be recognized that many of the design criteria are of necessity selected on a somewhat arbitrary basis. In some cases, such as peak/average power distribution, the choice of a parameter value is based on experience as to those values which have proved appropriate in similar designs. In other cases such as blanket composition, the results insofar as core design is concerned are relatively insensitive to the selection. In a few cases such as the choice of reactor power rating and the core height/diameter ratio, extension of the analysis to include the effect of varying these parameters appears to be warranted, as discussed in the section on future work.

One critical design criterion which cannot be readily categorized in one of the preceding classes is the assumption of a constant fuel thermal conductivity and radial temperature drop. The values taken for these two parameters are equivalent to assuming a constant peak linear power generation of 423 watts/cm and correspond to assuming a central fuel temperature (approximately the melting point) limitation on power density.

A similar design criterion has been used for other oxide fuel elements and its validity is being experimentally investigated in the fuel development phase of the Fast Oxide Breeder program. When and if a better criterion for determining fuel power density is established, the parametric analysis can be modified accordingly.

Table I summarizes the input fuel compositions and other design criteria used in this work. For the annular core discussed in Section V, most of the above design criteria apply. The exceptions are covered in Section V.

TABLE I

Design Criteria for Scoping Studies

<u>Reactor Power</u>	
Thermal	812 MW
Electrical, at 37% efficiency	300 MW
<u>Core Fuel Conditions</u>	
Fuel density variable from 65% to 90% theoretical density *	
Fuel rod diameter variable from 0.125 to 0.280 inch	
Peak temperature rise (surface to center of fuel)	3500°F
Thermal conductivity	1.0 BTU/hr-ft°F
Peak-to-average specific power	2.5
Specific fuel irradiation	100,000 MWD/t
Oxide fuel (PuO ₂ -UO ₂) mixture	
<u>Input Fuel Isotopic Composition</u>	
Uranium entirely U-238	
Plutonium as follows:	
Pu-239	60%
Pu-240	25%
Pu-241	12%
Pu-242	3%
<u>Sodium Coolant Conditions</u>	
Inlet temperature	750°F
Average outlet temperature	1,000°F
Flow rate	23 ft/sec
<u>Clad and Structure</u>	
Material: stainless steel (72 atom percent Fe, 28 atom percent Ni.)	
Clad thickness variable from 5 to 20 percent of fuel rod diameter.	
Structural volume equals 10 percent of clad volume.	
<u>Core Dimensions</u>	
Height/Diameter = 1. The core diameter is then uniquely determined by the fuel rod diameter and the steel clad thickness, utilizing the above design criteria. Two percent of the total core volume is void and reserved for control rod passages.	

* Theoretical density taken to be 10.9 g/cc (variation with Pu/U ratio neglected)

TABLE I (continued)

Blanket Properties*

Thickness	18 inch
Volume fractions:	
UO ₂ (entirely U-238)	0.50
Sodium	0.30
Steel	0.20

Atom Constituents of Materials

Fuel	0.0243×10^{24} atoms/cc**
Steel	0.0840×10^{24} atoms/cc
Sodium	0.0218×10^{24} atoms/cc

*Axial and radial blankets were assumed identical for scoping studies.

**Atoms/cc of (U+Pu) at 100% of theoretical density.

IV. Calculational Results for a Cylindrical Core Surrounded by a Fertile Blanket

A. Breeding Ratio, Plutonium Loading and Doubling Time

1. Definition of Terms

The term breeding ratio is intended to define the rate at which fissile isotopes are being created in the breeder reactor to the rate at which they are being destroyed. With a plutonium fuel containing an appreciable content of Pu-240 and higher isotopes, this ratio is difficult to define uniquely. Pu-240, and presumably Pu-242, has an appreciable fissile value in a fast reactor, but less than that of either Pu-239 or Pu-241. Therefore, breeding ratios and plutonium mass may be defined, at one extreme, in terms of the content of Pu-239 and 241. It is to be expected that as the utilization of plutonium fuel in reactors becomes more common, economic values will be assigned to the various plutonium isotopes in accordance with their fissile value (2). On the other hand, current fuel cost analyses usually are based on a fixed price for plutonium, independent of the isotopic composition. For this purpose, it is convenient to base breeding ratio and plutonium loading on the "Total Plutonium" content. This results at the other extreme in an alternate definition for the breeding ratio. Results based upon this definition are presented in Appendix A. The latter definition yields consistently higher values for the breeding ratio and fuel loading, one tending to compensate the economic effects of the other.

In this section the "Breeding Ratio" will be defined as the ratio of non-fission capture of U-238 and Pu-240 to the fission and capture rates for Pu-239 and Pu-241. That is, the rate at which Pu-239 and Pu-241 isotopes are being formed divided by the rate at which these isotopes are being destroyed. The mass of fissile material, plutonium mass, will likewise be based upon Pu-239 and Pu-241 content.

2. Effects of Fuel Rod Diameter and Fuel Density

Figure 1 shows the internal (core) breeding ratio and the total breeding ratio versus fuel rod diameter for fuel with a density of 65 or 90 percent of the theoretical $\text{PuO}_2\text{-UO}_2$ density. The steel clad thickness in these cores is ten percent of the fuel rod diameter. The breeding ratios are significantly larger for the 90 percent fuel density due to the smaller core leakage and parasitic capture, and the increased rate of U-238 and Pu-240 fissions. The internal breeding ratio is, of course, highly dependent on the core size and consequently, on the fuel rod diameter. The numbers at the top of the figure are the core diameters corresponding to three different fuel rod diameters. The total breeding ratio ranges from 1.1 for the smallest rods to 1.3 for the largest rods. Increasing blanket thickness would increase these values slightly, the increase being greater for small fuel rod diameters. Thus, the breeding ratio advantage for the large fuel rods and high density fuel may be decreased somewhat.

The internal breeding ratio ranges from 0.3 for the small rods to 0.95 for the large rods, rising appreciably with increasing fuel rod diameter, or core size. The incentive for large core sizes and high fuel density is very evident when one considers the internal breeding ratio. Internal breeding is an important economic factor for selecting design parameters for a fast breeder since plutonium produced in the core can be burned directly, without undergoing processing or fabrication steps as is required for plutonium made in the blanket. In addition, a high internal breeding ratio eases control problems by sustaining reactivity with burnup. The required plutonium (Pu-239 and 241) fuel loading was calculated for the equilibrium core (i.e., the average core composition corresponding to the input fuel irradiated to

Figure 1 Breeding ratio versus fuel rod diameter and fuel density.

50,000 MWD/t, one-half of the irradiation for the average discharge fuel). Figure 2 shows the variation of plutonium mass over the range of rod size, core size and fuel density. The plutonium mass rises appreciably with increasing fuel rod size, resulting in higher fuel inventory costs. This opposes the increased plutonium credit due to the higher total breeding ratio found in Figure 1. Consequently, the optimum fuel rod diameter from economic considerations, involving plutonium mass and breeding ratio would depend highly on fuel inventory rates. The plutonium mass does not increase nearly as fast as core size since the plutonium concentration decreases appreciably with increasing core size. Over the range of fuel rod diameters considered, the plutonium mass is slightly higher for fuel of 90 percent density than for 65 percent density, the difference becoming smaller with reduced core size (higher neutron leakage). Neutron economy would be more sensitive to steel content in the 65 percent fuel density, although in these calculations steel content was varied only for the 90 percent fuel density.

Table II lists design information such as core diameter, composition and average fuel specific power while Table III contains breeding ratios and fuel loadings over the range of interest. The isotopic fuel compositions for the equilibrium cores are listed in Table IV. The equilibrium core thus defined has sufficient excess reactivity to allow a four month interval between shutdowns for refueling. The initial and discharge isotopic compositions are also given based upon this refueling schedule.

The atom ratio of uranium to plutonium is shown in Figure 3 for input and discharge compositions. This ratio is 2.5 for the smallest core at 65% fuel density and reaches 9 for the

TABLE II
Core Dimensions, Compositions and Thermal Properties.

Fuel Rod Diam. Inch	Fuel Dens. % of Theor.	Ratio of Steel Clad Thickness to Fuel Rod Dia. Ft.	Core Diam. Ft.	Core Volume Fractions *			Average Specific Power		Peak Heat Flux at 2.5 peak to average $\frac{\text{BTU}}{\text{hr.} \cdot \text{ft.}^2}$	Fuel Res. Time at 100,000 MWD/t & 0.7 load fac. years
				Fuel	Steel	Sodium	kw/kg **	kw/kg ***		
0.125	90	0.05	4.04	0.255	0.059	0.666	246	1430	1.36×10^6	1.67
0.200	90	0.05	4.75	0.404	0.093	0.483	96	913	0.85×10^6	4.67
0.280	90	0.05	5.51	0.505	0.117	0.358	49	569	0.61×10^6	8.67
0.125	90	0.10	4.15	0.235	0.114	0.631	246	1391	1.36×10^6	1.67
0.200	90	0.10	4.94	0.360	0.174	0.446	96	867	0.85×10^6	4.67
0.280	90	0.10	5.77	0.440	0.213	0.327	49	530	0.61×10^6	8.67
0.125	90	0.20	4.38	0.201	0.212	0.567	246	1344	1.36×10^6	1.67
0.200	90	0.20	5.31	0.289	0.305	0.386	96	781	0.85×10^6	4.67
0.280	90	0.20	6.28	0.342	0.362	0.276	49	472	0.61×10^6	8.67
0.125	65	0.10	4.15	0.235	0.114	0.631	341	1398	1.36×10^6	1.33
0.200	65	0.10	4.94	0.360	0.174	0.446	133	950	0.85×10^6	3.33
0.280	65	0.10	5.77	0.440	0.213	0.327	68	619	0.61×10^6	7.00

* 2% of volume reserved for control rod passages.

** Total U and Pu

*** Pu(239 & 241)

TABLE III
Breeding Ratios and Fuel Loadings*

Fuel Rod Diam Inch	Fuel Dens. % of Theor.	Ratio of Steel Clad Thickness to Fuel Rod Diam.	Breeding Ratio *			Core Pu(239 + 241) Mass Kg	Core U/Pu (239 + 241) Ratio at Input
			Core	Blanket	Total		
0.125	90	0.05	0.4444	0.7750	1.2194	568	3.371
0.200	90	0.05	0.8143	0.5382	1.3525	889	7.017
0.280	90	0.05	1.0338	0.4187	1.4525	1427	10.244
0.125	90	0.10	0.4310	0.7454	1.1764	584	3.221
0.200	90	0.10	0.7568	0.5056	1.2624	940	6.422
0.280	90	0.10	0.9479	0.3624	1.3103	1533	8.918
0.125	90	0.20	0.4053	0.6979	1.1032	604	2.996
0.200	90	0.20	0.6630	0.4575	1.1205	1040	5.372
0.280	90	0.20	0.8150	0.3285	1.1435	1721	7.154
0.125	65	0.10	0.2854	0.8273	1.1127	581	2.024
0.200	65	0.10	0.5768	0.6214	1.1982	855	4.462
0.280	65	0.10	0.7726	0.4658	1.2384	1312	6.548

* Both breeding ratio and Pu mass are evaluated for the equilibrium core condition in which the average fuel isotopic composition is that attained after 50,000 MWD/t burnup, one-half the burnup of the most depleted fuel in the core. Only Pu-239 and Pu-241 considered for evaluation of breeding ratio, Pu mass, and U/Pu ratio.

TABLE IV
Isotopic Fractions of Input (Clean) Fuel

Fuel Rod Diam. Inch	Fuel Dens. % of Theor.	Ratio of Steel Clad Thickness to Fuel Rod Diam.	U-238	Pu-239	Pu-240	Pu-241	Pu-242	Total Pu
0.125	90	0.05	0.7080	0.1752	0.0730	0.0350	0.0088	0.2920
0.200	90	0.05	0.8350	0.0990	0.0412	0.0198	0.0050	0.1650
0.280	90	0.05	0.8810	0.0714	0.0297	0.0143	0.0036	0.1190
0.125	90	0.10	0.6990	0.1806	0.0753	0.0361	0.0090	0.3010
0.200	90	0.10	0.8220	0.1068	0.0445	0.0214	0.0053	0.1780
0.280	90	0.10	0.8650	0.0810	0.0338	0.0162	0.0040	0.1350
0.125	90	0.20	0.6830	0.1902	0.0793	0.0380	0.0095	0.3170
0.200	90	0.20	0.7950	0.1230	0.0512	0.0246	0.0062	0.2050
0.280	90	0.20	0.8370	0.0978	0.0407	0.0196	0.0049	0.1630
0.125	65	0.10	0.5930	0.2442	0.1018	0.0488	0.0122	0.4070
0.200	65	0.10	0.7630	0.1422	0.0593	0.0284	0.0071	0.2370
0.280	65	0.10	0.8250	0.1050	0.0438	0.0210	0.0052	0.1750

TABLE IV(continued)

Isotopic Fractions of Core Equilibrium (50,000 MWD/t) Fuel

Fuel Rod Diam. Inch	Fuel Dens. % of Theor.	Ratio Steel Clad Thickness to Fuel Rod Dia.	U-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-243	Fission Prod. Pairs	Total Pu
0.125	90	0.05	0.6812	0.1463	0.0755	0.0285	0.0098	0.0004	0.0583	0.2601
0.200	90	0.05	0.7827	0.0924	0.0457	0.0146	0.0060	0.0004	0.0582	0.1587
0.280	90	0.05	0.8132	0.0783	0.0353	0.0102	0.0045	0.0004	0.0581	0.1283
0.125	90	0.10	0.6731	0.1500	0.0785	0.0294	0.0102	0.0005	0.0583	0.2681
0.200	90	0.10	0.7731	0.0967	0.0495	0.0158	0.0065	0.0001	0.0583	0.1685
0.280	90	0.10	0.8023	0.0828	0.0396	0.0116	0.0051	0.0005	0.0581	0.1391
0.125	90	0.20	0.6560	0.1582	0.0847	0.0313	0.0110	0.0005	0.0583	0.2852
0.200	90	0.20	0.7515	0.1062	0.0573	0.0185	0.0076	0.0007	0.0582	0.1896
0.280	90	0.20	0.7816	0.0913	0.0477	0.0142	0.0063	0.0007	0.0582	0.1595
0.125	65	0.10	0.5769	0.2049	0.1044	0.0417	0.0133	0.0005	0.0583	0.3643
0.200	65	0.10	0.7270	0.1201	0.0636	0.0222	0.0084	0.0004	0.0583	0.2143
0.280	65	0.10	0.7743	0.0959	0.0492	0.0155	0.0065	0.0004	0.0582	0.1671

TABLE IV (continued)
Isotopic Fractions of Discharged Fuel

Fuel Rod Diam. Inch	Fuel Dens. % of Theor.	Ratio of Steel Clad Thickness to Fuel Rod Dia.	U-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-243	Fission Prod. Pairs	Total Pu
0.125	90	0.05	0.6513	0.1218	0.0764	0.0231	0.0105	0.0008	0.1161	0.2318
0.200	90	0.05	0.7304	0.0859	0.0490	0.0112	0.0066	0.0008	0.1161	0.1527
0.280	90	0.05	0.7496	0.0801	0.0406	0.0078	0.0050	0.0008	0.1161	0.1335
0.125	90	0.10	0.6448	0.1238	0.0797	0.0237	0.0110	0.0009	0.1161	0.2382
0.200	90	0.10	0.7229	0.0878	0.0531	0.0122	0.0072	0.0007	0.1161	0.1603
0.280	90	0.10	0.7421	0.0815	0.0449	0.0088	0.0058	0.0008	0.1161	0.1410
0.125	90	0.20	0.6285	0.1297	0.0871	0.0255	0.0121	0.0010	0.1161	0.2544
0.200	90	0.20	0.7064	0.0924	0.0615	0.0142	0.0086	0.0008	0.1161	0.1767
0.280	90	0.20	0.7273	0.0846	0.0532	0.0108	0.0071	0.0009	0.1161	0.1557
0.125	65	0.10	0.5581	0.1704	0.1051	0.0354	0.0141	0.0008	0.1161	0.3250
0.200	65	0.10	0.6884	0.1019	0.0661	0.0174	0.0092	0.0009	0.1161	0.1946
0.280	65	0.10	0.7233	0.0874	0.0532	0.0119	0.0072	0.0009	0.1161	0.1597

large cores at a fuel density of 90%. The solid curve indicates the ratio for the input fuel and the dashed curve is for fuel which is discharged after an irradiation of 100,000 MWD/t burnup. It is interesting to note that as the internal breeding ratio increases for the large cores, the discharge uranium-to-plutonium atom ratio becomes less than the input ratio.

The composite effects of breeding ratio and fuel loading or specific power are embodied in doubling time. From the standpoint of fuel conservation, the doubling time is perhaps the most significant figure of merit. Figure 4 shows the doubling time over the range of fuel rod size studied for a 0.1 steel clad thickness to rod diameter, a 0.7 plant load factor, and an out-of-pile inventory equal to that of two fuel batches (the batch size allows a four month refueling interval). The doubling time is significantly reduced for the 90 percent fuel density, with the minimum doubling time (17 years for the cases covered in Figure 4) shifting toward smaller diameter rods as the fuel density increases.

3. Effect of Steel Clad Thickness.

Figures 5 and 6 show the breeding ratio and equilibrium plutonium loading versus the ratio of stainless steel clad thickness to fuel rod diameter. Table V shows fuel doubling time versus the steel clad ratio. The effect of clad material in a fast reactor is seen to be significant and the incentive for minimizing the steel content is appreciable so far as the breeding ratio penalty is concerned. Even the plutonium loading is appreciably influenced by the capture and spectrum effects with increasing steel content. As one attempts to increase the internal breeding ratio by increasing the core size and fuel rod diameter, the penalty for steel clad increases. Throughout the scoping studies the stainless steel was considered to be 72 atom percent iron and 28 atom percent nickel.

As the steel content rises the U-238 fission fraction decreases due to the inelastic scattering of the iron and nickel. The capture to fission ratio of plutonium-239 is appreciably increased by the spectrum degradation due to the steel. Table VI shows these effects for steel and lists other important trends as they vary over the range of parameters. The fuel specific power, Kw/Kg (Pu-239 and 241) show the composite effect of the trends listed in the other columns. As the core size increases, the fraction of fissions due to uranium-238 increases appreciably due to the larger uranium-to-plutonium ratio. The fraction of fissions due to plutonium-240 decreases for the same reason. The capture-to-fission ratio of plutonium-239 decreases with increasing core size due to the degradation of the neutron spectrum. Figures 7 and 8 show the neutron spectrum and power histogram as a function of lethargy for the 0.20 inch diameter rod and 90 percent fuel density.

TABLE V
Doubling Time versus Steel Clad Thickness

Steel Clad Thickness Fuel Rod Diameter	Doubling Time - Years [*] for Indicated Fuel Rod Diameter (inches)		
	<u>0.125</u>	<u>0.200</u>	<u>0.280</u>
0.05	13.7	12.0	14.7
0.10	19.1	17.5	23.0
0.20	28.5	37.5	50.5

- * The doubling time is based upon a 0.7 load factor and fast fission in U-238 and Pu-240. The specific power and breeding ratio were evaluated for Pu-239 and Pu-241 to obtain the above doubling time.

TABLE VI
Mean Fission Energy, Capture-to-Fission Ratios,
and Fissions in Fertile Materials

Fuel Rod Diam. Inch	Fuel Dens. % of Theor.	Ratio of Steel Clad Thick. to Fuel Rod Dia.	Core Diam. Feet	Mean Neutron Energy Kev.*	Fission Fraction in Core		Capt. to Fis. Ratio of Pu-239	Dollars Excess Operating Reactivity (4 mo. in- terval.)
					U-238	Pu-240		
0.125	90	0.05	4.04	280	0.102	0.070	0.246	6.70
0.200	90	0.05	4.75	231	0.144	0.054	0.271	1.90
0.280	90	0.05	5.51	219	0.161	0.047	0.276	0.33
0.125	90	0.10	4.15	238	0.089	0.066	0.259	7.57
0.200	90	0.10	4.94	185	0.123	0.053	0.283	2.25
0.280	90	0.10	5.77	176	0.136	0.047	0.287	0.66
0.125	90	0.20	4.38	170	0.069	0.060	0.282	7.92
0.200	90	0.20	5.31	137	0.091	0.050	0.303	3.50
0.280	90	0.20	6.28	128	0.099	0.046	0.308	1.37
0.125	65	0.10	4.15	268	0.061	0.073	0.242	9.00
0.200	65	0.10	4.94	193	0.098	0.058	0.276	4.36
0.280	65	0.10	5.77	172	0.113	0.052	0.286	1.61

* Half of the fissions are caused by neutrons of higher energy than listed.

Figure 7 Neutron spectrum.

B. Neutron Balances.

Neutron balances for an equilibrium core and clean blanket are tabulated for three different fuel rod diameters in Tables VII through IX. These balances yield a detailed description of the several processes that contribute to the breeding ratio and account for the differences between the various core sizes. The U-238 contribution to the total fission rate is seen to increase with increasing core size due to the increasing uranium-to-plutonium atom ratio. The capture-to-fission ratio for the various plutonium isotopes increases slightly with larger core size, due to the degradation of the neutron energy spectrum with increased core size. The spectrum hardening increases with increased leakage as seen in Table VI.

There is a significant penalty in breeding ratio associated with long fuel irradiation. This is evident from the 3.0 percent neutron capture by fission products in the equilibrium core, which almost equals the total capture by the steel clad in core and blanket. This fission product capture results in a 0.09 reduction of the breeding ratio.

TABLE VII

Neutron Balance for 0.125-Inch Fuel Rod Diameter

(Fuel Density 90% of Theoretical, Steel Clad Thickness = 12.5 mils)

	<u>Core</u>	<u>Blanket</u>	<u>Total</u>
U-238 fissions	0.0288	0.0153	0.0441
Pu-239 fissions	0.2494	0	0.2494
Pu-240 fissions	0.0213	0	0.0213
Pu-241 fissions	0.0489	0	0.0489
Pu-242 fissions	0.0028	0	0.0028
<u>Am-243 fissions</u>	<u>0.0001</u>	<u>0</u>	<u>0.0001</u>
Sum Fissions	0.3513	0.0153	0.3666
U-238 captures	0.1418	0.2861	0.4279
Pu-239 captures	0.0702	0	0.0702
Pu-240 captures	0.0241	0	0.0241
Pu-241 captures	0.0138	0	0.0138
Pu-242 captures	0.0031	0	0.0031
Am-243 captures	0.0001	0	0.0001
F. P. captures	0.0190	0	0.0190
steel captures	0.0158	0.0163	0.0321
<u>sodium captures</u>	<u>0.0025</u>	<u>0.0007</u>	<u>0.0032</u>
Sum Captures	0.2904	0.3031	0.5935
Leakage		0.0399	<u>0.0399</u>
			1.0000

TABLE VIII

Neutron Balance for 0.20-Inch Fuel Rod Diameter

(Fuel Density 90% of Theoretical, Steel Clad Thickness = 20.0 mils)

	<u>Core</u>	<u>Blanket</u>	<u>Total</u>
U-238 fissions	0.0410	0.0083	0.0493
Pu-239 fissions	0.2461	0	0.2461
Pu-240 fissions	0.0175	0	0.0175
Pu-241 fissions	0.0398	0	0.0398
Pu-242 fissions	0.0023	0	0.0023
<u>Am-243 fissions</u>	<u>0.00004</u>	<u>0</u>	<u>0.00004</u>
Sum Fissions	0.34674	0.0083	0.35504
U-238 captures	0.2570	0.1874	0.4444
Pu-239 captures	0.0714	0	0.0714
Pu-240 captures	0.0255	0	0.0235
Pu-241 captures	0.0116	0	0.0116
Pu-242 captures	0.0031	0	0.0031
Am-243 captures	0.00006	0	0.00006
F. P. captures	0.0299	0	0.0299
Steel captures	0.0242	0.0106	0.0348
<u>Sodium captures</u>	<u>0.0018</u>	<u>0.0004</u>	<u>0.0022</u>
Sum Captures	0.42256	0.1984	0.62096
Leakage		0.0240	<u>0.0240</u>
			1.0000

TABLE IX

Neutron Balance for 0.28-Inch Fuel Rod Diameter

(Fuel Density 90% of Theoretical, Steel Clad Thickness = 28.0 mils)

	<u>Core</u>	<u>Blanket</u>	<u>Total</u>
U-238 fissions	0.0461	0.0053	0.0514
Pu-239 fissions	0.2458	0	0.2458
Pu-240 fissions	0.0157	0	0.0157
Pu-241 fissions	0.0345	0	0.0345
Pu-242 fissions	0.0021	0	0.0021
<u>Am-243 fissions</u>	<u>0.0002</u>	<u>0</u>	<u>0.0002</u>
Sum Fissions	0.3444	0.0053	0.3497
U-238 captures	0.3211	0.1315	0.4526
Pu-239 captures	0.0724	0	0.0724
Pu-240 captures	0.0229	0	0.0229
Pu-241 captures	0.0102	0	0.0102
Pu-242 captures	0.0030	0	0.0030
Am-243 captures	0.0003	0	0.0003
F. P. captures	0.0349	0	0.0349
Steel captures	0.0290	0.0073	0.0363
<u>Sodium captures</u>	<u>0.0013</u>	<u>0.0003</u>	<u>0.0016</u>
Sum Captures	0.4951	0.1391	0.6342
Leakage		0.0161	<u>0.0161</u>
			1.0000

C. Excess Operating Reactivity

In a fast reactor, with its extremely short neutron prompt generation time, safety considerations favor low operating reactivity (i.e., the excess reactivity held down by control rods is low). If the operating reactivity i.e. the excess provided for burnup, is held below one dollar, the reactor cannot go prompt critical as a result of rapid control rod withdrawal. Even with a reactivity above one dollar, the control system can be designed with weak shim rods for safety, so that the insertion of an excessive reactivity cannot occur under any reasonable circumstances.

While safety favors a low operating reactivity, the time interval between shutdowns for partial core refueling decreases with operating reactivity. Operating cost is lowered by increasing the refueling interval. The operating reactivity required to yield a given refueling interval depends mainly on the internal breeding ratio and the fuel power density.

Figures 9 and 10 show the required operating reactivity versus refueling interval for 90 and 65 percent theoretical fuel density. As the fuel rod diameter and core size increase the internal breeding ratio increases and the rate of fuel burnup decreases, resulting in a slower reactivity decrease. One dollar operating reactivity can be easily accommodated with the large core containing the 0.28 inch diameter fuel rod, with a six month refueling interval for the 90 percent density fuel. For an intermediate size core with the 0.20 inch fuel rod diameter and 90 percent fuel density, the refueling interval is almost two months for an excess operating reactivity of one dollar. The refueling interval is reduced to only a few weeks for the smallest core with 0.125 inch diameter fuel rod and 90 percent density.

An important conclusion can be drawn concerning small core fast reactors with high fuel specific power, and which are not fueled during operation. In order to achieve an economical refueling interval the operating reactivity must be allowed to exceed one dollar and necessary control and instrumentation provided. This may even be necessary for an intermediate sized core to obtain low operating costs.

D. Sodium Temperature Coefficient of Reactivity.

For reactor stability, it is highly desirable, and perhaps absolutely essential, that the temperature coefficient of reactivity due to the coolant be small and negative. Removal of sodium by thermal expansion results in a negative reactivity effect due to the slightly increased neutron leakage. A positive contribution arises from the increase of U-238 fissions and improved plutonium fission-to-capture ratio accompanying the spectrum hardening produced by the reduced sodium density.

The coolant temperature coefficient of reactivity is shown in Figure 11 for 65 and 90 percent fuel density as a function of rod diameter or core size. The numbers adjacent to the points indicate the ratio of U-238 to Pu-239 and Pu-241 atoms. The sodium temperature coefficient is quite sensitive to this ratio(3). The coefficient becomes more negative as the core leakage increases, either as a result of a smaller core size or a lower fuel density. As the core size is increased, the magnitude of the negative coefficient decreases and is positive for core diameters above 5.8 feet with a 90 percent fuel density. The point at which the coefficient becomes positive at lower fuel densities is shifted toward larger core sizes. The possibility of a positive coolant coefficient probably imposes an upper limit on the core size from stability considerations. These calculations were based on an equilibrium core, clean blanket, and unit height-to-diameter ratio. As the blanket builds up plutonium with time, the effective size of the core increases, resulting in a positive contribution to the sodium temperature coefficient. A preliminary calculation made for a blanket containing 5 percent plutonium in the first four inches of the blanket (next to the core) and one percent in the remainder of the blanket reduced the magnitude of the coefficient from -6.18 to -5.05×10^{-6} per $^{\circ}\text{C}$ for the 0.2 inch diameter, 90 percent density fuel. Hence, all the coefficients shown in Figure 11 should be made slightly less negative (more positive) for equilibrium blanket concentrations. When the steel ratio (clad thickness to rod diameter) is decreased from 0.1 to 0.05 for the 0.2 inch rod and 90% fuel density case, the coolant temperature coefficient changes from -6.18 to -3.35×10^{-6} per $^{\circ}\text{C}$. Reducing the steel content yields a positive contribution to the coefficient since the sodium then exerts a greater influence on the neutron spectrum.

Figure 11 Sodium temperature coefficient of reactivity versus fuel rod diameter and fuel density.

V. Calculational Results for Annular Core with a Central and Outer Blanket.

An annular core was investigated in order to evaluate the potential advantages of this core and blanket configuration. A central blanket, 18 inches in diameter was surrounded by an annular core which was 23 inches thick and 46 inches high. This cylindrical configuration was entirely surrounded by an external blanket which was 16 inches thick. The design criteria are the same as listed in Table I, with the following exceptions:

1. The central blanket material composition is the same as that of the core, except the input fuel isotopic composition is entirely U-238 oxide.
2. The outer blanket thickness is reduced from 18 to 16 inches, but the composition is the same as the blanket composition listed in Table I.
3. The core height is twice the thickness of the annular core region.
4. The peak-to-average power density in the core is reduced from 2.5 to 2.2 due to the extremely flat radial power distribution in the annular core.
5. The fuel rod diameter is 0.20 inch, steel clad thickness is 10 percent of the fuel diameter (20 mils), and 10 percent excess is allowed for structure. The fuel density is 90 percent of the theoretical value (10.9 gm/cm^3 for UO_2).

No attempt has been made to optimize the geometrical configuration of the annular reactor. The central blanket diameter represents a guessed optimum size in order to evaluate its potential and to determine whether it deserved more extensive study.

As plutonium is produced in both blankets of this design the differences between the annular and cylindrical reactors are lessened. An annular core with no plutonium in the blankets might be expected to exhibit some of the following advantages: (1) radial power flattening (2) increased refueling interval or less required excess reactivity due to breeding in the central blanket (3) increased delayed neutron fraction by promoting U-238 fissions in the central blanket in a region of high neutron importance. Most of these advantages did not materialize as shown by the comparison of an annular core to a cylindrical core in Table X. In spite of the improved radial power distribution, the power density is lower for the annular core since a higher plutonium loading is required to override the higher core leakage. Both the internal and total breeding ratio are slightly less than for the cylindrical core. The operating reactivity is not

TABLE X

Comparison of Annular and Cylindrical Core Reactors.
(0.20 inch rod diameter, 90% fuel density, 20 mil clad)

	<u>Annular Core</u>	<u>Cylindrical Core</u>
Height (in.)	46	60
Material Volume Fractions in Core		
Fuel	0.385	0.360
Steel	0.185	0.174
Sodium	0.415	0.446
Peak/Average Radial Power Density	1.28	1.64
Fuel Loading, Kg. Pu(239 & 241)	965	937
Fuel Loading, Kg. Pu (total)	1440	1400
Average Specific Power Kw/Kg Pu (239 & 241)	841	867
Leakage from Outer Blanket - percent	3.4	2.4
Total Breeding Ratio	1.20	1.26
Internal (Core) Breeding Ratio	0.66	0.76
Central Blanket Breeding Ratio	0.09	
Operating Reactivity for 4-Month Refueling Interval, Dollars	1.9	2.4
Delayed Neutron Fraction	0.00382	0.00388
Mean Neutron Energy Causing Fission, Kev.	252	185
Sodium Temperature Coefficient of Reactivity, $\Delta \rho / ^\circ\text{C}$	-3.85×10^{-6}	-6.18×10^{-6}

reduced as much as expected because of the breeding ratio reduction. The effective delayed neutron fraction, obtained from a flux squared spatial weighting of the actual delayed fraction, is not much different for the two cores. This is due to the fact that the importance of the center blanket is not much greater than the outer blanket and its volume is very small relative to that of the outer blanket. The sodium temperature coefficient is slightly less negative, but this could have been altered by the different height-to-diameter ratio. In view of this comparison the annular core does not appear to warrant the added mechanical design complexity over the simpler cylindrical core.

A neutron balance for the annular core is given in Table XI for comparison with the similar balance for the cylindrical core given in Table VIII.

TABLE XI

Neutron Balance for Annular Core Breeder

(0.20-Inch Fuel Rod Diameter, 90% Theoretical Fuel Density, 20-Mil Steel Clad)

	<u>Core</u>	<u>Center Blanket</u>	<u>Outer Blanket</u>	<u>Total</u>
U-238 fissions	0.0399	0.0024	0.0089	0.0512
Pu-239 fissions	0.2525	0	0	0.2525
Pu-240 fissions	0.0197	0	0	0.0197
Pu-241 fissions	0.0435	0	0	0.0435
Pu-242 fissions	0.0025	0	0	0.0025
<u>Am-243 fissions</u>	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
Sum Fissions	0.3581	0.0024	0.0089	0.3694
U-238 captures	0.2235	0.0339	0.1711	0.4285
Pu-239 captures	0.0694	0	0	0.0694
Pu-240 captures	0.0241	0	0	0.0241
Pu-241 captures	0.0120	0	0	0.0120
Pu-242 captures	0.0031	0	0	0.0031
Am-243 captures	0.0001	0	0	0.0001
F. P. captures	0.0247	0	0	0.0247
Steel captures	0.0209	0.0022	0.0092	0.0323
<u>Sodium captures</u>	<u>0.0014</u>	<u>0.0002</u>	<u>0.0004</u>	<u>0.0020</u>
Sum Captures	0.3792	0.0363	0.1807	0.5962
Leakage			0.0344	<u>0.0344</u>
				1.0000

VI Calculational Techniques.

The basic cross section set was the 17-group library of microscopic cross sections associated with MUG-O (4). The MUG-O condensation routine was utilized to prepare 11-group FINE (5) cross section libraries from the 17-group MUG-O library. Table XII shows the lethargy and energy breakdown of the condensed eleven group set. The first eleven energy groups of the MUG-O library were obtained largely from the Argonne compilation of fast reactor cross sections (6). The remaining six groups were compiled at APED. Since fast cross sections for Pu-241 and Pu-242 were not available, the Pu-241 cross section was assumed identical with Pu-239, except for the ν value which was taken to be 3.01. The cross section for Pu-242 was set equal to that of Pu-240. Fission product capture cross sections were obtained from the G.H.S. analysis (7).

In addition to the MUG-O cross section condensation option, the zero-dimensional multigroup calculation option was extensively utilized for criticality surveys. FABU (8) facilitated the burnup calculations so that the plutonium enrichment range required for the VALPROD (9) calculations were considerably narrowed. The equilibrium core isotopic fuel composition was obtained from FABU at the desired irradiation exposure.

Multi-group one-dimensional diffusion theory, utilizing eleven energy groups, constituted the basic computational technique for these scoping studies. This computation was carried out with VALPROD (9) which permits inelastic scattering into any of the first eight groups. Input cross section generation for VALPROD was facilitated with FINE, and NUB was used to edit the output of VALPROD to obtain neutron balances and breeding ratios. Since only one-dimensional cylindrical geometry was used in the VALPROD calculations, the neutron events in the axial blanket and axial leakage were obtained from the ratio of axial-to-radial core buckling squared for each calculation. (Utilizing a 15 cm. reflector savings, with leakage proportional to the buckling squared yielded good reactivity agreement between one-dimensional VALPROD calculations and the zero-dimensional MUG-O calculations.)

A new burnup code, SCORCH (11) which is a multi-region version of FABU was utilized to evaluate isotopic fuel changes in various regions. SCORCH requires the average one-group region dependent cross sections obtained from NUB and FINE

TABLE XII

Lethargy and Energy Limits for 11-Group Cross Section Set

<u>Group No.</u>	<u>Lethargy Interval</u>	<u>Lower Energy Limit</u>
1	1.53	2.25 Mev
2	0.5	1.5 "
3	0.5	0.825 "
4	0.5	0.500 "
5	0.5	0.300 "
6	1.0	0.110 "
7	1.5	0.025 "
8	2.0	3350 ev
9	2.82	195 "
10	6.21	0.39 "
11	1.00	0.15 "

for the burnup calculations. SIN (12) was used to facilitate the input preparation for SCORCH.

SCORCH was particularly useful for the more complex annular core geometry. The reactivity change with burnup for a fixed refueling interval for the annular core was obtained by considering the central blanket and the external blanket each to be composed of several regions. A VALPROD calculation for the equilibrium core and clean blankets then yielded the region dependent cross sections required for the burnup calculation.

Survey calculations of the sodium temperature coefficient of reactivity were obtained from MUG-0. More refined coefficients were then calculated with VALPROD in order to determine the effect of the blanket U-238 fissions on the coefficient. These coefficients were determined from the reactivity change with sodium (density) and involve small differences which should be verified by more refined calculations.

VII Future Work.

1. Higher Power Rating.

These studies were performed for a 300 MW(e) or 812 MW(t) reactor. With higher power ratings resulting in larger core sizes, smaller fuel rods with their high specific power can be used without penalty to the breeding ratio. Scoping studies should be made, therefore, for higher ratings, perhaps 600 and 900 MW(e), with rod diameters ranging from 0.125 to 0.20 inches.

2. Core Height Less Than Core Diameter.

There may be some thermal and hydraulic advantages to rendering H/D less than unity. In addition, the sodium volume fraction will be reduced, resulting in improved neutron economy. Scoping studies should be made for H/D values around 0.7, with perhaps fixed fuel rod diameter and fuel density selected from the results of the present study.

3. Reference Design.

The results of the present study should be adequate to permit selection of parameters for a 300 MW(e) reference design. Calculational techniques utilized for these scoping studies should be reexamined for their adequacy in yielding highly accurate results for more detailed reactor physics design. For example, the adequacy of diffusion theory for fast reactor analysis should be checked by comparison with a calculation utilizing the Sn approximation of neutron transport theory. Cross sections should be checked to determine whether they include the most recent and reliable data. Methods and cross sections should be evaluated with fast reactor critical experiments. Reactor physics design calculations on a reference reactor must include, in addition to more precise evaluation of breeding ratio, critical mass, excess operating reactivity, and sodium temperature coefficient, the following:

- (1) radial, axial, and local power distributions;
- (2) control rod design;
- (3) blanket design and optimization;
- (4) refueling patterns and intervals for both core and blanket;
- (5) prompt and delayed temperature coefficients;
- (6) stability and reactor dynamics;
- (7) delayed neutron lifetime;
- (8) two dimensional diffusion theory calculations for the reference core design.

Appendix A

Alternate Fuel Definition.

In section IV-A the definition of breeding ratio and plutonium fuel loading was discussed in terms of $\text{Pu}(239 + 241)$ and total plutonium (i.e. Pu-239, 240, 241 and 242). This section contains the results based upon this latter definition, as shown in Figures 12 through 16.

Comparison of the important terms are shown in Table XIII based on the two definitions of plutonium content. The results are for the 0.20 inch fuel rod diameter, 20 mil steel clad, and 90 percent fuel density. The values in the left hand column are based upon all plutonium isotopes included in the fuel term while the right hand column includes only Pu-239 and Pu-241. The total plutonium fuel definition results both in higher breeding ratios and higher plutonium fuel loading, and a slightly longer doubling time.

Figure 13 Total plutonium loading versus fuel rod diameter and fuel density.

TABLE XIII
Comparison of Breeding Characteristics for
Alternate Fuel Definitions

	<u>Total Pu</u>	<u>Pu-239 and Pu-241</u>
Core Fuel Loading-Kg.	1400	940
Internal Breeding Ratio	0.83	0.76
Total Breeding Ratio	1.43	1.26
Doubling Time-Years	19.1	17.5

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