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A THERMAL HYDRAULIC ANALYSIS
OF THE
PACIFIC NORTHWEST LABORATORY
FUSION-FISSION HYBRID REACTOR CONCEPT

Battelle

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A THERMAL HYDRAULIC ANALYSIS
OF THE
PACIFIC NORTHWEST LABORATORY
FUSION-FISSION HYBRID REACTOR CONCEPT

by

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June 1973

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THERMAL HYDRAULIC ANALYSIS OF THE PNL FUSION-FISSION HYBRID REACTOR CONCEPT

A. M. Sutey and C. W. Stewart

INTRODUCTION

The PNL Fusion-Fission Hybrid concept^(1,2,3) consists of a sub-Lawson D-T fusion plasma surrounded by a helium-cooled neutron converter and a helium-cooled graphite-uranium subcritical fission lattice. This type of hybrid exhibits a high potential for a viable power plant while utilizing a fusion plasma whose operating requirements are more than an order of magnitude less stringent than break-even conditions for a pure fusion device.⁽³⁾

A conceptual design has been developed in an attempt to characterize a fission blanket system which will provide maximum power multiplication of D-T source neutrons. To date the dimensions of the lattice concepts have been somewhat arbitrarily chosen to be suitable primarily from a physics standpoint. The purpose of this study is to evaluate these concepts on a thermal-hydraulic basis to further define the technical feasibility of the hybrid concept.

SUMMARY

A preliminary thermal hydraulic analysis was performed on a helium-cooled subcritical blanket of a fusion-fission reactor concept having an overall power level of 0.75 MW/m^3 . Three candidate fuel rod materials, metallic uranium, uranium carbide and uranium oxide, were considered. Helium coolant operating conditions considered were similar to those typical of an HTGR.⁽⁵⁾ Both steady state and pulsed power modes of operation were studied.

If pulsed mode of operation is necessary, the total cycle period should be less than one second if the pulse time is short (0.1-0.2 sec) or less than two seconds if the pulse time is longer (0.5-1.0 sec). The limiting factor is the magnitude of the resulting cyclic temperature variation in the lattice area. Uranium oxide rods may be able to tolerate a somewhat longer period since the maximum rod temperature for this material is much higher than uranium or uranium carbide.

Analysis shows that to achieve a thermodynamic efficiency comparable to present power reactors, the helium temperature at the reactor exit should be at least 1000°F. This criterion excludes the use of metallic uranium in the reference lattice design since its peak center-line temperature may be as high as 1700°F for the single-rod design which is too near its melting point for practical use. If the reference lattice were modified to use four small rods in each coolant channel instead of the single large rod, the peak center-line temperature could be reduced to 1350°F which might be acceptable.

Low temperatures are desirable in the converter to reduce tritium diffusion rates from the lithium into the helium and to prevent helium embrittlement of the first wall and cladding. This study has shown that the converter can be used as a preheater for the lattice coolant. By adjusting the flow rate and module length, the helium can be maintained at low temperatures in the converter while thermodynamically acceptable high temperatures can be maintained at the lattice exit.

REFERENCE DESIGN

GEOMETRY

The conceptual design of the subcritical blanket proposed by Leonard and Wolkenhauer^(3,4) has two major power producing components: a neutron converter at the inner wall, and the fission lattice. As shown in Figure 1, the converter is 8 cm in thickness and consists of closely spaced 0.5 inch diameter uranium rods and 0.25 inch I.D. lithium passages. The uranium rod cladding and the walls of the lithium tubes are 20 mil stainless steel. The converter is shielded inside and outside by stainless steel shells 0.5 cm thick. The inner shield is the first wall from the plasma. In the remaining space helium coolant flows parallel to the uranium rods and lithium tubes.

In the lattice, uranium fuel rods are distributed in a homogeneous bed of graphite 150 cm in thickness with 7-inch spacing between all rods. The original design used a single 1.0 inch diameter rod in a 2.72 inch O.D. annular helium flow passage. A four-rod channel with 0.5 inch rods in a 2.4 inch square helium passage was also investigated to determine if center-line temperatures could be reduced significantly. Unit cells for both these models are shown in Figure 2. The actual arrangement of the lattice cells has not yet been set. To provide a model for heat transfer studies, the cells were arbitrarily arranged in concentric shells such that a 7-inch circumferential rod spacing was maintained and the number of cells in any shell was an exact integer. The resulting assembly, shown in Figure 3, consists of eight concentric rows of shells. The average radial thickness of the new cell became 7.8 inches with the inner and outer shells truncated due to the geometry.

A graphite reflector and lithium absorber form two additional outer shells beyond the lattice. Since very little power is produced in these regions, they were not considered in this analysis. The entire assembly may be arranged in a toroidal shape as in the Tokamak concept⁽⁶⁾ or as a simple cylinder. The latter case was assumed to simplify calculations.

UNIT CELL DATA

FLOW AREA (IN ²)	0.691
WETTED PERIMETER (IN)	5.45
HEATED " "	5.05
HYDRAULIC DIAMETER (IN)	0.508
AVG. HEAT TRANS. COEF. (BTU/HR-FT ² -°F)	265.0
HEAT TRANS. AREA PER FT	0.0352 FT ² /FT

ASSEMBLY DATA (1308 CELLS)

TOTAL VOLUME PER FT.	12.725 FT ³ /FT
VOLUME FRACTION HELIUM	49.52%
" " LITHIUM	5.26%
" " URANIUM	28.03%
" " STEEL	17.19%
TOTAL HEAT TRANS. AREA	46.04 FT ² /FT

FIGURE 1. Converter Detail

SINGLE ROD MODEL

4-ROD MODEL

CHANNEL DATA

4.9	FLOW AREA (IN ²)	4.9
11.94	WETTED PERIMETER (IN)	16.18
3.40	HEATED " "	6.53
1.642	HYDRAULIC DIAMETER (IN)	1.212
200	AVG. HEAT TRANS. COEF. (BTU/HR-FT ² -°F)	185
132000	MAX. HEAT FLUX (BUT/HR-FT ²)	67800
0.0236	HEAT TRANS. AREA (FT ² /FT)	0.0454

ASSEMBLY DATA (804 RODS)

TOTAL VOLUME PER FT	287.64 FT ³ /FT
VOLUME FRACTION HELIUM	9.51%
" " FUEL	1.52%
" " Clad	0.25% (1-ROD)
" " "	0.124% (4-ROD)
" " GRAPHITE	88.71% (1-ROD)
" " "	88.84% (4-ROD)
TOTAL FLOW AREA	27.36 FT ²
HEAT TRANS. AREA PER FT	18.94 FT ² /FT (1-ROD)
" " " "	36.48 FT ² /FT (4-ROD)

FIGURE 2. Lattice Detail

FIGURE 3. Fission Blanket Assembly

POWER DISTRIBUTION

Besides a neutron thermal inner wall flux of 0.05 MW/m^2 , the power produced by the device may be attributed entirely to fission. The radial power distribution through the device was based on a fission density distribution normalized by volume for an average of 0.75 MW/m^3 . Since fission density was determined by assuming completely homogeneous distribution of the various component materials, it was necessary to "re-concentrate" the power production in the fuel rods.

In the lattice, the total power produced in each previously described concentric shell was attributed to the uranium in that shell which was about 1.5% of the total shell volume. This led to power densities of up to 94 times the device average. In the converter, the total power was divided between the uranium and lithium. Distribution was made by dividing the converter into four concentric volumes corresponding to the four rows of uranium rods. Lithium was initially apportioned equally among the four volumes and afterwards re-combined into the three actual tube rows. This procedure gave uranium and lithium power densities up to 11.9 and 6.9 times the overall average, respectively. No power generation was assumed for the graphite, helium or stainless steel in the converter or lattice. The results are summarized in Figure 4 and Table 1.

OPERATING CONDITIONS

Prior to the performance of thermal hydraulic calculations it was necessary to determine acceptable operating conditions. The overall power density was set at 0.75 MW/m^3 , and the radial power distribution was determined as described in the preceding section. It was assumed that the device will eventually be used for power generation, and thus, the coolant outlet temperature must be compatible with the demands of the steam cycle associated with the reactor. As shown in Figure 5, a standard Rankine cycle⁽⁷⁾ with superheat was considered to determine the lower bound on steam temperature for reasonable efficiency. The Rankine cycle is described schematically in Figure 5. Since the sink conditions used

FIGURE 4. Radial Power Distribution

TABLE 1. Rod Power Density

Converter (Figure 1)

Rod no.	Material	p/p_{avg}
1 (inner)	Uranium	11.87
2	"	10.15
3	"	8.569
4 (outer)	"	7.63
5 (inner)	Lithium	6.88
6	"	5.85
7 (outer)	"	5.06

Lattice (Figure 4)

Rod no.	p/p_{avg}
1 (inner row)	47.77
2	91.41
3	94.35
4	83.88
5	69.68
6	54.10
7	37.97
8 (outer row)	18.34

ENTROPY
IDEAL RANKINE CYCLE (SUPERHEATED STEAM)

- 4-1 CONSTANT PRESSURE HEAT ADDITION
- 1-2 CONSTANT ENTROPY EXPANSION (TURBINE)
- 2-3 CONSTANT PRESSURE COOLING (CONDENSER)
- 3-4 CONSTANT ENTROPY COMPRESSION (PUMP)

FIGURE 5. Effect of Pressure and Temperature on a Possible Rankine Cycle

in Figure 5 were chosen arbitrarily, the only purpose of the figure is to show trends. It can be seen that turbine inlet pressure has the greatest effect on efficiency. However, the temperature must be at least 750°F, for example, to allow operation at 2500 psi for an ideal efficiency of 39%. Below this temperature, the higher pressures are not available and efficiency falls rapidly. A temperature of 800°F was set as a desirable design point. Assuming a 200°F helium to steam temperature difference in the steam generating system, the minimum helium delivery temperature should be at least 1000°F.

The specifications for the Fort St. Vrain HTGR⁽⁵⁾ served as a guide for values of flow rate and pressure. The major difference between the Fort St. Vrain core cooling system and that of the hybrid is that the coolant passages are much smaller and the helium does not contact the fuel rods directly in the St. Vrain system. Except for differences in pressure losses, the operation of the two systems is similar. A comparison of Fort St. Vrain specifications and range of parameters considered in this analysis are shown in Table 2.

TABLE 2. Hybrid Operating Conditions

	<u>Range Considered in Hybrid Analysis</u>	<u>Fort St. Vrain HTGR</u>
Core helium outlet temp (°F)	980-1000	1236-1427
Core helium inlet temp (°F)	730-1000	569-742
Core helium pressure drop (psi)	1.0	0.37-3.6
Core helium flow rate (lb/hr-ft ²)	5.5-7.8(10 ⁴)	1.7-6.1(10 ⁴)
Core helium pressure (psia)	600	589-686
Core heat flux (BTU/hr-ft ²)	0.08-1.31(10 ⁵)	0.45-1.2(10 ⁵)
Heat transfer coef (BTU/hr-ft ² °F)	180-260	230-400
Steam outlet temp (°F)	800	1000
Steam inlet temp (°F)	--	300-400
Steam outlet pressure (psia)	2500	2500

MATHEMATICAL MODELING

The helium passages in the device are well suited to subchannel analysis. For this reason the COBRA computer code,⁽⁸⁾ which was developed for subchannel analysis of nuclear reactor rod bundles, was adapted for the single phase gas coolant. COBRA uses a mathematical model which considers both turbulent and diversion cross flow between adjacent subchannels. Flow in each subchannel is assumed to be one-dimensional with boundary conditions of inlet temperature or enthalpy, inlet mass velocity and exit pressure. Transients in pressure, temperature, flow rate and heat flux may be specified individually or simultaneously in any combination. Fuel rod heat transport is considered in transient solutions which permits investigation of power pulsing effects on fuel performance.

The ideal gas law was used to determine helium properties with viscosity and thermal conductivity defined by linear functions of temperature. Since the code does not account for the temperature dependence of fuel and cladding properties, appropriate constant material properties were input corresponding to the average expected temperature for each run.

The surface heat transfer coefficient for the converter was estimated by relating the parallel flow tube bundle geometry to the well documented circular tube case. A correlation has been developed by Diessler and Taylor⁽⁹⁾ for this purpose which relates the ratio of Nusselt numbers for the parallel bundle and circular tube to the tube pitch to diameter ratio. For the reference converter design, the ratio of Nusselt numbers is 1.19. For the circular tube Nusselt number, Kays⁽⁷⁾ recommends

$$Nu = 0.022 Pr^{0.6} Re^{0.8}$$

so that the final correlation for the converter becomes:

$$Nu = 0.0262 Pr^{0.6} Re^{0.8}.$$

This correlation also applies to the 4-rod lattice channel. Although the single rod lattice channel is an annulus instead of a parallel flow tube

bundle, the Nusselt number predicted with the tabulated annulus solution corresponds very closely to that of the above correlation for the Reynolds and Prandtl numbers of interest. For this reason, the same heat transfer correlation was applied to all areas of the device. The friction factor was determined from

$$f=0.046 \text{ Re}^{-0.2}$$

which approximates the Karman-Nikuradse equation in the range of Reynolds number expected.⁽¹⁰⁾ This approximation was also applied uniformly to both lattice channels and converter.

STEADY STATE OPERATION

An important consideration in the safe steady state operation of the hybrid device is the maximum centerline temperature of the hottest lattice fuel rod. Two other areas which further restrict the choice of operating conditions are also important. In the converter region, it is necessary to limit the temperature of the lithium to below 900°F ⁽¹¹⁾ in order to reduce tritium diffusion into the coolant and to prevent helium embrittlement of the stainless steel first wall and fuel rod cladding. The second area is the steam power cycle which must operate at a sufficiently high temperature to provide reasonable thermal efficiency. This was shown in the preceding section to require a helium outlet temperature of about 1000°F . Thus, there is both an upper and lower limit to permissible operating temperature.

CONVERTER:

It was found that the above conditions could be met by connecting the converter and lattice coolant flow in series and using the converter as a preheater with a significantly lower inlet temperature. A 15.75 ft section of the device was modeled with the aid of COBRA to test the feasibility of this configuration. For this sample calculation, uranium carbide fuel rods in single-rod channels were used in the lattice and uranium fuel rods in the converter. Helium was supplied to the converter at 730°F and 120 ft/sec. The maximum helium temperature in the converter for this case was 800°F and the temperature of the lithium was below 850°F . The required 1000°F at the lattice outlet was achieved with a maximum uranium carbide rod center temperature of 1960°F . The axial temperature distribution for this postulated section is shown graphically in Figure 9. This configuration may prove highly flexible. By using a longer length or lower helium flow rate (or both), lower converter inlet temperatures would be necessary. This would maintain much lower temperatures in the converter while the lattice outlet could be kept at 1000°F . On the other hand, a shorter length and higher flow rate could permit the utilization of lower helium operating pressures.

A final possibility which might aid in reducing the coolant temperature in the converter is to use the lithium as an additional coolant. Its high heat capacity and thermal conductivity would make it very effective as a heat sink. No flow of lithium was assumed in this study, but since it must be circulated in some way for extraction of tritium, the use of an in-line heat exchanger would appear convenient.

High fuel rod center temperature is not a problem in the converter since the rods are small and the maximum power level is 1/7 that of the lattice. Also, the stainless steel first wall is able to transmit the specified 0.05 MW/m^2 wall flux with an acceptably small temperature difference. However, a much thicker wall will probably be required to withstand the compressive force of the 600 psi helium. An increase in wall thickness will not be critical to heat transmission since the temperature difference will remain relatively small.

LATTICE:

Since the lattice fuel rods operate at the highest power levels, fuel rod center-line temperature is the limiting factor in determining the maximum power of the device. An expression for rod center temperature as a function of the pertinent geometrical and material parameters is as follows:

$$T_c = T_e + Q \left(\frac{V_x}{4C_p m} + \frac{D}{4h} + \frac{D^2}{16k} \right)$$

where

T_c = rod center temperature

T_e = helium inlet temperature

Q = fuel power density (Btu/hr-ft³)

V_x = volume of fuel in a section x feet in length which is in contact with the helium in question

D = rod outside diameter

m = helium mass flow rate (lb/hr)

C_p , h , k = helium specific heat, surface conductance
and fuel thermal conductivity, respectively.

This relation shows the importance of fuel rod diameter in determining the centerline temperature and should be considered when optimizing fuel rod diameter. The thermal performance of several fuel materials was evaluated with this relation for 1000°F coolant temperature and for rod diameters and power densities ranging from 0.25 to 1.25 inches and 25 to 175 MW/m³, respectively. Variable thermal conductivity was also included. The results are shown in Figures 6, 7 and 8. For the 1.0 inch diameter rod of the single-rod channel in the reference design (Figure 2), the centerline temperature of uranium would be unacceptably high (close to the 2070°F melting point) at the peak lattice rod power density of 70 MW/m³. However, centerline temperatures for uranium carbide and uranium oxide would be tolerable. Increasing the helium flow rate would not remedy the problem with the uranium single-rod design since this would only decrease the film resistance and lower the center temperature by 300°F at most, whereas, 800°-900°F is required.

To lower the centerline temperature of the lattice fuel, it was decided to use rods of reduced diameter. A four-rod design as shown in Figure 2 was investigated for this purpose. The higher surface-to-volume ratio would allow any of the three candidate fuel materials to be utilized. The dramatic temperature reduction can be seen in Figures 6, 7 and 8, by comparing the behavior of the original 1.0 inch rods to the 0.5 inch rod diameter of the four-rod model. The cladding surface temperature in this new configuration is about 300°F lower than the single rod case which may also be an important consideration.

FIGURE 6. Rod Center Temperatures, Uranium, 1000°F Coolant

FIGURE 7. Rod Center Temperatures, Uranium Carbide, 1000°F Coolant

FIGURE 8. Rod Center Temperatures, Uranium Oxide, 1000°F Coolant

FIGURE 9. 15.75 Ft Module, Series Flow

PULSED POWER OPERATION

Some of the proposed fusion machines will not produce a steady state fusion reaction. For these designs, power may be generated in a series of pulses with each pulse followed by a "rest period". For the purposes of this study, "period" refers to the total time required for one pulse and one rest period and "duty ratio" is defined as the ratio of the pulse time to the period. The presently conceived pulsed fusion machines (excluding "quasi-DC" devices) may operate with periods on the order of a few seconds and duty ratios of a few hundredths to a few tenths.⁽¹²⁾ A steady state device would have an infinite cycle time and a duty ratio of unity. The pulse characteristics of future fusion devices are only speculative at this time. Nevertheless, an evaluation was made of the effect of pulsed power on the thermal performance of the fusion-fission hybrid concept in an attempt to determine possible future design limits.

The effect of pulsed power is to cause a cyclic temperature variation in all areas of the system. This would probably not be critical except that the temperature variation causes cyclic thermal stresses which may cause fatigue. Also, if the period is long and the duty ratio is small, the power pulse may need to be impractically severe to produce the required overall average power level. The magnitude of the temperature variation depends more on the period than on the duty ratio. The rod is only required to store that amount of energy which will be dissipated to the helium by normal cooling during a cycle and which also must be equal to the power produced at the equivalent steady state power level. Since the cooling rate depends only on the material properties, geometry, and temperature difference, very low duty ratios create no problems if the period is sufficiently short so that the material is not allowed to cool significantly between pulses. Over small temperature differences the rate of temperature decay is approximately constant, and the total change in temperature in a pulse cycle can be shown to be

$$\Delta T = \frac{\bar{P}}{C} \theta \left(1 - \frac{\Delta \theta}{\theta}\right)$$

where ΔT = cyclic change in avg rod temperature
 $C = \frac{\pi D^2 L}{4} \rho C_p$, total rod heat capacity (Btu/°F)
 $\bar{P}$ = time averaged rod power density (Btu/hr-ft³)
 θ = period
 $\Delta\theta$ = pulse time.

Note that $\Delta\theta/\theta$ has been defined as the duty ratio. It can be seen that the cyclic temperature variation increases as the duty ratio decreases (pulse time becomes shorter for constant period) but reaches a finite maximum value for an instantaneous pulse. Thus, the controlling parameter for the temperature variation is the period and not the duty ratio. Figure 10 shows the magnitude of the cyclic temperature variation in the inner wall and the hottest rods of the converter and lattice. This information was derived from COBRA output at standard helium operating conditions (1000°F, 600 psi, 100 ft/sec) with a 1/8 duty ratio (0.125 sec pulse each second at 8 times the device average power). It can be seen that metallic uranium responds to the pulse much more quickly due to its higher conductivity and low specific heat while the uranium oxide and carbide exhibit a 25% lower temperature variation.

The hot lattice rod is definitely the critical area whereas, the converter experiences very little variation. Figures 11, 12 and 13 show the pertinent temperatures as a function of time through one complete pulse cycle. The equilibrium center temperatures (shown as dashed lines) may not correspond to what might be expected from Figures 6-8 since the variable thermal conductivity is not accounted for by COBRA. Note that the rod cladding experiences very little temperature change throughout the cycle, and the heat transfer rate to the helium remains essentially constant. The large mass of graphite in the lattice, and the lithium and steel in the converter, effectively damp out almost all oscillation in helium temperature. If the period is maintained at less than one second, it is expected that the thermal hydraulic problems associated with power pulsing can be tolerated.

HOTTEST FUEL RODS, AS INSTALLED, WITH
INSTANTANEOUS PULSE (DUTY RATIO=1/∞).
FOR FINITE PULSE TIME, MULTIPLY TEMP.
VARIATION FROM GRAPH BY (1.0-DUTY RATIO).

FIGURE 10. Power Pulse Response

HELIUM INLET CONDITIONS

600 PSIA
1000 °F
 5.519×10^4 LB/HR-FT²

POWER CYCLE

PULSE 0.125 SEC. @ $8 \times P_{AVG.}$
PERIOD 1.0 SEC.
 P_{AVG} 0.75 MW/m³

FIGURE 11. Converter Fuel Rod Temperatures

INLET CONDITIONS

600 PSIA
1000 °F.
 5.519×10^{-4} LBm/HR-FT²

POWER CYCLE

PULSE 0.125 SEC. @ $8 \times P_{AVG}$
PERIOD 1.0 SEC.
 P_{AVG} 0.75 MW/m³

FIGURE 12. Lattice Single Rod Model Hot Channel Temperatures

INLET CONDITIONS

600 PSIA
1000 °F
 5.519×10^{-4} LB/HR-FT²

POWER CYCLE

PULSE 0.125 SEC @ $8 \times P_{AVG}$
PERIOD 1.0 SEC
 P_{AVG} 0.75 MW m³

FIGURE 13. Lattice Four Rod Model Hot Channel Temperatures

CONCLUSIONS

I. RESTRICTIONS:

- a) Uranium metal cannot be utilized in the lattice with 1-inch rod diameter fuel.
- b) Helium outlet temperature should be above 1000°F for reasonable thermal efficiency in steam cycle operation. Lattice channels must be orificed.
- c) The vacuum wall should be thick enough to withstand 600 psi compressive pressure from the helium, however pressures could be lowered somewhat without sacrificing thermal hydraulic performance.
- d) Pulse period should be limited to under 2 seconds. No restriction on duty ratio.

II. RECOMMENDATIONS:

- a) Use four-rod channel design in the lattice for lower rod temperatures and more efficient cooling.
- b) Use helium operating conditions 600 psi and 700-1000°F.
- c) Connect the converter and lattice in series to use the converter as a preheater for the lattice.
- d) Utilize uranium carbide for lattice rods for best factor of safety.

III. OPTIONS:

- a) Use uranium metal in the converter and cooler lattice positions and use uranium carbide in hot lattice rod positions if neutronicly feasible.
- b) Provide additional cooling with the converter lithium.
- c) Increase helium velocities up to 500 ft/sec, if necessary.
- d) Increase power level up to 2 MW/m³ (if UC used in lattice).

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