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FFTF CLOSED LOOP TEST CHARACTERISTICS

R. E. Keyes

June 1970

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FFTF CLOSED LOOP TEST CHARACTERISTICS

by

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Reactor and Plant Technology Department
FFTF Division

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ABSTRACT

This document provides a set of curves which characterize the FFTF closed loop testing capability. Limiting parameters for candidate breeder-reactor fuels are discussed. Heat removal requirements of the ex-reactor loop system are set by the maximum of a variety of possible test designs. The study concludes that a 2 MW heat removal capacity is adequate for a maximum test diameter of 2.5 in.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

ERRATA

Page 5 - Line 13-14 should read:

A test diameter of at least 3 in. is necessary to accommodate the 37-pin bundle.

FFTF CLOSED-LOOP TEST CHARACTERISTICS

R. E. Keyes

INTRODUCTION

The Fast Flux Test Facility (FFTF) is a sodium-cooled 400 MW fast reactor. Closed-loop test sections in the reactor core each have separate sodium cooling systems. A test assembly inserted into an in-reactor test section can be irradiated and cooled in an environment prototypic of a fast breeder reactor core.

The test section is a tubular structure, approximately 35 ft long, penetrating the vessel cover and extending down through the fueled region of the core. The reference concept uses a reentrant flow path with test coolant flowing down an annulus and back up through a central test section. The test coolant system is coupled to the tube via a nozzle spool-piece located above the vessel cover. The in-reactor tube consists of a pressure tube and an inner flow-tube. The ID of the inner flow tube establishes the test diameter (TD).* (See Figure 1 for schematic arrangement.) Both tubes are double-walled with the intervening gaps providing heat resistance. Heat transfer to the reactor, and between the closed loop outlet and inlet above the core, is restricted and the thermal gradients across the tube and nozzle walls are within acceptable stress levels. Argon is used for the gap insulation above and below the core. At the core region the insulation is sintered stainless steel to promote decay heat transfer to the reactor coolant after shutdown.

The cross section available for inserting the test assembly into the closed loop tube was expected to be between 3.0 and 2.5 in. in diameter. The 2.5 in. diameter was established as a minimum requirement because it could accommodate all prospective candidate fuel pin sizes within a 19-pin bundle. A 19-pin bundle was considered a minimum size acceptable for simulating the thermal hydraulics of larger reactor fuel bundles.

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* *The term "test diameter" (TD) is, by definition, the ID of the flow tube. (See Figure 1) This diameter is equivalent to the maximal dimension limit of a test bundle.*

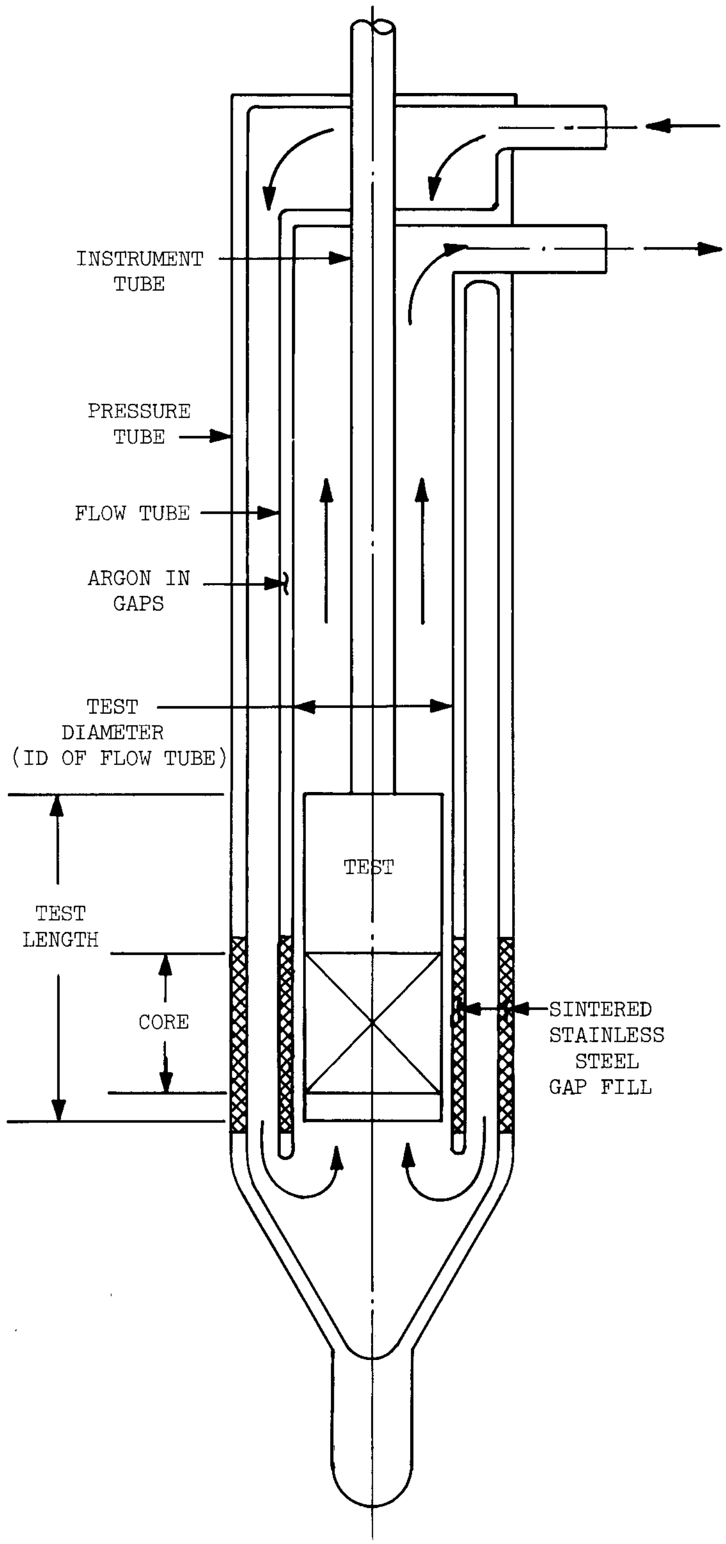

FIGURE 1. *Schematic of Closed Loop In-Reactor Tube*

The ex-reactor closed-loop system includes a radioactive sodium primary loop, a non-radioactive secondary loop, and an air blast heat exchanger, all constructed of austenitic stainless steel (type 304). Two 2 MW systems are to be installed initially with 4 MW thermal power as the highest design value of closed-loop systems for future installation.

This study was undertaken to determine acceptable test limits and characteristics of the FFTF closed-loop test facilities. There are bounds to test capability in characteristics of coolant velocity, heat flux through the cladding, linear power generation, and $^{238}U/^{239}Pu$ ratio of the fuel. These bounds are discussed in this report but are not rigidly quantified because of the paucity of applicable irradiation data.

The following assumptions were predicated for this study:

- A 72 in. long pin is used for the closed-loop test assembly.
- A 12 in. pitch is employed for spirally wrapped spacer wire.
- A 90 psi pressure drop for all test assemblies is reserved for system pressure drop calculations for the pump head required.
- 1.1 to 1.3 is the triangular pitch ratio range for all pin bundles in a hexagonal array.
- A radial clearance of 0.125 in. is allowed for shroud wall plus clearance between it and the test diameter; i.e., $TD-D_t = 0.125 \times 2 = 0.25$ in.*
- All pins in each assembly are fuel pins. Actual test assemblies will probably be instrumented, at least with flux wires and thermocouples, with some fuel pins thus being displaced.
- 36 in. is the length of fuel in the test specimen to conform to the core length, although shorter test fuel length need not be eliminated.
- Total power is the heat picked up by the test coolant, which is the product of coolant mass flow rate, specific heat, and the temperature

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* *See Appendix A for nomenclature.*

increase in its passage through the test assembly. It should be recognized that not all of this heat is generated in the fuel. Gamma heating in the hex shroud tube, the closed-loop tube walls, the wire wrap, and other areas also contribute heat to the coolant. For high-power tests the gamma heating from the above sources is minor; but the gamma heating is significant as test power becomes small. This is an important consideration when examining the significance of linear power and heat flux. This study uses total power, defined above, as the basis for deriving linear power and heat flux.

- A temperature of 1000°F is assumed as the average sodium condition because this temperature is in the range of broadest interest. The corresponding physical properties are 51.4 lb/ft^3 density, 0.555 lb/ft-hr viscosity, and 0.3006 Btu/lb-°F specific heat.
- Axial peaking factor = 1.24 is considered applicable to the initial core.
- The range of interest for TD's is 1.8 to 3.0 in.
- Bundle sizes of interest are 61, 37 and 19 pins hexagonally arrayed with a triangular pitch-to-diameter ratio ranging from 1.1 to 1.3.
- Maximum linear power as follows:
  - 20 kW/ft for Pu-U oxide fuels gas-bonded
  - 40 kW/ft for Pu-U carbide fuels gas-bonded
  - 55 kW/ft for Pu-U carbide fuels sodium-bonded
- A maximum coolant velocity of 50 ft/sec.
- A maximum heat flux through the cladding of 2.0×10^6 $Btu/hr\text{-}ft^2\text{-}°F$.
- The $^{238}U/^{239}Pu$ ratio, a measure of the fertile-to-fissile content of the fuel, affects the Doppler coefficient. The perturbation by a single test assembly can be ignored in the FFTF core. This ratio can be the experimenter's choice, probably determined by the desired power.

CONCLUSIONS

This study provides a set of envelopes of the FFTF closed-loop test characteristics for test diameters ranging from 1.8 to 3.0 in. The characteristics for other conditions can be derived by means of equations and other information in the appendices. From this study it is concluded that:

- A TD of 2.5 or 2.4 in. is adequate for testing candidate fuels[1] assuming a 19-pin bundle. A 2 MW rated coolant system is sufficient to dissipate the thermal power for a TD of 2.5 in.
- It is possible with the 2.5 or 2.4 in. TD and 2 MW power to perform tests at peak linear powers as high as 55 kW/ft for sodium-bonded carbide fuels.
- A future ex-reactor heat removal system capacity of 4 MW is unnecessary unless 37-pin bundles are required. A test diameter of at least 3 in. is unnecessary to accommodate the 37-pin bundle.

DISCUSSION

The closed loop test facilities are being designed for a wide variety of tests to suit the needs of researchers. The test capability to meet fuel test needs is characterized by figures contained in this report and some additional limiting considerations. These limiting considerations, as applicable to liquid metal cooled fast reactors, are discussed below. Table 1 is provided to specifically relate FFTF to LMFBR and future breeder reactors. Projected characteristics for 1000 MWe designs (as proposed in 1968) are listed in this table. A study of carbide fuels proposed eight "most useful" tests for potential FFTF carbide fuels. Data for these are listed in Table 2.

PIN DIAMETER

The size of the closed loop TD places a rigid restriction on the number of pins in a test bundle particularly since pins less than 0.2 in. diameter are not of interest in the LMFBR program.

TABLE 1. *Parameters from 1000 MWe Design[1] Studies of 1968*

Manufacturer Symbol	A	B	C	D	E
Fuel	oxide	oxide	oxide	carbide	carbide
Bond	gas	gas	gas	sodium	sodium
Cladding, SS Type	304	304/316	316	316	316
Cladding Thickness, in.	0.015	0.015	0.014	0.011	0.014
Temperature, °F					
Fuel Peak				2310	2313
Cladding Peak		1130		1169	1295
Coolant Out	1100	1060	1150	1090	1028
Coolant Rise	300	300	295	305	278
Core Height, ft	2.9	4.16	3.0	2.0	3.88
Fuel Pin, OD, in.	0.28	0.25	0.23	0.399	0.30
Triangular p/d	1.20	1.34[a]	1.20	1.18	1.24
Max. Vel. ft/sec		32.8	38.2	42	28.7
Peak Linear Power, kW/ft	17	15	18.1	43.5	30.6
Clad Heat Flux Btu/hr-ft^2 x 10^{-6}		0.78	1.02	1.36	1.33

(a) Estimated from parameters given

TABLE 2. *Parameter for Proposed Carbide[2] Fuel Tests*

Case (Run No.)*	1(2)	2(1)	3(1)	3(5)	4(3)	5(1)	6(2)	7(2)
Bond	sodium	sodium	sodium	sodium	helium	helium	helium	helium
Max. Center Line Temp. °F	1900	1800	2700	2700	3600	3370	3800	3770
Fuel Pin OD, in.	0.207	0.246	0.324	0.233	0.210	0.253	0.213	0.246
Linear Power, Peak, kW/ft	37.1	31.0	57.6	56.0	26.0	36.4	28.1	28.5
Clad Heat Flux Btu/hr-ft^{2}x10^{-6}	2.3	1.6	2.3	2.9	1.6	1.8	1.7	1.5

* This is the designation used in Reference 2.

COOLANT TEMPERATURE RISE

The maximum temperature rise of 400°F for the coolant through the test is an arbitrarily established requirement which may be dependent upon other actual limitations such as fuel, cladding, or coolant temperatures.

FLOW-DIMENSIONS

Flow in gal/min and ft/sec velocity are related to the test specimen dimensions (in Figures 2, 3 and 4) for hexagonal, triangular-pitch bundle arrangements for 61, 37, and 19 pins, respectively. Equations used for pressure drop and flow are given in Appendix B.

VELOCITY

The coolant velocity through the specimen is optional. However, it should be recognized that fretting corrosion and vibration-induced wear are potentially present in any multi-component fuel rod assemblies that are exposed to flowing coolant.[(3)*] Coolant flow also may produce test assembly oscilliatory motions which lead to galling, fatigue damage, and/or mechanical anomalies in the fuel assemblies.[(3)**] At the time of this report a firm limiting velocity has not been established. For reactor use a coolant velocity of 50 ft/sec is considered an upper limit,*** however, velocities of 40 ft/sec or greater require investigation. Accordingly, the researcher can judge from the velocity parameter, which tests may be difficult because of high velocity.

POWER

Power as represented by coolant heat removal, linear power (heat generation rate per unit length) and heat flux through the cladding (based on the outside area) are shown in Figures 5, 6 and 7 for 61-, 37- and 19-pin bundles for coolant temperature rises of 400, 300 and 200°F. The The figures are made by superimposing the power characteristics of the flow-dimensional characteristic envelopes of Figures 2, 3 and 4.

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* *See Page 19 of the reference*

** *See Page 21 of the reference*

*** *Private communication, J. M. Yatabe, BNWL, FFTF Division*

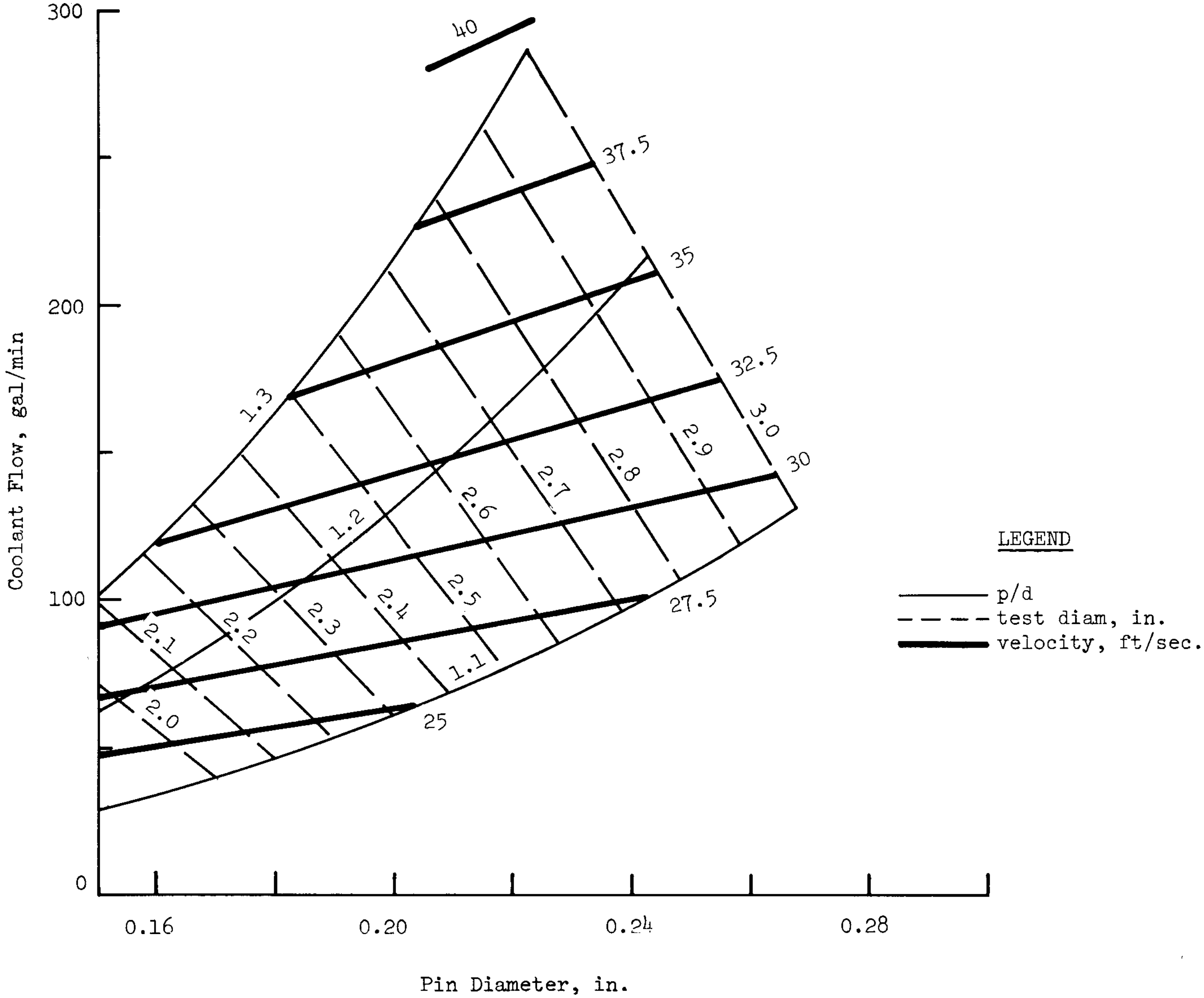

FIGURE 2. *Test Characteristics for 61-Pin Bundle Flow (ΔP = 90 psi)*

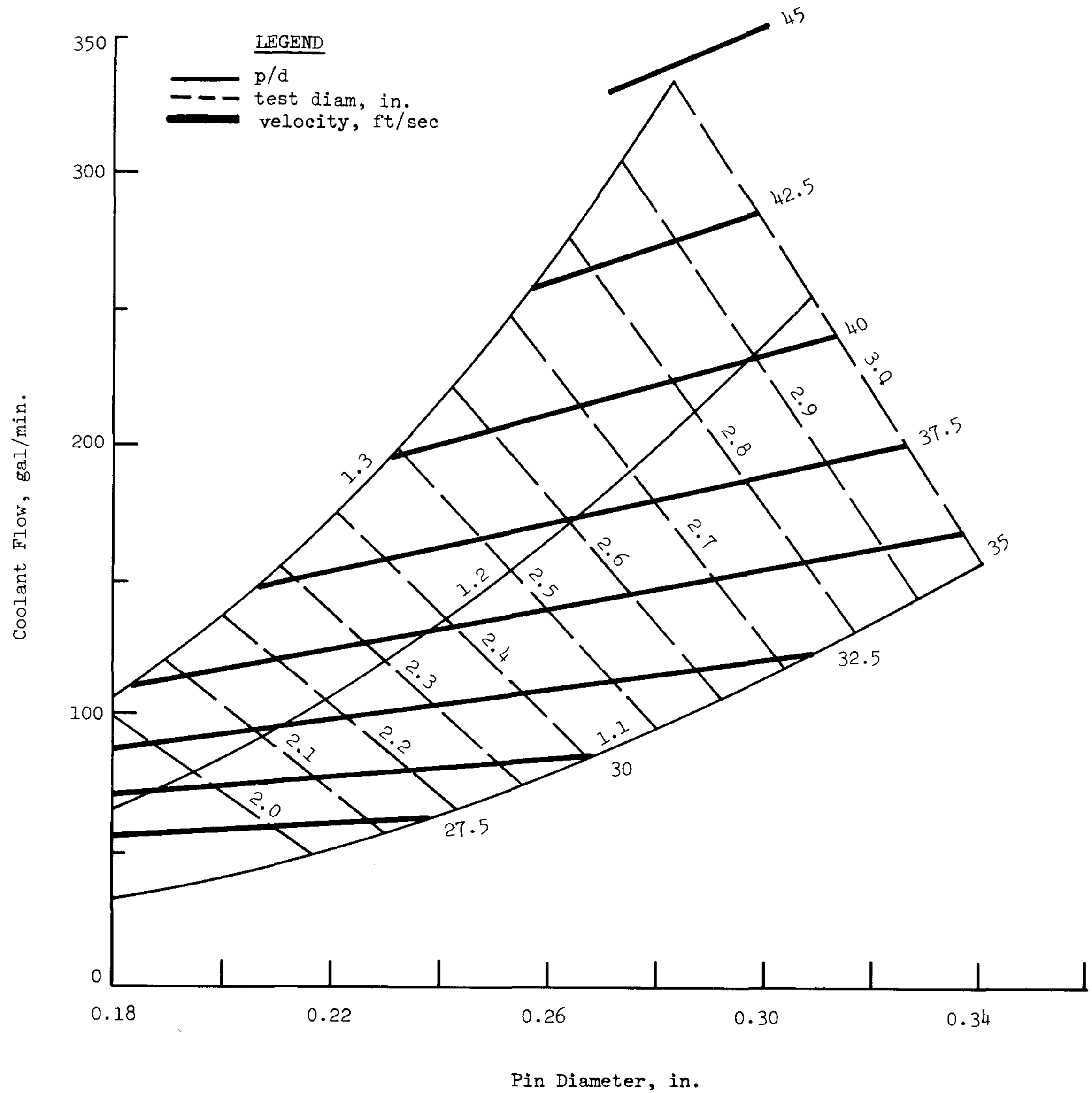

FIGURE 3. *Test Characteristics for 37-Pin Bundle Flow (ΔP = 90 psi)*

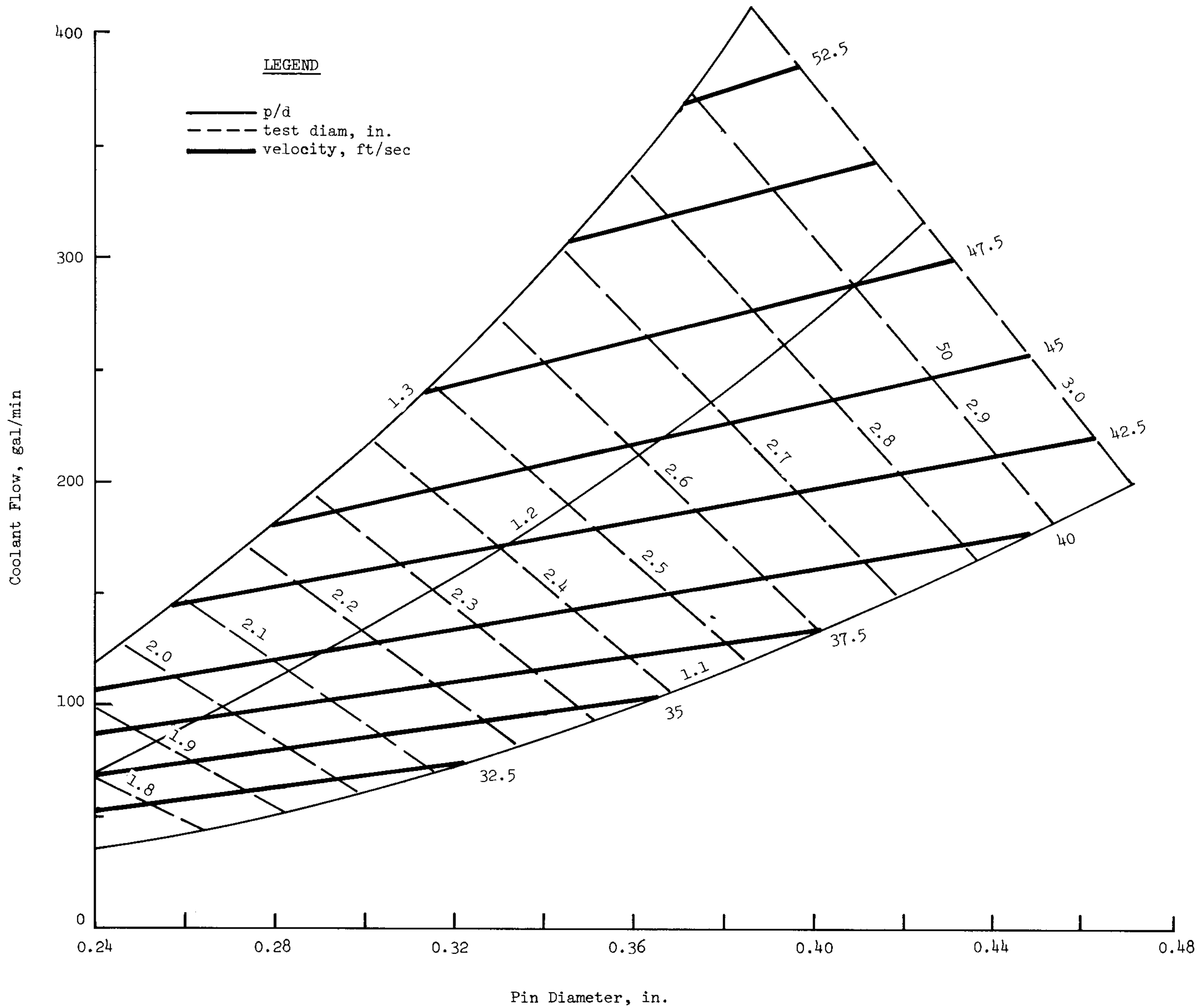

FIGURE 4. *Test Characteristics for 19-Pin Bundle Flow (ΔP = 90 psi)*

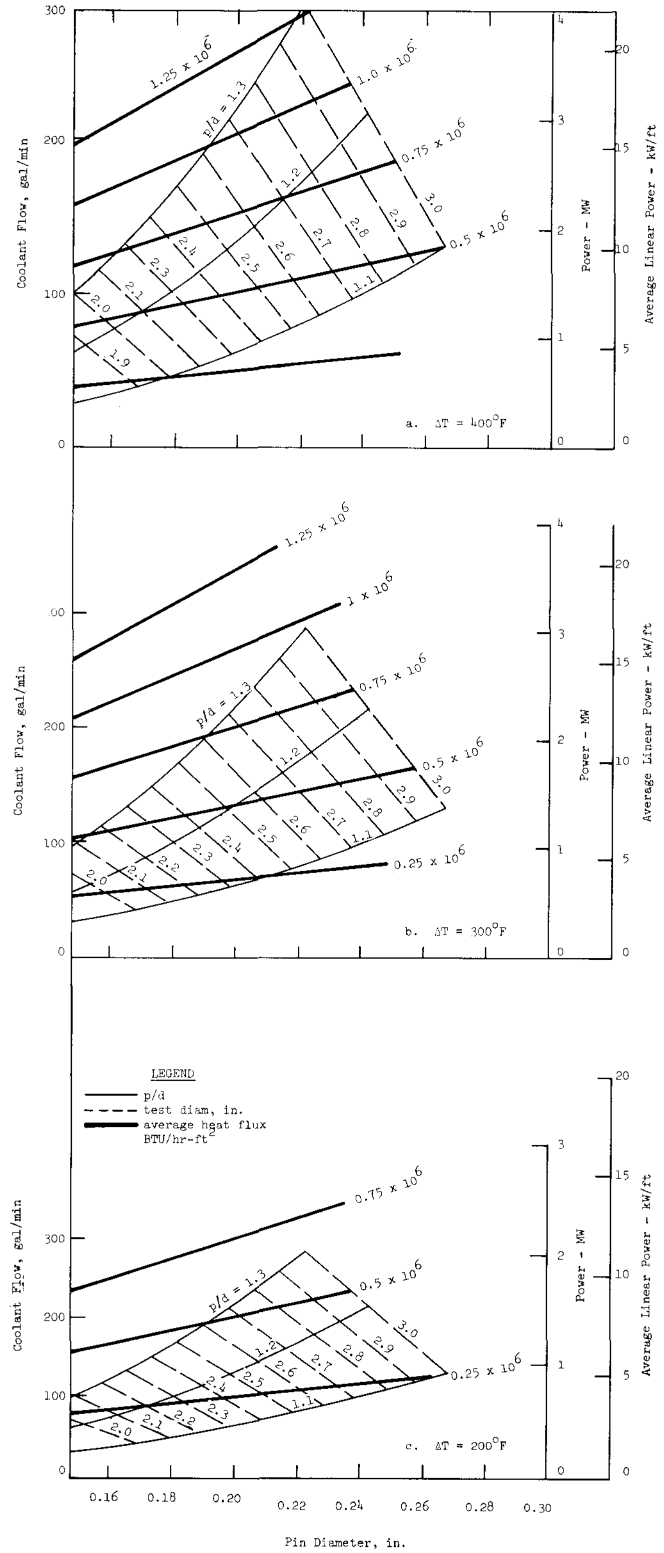

FIGURE 5. *Test Characteristics for 61-Pin Bundle (Heat Flux and Power)*

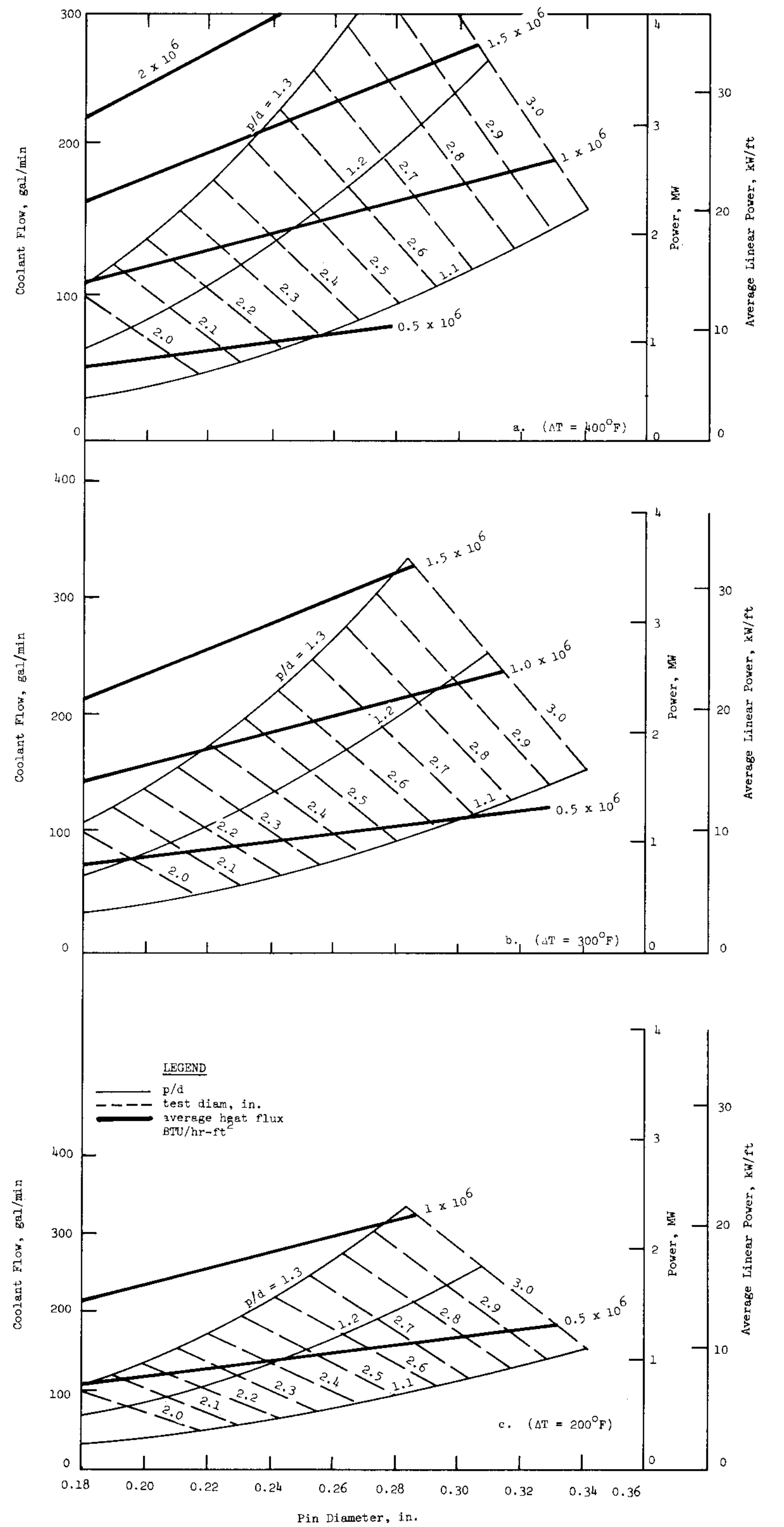

FIGURE 6. *Test Characteristics for 37-Pin Bundle (Heat Flux and Power)*

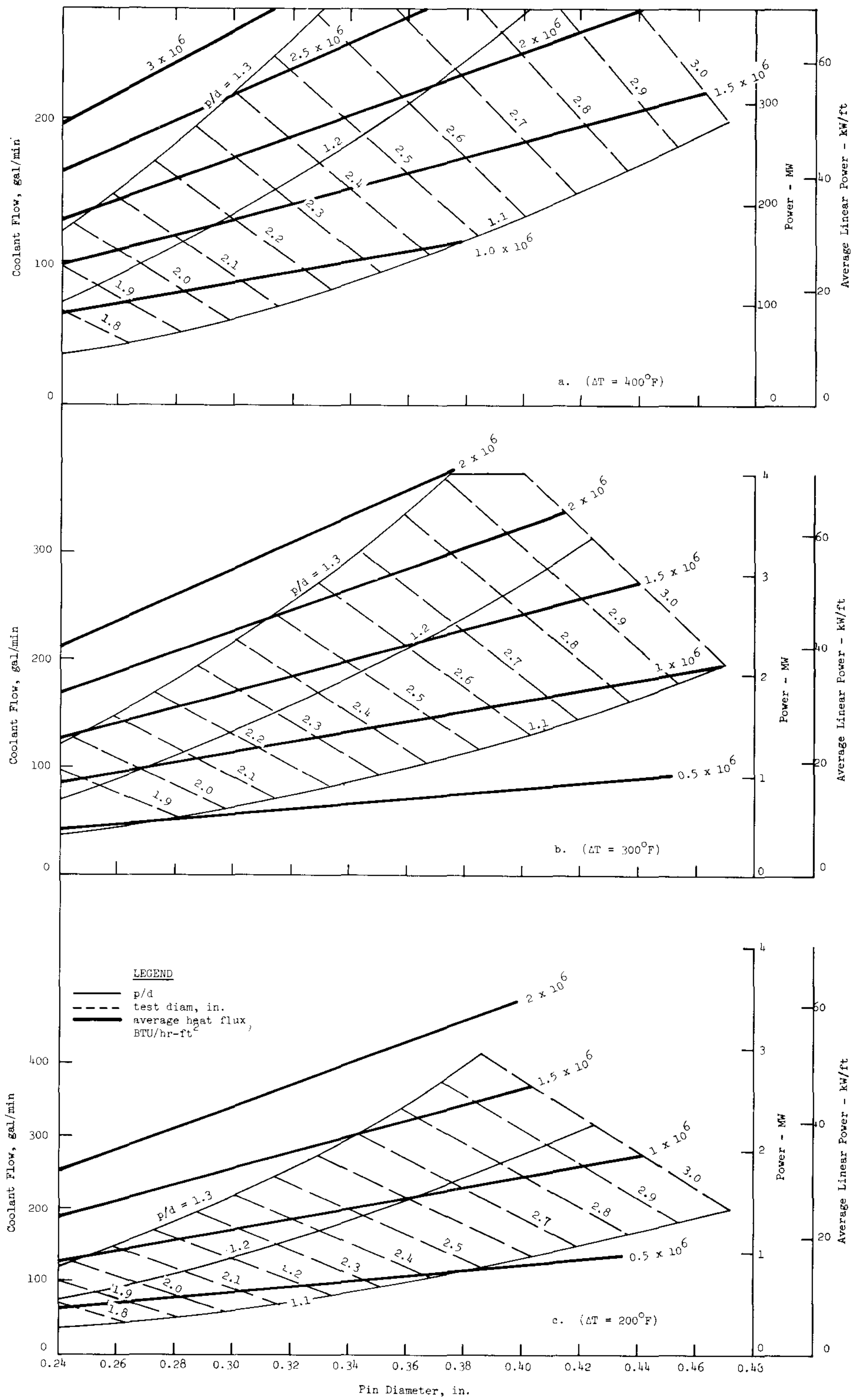

FIGURE 7. *Test Characteristics for 19-Pin Bundle (Heat Flux and Power)*

An important consideration is the effect of density on the flow and the power characteristic. This study was based on an average density of 51.4 lb/ft^3 at 1000°F for deriving the characteristics of closed-loop tests. Calculations for other sodium conditions and for other than 90 psi pressure drop can be made by using the equations and data of Appendix B.

HEAT FLUX

Heat flux through cladding is not a limit per se, but may be associated with limitations of cladding thermal stresses, burnout potential, and mass transfer.[(2)] There is extensive experience up to 1 x 10^6 Btu/hr-ft^2, limited experience in the range of 1.5 to 2.0 x 10^6 Btu/hr-ft^2, and no known experience at higher fluxes. The carbide fuel cases studied involved heat fluxes above 1 x 10^6 Btu/hr-ft^2.[(2)*] Another study [(3)**] predicts average heat fluxes of 0.7 to 1.0 x 10^6 Btu/hr-ft^2 and peak heat fluxes, including over-power conditions, of 1.4 x 10^6 to 2.0 x 10^6 Btu/hr-ft^2 for advanced cores including over-power conditions.

LINEAR POWER

The linear power rate becomes limiting in association with other limits. The limit usually associated is the temperature of the fuel or the cladding. Most experience has been with oxide fuels. One reason for carbide fuel in advanced cores is that it allows a lower temperature gradient than for oxide fuels. A typical representation of temperature profile for the different fuel pin schemes is shown in Figure 8. The steepness of the temperature gradient across the cladding is indicative of the linear power levels for the three fuel-pin schemes.

Appendix C discusses the relationship of linear power to temperature limits on fuel pins. The linear power scale in a characteristic curve is an average value over the assumed 3 ft test fuel length. A peak value is estimated to be 24% greater. Using this information, the experimenter can select the essential dimensions for a chosen linear power.

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* *See Pages 44-45 of the reference*

** *See Page 5 of the reference*

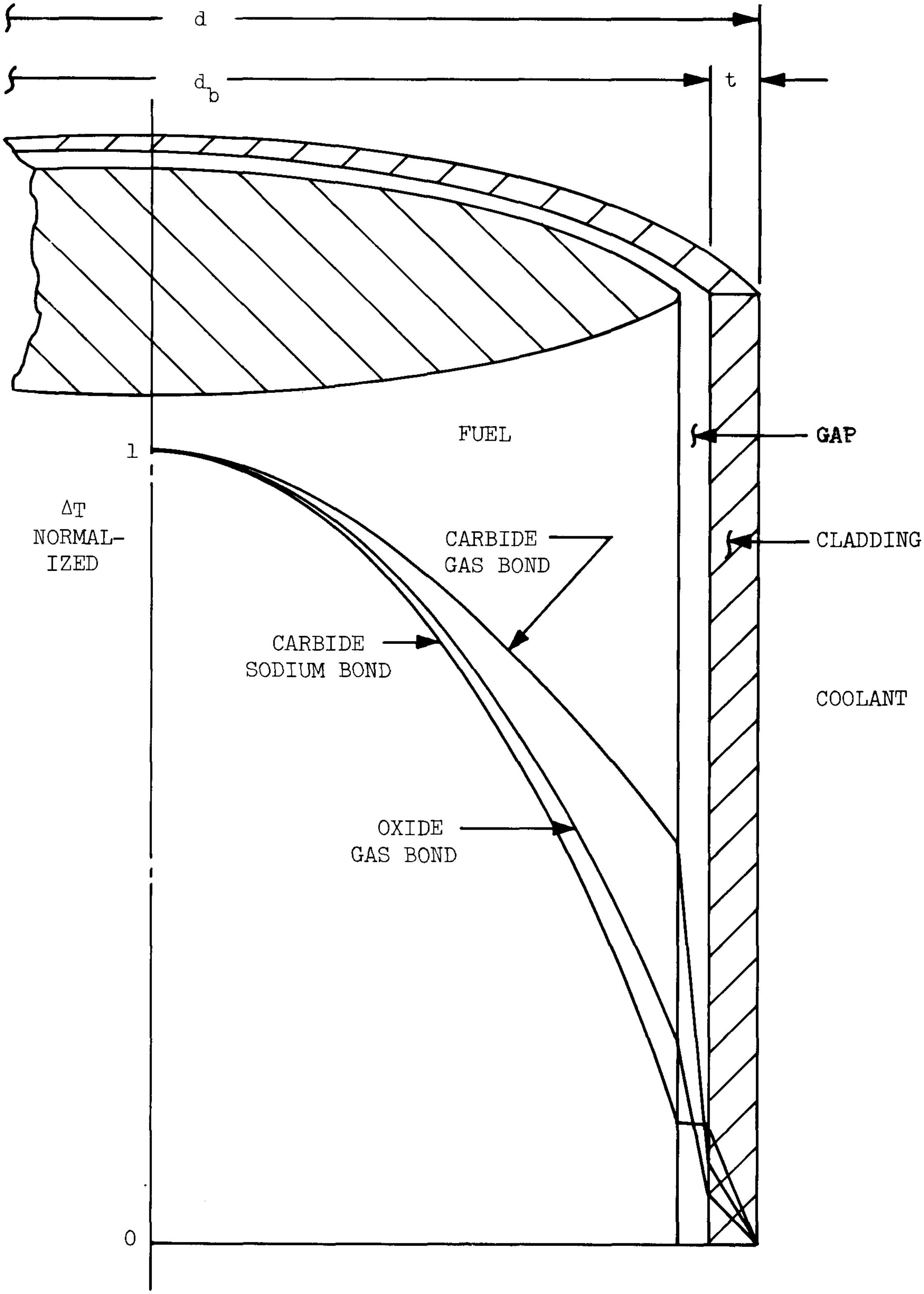


FIGURE 8. *Typical Temperature Profiles Through Fuel Pins*

$^{238}U/^{239}Pu$ RATIO

The $^{238}U/^{239}Pu$ ratio is important in large, fast reactors because of its bearing upon the Doppler coefficient, an important factor in the overall power coefficient of reactivity. For a closed loop test, the local Doppler coefficient (and hence the $^{238}U/^{239}Pu$ ratio) is not necessarily a limiting factor. The ratio, however, does affect the closed loop test performance capability and therefore could be of interest. The $^{238}U/^{239}Pu$ ratio parameters for carbide fuels are shown on the flow-dimension characteristic curves by Figures 9, 10, and 11 (for the 61, 37, and 19-pin bundles) for temperature rises of 400, 300 and 200°F, respectively. Appendix D gives the derivation of the equations for determining parameters.

The $^{238}U/^{239}Pu$ ratio has been labeled to indicate plutonium enrichment. ^{235}U is also an enrichment candidate. The parameters are still roughly applicable for the case where the ^{235}U is the fissile fuel. It is expected that test fuels will be formulated to simulate the proposed fuel chemistry by Pu/U ratio. If necessary, ^{235}U will be used to achieve the desired power.

The parameters of $^{238}U/^{239}Pu$ ratio are based on an assumed first-core, total peak flux of 0.7 x 10^{16} nv.(5) This flux value is taken at the midpoint of the most central closed-loop. The closed loops away from the center of the core will have lower flux values (about 0.35 x 10^{16} nv) and the parameters of the characteristic flux curves must be lowered proportional to the flux value. Advanced FTR cores have been projected to 470 MW with flux levels up to 1.0 x 10^{16} nv. Accordingly, the parameters of fertile-to-fissile ratio must be adjusted upward proportional to the flux.

Note that the Figures 9, 10 and 11 are based on a theoretical density of 85% which is half-way between the 80% and 90% extremes indicated by the 1000 MWe design studies.(1) The $^{238}U/^{239}Pu$ parameters are proportional to density and can be adjusted for theoretical densities other than 85%.

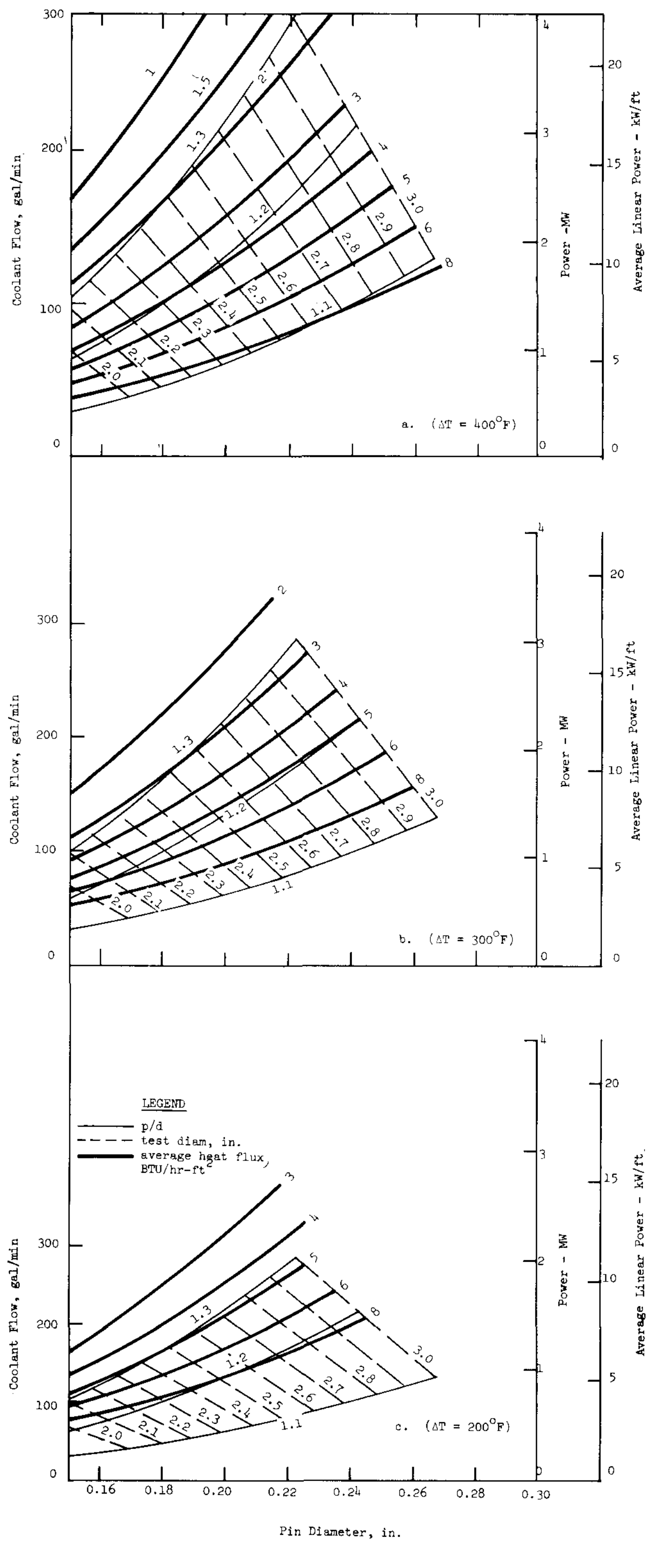


FIGURE 9. *Test Characteristics for 61-Pin Bundle $^{238}U/^{239}Pu$ (Carbide Fuels, 85%ρ, ϕ_{Pk} = 0.7 x 10^{16} nv, F_a = 1.24)*

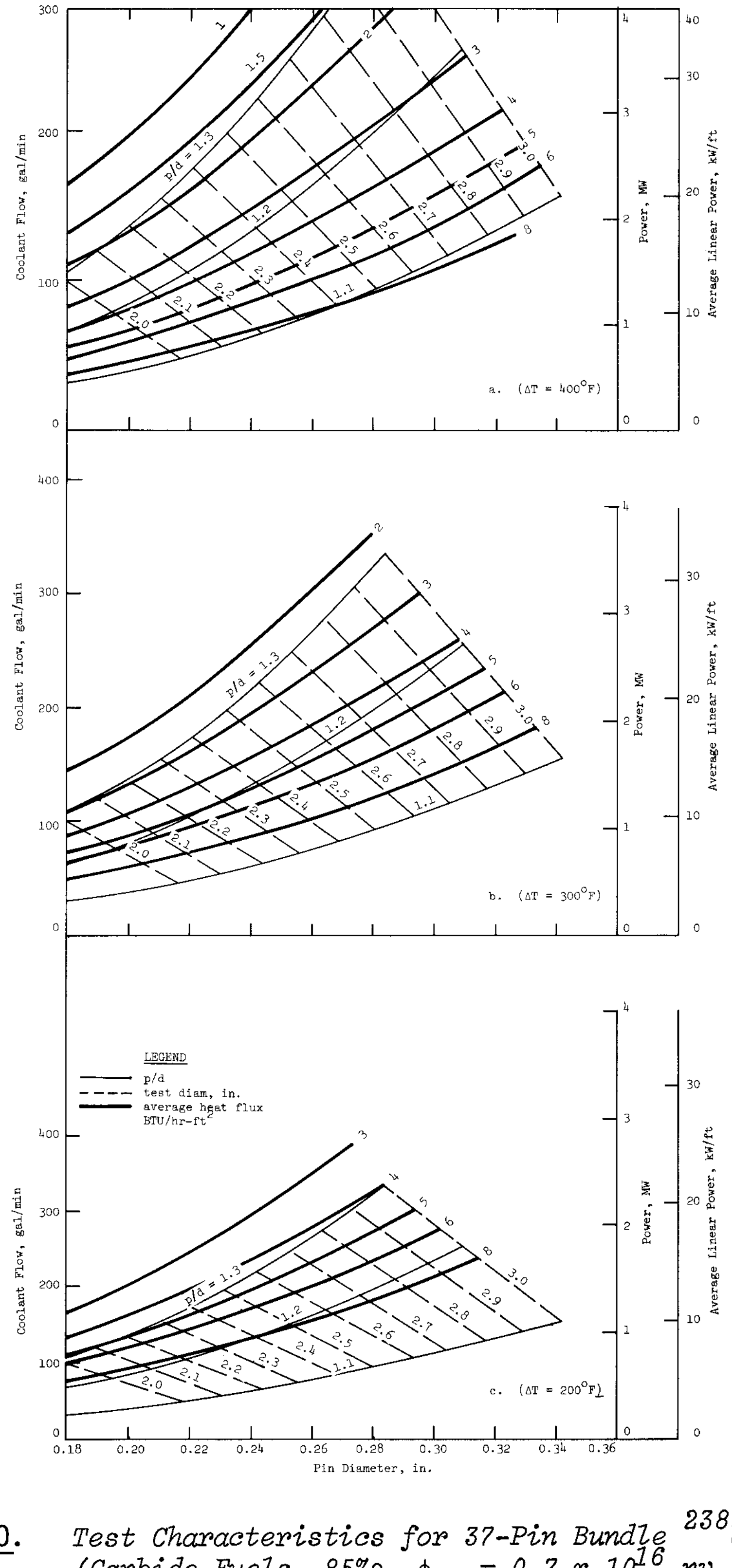


FIGURE 10. *Test Characteristics for 37-Pin Bundle $^{238}U/^{239}Pu$ (Carbide Fuels, 85%ρ, ϕ_{Pk} = 0.7 x 10^{16} nv, F_a = 1.24)*

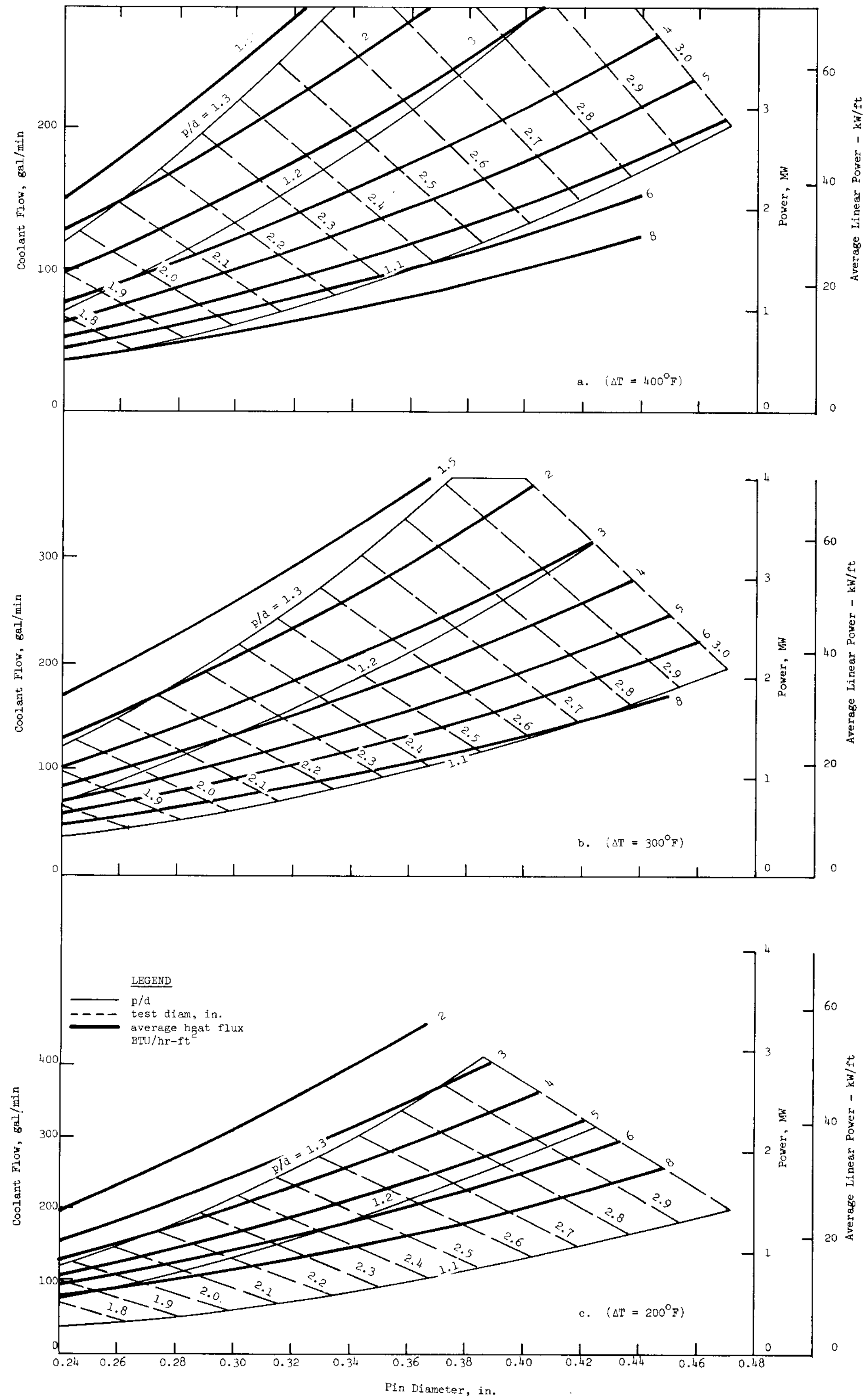


FIGURE 11. *Test Characteristics for 19-Pin Bundle $^{238}U/^{239}Pu$)Carbide Fuels, 85%ρ, $\phi_{Pk} = 0.7 \times 10^{16}$ nv, $F_a = 1.24$)*

PITCH-TO-DIAMETER RATIO

The characteristic curves show that the upper boundary of the pitch-to-diameter ratio, p/d, may be set by upper limits of velocity, heat flux, or linear power. These limits indicate that, for small pin diameters, safe tests can be performed with pin bundles having p/d values greater than 1.3. For large pin diameters the experimenter will tend to select low p/d values to stay within the limits of heat flux, or linear power. Low p/d values, however, have limits which need to be explored. A study[6] discloses that the Nusselt number (hD_e/k) for sodium decreases rapidly as p/d goes from 1.4 to 1.0. The equivalent diameter decreases to a limiting value at p/d = 1 (d_w = 0). Therefore the sodium film coefficient approaches zero at p/d = 1 if the conductivity remains constant. This means that the film coefficient cannot be ignored at low p/d values as is indicated by Figure 8.

TEST DIAMETER, TOTAL POWER, AND 19- OR 37-PIN BUNDLES

The capability to accommodate test bundle pin and pitch sizes of the 1000 MWe studies is shown in Table 3. All dimensions, excepting those proposed by Manufacturer D, can be fitted by 19-pin bundles in less than a 2.5 in. TD. Table 1 data for Manufacturer D show a 42 ft/sec coolant velocity which would necessitate hydraulic testing to prove that it is acceptable before nuclear testing. If Manufacturer D data is disregarded, the two appropriate combinations for test loop design are as follows:

- For a TD of 2.5 in. of somewhat less, use 19-pin bundles and a 2 MW external system.
- For a TD of 3.0 in., use 37-pin bundles and a 3 MW power external system

The important relationship between pin size and TD is shown by parameters of total power (and linear power) in Figure 12 for 19- and 37-pin bundles. There is no known reported experience at heat fluxes higher than the boundary of 2 x 10^6 Btu/hr-ft^2. The 55 kW/ft peak linear power limit may be a maximum for sodium bonded carbide fuels (see Appendix C). Two significant conclusions can be drawn from Figure 12 as follows:

1. A 2 MW heat removal system couples well with a 19-pin bundle and a TD of 2.5 in. or somewhat less. The 44 kW/ft peak linear power is more than enough for testing gas-bonded carbide fuels. If a 2.5 MW system were used, a 55 kW/ft peak linear power for 19-pins would permit testing sodium-bonded fuel pins in a 2.5 in. TD.
2. A 4 MW system is an over-design for 37-pin bundles within a 3.0 in. or less TD. Even if this combination were to be possible for the closed-loop tube, high linear power tests of sodium bonded fuels would still have to use 19-pin bundles.

FUTURE TEST CAPABILITY

Table 2 lists some suggested test parameters for carbide fuels contemplated for the advanced FTR core. Table 4 lists the power required to perform these tests in the closed-loop tube and system. All the helium-bonded tests can be made with 2 MW power and 19-pin bundles and also with 4 MW power and 37-pins. The sodium bonded fuel pins are more of a problem. Only two of the four tests could be done with a 2 MW system. The future 4 MW system could do all four with 19-pin bundles used for the 3(1) and 3(5) tests. Reference to Table 2 discloses that the heat flux through the cladding in three of the four tests is greater than 2×10^6 Btu/hr-ft^2. This heat flux will bear reassessment, particularly No. 3(5) with 2.9×10^6. High heat fluxes are desirable to gain the most benefit from sodium-bonded fuels, however, experience has not yet established a safe limiting value. The proposed tests were presented almost three years ago and it is possible that the authors would revise the values at the present time.

The 61-pin bundle has been ignored because there really is not much in its favor. A 2.5 in. TD would limit testing to pin dameters under 0.22 in. and at little more than 10 kW/ft linear power. A 3.0 in. TD and 4 MW power is a little better. A limit on linear power of 22 kW/ft average would accommodate only oxide fuels or low-powered carbides.

TABLE 3. *Closed Loop Mock-up of 1968 - 1000 MWe Design Studies*

Mfgr.	Fuel	Dimensions d, in.	Dimensions p/d	Test Diameter 19 Pins	Test Diameter 37 Pins	Average Lin. Pwr. kW/ft	Power, MW 19 Pins	Power, MW 37 Pins
A	oxide	0.28	1.20	2.00	2.73	13.7	0.78	1.52
B	oxide	0.25	1.34	2.10	2.74	12.1	0.69	1.34
C	oxide	0.23	1.20	~1.70	2.30	14.6	0.83	1.62
D	carbide	0.399	1.18	2.78	>3.5	35.1	2.00	3.90
E	carbide	0.30	1.24	2.28	3.0	24.7	1.41	2.74

TABLE 4. *Power for Carbide Fuel Tests*

Case(Run No.)	1(2)	2(1)	3(1)	3(5)	4(3)	5(1)	6(2)	7(2)
Bond	← sodium			→	← helium			→
Av. kW/ft	29.9	25.0	46.5	45.2	21.0	29.4	22.7	23.0
Power, MW 19 Pins	1.70	1.43	2.65	2.58	1.20	1.68	1.29	1.31
37 Pins	3.31	2.78	5.19	5.02	2.33	3.27	2.51	2.55

APPENDIX A

NOMENCLATURE

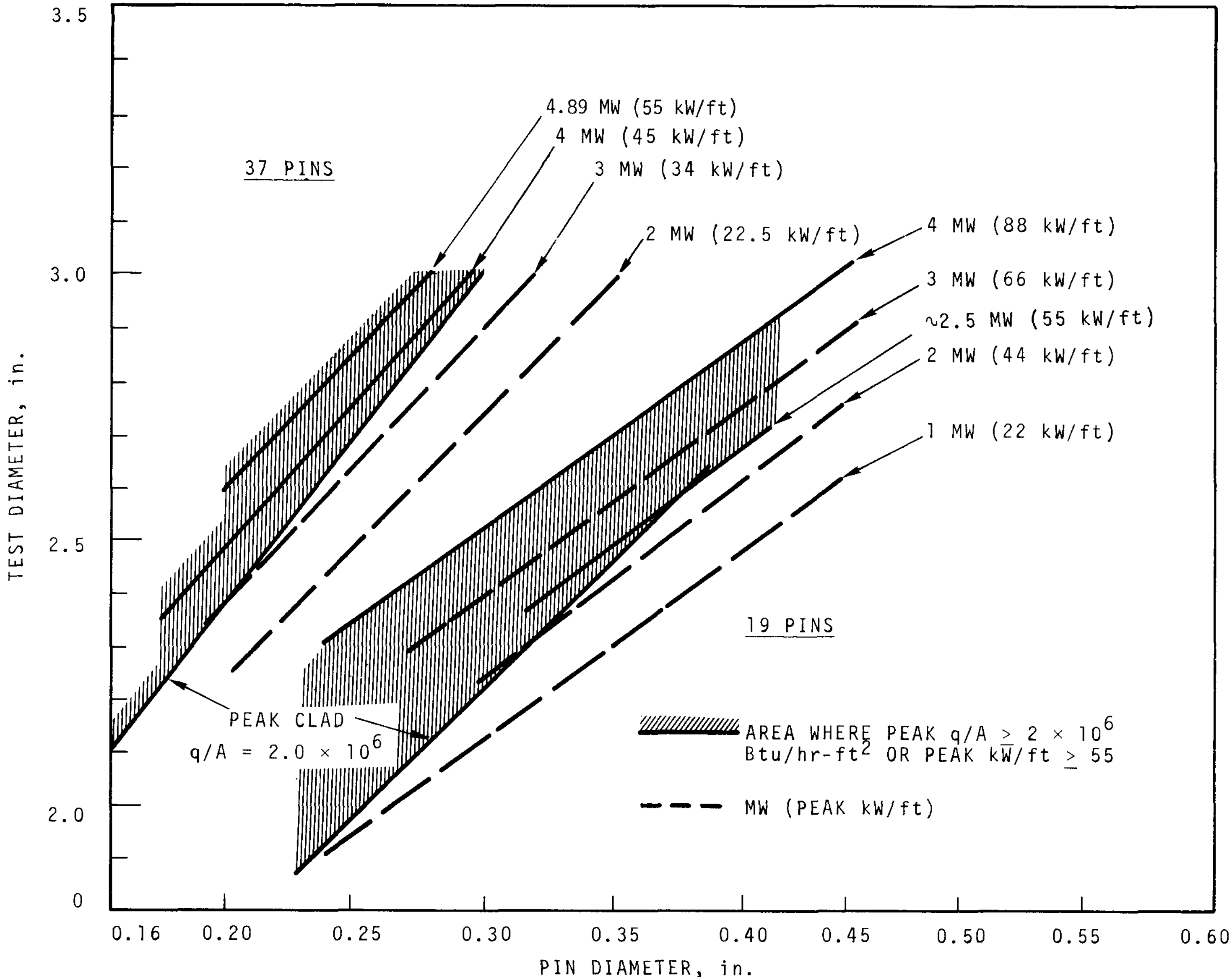


FIGURE 12. *Closed Loop Test Bundle Capability for 400°F Temperature Rise*

EFFECTS OF CHANGES IN ASSUMPTIONS

This preliminary study provided a limited characterization of the closed-loop test capability. Additional analyses are necessary to cover thoroughly the full range of test potentials. Based on information contained herein, other assumptions can determine test characteristics beyond those covered here. Some possibilities are as follows:

- Raising the test pressure drop above 90 psi would raise the flow and power capabilities above those shown in the figures of this study. When the pressure drop of the closed loop primary system is characterized, this prospect can be explored in more detail.
- Shortening the test pin length from the assumed 72 in. would reduce the flow resistance with a corresponding flow and power increase for the same pressure drop.
- Shortening the length of the test fuel would permit driving the pins at higher linear power for the same temperature rise. This is a means to achieve a 55 kW/ft peak linear power when the coolant system is limited to 2 MW.
- Leaving one or more pins unfueled would achieve higher linear powers for the fuel within the temperature rise and power limits. Unfueled pins may be desirable in order to use them for instrumentation.

NOMENCLATURE

FOR PIN BUNDLE

d = pin diameter, in.

d_w = wire diameter, in.

p = triangular pitch $(d + d_w)$

t = thickness of cladding, in.

D_F' = hex dimension across flats for snug-fit bundle, in.

D_F = hex dimension across flats including fit allowance, in.

D_T' = hex dimension across corners for snug-fit bundle, in.

D_T = hex dimension across corners including fit allowance, in.

TD = test diameter (flow tube ID)

L = length, in.

F = fit factor, D_F/D_F' or D_T/D_T'

A = flow area, in.^{2}

A_f = flow area, ft^2

N = number of pins in bundle

N_D = number of pins in long hex dimension

D_e = equivalent diameter, in.

FOR FLOW AND PRESSURE DROP

W = flow, lb/hr

M = gal/min

V = velocity, ft/sec

ΔP = total pressure drop, psi

ΔP_f = friction pressure drop, psi

ΔP_c = entrance and exit pressure drop, psi

K = dimensionless resistance coefficient

g = gravitational constant, 32.2 ft/sec^2

A = flow area, in.^{2}

B = wetted perimeter, in.

Re = Reynold's Number

FOR PROPERTIES

ΔT = temperature rise or temperature difference, °F

ρ = density, lb/ft^3

ρ_g = density, gm/cc

μ = viscosity, lb/hr-ft

c_p = specific heat, Btu/lb-°F

k = thermal conductivity, Btu/hr-ft-°F

h = local heat transfer coefficient, Btu/hr-ft^2-°F

E = modulus of elasticity

ν = Poisson's ratio

α = coefficient of expansion, °F^{-1}

FOR OTHERS

σ = stress

q = power, MW

q_L = linear power, kW/ft

w = weight

ϕ = average neutron flux, nv

ϕ_{pk} = peak neutron flux, nv

F_a = axial peaking factor

F_r = radial peaking factor

C = constant or coefficient

APPENDIX B

EQUATIONS AND DATA

EQUATIONS AND DATA

This appendix contains equations, coefficients, sodium properties, and plots for deriving the pressure drop of test bundles and the heat removal power. These were used for deriving values for the characteristic curves. They should be helpful to determine data points other than those on the curves presented in this study.

The dimensional arrangement of the hex bundles is shown in Figures B-1 and B-2.

EQUATIONS

Snug-Fit Dimensions

$$D_T' = (N_D-1)(d + d_W) + \frac{2}{\sin 60°}\left(\frac{d}{2} + d_W\right)$$

$$= (N_D + 0.1547)\ d + (N_D + 1.3094)\ d_W$$

$$= (N_D P + 1.3094\ P - 1.1547)\ d = C_D d$$

Where C_D, diagonal coefficient, is listed in Table B-1

$$D_F' = D_T' \sin 60° = 0.86603\ D_T'$$

Shroud Dimensions, Inside

$$D_F = D_F' + \text{allowance}$$

$$D_T = D_F \sin 60° = 1.1547\ D_F$$

Fit - Factor, F

$$F = D_F/D_F' = 1.007 \text{ (based on FFTF fuel assembly)}$$

Cross-Section

$$A = d^2 \{0.6495[F(N_D P + 1.3094\ P - 1.1547)]^2 - 0.7856\ N(P^2 - 2P + 2)\} = C_A d^2$$

Where C_A, area coefficient, is listed in Table B-1.

Perimeter

$$B = 3\ D_T + N\ \pi\ P\ d = C_B\ d$$

Where C_B, perimeter coefficients, is listed in Table B-1

Equivalent Diameter

$$D_e = 4\ A/B\text{; can be selected from Figure B-3}$$

Pressure Drop Equations

$$\Delta P = \Delta P_f + \Delta P_c = \frac{V^2\rho}{288g}\left[\frac{fL}{D_e} + K\right]\text{, psi}$$

Where f can be found from Figure B-4 or is

$$f = \frac{0.20}{Re^{0.2}} \text{ for } 10^4 < Re < 2 \times 10^5\text{, derived from Ref. 7 data}$$

K = 1.5 is the assumed combined resistance coefficient for entrance and exit of the bundle

$$V = \frac{W}{25\rho A} = 0.321\ \frac{M}{A}$$

$$Re = \frac{W\ D_e}{12\ A_f} = \frac{12W\ D_e}{\mu\ A}$$

De, see Figure B-3

ρ, k, c and μ are listed in Table B-2

By using $f = 0.20/Re^{0.2}$, L = 72 in. and the 1000°F properties,

$$\Delta P = C_f\ M^{1.8} + C_c M^2\text{, psi}$$

Where

$$C_f = 2.525 \times 10^{-4}\ \left(\frac{C_B^{\ 1.2}}{C_A^{\ 3}\ d^{\ 4.8}}\right)$$

$$C_c = 8.57 \times 10^{-4}/(C_A^{\ 2} d^4)$$

Power Equation

The heat transport equation for the coolant flowing through the test is

$$q = \frac{W\ C_p\ \Delta T}{3{,}413{,}000},\ \mathrm{MW}$$

$$= \frac{8.03\ \rho\ M\ C_p\ \Delta T}{3{,}413{,}000}$$

at an average temperature of 1000°F,

$$q = 3.635 \times 10^{-5}\ M\ \Delta T,\ \mathrm{MW}$$

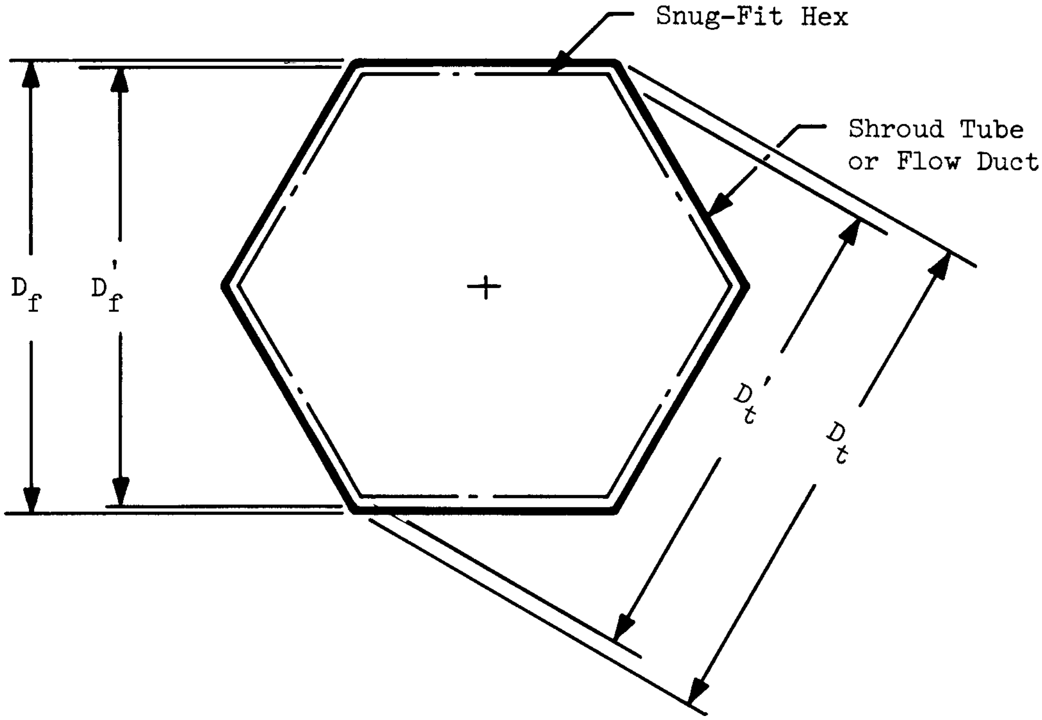


FIGURE B-1. *Hex Dimension of Pin Bundle*

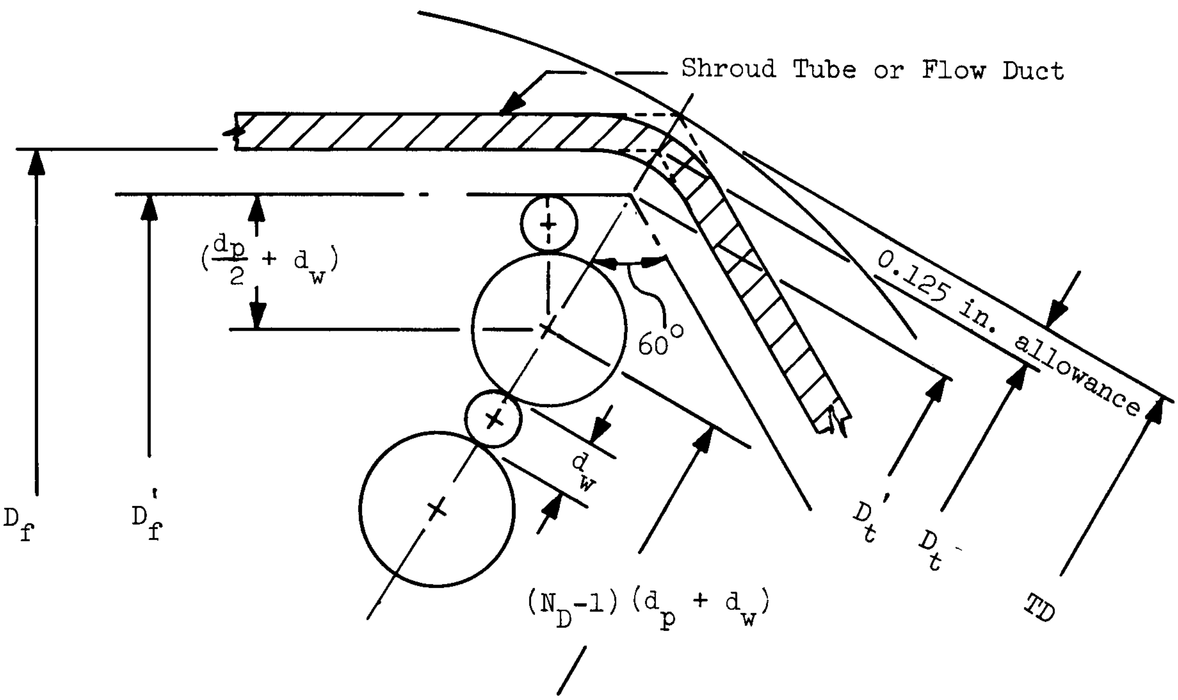


FIGURE B-2. *Hex Corner Details*

TABLE B-1. *Coefficients*

Coef. N/P	Diagonal C_D 61	37	19	7	Area Coefficient, C_A 61	37	19	7	Perimeter Coefficient, C_B 61	37	19	7
1.10	10.186	7.986	5.786	3.586	19.93	12.64	6.970	2.913	241.6	152.0	83.14	35.02
1.12	10.392	8.152	5.912	3.672	22.51	14.28	7.877	3.301	246.0	154.8	84.71	35.72
1.14	10.598	8.318	6.038	3.758	25.11	15.93	8.795	3.694	250.5	157.6	86.29	36.42
1.16	10.804	8.484	6.164	3.844	27.73	17.60	9.717	4.093	254.9	160.5	87.86	37.12
1.18	11.010	8.650	6.290	3.930	30.36	19.27	10.65	4.496	259.4	163.3	89.44	37.82
1.20	11.217	8.817	6.417	4.017	33.02	20.96	11.59	4.906	263.9	166.1	91.01	38.52
1.22	11.423	8.982	6.543	4.103	35.69	22.67	12.55	5.320	268.3	169.0	92.59	39.22
1.24	11.629	9.148	6.669	4.189	38.38	24.38	13.51	5.740	278.8	171.8	94.16	39.92
1.26	11.835	9.315	6.795	4.275	41.09	26.12	14.48	6.166	277.2	174.6	95.74	40.62
1.28	12.041	9.481	6.921	4.361	43.81	27.86	15.45	6.597	281.7	177.4	97.31	41.32
1.30	12.247	9.647	7.047	4.447	46.55	29.62	16.44	7.033	286.1	180.3	98.89	42.02

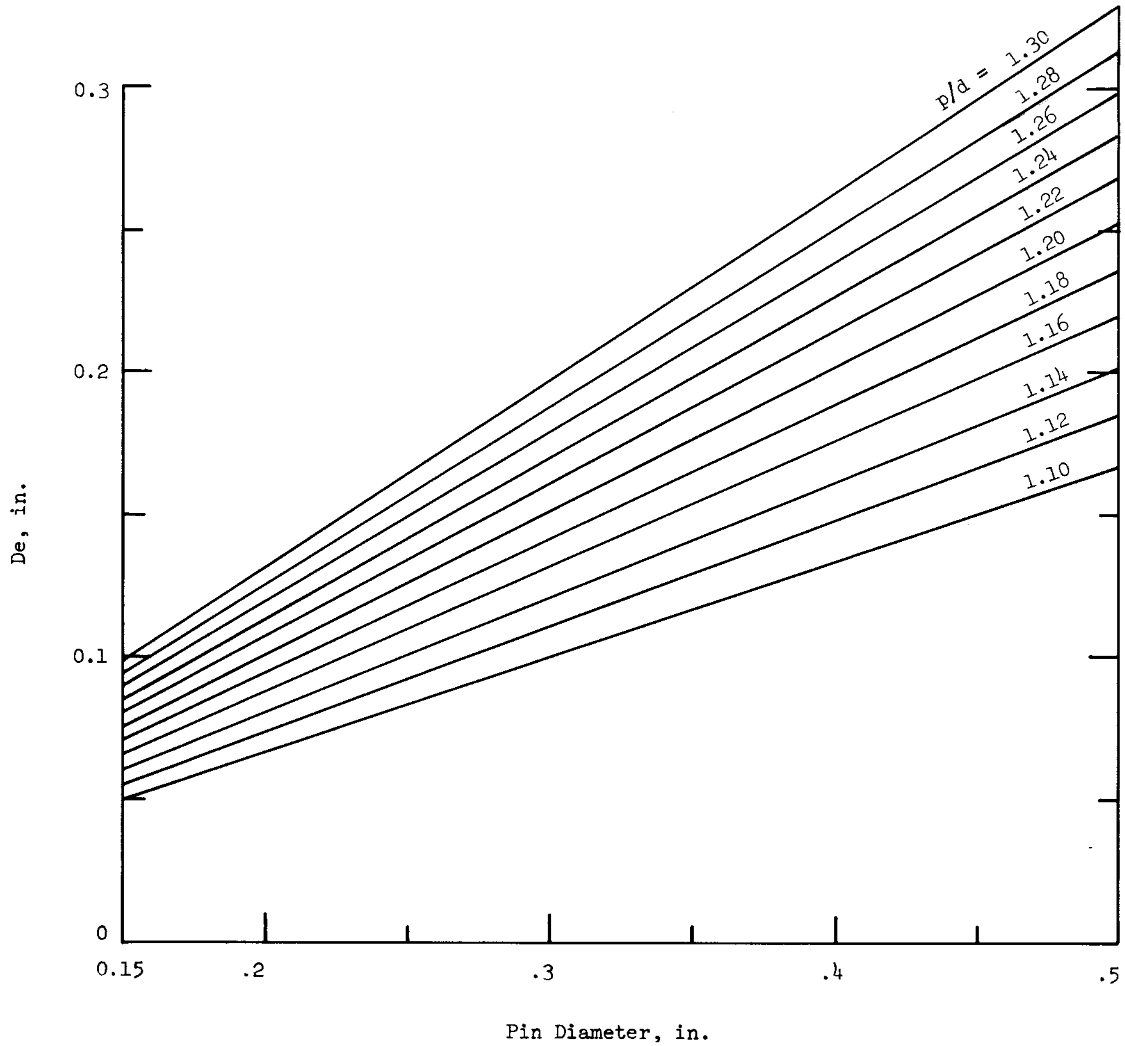


FIGURE B-3. *Equivalent Diameters for Hexagonal Bundles (61-, 37-, 19- and 7- Pins)*

Drawn from Reference (7) results

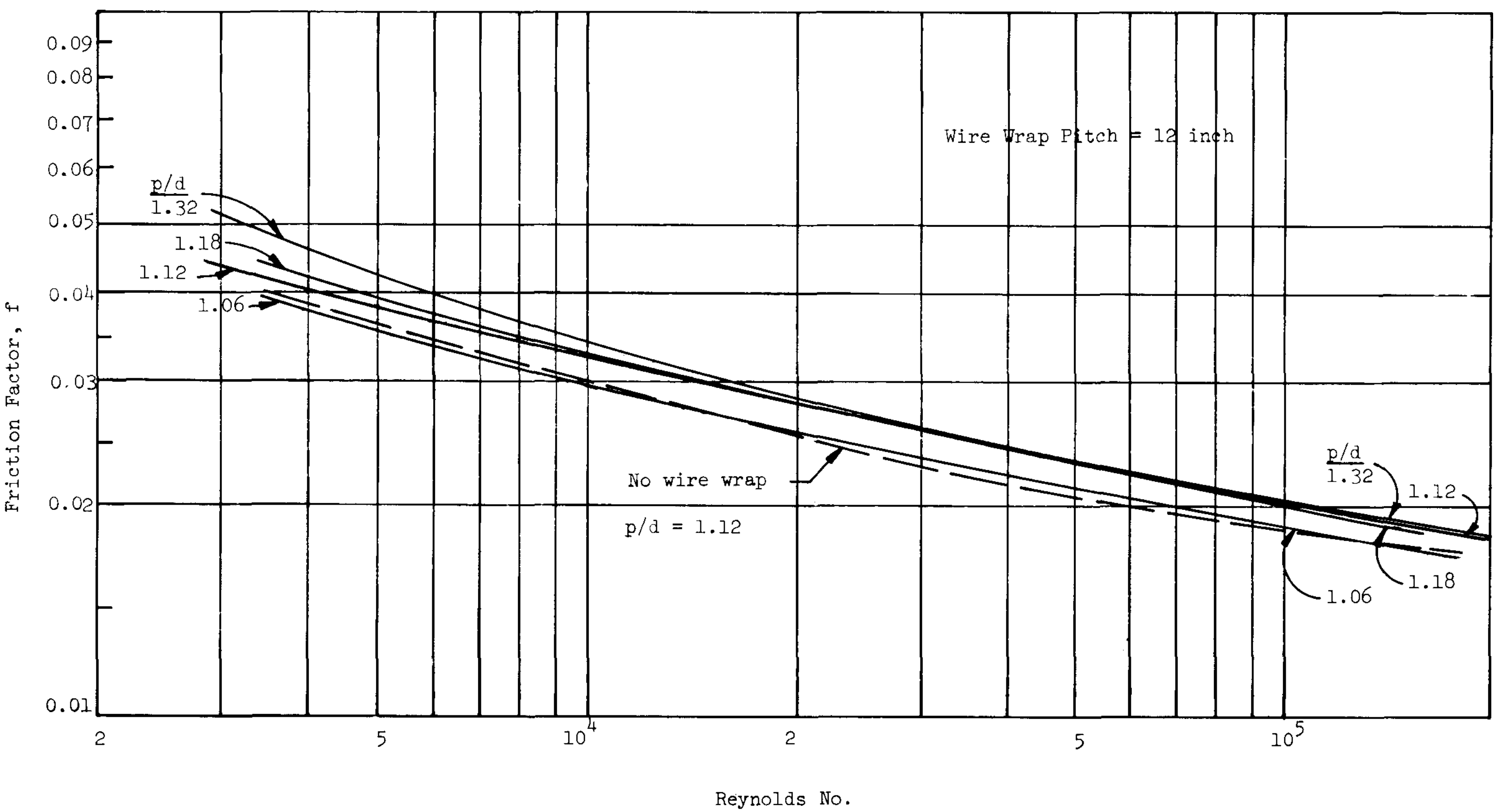


FIGURE B-4. *Friction Factor for Hex Pin Bundles*

TABLE B-2. *Properties of Liquid Sodium*

Temp, °F	Density lb/ft^3	Thermal Cond. Btu/hr-ft-°F	Specific Heat Btu/lb-°F	Viscosity lb/ft-hr
400.0	56.345	47.253	0.31950	1.07779
450.0	55.937	46.467	0.31699	0.99111
500.0	55.528	45.680	0.31465	0.91913
550.0	55.118	44.894	0.31248	0.85852
600.0	54.707	44.107	0.31048	0.80686
650.0	54.295	43.321	0.30864	0.76236
700.0	53.882	42.534	0.30698	0.72367
750.0	53.469	41.748	0.30549	0.68973
800.0	53.054	40.961	0.30417	0.65973
850.0	52.639	40.175	0.30303	0.63304
900.0	52.224	39.389	0.30205	0.60914
950.0	51.807	38.602	0.30124	0.58237
1000.0	51.390	37.816	0.30060	0.55452
1050.0	50.973	37.029	0.30013	0.52960
1100.0	50.555	36.243	0.29984	0.50719
1150.0	50.137	35.456	0.29971	0.48694
1200.0	49.718	34.670	0.29975	0.46854
1250.0	49.299	33.884	0.29997	0.45178
1300.0	48.879	33.097	0.30035	0.43643
1350.0	48.460	32.311	0.30091	0.42233
1400.0	48.040	31.524	0.30163	0.40933

APPENDIX C

TEMPERATURE LIMITS ON FUEL PINS

TEMPERATURE LIMITS ON FUEL PINS

FUEL TEMPERATURE LIMIT

The maximum fuel temperature (assumed to be incipient centerline melting) is one obvious limiting criterion worth examining. It is equal to the coolant temperature plus the temperature difference to the fuel centerline. The general equation for the temperature difference is:

$$\Delta T = \frac{q_L}{\pi}\left[\frac{1}{4k_f} + \frac{12}{h_b d_b} + \frac{t}{dk_s} + \frac{12}{h_c d}\right] \times 3410, \ ^\circ F$$

where the bracketed terms apply respectively to fuel, bond, cladding and coolant film. The slab geometry equation is used for the cladding because thickness is small compared to radius. The last term is usually small, therefore, it is disregarded.* The bond term, only when applied to sodium bond, is disregarded. The applicable equation for temperature difference then becomes:

$$\Delta T = \frac{q_L}{\pi}\left[\frac{1}{4k_f} + \frac{12}{h_b d_b} + \frac{t}{dk_s}\right] \times 3410 \text{ for gas-bonded fuel pins}$$

and

$$\Delta T = \frac{q_L}{\pi}\left[\frac{1}{4k_f} + \frac{t}{dk_s}\right] \times 3410 \text{ for sodium-bonded fuel pins}$$

These equations are used for arriving at the relationships between temperature difference and linear power in Figures C-1 and C-2.

The properties and other data used are tabulated as follows:

For fuels:	oxide	k = 2 Btu/hr-ft-°F
	carbide	k = 10 Btu/hr-ft-°F
cladding		k = 13 Btu/hr-ft-°F
cladding thickness		t = 0.0652 d, in.; this ratio suits FFTF pins but is not necessarily a valid rule
gap coefficient		h = 2000 Btu/hr-ft^2-°F [(8)]**
coolant temperature		T = 1000°F

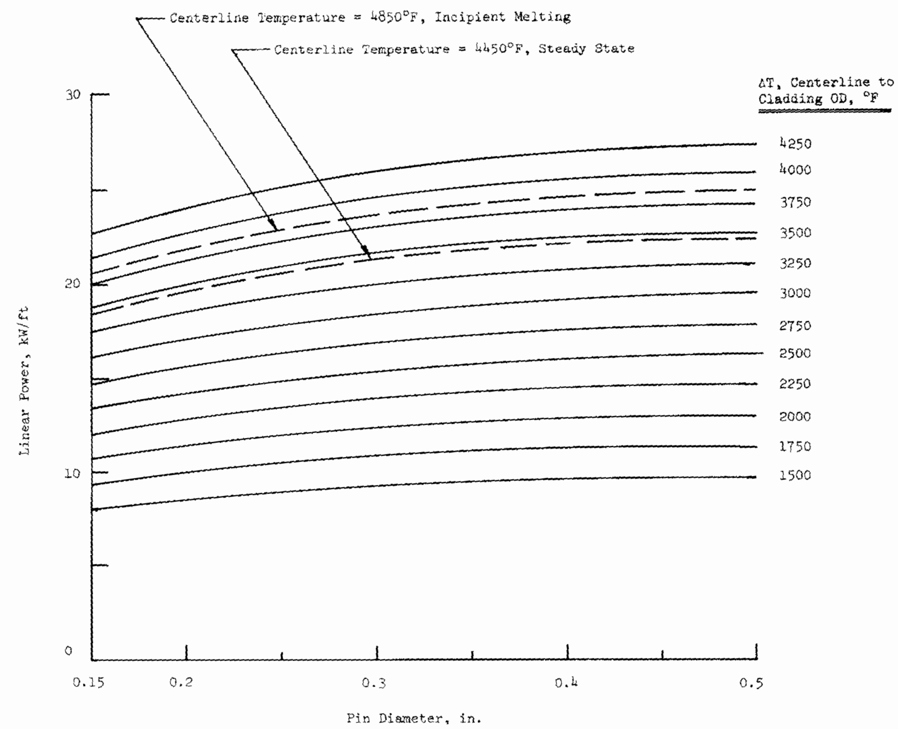
* See pitch-to-diameter ratio discussion in text
** See Figure 1.2.2 of the Reference

Note that for gas-bonded pins there is a dependence upon the pin diameter wnich does not apply to sodium-bonded pins.

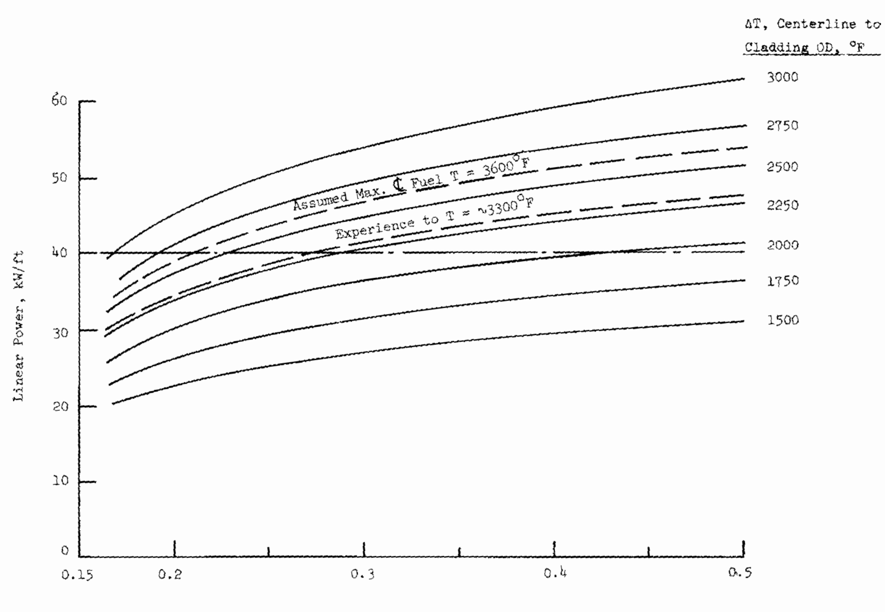
In Figure C-1a for gas-bonded oxide fuels, the maximum melt temperature and the steady-state temperature are those specified for the FTR core. In Figure C-1b for gas-bonded carbide fuels, the two maximum temperatures shown are based on a statement, page 44 of Reference 2: "With helium-bonded carbide fuel some temperature experience exists to 1800°C (about 3300°F), but the allowable maximum centerline temperature is unknown. For the fuel cases, a maximum beginning-of-life temperature of 2000°C (about 3600°F) was assumed (operating temperature are expected to be much lower)." In Figure C-2 for carbide fuels sodium bonded, page 44 of Reference 2 offers another quote for the temperature maximum. It is, "With sodium-bonded stoichiometric (or stabilized) fuel, the probable maximum centerline temperature which should be allowed is 1400 to 1500°C (2552 to 2732°F)."

Oxide fuel pins tend to be smaller than carbide fuel pins because of thermal conductivity differences. Pin diameters of 0.15 to 0.25 in. are used for oxide fuels and diameters of 0.30 to 0.40 in. are projected for carbide fuels. Examining Figures C-1a and C-1b over these ranges respectively the linear powers of 20 and 40 kW/ft are good rule-of-thumb limits. In Figure C-2, the maximum temperature limits suggested herein, results in a linear power limit close to the 55 kW/ft mentioned in the text for sodium-bonded carbide fuel pins. More experience with the latter is required before a firmer linear power limit is established.

A number of simplifying assumptions such as lumped properties over the temperature range, coolant temperature, t/d ratio, etc., were made in order to provide a rational basis for limits of linear power. When specific test bundles are to be tested, a more rigorous analysis needs to be conducted and the temperature limits carefully considered. The temperature limits that may be applicable to the cladding should not be overlooked. These limits are discussed next.

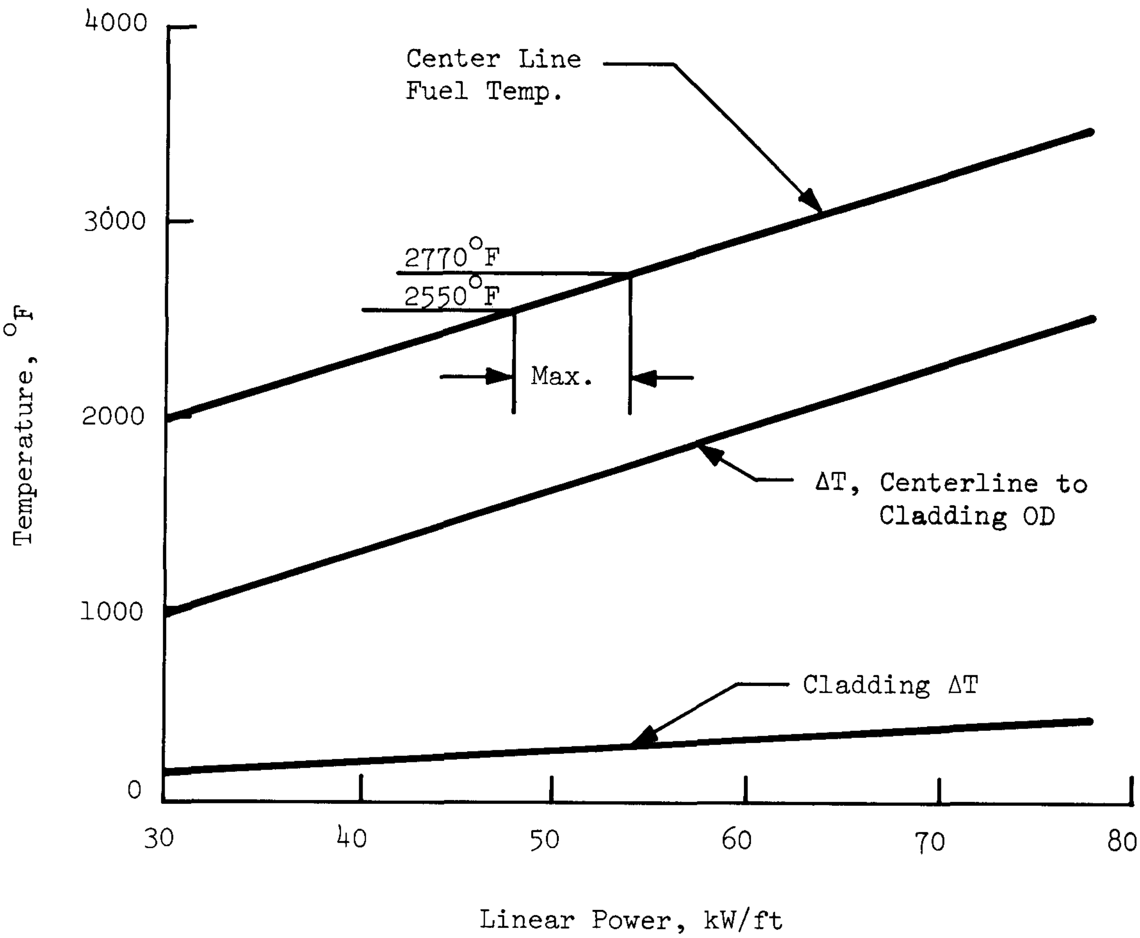


a. oxide fuels



b. carbide fuels

FIGURE C-1. *Fuel Pin Temperature Difference for Gas Bonded Fuels*



<u>FIGURE C-2.</u> *Fuel Pin Temperature Difference for Carbide Fuel Sodium Bonded*

CLADDING DIFFERENTIAL TEMPERATURE LIMITS

Austenitic stainless steel used for cladding has a firm upper temperature limit. The maximum interface temperature is limited to less than 1800°F since the stainless steel, fuel (carbide) eutectic melts at about 1000°C (1832°F).[(2)*] The interface temperature difference is dependent upon the coolant temperature plus the cladding temperature difference which in turn is a function of the heat flux. There is discussion of the latter already in the text.

Calculation of thermal stress in tubing can be estimated from the equation:

$$\sigma = \frac{0.7\ E\ \alpha\ \Delta T}{1-\nu}$$

Using this equation, and the properties and allowable stresses for 1200 and 1400°F, the temperature difference allowed is in the neighborhood of 100°F. A higher temperature difference is accepted for fuel pin cladding based on reactor experience. An example of the temperature difference expected from Rochrock's study lists cladding ΔT = 230°F.[(4)**] Figure C-2 shows the cladding temperature difference as a function of linear power for the given t/d ratio of 0.0652. It is recognized that the cladding temperature also may be limited by pin hoop-strength requirements.

The subject of metal behavior in the neutron environment of a core is complex, and the limiting factors differ from those applicable ex-reactor. Neutron bombardment at certain energy levels causes the rearrangement of the metallic atoms of the cladding and thereby relieves high thermal stresses. The dimensional changes which occur because of this phenomenon, and concurrent swelling, is a matter of widespread concern. Core exposure limits of the cladding cannot be specified for certain at the present time.

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* See Page 44 of the reference

** In Table 4 of the reference

APPENDIX D

$^{238}U/^{239}PU$ RATIO

$^{238}U/^{239}PU$ RATIO

The parameters of fertile-to-fissile ratio, $^{238}U/^{239}Pu$, relationship to power, number of pins, pin diameter and flux is derived from the approximation:

$$q = C\,(\phi \times 10^{-16})\; w_{^{239}Pu}, \quad MW$$

$$= C\,\frac{(\phi_{pk} \times 10^{-6})}{F_a \times F_r}\; w_{^{239}Pu}$$

The product of peaking factors, $F_a \times F_r$, is 1.30 x 1.33 = 1.74.(2)* At present, for the FFTF, F_a = 1.24 and F_r = 1.40 the product of which is the same 1.74. Using this value the proportionality constant, C, was found from Appendix A studies of Reference 2 to be about 1.83 for ^{239}Pu and 1.65 for ^{235}U for projected FFTF advanced cores.

The power equation applicable to a closed loop position is:

$$q_t = 1.83 \times 10^{-16}\,\frac{\phi pk}{F_a}\, w_{^{239}Pu} = 1.476 \times 10^{-16}\,\phi_{pk}\, w_{^{239}Pu}, \quad MW$$

where ϕ_{pk} is that of the test position, and F_a is 1.24. F_r is not applicable.

The next step is to find the test $w_{^{239}Pu}$ in terms of the fertile-to-fissile ratio, defined as the ratio by weight of $^{238}U/^{239}Pu$. Account is taken of the presence of ^{240}Pu in the fuel by assuming it to be 12% of the total Pu. The total weight of fuel is:

$$w = w_{^{239}Pu} + w_{^{240}Pu} + w_{^{238}U} + w_{carbon}$$

Knowing that carbide fuel is 5% by weight of carbon, the equation for the weight of fissile plutonium becomes:

$$w_{^{239}Pu} = \frac{w}{1.196 + 1.052\,(w_{^{238}U}/w_{^{239}Pu})}$$

---

* Page 56 of the reference

The total fuel weight, w, can be found as a product of volume and density. Using 13.6 g/cm^3 for theoretical density and 85% smeared density to allow void volume for swelling, the equation for total weight is:

$$w = \frac{N\pi(d-2t)^2 \; L \quad 0.85 \times 16.39 \; \rho_g}{4 \times 1000}, \text{ kg}$$

Substituting L = 36 in. and simplifying:

$$w = 5.36 \; N \; (d-2t)^2, \text{ kg}$$

Substituting equation (3) into (2):

$$w_{^{239}Pu} = \frac{5.36 \; N \; (d-2t)^2}{1.196 + 1.052 \; (w_{^{238}U}/w_{^{239}Pu})}, \text{ kg}$$

Applying this expression for the weight of fissile plutonium the closed loop power is:

$$q_t = \frac{7.90 \times 10^{-16} \; \phi_{pk} \; N \; (d-2t)^2}{1.196 + 1.052 \; (w_{^{238}U}/w_{^{239}Pu})}$$

This is the equation used for determining the $^{238}U/^{239}Pu$ parameters of Figures 9, 10 and 11. For cladding thickness, t, the equation of Rothrock's from Reference 4 was used, t = 0.005 + (0.010/0.235), d, and the gap between fuel and cladding was disregarded.

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