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COMPONENT OPTIMIZATIONS FOR NUCLEAR-POWERED CLOSED-CYCLE GAS TURBINE POWER PLANTS

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ABSTRACT

Nuclear-powered, closed-cycle gas turbine power plants with directly-cooled reactor cores for relatively small power outputs are examined. The results of preliminary economic optimizations for the various components comprising such plants are presented. The reactor core is examined, both from the nuclear and the heat exchange viewpoint, and results of regenerator and turbomachine optimizations are discussed.

1.0 INTRODUCTION

Consideration is given to the possible development of a nuclear-powered, closed-cycle gas turbine employing a reactor concept into which is built the necessary heat transfer for the extraction of heat power. In a previous paper,[1*] efforts were made to evaluate power plant performance of cycles employing a thermal, heterogeneous, gas-cooled reactor, as being developed by the General Atomics Division of General Dynamics Corporation, and a thermal, bismuth-uranium type when conditions are extrapolated to levels permitting effective use of closed-cycle gas turbines. The bismuth-uranium, molten metal-fueled type is a development of the Brookhaven National Laboratories. Presently, the Babcock and Wilcox Company is undertaking under contract with the Atomic Energy Commission the development of a liquid metal-fueled reactor experiment termed "LMFRE."

Some of the advantages which are given in the paper for a closed-cycle gas turbine design are:

1. Increased capacity for a given size and/or weight.
2. Wider range of loads without substantial loss of efficiency, since power may be reduced by reducing pressure level without affecting temperature, pressure ratios or speed.
3. Reduction of size tends to control difficulties resulting from differential temperature extraction.
4. In nuclear-powered, closed-cycle gas turbine systems, it is possible to contain the working fluid in a closed system so that direct extraction of heat from the reactor into a working fluid

*Numbers refer to similarly numbered references in bibliography at end of paper.

can be achieved without release of radioactive gas to the atmosphere.

The influence of working fluid characteristics on the design of closed-cycle gas turbines has been discussed by the present writers,[1] by S. T. Robinson,[2] and by Mr. S. T. Dennison.[3] These authors provide a basis by which a suitable working fluid for a power plant can be selected with the necessary information and data regarding the thermodynamic and heat transfer relationships of the working fluid.

This paper deals with summaries of results and calculations of a nuclear-powered gas turbine system which employs a reactor which is internally cooled in core and reflector by the working fluid which is used directly in a regenerative closed-cycle gas turbine system. Efforts have been made to consider optimum selection of the various components so that the integrated system of components can be evaluated in terms of reactor heat power, maximum temperature in reactor core and reflector, choice of working fluid, selection of pressure ratio, and maximum working fluid pressure.

To illustrate the techniques employed for optimization of these parameters for a reactor coupled with a closed-cycle gas turbine, the authors have selected a reactor heat power output in the range of 5-10 megawatts from which it appears that the economically optimized electrical power outputs are in the range of 1400-2800 kilowatts.

Although several possible types of molten reactor fuels can be employed in reactor designs, it appears that there are immediate possibilities for U-235 or plutonium dissolved in bismuth metal, and operated at high temperatures. Such a concept is being considered by the Brookhaven National Laboratories.

From those data which have been made available to the authors, programs and projects have been conducted to project these data to a practical, small-scale "burner" nuclear power plant, the success of which is contingent upon a vast amount of additional research, development, and engineering. The results of the analyses do indicate that this type of reactor with built-in heat transfer surfaces has a potential for application in the lower reactor heat power range where light weight and high efficiencies in the consumption of nuclear fuels are extremely important.[4]

2.0 DESCRIPTION OF THE FAST REACTOR SYSTEM COUPLED WITH A SIMPLE CLOSED-CYCLE GAS TURBINE

A simplified flow system of a reactor with built-in heat transfer in core and reflector, coupled with a regenerative closed-cycle gas turbine power plant, is given in Figure 1. In general, the flow system considered can be described as follows.

The working fluid, which may be helium, carbon dioxide, nitrogen or other suitable gas, enters the reactor system from a regenerator. The flow of gas from the regenerator is split between heat transfer surfaces provided in the core and heat transfer provided in the reflector. Heat is transferred from the fissioning mass in the core by conductive and natural convective heat transfer through tubes into the working fluid.

Since peak neutron fluxes occur in the reflector, and since the reflector is considered to be U-238 in which both fast fission of U-238 and fission of plutonium occurs, considerable additional power is generated in the reflector.

U-235 or Pu-239 dissolved in bismuth, as developed by the Brookhaven National Laboratory,[5] have been selected as possible fuels for one type of reactor. The fuel will be in the molten state operating at from 650°C to 800°C. The power level of the reactor is sustained by the continuous addition of molten fuel to the core and the continuous removal of fuel and fission product gases from the reactor core.

The core fuel is contained in a tantalum vessel into which a maximum of heat transfer tubes of the U-tube type is located. Surrounding the core design is a eutectic of U-238 and iron in which the maximum temperature is sustained at 600°C to 650°C by the proper installation of heat transfer areas into the molten reflector. The reflector is again contained in a tantalum vessel with provisions for the continuous addition of uranium, as an iron-eutectic, and the removal of a continuous amount of reflector material as a product for plutonium recovery.

The working fluid leaves the core and reflector of the reactor unit and flows at a suitable temperature and pressure to a high pressure turbine in which it is expanded and from which useful work is obtained to drive the high and low pressure compressor units (Compressor #1 and Compressor #2), which are directly connected shaft-wise to the high pressure turbine. The expanded gas leaves the high pressure turbine and is expanded again in the low pressure turbine to obtain useful work in the form of shaft horsepower, which can be converted to either electrical energy or to mechanical work. The gas leaving the low pressure turbine flows then through a regenerator, exchanging heat with the incoming gas from Compressor #2. The gas flowing from the regenerator enters Cooler #1, where it is cooled to the lowest temperature of the system and is also at the lowest pressure of the system. Compressor #1, driven by the high pressure turbine, compresses the gas to an intermediate pressure. The compressed gas then enters Cooler #1, where it is again cooled to ambient temperatures by a suitable coolant. Compressor #2 compresses the gas to the highest pressure of the system.

The choice of suitable absolute pressure level depends upon the power output. An increase in power will demand increased pressure to obtain the requisite mass flow and heat transfer coefficients. The gas at this maximum pressure flows through the regenerator, exchanging heat with the exhaust from the low pressure turbine. The gas leaving the regenerator is at the temperature where the cycle becomes complete.

3.0 PARAMETRIC STUDIES

The coupling of a fissioning, bismuth, internally gas-cooled reactor to a closed-cycle gas turbine requires that the nuclear parameters be carefully evaluated in terms of heat transfer and heat cycle analysis. Optimum designs for the respective components will be affected by power level, temperature level, choice of working fluid, and choice of working pressure (in the case that limits external to the cycle are set upon this parameter). The optimum will also depend upon the application, since in most cases it will be desirable to optimize the cost, in others the weight and/or size.

Nuclear power plants in the range of outputs from approximately 600 to 5000 shaft horsepower have been considered in such optimization studies. It is assumed that the gaseous working fluid will be used to cool the reactor core directly with possible pre-heat in the reflector. Preliminary results of such studies have been reported in a previous paper by the present authors.[1] As explained therein, the low pressure ratio, highly regenerative cycle was selected as most applicable. Curves were presented for such a cycle delineating the individual effect on thermal efficiencies of a change in efficiency or effectiveness of any one component, and also the effect of a change in temperature or pressure ratio. Thus, a basis was provided for ready estimation of thermal efficiency for such a cycle with any combination of component efficiencies and effectivenesses. Further, an evaluation of the effect on the attainable efficiency of the turbine and compressor components of power level, pressure level, temperatures, and fluid selections was made. The degree of reduction in efficiency due to reduced power level or increased pressure level (i.e., reduced flow path dimensions), and the effect on efficiency and also on machine size of fluid selection was shown.

Using such data, it becomes desirable to consider the economic optimization of the principal heat exchanger components in the context of the overall nuclear power plant. For those cases in which minimum cost is the controlling factor, it would then become possible to select intelligently the most suitable working pressures and cycle temperatures, assuming that the reactor temperature is fixed by considerations of feasibility. Also, an intelligent choice of working fluid for any given conditions of reactor temperature and heat sink temperature at a given power level would be possible unless, of course, the choice were dictated by other factors, such as induced radioactivity.

The principal heat exchanger components with which we must concern ourselves are the nuclear reactor itself and the regenerator. The controlling factors in the two cases are quite different, in that the cost of heat exchange surface within the reactor is considerably greater than the cost of the heat exchange surface for the regenerator. In either case, the general problem is complicated so that certain simplifying assumptions must be made before a practical analysis can be attempted.

A preliminary analysis of this situation has been made for a nuclear power plant wherein a counterflow heat exchanger with passages of approximately 1/16 inches hydraulic diameter was assumed. Utilizing assumed costs for nuclear fuel, heat exchange surface, amortization rates, etc., and the turbomachinery data developed in Reference 1, optimized regenerator effectivenesses were computed as a function of maximum cycle pressure and temperature and power level for air and helium. These results are summarized in Figures 2 through 7. It will be noted that the economically optimum regenerator effectiveness under the assumptions of this study is not primarily a function of power level. However, it is very definitely a function of pressure level, being substantially reduced in all cases for reduced pressure. This reduction is somewhat less for helium than for air. The corresponding economically optimized hot side film coefficients are shown in Figures 8 through 13. The coefficients for the cold side are of similar magnitude. It is noted that these also increase substantially for higher pressures.*

It should be mentioned that the increase of economically justified regenerator effectiveness for higher pressures more or less overcomes the reduced turbomachinery efficiency at high pressure (for a given power level) so that the overall economically justified thermal efficiency is not a strong function of pressure level.

The reactor theory of low power kinetics of circulating-fueled reactors with several groups of delayed neutrons[6] has provided a basis for computing reactor periods of such reactors by finding the eigenvalues of a system of differential equations. Such evaluations are applicable to internally-cooled reactor cores when natural convective heat transfer and fluid motion of fuel are similar to the circulating fuel type discussed by J. A. Fleck.[6] The characteristics of liquid bismuth as a fuel solvent and a heat transport

* The authors are indebted to W. J. Yang, formerly of The University of Michigan research staff, for the calculations on which these data are based.

medium for nuclear reactors in terms of pumping power, metallurgical problems, heat transfer, radioactivity, chemical activity, and nuclear considerations has been evaluated at the Brookhaven National Laboratories.[7] This work has indicated the qualifications of liquid bismuth as a heat transfer medium in nuclear power reactors, and shows characteristics which influence design and operation of reactor systems in which it is used.

In the discussions of optimization of nuclear reactor parameters for adaptation to closed-cycle gas turbines where the size of the core should be of total minimum weight, the authors have endeavored to evaluate the operation of a bismuth-fueled system in the fast range.

An internal heat exchange device as suggested by Robert J. Teitel[4] has been adapted to the specific purpose of employing a gas coolant through channels located in the core. The use of plutonium dissolved in bismuth in the intermediate or fast range also offers promise of increasing specific power and plutonium concentration considerably above that which can be achieved by a liquid metal-fueled solution of uranium and bismuth. The solubilities of uranium, thorium and plutonium in bismuth as a function of temperature are shown in Figure 14. Thus, if the reactor average operating temperature is in the neighborhood of 800°C, it would be possible to vary the concentration from several hundred parts per million to a maximum value of about 18% by weight if plutonium were the fuel. The variation of concentration provides one means of reactor control, and correlation of fuel volume to available heat transfer surface.

Thus, it appears possible to give consideration to a technically feasible nuclear reactor core employing fissile material dissolved in bismuth which is surrounded by a reflector comprised of U-238-iron and is contained in the shell side of a standard U-tube heat exchanger. Calculations have been made, premised upon a reactor vessel which is fabricated from tantalum, consisting of 77 U-tubes, 1/2 inch O.D. by 16 BWG on 1-1/33 inch triangular pitch with a 1-3/4 inch radius (minimum) contained in a 16 inch O.D. by 1/4 inch thick tantalum shell with a nominal tube length of 6 inches. For such a reactor, the heat transfer surface based on outside tube dimensions is 152.2 square feet.

Provisions are made for addition of proper concentrations of fissile atoms dissolved in bismuth so that the power level of the reactor can be maintained constant. Tantalum tubes, tube sheets, and heat exchanger shell are provided. The maximum operating temperature of the bismuth-plutonium fuel is estimated to be 850°C (1562°F). The temperature of the working fluid, helium, entering the reflector is calculated to be 1000°F to maintain the reflector at a temperature below 700°C. The temperature of the working fluid entering the reactor is 1150°F. During normal operation, the temperature of the working fluid leaving the reactor is 1350°F. The economically optimized temperature calculated leaving the regenerator and entering the reactor reflector in the horsepower range from 1000 to 10,000 shaft horsepower lies in the range from 825°F to 880°F. Arrangements of heat transfer area in the reflector to maintain reflector temperatures not in excess of 700°C provide a working fluid temperature entering the core between 950°F and 1050°F. Under such conditions, and for a maximum fuel temperature of 850°C for the core configuration considered, the average temperature of the working fluid leaving the reactor core is calculated to be 1350°F. Table I summarizes some general characteristics of the core design.

3.1 Nuclear Parameters

The reactor concept evaluated consists of an intermediate to fast bismuth-plutonium-fueled core contained in the shell side of a standard tantalum U-tube heat exchanger in which the working fluid (helium) flows through the tubes. The resolution of the nuclear parameters requires that multigroup calculations be conducted to evaluate criticality, flux and flux distribution, power level, etc. Final designs would be dependent upon trial and error machine computations so that adjustment of size and core composition is compatible to heat transfer and cycle performance as a function of flux time.[8]

The calculations conducted have been based upon the following assumptions:[9]

1. Fuel is plutonium dissolved in bismuth over range of concentrations.
2. Average reactor fuel operating temperature is 1500-1600°F.
3. Reflector material is U-238 - Fe at M.P. of 650°C.
4. Published cross-section data have been used.

Table II presents results of lethargy group characteristics per megawatt of reactor power (base reactor).

Nuclear parametric studies in the intermediate and fast energy range are dependent upon establishment of the proper analytic approach with suitable boundary conditions. One approach is to use the age diffusion equations in proper energy groups.[10] The age diffusion equation is thus reduced to a finite set of coupled-ordinary differential equations in which each equation represents the average neutron behavior within a given energy group. In cases where the critical configuration and dimensions are large compared to the mean free path of neutrons, the age diffusion equation can be written as follows:

$$\nabla D \nabla (nv_i) - \Sigma_a nv_i - \frac{\partial q}{\partial u} + S(r_i u) = \frac{\partial n(r_i u)}{\partial t} \qquad (3.1)$$

$$\nabla D_T \nabla (nv_i) - \Sigma_{a_T} nv_T + q_T = \frac{\partial n(r,T)}{\partial t} \qquad (3.2)$$

where

$$D = \frac{1}{3\Sigma_{tr}} = \frac{1}{3\Sigma_s(1 - \bar{\mu}_o)}$$

Σ_{tr} = macroscopic transparent cross-section

$\bar{\mu}_o$ = average cosine of scattering angle

Σ_a = macroscopic absorption cross-section

ζ = average logarithmic energy decrement

u = lethargy

$$nv = \text{neutron flux}$$

$$S(r,u) = \text{source generation term}$$

$$\frac{\partial n(r,u)}{\partial t} = \text{rate of change of neutron density with time}$$

The source of neutrons resulting from fission can be expressed as:

$$S_f(r,u) = \nu \left[\int_{-\infty}^{u_T} \Sigma_f \, nv \, du' + \Sigma_{f_T} nv_T \right] \chi(u) . \qquad (3.3)$$

The source of neutrons resulting from inelastic scattering can be expressed as:

$$S_{is}(r,u) = \int_{-\infty}^{u} \Sigma_{is} \, nv \quad (u' \rightarrow u) du' \qquad (3.4)$$

where

$$\nu = \text{number of neutrons per fission}$$

$$\chi(u) = \text{normalized fission spectra}$$

$\chi(u' \rightarrow u)$ = fraction of neutrons which are inelastically scattered and degraded from u' to u.

If Σ_s and Σ_{tr} can be considered constant over small spatial regions, the equation for steady state reactor operations can be expressed as:

$$\frac{1}{3\zeta \Sigma_f \Sigma_{tr}} \nabla^2 q - \frac{\Sigma_a}{\zeta \Sigma_s} q - \frac{\partial q}{\partial u} + S(r,u) = 0 \qquad (3.5)$$

The above equation can be expressed in integrated form over the limits of the logarithmic energy range, U_i, in the i-th lethargy group as follows:

$$U_i \left(\frac{\nabla^2 q}{3\zeta \Sigma_s \Sigma_{tr}}\right)_{i(avg)} - U_i\left(\frac{\Sigma_a q}{\zeta\Sigma_s}\right)_{i(avg)} - q_i^o + q_i^I + \int_{U_i} S(r,u)du = 0. \tag{3.6}$$

The term, q_i^o, is the degradation of neutrons out of the i-th group, and q_i^I is the degraded group entering the i-th group from the previous higher energy group.

In cases where the slowing down density and macroscopic cross-sections encounter small changes within the energy group, Equation (3.6) can be written in approximate form as products of averages,

$$\left(\frac{U_i}{3\zeta \Sigma_s \Sigma_{tr}} \nabla^2 q\right)_{i(avg)} - \left(\frac{U_i \Sigma_{aq}}{\zeta \Sigma_s}\right)_{i(avg)} - q_i^o + q_i^I + \int_{u_i} S(r,u)du = 0 \tag{3.7}$$

Neutrons produced from fission and resulting from inelastic scattering can be written as

$$\int_{U_i} S(r,u)du = \left[\sum_{j=1}^{n}\left(\frac{U_i\Sigma_f}{\zeta \Sigma_s}\right)_{j(avg)} X_{f_{ij}} + \left(\frac{U \Sigma_{is}}{\zeta \Sigma_s}\right)_{j(avg)} X_{(is)(jj)}\right] q_i(avg) \tag{3.8}$$

We can equate the slowing down density within the i-th group to the slowing down density of the previous group:

$$q_i^I = q_{i-1}^o$$

Solution of Equation (3.8) is possible if the relationship of q_{avg} and q_o is linear:

$$q_i^o = w_i (q_i)_{avg}$$

where

w_i = an appropriate constant for each group

Normally, the coefficients, w_i, will be less than unity in those groups and regions which do not contain source terms. By substitution of Equation (3.8) into Equation (3.7) and simplifying, we have:

$$\nabla^2 (q_i)_{avg} + \sum_{j=1}^{n} c_{ij} (q)_{j(avg)} = 0 \qquad (3.9)$$

where

$$c_{ij} = \frac{1}{\left(\frac{U}{3\zeta \Sigma_s \Sigma_{tr}}\right)_{avg}} \left\{ w_{i-1} \delta_{i-1} - \left[\left(\frac{U \Sigma_a}{\zeta \Sigma_s} \right)_{i(avg)} + w_i \right] s_i + \left(\frac{U \Sigma_t}{\zeta \Sigma_s} \right)_{j(avg)} \chi_{f_{ij}} + \left(\frac{U \Sigma_{is}}{\zeta \Sigma_s} \right)_{j(avg)} \chi_{is(ij)} \right\}$$

when

$$\delta_k = \begin{cases} 1 & \text{if } j = k \\ 0 & \text{if } j \neq k \end{cases}$$

$\chi_{f_{ij}}$ = fraction of neutrons occurring in j-th group resulting from neutrons born in the i-th group.

$\chi_{(is)ij}$ = fraction of neutrons inelastically scattered in the j-th group which are released in the i-th group.

Recalling that the homogeneous set of differential equations are of the form

$$\nabla^2 (q_i)_{avg} = -B^2 (q_i)_{avg}, \tag{3.10}$$

substitution of Equation (3.10) into Equation (3.9) gives:

$$\sum_{j=1}^{n} (B^2 \partial_i - C_{ij})(q_i)_{avg} = 0 \tag{3.11}$$

This equation can be expanded in matrix form:

$$\begin{pmatrix} B^2 - C_{11} & -C_{12} & -C_{13} & \cdots & C_{1n} & q_{1(avg)} \\ -C_{21} & B^2 - C_{22} & -C_{23} & \cdots & C_{2n} & q_2 \\ -C_{31} & -C_{32} & B^2 - C_{33} & \cdots & -C_{3n} & q_3 \\ \vdots & \vdots & \vdots & & & \vdots \\ -C_{n1} & -C_{n2} & -C_{n3} & \cdots & B^2 - C_{nn} & q_{n(avg)} \end{pmatrix} = 0$$

(3.12)

Equation (3.11) can be satisfied if the determinant of the coefficient matrix is zero. The determinant is a system of polynomials of degree n.

3.2 Reactor Core Heat Transfer Parameters

It is to be expected that the economic optimization of the reactor core as a heat exchanger will show optimum heat transfer coefficients differing considerably from those of the regenerator because of the interplay of the nuclear parameters--fissile material inventory, costs of the core and shielding, etc.--with the heat transfer parameters. A general investigation of the situation would be extremely difficult because of the many types of reactor cores to be considered and the unknown magnitude of many of the most significant cost factors. However, a preliminary investigation has been made for a particular configuration in order to at least determine the significant trends.*

The configuration investigated is similar to that described in this paper in that heat is removed from a liquid-metal-fueled core by the passage of coolant gas through tubes in the core. This coolant gas is the working fluid for a closed-cycle gas turbine. It differs in that a graphite cylindrical core (height equal to diameter), fitted with vertical wells for the liquid metal fuel and holes for passage of the coolant gas, is considered. The nuclear calculations are based upon the work of J. Chernick,[8] and very rough corrections are made for the presence of steel coolant tube liners. While no attempt was made to work out the detailed features of the construction, it was felt prudent to include allowance for metallic liners of some sort.

*The authors are indebted to Y. Hwang, of the University of Michigan research staff, for these calculations.

It was assumed that the temperature limitation for the overall power plant would be within the reactor rather than the turbine equipment. It then becomes a problem of determining the economically justifiable degree of approach between the working fluid temperature at turbine inlet and the inside wall temperature of the coolant tubes. A closer approach requires additional heat exchange surface area within the reactor, but increases overall thermal efficiency (for a fixed reactor temperature). It is thus a balance between capital cost and operating cost.

Under the assumptions described above, calculations have been conducted for air for power outputs ranging between 600 and 60,000 horsepower and maximum cycle pressures ranging between 400 and 1000 psia. A reactor temperature limitation of 1700°F for the inside wall of the coolant tubes was assumed. The gas inlet temperature to the reactor was determined from the regenerator optimizations previously described in this report. The outlet temperature was economically optimized considering amortized capital costs of the reactor, fuel inventory costs, and pumping power cost. The reactor cost estimation was necessarily very rough. Corrections for increased gas pressure were applied. To date, the reactor investigation has only considered air (or nitrogen) as the working fluid. It is hoped that it may be extended to helium also.

The results for air are shown in Figures 15 through 19. The optimized turbine inlet temperature, log mean temperature difference, gas side film coefficient, the temperature increase through the reactor, and the core diameter are plotted as functions of power output and pressure level. Table III shows the economically justified pressure drop in absolute values, and as a portion of the pressure level, across the reactor for the various cases. It is noted that this is less than 3% even for 60,000 horsepower at the minimum pressure considered and becomes practically insignificant for the smaller powers and higher pressures.

It is noted that the approach of the gas temperature to the coolant tube inner wall temperature is closest for the low powers and high pressures. Also, of course, the log mean temperature difference is minimized in these cases. The rather peculiar shape of these curves may be explained in the following manner. For the low power units, the size of the core, and hence, the heat transfer area available, is controlled by considerations of critical mass and uranium inventory. In other words, the reactor as a heat exchanger is oversized so that the pressure and temperature differentials are very small. As the power level is increased or the pressure level decreased, the core size is still controlled by the nuclear parameters so that the gas pressure and temperature drops increase. As yet, however, the cost of fissionable material burn-up (i.e., the thermal efficiency effect) is small. However, as the power is further increased, this burn-up cost begins to control, so that it becomes desirable to increase the reactor size at some capital cost to improve the overall thermal efficiency by reducing the required temperature differentials and pressure drops.

The results which have been listed apply directly only to a very specialized example. However, it is felt that the trends exhibited are illustrative and do have more general application.

4.0 RESULTS OF PARAMETRIC STUDIES AS APPLIED TO LIQUID-METAL-FUELED FAST REACTOR

4.1 General

The overall thermal efficiencies with optimized turbomachinery component efficiencies at various temperature and pressure levels can be considered in the range of energy outputs equivalent to 1000 to 3000 BHP. Such curves are shown in Reference 1. Thermal efficiency versus pressure ratio at 1200°F and 1500°F are also plotted for component efficiencies varied individually.

As a result of the calculations on which these curves are based, it becomes possible to evaluate the range of optimum temperatures for the working fluid entering the reactor for a fixed working fluid outlet temperature from the reactor.

Figure 20 is a plot of the optimized temperature for the working fluid (helium) leaving the regenerator and entering the reactor versus horsepower output when the outlet working fluid temperature is fixed at 1500°F for various pressures.

Figure 21 presents a similar plot when the working fluid (helium) leaving the reactor is 1200°F.

Thus, within limits of materials of construction, it is possible to establish specific temperature differences for the working fluid flowing through a reactor core wherein the heat transfer area is contained.

The heat transfer problem to be considered in a molten metal-fueled core into which heat transfer area is incorporated is a complex problem and warrants considerable research and development before final correlations between the source generation term, critical configuration, fluid mechanics and heat transfer can be achieved.

An optimum design of a reactor configuration capable of heat-power generation in terms of heat transfer area and maximum temperature limits becomes a function of power level, flux distribution, maximum allowable temperatures, thermal stress, arrangement of coolant passages, materials of construction. A specific idealized example was described in the last section.

4.2 Heat Transfer from Molten Fuel to Working Fluid

The calculations to date give results where it is possible to consider the heat transfer from a mass of molten fissioning fuel to a working fluid. The reactor engineer can select optimum conditions of temperature, pressure ratio, thermal efficiency for a nuclear power plant of a given power output. From such conditions, it becomes possible to consider the heat transfer characteristics of a surface incorporated into the core geometry of a molten metal-fueled reactor.

From heat engine cycle analyses and the rapid advancements of technologies in materials for gas turbine components, it appears that the maximum temperature of the system is limited by the container material for the molten fuel. Investigations indicate that it is possible to achieve a coolant tube wall temperature of 1700°F, and this value was used in the reactor optimization described in the last section. Thus, a working fluid temperature for reactor outlet conditions at 1500°F appears within reach.

For such conditions and a reactor working fluid pressure of 1000 psig when delivering 3000 BHP in output power, the optimum inlet temperature of the working fluid entering the reactor core is 875°F. If we select a $\Delta p/p$ as .02, which appears reasonable for the system from the previously described calculations, the economically justified thermal efficiency expected would be 28%. Thus, the heat power level average for the reactor would be

$$\frac{(3000)(7457)}{.27} = 8.3 \text{ megawatts} .$$

Thus, to allow for thermal losses of heat to structural materials, shielding, etc., it would appear that a heat power generation between 8.5 and 9.0 megawatts would suffice. Premised upon multigroup calculations by Chernick, the size and geometry of reactor configuration is dependent upon heat transfer surface and arrangements in the reactor core.

4.3 Determination of Optimized Core Diameter

As previously mentioned, summaries of calculations based upon Chernick's work, showing relationships of core diameter to output power, are shown in Figure 19. A maximum core diameter is obtained at 2000 horsepower. A minimum appears at about 10,000 BHP. These calculations have been premised upon air as the coolant but relation curves can be expected for other working fluids.

4.4 Selection of Maximum Turbine Temperature (Reactor Outlet Temperature) Versus Power Output

Figure 15 summarizes calculations wherein the gas turbine inlet temperature is determined as a function of output horsepower. A minimum appears in the range of 20,000 to 30,000 BHP for the range of working fluid pressures from 400 to 1000 psia.

4.5 Determination of Optimum Log Mean Temperature Difference

Based upon a tube wall temperature of 1700°F, calculation summaries for log mean temperature difference (average) versus plant output horsepower are shown in Figure 16. Maxima appear at the range of horsepower outputs for various pressures in the range of 20,000 to 40,000 BHP.

4.6 Determination of Film Coefficient for Working Fluid at Constant Reactor Tube Wall Temperature

For an optimized core geometry at constant tube wall temperature, calculations showing the effect of gas coolant film coefficients are shown in Figure 17. Calculations were made for various gas flow rates for a fixed heat transfer geometry in terms of total average working fluid pressure in the reactor core. It is to be noted that at low reactor power levels, the film coefficient appears to remain constant at about 40 for all pressures. As power is increased, the film coefficient approaches a maximum of 560 at 1000 psi, 370 at 700 psi, and 300 at 400 psi. Note that these values are in the power output range of about 60,000 BHP. This fact arises from the situation wherein the uranium-bismuth inventory and critical dimensions control for low power outputs but become a much less controlling factor for higher power outputs.

4.7 Temperature Rise of Working Fluid Through Reactor

Summaries of calculations of the temperature rise of gas for various pressuresas a function of BHP are shown in Figure 18. When the pressure is 1000 psia, a maximum rise occurs at 7000-9000 BHP and a minimum rise occurs between 20,000 and 30,000 BHP.

Although the calculations presented are premised upon air as the working fluid, similar relationships are expected for other working fluids.

TABLE I

GENERAL CHARACTERISTICS OF BISMUTH-PLUTONIUM FUEL CORE DESIGN WITH URANIUM-IRON REFLECTOR

Reactor Heat Power	8.8 MW
Core Fuel	Plutonium dissolved in bismuth by weight
Addition of Fuel	Continuous or semi-continuous
Fission Product Removal	Continuous removal of fission product gases, about 95 of total fission product
Maximum Temperature of Reactor Fuel	850°C
Temperature of Working Fluid Entering Reflector	825°F - 880°F
Temperature of Working Fluid Entering Core	950°F - 1050°F
Temperature of Working Fluid Leaving Reactor Core	1350°F
Reactor Construction Material	Tantalum
Maximum Allowable Surface Temperature	900°C
Reactor Type	Shell and tube unit with U_2 bundle in core
Fuel on Outside of Tubes	Working fluid through tubes
Heat Transfer Surface Available	152.2 square feet in core
Reflector	U^{238}-iron eutectic, canned and pierced with tantalum tubes
Core Dimensions	16" O.D. by 6" nominal tube length
Reflector Dimension	8" annular thickness surrounding tantalum core

TABLE II

LETHARGY GROUP CHARACTERISTICS PER MEGAWATT OF REACTOR POWER (BASE REACTOR)

Lethargy Group	Energy (ev)	Space Avg. Relative Slwng. Dwn. Density (Grp. Avg.)	Space Average Fissioning Rate per cm^3	Space Average Neutron Flux	Space Average Group Neutron Flux per Unit Lethargy
1	$1.0 - .15 \times 10^7$	1	5.04×10^9	5.96×10^{12}	3.14×10^{12}
2	$1.5 \times .15 \times 10^6$	.979	4.06×10^9	4.23×10^{12}	1.84×10^{12}
3	$1.5 \rightarrow .15 \times 10^5$	.796	2.83×10^9	2.13×10^{12}	9.25×10^{11}
4	$1.5 \rightarrow .15 \times 10^4$	.512	4.33×10^9	1.34×10^{11}	5.83×10^{11}
5	$1500 \rightarrow 150$	.224	5.83×10^9	5.70×10^{11}	2.48×10^{11}
6	$150 \rightarrow 24$	.0383	2.27×10^9	8.10×10^{10}	4.39×10^{10}
	24		4.91×10^8		

Note: The space average of total neutron flux within the six lethargy groups per megawatt of reactor power for a base reactor is:

$$\phi = 1.43 \times 10^{13} \text{ nv}$$

TABLE III

PRESSURE DROP VS. PLANT OUTPUT FOR OPTIMIZED BISMUTH-URANIUM GRAPHITE MODERATED REACTOR

$P_{max} \rightarrow$	1000 psia		700 psia		400 psia	
Output HP $\downarrow$	ΔP psia	$\frac{\Delta P}{P_{max}}$	ΔP psia	$\frac{\Delta P}{P_{max}}$	ΔP psia	$\frac{\Delta P}{P_{max}}$
60,000 HP	12.8	0.0128	9.4	0.0134	11.4	0.0285
20,000 HP	3.8	0.0038	5.22	0.0075	5.3	0.0132
6,000 HP	0.38	0.00038	0.583	0.00083	0.739	0.00185
2,000 HP	0.059	0.00006	0.088	0.00013	0.154	0.000385
600 HP	0.0093	0.00001	0.0131	0.00002	0.0224	0.000056

BIBLIOGRAPHY

1. Hammitt, F. G. and Ohlgren, H. A., "Nuclear-Powered Gas Turbines for Light Weight Power Plants," Paper No. 57-NESC-79, ASME, EJC Congress, February, 1957.

2. Robinson, S. T., "Influence of Working Fluid Characteristics on the Design of Closed-Cycle Gas Turbines," ASME, 1957 National Conference of the Gas Turbine Power Division, March, 1957.

3. Dennison, S. T., Discussion of Reference 2, private communication, February, 1957.

4. Teitel, R. J., "An Internally Cooled Liquid Metal-Fueled Reactor Design," ANS, Preprint 34, 1955.

5. Teitel, R. J., Gurinsky, D. H., and Bryner, J. S., "Liquid-Metal Fuels and Liquid-Metal Breeder Blankets," Chemical Engineering Progress Symposium (Series #13), Nuclear Engineering, Part III, Volume 50, 1954.

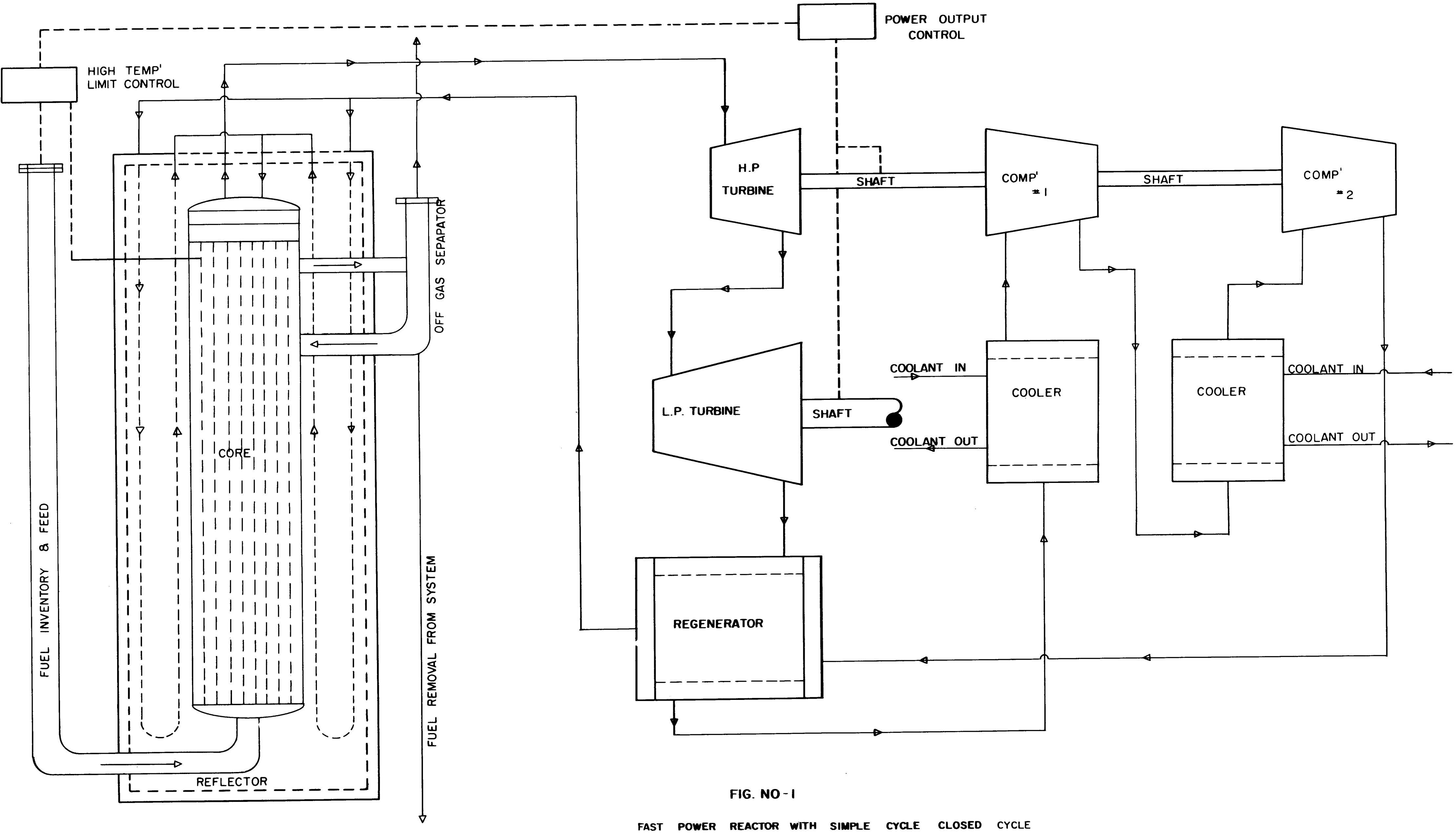
6. Fleck, J. A., "Theory of Low Power Kinetics of Circulating Fueled Reactors with Several Groups of Delayed Neutrons," BNL-334, April, 1955.

7. Dwyer, O. E., Gurinsky, H., and Weeks, J. R., "Liquid Bismuth as a Fuel Solvent and Heat Transport Medium for Nuclear Reactors," ASME, Nuclear Engineering and Science Congress, December, 1955.

8. Chernick, J., "Small Liquid Metal-Fueled Reactor Systems," Nuclear Science and Engineering, 1, pp. 135-155 (1956).

9. Mozer, B., "Evaluation of the Univac Data for a Liquid Metal Fuel Reactor," BNL-2395, Brookhaven National Laboratories, July 28, 1954.

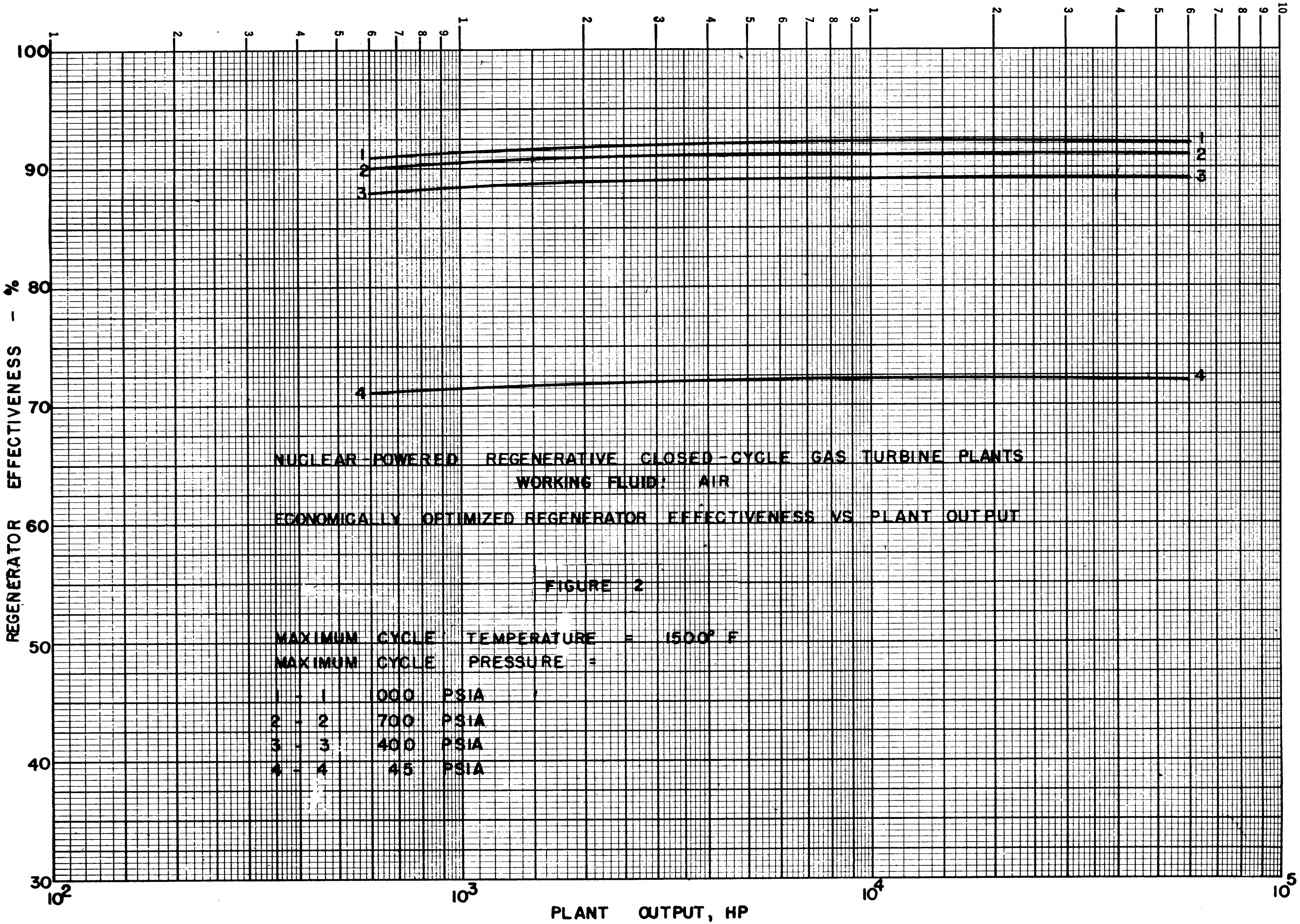
10. Ehrlich, R and Hurwitz, H., "Multigroup Methods of Neutron Diffusion Problems," Nucleonics, 12, 223 (1954).



REACTOR UNIT WITH HEAT TRANSFER BUILT INTO CORE & REFLECTOR

FIG. NO-1

FAST POWER REACTOR WITH SIMPLE CYCLE CLOSED CYCLE GAS TURBINE POWER PLANT.


100
90
80
70
60
50
40
30
REGENERATOR EFFECTIVENESS - %
10^2
10^3
10^4
10^5
PLANT OUTPUT, HP
1
2
3
4
NUCLEAR POWERED REGENERATIVE CLOSED-CYCLE GAS TURBINE PLANTS
WORKING FLUID: AIR
ECONOMICALLY OPTIMIZED REGENERATOR EFFECTIVENESS VS PLANT OUTPUT
MAXIMUM CYCLE TEMPERATURE = 1500° F
MAXIMUM CYCLE PRESSURE =
1 - 1 1000 PSIA
2 - 2 700 PSIA
3 - 3 400 PSIA
4 - 4 45 PSIA
FIGURE 2

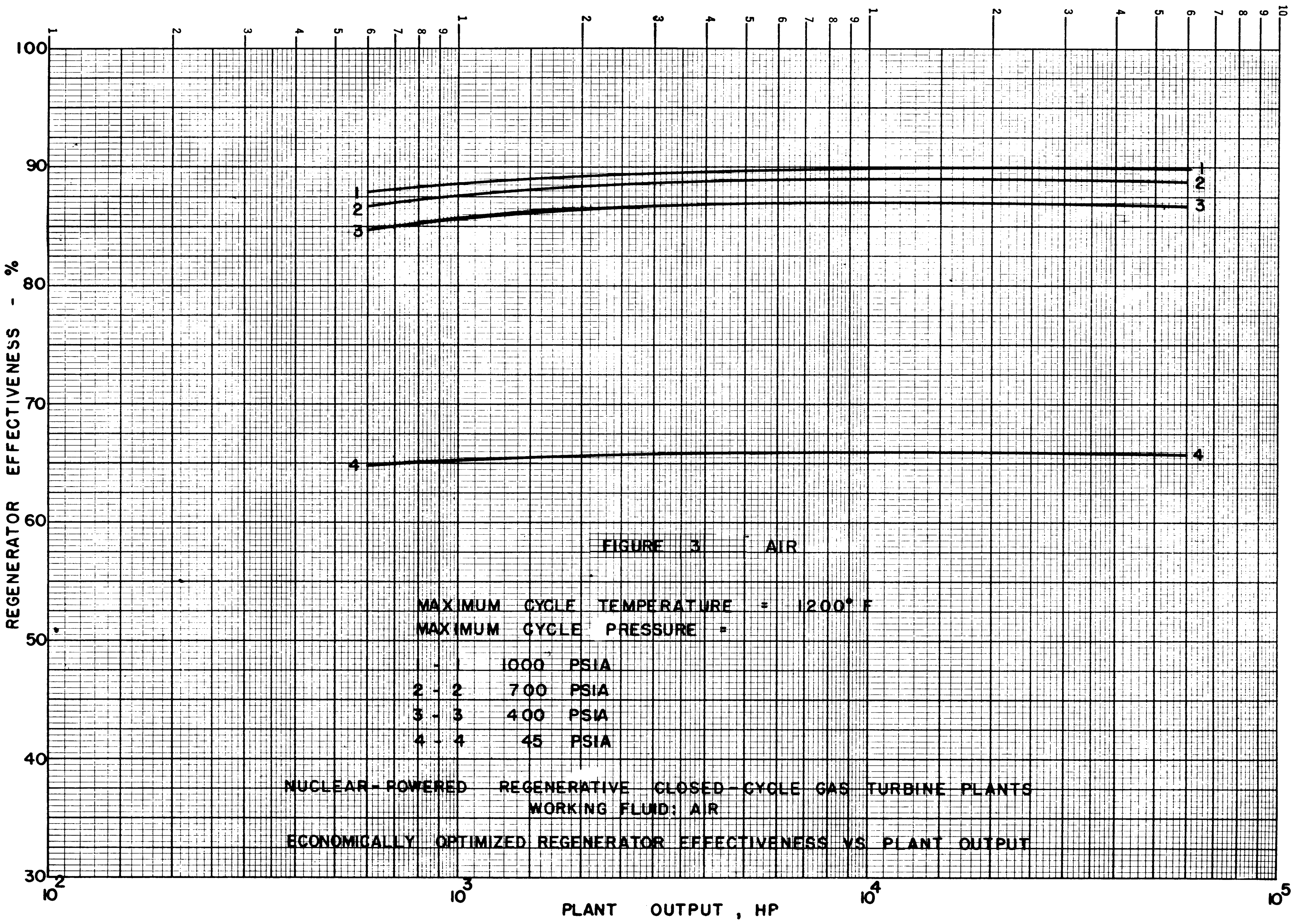
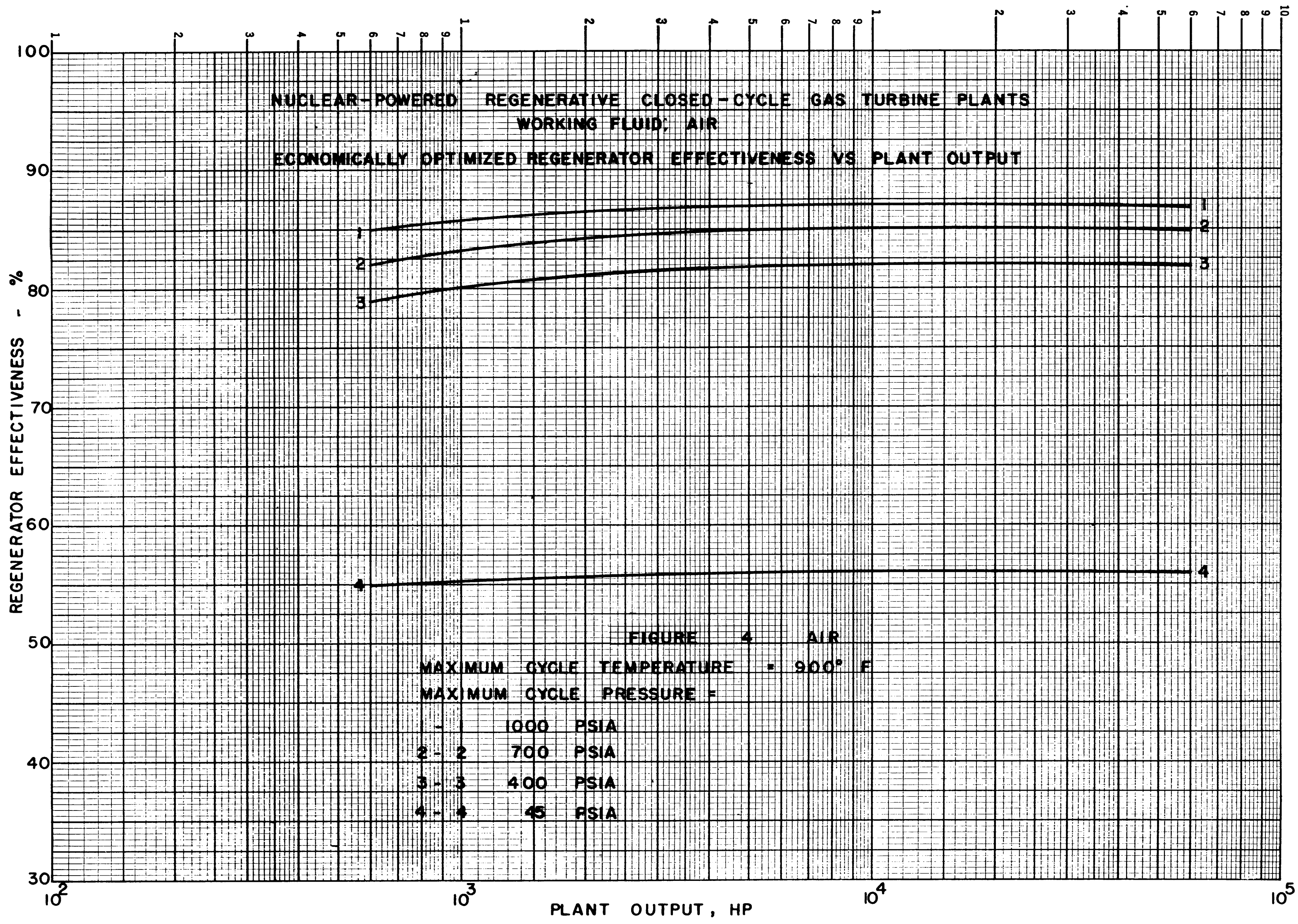
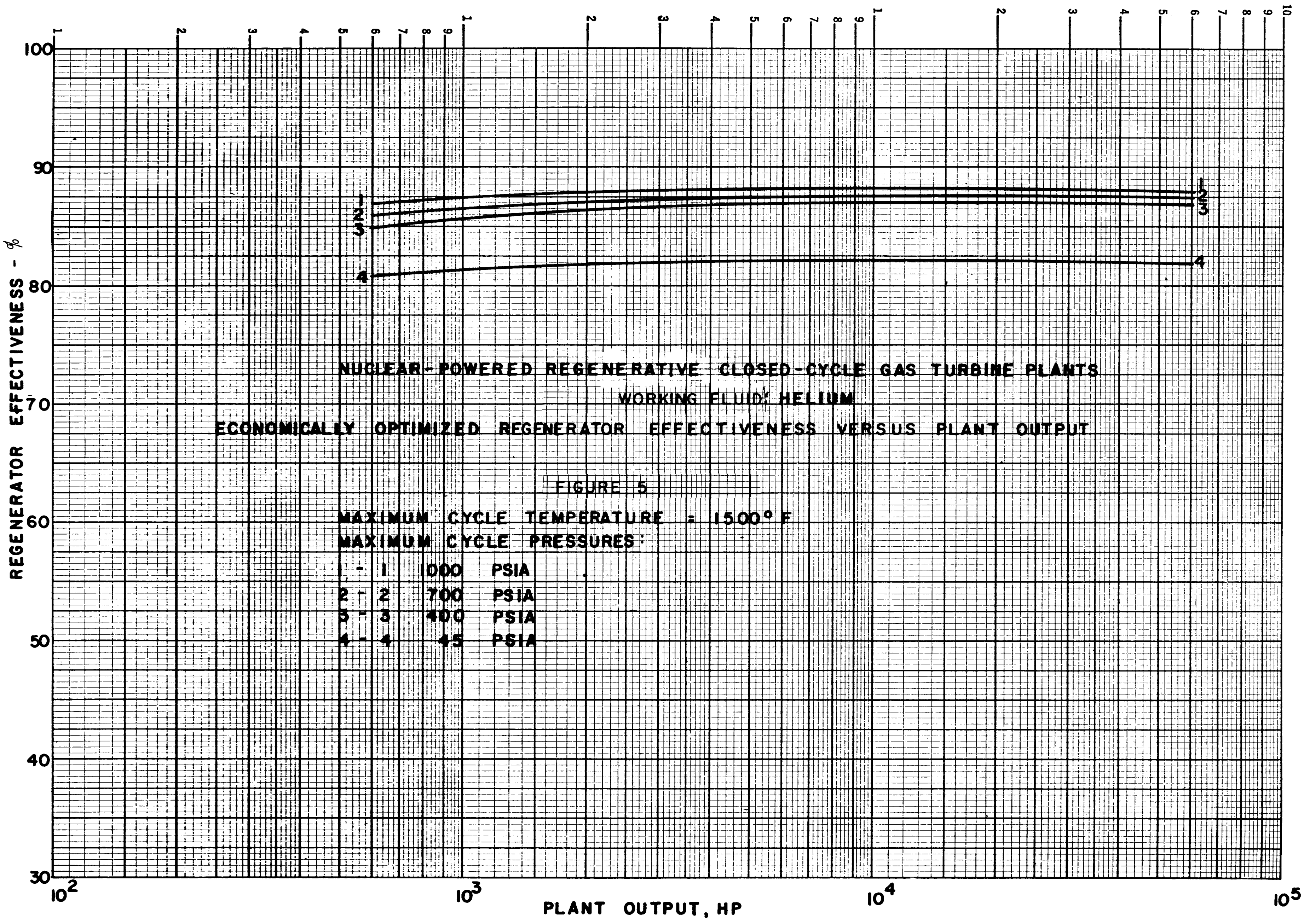


FIGURE 3 AIR
MAXIMUM CYCLE TEMPERATURE = 1200° F
MAXIMUM CYCLE PRESSURE =
1 - 1 1000 PSIA
2 - 2 700 PSIA
3 - 3 400 PSIA
4 - 4 45 PSIA
NUCLEAR-POWERED REGENERATIVE CLOSED-CYCLE GAS TURBINE PLANTS
WORKING FLUID: AIR
ECONOMICALLY OPTIMIZED REGENERATOR EFFECTIVENESS VS PLANT OUTPUT
REGENERATOR EFFECTIVENESS - %
PLANT OUTPUT , HP



NUCLEAR-POWERED REGENERATIVE CLOSED-CYCLE GAS TURBINE PLANTS
WORKING FLUID; AIR
ECONOMICALLY OPTIMIZED REGENERATOR EFFECTIVENESS VS PLANT OUTPUT
FIGURE 4 AIR
MAXIMUM CYCLE TEMPERATURE = 900° F
MAXIMUM CYCLE PRESSURE =
1 - 1 1000 PSIA
2 - 2 700 PSIA
3 - 3 400 PSIA
4 - 4 45 PSIA
REGENERATOR EFFECTIVENESS - %
PLANT OUTPUT, HP
100
90
80
70
60
50
40
30
10^2
10^3
10^4
10^5
1
2
3
4


NUCLEAR-POWERED REGENERATIVE CLOSED-CYCLE GAS TURBINE PLANTS
WORKING FLUID: HELIUM
ECONOMICALLY OPTIMIZED REGENERATOR EFFECTIVENESS VERSUS PLANT OUTPUT
FIGURE 5
MAXIMUM CYCLE TEMPERATURE = 1500° F
MAXIMUM CYCLE PRESSURES:
1 - 1 1000 PSIA
2 - 2 700 PSIA
3 - 3 400 PSIA
4 - 4 45 PSIA
REGENERATOR EFFECTIVENESS - %
100
90
80
70
60
50
40
30
10^2
10^3
10^4
10^5
PLANT OUTPUT, HP

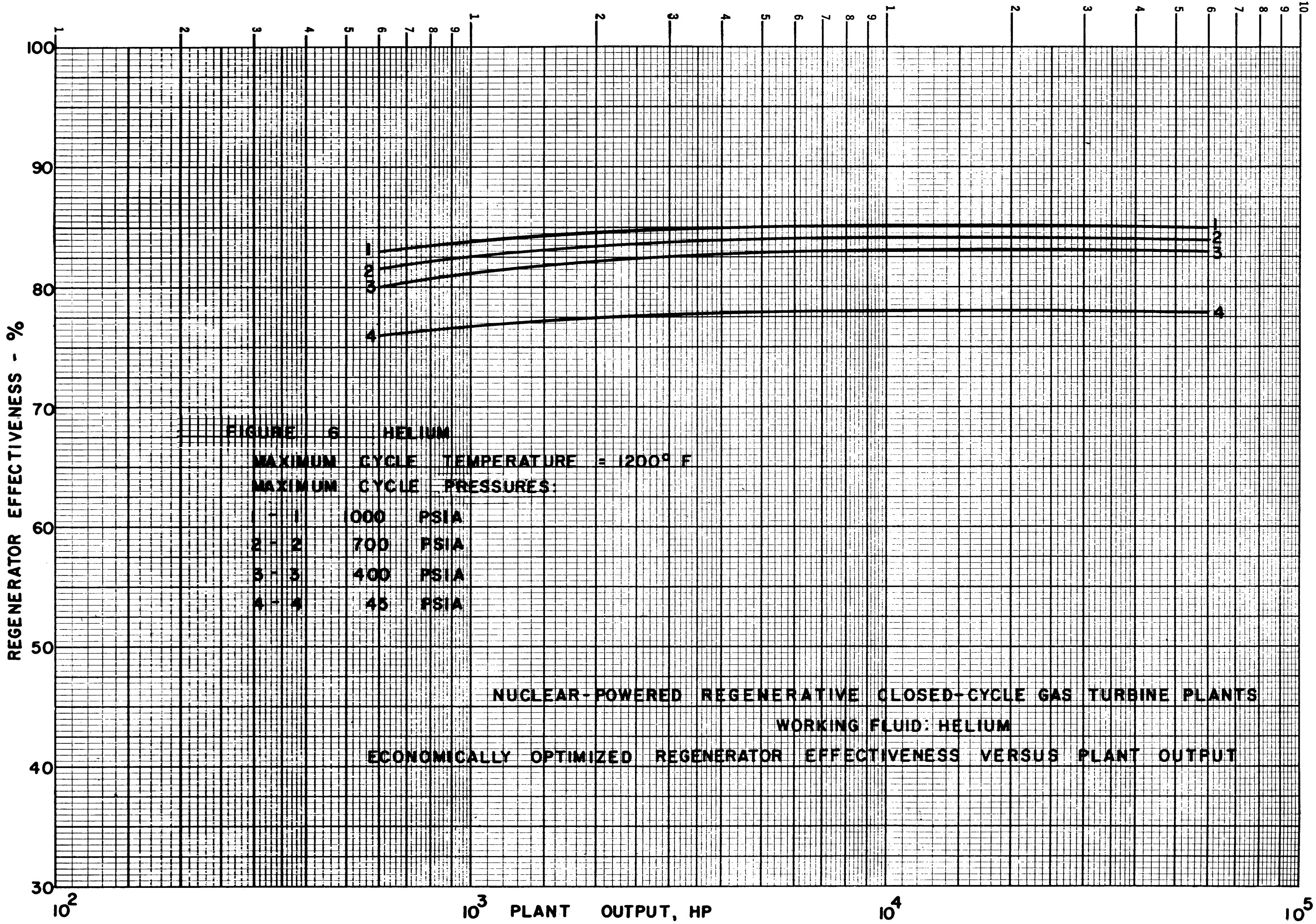


FIGURE 6 HELIUM
MAXIMUM CYCLE TEMPERATURE = 1200° F
MAXIMUM CYCLE PRESSURES:
1 - 1 1000 PSIA
2 - 2 700 PSIA
3 - 3 400 PSIA
4 - 4 45 PSIA
NUCLEAR-POWERED REGENERATIVE CLOSED-CYCLE GAS TURBINE PLANTS
WORKING FLUID: HELIUM
ECONOMICALLY OPTIMIZED REGENERATOR EFFECTIVENESS VERSUS PLANT OUTPUT
REGENERATOR EFFECTIVENESS - %
PLANT OUTPUT, HP
100
90
80
70
60
50
40
30
10^2
10^3
10^4
10^5
1
2
3
4

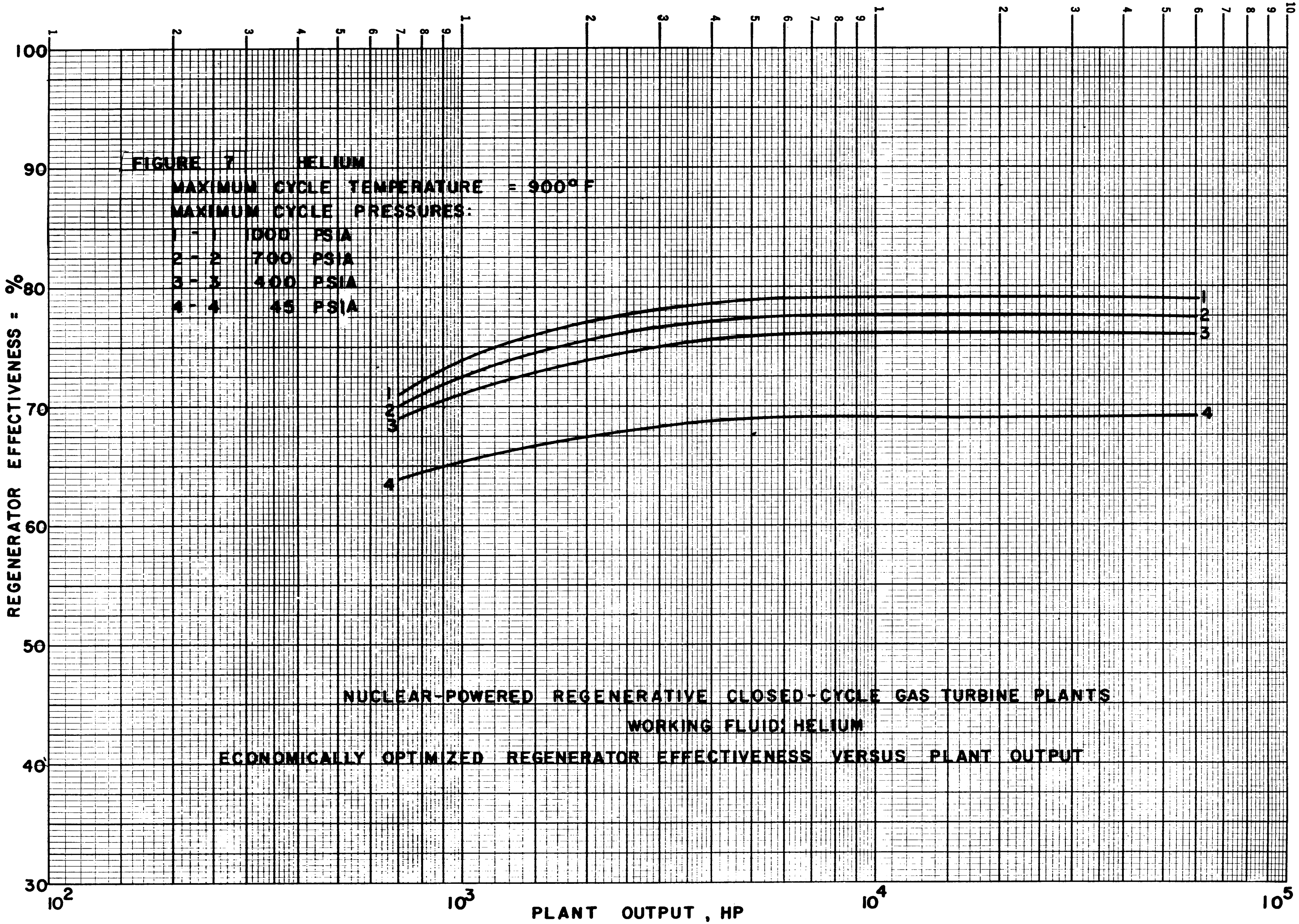
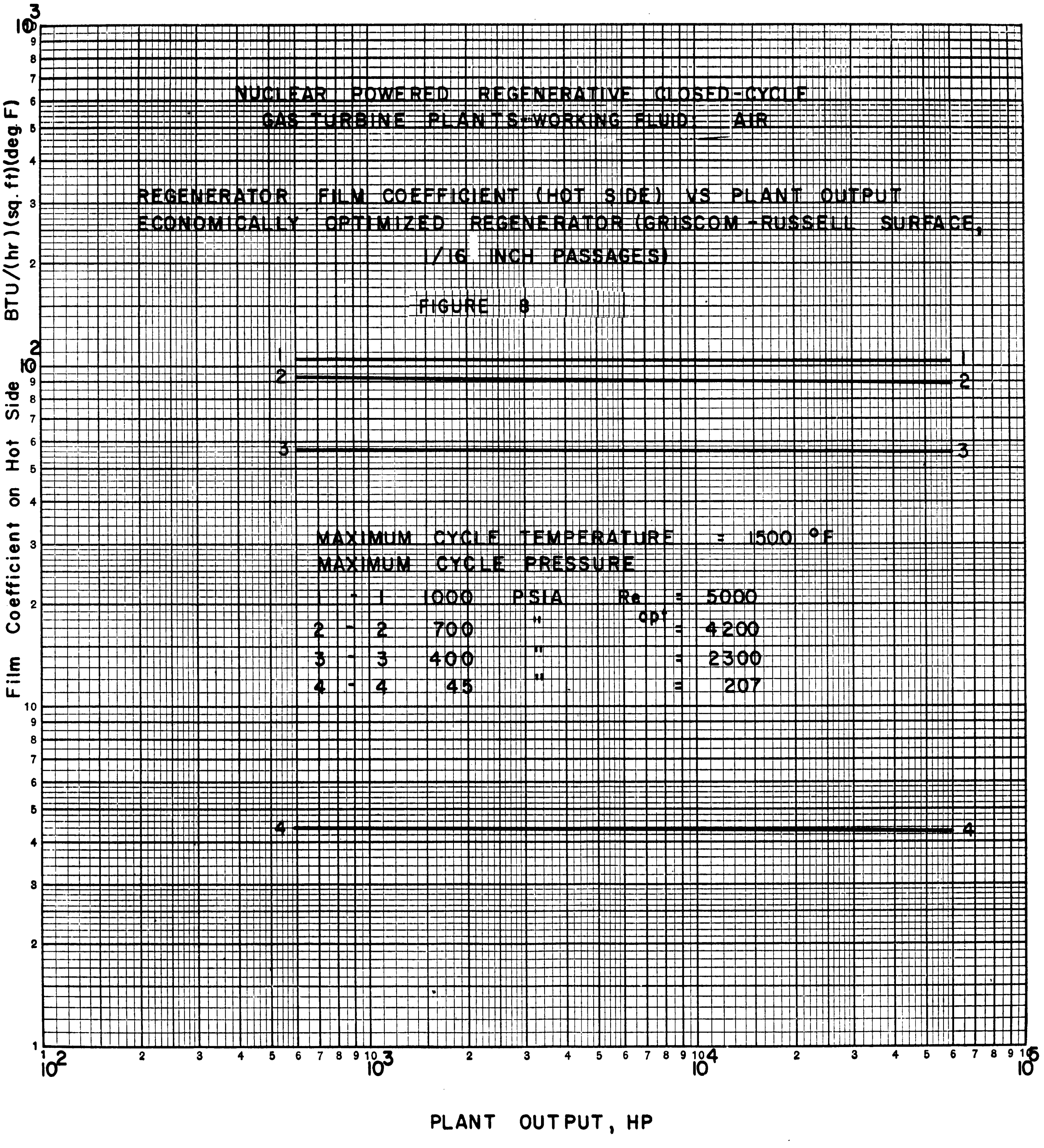
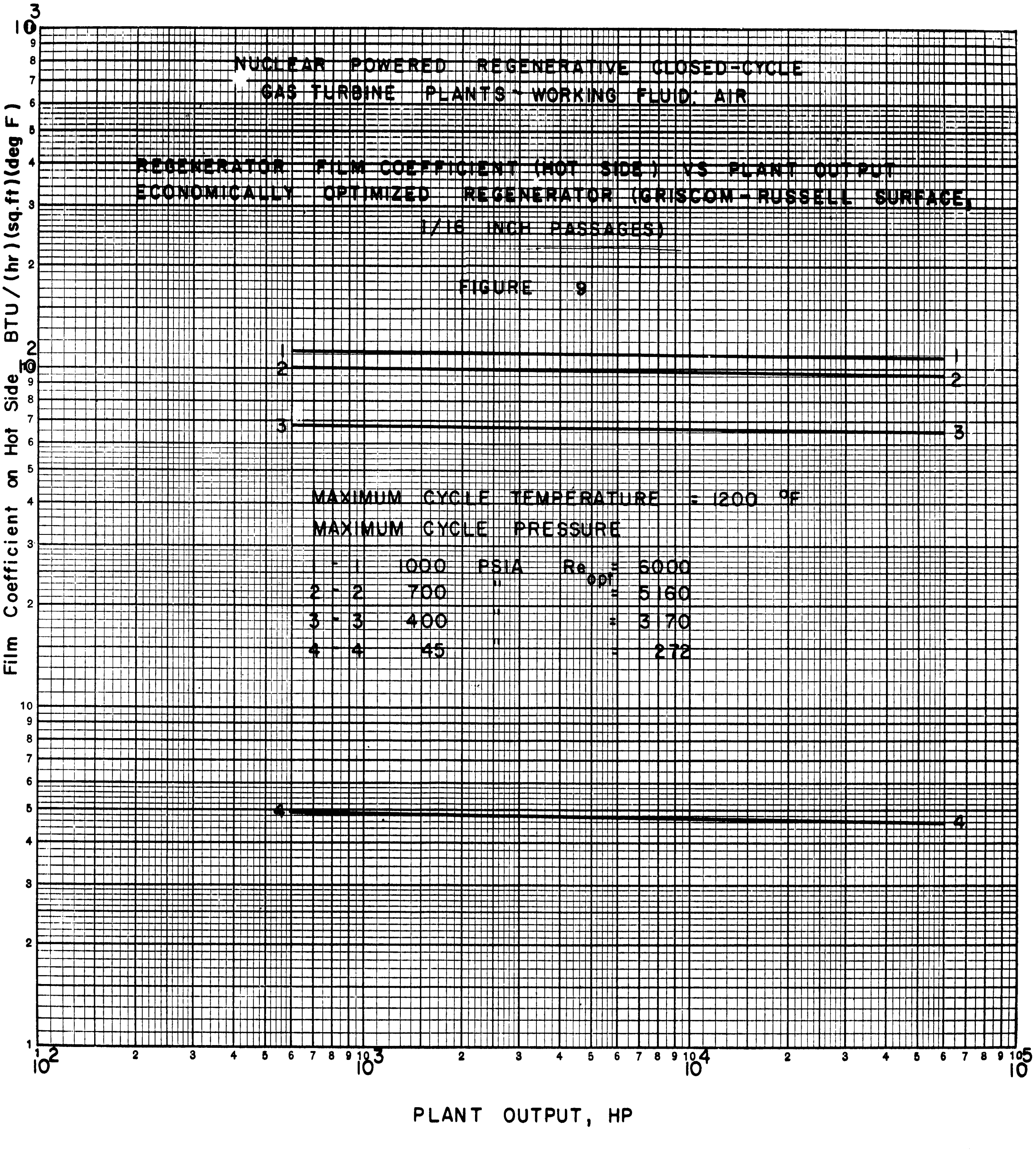


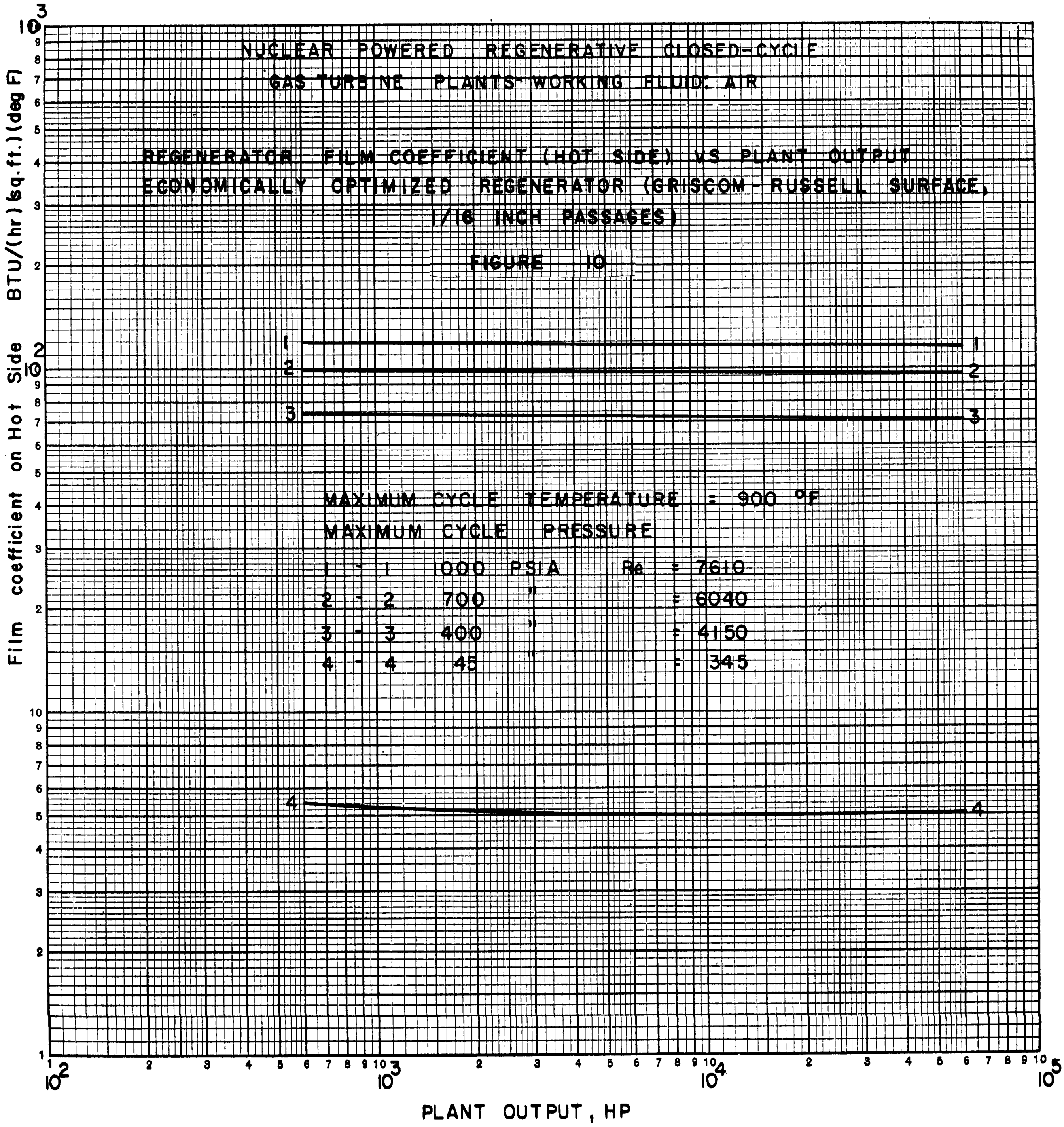
FIGURE 7 HELIUM
MAXIMUM CYCLE TEMPERATURE = 900° F
MAXIMUM CYCLE PRESSURES:
1 - 1 1000 PSIA
2 - 2 700 PSIA
3 - 3 400 PSIA
4 - 4 45 PSIA
NUCLEAR-POWERED REGENERATIVE CLOSED-CYCLE GAS TURBINE PLANTS
WORKING FLUID: HELIUM
ECONOMICALLY OPTIMIZED REGENERATOR EFFECTIVENESS VERSUS PLANT OUTPUT
REGENERATOR EFFECTIVENESS = %
PLANT OUTPUT, HP
100
90
80
70
60
50
40
30
10^2
10^3
10^4
10^5

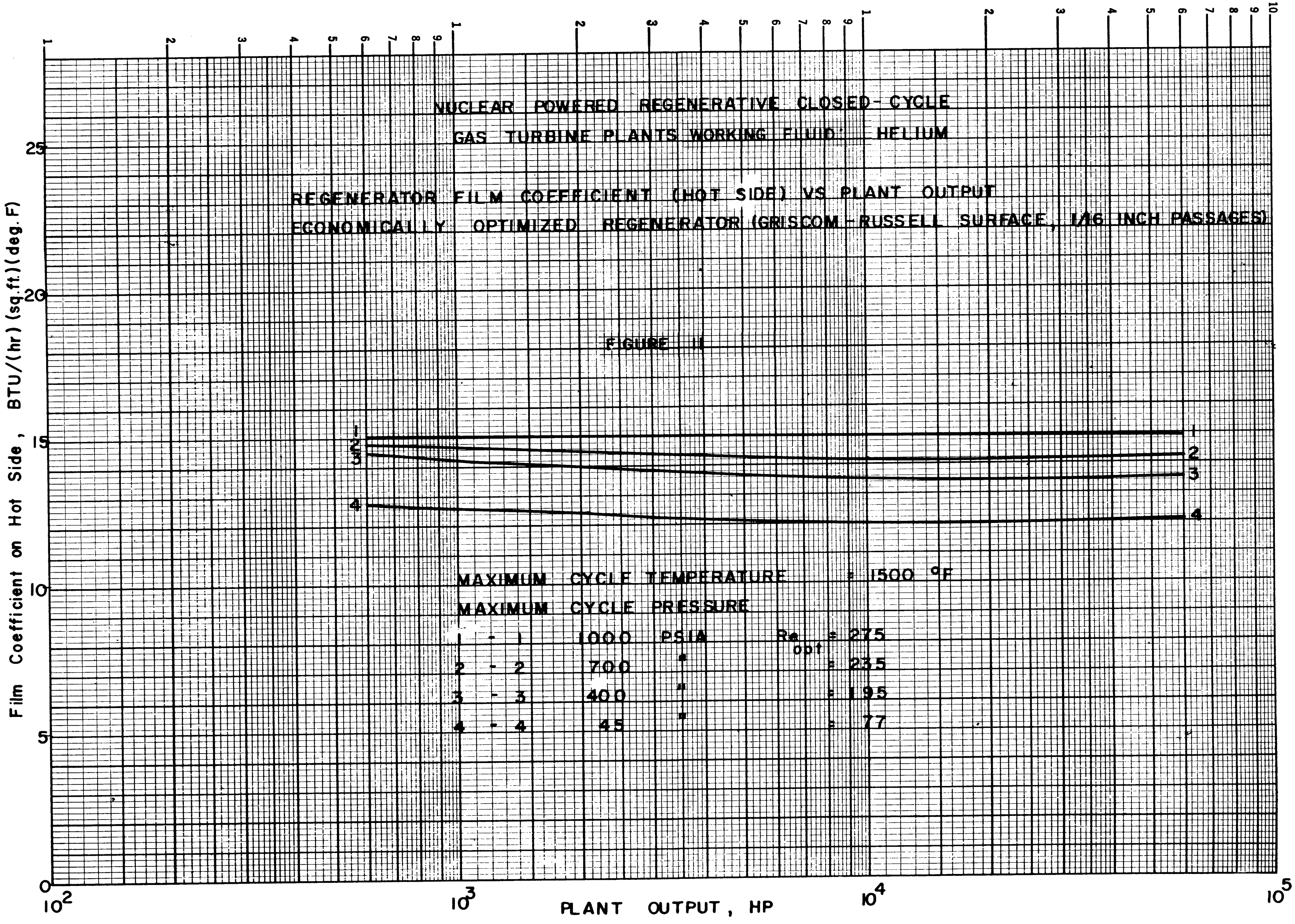


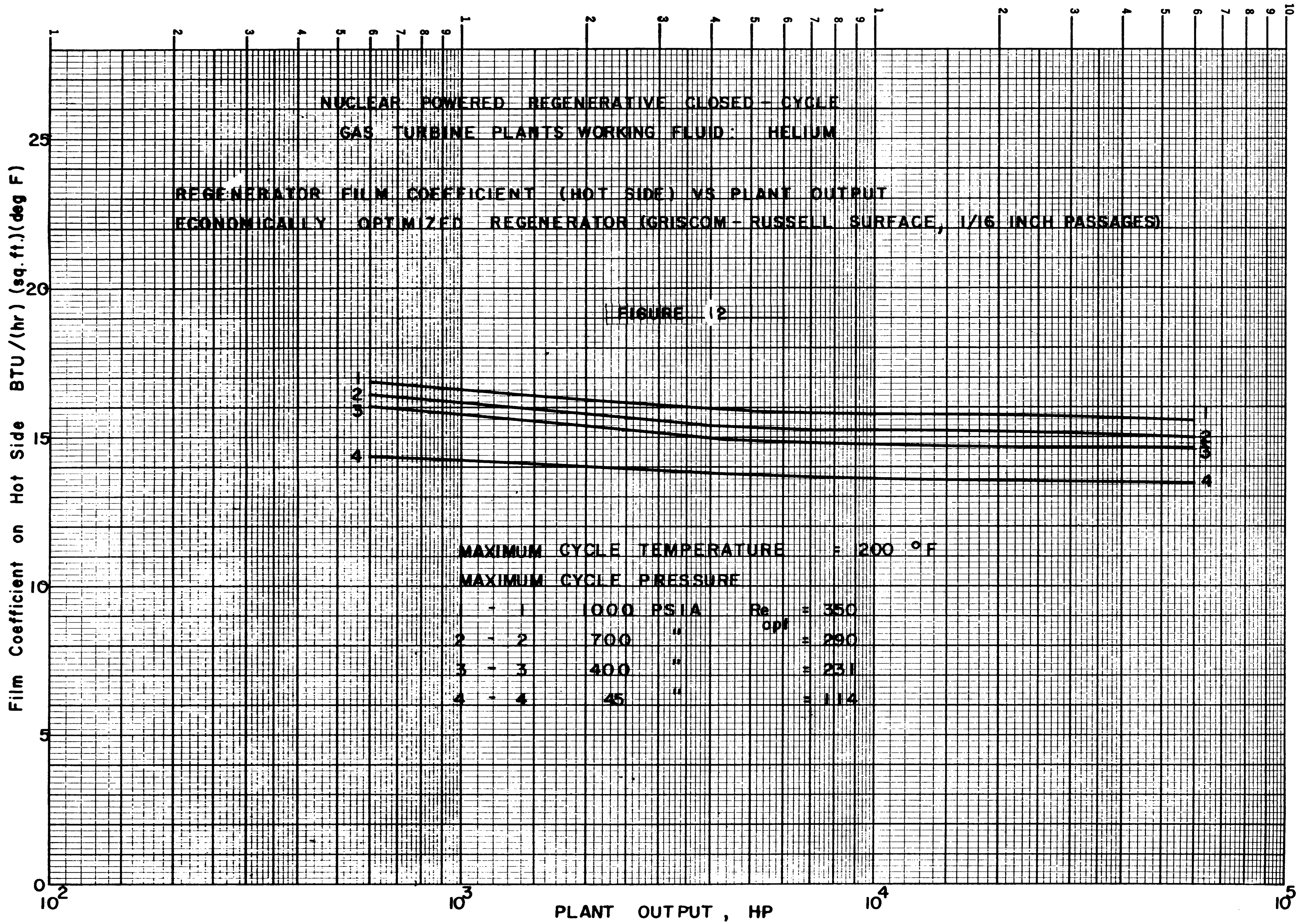
NUCLEAR POWERED REGENERATIVE CLOSED-CYCLE
GAS TURBINE PLANTS-WORKING FLUID: AIR
REGENERATOR FILM COEFFICIENT (HOT SIDE) VS PLANT OUTPUT
ECONOMICALLY OPTIMIZED REGENERATOR (GRISCOM-RUSSELL SURFACE,
1/16 INCH PASSAGES)
FIGURE 8
MAXIMUM CYCLE TEMPERATURE = 1500 °F
MAXIMUM CYCLE PRESSURE
1 - 1 1000 PSIA Re = 5000
2 - 2 700 " opt = 4200
3 - 3 400 " = 2300
4 - 4 45 " = 207
Film Coefficient on Hot Side BTU/(hr)(sq. ft)(deg. F)
PLANT OUTPUT, HP

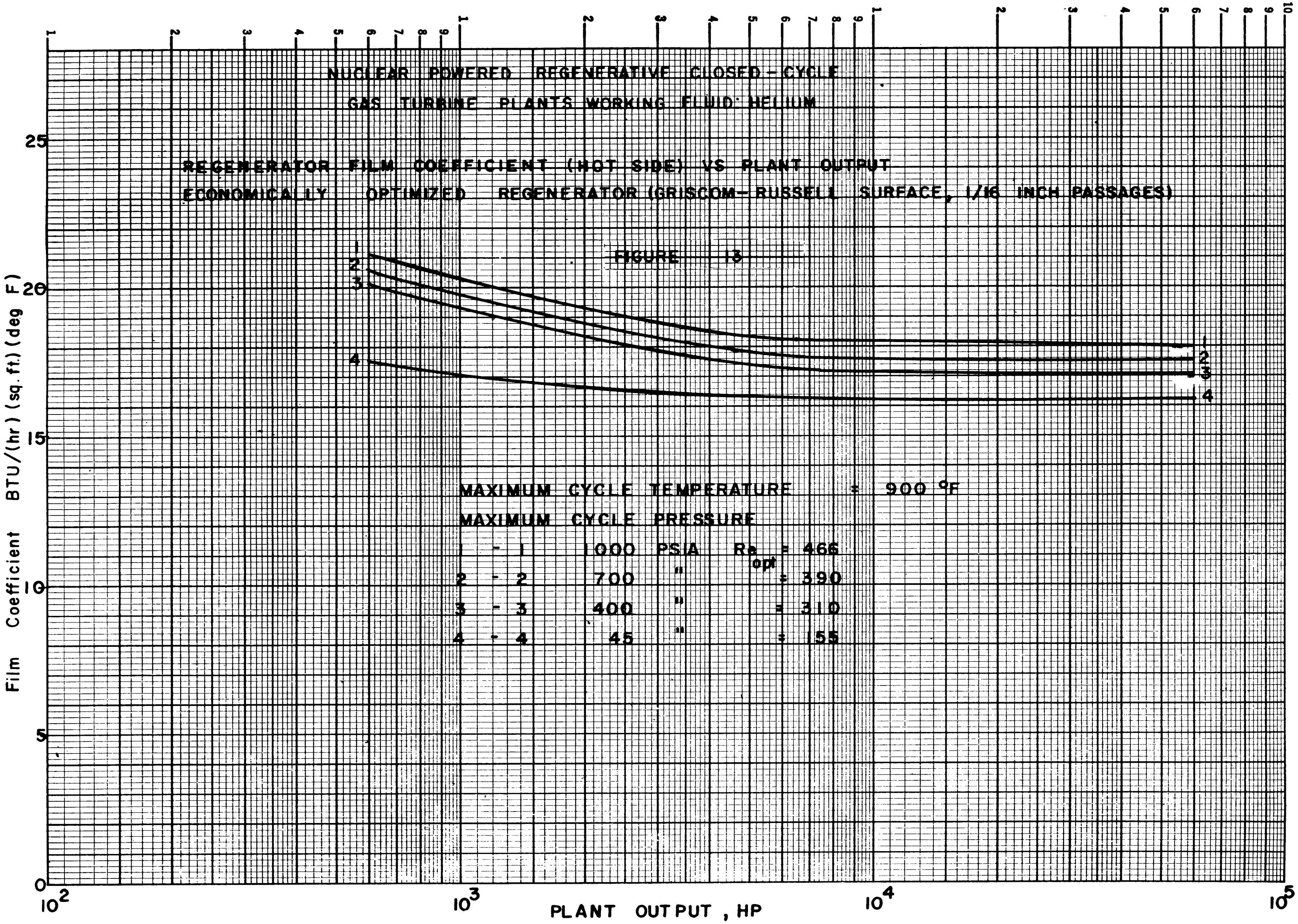


NUCLEAR POWERED REGENERATIVE CLOSED-CYCLE
GAS TURBINE PLANTS - WORKING FLUID: AIR
REGENERATOR FILM COEFFICIENT (HOT SIDE) VS PLANT OUTPUT
ECONOMICALLY OPTIMIZED REGENERATOR (GRISCOM-RUSSELL SURFACE, 1/16 INCH PASSAGES)
FIGURE 9
MAXIMUM CYCLE TEMPERATURE = 1200 °F
MAXIMUM CYCLE PRESSURE
1 - 1 1000 PSIA Re_{opt} = 6000
2 - 2 700 " = 5160
3 - 3 400 " = 3170
4 - 4 45 " = 272
Film Coefficient on Hot Side BTU / (hr) (sq.ft) (deg F)
PLANT OUTPUT, HP


NUCLEAR POWERED REGENERATIVE CLOSED-CYCLE
GAS TURBINE PLANTS- WORKING FLUID: AIR
REGENERATOR FILM COEFFICIENT (HOT SIDE) VS PLANT OUTPUT
ECONOMICALLY OPTIMIZED REGENERATOR (GRISCOM-RUSSELL SURFACE,
1/16 INCH PASSAGES)
FIGURE 10
MAXIMUM CYCLE TEMPERATURE = 900 °F
MAXIMUM CYCLE PRESSURE
1 - 1 1000 PSIA Re = 7610
2 - 2 700 " = 6040
3 - 3 400 " = 4150
4 - 4 45 " = 345
Film coefficient on Hot Side BTU/(hr)(sq.ft.)(deg F)
PLANT OUTPUT, HP


NUCLEAR POWERED REGENERATIVE CLOSED-CYCLE
GAS TURBINE PLANTS WORKING FLUID: HELIUM
REGENERATOR FILM COEFFICIENT (HOT SIDE) VS PLANT OUTPUT
ECONOMICALLY OPTIMIZED REGENERATOR (GRISCOM-RUSSELL SURFACE, 1/16 INCH PASSAGES)
FIGURE 11
MAXIMUM CYCLE TEMPERATURE = 1500 °F
MAXIMUM CYCLE PRESSURE
1 - 1 1000 PSIA Re_{opt} = 275
2 - 2 700 " = 235
3 - 3 400 " = 195
4 - 4 45 " = 77
Film Coefficient on Hot Side, BTU/(hr)(sq.ft.)(deg.F)
PLANT OUTPUT, HP


NUCLEAR POWERED REGENERATIVE CLOSED - CYCLE
GAS TURBINE PLANTS WORKING FLUID: HELIUM
REGENERATOR FILM COEFFICIENT (HOT SIDE) VS PLANT OUTPUT
ECONOMICALLY OPTIMIZED REGENERATOR (GRISCOM - RUSSELL SURFACE, 1/6 INCH PASSAGES)
FIGURE 12
MAXIMUM CYCLE TEMPERATURE = 200 °F
MAXIMUM CYCLE PRESSURE
1 - 1 1000 PSIA Re_{opt} = 350
2 - 2 700 " = 290
3 - 3 400 " = 231
4 - 4 45 " = 114
Film Coefficient on Hot Side BTU/(hr)(sq.ft.)(deg F)
PLANT OUTPUT, HP
10^2
10^3
10^4
10^5
0
5
10
15
20
25



NUCLEAR POWERED REGENERATIVE CLOSED-CYCLE
GAS TURBINE PLANTS WORKING FLUID: HELIUM
REGENERATOR FILM COEFFICIENT (HOT SIDE) VS PLANT OUTPUT
ECONOMICALLY OPTIMIZED REGENERATOR (GRISCOM-RUSSELL SURFACE, 1/16 INCH PASSAGES)
FIGURE 13
MAXIMUM CYCLE TEMPERATURE = 900 °F
MAXIMUM CYCLE PRESSURE
1 - 1 1000 PSIA Re_{opt} = 466
2 - 2 700 " = 390
3 - 3 400 " = 310
4 - 4 45 " = 155
Film Coefficient BTU/(hr) (sq. ft.) (deg F)
PLANT OUTPUT, HP
10^2
10^3
10^4
10^5
0
5
10
15
20
25

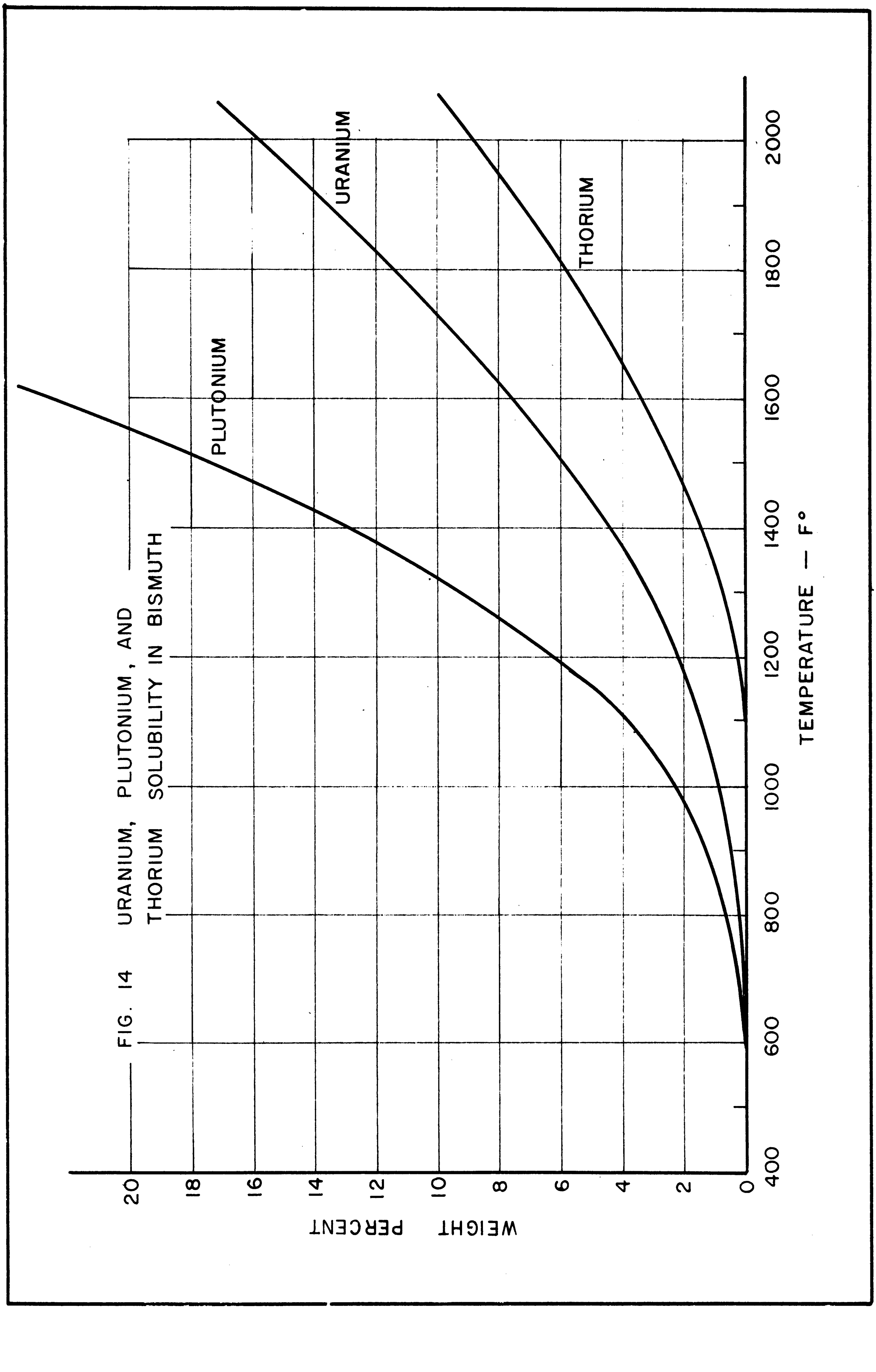
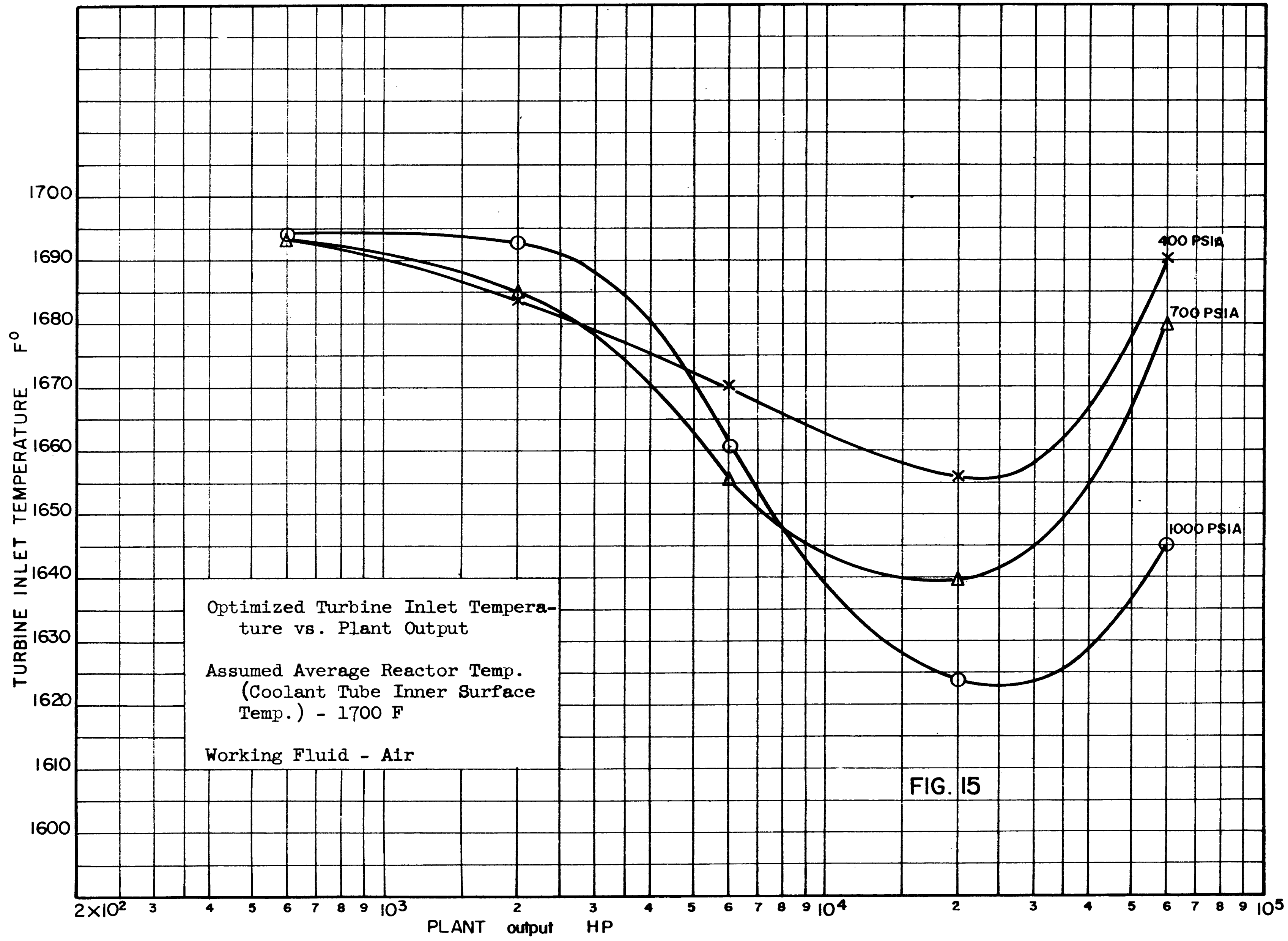
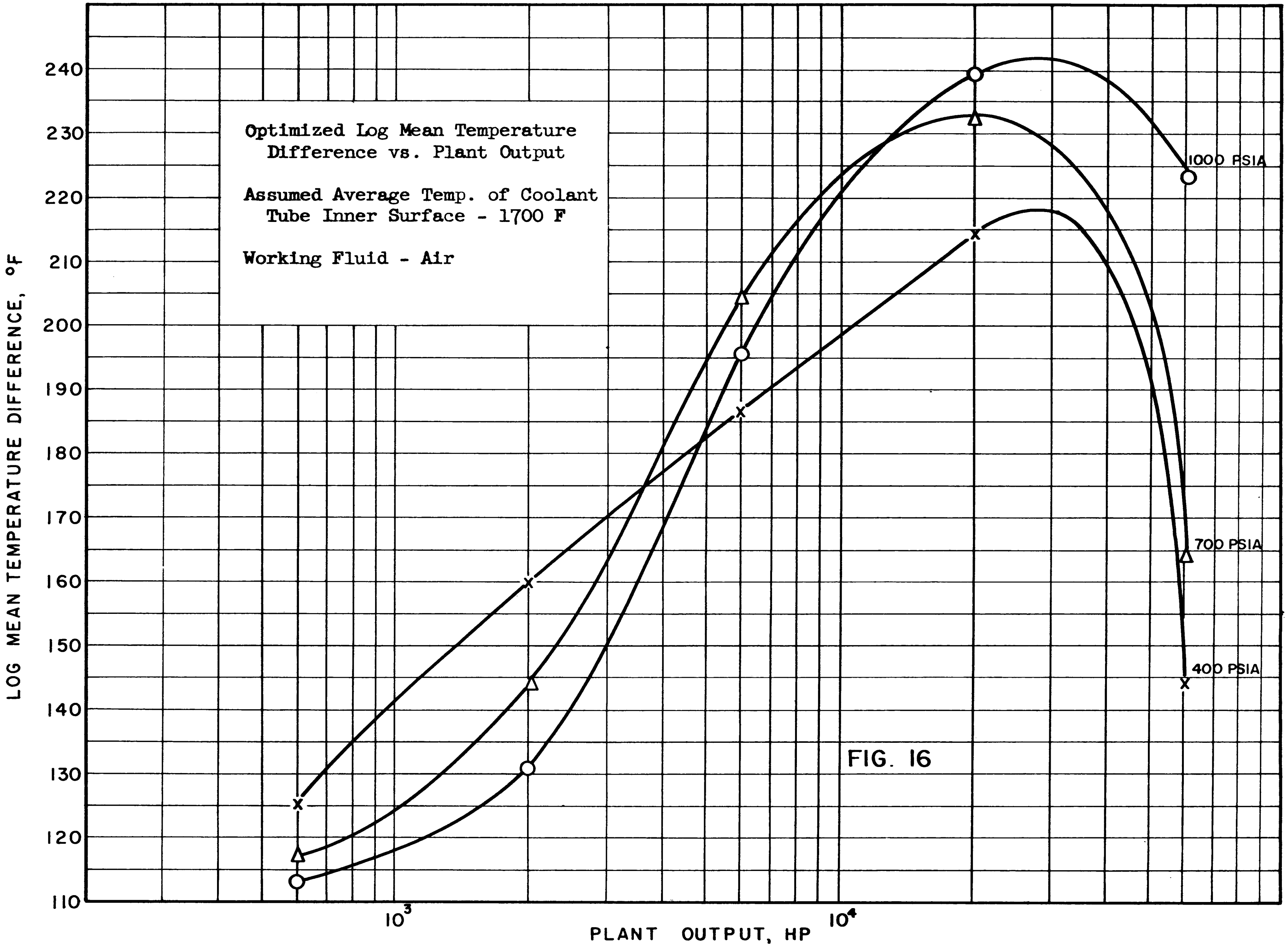


FIG. 14 URANIUM, PLUTONIUM, AND THORIUM SOLUBILITY IN BISMUTH



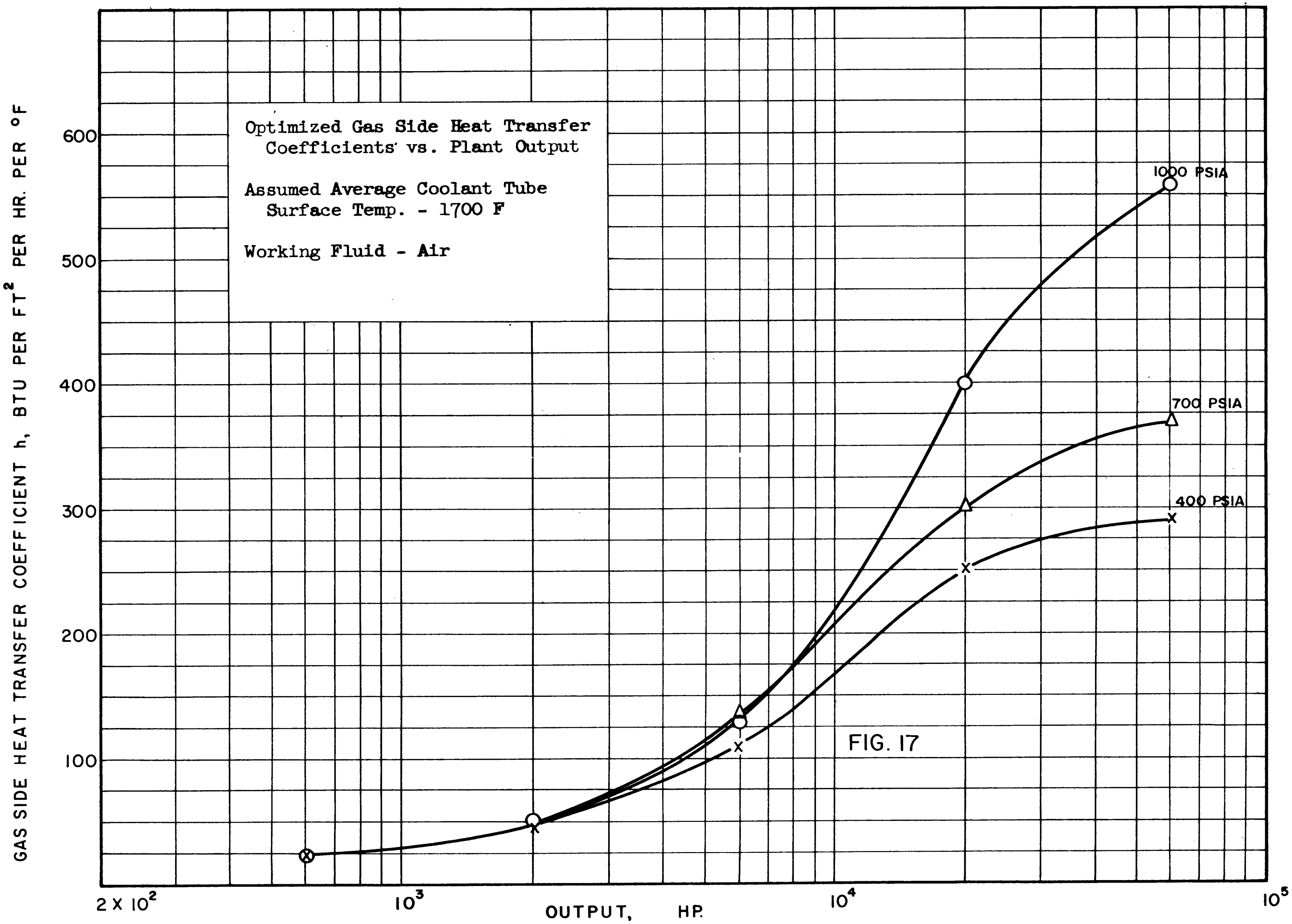
1700
1690
1680
1670
1660
1650
1640
1630
1620
1610
1600
TURBINE INLET TEMPERATURE F°
2×10²
3
4
5
6
7
8
9
10³
2
3
4
5
6
7
8
9
10⁴
2
3
4
5
6
7
8
9
10⁵
PLANT output HP
400 PSIA
700 PSIA
1000 PSIA
Optimized Turbine Inlet Temperature vs. Plant Output
Assumed Average Reactor Temp. (Coolant Tube Inner Surface Temp.) - 1700 F
Working Fluid - Air

FIG. 15



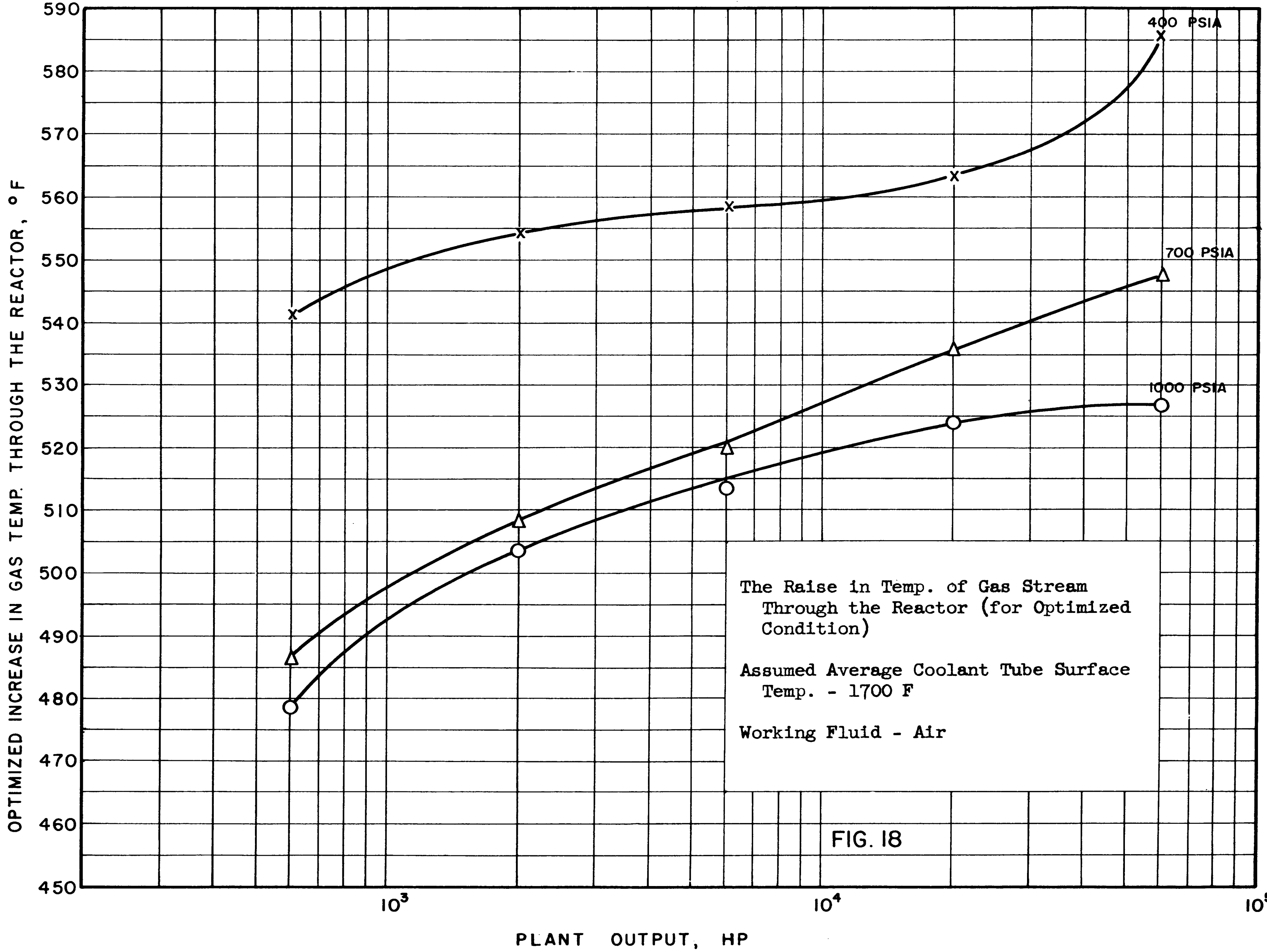
Optimized Log Mean Temperature
Difference vs. Plant Output
Assumed Average Temp. of Coolant
Tube Inner Surface - 1700 F
Working Fluid - Air
1000 PSIA
700 PSIA
400 PSIA
240
230
220
210
200
190
180
170
160
150
140
130
120
110
LOG MEAN TEMPERATURE DIFFERENCE, °F
10^3
10^4
PLANT OUTPUT, HP

FIG. 16



Optimized Gas Side Heat Transfer Coefficients vs. Plant Output
Assumed Average Coolant Tube Surface Temp. - 1700 F
Working Fluid - Air
1000 PSIA
700 PSIA
400 PSIA
GAS SIDE HEAT TRANSFER COEFFICIENT h, BTU PER FT² PER HR. PER °F
600
500
400
300
200
100
2 X 10²
10³
10⁴
10⁵
OUTPUT, HP.

FIG. 17


OPTIMIZED INCREASE IN GAS TEMP. THROUGH THE REACTOR, °F
590
580
570
560
550
540
530
520
510
500
490
480
470
460
450
400 PSIA
700 PSIA
1000 PSIA
The Raise in Temp. of Gas Stream Through the Reactor (for Optimized Condition)
Assumed Average Coolant Tube Surface Temp. - 1700 F
Working Fluid - Air
FIG. 18
10^3
10^4
10^5
PLANT OUTPUT, HP

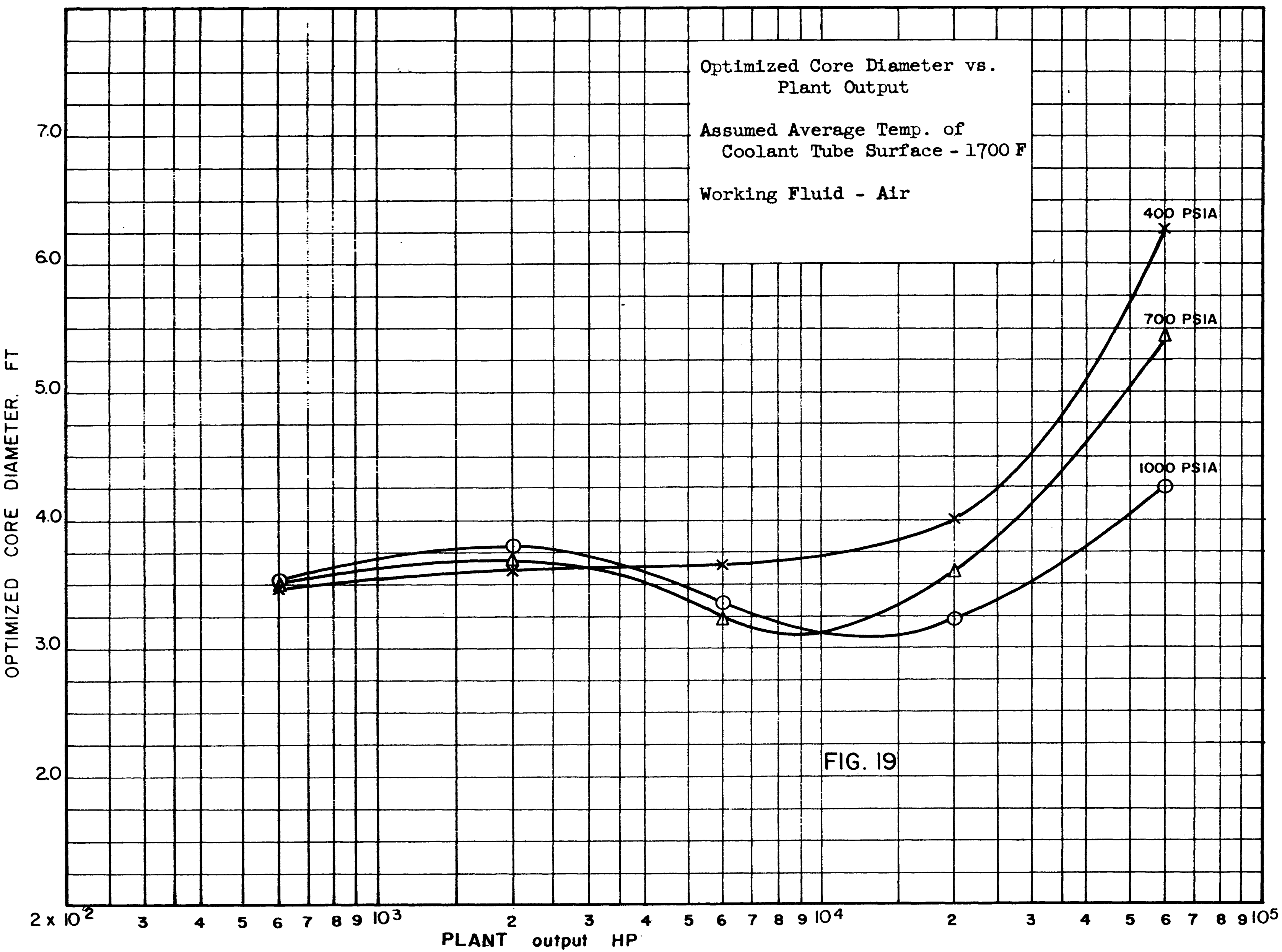

Optimized Core Diameter vs. Plant Output
Assumed Average Temp. of Coolant Tube Surface - 1700 F
Working Fluid - Air
400 PSIA
700 PSIA
1000 PSIA
OPTIMIZED CORE DIAMETER. FT
7.0
6.0
5.0
4.0
3.0
2.0
2 x 10²
3
4
5
6
7
8
9
10³
2
3
4
5
6
7
8
9
10⁴
2
3
4
5
6
7
8
9
10⁵
PLANT output HP

FIG. 19

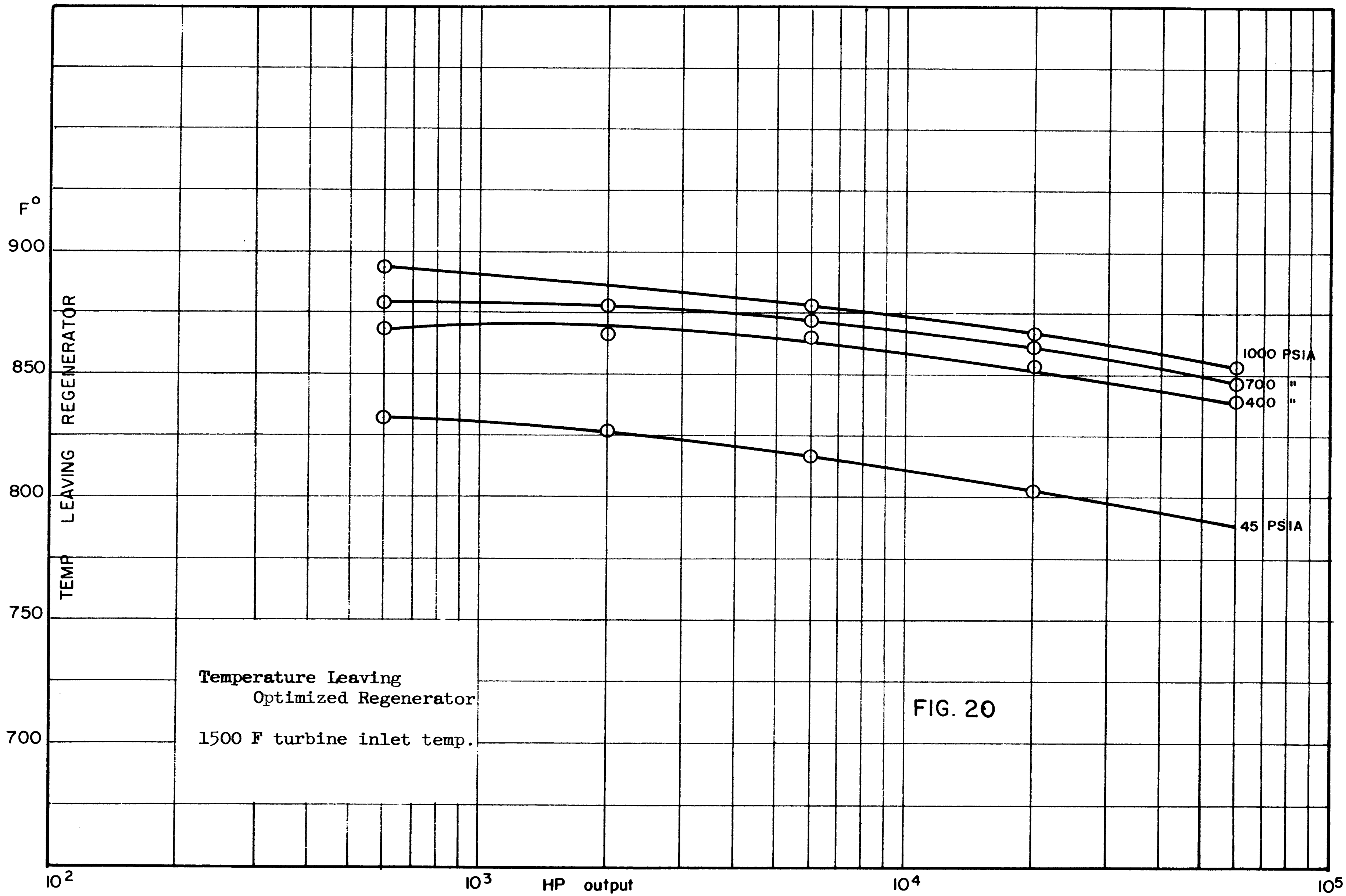

F°
900
850
800
750
700
TEMP LEAVING REGENERATOR
1000 PSIA
700 "
400 "
45 PSIA
Temperature Leaving
Optimized Regenerator
1500 F turbine inlet temp.
10²
10³
HP output
10⁴
10⁵

FIG. 20

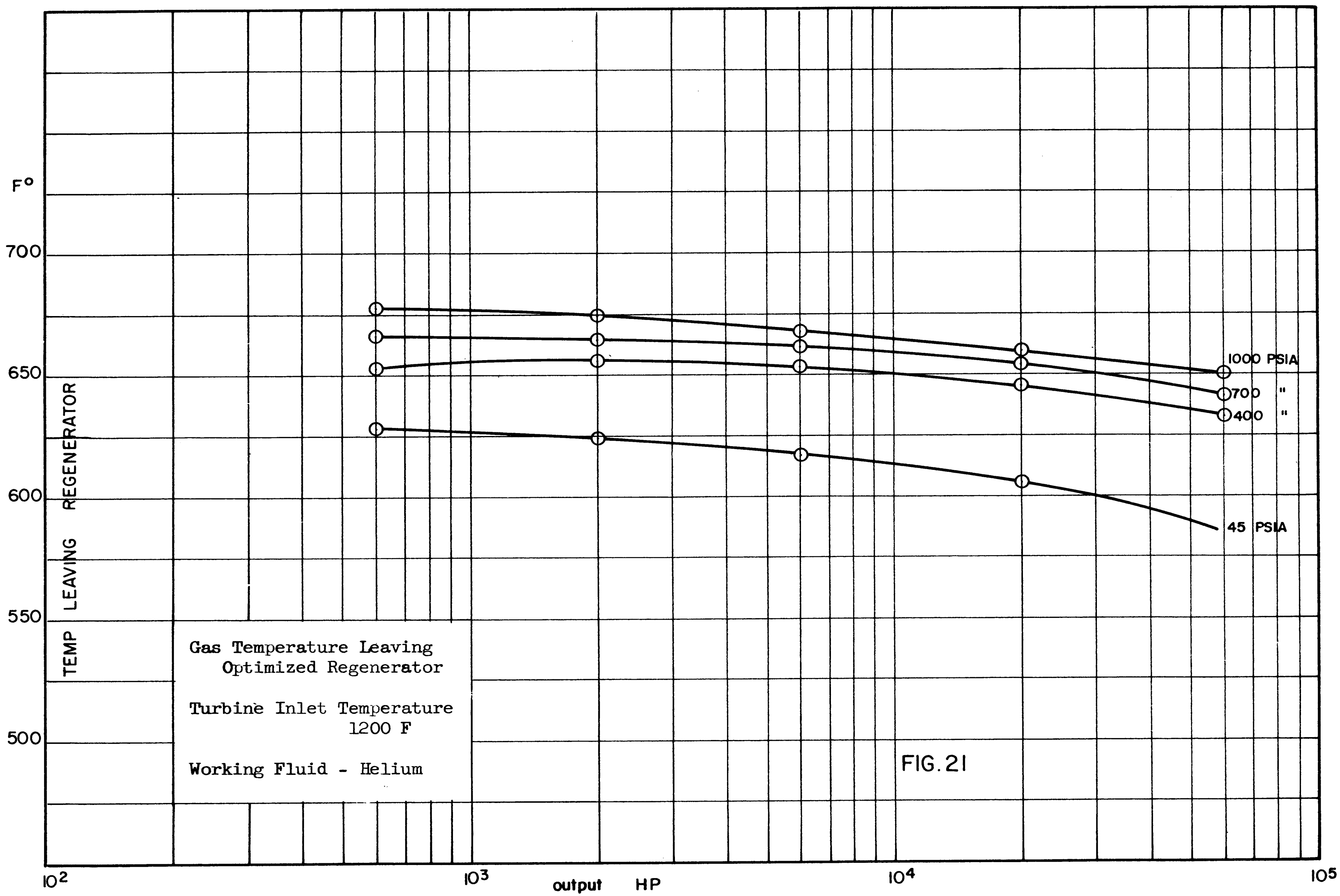

F°
700
650
600
550
500
TEMP LEAVING REGENERATOR
Gas Temperature Leaving
Optimized Regenerator
Turbine Inlet Temperature
1200 F
Working Fluid - Helium
1000 PSIA
700 "
400 "
45 PSIA
10^2
10^3
10^4
10^5
output HP

FIG. 21

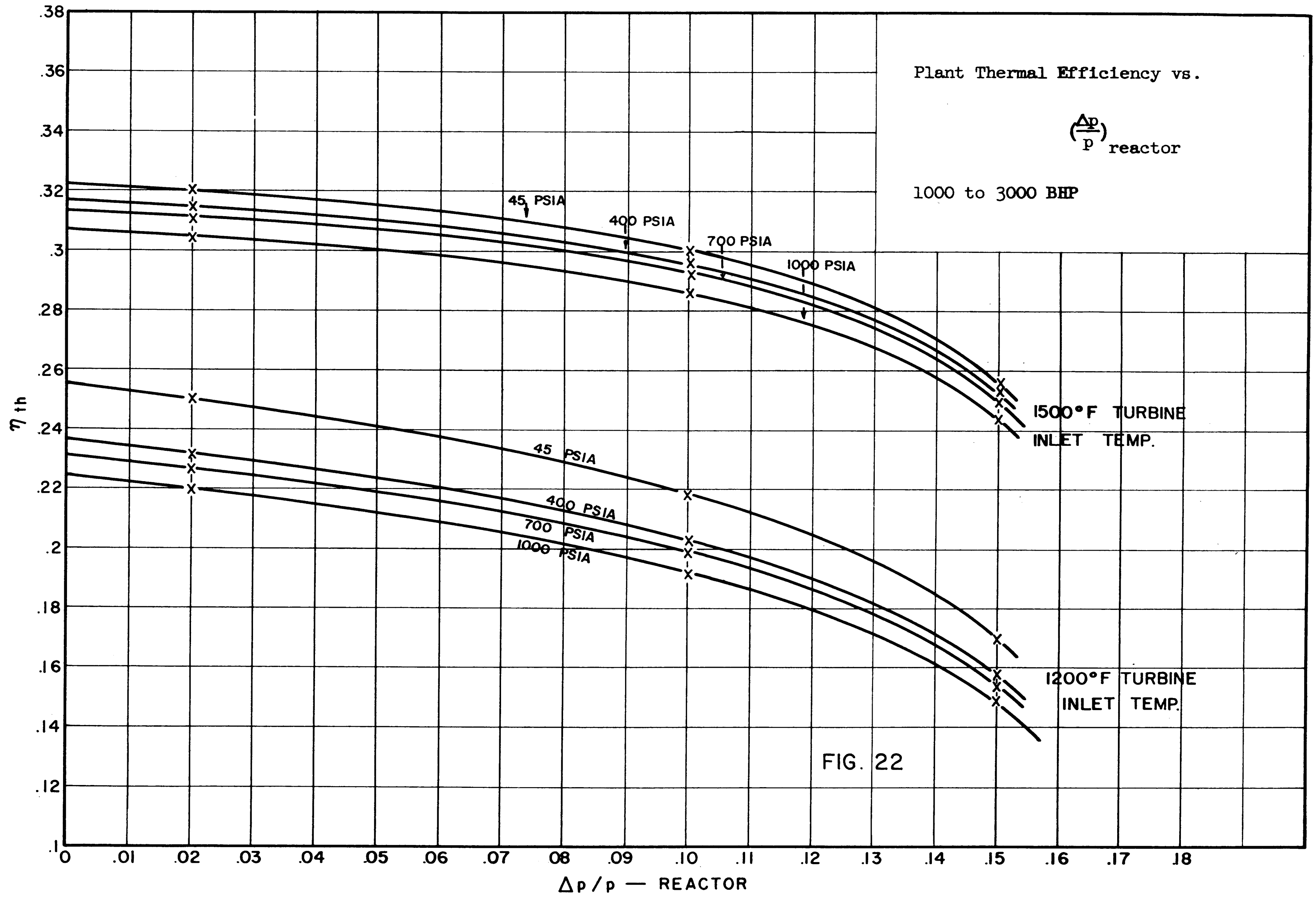

Plant Thermal Efficiency vs.
$(\frac{\Delta p}{p})_{reactor}$
1000 to 3000 BHP
45 PSIA
400 PSIA
700 PSIA
1000 PSIA
1500°F TURBINE INLET TEMP.
45 PSIA
400 PSIA
700 PSIA
1000 PSIA
1200°F TURBINE INLET TEMP.
η_{th}
.38
.36
.34
.32
.3
.28
.26
.24
.22
.2
.18
.16
.14
.12
.1
0
.01
.02
.03
.04
.05
.06
.07
.08
.09
.10
.11
.12
.13
.14
.15
.16
.17
.18
$\Delta p / p$ — REACTOR

FIG. 22