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STEAM-COOLED FAST BREEDER REACTORS

D. T. Aase**

MASTER

Although most development work on Fast Breeder Reactors has been devoted to the use of liquid metal cooling, in the early 1960's several studies were begun concerning the use of alternate coolants as a means of circumventing some of the problems associated with liquid metal systems. Among the concepts considered since that time, several have involved using steam as coolant.

Proponents of steam cooling recognized that breeding ratios somewhat lower than in other fast reactor concepts could be expected, due to the softer neutron spectrum resulting from the steam. The interest, however, was to introduce fast breeder economics into the nuclear power economy by utilizing the large background of light water reactor technology and utility experience with steam systems.

In general, steam coolant technology was felt to be simpler than sodium. The super heat ~~program~~ ^{had} shown that steam-cooled reactors could operate on a direct cycle, thus eliminating the primary heat exchangers and offering a more economical steam supply system. Some strong additional incentives for considering steam cooling included the possibility of flooding the reactor with a transparent medium (water) during rearing and maintenance operations, and the possibility of using water systems for emergency cooling.

Early studies involved the use of super critical pressure steam a coolant. There were two outstanding reasons for the selection of this

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pressure. The first was to obtain the high thermal efficiencies of modern fossil fueled power plants (43%). It was also recognized that the core power density would have to be high. The super critical fluid possesses better heat transfer characteristics than lower pressure systems, thus allowing higher power densities to be obtained by the core designers. Some of the concepts "boiled" the super critical coolant in the core itself. These concepts however required decoupling of the various core regions. The fast core regions, containing the lower density coolants, were separated neutronically from the boiler core region containing the higher density coolant which resulted in a softer neutron spectrum. The power system for this concept was identical to that for a fossil fueled super critical plant, thus representing a minimum of departure from current steam system technology. To obtain reasonable breeder performance, however, the core involved in the use of an inverted fuel element (coolant flows through the tubes with fuel packed around them). This represented a considerable departure from current fuels technology. It also resulted in a great deal of complexity in the headering design between the inlet lines, the coolant lines, *through the fuel and the exit lines*, from the reactor vessel.

A more interesting cycle concept was introduced in these early studies. It utilized super critical steam as a coolant, but the coolant was passed through the critical point, or boiled outside the core. This resulted in only low density coolant going through the core region of the reactor. Lowering the coolant density in the core eliminated the need for core segmentation and resulted in a pin type fuel design, *with* adequate power densities and breeder performance. The Loeffler cycle has since been adapted in all recent steam-cooled

fast breeder reactor studies. This cycle is shown schematically in Figure I. The steam exiting from the reactor vessel is ^{divided} divided with one part going to the power conversion portion of the cycle, the turbines, condenser, feedwater heaters, etc., and returning ^{to} the Loeffler boiler. Approximately six times as much flow goes directly from the reactor exit to the Loeffler boiler. The Loeffler boiler is a direct contact boiler where the six parts of steam coming directly from the reactor are mixed with the one part of ^{feedwater} steam returning from the power conversion cycle to convert the latter back to steam. The Loeffler boiler also contains steam separators to ensure only dry saturated steam enters the circulators. This ^s saturated steam leaves ^{ing} the Loeffler boiler is then pumped by the steam circulators through the reactor core and the cycle is repeated.

The Loeffler boiler and steam circulator present additional components to the normal steam power cycle. There is considerable experience with Loeffler boilers in Europe, however, not of the sizes needed for steam-cooled fast breeder reactors. Proposed steam circulators have included both axial flow and centrifugal types, depending on the ^{particular} plant arrangement. Although there is also some experience with steam circulators, it is not with the sizes needed to cool a 1000 MW steam-cooled fast breeder reactor. Primary development emphasis in scaling-up steam circulators would be in bearing and seals technology.

The Loeffler cycle has allowed the utilization of pin type fuel and resulted in a harder neutron spectrum. Performance resulting from ^{an} the early designs ^{of} utilizing the super-critical Loeffler cycle, showed a breeding ratio ^{of} 1.10 and about a 40-year doubling time. If the steam-cooled fast breeder is competing with light water reactors and fossil

fueled plants, this breeder performance is perhaps adequate. If, however, it is being compared with LMFBR's or gas-cooled breeders, as it is today, it is indeed a modest breeder performance.

To make the steam-cooled fast breeder more competitive with the LMFBR or the gas-cooled breeder, recent design studies have centered around lower pressures ranging from the 1250 psi B&W design to a 2600 psi Karlsruhe design. The B&W low pressure and earlier super-critical-pressure designs, along with the Karlsruhe intermediate-pressure design, were used by the Alternate Coolant Task Force to evaluate the potential breeder performance, economics, and to assess the R&D requirements, for steam-cooled fast breeder reactors. Along with these evaluations, a parametric ~~study~~^{study} was ~~which~~^{which} performed at Battelle-Northwest, attempted to predict the importance of the various design and operating parameters, such as core size and shape, system pressure and temperature on the breeder performance and economics of the SCFBR. The remainder of this paper is a brief summary of the results of this ~~study~~^{study}, as well as a discussion of the elements of basic technology, such as neutronics, fuels and materials, ^{and} thermal hydraulics, ^{which} must be answered before the ultimate ^bcapacities of the steam-cooled fast breeder reactor can be assessed.

The parametric study was performed in two parts. The first part was a neutronics ^esurvey. From a neutronics point of view, the main consideration is the atom densities of the various components in the reactor. With this in mind, three steam densities and three coolant fractions were chosen. Figure 2 shows the resulting nine cases and their gross characteristics. The atom densities corresponding to these nine cases more than cover the ranges of temperature, pressure and volume fractions of interest in the steam-cooled fast breeder reactor. The units on core steam inventory, throughout this study, are pounds per cubic foot.

core. It is a smeared steam density.

The study included the determination of enrichment, breeding ratios and reactivity coefficients for 1000 MW_e reactors using an average fuel exposure of 50,000 MWD/t of heavy metal. The sensitivity of these parameters to changes in steam volume fraction, steam inventory, core size, and core shape was examined.

The statics calculations were performed using a 1-D diffusion theory code, HFN. Eight-group cross sections were generated by the HRG and TEMPEST codes from a GAM-II cross section set revised by Oak Ridge National Laboratory to more closely agree with new data. Plutonium constants in this set are the preliminary ENDFB values by Greebler.

The 1-D calculations provided the necessary relationships between breeding ratio and plutonium enrichment. But burnup calculations had to be performed to determine the initial enrichment necessary to achieve a batch core exposure of 50,000 MWD/t. The burnup calculations were performed in the following manner. A minimum value of k , of 1.01 was selected. This allowed a reasonable amount of maneuvering room at the beginning of life, and accounted for nonuniformities at the end of life. A reflector plus blanket core saving of 35 centimeters based on preliminary 1-D statics calculations, was assumed. A zero dimensional, but fundamental mode burnup code, FCC, was used to trace the reactivity vs exposure for the various cores. If an initial k of 1.01 was insufficient to provide 50,000 MWD/t, then the initial reactivity was raised by increasing the initial enrichment. This method, by assuming no blanket contribution to the reactivity as plutonium builds in, causes slightly higher than actual initial enrichments to be calculated.

The other set of calculations performed in the neutronic survey were the calculation of voiding and flooding coefficients. The void coefficient is determined with the entire reactor, including the blanket void of steam. This situation is different from the familiar local sodium

void coefficient which is caused by a local over-power condition. Loss of steam occurs only when an external pipe rupture allows depressurization of the entire system. The void calculations were made with code combinations HRG, TEMPEST, and HFN. The flooding coefficient was determined for only the 8,000 liter cores. The one dimensional code HFN was used for the flooding calculations, but THERMOS was used with HRG to determine the thermal group cross sections. Both the core and blanket were flooded with 20°C water. Figure 3 is a summary of the information obtained for one core volume. Cores having a low steam volume fraction and low steam inventory yield the best breeder performance. It is also interesting to note that the reactors with the best breeder performance have the best loss of coolant characteristics. This type of information was used to guide our selection of design variables in the design and economics survey. Another way of representing the same data is shown in Figure 4, where any design falling in the area underneath the curve will have a breeding ratio as good as that for the curve and a void coefficient less than that for the curve. That is, any design having a coolant fraction and steam inventory which falls below the curve, C would have a breeding ratio greater than 1.1 and Δk void of less than zero.

The breeding ratios obtained in this neutronics survey were slightly lower than those obtained for specific designs in the Alternate Coolant Task Force Evaluations. There were two reasons for this. The first and probably most important reason was that the fifteen volume percent structural material in the core was all taken to be Inconel-525. This particular alloy has approximately 4% tantalum and 9% molybdenum as alloying agents. These are both strong neutron absorbers. The second reason was the slightly higher initial enrichments, resulting from our simplified burnup analysis, caused lower breeding ratios. The overall correlation,

obtain this data

however, was fairly good and the trends predicted would apply to specific reactor designs of the geometries considered currently.

It was concluded, after performing the neutronics survey, that one of the most critical needs of the steam-cooled fast reactor designer is to have some critical experiments performed with representative neutron spectrums. During the survey calculations the results were compared using a modified Russian cross section set and the Oak Ridge cross section set. These two sets of cross sections did not predict grossly different breeder performance; they did predict different void performance. The void coefficient necessary in steam-cooled fast reactors, however, is not well enough defined to be used as a design constraint in the preliminary studies that have been made to date.

One cross section uncertainty that has not been quantitatively evaluated is the α value of plutonium in the 1 to 25 keV neutron energy range. α is the ratio of capture to fission cross section and is, thus, very important in determining the initial core enrichment. The initial core enrichment determines the breeding ratio. Intuitively it seems, that since the steam-cooled fast breeder reactor has a relatively soft neutron spectrum, the higher α values for plutonium would severely impair its performance as a breeder. This speculation ^{has some merit for} ratios of less than ~~1.2~~ ^{very low performance breeders such as} ~~1.2~~ ^{as with breeding} ~~Because they~~ have mean neutron energies less than 100 keV and a significant neutron population in the energy ^{range} of 1 to 25 keV, where the uncertainty in the plutonium α exists. If, on the other hand, you look at one of the steam-cooled reactors with a higher breeding ratio, such as 1.4, ^{the} ~~that~~ means neutron energy is up over 160 keV. This is actually higher than some of the sodium-cooled fast reactor concepts that

reactors as
sodium

have a softened neutron spectrum to increase the Doppler coefficient.

In this particular case you would expect the performance to be effected even less than the sodium-cooled fast reactor design.

This is not the only comparison that should be made, however, when considering the affect of plutonium α on the steam-cooled fast breeder reactor. The designs considered in our neutronics survey have a very large fraction of the breeding occurring in the core and very little in the blanket. For instance, a reactor with a 1.4 breeding ratio has approximately 1.2 of that in the core and the other .2 in the blankets. The uncertainty of the α value of plutonium will have its greatest affect on the core conversion ratio. Perhaps the first thing to consider in designing a steam-cooled fast reactor with a higher value of plutonium α , is to get the breeding to take place more in the blankets and less in the core. One means of doing this is to design a core with higher leakage. Therefore, a designer might consider more seriously annular cores, segmented cores, or a modular core. In this manner, and possibly some other ways, the affect of an increase in the plutonium α value could be minimized to the performance of a steam-cooled fast breeder reactor. It's definitely premature to put any quantitative penalty on breeder performances without having a means of considering the design options.

The Heavy Water Reactor Program Office also requested that we look into the use of heavy-water-steam-cooling in fast breeder reactors. Two sets of neutronics calculations were performed on this reactor concept. First, D_2O was substituted in the lattice of the B&W super-critical pressure reactor design. The physics calculations performed on this design were made with a rather detailed model for both statics and burnup cases. Also for this design, the heavy water inventory was estimated for purposes of adjusting capital and operating and maintenance costs. The resulting of the neutronics calculations for this reactor concept were

interesting. The breeding ratio increased from 1.06 to 1.36. This factor of 6 increase in the breeding gain brought the doubling time ^{down} to approximately 10 years and gives the concept interesting performance as a breeder. The heavy water inventory in a very high pressure systems is comparatively low. There is about 350,000 lbs. of heavy water in the entire system. This, ^{results in} ~~reflects~~ a capital cost of from \$6,000,000 to \$7,000,000 -- considerably lower than any other heavy water reactor concept. Heavy water was also substituted into our survey model and calculations were made on our array of nine combinations of volume fractions and steam inventories. These calculations were carried out, however, for only one volume, an 8,000 liter spherical core. The resulting breeder performance is shown in Figure 5. These results were presented by Dr. S. J. ^{Altschuler} ~~Altschuler~~ at the Toronto ANS meeting in June of this year. Figure 5 shows ~~that~~ the largest gains are made in the marginal breeders such as in Case1B compared to Case0A, which had good ^{light-water} ~~H₂O~~ breeder performance. We are not absolutely sure, however, that this trend is valid. The 8,000 liter spherical core is rather large for an SCFBR and has most of the breeding occurring in the core. In general, the better cores are smaller than 8,000 liters and are highly pancaked with an H/D ratio, more typically, of 0.2. Thus, much more of the breeding occurs in the blanket. If we had redone the survey calculations for a smaller sphere, to get higher leakage, we may very well ^{have gotten a different trend} have heard estimates of increases ^I in breeding gains of 0.3, even at the dry conditions, when D₂O is substituted ^{the} lattice. This would result in breeding ratios ^{near} 1.6. Though this is hard to believe, the survey results may well underestimate the D₂O breeder performance at these drier conditions, when ~~D₂O is substituted into the lattice.~~

In summing up the conclusions of the neutronics survey, it can be said that designers need better ^Gross section information in typical neutron energies. In future steam-cooled fast breeder reactor concepts

designs, I would expect that you might see some using D₂O as coolant, ^{and} possibly with an indirect cycle. If the α of plutonium does prove to be higher in the 1 to 25 kev range, than currently assumed in this country, you may see some spoiled geometry designs such as annular, segmented or modular cores.

The second part of the parametric study was ^{to} perform a design and economics survey. The calculations for this survey used the analog simulation techniques developed by BNW for the preliminary study of potential FFTF core designs. The simulation consists of two parts, a core design model and an economics model. The enrichment and breeding ratios resulting from the neutronics survey were correlated with geometric buckling, fuel volume fraction and steam inventory. These correlations were incorporated into the design simulation, along with the normal thermal hydraulic analysis and calculations for power plant cycle thermal efficiency. The economics simulation was patterned after the ORNL treatment in order to obtain results that could be compared with their evaluation studies. Because the overall simulation was large and algebraic in nature, the actual calculations were performed with ^a digital computer program, MIMIC. MIMIC is one of many available digital analog simulators. The simulation was used to estimate the effect on breeding ratio doubling time, and fuel cycle costs of steam pressures from 1,000 to 3,000 psi, steam temperatures from 900 to 1100 °F, core height to diameter ratios from .2 to .6 and core volumes from 6,000 to 10,000 liters.

~~This may be the most severe limitation to the ultimate performance of steam-cooled fast breeder reactors. They will have to develop a separate advance fuel technology.~~

~~A great deal of effort was expended in the super heat program to identify possible clad materials. There were two alloys that seemed~~

There are several facets of design technology, which have a bearing on the performance of steam-cooled fast breeder reactors. The fuel form for SCFBR's is a mixed oxide fuel very similar to that being considered for LMFBR's or gas-cooled fast reactors. The status of fast reactor oxide fuels will be discussed, I'm sure in applicable technical discipline sessions later in this conference. One of the design constraints incorporated into BNW design simulation was that the maximum fuel centerline temperature would not exceed 4850°F. As an independent design parameter was varied, such as system pressure, the simulation would automatically adjust the number of fuel pins in the core to maintain the fuel centerline temperature constant.

One of the problems the steam-cooled fast breeder LMFBR's and gas-cooled reactors are compatible with the steam environment. This may be the most severe limitation to the ultimate performance of steam-cooled fast breeder reactors. They will have to develop a separate advanced fuel technology.

A great deal of effort was expended in the super heat program to identify possible clad materials. There were two alloys that seemed to show promise

to show promise for application in steam-cooled fast breeder reactors. One is Inconel-625, which has been mentioned as being used in the BNW parametric studies, and the other is Incoloy-800, which has two advantages for SCFBR application. It uses titanium and aluminum as alloying agents to obtain long-term stress rupture properties, as opposed to tantalum and molybdenum used by Inconel-625 in large amounts. Incoloy-800 also appears to retain its ductility better at high neutron exposures. There is available from the super-heat program yield stress data as well as creep and stress rupture data for these alloys. There may, however, be a need for high and low cycle fatigue data when all of the possible modes of failure in SCFBR fuel are identified.

There are several alternate clad design techniques which lead to different failure modes. All of the steam-cooled fast breeder reactor concepts, published to date, have used prepressurized pins to keep the clad from collapsing during initial operation. The failure mechanisms of concern in this concept are the creep and stress rupture possibilities near end of life when fission products have built up ⁱⁿ a high burnup pin. Also of concern at the end of life, is the burst strength in case of system depressurization. At any rate, the performance of pre-pressurized fuel pins has not been demonstrated. Another design constraint used in the design simulation, was the maximum allowable clad temperature. Again, analog techniques were employed so, as an independent design parameter was varied, the fuel pin diameter would be adjusted to maintain ^{constant} clad temperature of 1390° F maximum.

Venting or pressure equalization systems have not, as yet, been proposed for SCFBR applications. They would present problems in a direct cycle, unless a highly reliable system can be designed and demonstrated.

Another aspect of fuel design which differs from other fast reactor concepts is the fuel bundle. As was noted from the neutronics survey, to obtain good breeder performance the coolant fraction must be low. At 25 volume percent coolant, with typical SCFBR fuel pin dimensions, the pitch to diameter ratio of the pins in a bundle would be approximately 1.05. This is a low enough value to cause concern in the accuracy of the heat transfer and hydraulics calculations, as well as presenting a possible problem in emergency core coolant techniques. Therefore, the minimum coolant fraction allowed in the design simulation was 0.25.

The super heat program, as well as an extensive experimental program by Westinghouse over a wide range of temperatures and pressures, has provided the basic correlations necessary to predict the heat transfer and hydraulic performance of SCFBR cores. The uncertainties in this area lie in the effect of various potential rod spacer designs. The Karlsruhe intermediate pressure design, with a coolant fraction of 32 volume percent, utilizes integral spiral fins as a spacer design. They have performed considerable amount of experimental work verifying the thermal and hydraulic performance of this concept. The BNW low pressure design, however, with very close pitched pins, has proposed an egg crate spacer design where the fuel rods are necked down in the area of the spacers. By necking the fuel down in the spacer region, the coolant flow cross section is kept approximately the same all along the coolant channel. While this would appear to enhance the thermal and hydraulic performance, it would certainly require some experimental program to verify the performance. A friction factor multiplier was specified in the design simulation to predict the effect on the hydraulic performance of different fuel pin spacer designs. This factor simply multiplies the smooth channel fanning friction factor according to experimental

results for various spacer designs that were developed by deStordeur, in this country and by the people at Karlsruhe.

Steam velocities ^s over 200 feet per second have resulted from some of the designs considered. This may present a problem in flow induced vibrations. It may also present a problem in erosion of clad material or other structural material in the core. Experimental consideration of these problems may well require lower steam velocity designs or cause new failure modes to be considered in clad design.

Steam circulators must move as much as 35×10^6 lbs/hr through the reactor and Loeffler boiler portions of the cycle. This results in high pumping powers. The Karlsruhe design has a rather conventional plant layout with most of the components of the Loeffler cycle and power conversion system outside of the pressure vessel. They have a pumping power of 66 MW, half of which is used to move the steam through the external circuit. A large fraction of the other half is taken up in the headers necessary to get the steam in and out of the core region. Thus, only a small fraction of the pumping power is used to move the coolant past the heat transfer surfaces in the core. If this plant arrangement is used at lower steam pressures, it soon becomes obvious that to keep the pumping power reasonable will require large components which will be very costly. A more streamlined plant arrangement would have a significant economic advantage. ^{PP} Figure 6 shows the design, note that only the steam going to the turbines leaves the pressure vessel. The steam necessary ⁱⁿ to the Loeffler boiler, ^e remains in the pressure vessel. Within the pressure vessel are the Loeffler boiler, the steam circulators and the steam separators. This plant arrangement, along with the streamlining of the fuel by necking down the fuel pins in the spacer region, allows BNW to concentrate their 100 MW of pumping power on the heat transfer

PP Figure 6 shows the plant arrangement used by BNW in their low pressure steam-cooled fast reactor concept.

surfaces in the core. Different plant arrangements were symbolized in the design simulation by specifying the out-of-core pressure drop.

Figure 7 shows the relationship of pumping power to system pressure for three relatively different design concepts from the parametric survey. They are different not only in core dimensions but also in plant arrangement and the core hydraulics assumed. The first design on this slide is modeled after the Karlsruhe D-1 design in its plant arrangement and fuel design. The second design is modeled after the B&W low pressure design as far as plant arrangement and fuel design is concerned. Different fuel designs are indicated by the friction factor multipliers. ^{There is} ~~We have~~ a factor of 2.0 for the integral fin tubes, and a factor of 1.⁰ for the constant flow cross section B&W fuel design. The third design is identical to the second design except the out-of-core pressure drop is 80 psi and the friction factor multiplier is 1.4. Note in Figure 7 that the streamlined design is necessary to achieve the very low pressures, around a 1000 psi, and still maintain a reasonable pumping power.

Figure 8 shows the relationship between breeding ratio and system pressure for the same three designs. A significant gain in breeding ratio, at a constant pressure, is obtained by redesigning the core. Designs 2 and 3 have core volumes of ~~the~~ 7,000 liters and a height-to-diameter ratio of 0.2. The increase in the breeding ratio is caused by the lower height to diameter ratio. This results in a significantly higher fuel volume fraction or ^C conversely a lower coolant fraction. The gain at constant pressure is similar in magnitude to the change in performance obtained by reducing the pressure to a 1000 psi in the streamlined designs.

Figure 9 shows the relationship of doubling time with system pressure for the same three designs. Again, note that redesigning at constant pressure has gained as much as varying the pressure to lower values. The gain in doubling time is a function of two things, comparing designs 2 and 3 with design 1. First of all, the increased breeding ratio causes a decrease in doubling time, but the smaller core volume of 7,000 liters reduced the fissile inventory, which ^a gives an additional increase in doubling time. Doubling times close to 10 years at very low system pressures were obtained.

Figure 10 showing the fuel cycle costs is rather interesting in some respects. ^{It indicates} We found you neither gain nor lose, in fuel cycle economics, by designing a better breeder reactor. Although you would expect cheaper fuel cycle cost with increased breeding, the gains made in plutonium revenue are offset by higher fuel fabrication costs. The unit fabrication costs used in the BNW study were those developed by ORNL for use in evaluating the alternate coolant fast reactor concepts. The particular values we used were based on a 15,000 MW_e industry size. The unit costs were put in our simulation as a function of fuel pin diameter.

The results of this parametric study, as well as the way we had to perform it, lead us to the following conclusions: (1) fuel cycle costs are very weakly affected by choice of steam pressure (2) choice of steam pressure in a SCFBR will depend on whether the chief objective in a given design is to minimize capital cost or to emphasize breeding ratios and doubling time (3) performance in terms of breeding ratio, doubling time and fuel cycle costs is affected as much by design approach and assumed limitations as by choice of steam pressure, (4) designs which reduce hydraulic resistance, throughout the system, contribute significantly to obtaining the potential benefits in breeder performance at low steam pressures.

FIGURE 1

LOEFFLER CYCLE SCHEMATIC

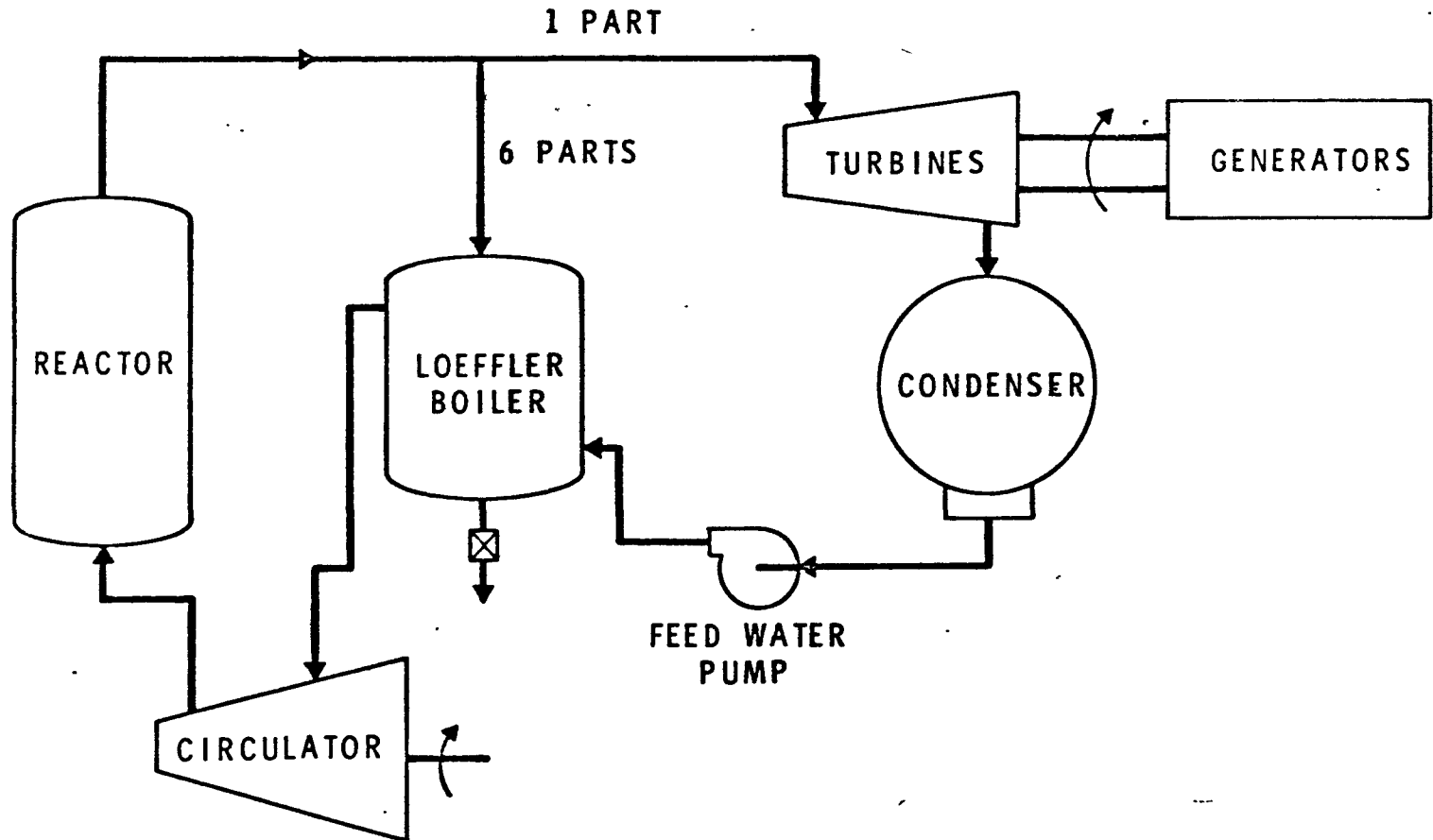


Figure 2

Parametric Study Core Characteristics

<u>Core Steam</u>	<u>Core Volume Fractions</u>			<u>Average Steam</u>	
Inventory, (lbs/ft ³)	Fuel	0.60	0.45	0.30	Temperature,* °F
	Steam	0.25	0.40	0.55	
	Structure	0.15	0.15	0.15	
<hr/>					
0.367	Case OA 1400 psi	Case IA 880 psi	Case 2A 650 psi		1200
1.18	Case OB 3400 psi	Case 1B 2300 psi	Case 2B 1700 psi		1000
2.83	Case OC 4100 psi	Case 1C 3400 psi	Case 2C 2800 psi		800

* *Temperatures and pressures contained in the tables are typical values which result in a density that will yield the steam inventory shown in column one.*

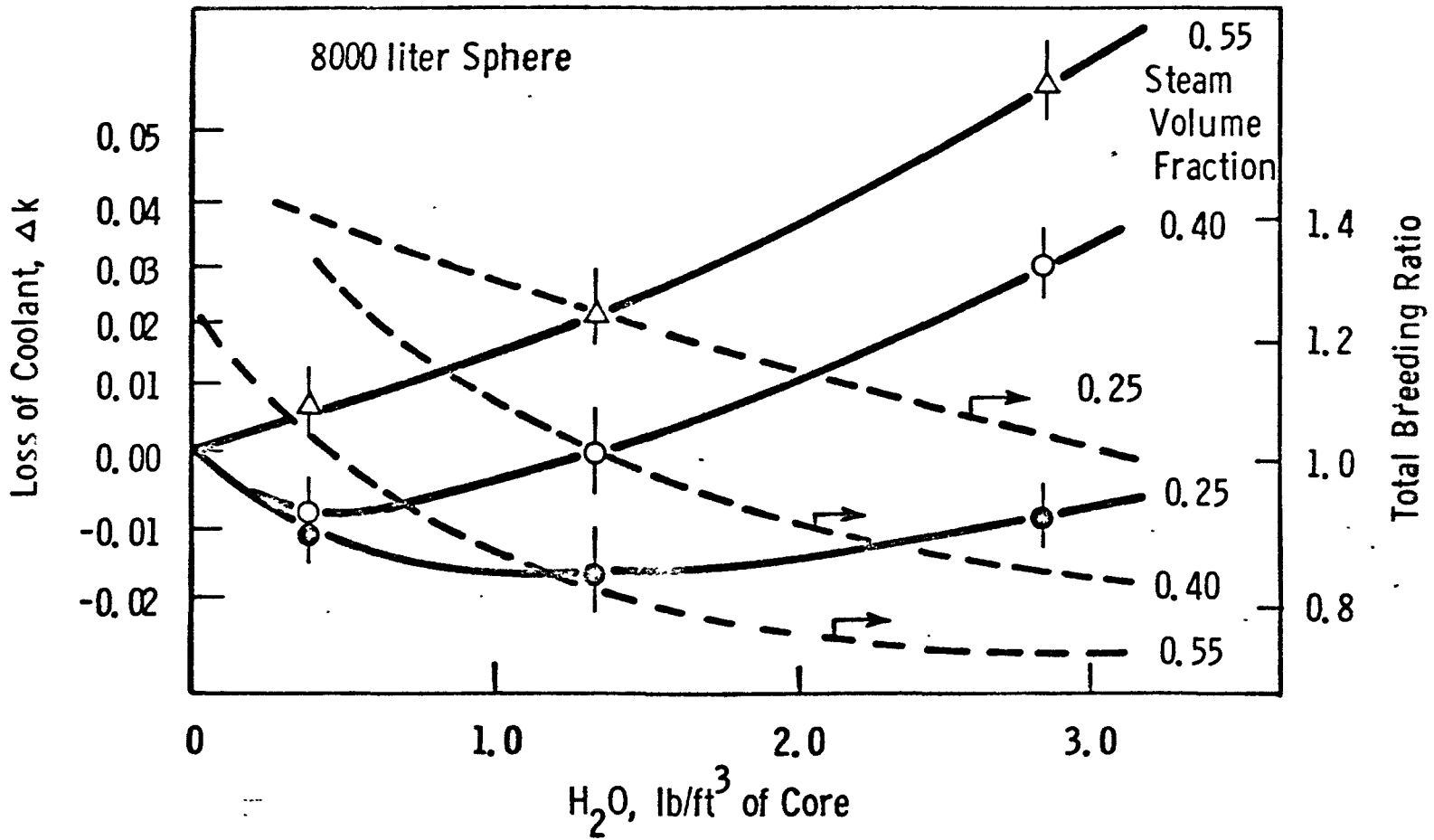


FIGURE 3
 Loss of Coolant Characteristics
 for 8000 Liter SCFBR Cores

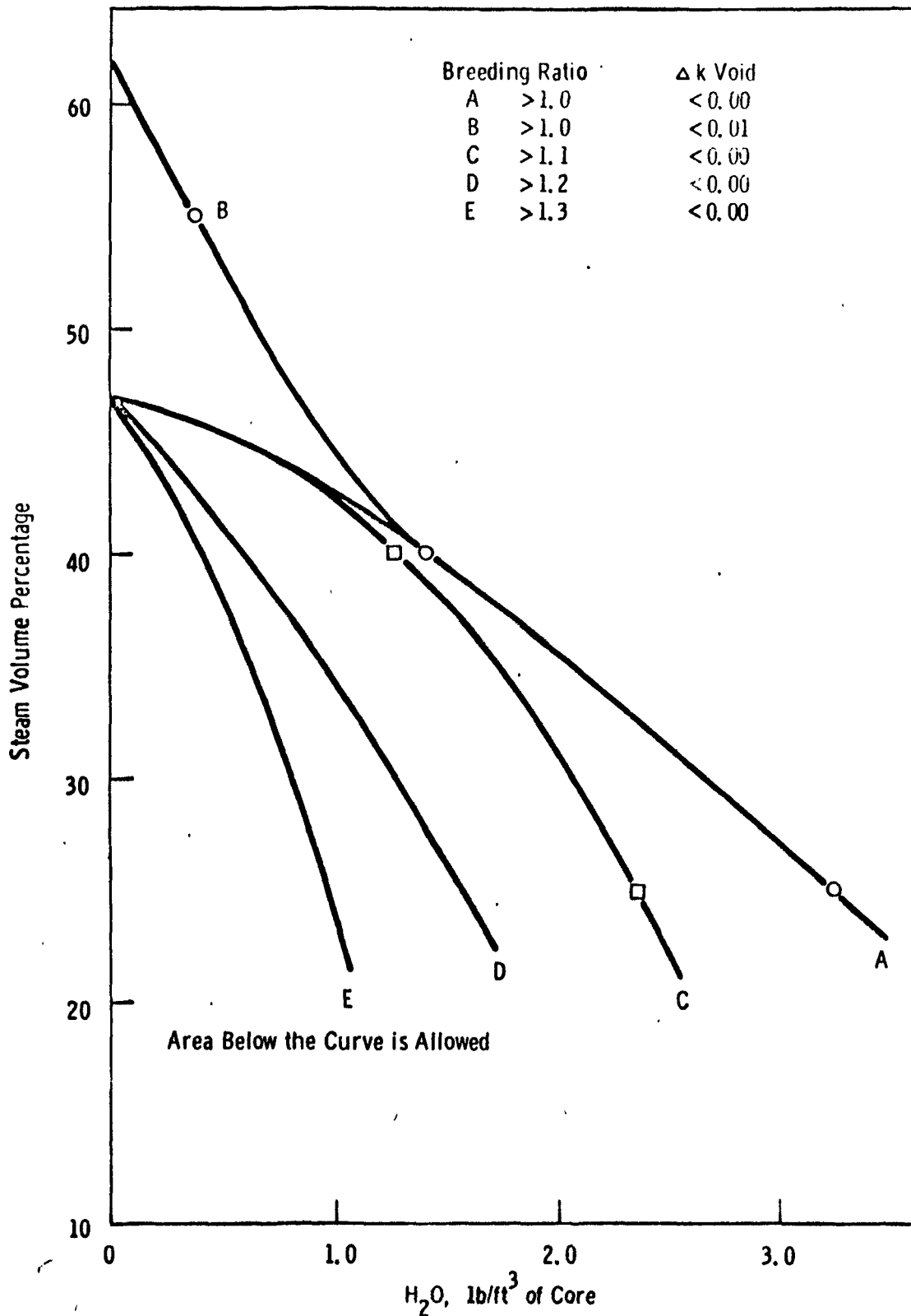


FIGURE 4

Allowable Combinations of Coolant Fraction and Steam Inventory for an 8000 Liter SCFBR Core

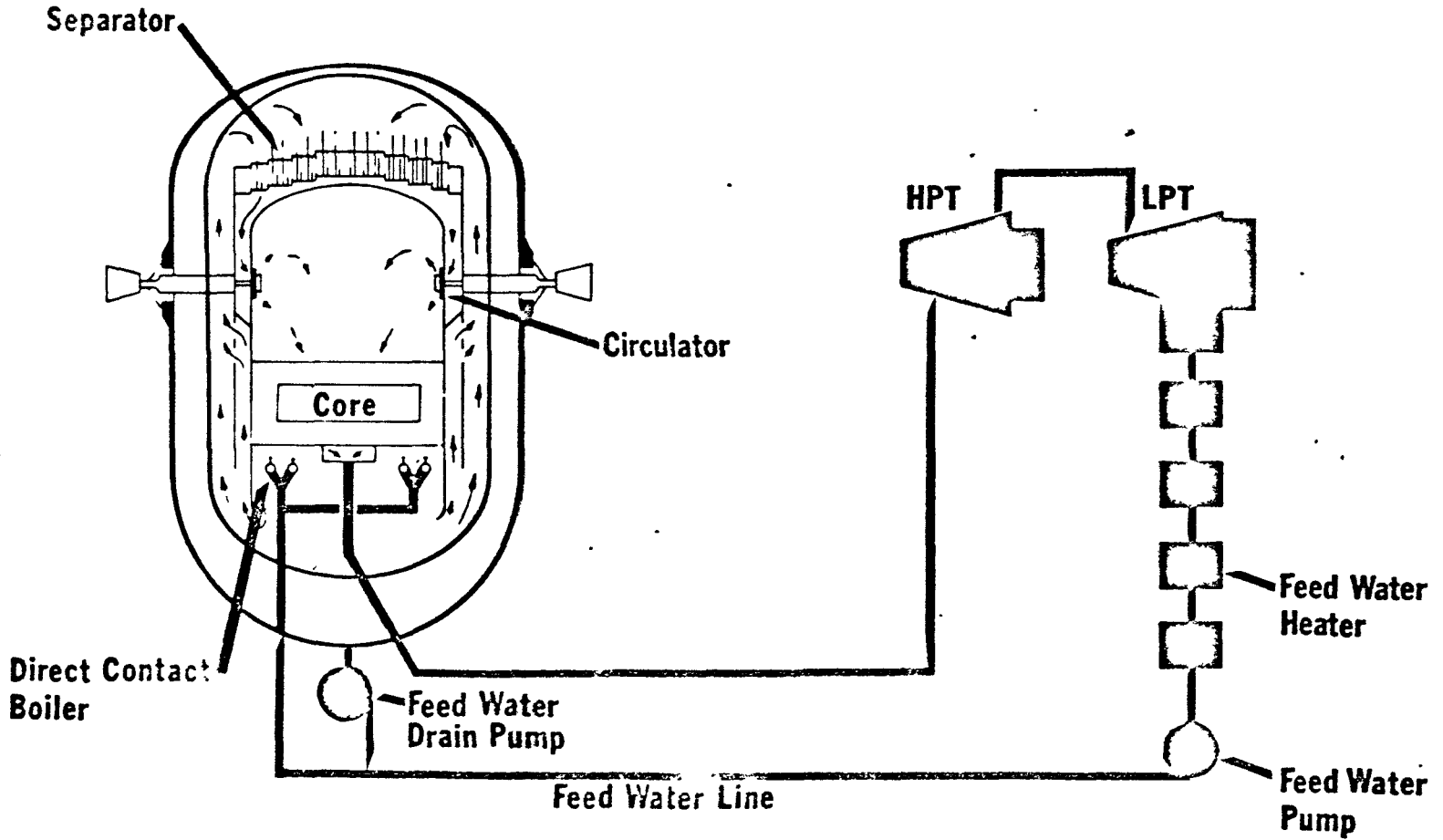
Figure 5

COMPARISON OF HEAVY AND LIGHT STEAM
FOR 8000ℓ SCFBR'S

CASE	INITIAL k_{eff}	ENRICHMENT % Pu	BREEDING RATIO		AVERAGE TOTAL	DOUBLING TIME, YR
			AVERAGE CORE	INITIAL BLANKET		
0A	1.0100	11.30	1.125	0.187	1.312	11.54
	1.0100	11.44	1.136	0.224	1.360	10.12
0B	1.0100	11.29	1.061	0.147	1.208	17.29
	1.0100	11.78	1.120	0.203	1.323	11.62
0C	1.0260	11.63	0.943	0.107	1.050	74.1
	1.0100	11.98	1.095	0.183	1.278	13.73
1A	1.0115	13.14	0.997	0.244	1.241	13.03
	1.0100	13.03	1.044	0.267	1.311	10.02
1B	1.0350	13.75	0.905	0.180	1.085	39.2
	1.0100	13.31	1.023	0.244	1.267	12.08
1C	1.0600	14.65	0.790	0.104	0.894	-----
	1.0155	13.86	0.980	0.198	1.178	18.61
2A	1.0685	18.32	0.750	0.271	1.021	139.0
	1.0585	17.85	0.799	0.295	1.094	30.3
2B	1.0830	19.13	0.680	0.213	0.893	-----
	1.0655	18.72	0.765	0.264	1.029	103.0
2C	1.0985	20.53	0.607	0.130	0.737	-----
	1.0717	19.58	0.735	0.225	0.960	-----

FIGURE 6

SCDR SCHEMATIC



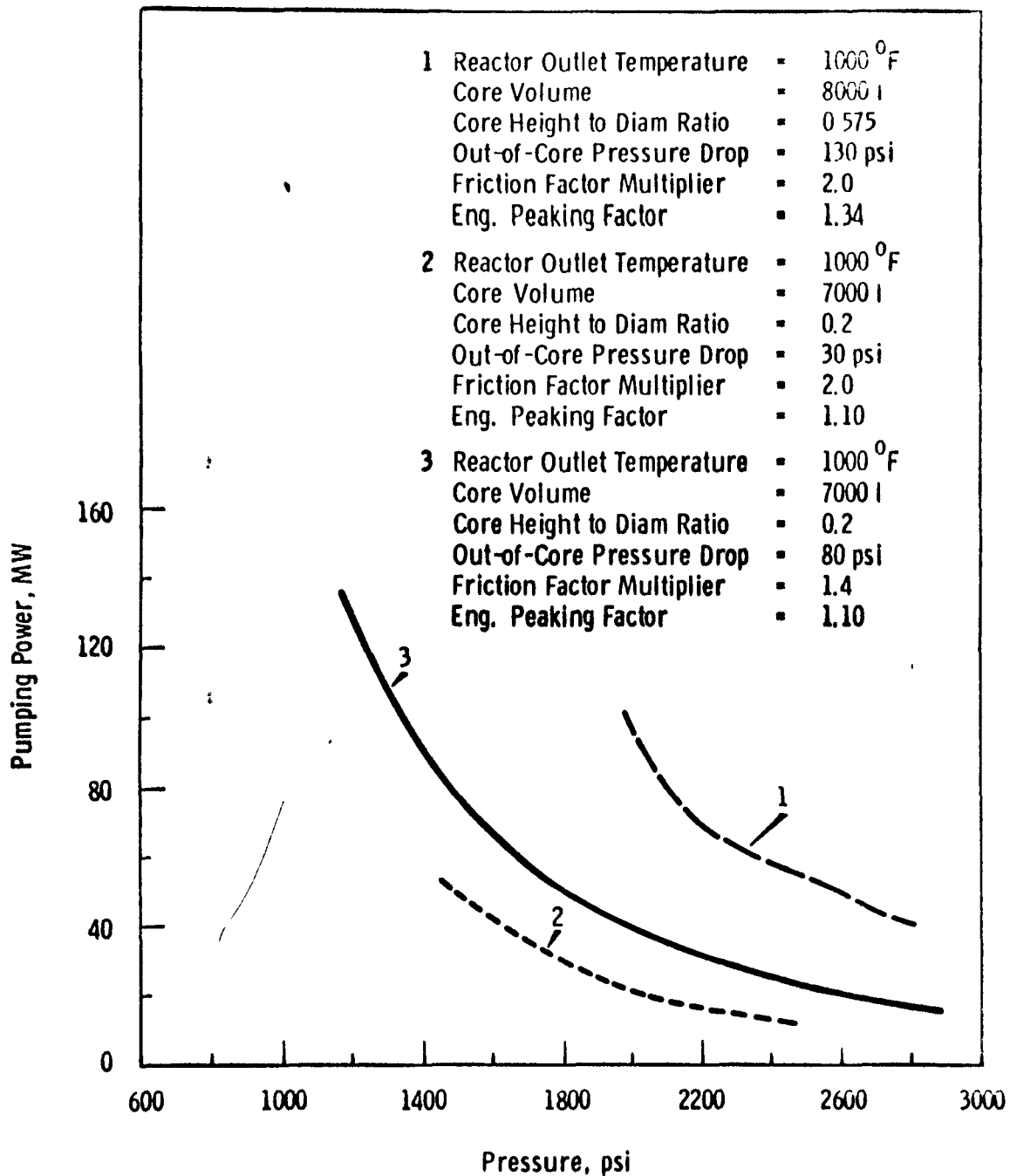


FIGURE 7

Pumping Power Versus
System Pressure

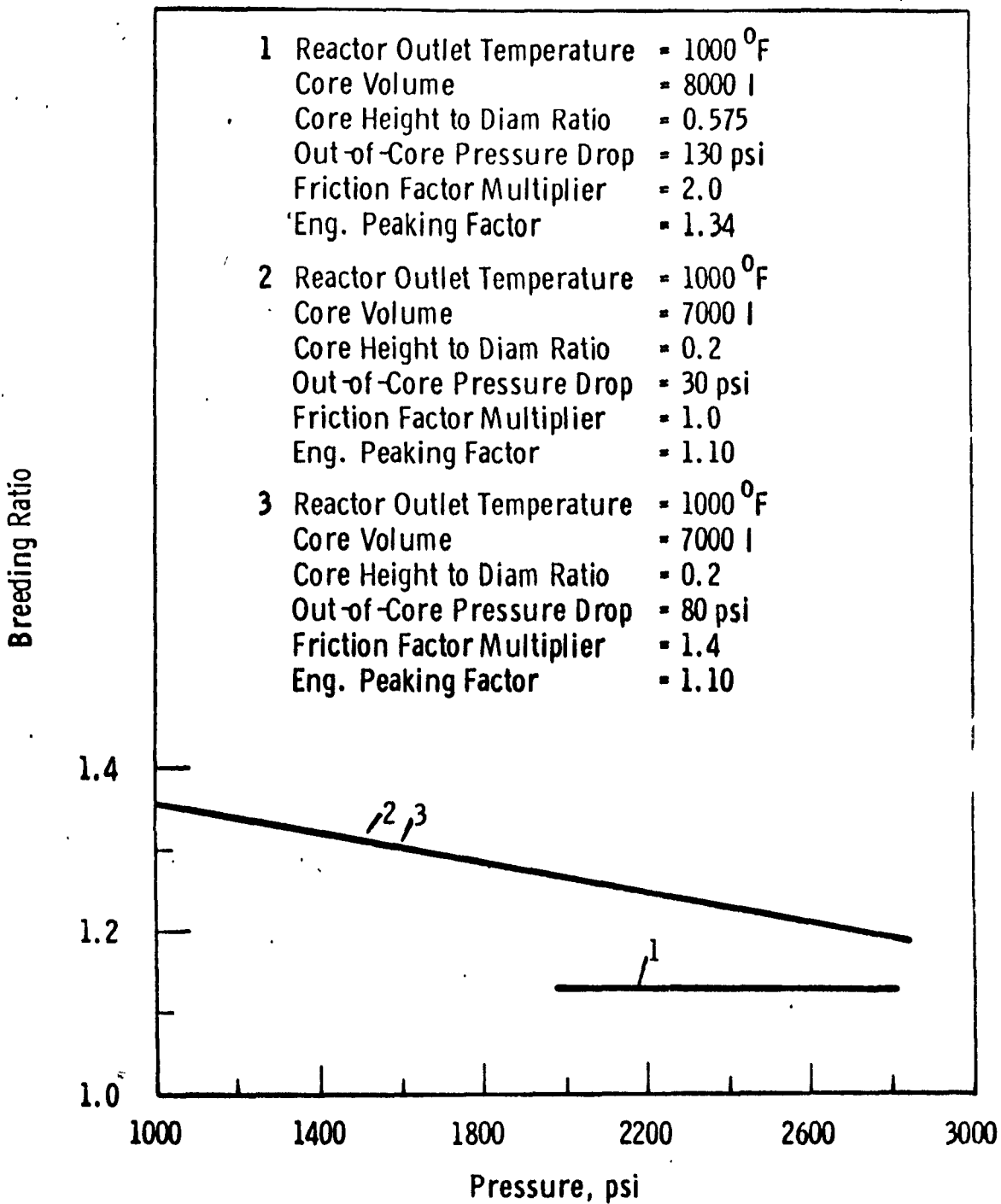


FIGURE 8

Breeding Ratio Versus
System Pressure

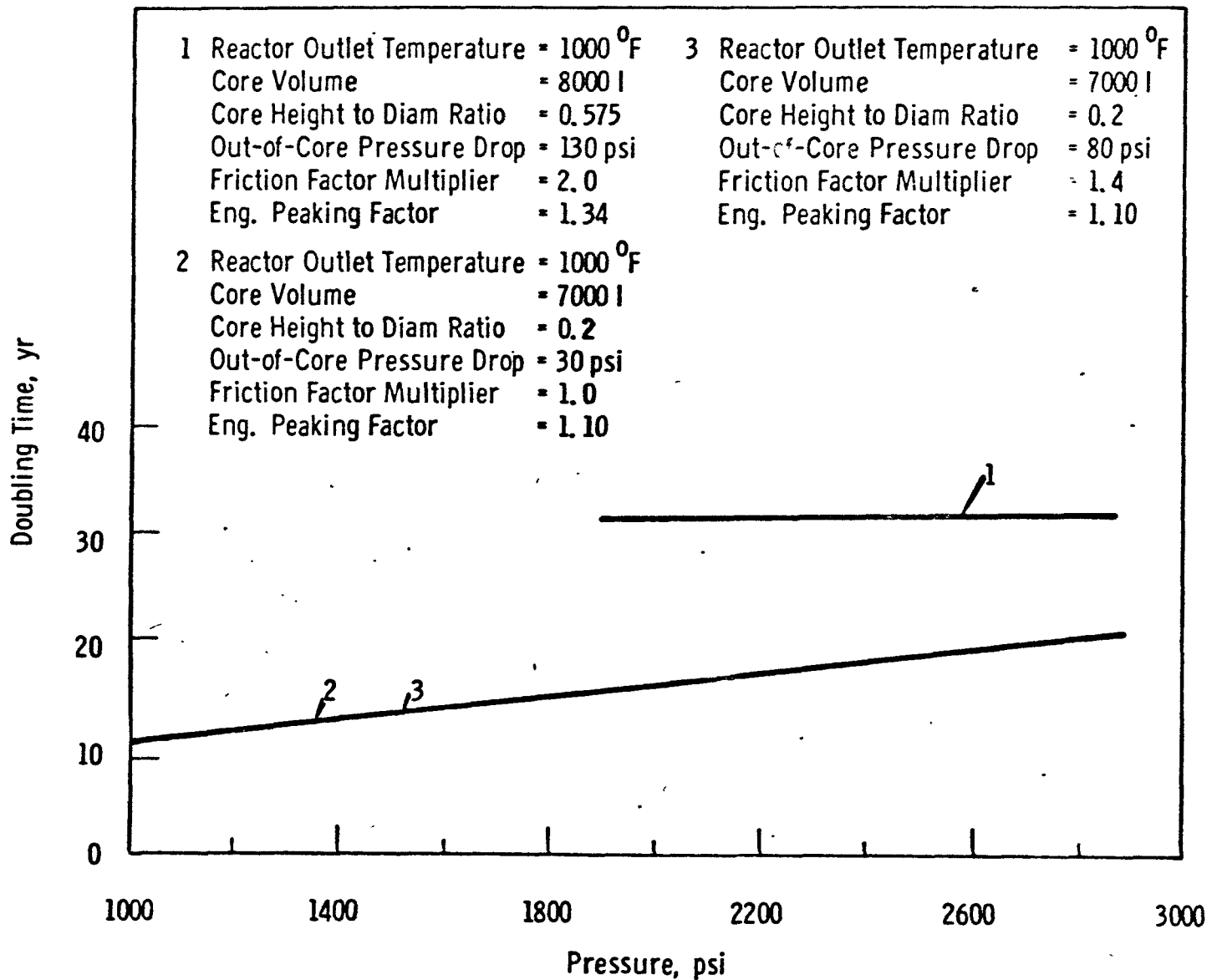


FIGURE 9

Doubling Time Versus System Pressure

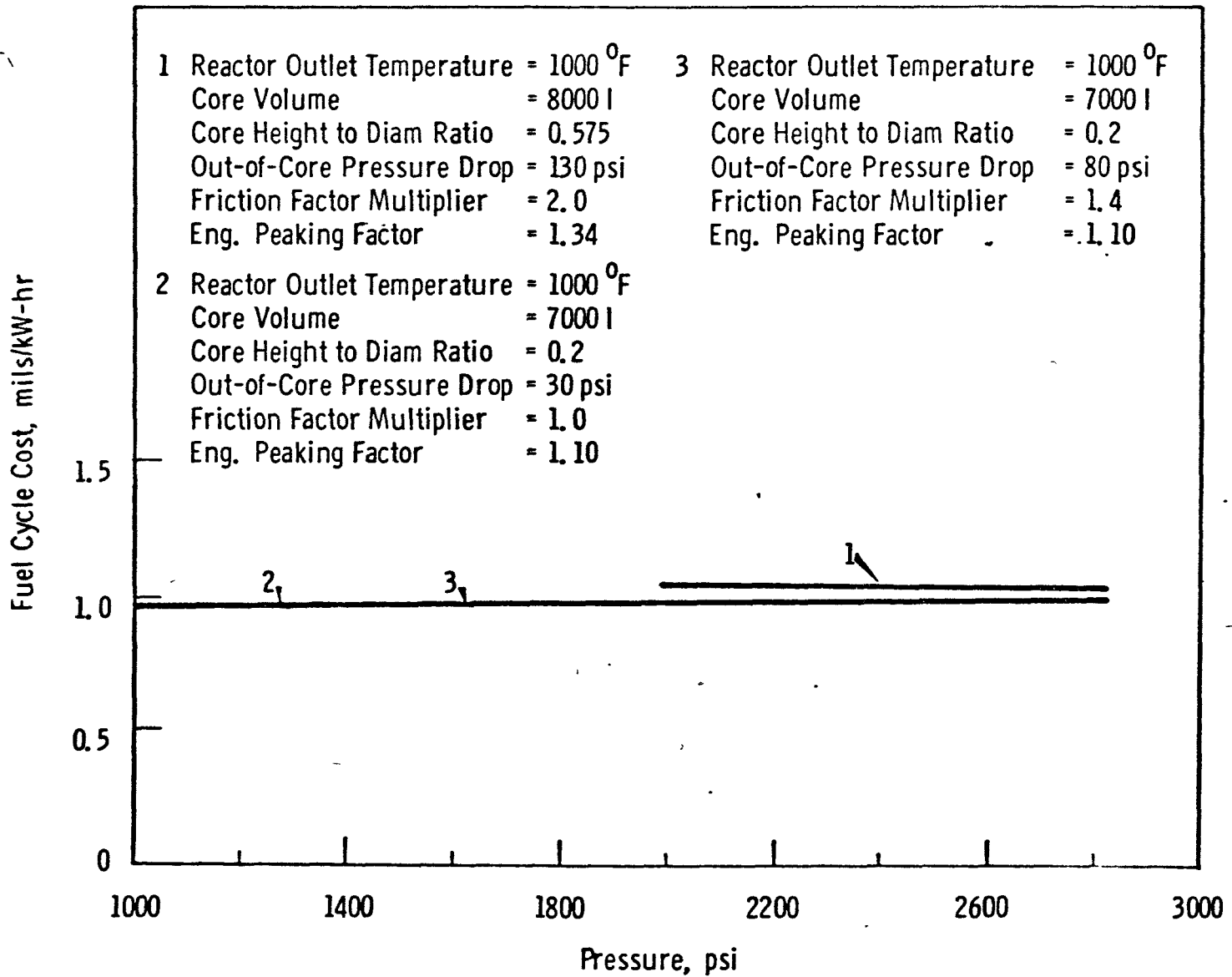


FIGURE 10

Fuel Cycle Costs
 Versus System Pressure