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FAST FLUX
TEST FACILITY
PROJECT REPORT

TYPICAL CHARACTERISTICS
OF ADVANCED CORES
IN THE DISPERSED FTR

December 1968

AEC RESEARCH &
DEVELOPMENT REPORT

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TYPICAL CHARACTERISTICS OF ADVANCED CORES
IN THE
DISPERSED FTR

R. B. Rothrock

I. INTRODUCTION

A. OBJECTIVE

Future testing capability goals for the FTR include a peak flux of 10^{16} n/cm²-sec and coolant bulk outlet temperatures up to 1200 °F. It is the objective of this study to predict core design constraints that will determine future high performance core designs; to develop a typical advanced core design utilizing a specific and realistic fuel system, in sufficient detail to confirm that the dispersed core arrangements are capable of meeting future testing goals. Other objectives are to identify features of the reactor and plant design that are essential to the capability of accommodating future high performance cores such as system power level, and reactor lattice dimension.

Two dispersed core arrangements (Annular Cone and Annular Skew) developed by PNL to yield a relatively compact core, meets the initial testing goals with mixed oxide fuel while providing sufficient separation of subassemblies at the cover for individual nozzle access to each core position for fuel handling and instrumentation.⁽¹⁾ The two arrangements are similar in general configuration, having 91 total core positions arranged in a hexagonal lattice with the central seven core positions (the "Test Region") modified to accept three vertical closed loops, while additional closed and open loop tests can be installed in any of the remaining 84 lattice positions. The core and skew arrangements differ principally in the means by which separation of the fueling nozzles at the reactor cover is achieved. The core subassemblies (except for the three vertical loops) are tilted outward in a radial plane in the cone, at an inclination of 0.9 degrees per row. In the skew, they are rotated about a radius at the axial midpoint of the fuel (the twist plane) through an angle of about two degrees per row of lattice positions. The

important consequences of this difference in geometry with respect to conceptual design of advanced cores are:

- (1) The skewed arrangement is more compact in the core region.
- (2) Sufficient clearance is developed between adjacent subassemblies in the core because of the radial tilt to permit a transition in the subassembly flow ducts from hex to round cross section at a point shortly above the top of the fuel. This increases the flow area available above the fueled region and reduces the hydraulic penalty associated with long gas plenums. A similar transition cannot be completed in the skewed array until about six feet above the twist plane, although a gentle taper can be started at a lower point to provide some relief.
- (3) The cone geometry permits locating three full size driver subassemblies or open tests in inclined lattice positions adjacent to the three vertical closed loops, within the central 7-cluster of lattice positions. This cannot be done with the skewed geometry unless either the vertical closed loops are converted to utilize inclined lattice positions, or special fuel subassemblies are used.

Since these differences between core arrangements affect the design and performance of advanced cores, typical core designs will be developed for both the annular cone and the annular skew geometries.

B. ADVANCED FUEL SYSTEM SELECTION

The specific core designs to be developed in this study will be "near term" high performance cores; i. e., probably the first FTR replacement core after the initial oxide core operation. This replacement core might be loaded within three to four years after initial criticality, and its design and procurement would therefore be initiated in the early to mid 1970's. The chronology is important in order to judge the degree of extrapolation of current fuel technology that should be made in predicting future fuel capabilities. This schedule places the procurement of the FTR replacement core along with or shortly after the fabrication of one or more LMFBR demonstration plant cores. Manufacturing technology and capacity acquired for this purpose can be available for the FTR if similar fuel is used. In regard to irradiation

performance, it is expected that any fuel system currently under active development for LMFBR use will accumulate sufficient irradiation experience to provide a basis for FTR replacement core design. In general, fuel systems for which there is no visible commercial interest at present, as judged by the 1000 MWe studies and conceptual designs for demonstration plants, are considered unrealistic for FTR replacement core use on a short-term basis even if they promise considerable performance advantages. This judgement is based on the lead time needed to establish commercial facilities to manufacture and to reprocess several FTR core loads of a "new" fuel type. It is assumed that the expense of developing this capability could not be justified by possible performance gains in the FTR alone. This argument excludes some potentially attractive fuels such as UN-PuN for the first replacement core, but not necessarily for subsequent longer-term replacement core designs.

Within a limited total heat removal capability, an increase in peak flux to the 10^{16} range (core average specific power of $1 \text{ MW}_t/\text{kg}$ fissile) requires a substantial reduction in core volume and loading from the 1000 liter, 500 to 600 kg designs that have been developed for the initial oxide fueled dispersed cores, and a corresponding increase in power density and fissile loading density. Improved thermal performance and high fissile atom density are therefore important in selecting potential fuels for advanced FTR cores. Mixed UC-PuC is the only fuel system now under active development which offers a significant improvement over $\text{UO}_2\text{-PuO}_2$ in these two areas and which appears likely to be commercially available in about the mid 1970's, judging from current LMFBR studies and programs. Consequently, the advanced core designs developed in this survey use UC-PuC fuel.

A conceptual cladding selection must also be made in order to complete a consistent advanced core design. Fortunately, the choice of cladding material does not appear to be crucial to the flux capability, but it does determine the maximum operating temperatures that can be achieved. The FTR goal of 1200 °F bulk outlet can be achieved only with a high temperature cladding material such as the refractory metal alloys

under current development. Current conceptual designs for demonstration plants in the 1970's are based on austenitic cladding and outlet temperatures in the range 1000 to 1100 °F. It appears unlikely that commercial sources of refractory alloy cladding will be developed in time to be utilized for the first FTR replacement cores, and stainless steel cladding is therefore assumed to be used. It is probable that the FTR would not reach the same bulk outlet temperature that a demonstration plant could achieve, if both cores were designed for the same peak cladding temperature, because of the higher coolant ΔT and probably higher peaking factors in the FTR. A bulk outlet temperature of 1000 °F will be used as an objective in developing a conceptual carbide core design, and the coolant exit temperature of individual test positions can be raised above this value as necessary by adjusting test position flow.

The selection of gas or liquid metal bonded fuel also has an important effect on the fuel characteristics and performance. A previous survey of carbide fuel performance for the FTR⁽²⁾ concluded that gas bonded fuel could be developed for short term use with greater assurance than liquid metal bonded fuel. In addition, gas bonded fuel appears to be preferred for carbide fueled demonstration plants and would probably yield lower fabrication costs. For these reasons, gas bonding is assumed for this conceptual advanced core study.

Wire wrap spacers will be assumed in developing conceptual designs, because of their favorable pressure drop characteristics.

II. CARBIDE CORE DESIGN CONSIDERATIONS

A. CORE POWER, SIZE, AND LATTICE DIMENSIONS

An increase in outlet temperature as planned for future cores will increase the HTS heat dissipation capability above its 400 MW (900 °F outlet) design basis;⁽¹⁾ normal engineering design margin, or intentionally built in stretch capability, might also permit an increase in system power for future cores. However, for the initial replacement cores it is more prudent to try to achieve the desired specific power goals within the 400 MW design basis, reserving any additional stretch

capability for further performance improvements in subsequent cores. The designs to be developed in this memo will assume a 400 MW system capability. To reach the goal specific power of 1MW/kg fissile, a core fissile inventory of 400 kg is evidently required. This loading can be obtained in the annular geometry with a total core volume of about 600 to 650 liters. For a 32 in. fuel length and lattice dimensions, at the pinch plane in the 4 to 5 in. range (permitting the re-entrant closed loop tube design with annular downcomer), the number of core lattice positions for a 650 liter core is tabulated below, based on conical geometry with a 0.9 degree per row inclination.

<u>Lattice Pitch at Bottom of Fuel, in.</u>	<u>Volume of One 32 in. Core Position, liters</u>	<u>Number Core Positions for 650 liters</u>
4.0	8.4	77
4.5	10.5	62
5.0	12.8	51

Previous core arrangement studies favored a position core for both cermet and carbide fuel;^(2, 3) arrangements containing fewer than 61 positions have excessive worth of core components and result in using a relatively large proportion of the core volume for parasitic components such as low-loaded tests and control rods.

A 61-position core also has the advantage of having (for a 650-liter volume) about the same lattice dimension as the 91-position conical dispersed core designs currently being studied for the initial oxide cores. Therefore, a 61-position conical core provides an arrangement capable of meeting the goal specific power with an acceptable lattice dimension (4.5-in.) and fuel length. For compatibility with current dispersed oxide core designs, the subassembly outside dimension across flats should be chosen as 4.420 inch. This skewed

arrangement yields tighter packing in the core, so that a smaller core volume and higher specific power is obtained. Dimensions for the two cores will be taken as:

	<u>Annular Cone</u>	<u>Annular Skew</u>
Subassembly outside dimension across flats	4.420 in.	4.420 in.
Mean core lattice pitch	4.75 in.	4.55 in.
Total number core positions	61	61
Total core volume, for 32 in. fuel	627 liters	574 liters

B. CORE ARRANGEMENT

In addition to driver fuel subassemblies, the core lattice must include open and closed loop tests and (possibly) in-core control and safety rods. Five closed loop and eight open loop tests are assumed, consistent with the testing requirements developed for the annular cone conceptual dispersed core design.⁽⁴⁾ Control rod worth requirements for smaller cores can be projected from those developed for the initial oxide cores as follows:

	<u>Worth, oxide core</u>	<u>Assumed Worth, advanced core</u>
<u>Primary Safety System</u>		
Total Worth (3-rod system)	3 to 4% $\Delta\rho$	3 to 4% $\Delta\rho$
<u>Shim-Secondary Safety System:</u>		
Burnup Δk per cycle	2%	3%
Test shim	1%	1%
Temperature defect	1%	1%
Shutdown margin	1%	1%
Uncertainty allowance	1%	1%
Stuck rod allowance	Highest worth rod	

The primary safety system total worth requirement is not expected to change, and three in-core rods are allocated per primary safety system, each worth 1 to 1 1/2% $\Delta\rho$. The total shim control required is expected to increase by about 1% $\Delta\rho$, because of higher burnup allowance. This is intended to allow for lower total fissile inventory, and possibly lower breeding ratio, assuming that the operating

cycle would not be shortened. Alternatively, if increased shim worth proves costly to obtain in terms of core performance, the operating cycle might be reduced. It is assumed that the total allowance for temperature defect, shutdown, and uncertainty need not be changed significantly.

To provide maximum utilization of the core region for driver fuel and irradiation tests, it is desirable to use as few in-core control rod positions as possible. As a minimum, three in-core positions are required for the primary safety system. Most if not all of the worth required for the shim-secondary safety system can be provided by reflector control rods. Typical full reflector bank control worths calculated for unzoned PuO_2 -SS cermet and PuO_2 - UO_2 fueled cores of varying size are tabulated below.⁽⁵⁾ (50/B₄C rods)

<u>Core Volume, liters</u>	<u>UO_2-PuO_2, % $\Delta\rho$</u>	<u>PuO_2-SS, % $\Delta\rho$</u>
400	9.38	14.7
600	7.19	12.6
800	5.79	10.9
1000	4.77	9.64

Although comparable calculations for PuC-UC fuel are not presently available, it appears probable that sufficient worth would be available with a nearly-complete reflector bank and possible enriched B₄C to meet the control requirements of 6 to 7% $\Delta\rho$ (depending on burnup and uncertainty allowances) plus the highest-worth rod. For this reason, the core arrangements to be studied will have three in-core safety rod positions, no in-core shim rods, and nearly full reflector banks.

Both the annular cone and the annular skew dispersed arrangements have one or more lattice positions in the central 7-cluster that cannot be used for full-size tests or drivers because of interference with the three vertical closed loops. However, it will be possible to convert the reactor cover, and either convert or replace the inlet plenum tube-sheet, to accommodate a "true skew" or "true cone" array for more compact cores, assuming that previous experiments in inclined test positions show that it is not essential to retain more than one plumb

vertical core position.⁽¹⁾ Conversion of the interfering vertical closed loops to fit the regular lattice geometry is especially important for the skewed core. The consequences for the cone are not as great, but for consistency it will be assumed that both cone and skewed cores would be converted to regular geometry when an advanced core is loaded. A typical core map for a 61-position, regular geometry core is shown in Figure 1, applicable to either cone or skewed array. Core position allocations are as follows:

Total number core positions	61
In-core safety rods	3
Closed loops	5
Open loops	8
Driver fuel	45
Number of "equivalent" fueled positions, counting open loops at 75%	51
Total first row reflector positions	30
Reflector control rods	24
Reflector elements or open tests, in first row reflector	6

C. CORE DESIGN LIMITS

1. Peaking Factors

Nuclear and engineering peaking factors appropriate to the split cone design were tabulated in the System 35 CSDD, A-0035-R. Revised radial nuclear factors were calculated for the annular cone design, although time did not permit optimization of the driver zoning ratio and the thermal analysis is complicated by the use of reflector shim control assumed for this design and the accompanying changes in radial power shape during a cycle. It is expected that overall radial factors for advanced cores will also be relatively higher. This is particularly true of the ratio of the hot channel coolant ΔT to the hot subassembly bulk coolant ΔT , which would be expected to increase for smaller cores with fewer subassemblies. The values assumed for various peaking factors are shown in Table I, along with comparable values for the split cone.

FIGURE 1. Typical 61-Position Core Map

TABLE I. Peaking Factors

<u>Factor</u>	<u>Value Assumed for Carbide Core</u>	<u>Split Cone Value</u>	<u>Reference for Split Cone</u>
Local Radial Nuclear Factor	1.40	1.30	
Local Axial Nuclear Factor	1.24	1.24	a
Local Engineering Factor on Fuel Cl Temperature	1.14	1.14	
Engineering Factor on Flow Maldistribution	1.10	1.20	a
Hot Subassembly Bulk ΔT Core Avg. ΔT (Unorificed)	1.16	1.14	b
Hot Channel Coolant ΔT Hot Subassembly Bulk ΔT	1.20	1.16	c

Table References:

- a. System 35 CCDD, A-0035-R, Appendix D
- b. FFTF Monthly Progress Report for November 1967, BNWL 638, p. 3.20
- c. FFTF Monthly Progress Report for December 1967, BNWL 659, p. 3.28

2. Thermal-Hydraulic Limitations

The following plant and system limits are assumed, consistent with current system designs:

Reactor coolant ΔT	300 min/300 max
Reactor $\Delta \rho$	100 to 120 psi
Power level, drivers plus open loop tests	400 MW

3. Safety-Related Limitations

The maximum acceptable subassembly worth and the minimum acceptable Doppler coefficient can strongly influence the core size and the configuration. These design limitations are related to the safety criteria of containment of a DBA and limitation of the consequences of a single fuel element meltdown. Design bases developed for the initial oxide cores are:⁽⁶⁾

Number of driver subassemblies 70

Doppler constant, T_{dT}^{dk} -0.004 (sodium in)
-0.002 (voided)

For advanced carbide cores, the use of a 61-position core arrangement presumes that the subassembly worth limitation may be relaxed. This can be qualitatively justified on the basis of improved safety system response speed and the probably greater transient over-power capability of carbide fuel when designed for maximum temperatures (3600 °F) in normal operation. A quantitative check on this assumption should be made.

In regard to the Doppler coefficient for small carbide cores, the minimum values of $T_{dT}^{dk} = -0.002/-0.004$ developed for 1000 liter oxidized cores will be used. The exact relation of DBA energy release to Doppler constant need not be the same for small carbide as for large oxide cores, but the oxide criteria will be taken as typical pending a DBA analysis for advanced cores. Calculations of T_{dT}^{dk} reported in Reference 7 for various core sizes and compositions are shown in Table II.

TABLE II. Calculations of Doppler Coefficients

Core Composition	Volume, liters	$^{238}\text{U}/^{239}\text{Pu}$	^{239}Pu Loading, kg	Doppler T_{dT}^{dk}	
				Na/in.	Voided
UC-PuC	600	5.24	375	-0.004	-0.0021
UC-PuC-BeO*	600	4.48	355	-0.0057	-0.0035
UC-PuC	800	5.67	468	-0.0047	-0.0026
UC-PuC-BeO*	800	4.87	443	-0.0066	-0.0042
UO ₂ -PuO ₂	600	4.05	349	-0.0037	-0.0017
UO ₂ -PuO ₂	800	4.42	443	-0.0044	-0.0022

* BeO is assumed to occupy 5% of the core.

A Doppler constant of -0.004 requires a higher $^{238}\text{U}/^{239}\text{Pu}$ ratio in (unmoderated) carbide cores than in oxide cores of equal size. This is difficult to achieve in a 600 liter core because of the limited fuel volume available. For this reason, use of BeO or other moderator is

likely in a carbide core if a Doppler constant of -0.004 is needed. From these calculated values, the minimum $^{238}\text{U}/^{239}\text{Pu}$ ratios needed to provide a Doppler constant $T \frac{dk}{dT} - 0.004$ in a 600 l. carbide core are estimated as 5.2 for unmoderated cores and about 3.1 with 5% BeO.

D. FUEL DESIGN LIMITS

The typical carbide fuel performance parameters developed in Reference 2 will be used as a guide to conceptual replacement core designs. In Particular, a peak operating temperature of 3600 °F or a linear heat rate of about 40 kW/ft will be adopted, and a goal smeared density of 80% TD (either packed powder or low density pellets) to allow burnups to 100,000 MWd/MT peak. It will be assumed that the smeared density may be increased to 90% TD for initial core loadings if this is necessary to provide sufficient ^{238}U to meet the Doppler coefficient requirement.

III. CARBIDE CORE PARAMETRIC SURVEY

A. FUEL LENGTH AND NUMBER OF PINS

The peak linear power of 40 kW/ft recommended in Reference 2 for gas bonded carbide fuel pins can be used to determine acceptable ranges of fuel length and numbers of fuel pins; peak fuel temperatures can then be calculated to check out selected designs. Based on 400 MW system power divided among 51 equivalent power generating subassemblies, a heat deposition in the fuel of 91%, and the peaking factors of Table II (1.98 overall, including engineering factors) the following peak operating linear heat rates are calculated:

<u>Number pins per S/A</u>	<u>Fuel Length</u>	
	<u>32 in.</u>	<u>38 in.</u>
127	42	35.2 kW/ft
169	31.4	26.4
217	24.4	20.6

Within the range of fuel lengths, the 169-pin design is preferred for the 61-position cores, for the following reasons:

- (1) It permits reduction to the minimum fuel length (32 in.) judged to provide sufficiently prototypical irradiation facilities, without exceeding a linear heat rate of 40 kW/ft.
- (2) It provides some margin, even at 32 in. fuel length, for the use of BeO if this should be necessary to obtain a satisfactory Doppler coefficient.
- (3) The use of 127-pin bundles in a 61-position core would require longer fuel and consequently greater core volume and lower specific power if linear heat rates are limited to 40 kW/ft. However, 127-pin bundles would be appropriate for higher performance fuel or for core arrangements using 70 total positions. These alternatives will not be explored in this study.

B. FUEL DESIGN INTERRELATIONS

1. Pin, Spacer Wire, and Duct Dimensions

The following overall dimensions will be used, consistent with the current dispersed core arrangements:

Minimum Core Lattice	4.450 in.
Flow Duct OD	4.420 in. across flats
Flow Duct ID	4.120 in. across flats
Nominal clearance around fuel pin bundle	0.035 in. diametral
Fuel pin bundle enclosing dimension	4.085 in. across flats

The relation between fuel pin OD and spacer wire diameter to maintain a constant fuel pin bundle dimension of 4.085 in. for 169-pin bundles is shown on Figure 2, along with the flow area per subassembly which provides an indication of the relative coolant heat removal capability.

2. Coolant Heat Removal Capacity

The limiting heat removal capability for the hottest subassembly is determined by the mean coolant flow velocity, flow area, and allowable bulk coolant temperature rise. The flow velocity may be limited either by excessive pressure drop or by other effects of high velocity flow such as vibration and mass transfer. Heat removal estimates made

FIGURE 2. Spacer Diameter and Net Flow Area
Versus Pin OD 169-Pin Bundle, 4.120 in. Duct ID

for a maximum fuel pin bundle ΔP of 70 psi and a maximum mean flow velocity of 40 ft/sec are plotted on Figures 3a and 3b for bulk coolant ΔT 's of 350, 400 and 450 °F. The ΔP 's are based on a total pin bundle length of 102 in. excluding the lower reflector, which allows about 5 ft for fission gas plenums. For calculational convenience, the total pin bundle ΔP is taken to be equivalent to the friction ΔP for an "equivalent length," within a straight hex duct. This length is taken as 57 in. for the cone and 73 in. for the skew. A friction factor of $f = 0.20 (\text{Re})^{-0.2}$ was used. Peak subassembly power levels are based on 95% of the total fission heat in the hottest subassembly removed by the active coolant within the duct. A peak subassembly power rating of about 9 MW is desired for the 61-position core based on a 1.16 hot subassembly to-average subassembly peaking factor and a 1.10 factor for flow distribution among subassemblies.

3. Fissile Loading and $^{238}\text{U}/^{239}\text{Pu}$ Ratio

If three or four tests are clustered near the core center and the remaining tests and in-core safety rods dispersed throughout the remaining core volume, as suggested in the core map (Figure 1), then the core can be approximately represented as a central circular test zone of low fuel fraction, surrounded by a driver annulus of high fuel fraction. Survey (1-D) nuclear calculations for mixed oxide-fueled cores of this geometry⁽⁸⁾ yield ^{239}Pu inventories as a function of test zone and driver zone volumes as shown in Figure 4, for a smeared PuO_2 concentration of 2% in the central zone. These results can be compared with a more detailed calculation⁽⁹⁾ of a 1010 liter core layout in 2-D with zoned driver, which yielded a ^{239}Pu loading about 4% higher than the values of Figure 4 with a smeared PuO_2 fraction of about 3.25% over the central seven lattice positions. Because the center core position in advanced cores with slightly inclined closed loops, is potentially available the initial advanced cores will be assumed to correspond to a 4% PuO_2 fraction. This is expected to reduce the correction factor that is applied to the loadings of Figure 4 to about 1.03, for mixed oxide cores. The

FIGURE 3a. Subassembly Power Capability and $^{238}\text{U}/^{239}\text{Pu}$ Ratio for 169-Pin Bundle, 4.420 in. Duct OD Effective Hydraulic Length = 57 in. (Cone Geometry)

FIGURE 3b. Subassembly Power Capability and $^{238}\text{U}/^{239}\text{Pu}$ Ratio for 169-Pin Bundle, 4.420 in. Duct OD Effective Hydraulic Length = 73 in. (Skew Geometry)

loadings predicted for oxide cores by Figure 4 were converted to equivalent carbide core loadings by using the ratio of ^{239}Pu inventories for 600 liter carbide and oxide cores of Table II; this routine was used for both moderated and unmoderated cores.

FIGURE 4. Fissile Loading for Annular $\text{PuO}_2\text{-UO}_2$ Cores, 2 vol% PuO_2 in Central Test Zone

The resulting ^{239}Pu loadings estimated for skewed and conical 61-position carbide cores are shown in Table III.

TABLE III. Estimated Loadings for 61-Position Core with 4.420 in. OD Subassemblies

	<u>Cone</u>	<u>Skew</u>
Smeared PuC fraction in test zone (central 7-cluster)	4.0%	4.0%
Core average $^{238}\text{U}/^{239}\text{Pu}$ ratio	3 to 4	3 to 4
Test region volume (32 in. fuel)	72 liter	66 liter
Driver region volume (32 in. fuel)	555 liter	510 liter
Estimated ^{239}Pu loading, no BeO	400 kg	355 kg
Estimated ^{239}Pu loading, 5% dist. BeO	380 kg	335 kg
Assumed ^{239}Pu loading of closed loop tests (total)	6 kg	6 kg
Estimated total ^{239}Pu loading of drivers plus open loops (no BeO)	394 kg	349 kg
Estimated total ^{239}Pu loading of drivers plus open loops (5% BeO)	374 kg	329 kg

The average driver $^{238}\text{U}/^{239}\text{Pu}$ ratio for the unmoderated cone and skew arrangements is also shown on Figures 3a and 3b, based on driver plus open loop ^{239}Pu loadings of 394 and 349 kg. The total U + Pu loadings from which the $^{238}\text{U}/^{239}\text{Pu}$ ratios are calculated are based on a theoretical density of 13.6 g/cm^3 , 80% smeared density, and cladding thickness determined by

$$t_{\text{cladding}} = 0.005 + 0.010 \frac{\text{fuel pin OD}}{0.235} \text{ inch.}$$

The Pu composition is 12% ^{240}Pu , and the U is natural.

4. Selection of Pin Size and Coolant ΔT

The $^{238}\text{U}/^{239}\text{Pu}$ ratios can be estimated from Figures 3a and 3b for 61-position cores having hot subassembly coolant ΔT 's of 350, 400 and 450 °F.

<u>Hot Subassembly Bulk Coolant ΔT</u>	<u>$^{238}\text{U}/^{239}\text{Pu}$ Ratio</u>	
	<u>Cone</u>	<u>Skew</u>
350 °F	3.37	3.53
400 °F	3.68	3.92
450 °F	3.91	4.23

Increasing the ΔT from 350 to 400 °F permits about a 10% increase in $^{238}\text{U}/^{239}\text{Pu}$ ratio (or Doppler coefficient), while a further increase to 450 °F permits about a 7% additional increase in Doppler coefficient. Limiting the hot subassembly coolant ΔT to 400 °F in initial carbide replacement cores has the advantage of allowing orificing of flow ducts to produce a uniform outlet temperature without exceeding the 400 °F maximum reactor ΔT , if uniform duct outlet temperatures should prove necessary to reduce thermal bowing problems. Alternatively, if non-uniform temperatures are acceptable, designing the initial carbide cores for 400 °F peak duct ΔT at 400 MW permits a later increase in reactor power to about 450 MW without exceeding the 400 °F reactor ΔT limitation. For either reason, it seems desirable to develop the initial carbide core conceptual design on the basis of a 400 °F maximum subassembly bulk ΔT .

The selection of a 400 °F maximum coolant ΔT results in fuel pin OD's of about 0.260 and 0.250 in. for the cone and skew arrangements, with $^{238}\text{U}/^{239}\text{Pu}$ ratios of 3.7 and 3.85, respectively. If the fuel smeared density is increased to 90% TD in order to obtain increased ^{238}U loading, the $^{238}\text{U}/^{239}\text{Pu}$ ratios are increased to about 4.3 and 4.5 for the cone and skew, respectively. Since these are substantially below the value of 5.2 estimated in Section II.C to be required to obtain a Doppler constant of $T \frac{dk}{dT} - 0.004$ in an unmoderated core, it appears to be necessary to use BeO (or other moderator) for Doppler augmentation.

C. USE OF BeO IN COMPACT CARBIDE CORES

Since a Doppler constant of about $T \frac{dk}{dT} - 0.004$ is required by present safety criteria (developed, however, for large oxide fueled cores) it is necessary to include in the conceptual design of at least the initial replacement carbide cores the capability for attaining this Doppler

coefficient by the use of BeO or other moderator. In addition to the calculations of Table II, several nuclear calculations have been made of the use of lumped or distributed BeO in large oxide cores including effects on k_{eff} , Doppler, and local power peaking.⁽¹⁰⁾ The BeO locations considered were the radial reflector, individual pins within fuel elements, and homogenous mixture with fuel. Since the radial reflector positions are expected to be used for control rods and tests in the advanced compact cores, and in any case the use of BeO in the radial reflector results in only a relatively small improvement in Doppler with an accompanying local power peak,⁽¹⁰⁾ it is assumed that the BeO would be located in individual pins within the driver fuel subassemblies, which replace a small fraction of the regular fuel pins. Consequences of exchanging active fuel pins for BeO pins in the 169-pin bundles are estimated in Figure 5 for both the cone and skew arrangements, under the following assumptions.

- (1) Only driver fuel elements have BeO pins.
- (2) Effects of U/Pu ratio and v/o BeO on Doppler and fissile inventory are interpolated linearly from the data of Table II for 600 liter carbide cores.
- (3) The BeO pins have 0.010 in. cladding and 90% smeared density BeO.
- (4) Local power peaking due to lumped BeO in sizes characteristic of fuel pins is negligible for a single pin.⁽¹¹⁾ It will be assumed that the number of BeO pins needed can be distributed through the subassembly in such a manner that there is no significant increase in the radial peaking factor.

The intersections on Figure 5 show that a Doppler constant of about $T \frac{dk}{dT} = -0.004$ can be obtained in the cone and skew arrangements by using 33 and 29 BeO pins per 169-pin driver subassembly, respectively. Because of the approximate nature of the calculation, a round number of 30 BeO pins per subassembly will be used for core performance estimates, for both cone and skew arrangements. The loss of these 30 fuel pins increases the peak linear heat rate to about 37 kW/ft.

FIGURE 5. $^{238}\text{U}/^{239}\text{Pu}$ Ratio and Linear Heat Rate
Versus Number of BeO Pins per Subassembly

IV. TYPICAL CARBIDE CORE PERFORMANCE

A. SUMMARY OF DESIGN CHARACTERISTICS

Design and performance characteristics for the typical carbide core designs developed in Sections II and III are summarized in Table IV.

TABLE IV. Typical Design Characteristic's

<u>Core Dimensions and Arrangement</u>	<u>Cone</u>	<u>Skew</u>
Total number of lattice positions	61	61
Lattice inclination, per row	0.9 degree	2 degree
Mean lattice pitch	4.75 in.	4.55 in.
Total core volume	627 liters	576 liters
Number of driver subassemblies	45	45
Number of open loop tests	8	8
Number of closed loops	5	5
Number of in-core control rods	3	3
Number of first-row reflector positions	30	30
Number of reflector control rods	24	24
Number of first-row reflector tests	6	6
Operating power level, exclusive of closed loops	400 MW	400 MW
<u>Driver Subassembly Design</u>	<u>Cone</u>	<u>Skew</u>
Flow duct outside dimension	4.420 in.	4.420 in.
Flow duct inside dimension	4.120 in.	4.120 in.
Cold nominal clearance around fuel pin bundle	0.035 in.	0.035 in.
Fuel pin OD	0.260 in.	0.250 in.
Fuel pin ID	0.228 in.	0.219 in.
Spacing between fuel pins	0.048 in.	0.057 in.
Net flow area	5.4	6.0 in. ²

TABLE IV. Typical Design Characteristic's (contd)

<u>Driver subassembly Design</u>	<u>Cone</u>	<u>Skew</u>
Total number pins per bundle	169	169
Pin spacing system	(Spiral wire wrap)	
Number of fuel pins per bundle	139	139
Number of BeO pins per bundle	30	30
Fuel length	32 in.	32 in.
Gas plenum length	60 in.	60 in.
Overall fuel pin length, exclusive of lower axial reflector	102 in.	102 in.
Effective hydraulic length for $\Delta\rho$ calculations	57 in.	73 in.
<u>Driver Fuel Thermal- Hydraulic Performance</u>	<u>Cone</u>	<u>Skew</u>
Mean flow velocity in hex duct	36.6 ft/sec	34.1 ft/sec
Pin bundle $\Delta\rho$	~70 psi	~70 psi
Coolant ΔT in hottest subassembly, with 1.1 plenum factor on flow	400 °F	400 °F
Nominal coolant ΔT in hottest subassembly	365 °F	365 °F
Reactor ΔT (orificed)	365 °F	365 °F
Reactor outlet temperature	1000 °F	1000 °F
Reactor inlet temperature	635 °F	635 °F
Hot subassembly bulk outlet temperature (with plenum factor)	1035 °F	1035 °F
Hot subchannel coolant outlet temperature	1115 °F	1115 °F
Heat flux at hot end of pin	1.4×10^6	1.5×10^6 Btu/hr-ft ²
Assumed film coefficient	30,000	30,000 Btu/hr-ft ² -°F
Peak clad temperature (at OD)	1115 °F	1165 °F

TABLE IV. Typical Design Characteristic's (contd)

Driver Fuel Thermal-Hydraulic Performance	Cone	Skew
Peak clad temperature (at ID)	1310 °F	1325 °F
Average linear heat rate	18.7	18.7 kW/ft
Peak linear heat rate	37.0	37.0 kW/ft
Assumed coolant ΔT to axial mid-point of hottest coolant subchannel	240 °F	240 °F
Peak heat flux at hot spot, at cladding OD	1.78×10^6	1.85×10^6 Btu/hr-ft ²
ΔT from coolant to cladding ID at hot spot	220 °F	230 °F
Assumed pellet-to-cladding coefficient	2000	2000 Btu/hr-ft ² -°F
Pellet-to-clad ID ΔT	1000 °F	1050 °F
Assumed fuel conductivity characteristic	-28 °F/kW-ft	-----
ΔT from pellet edge to &	1040 °F	1040 °F
Peak fuel & temperature	3135 °F	3195 °F
Nuclear Parameters	Cone	Skew
BeO volume fraction in core	4.6 vol%	4.6 vol%
Est. ²³⁹ Pu loading without BeO	400 kg	355 kg
Average specific power without BeO	1.0 MW/kg fissile	1.13 MW/kg fissile
Est. loading with BeO	382 kg	339 kg
Average specific power with BeO	1.05 MW/kg fissile	1.18 MW/kg fissile
Est. total flux at center of core, (with or without BeO)	1.0×10^{16}	1.1×10^{16} n/cm ² -sec

B. PRINCIPAL CORE DESIGN PROBLEMS AND UNCERTAINTIES

1. Safety Design Constraints for Carbide Cores

Safety requirements developed for the initial oxide cores concerning maximum acceptable subassembly worth and minimum acceptable

Doppler coefficient have a significant impact on advanced core designs, and on the design of the permanent facilities which must eventually accommodate the advanced cores. It appears that Doppler coefficients (sodium in) up to about $T \frac{dk}{dT} = -0.004$ or -0.005 can be obtained by the use of BeO in a compact carbide core, at the cost of additional complexity in fuel subassembly design and manufacture. However, the subassembly worth for advanced cores is unavoidably higher than for the initial oxide cores, and the designs proposed in this study do not meet the present requirement for oxide cores.⁽⁶⁾ It is therefore necessary to develop analogous safety criteria for carbide fuel in order to determine whether a core lattice of ~ 4.5 in. is adequate for advanced cores, and to establish the minimum Doppler coefficient needed for limiting the calculated DBA energy release and to ensure safe power operation.

2. Use of BeO Pins in Fuel Subassemblies

If BeO pins are needed within the driver subassemblies, as suggested in this study, some design and analytical effort will be needed to determine the best location of these nongenerating pins, and the thermal and nuclear consequences. The addition of moderators to either the axial reflectors or insulating pellets at the ends of the fuel stack might also be investigated as an additional means to augment the Doppler coefficient.

3. Pressure Drop in Gas Plenum Regions

Because of the assumption made in this study of long (5-ft) fission gas plenums, the capability of enlarging the flow duct cross section from a hex to a full round shortly above the active fuel is important in determining the power density that can be achieved in the core. The $\Delta\rho$ calculations on which the choice of fuel pin size and the estimate of $^{238}\text{U}/^{239}\text{Pu}$ ratio are based, are sensitive to the pressure drop in the gas plenum region. Consequently, for core designs utilizing long gas plenums, the duct contours permitted by the cone and skew geometries should be determined, and a better estimate made of the pressure drop for the portion of the fuel pin bundle within the variable-area duct.

V. IMPACT OF ADVANCED CORE ON REACTOR AND PLANT DESIGN

A. LOCATION OF DRIVER FUEL IN CENTRAL SEVEN POSITIONS

For the annular skew arrangement with three vertical closed loops, the low fuel fraction in the center "test zone" causes a penalty of probably 10 to 20% on peak flux⁽⁸⁾ and a further penalty on $^{238}\text{U}/^{239}\text{Pu}$ ratio (higher ^{239}Pu loading). This can be remedied either through the conversion of the three vertical loops to fit the skew geometry (at an inclination of 2 degrees) or by using special undersize subassemblies to eliminate interferences while retaining the three plumb vertical loops. In order to preserve the option of increasing the fuel fraction near the core center for advanced cores without requiring special subassemblies, it is desirable to utilize the replaceable insert in the reactor cover and inlet plenum upper tubesheet, if possible, to permit conversion of the vertical loops to skewed geometry.

A similar conversion can be made for the annular cone arrangement; however the incentive is not as great (the conical arrangement allows three drivers in the test zone along with the vertical loops) and the ligament available at the cover and tubesheet for securing the insert is also severely restricted, so that the replaceable test zone insert design is not judged as desirable in the cone as in the skew.

B. INTERCHANGEABILITY OF CORE COMPONENTS

To achieve a core volume of ~600 liters in future cores with a fuel length adequate for testing needs, it is essential to be able to replace the outside ring of core subassemblies in the 91-position oxide core designs with reflector elements - i. e., reflector control rods, reflectors, or reflector test positions. It is also desirable to be able to relocate in-core control and test positions to more optimum locations for compact cores. If the initial 91-position cores are orificed, it is preferable that the orifices not be permanently installed in the inlet plenum since a different orificing pattern would be required for the 61-position core.

C. CLOSED LOOP AND CORE INSTRUMENTATION ACCESS RELiance ON REFLECTOR SHIM CONTROL

Reliance on reflector in advanced cores, as has been assumed in this study, leaves little room between control rod drives for closed loop piping and instrument lead access. Consideration should be given to this possible need for access to a 61-position core through a nearly complete ring of control rod drives when selecting conceptual arrangements for closed loop piping and instrument lead routing and locating the external connections and instrument trays, so that a minimum of rearrangement would be needed to adapt the facility to an advanced core. Potential cores should be compared with regard to potential closed loop locations so that common closed loop facilities can be used if possible.

D. HEAT TRANSPORT SYSTEM CAPABILITIES

It was found possible to develop conceptual carbide cores in the dispersed arrangements meeting the goal specific power of one megawatt per fissile kilogram without exceeding the 400 MW system rating. Therefore, the HTS sizing appears adequate for near-term advanced cores. Stretch capabilities up to ~500 MW would be useful however to permit further improvements in core flux and to provide the alternative of using a larger core (e.g. 73-positions) for greater testing volume with the same specific power.

E. REACTOR LATTICE DIMENSION

The core lattice and subassembly size for future cores that use the same reactor cover and inlet plenum are fixed by the choice of dimensions made for the first oxide core. If core designs with ~45 driver elements are acceptable, then it appears that future testing goals can be met with a subassembly size of ~4.5 in., as is now being used in the initial oxide core designs for the 91-position dispersed core arrangements.

F. IRRADIATED FUEL HANDLING AND STORAGE

The increase in operating power density necessary to meet future flux goals results in a higher decay heat load, which must be

accommodated by the irradiated fuel handling and storage equipment. Radiation sources used in shielding design should also take into account future high power density operation.

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