KERNFORSCHUNGSZENTRUM

KARLSRUHE

Juli 1966

KFK 447

Institut für Neutronenphysik und Reaktortechnik

Comparison of He, CO_2, and steam as coolants of a 1000 MWe Fast Reactor

Mario Dalle Donne

GESELLSCHAFT FUR KERNFORSCHUNG M. B. H.

KARLSRUHE

KERNFORSCHUNGSZENTRUM KARLSRUHE

July 1966 KFK 447

Institut für Neutronenphysik und Reaktortechnik

Comparison of He, CO_2, and steam as coolants of a 1000 MWe Fast Reactor+)

by

Mario Dalle Donne

+) Work performed within the association in the field of fast reactors between the European Atomic Energy Community and Gesellschaft für Kernforschung mbH., Karlsruhe

1. INTRODUCTION

The safety problems arising with the sodium void coefficient in big fast reactor cores make the possibility of using a much less dense coolant, like a gas, very interesting.

The recent discovery of the explosive character of sodium boiling [1, 2] makes these problems even more serious.

In the present paper the performances of three fluids: helium, carbon dioxide and water steam, as coolant of a big fast reactor core are compared.

2. EQUATIONS GOVERNING THE HEAT TRANSFER IN THE CORE+)

Assuming a chopped cosine axial power distribution and a core geometry given by a cluster of parallel smooth fuel rods, the hydraulic diameter of the cooling channel is given by:

$$d_h = K_o \left(\frac{P'}{P}\right)^{0.8} \frac{H\, P'^{0.2}}{Q^{0.2}} \qquad (1)$$

where

$$K_o = 0.0193\,(T_2 - T_1)^{0.2}\ \frac{1}{\nu}\left(\frac{\rho_w}{\rho}\right)^{0.8}\left(\frac{c_{pm}^{0.2}\ c_{pw}^{0.8}}{c_{pM}}\right)\frac{k_w^{0.2}}{Pr_w^{0.4}}\left(\frac{H'}{H}\right)$$

+) The formulae shown in this paragraph are obtained in Appendix I.

$$\nu = \frac{1}{\sqrt{\left(2\varphi \frac{c_{pM}}{c_{pm}} - 1\right)^2 \sin^2 \frac{\pi H}{2H'} - 1}}$$

$$\varphi = \frac{T_{wM} - T_1}{T_2 - T_1}$$

K_o includes all the parameters depending on the coolant physical properties, but it depends also on the ratio $\frac{H'}{H}$, on the coolant temperatures at the inlet and outlet of the core and on the maximum fuel element surface temperature. However, once these parameters are chosen, K_o is fixed and it is characteristic of the coolant chosen. It is interesting to notice at this point, that K_o is pressure indipendent for a perfect gas.

The total pressure drop in the core is the sum of various different contributions:

Δp_1, due to fiction of the coolant against the wall of the coolant channel

Δp_2, due to turbulent dissipation caused by the grids supporting the fuel rods

Δp_3, due to the momentum loss necessary to accelerate the coolant in the cooling channel (acceleration due to increase of temperature and decrease of pressure) and acceleration losses and recoveries at the inlet and outlet of the channel

Assuming a core geometry given by smooth fuel rods supported by transversal grids, the analytical expressions of these pressure drops are:

$$\Delta p_1 = \frac{1.115}{(T_2-T_1)^{1.8}} \; \frac{\mu_m^{0.2}}{\rho_m \, c_{pm}^{1.8}} \; \frac{(P'/P)^{1.8} \, H}{d_h^3 \; P'^{1.8}} \; Q^{1.8} \qquad (2)$$

$$\Delta p_2 = \frac{16 \xi}{(T_2-T_1)^2} \; \frac{1}{\rho_m \, c_{pm}^2} \; \frac{(P'/P)^2 \, Q^2}{d_h^2 \; P'^2} \qquad (3)$$

$$\Delta p_3 = \frac{16}{(T_2-T_1)^2} \left(\frac{T_2-T_1}{T_m} + \lg n \frac{p_1}{p_2} + \frac{1.1T_1-0.55T_2}{2\,T_m} \right) \frac{1}{\rho_m \, c_{pm}^2} \; \frac{(P'/P\,)^2 \, Q^2}{d_h^2 \; P'^2} \qquad (4)$$

To these pressure drops correspond three contributions to the total pumping power required to circulate the coolant in the core. The pumping power, as usual, is referred to the heat quantity to be abducted:

$$\frac{N_1}{Q_{th}} = K_1 \; (P/P')^{0.6} \; \frac{Q^{2.4}}{H^2 \; P'^{2.4}} \qquad (5)$$

$$\frac{N_2}{Q_{th}} = K_2 \; (P'/P)^{0.4} \; \frac{Q^{2.4}}{H^2 \; P'^{2.4}} \qquad (6)$$

$$\frac{N_3}{Q_{th}} = K_3 \; (P'/P)^{0.4} \; \frac{Q^{2.4}}{H^2 \; P'^{2.4}} \qquad (7)$$

where:

$$K_1 = \frac{1.553 \cdot 10^5}{\eta} \frac{\nu^3}{(T_2-T_1)^{3.4}} \frac{(\rho/\rho_w)^{2.4}}{\rho_1 \, \rho_m} \frac{c^3_{pM}}{c^{3.4}_{pm} \, c^{2.4}_{pw}} \frac{\mu_m^{0.2} \, Pr_w^{1.2}}{k_w^{0.6}} \left(\frac{H}{H'}\right)^3$$

$$K_2 = \frac{4.29 \cdot 10^4 \xi}{\eta} \frac{\nu^2}{(T_2-T_1)^{3.4}} \frac{(\rho/\rho_w)^{1.6}}{\rho_1 \, \rho_m} \frac{c^2_{pM}}{c^{3.4}_{pm} \, c^{1.6}_{pw}} \frac{Pr_w^{0.8}}{k_w^{0.4}} \left(\frac{H}{H'}\right)^2$$

$$K_3 = \frac{4.29 \cdot 10^4}{\eta} \left(\frac{T_2-T_1}{T_m} + \lg n \frac{p_1}{p_2} + \frac{1.1 \, T_1 - 0.55 \, T_2}{2 \, T_m}\right) \frac{\nu^2}{(T_2-T_1)^{3.4}} \cdot$$

$$\cdot \frac{(\rho/\rho_w)^{1.6}}{\rho_1 \, \rho_m} \frac{c^2_{pM}}{c^{3.4}_{pm} \, c^{1.6}_{pw}} \frac{Pr_w^{0.8}}{k_w^{0.4}} \left(\frac{H}{H'}\right)^2$$

Therefore the total pumping power required by the core is:

$$\frac{N_t}{Q_{th}} = \left[K_1 \, (P/P')^{0.6} + K_2 \, (P'/P)^{0.4} + K_3 \, (P'/P)^{0.4}\right] \cdot$$

$$\cdot \frac{Q^{2.4}}{H^2 \, P'^{2.4}} = K_4 \frac{Q^{2.4}}{H^2 \, P'^{2.4}} \tag{8}$$

3. GENERAL COMPARISON BASED ON HEAT TRANSFER PROPERTIES

For a certain fuel element rod (constant $\frac{P'}{P}$, P', H) and a certain amount of heat Q to carry away from any coolant channel of the core, d_h is proportional to K_o.

We assume that the core is formed by parallel fuel rods, without extented heat transfer surfaces (fins etc). The coolant volume fraction in the fuel boxes is given by:

$$\alpha = \frac{1}{1 + \frac{d}{d_h}} \tag{9}$$

Equation (19) of [3] shows that the fuel rod diameter optimum from the point of view of fuel cycle considerations is proportional to $\frac{\chi^{0.25}}{H^{0.25}}$, for constant density, enrichment and average burn up of the fuel. χ H is the heat produced by each fuel element rod and it is equal to Q, heat abducted in any coolant channel, in case of square array of rods, or equal to $\frac{1}{2}$Q in case of a triangular array. Thus we can write:

$$d = c_1 \frac{Q^{0.25}}{H^{0.5}} \tag{10}$$

From (1), (9), and (10) one obtains:

$$\alpha = \frac{1}{1 + \frac{d^{2.8}}{K_o \, c_1^2 \, c_2^{0.2} \, Q^{0.3}}} \tag{11}$$

where $P' = c_2 d$

Equation (11) gives the physical meaning of K_o: for constant fuel rod diameters an increase in K_o means an increase in coolant volume fraction, for constant coolant volume fractions an increase in K_o requires an increase in fuel rod diameter.

In the previous reasoning χH, or Q, is held constant: this is significant because, as shown in [3], χH is one of the main parameters from which the fuel cycle depends.

The physical meaning of K_4 is easier to understand: for a certain fuel element rod and certain thermal output for coolant channel Q, K_4 is proportional to the total pumping power requested to circulate the coolant in the core referred to the total thermal output.

Now if we want to compare the value of K_o and K_4 for He, CO_2, and steam as coolant of a fast reactor core, we have to make a series of estimates in choosing the numerical values of some of the parameters which appear in their expressions:

<u>Coolant pressure at the core inlet</u>: the coolant physical properties depend on the coolant pressure. The pressure variation along the cooling channels is generally quite small in comparison with the absolute value (of the order of few percents for acceptable reactors), so as characteristic of the pressure field we can take the pressure at only one section of the core. The most significant pressure is the maximum, because it influences the design of the pressure vessel, and it is generally at the inlet of the core. We have performed calculations for three values of p_1 equal to 70, 100, and 150 Atms. Such high pressures

are necessary with gas cooling and technically possible with the use of prestressed concrete pressure vessels.

Maximum cladding surface temperature: this temperature depends on the material used for cladding and on the can concept used. If one thinks in terms of stainless steel, a maximum surface temperature of 650 °C would be probably appropriate with free-standing clad, and 700 °C with collapsed clad. These figures, however, are only indicative: the cladding problem requires a lot of experimental investigation yet. Calculations were performed for both temperature values. It is interesting to notice that these values are nominal maximum and that they would correspond to maximum hot spot temperatures of 750 - 800 °C and 800 - 850 °C, respectively.

Coolant temperature at the core outlet: in the case of steam cooling no secondary circuit is used (see Fig. 1). Thus this temperature is practically equal to the maximum superheated steam temperature required by the turbines. Modern big turbines require temperatures in the range 538 °C [4, 5] to 565 °C or more [6]. With the first temperature ferritic steels can still be used, with the second austenitic steels are necessary. In our calculations we will use an average value of 550 °C. In the case of He or CO_2 heat exchangers are required. A difference between outlet gas temperature and maximum steam temperature in the heat exchangers often used is 30 °C [7]. In recent studies of gas cooled fast reactors such difference has been assumed equal to 60 °C [4] or 83 °C [5]. As a reasonable compromise we choose this difference equal to 50 °C. The resulting

He and CO_2 outlet temperature is equal to 600 °C. In reality this temperature should be obtained with an optimisation process based on the cost of the KWhe, because greater is T_2 (for a constant maximum steam temperature) smaller are the sizes of the heat exchangers and (for a constant value of T_{wM}) bigger the pumping power.

Coolant temperature at the core inlet: The gas inlet temperature (He, CO_2) should be also obtained by an optimisation process. Indeed the higher it is, the higher is the efficiency of the steam cycle, but the coolant mass flow must be increased to carry away the same quantity of heat from the core and the pumping power increases considerably. In the studies indicated above [4, 5] T_1 has been chosen equal to 260 °C, which seems a reasonable value for the considerations just mentioned. The same value will be chosen in the present study. However, it is felt that the range in which T_1 could possibly vary is much larger than that of T_2, and it depends on the coolant pressure (more precisely higher values of p_1 allow higher T_1's). In a very recent study [8], released when the majority of the calculations of the present work were already performed, T_1 has been assumed equal to 300 °C for p_1 = 70 Atm and equal to 340 °C for p_1 = 140 Atm. For steam the choice of T_1 is not free. Indeed, Fig. 1 clearly shows that T_1 depends on the pressure p_1 and the pumping power N_t/Q_{th} which fixes the degree of superheating of the steam at the core inlet. In Appendix II the relationship between T_1, p_1 and $\frac{N_t}{Q_{th}}$ is studied. The values of T_1 shown in Table I are obtained from the $\frac{N_t}{Q_{th}}$'s of the numerical example shown in the next paragraph.

The indicated values of T_1 are then less general than those for the other two gases. However, the variation of T_1 with $\frac{N_t}{Q_{th}}$ in the range of practical $\frac{N_t}{Q_{th}}$'s is quite small.

<u>Other parameters appearing in the expressions of K_o and K_4.</u>

<u>We assume</u>:

$P'/P = 1$, because we refer to fuel element without heat transfer extended surfaces

$\frac{H}{H'} = 0.821$, which refers to the numerical example of the next paragraph

$\xi = 0.21$, which corresponds to a fuel element grid support studied at Karlsruhe [9]

$\eta = 0.726$ [10]

The coolant pressure at the core outlet p_2 appears also in the expression of K_4, but the influence on K_4 is very weak. Thus, like in the case of T_1 for steam, the values of p_2 have been obtained from the numerical example of next paragraph. The helium physical properties are from [11, 12, 13]. The carbon dioxide transport properties are from [14] and the thermodynamic properties from [15], the steam properties from [16, 17, 18, 19, 20].

Table I shows the values of K_o and K_4 calculated with the assumptions outlined above. These values are fairly general because they depend mainly on the gas chosen, and on the pressure and temperatures indicated in the table.

One can see that:

K_o is practically not effected by the pressure.

Higher T_{wM}'s mean higher coolant volume fractions for constant fuel rod diameters or higher fuel rod diameters for constant coolant volume fractions.

Carbon dioxide requires smaller coolant volume fractions or smaller fuel rod diameters than those for helium or steam.

In any case steam requires the minimum amount of pumping power. Helium is the second best.

4. NUMERICAL EXAMPLE

The above considerations although useful and rather general, are not really conclusive. We know for instance that steam requires less pumping power than the other two gases, but this doesn't really mean much if we don't know something about the core dimensions, plant efficiency, fuel rating etc.

We decided therefore to calculate particular reactors using the above values of K_o and K_4 and the following parameters:

plant net electrical output:	1000 MWe
pin diameter in the core:	0.635 cm (because smooth fuel rods, roughened surface rods would allow higher diameters)

core height: 183 cm (chosen in a way to have reactors with H/D $\cong$ 0.7)

cladding material: stainless steel

cladding thickness: o.38 mm

average power per pin length: 211 W/cm

maximum power per pin length: 384 W/cm (that is maximum oxide fuel temperature equal to 2200 ^{o}C)

pin length due to axial blanket and gaseous fission products store chamber: the same as the core height

pin diameter in gaseous fission products store chamber: 80 % of that in the core

pumping power required by the primary circuit other than reactor: 2 % of the total thermal output for He and CO_2, 1 % for steam at 70 and 100 Atms, 0.5 % for steam at 150 Atms

core blockage factor: 14 % in volume, of which 13 % made up of structural material and 1 % of coolant

the steam cycle thermodynamic efficiencies (which depend mainly on T_1 and T_2) have been calculated with the optimum number of regenerative preheatings [21]

the auxiliary power other than that required by the primary coolant circulators and by the feed-water pumps in the turbine circuit, has been neglected

the primary coolant circulators are driven by high pressure steam turbines

These parameters and assumptions can be considered typical for big gas cooled reactors. Some of them are from previous studies on the subject [4, 5, 8], some come from estimates of what could be the best or most likely values of these parameters. The choosing of such parameters reduces considerably the generality of the comparison performed in this paper. Indeed, the three coolants should be compared not for the same parameters but in the conditions in which their different properties have been used at their best. In practice this is very difficult, especially because of the great uncertainties in estimating the plant capital cost variations when the above parameters, plus temperatures and pressure, vary. Nevertheless it is felt that this type of comparison is still significant because the physical properties of the three coolant compared are not extremely different. A comparison with sodium in these conditions would be naturally less significant.

The results of the calculations are reported in Table II and Figures 2 to 6. As expected carbon dioxide requires the minimum amount of coolant volume fraction, but produces the highest pressure drops in the core. As far as pumping power and plant efficiency are conderned steam is always the best, at parity of p_1 and T_{wM}, followed by helium and by carbon dioxide.

The comparison just performed was for constant core height and therefore constant fuel rod diameter (see equation (10)). It would be then very interesting to do a comparison in which H and d vary, but always remaining in the relationship (10), especially for carbon dioxide where the values of K_o are so much different from those of the other two coolants. Table III shows results of such calculations for CO_2, in which the coolant channel hydraulic diameter has been held equal to that obtained for helium (0.68 cm). One can see that the performance of CO_2 improves in respect of the case with H = constant, although it is still not as good as that of steam, but this improvement is paid by a considerable reduction in fuel rod diameter. This would probably produce an increase in number of supporting grid (increase in ξ) which was not taken into account in the calculation and which would probably completely cancel the improvement in $\frac{N_t}{Q_{th}}$ and η_t. Besides, H/D becomes about 1.1 and the core volume increases of more than 20 %. Thus we thought that the improvement was not real and we did not pursue this way further.

5. NEUTRONIC COMPUTATIONS

The thermal calculations have allowed to fix the dimensions, coolant, structural material, and fuel volume fractions in the core. These parameters known, it is possible with neutronic calculations to obtain the fuel enrichment, fuel rating, core internal and total breeding ratio, coolant void and Doppler coefficients.

These calculations are described in more detail in Appendix III. The results are shown in Table IV and in Figures 7 to 11. The rating does not vary very much due to the assumptions Q, d, and H equal constant, thus for constant total plant efficiency the fuel cycle cost is approximately the same for all reactors. The Doppler coefficient constant A_{Dop} is proportional to the Doppler coefficient because the fuel element temperature is approximately the same for all the cases considered.

The difference between the Doppler coefficient, calculated between the maximum fuel temperature of the average element and the fuel melting temperature, and the coolant void coefficient is shown in Figure 11. We consider this difference as an indicative coefficient of safety. In this one makes reference to a hypothetical accident due to instantaneous outflow of the coolant from the core and takes as a danger signal the melting of all the fuel with temperature above the peak temperature of the average fuel element. This coefficient of safety is only indicative, because such an accident is in practice impossible: the action of the negative Doppler coefficient is always much faster than that of the positive void coefficient in any possible forseen accident, especially if one contemplates the use of a per-stressed pressure vessel. A real safety evaluation could derive only by a detailed analysis of any possible credible accident.

6. CONCLUSIONS

The results of this study have not a definitive character, since too many assumptions have been made. Some of them are somewhat arbitrary. For instance, the fixing of the height of the core, of the coolant temperature at the core inlet for helium and carbon dioxide, and of the pumping power required by the primary circuit other than reactor. Furthermore, the assumed fuel element geometry can be improved. For instance, partial roughening of the fuel element rods or use of heat transfer extended surfaces for mechanical support would probably allow the use of rods with bigger diameters. It is clear also that partial roughening will improve the performance more with carbon dioxide than with helium or steam because of the tendency of that gas to require very small rod diameters with smooth rods. On the other hand, use of extended surfaces will improve the performance more with steam and helium because in the core there is more space to accomodate them than with carbon dioxide.

These are the limitations of the present study. However, we feel that the similarity of the physical properties of the coolants considered allows a fairly accurate comparison although the absolute values may be not the best for the three coolants.

The conclusions can be summarized as follows:

1. When the maximum coolant pressure and the maximum fuel element clad surface temperature are held constant:

a) The steam is by far the best coolant from a thermal point of view, i.e. the most economical, followed by helium. Carbon dioxide is the worst coolant.

b) Carbon dioxide cooled reactors are the best breeders. Steam cooled reactors have the smallest breeding ratios.

c) If one takes the available Doppler minus the total coolant void reactivity as a criterion indicative for safety, then helium is the safest coolant, steam the most dangerous, with the difference between helium and carbon dioxide being very small.

2. The passage from maximum fuel cladding surface temperature of $700^{\circ}C$ (indicative for collapsed fuel clad) to a temperature of $650^{\circ}C$ (free standing clad) produces a considerable decrease in the total plant efficiency at low pressures, the effect being almost negligible at high pressures. The effect on breeding and safety coefficient is very small, the $650^{\circ}C$ temperature is in any case better.

3. In figures 12 and 13 the total breeding ratios and the safety coefficients (Doppler minus void) are plotted versus the total plant efficiency η_t, thus giving an indication of breeding versus economy and safety versus economy. From these figures it appears that for a constant plant efficiency η_t the carbon dioxide and helium cooled reactors are better breeders and safer than the steam cooled reactors. However, since steam is a better coolant from a thermal point of view, the com-

parison at constant η_t is much closer than at constant coolant pressure. Furthermore it should be stressed that these diagrams are biased rather strongly against steam because for a constant η_t and approximately the same fuel cycle cost, implicit in the assumptions made in this comparison, steam is more economical than carbon dioxide or helium. Indeed the capital costs are considerably lower with steam because the required coolant pressure is smaller and no heat exchangers are necessary. On the other hand the turbine costs would probably be less with carbon dioxide or helium, especially if a low coolant pressure is chosen.

Until these capital costs are known with a sufficient precision, it is not possible to draw a definitive conclusion regarding the choice of a gas coolant for a large fast reactor.

Table I

Coolant	Maximum Surface Fuel Elem. Temperature T_M (^oC)	Core Inlet Pressure p_1 (Atm)	Core Inlet Coolant Temperature T_1 (^oC)	Core Outlet Coolant Temperature T_2 (^oC)	Coolant Channel Size Factor $K_o \cdot 10^2 (cal^{0.2} cm^{-0.2} sec^{-0.2})$	Pumping Power To Thermal Output Ratio Factor $K_4 \cdot 10^6 (cal^{-2.4} cm^{4.4} sec^{2.4})$
He	650	70	260	600	1.468	4.133
		100	260	600	1.468	1.816
		150	260	600	1.468	0.7706
	700	70	260	600	2.132	1.603
		100	260	600	2.132	0.7599
		150	260	600	2.132	0.3310
CO_2	650	100	260	600	1.040	4.575
		150	260	600	1.039	1.331
	700	70	260	600	1.521	2.755
		100	260	600	1.543	1.111
		150	260	600	1.518	0.4242
H_2O steam	650	70	310.4	550	1.666	3.066
		100	319.1	550	1.644	0.9439
		150	344.5	550	1.584	0.3028
	700	70	295.6	550	1.972	1.228
		100	315.2	550	1.981	0.5080
		150	342.6	550	1.856	0.1791

Table II

Coolant	Maximum Surface Fuel Element Temperature T_{wM} (°C)	Core Inlet Pressure p_1 (Atm)	Coolant Channel Hydraulic Diameter d_h (cm)	Pressure Drop Across The Core p(Atm)	Pumping Power To Thermal Output Ratio N_t/Q_{th}	Power Plant Net Electrical Efficiency η_t	Reactor Thermal Output Q_{th} (MW)	Pumping Power N_t (MW)	Core Volume V(liters)	Core Coolant Volume Fraction	Core Steel Volume Fraction	Core Fuel Volume Fraction
He	650	70	0.468	15.6	18.0 %	27.2 %	3670	661	11120	37.4 %	24.2 %	38.4 %
		100	0.468	9.8	9.0 %	33.0 %	3030	274	9170	37.4 %	24.2 %	38.4 %
		150	0.468	6.2	5.0 %	35.7 %	2800	140	8490	37.4 %	24.2 %	38.4 %
	700	70	0.680	4.9	8.4 %	33.5 %	2990	250	10790	45.4 %	22.4 %	32.2 %
		100	0.680	3.3	5.0 %	35.6 %	2805	141	10130	45.4 %	22.4 %	32.2 %
		150	0.680	2.2	3.3 %	36.7 %	2720	90	9830	45.4 %	22.4 %	32.2 %
CO_2	650	100	0.332	51.4	17.1 %	26.3 %	3800	737	10080	30.5 %	25.7 %	43.8 %
		150	0.332	23.4	7.1 %	34.3 %	2910	206	7725	30.5 %	25.7 %	43.8 %
	700	70	0.485	21.4	12.6 %	30.7 %	3256	412	10020	38.2 %	24.0 %	37.8 %
		100	0.492	11.9	6.3 %	34.8 %	2872	181	8910	38.5 %	23.9 %	37.6 %
		150	0.484	7.5	3.6 %	36.5 %	2740	100	8430	38.2 %	24.0 %	37.8 %
H_2O steam	650	70	0.532	15.6	12.9 %	28.6 %	3490	451	11200	40.2 %	23.6 %	36.2 %
		100	0.525	8.4	4.7 %	37.0 %	2700	126	8600	39.9 %	23.6 %	36.5 %
		150	0.506	5.0	1.7 %	40.4 %	2470	42	7750	39.1 %	23.8 %	37.1 %
	700	70	0.629	7.3	5.8 %	35.5 %	2815	164	9790	43.7 %	22.8 %	33.5 %
		100	0.632	4.7	3.0 %	38.6 %	2588	78	8980	43.9 %	22.7 %	33.4 %
		150	0.592	3.1	1.2 %	40.9 %	2450	30	8250	42.5 %	23.0 %	34.5 %

Table III

CO_2, $T_1 = 260^\circ C$, $T_2 = 600^\circ C$, $T_{wM} = 700^\circ C$, $d_h = 0.680$ cm									
Core Inlet Pressure p_1(Atm)	Core Height H(cm)	Fuel Rod Diameter d(cm)	Pressure Drop Across The Core p(Atm)	Pumping Power To Thermal Output Ratio N_t/Q_{th}	Power Plant Net Electrical Efficiency η_t	Reactor Thermal Output Q_{th}(MW)	Pumping Power N_t(MW)	Core Volume V(liters)	Core Coolant Volume Fraction
70	266	0.526	15.9	9.6 %	32.7 %	3061	294	12 240	49.5 %
100	262	0.540	8.9	4.7 %	35.8 %	2790	132	11 360	48.9 %

Table IV

Coolant	Maximum Surface Fuel Elem. Temperature T_{wM} (°C)	Core Inlet Pressure p(Atm)	Enrichment (number of atoms of fissile plutonium over total fuel)	Rating (MWth/kg fiss)	Core Internal Breeding Ratio	Total Breeding Ratio	Void Coefficient ($)	Doppler Coefficient Constant A_{Dop}	Indicative Safety Coefficient ($)
He	650	70	0.0958	1.03	1.108	1.382	0.413	-0.00892	-1.040
		100	0.0976	1.01	1.083	1.382	0.608	-0.00870	-0.810
		150	0.0986	1.00	1.071	1.384	0.922	-0.00859	-0.478
	700	70	0.1037	0.952	1.015	1.336	0.555	-0.00810	-0.764
		100	0.1050	0.939	1.001	1.332	0.789	-0.00795	-0.506
		150	0.1050	0.939	1.000	1.330	1.183	-0.00850	-0.202
CO_2	650	100	0.0925	1.07	1.152	1.407	0.703	-0.00944	-0.835
		150	0.0949	1.04	1.121	1.400	1.185	-0.00937	-0.342
	700	70	0.0978	1.01	1.083	1.376	0.658	-0.00887	-0.788
		100	0.0992	0.994	1.068	1.387	1.053	-0.00890	-0.396
		150	0.0998	0.988	1.062	1.369	1.579	-0.00905	+0.105
H_2O steam	650	70	0.1014	0.972	1.004	1.257	5.40	-0.0133	+3.24
		100	0.1050	0.939	0.947	1.209	6.89	-0.0141	+4.59
		150	0.1068	0.924	0.899	1.148	8.71	-0.0152	+6.23
	700	70	0.1068	0.924	0.944	1.214	5.92	-0.0131	+3.78
		100	0.1092	0.902	0.899	1.162	7.47	-0.0141	+5.17
		150	0.1099	0.899	0.859	1.100	9.06	-0.0155	+6.54

APPENDIX I

EQUATIONS GOVERNING THE HEAT TRANSFER IN THE CORE

All the following considerations are valid for a central core subassembly where the maximum amount of heat will be produced. For the other subassemblies a certain amount of orificing is required.

Temperature distribution in cooling channels

We assume that heat flux distribution in the core is given axially by a cosine law:

$$q = q_o \cos \frac{\pi z}{H'} \tag{12}$$

Then, the coolant enthalpy along the cooling channel is given by:

$$G \left[I(z) - I_1 \right] = \int_{-H/2}^{z} q_o \cos \frac{\pi z}{H'} \, dz = \frac{q_o H'}{\pi} \left(\sin \frac{\pi z}{H'} + \sin \frac{\pi H}{2H'} \right) \tag{13}$$

The variation of pressure in a cooling channel is generally small in comparison with the absolute pressure value (of the order of few percents) so it is possible to write with good approximation in place of equation (13):

$$G \, c_p \left[T(z) - T_1 \right] = \frac{q_o H'}{\pi} \left(\sin \frac{\pi z}{H'} + \sin \frac{\pi H}{2H'} \right) \tag{14}$$

Where c_p is the specific heat averaged in the intervall $T(z)-T_1$. This approximation is, of course, better for helium and CO_2 than for steam.

For $T(z) = T_2$ the (14) becomes

$$Q = G\, c_{pm}(T_2 - T_1) = \frac{2q_o H'}{\pi} \sin \frac{\pi H}{2H'}$$

$$\therefore \quad q_o = \frac{\pi G\, c_{pm}(T_2 - T_1)}{2H' \sin \frac{\pi H}{2H'}} \tag{15}$$

By definition one has:

$$P' h \left[T_w(z) - T(z)\right] = q = q_o \cos \frac{\pi z}{H'} \tag{16}$$

$$\therefore \quad T_w(z) = T_1 + \frac{q_o H'}{\pi G\, c_p} \left(\sin \frac{\pi z}{H'} + \sin \frac{\pi H}{2H'}\right) + \frac{q_o}{P' h} \cos \frac{\pi z}{H'} \tag{17}$$

c_p and h are not independent of z because they depend on the temperature, which is not constant along the channel. However, except in cases of very high temperature gradients, this dependence is rather weak, so, for semplicity and to obtain general laws independent of temperature distributions, we assume in the following differentiation that c_p and h are independent of z. Differentiating (17) in respect of z and setting the derivative equal to zero, we obtain the section z_M where T_w has a maximum:

$$\left(\frac{\partial T_w}{\partial z}\right)_{z=z_M} = \frac{q_o}{G\, c_{pM}} \cos \frac{\pi z_M}{H'} - \frac{q_o \pi}{P' h H'} \sin \frac{\pi z_M}{H'} = 0$$

$$\therefore \quad \operatorname{tg} \frac{\pi z_M}{H'} = \frac{P'hH'}{G\ c_{pM}} \tag{18}$$

We have from (17):

$$T_{wM} = T_1 + \frac{q_o H'}{\pi G\ c_{pM}} \left(\sin \frac{\pi z_M}{H'} + \sin \frac{\pi H}{2H'} \right) + \frac{q_o}{P'h} \cos \frac{\pi z_M}{H'} \tag{19}$$

Replacing equations (15) and (18) in (19) we obtain:

$$\varphi = \frac{T_{wM} - T_1}{T_2 - T_1} = \frac{1}{2} \frac{c_{pm}}{c_{pM}} \left(1 + \frac{1}{\sin \frac{\pi z_M}{H'} \sin \frac{\pi H}{2H'}} \right) \tag{20}$$

$$\therefore \quad \operatorname{tg} \frac{\pi z_M}{H'} = \frac{1}{\sqrt{\left(2\varphi \frac{c_{pM}}{c_{pm}} - 1\right)^2 \sin^2 \frac{\pi H}{2H'} - 1}} \tag{21}$$

and

$$\frac{hP'H'}{\pi G\ c_{pM}} = \frac{1}{\sqrt{\left(2\varphi \frac{c_{pM}}{c_{pm}} - 1\right)^2 \sin^2 \frac{\pi H}{2H'} - 1}} = \nu \tag{22}$$

Replacing equations (15) and (18) in (14) for $z = z_M$ we obtain:

$$T(z_M) = T_1 + \frac{c_{pm}}{c_{pM}} \frac{T_2 - T_1}{2} \left(\frac{1}{\sin^2 \frac{\pi H}{2H'} \left(2\varphi \frac{c_{pM}}{c_{pm}} - 1\right)} + 1 \right) \tag{23}$$

Heat transfer condition in the section of maximum cladding temperature

The heat transfer coefficient by forced convection with gases in tubes in presence of high temperature has been extensively studied in the last 15 years. The conclusions of these studies and of experimental investigations is that the heat transfer coefficient is given by the equation:

$$Nu_w = 0.020 \, Re_w^{0.8} \, Pr_w^{0.4} \qquad (24) \quad [22]$$

Equation (24) is valid for tubes and for ducts which do not present very acute corners. So it is also valid in the case of parallel rods array when the pitch to rod diameter ratio p/d is not too near to one. Fortunately for big reactor cores the optimum p/d is considerably higher than one (1.2÷1.5) and the cooling channel doesn't present acute corners. Consequently the equation (24) is still valid for this geometry provided that the inner diameter of the tube is replaced by the hydraulic diameter d_h.

By definition equation (24) can be written:

$$\frac{h \, d_h}{k_w} = 0.020 \, \left(\frac{\rho_w \, v \, d_h}{\mu_w}\right)^{0.8} Pr_w^{0.4} = 0.020 \, \left(\frac{\rho_w}{\rho} \frac{4A\rho v}{P \, \mu_w}\right)^{0.8} Pr_w^{0.4}$$

$$\therefore \; d_h = \frac{0.020}{h} \, k_w \, Pr_w^{0.4} \left(\frac{\rho_w}{\rho}\right)^{0.8} \left(\frac{4 \, G}{P \, \mu_w}\right)^{0.8} \qquad (25)$$

but for equation (22) we have:

$$h = \frac{\pi G \; c_{pM}}{P'H'} \tag{26}$$

and

$$G = \frac{Q}{c_{pm} \; (T_2 - T_1)} \tag{27}$$

Replacing (26) and (27) in equation (25) and remembering that

$$Pr_w = \frac{\mu_w \; c_{pw}}{k_w}$$

one obtains equation (1)

Pressure drop in the cooling channels

The equations giving the pressure drop are the following:

$$\Delta p_1 = f_m \frac{2H}{d_h} \; \rho_m \; v_m^2 \tag{28}$$

$$\Delta p_2 = \xi \; \rho_m \; v_m^2 \tag{29}$$

$$\Delta p_3 = F(T_1, T_2, p_1, p_2) \; \rho_m \; v_m^2 \tag{30}$$

Where

$$f_m = \frac{0.046}{\left(\frac{\rho_m \; v_m \; d_h}{\mu_m}\right)^{0.2}} \quad \text{(Blasius's equation)} \tag{31}$$

$$F(T_1, T_2, p_1, p_2) = \left(\frac{T_2 - T_1}{T_m} + \lg_e \frac{p_1}{p_2} + \frac{1.1\ T_1 - 0.55\ T_2}{2\ T_m}\right) \qquad (32)$$

(see [23] page 40)

and ξ is a dimensionless constant depending on the shape, size and number of supporting grids.

Replacing equations (27), (31), (32) in equations (28), (29), and (30) and taking into account that $d_h = \frac{4\ A}{P}$ and $G = A\ \rho_m\ v_m$ one obtains equations (2), (3), and (4). When the units of the physical parameters on the right side of these equations are those of the c.g.s system, the resulting pressure differences are in dyne/cm^2. To have them in atmospheres it is necessary to multiply by $0.9869 \cdot 10^{-6}$.

Pumping Power

The total pumping power requested to circulate the coolant in the primary coolant circuit is established by the characteristics of the whole primary circuit. The part of the total pumping power relative to the core is given by:

$$N_t = \frac{1}{\eta}\ \frac{G_t}{\rho_1}\ (\Delta p_1 + \Delta p_2 + \Delta p_3) =$$

$$= \frac{1}{\eta}\ \frac{Q_{th}}{c_{pm}(T_2 - T_1)\rho_1}\ (\Delta p_1 + \Delta p_2 + \Delta p_3) \qquad (33)$$

We define N_1, N_2, N_3 in the following way:

$$N_1 = \frac{1}{\eta} \frac{Q_{th}}{c_{pm} (T_2-T_1) \rho_1} \Delta p_1$$

$$N_2 = \frac{1}{\eta} \frac{Q_{th}}{c_{pm} (T_2-T_1) \rho_1} \Delta p_2$$

$$N_3 = \frac{1}{\eta} \frac{Q_{th}}{c_{pm} (T_2-T_1) \rho_1} \Delta p_3$$

so $N_t = N_1 + N_2 + N_3$

Using equations (2), (3), and (4) one obtains:

$$\frac{N_1}{Q_{th}} = \frac{1.115}{\eta} \frac{(P'/P)^{1.8}}{(T_2-T_1)^{2.8}} \frac{\mu_m^{0.2}}{\rho_1 \rho_m c_{pm}^{2.8}} \frac{H Q^{1.8}}{P'^{1.8} d_h^3} \tag{34}$$

$$\frac{N_2}{Q_{th}} = \frac{16 \xi}{\eta} \frac{(P'/P)^2}{(T_2-T_1)^3} \frac{1}{\rho_1 \rho_m c_{pm}^3} \frac{Q^2}{P'^2 d_h^2} \tag{35}$$

$$\frac{N_3}{Q_{th}} = \frac{16}{\eta} \left(\frac{T_2-T_1}{T_m} + \lg_e \frac{p_1}{p_2} + \frac{1.1 T_1 - 0.55 T_2}{2 T_m}\right) \frac{(P'/P)^2}{(T_2-T_1)^3} \cdot$$

$$\cdot \frac{1}{\rho_1 \rho_m c_{pm}^3} \cdot \frac{Q^2}{P'^2 d_h^2} \tag{36}$$

Replacing equation (1) in (34), (35), and (36) one obtains equations (5), (6), and (7). When the units of the physical parameters on the right side of these equations are those of the c.g.s. system and the heat is in calories, the resulting N/Q's are in erg/cal. To obtain the N/Q's in dimensionless form it is necessary to divide by $4.187 \cdot 10^7$.

APPENDIX II

RELATIONSHIP BETWEEN T_1, p_1 AND $\frac{N_t}{Q_{th}}$ FOR STEAM

With reference to Figure 1 one can write that the power of the steam circulator is given by:

$$N_t = \frac{(I_1 - I_o)\ G_t}{\eta_M} \qquad (37)$$

where η_M is the ratio between the power given to the steam in the circulator and the power at the shaft of the driving turbine. η_M takes into account of the mechanical losses in the power transmission from turbine to circulator and of the mechanical losses in the circulator itself. We assume $\eta_M = 0.91$.

On the other hand the core thermal output is given by:

$$Q_{th} = G_t\ (I_2 - I_1) \qquad (38)$$

Combining (37) and (38) one obtains:

$$I_1 = \frac{I_o + \eta_M \frac{N_t}{Q_{th}} I_2}{1 + \eta_M \frac{N_t}{Q_{th}}} \qquad (39)$$

Once T_1, T_2, and p_1 are known, $\frac{N_t}{Q_{th}}$, p_2, and p_o must be estimated. From p_o and the condition $x = 1$ it is possible to obtain I_o and from T_2 and p_2 one has I_2. It is possible then to calculate I_1 from equation (39) and from p_1 and I_1 one obtains T_1. One can then with this value of T_1 make a calculation of the pressure drop Δp and pumping power necessary to circulate the steam in the primary circuit (see Appendix I) and check if the assumed values of $\frac{N_t}{Q_{th}}$ and p_2 are correct. If not the calculation must be repeated until they are.

The difference $p_2 - p_o$ takes into account of the pressure drop in the primary circuit other than in the reactor and it can be obtained by the relationship:

$$\frac{p_2 - p_o}{p_1 - p_2} = \frac{\text{pumping power required by reactor}}{\text{pumping power required by the rest of primary circuit}}$$

APPENDIX III

NEUTRONIC COMPUTATIONS

The objectives of the neutronic computations were to obtain the enrichment, breeding ratio and basic safety parameters for each of the 17 cases. Since this involves a considerable amount of computation, simplifying approximations were applied which primarily effect the coolant loss reactivity effect, but which, it is felt, give sufficient accuracy for this type of comparative study. These simplifying approximations are the following:

1. Calculations were performed in spherical geometry.

This approximation is important to the computation of the coolant loss reactivity effects, where core leakage makes an important contribution. The cylindrical cores considered (with $H/D \cong 0.7$) can be well simulated in spherical geometry.

2. No changes were made in the microscopic group constants going from coolant-in to coolant-out conditions.

With sodium coolant the elastic transfer cross sections of all materials in the range of the large sodium resonance ($\cong$ 3 keV) change considerably when the sodium is removed. For the gas coolants considered here such strong resonances do not exist, so it is expected that this contribution to the coolant loss effect will be small.

3. The total coolant loss effects will be in the same proportions as the maximum positive coolant loss effects.
The greatest positive reactivity effect due to coolant loss most likely occurs when only the center portion of the core is void of coolant. However, for the purpose of comparing coolants, the effect of total coolant removal from the core is sufficiently representative of the maximum positive value.

The plutonium isotopic composition used in the computations is that representative of fast reactor, recycled plutonium which has reached equilibrium levels [24], The fractions of Pu 239, Pu 240, and Pu 241 in the plutonium are 0.823, 0.159, and 0.018, respectively. A fission product pseudo-element representing long lived fission products was included with an atom density corresponding to 50 000 MWD/t burn-up in the fuel. The cross section set used is the 26-group set generated in the Soviet Union and adapted for use at Karlsruhe [25]. Plutonium "enrichments" were set to give a k_{eff} value of 1.01, and the values given in Table IV are based on total fuel plus fission product nuclear densities. The computations of the Doppler coefficient assumed both a uniform temperature distribution and change in the fuel (i.e., an isotopic coefficient).

Conversion from the actual cylindrical dimensions of the reactor cores to equivalent spherical dimensions was done by conserving core leakage in the fundamental mode approximation. That is, the axial and radial bucklings, B^2_{ax} and B^2_{rad}, were estimated from

$$B^2_{ax} = \left(\frac{\pi}{H + 40}\right)^2 \tag{40}$$

and

$$B^2_{rad} = \left(\frac{2.404}{\frac{D}{2} + 20}\right)^2 \tag{41}$$

where H and D are the core height and diameter, respectively, and the 20 cm extrapolation lengths are approximate for fast reactors. The spherical core radii R_{sp} are then given by

$$\left(\frac{\pi}{R_{sp} + 20}\right)^2 = B^2_{ax} + B^2_{rad} \tag{42}$$

A spherical blanket 40 cm thick was used. The composition of this blanket was taken to be a leakage weighted average of the axial and radial blanket compositions. Letting $N_{sp,m}$, $N_{ax,m}$, and $N_{rad,m}$ be the nuclear densities for material m in the spherical, axial and radial blankets, respectively, the value of $N_{sp,m}$ was obtained from

$$N_{sp,m} = \frac{N_{ax,m}\, B^2_{ax} + N_{rad,m}\, B^2_{rad}}{B^2_{ax} + B^2_{rad}} \tag{43}$$

The quantities $N_{ax,m}$ and $N_{rad,m}$ were determined by considering the axial blanket to have the same volume fractions as the core, and the radial blanket to have $\frac{1}{2}$ the core coolant volume fraction and the same volume ratio of steel to fuel as in the core. Blanket fuel was taken to be pure U 238 oxide.

The coolant loss reactivity effects were computed by running one-dimensional diffusion theory problems with and without the coolant present in the core. Doppler computations were accomplished with the 900 ^{o}C and 2100 ^{o}C values of the Pu 239 and U 238 cross sections available in the 26-group set. The small Doppler contribution due to Pu 240 was not included. It was then assumed that the Doppler coefficient follows the expression

$$\left(\frac{dk}{dT_f}\right) = \frac{A_{Dop}}{T_f} \qquad (44)$$

where A_{Dop} is a constant and T_f is the average absolute temperature of the fuel. The values of A_{Dop} have been reported for the 17 cases considered in this study and form a basis of comparison of the Doppler coefficients.

NOMENCLATURE

Geometrical parameters:

A = coolant channel cross section area (cm^2)

B^2_{ax} = core axial buckling (cm^{-2})

B^2_{rad} = core radial buckling (cm^{-2})

d = fuel element rod diameter (cm)

d_h = coolant channel hydraulic diameter (cm)

D = core diameter (cm)

H = core height (cm)

H' = core extrapolated height (cm)

P = coolant channel wetted perimeter (cm)

P' = coolant channel heat transfer perimeter (cm)

R_{sp} = radius of equivalent spherical core (cm)

z = distance from the core inlet of the coolant cross section considered (cm)

z_M = distance from the core inlet of the coolant cross section where the fuel surface temperature is maximum (cm)

Coolant physical properties:

c_{pm} = specific heat at constant pressure at temperature T_m and pressure p_m (cal/gr ^oC)

c_{pM} = specific heat at constant pressure between T_1 and $T(z_M)$ and at pressure p_m (cal/gr ^oC)

c_{pw} = specific heat at constant pressure at temperature T_{wM} and pressure p_m (cal/gr ^oC)

$I(z)$ = enthalpy at cross section z (cal/gr)

I_o = enthalpy upstream the coolant circulator (cal/gr)

I_1 = enthalpy at core inlet (cal/gr)

I_2 = enthalpy at core outlet (cal/gr)

x = steam quality

k_w = thermal conductivity at temperature T_{wM} and pressure p_m (cal/cm sec ^oC)

μ_m = dynamic viscosity at temperature T_m and pressure p_m (gr/cm sec)

μ_w = dynamic viscosity at temperature T_{wM} and pressure p_m (gr/cm sec)

ρ = density at temperature $T(z_M)$ and pressure p_m (gr/cm^3)

ρ_1 = density at temperature T_1 and pressure p_1 (gr/cm^3)

ρ_m = density at temperature T_m and pressure p_m (gr/cm^3)

ρ_w = density at Temperature T_{wM} and pressure p_m (gr/cm^3)

Other physical parameters:

G = coolant mass flow in the central coolant channels (gr/sec)

G_t = total coolant mass flow through core (gr/sec)

h = heat transfer coefficient between fuel element surface and coolant (cal/cm sec ^oC)

N_1 = pumping power required by friction losses in the core (MW)

N_2 = pumping power required by losses due to supporting grids in the core (MW)

N_3 = pumping power required by acceleration losses in the core (MW)

N_t = total pumping power required by the core (MW)

$N_{sp,m}$ = nuclear density of material m in spherical blanket (atoms/barn cm)

$N_{ax,m}$ = nuclear density of material m in axial blanket (atoms/barn cm)

$N_{rad,m}$ = nuclear density of material m in radial blanket (atoms/barn cm)

p_o = coolant pressure upstream the coolant circulator (Atm)

p_1 = coolant pressure at core inlet (Atm)

p_2 = coolant pressure at core outlet (Atm)

p_m = $\frac{p_1 + p_2}{2}$ (Atm)

Δp_1 = pressure drop required by friction losses in the core (Atm)

Δp_2 = pressure drop due to grids supporting fuel rods (Atm)

Δp_3 = pressure drop due to acceleration losses in the core (Atm)

Δp = $\Delta p_1 + \Delta p_2 + \Delta p_3$ (Atm)

q = heat to coolant per coolant channel and per unit length (cal/cm sec)

q_o = maximum value of q (cal/cm sec)

Q = heat output per coolant channel (cal/sec)

Q_{th} = total core heat output (MW)

T_1 = absolute coolant temperature at core inlet (^oK)

T_2 = absolute coolant temperature at core outlet (^oK)

T_m $= \dfrac{T_1 + T_2}{2}$ (^oK)

$T(z)$ = absolute coolant temperature at coordinate z (^oK)

$T_w(z)$ = absolute fuel element surface temperature at coordinate z (^oK)

T_{wM} = maximum absolute fuel element surface temperature (^oK)

T_f = average absolute temperature of fuel (^oK)

v = coolant velocity (cm/sec)

χ = heat produced per unit length of fuel rod (cal/cm sec)

τ_w = shear stress at the wall (dynes/cm^2)

Dimensionless groups:

f_m $= \dfrac{\tau_w}{\rho v/2}$ = Fanning friction factor

Nu_w $= \dfrac{h\ d_h}{k_w}$ = Nusselt number

Pr_w $= \dfrac{\mu_w\ c_{pw}}{k_w}$ = Prandtl number

Re_w $= \dfrac{\rho_w v\ d_h}{\mu_w}$ = Reynolds number

α = coolant fraction in fuel boxes

η = coolant circulator efficiency

η_M = coolant circulator mechanical efficiency

η_t = power plant total net efficiency

φ $= \dfrac{T_{wM} - T_1}{T_2 - T_1}$

$$\nu = \frac{1}{\sqrt{\left(2\varphi \frac{c_{pM}}{c_{pm}} - 1\right)^2 \sin^2 \frac{\pi H}{2H'} - 1}}$$

$$\zeta = \frac{\Delta p_2}{\rho_m v^2}$$ = pressure drop factor due to grids supporting fuel elements

Constants:

A_{Dop} = constant for $1/T_f$ variation of Doppler coefficient

$$c_1 = \frac{d\ H^{0.5}}{Q^{0.25}}$$

$$c_2 = P'/d = \begin{cases} \pi & \text{(square fuel element rod array)} \\ \pi/2 & \text{(triangular fuel element rod array)} \end{cases}$$

K_o = defined in equation (1)

K_1 =, K_2 =, K_3 = defined in equation (6), (7), (8)

$$K_4 = K_1(P/P')^{0.6} + K_2(P'/P)^{0.4} + K_3(P'/P)^{0.4}$$

ACKNOWLEDGEMENT

The author wishes to thank Professor K. Wirtz, Director of the Institut für Neutronenphysik und Reaktortechnik, who encouraged him to perform the work.

REFERENCES

[1] H.W. Hoffmann and A.I. Krakoviak:
"Convective boiling with liquid potassium"
Proceedings of the 1964 Heat Transfer and Fluid Mechanics Institute. Stanford, Calif.: Stanford University Press 1964

[2] M. Dalle Donne:
"A new and simple method of estimating the liquid superheat due to surface conditions in nucleate boiling and its application to sodium"
to be published in "Nukleonik", Bd. 7, Nr. 8

[3] W. Häfele:
"Principles and problems of the development of a fast reactor fuel element"
Proceeding of the conference on breeding, economics and safety in large fast power reactors, p. 265, October 7-10-1963

[4] P. Fortescue, R. Shanstrom, L. Mayer, W. Simon, P. Fischer:
"Gas cooling for fast reactors"
A/Conf 28/P/694 May 1964

[5] The Gas-cooled Reactor Project Staff:
"Gas-cooled fast reactor concepts"
ORNL 3642 September 1964

[6] G. Wensch:
"United States fast reactor fuel element development"
Conference held at Kernforschungszentrum Karlsruhe, the 8.11.1963

[7] D. Smidt et al.:
Kernreaktor Internal Report 1962

[8] P. Fortescue, R. Shanstrom, H. Fenech:
"Development of the gas cooled fast reactor concept"
ANS 100, p. 89, April 1965

[9] B. de Temple:
Internal Kernreaktor Report 1963

[10] H. Franze:
Internal Kernreaktor Report 1964

[11] J.B. Mann and N.C. Blais:
"Thermal conductivity of helium and hydrogen at high temperatures"
Los Alamos Scientific Laboratory Report LA 2316, 1959

[12] N.V. Tsederberg:
"Thermal conductivity of gases and liquids"
The M.I.T. Press Cambridge, Mass., 1965

[13] P.D. Stroom, W.E. Ibele, T.F. Irvine, jr.:
"Helium Prandtl number measurements and calculated viscosity and thermal conductivity"
1961 International Heat Transfer Conference, University of Colorado, 1961

[14] K. Sullivan:
"Thermal conductivity and viscosity of CO_2"
TRG. Report 438(R), 1963

[15] J. Hilsenrath et al.:
"Tables of thermodynamic and transport properties"
Pergamon Press, 1960

[16] M.P. Vukalovitch:
"Thermodynamic properties of water and steam"
VEB Verlag Technik Berlin, 1958

[17] VDI Wasserdampftafel, 5. Auflage

[18] VDI Wärmeatlas, Db 3, 1953

[19] Vargaftik and Tarasimanov:
Teploenergetika 8, 1961

[20] Mayinger:
Int. Journal of Heat and Mass Transfer 5, 1962

[21] H. Franze:
Private communication. Kernreaktor Karlsruhe, 1965

[22] M. Dalle Donne, F.H. Bowditch:
"Experimental local heat transfer and friction coefficients for subsonic laminar, transitional and turbulent flow of air or helium in a tube at high temperatures"
Dragon Project Report 184, April 1963
A.E.E. Winfrith, Dorset, England

[23] W. Hall:
"Reactor heat transfer"
Nuclear Engineering Monographs
Temple Press Ltd, London, 1958

[24] A. Jansen:

Private communication.Kernreaktor Karlsruhe 1965

[25] L.B. Abagjan et al.:

"Gruppenkonstanten schneller und intermediärer Neutronen

für die Berechnung von Kernreaktoren"

KFK-tr-144 (Translation from Russian), 1964

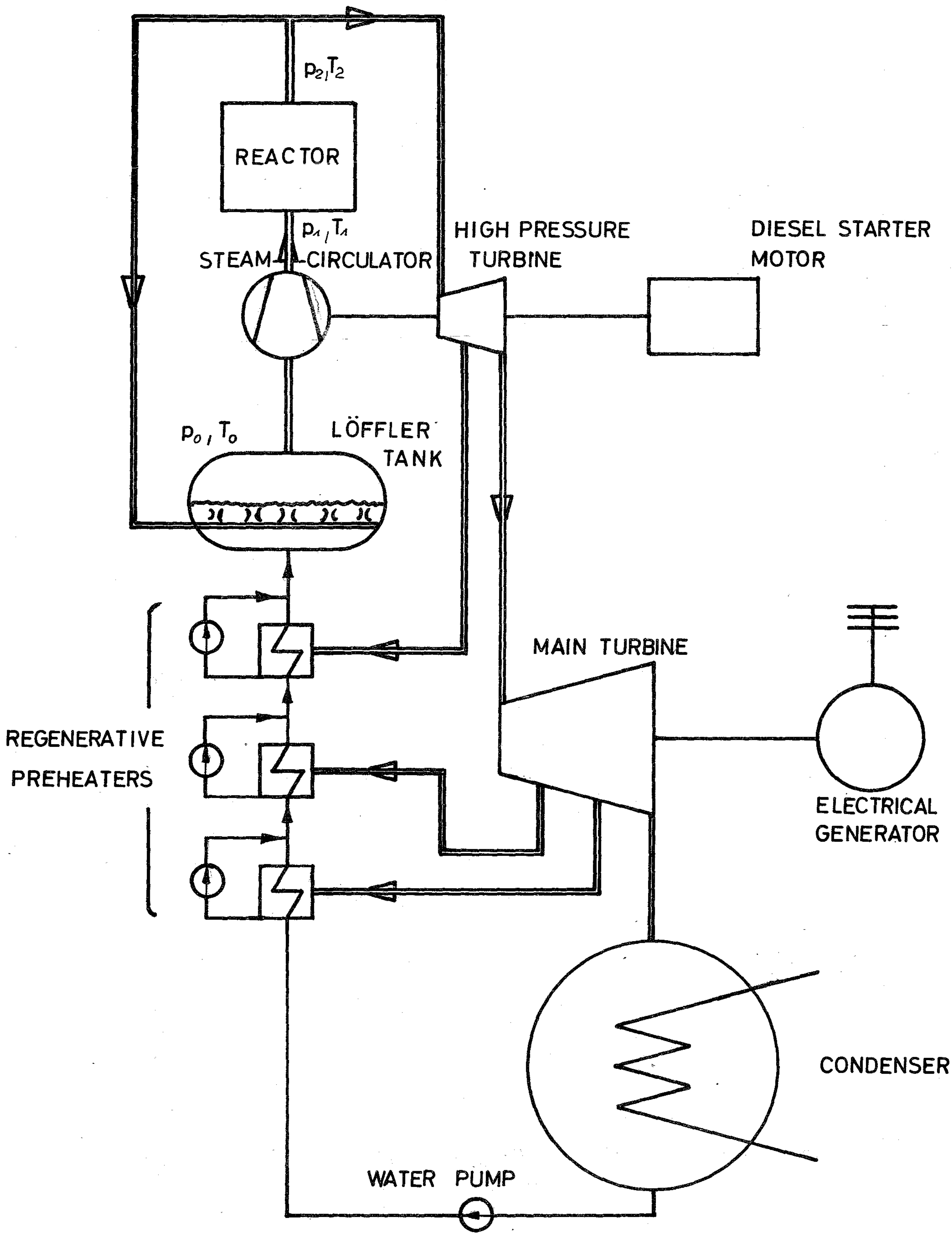

FIG. 1 SCHEMATIC FLOW DIAGRAM OF STEAM COOLED POWER PLANT

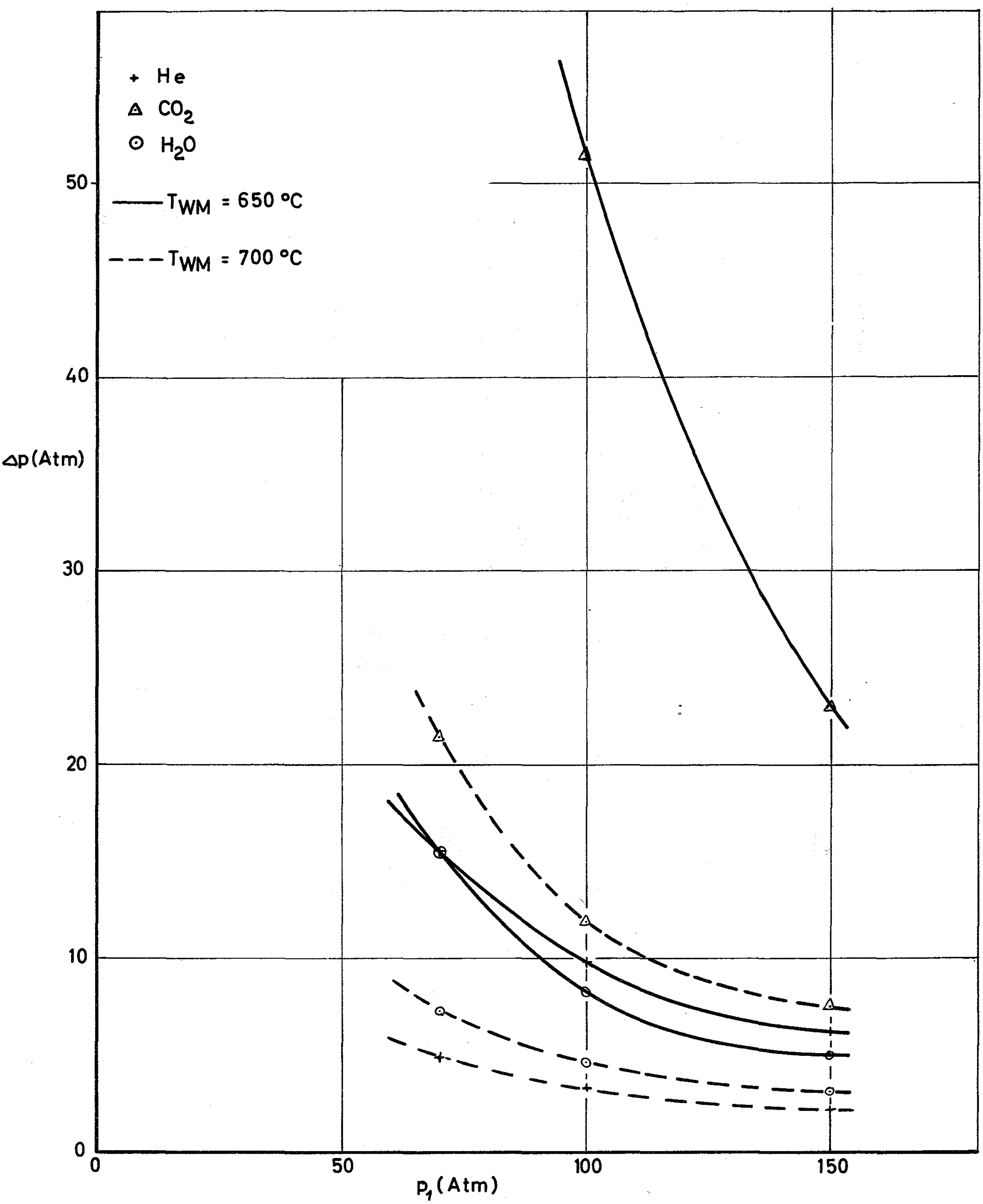

FIG. 2 PRESSURE DROP ACROSS THE CORE FOR A CONSTANT NETT ELECTRICAL OUTPUT OF 1000 MWe.

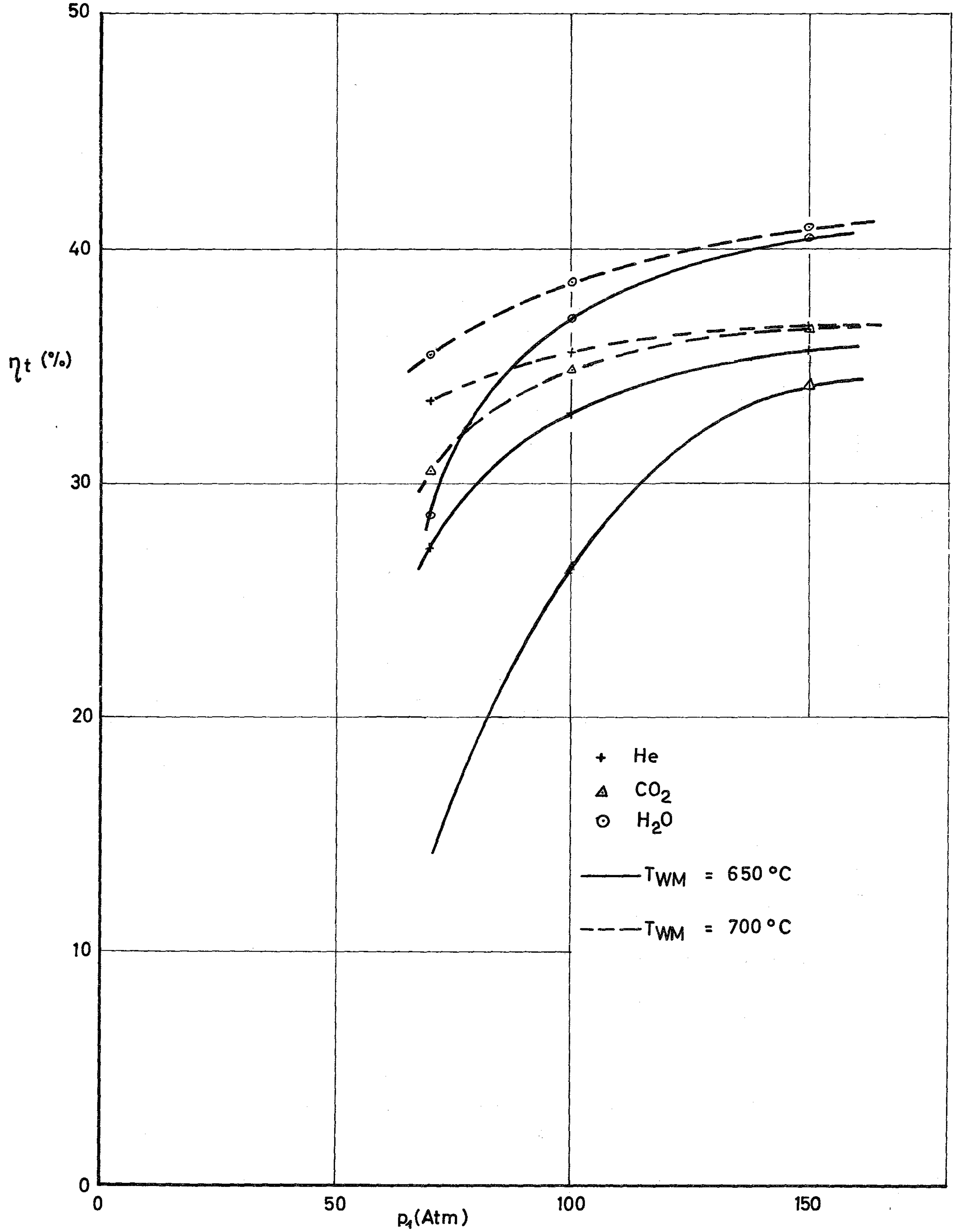

FIG. 3 NETT ELECTRICAL EFFICIENCY FOR A CONSTANT NETT ELECTRICAL OUTPUT OF 1000 MWe.

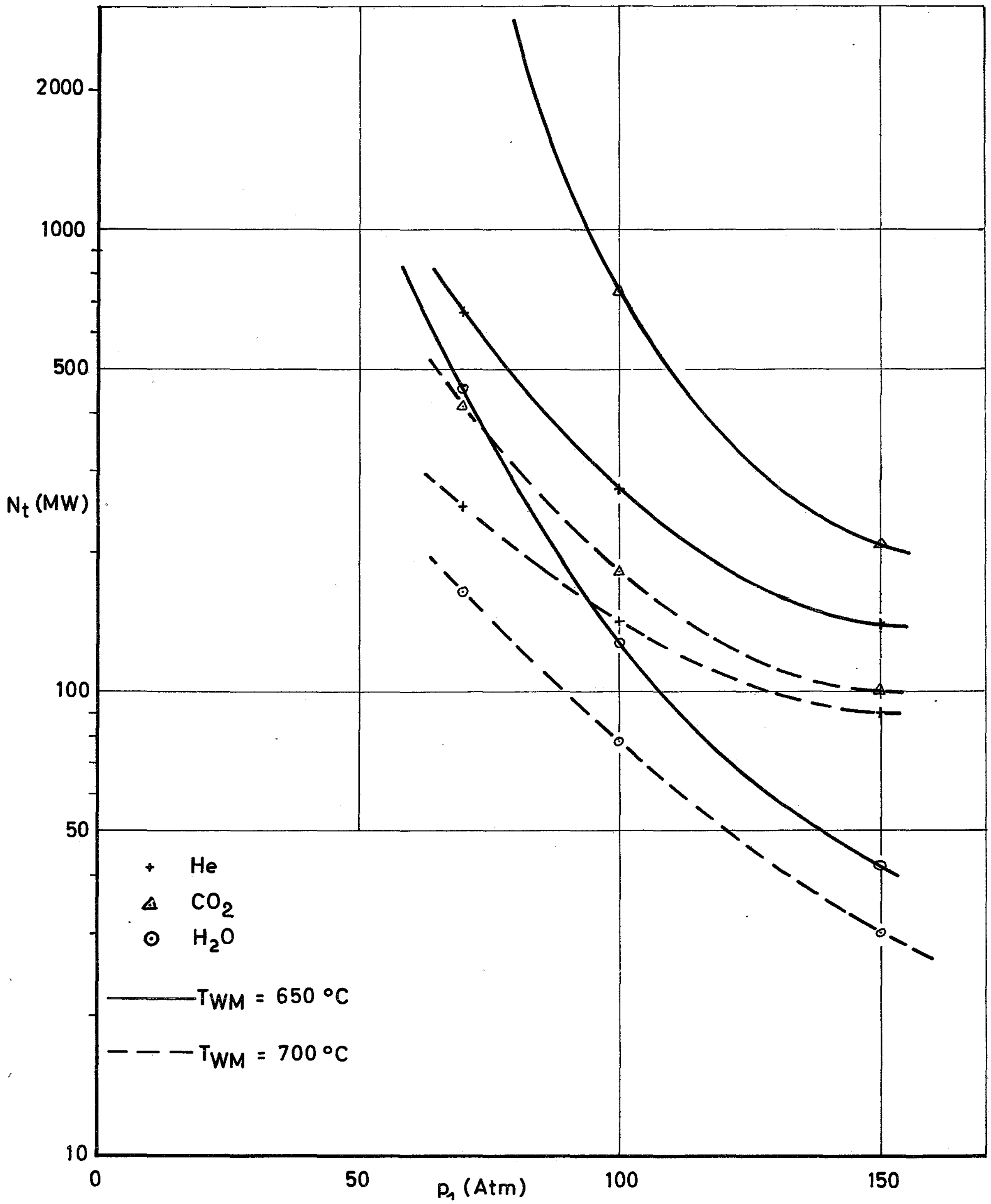

FIG. 4 PUMPING POWER REQUIRED BY THE COOLANT PRIMARY CIRCUIT FOR A CONSTANT NETT ELECTRICAL OUTPUT OF 1000 MWe.

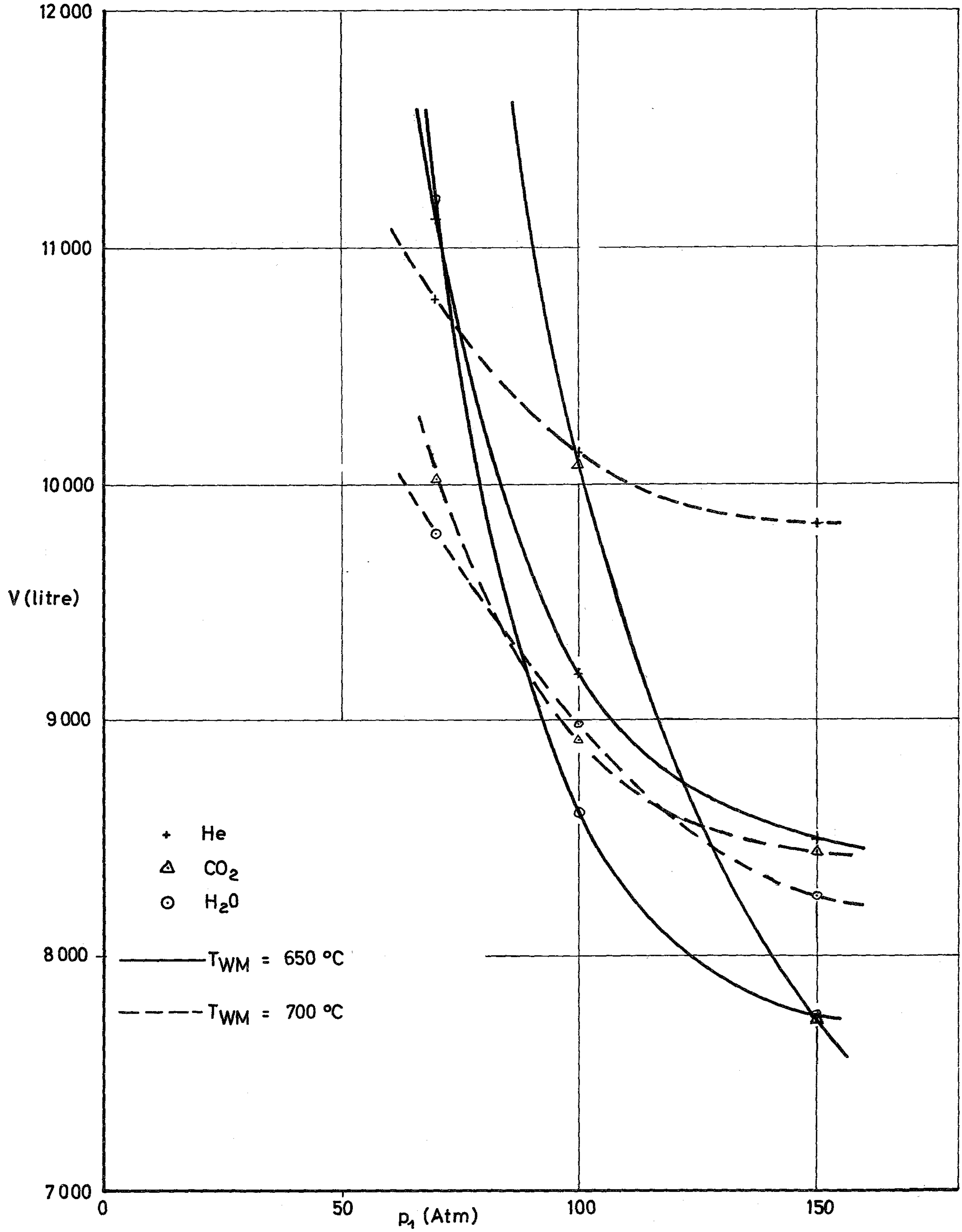

FIG. 5 CORE VOLUME FOR A CONSTANT NETT ELECTRICAL OUTPUT OF 1 000 MWe .

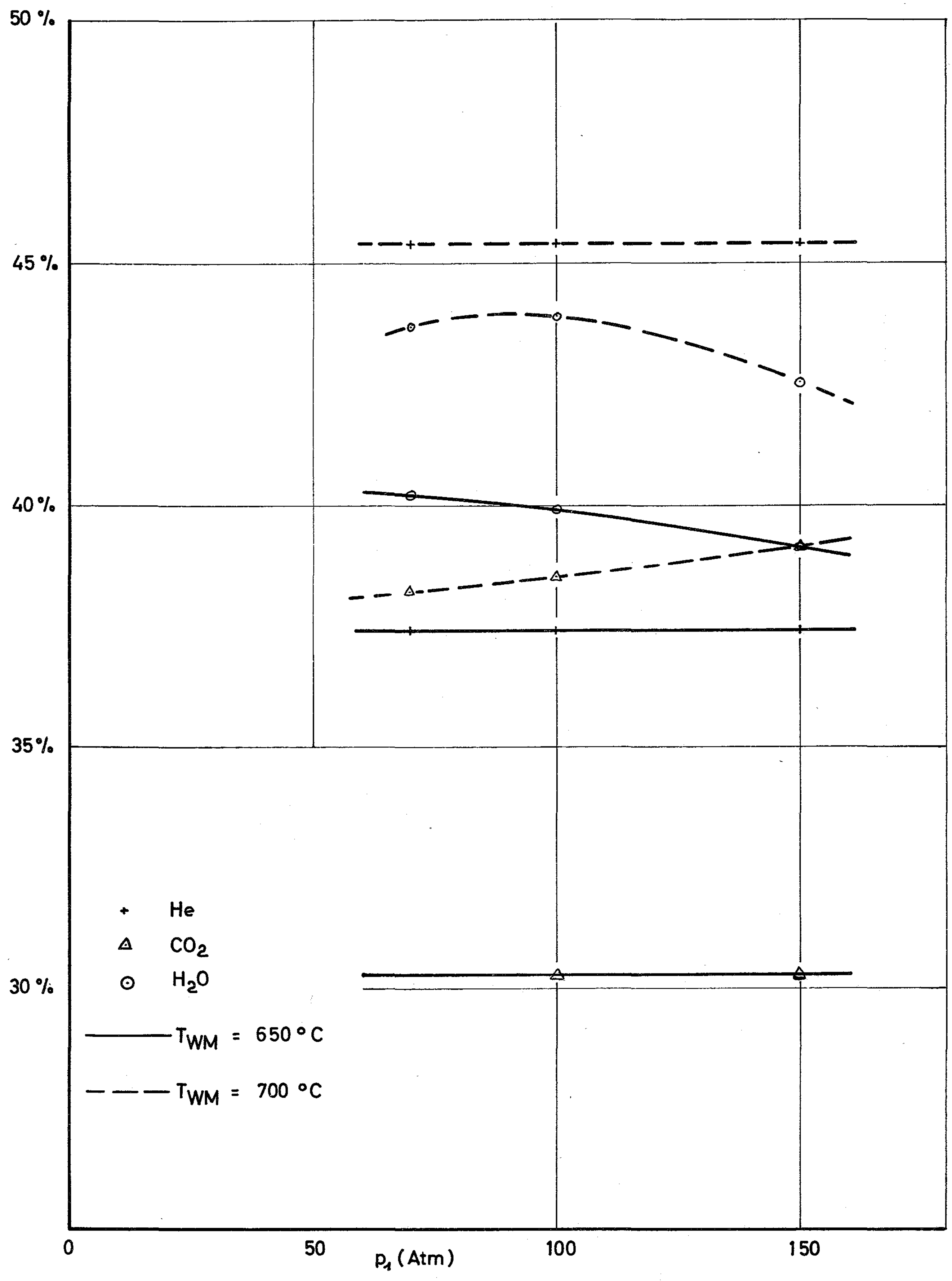

FIG. 6 CORE COOLANT FRACTION FOR A CONSTANT NETT ELECTRICAL OUTPUT OF 1000 MWe.

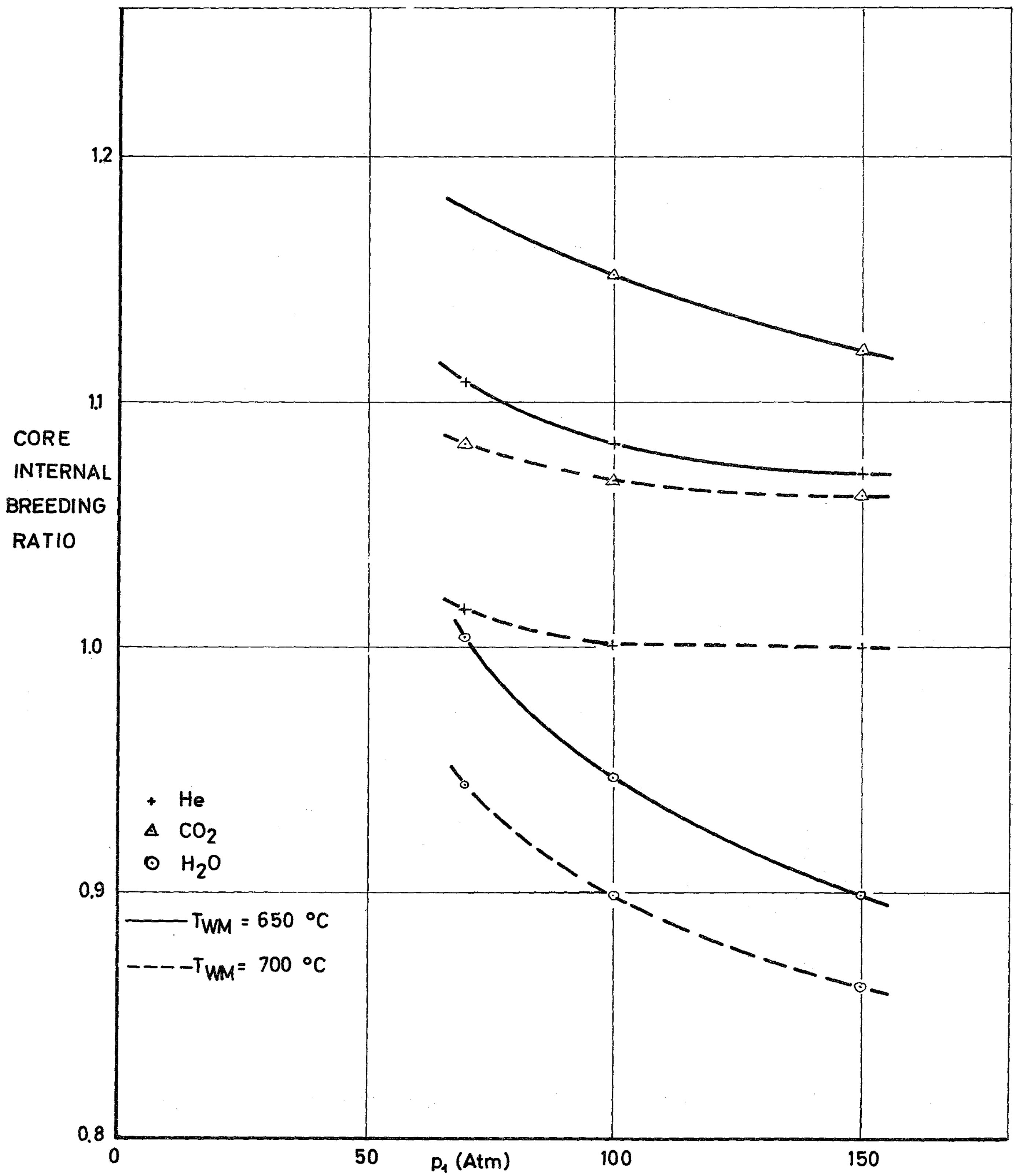

FIG. 7 CORE INTERNAL BREEDING RATIO.

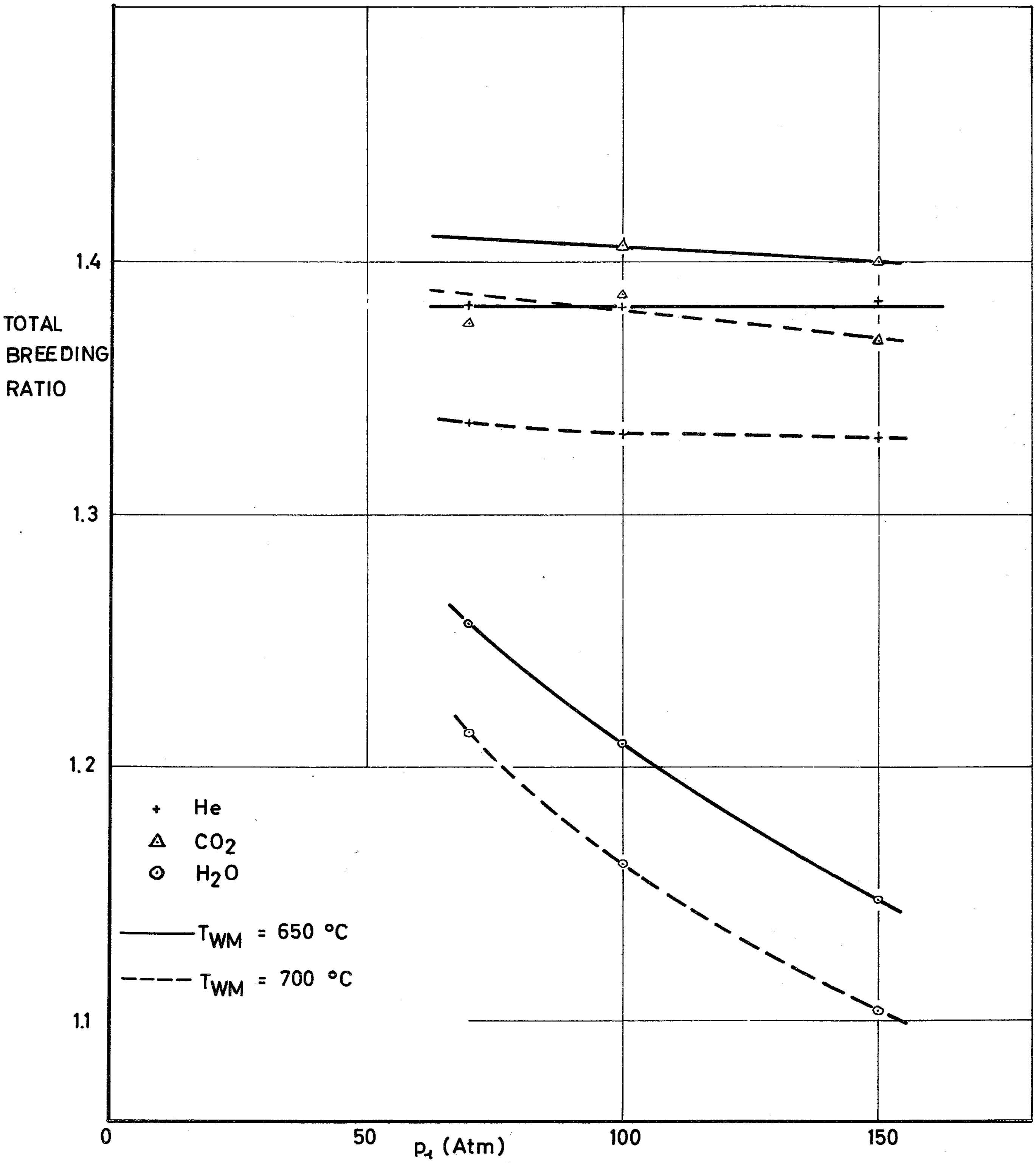

FIG. 8 TOTAL BREEDING RATIO

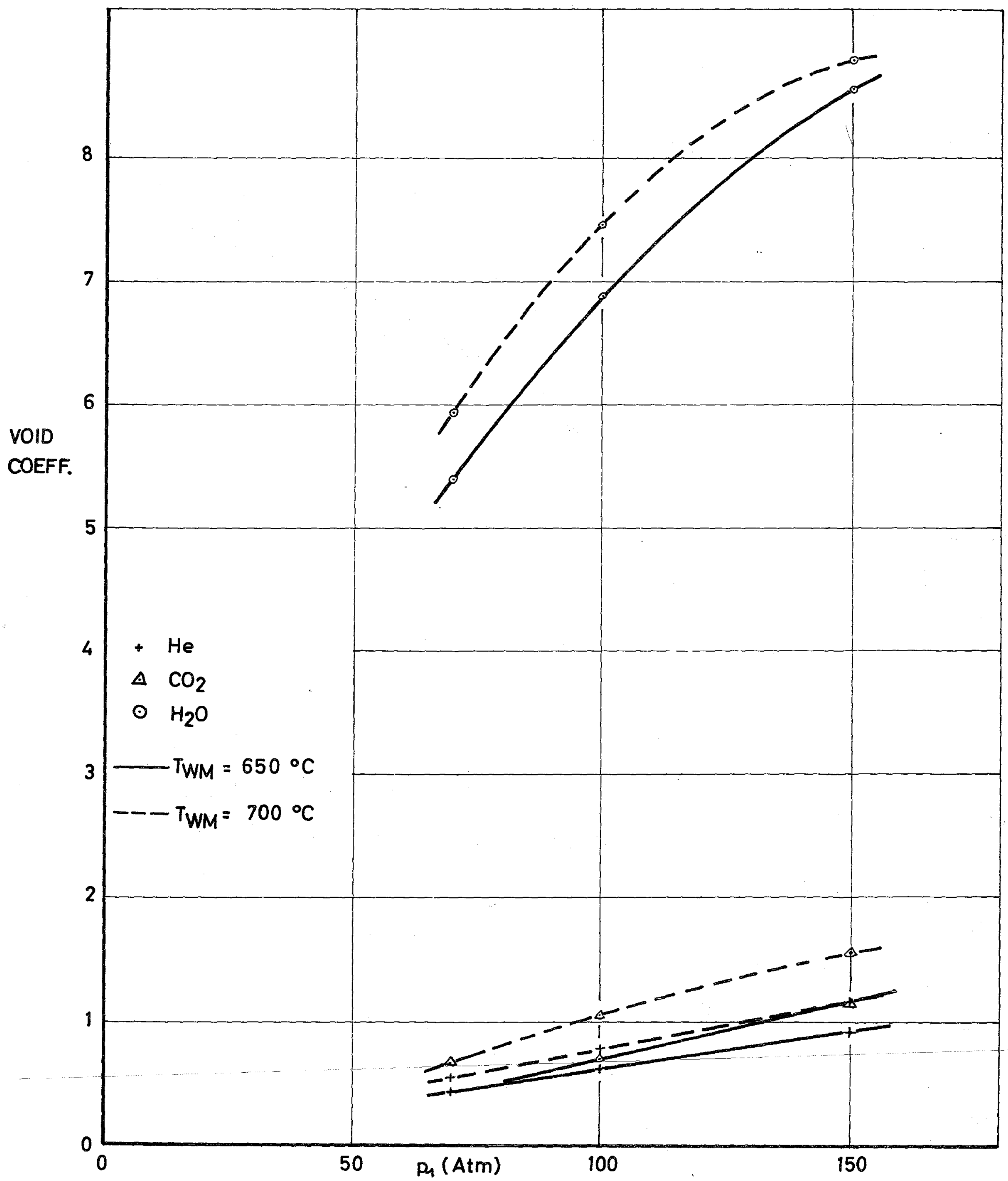

FIG. 9 VOID COEFFICIENT ($)

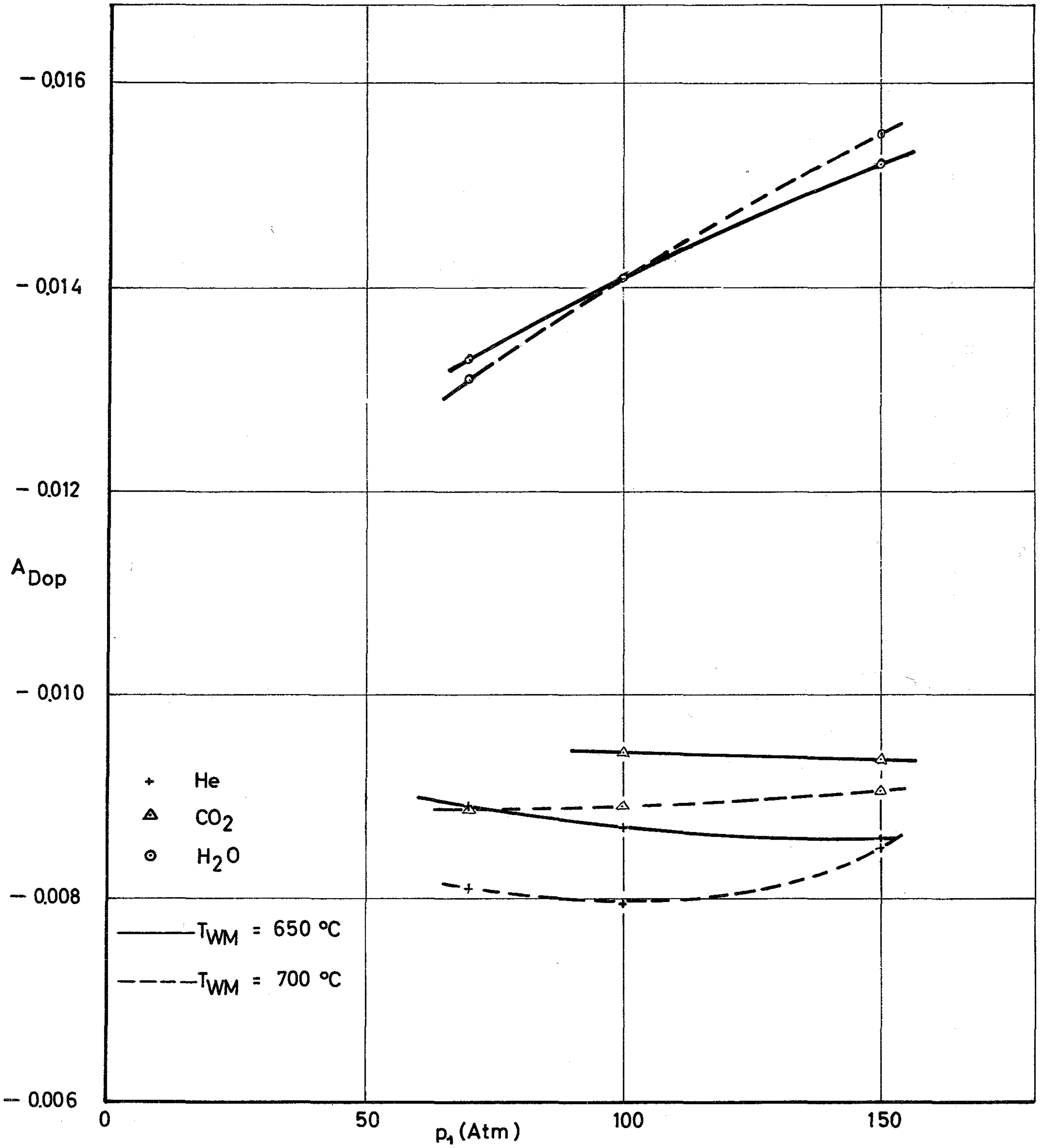

FIG. 10 DOPPLER COEFFICIENT CONSTANT A_{Dop}

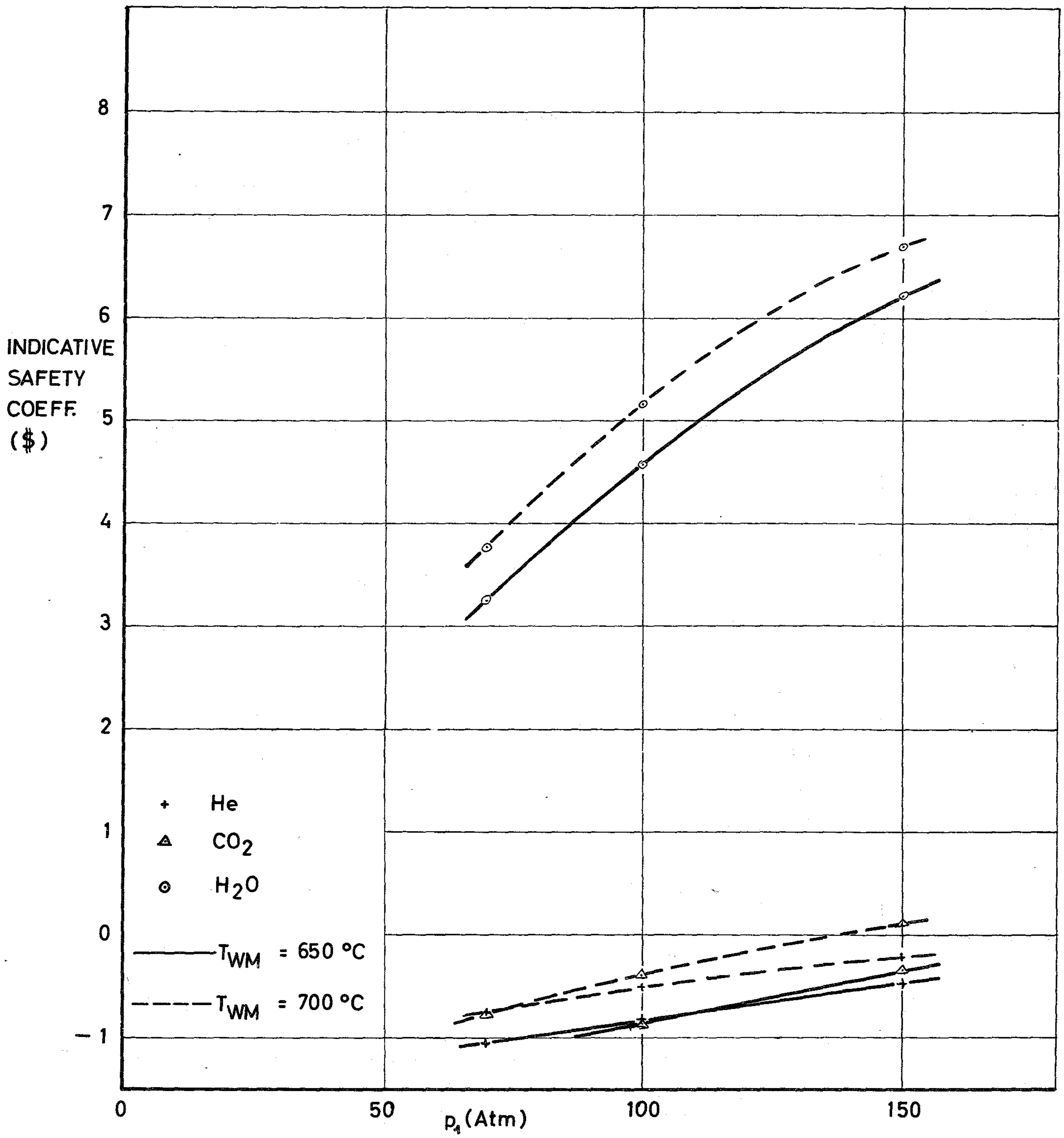

FIG. 11 INDICATIVE SAFETY COEFFICIENT

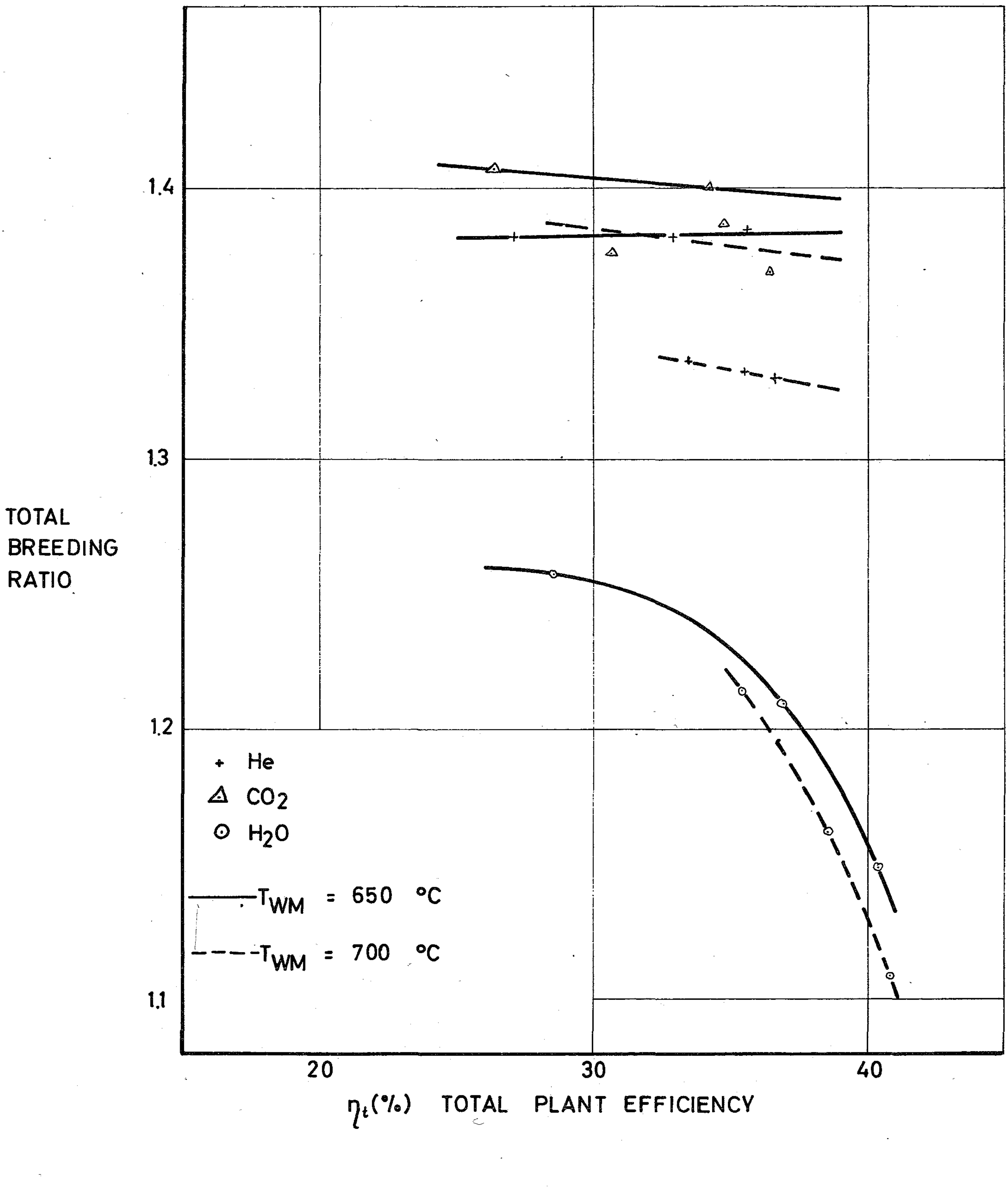

FIG. 12 BREEDING VERSUS ECONOMY

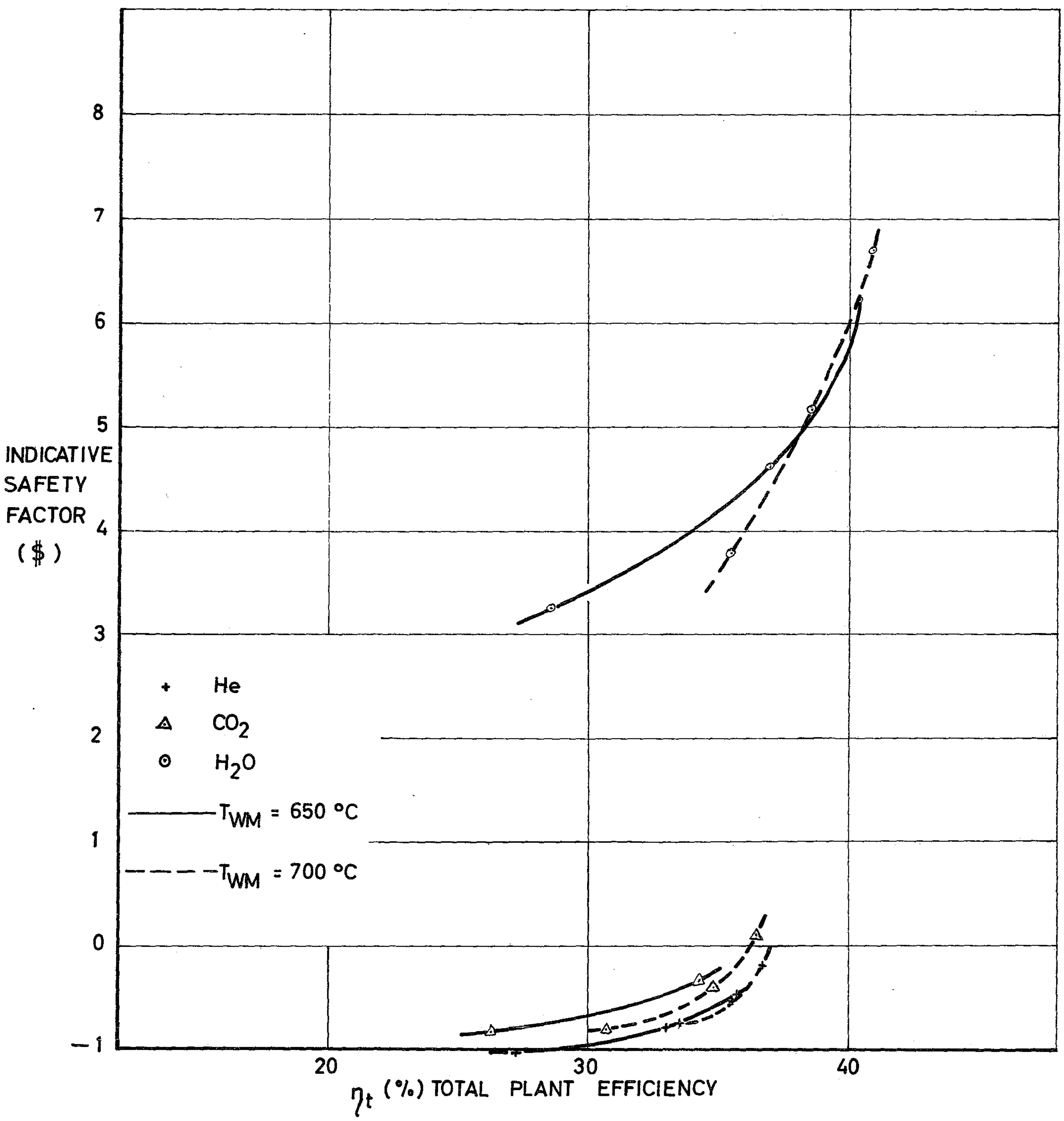

FIG. 13 SAFETY VERSUS ECONOMY