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Pressure Tube and Pressure Vessel Reactors; certain comparisons

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PRESSURE VESSEL AND PRESSURE TUBE DESIGNS; CERTAIN COMPARISONS

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Summary

In a comparison between pressure tube and pressure vessel type reactors for pressurized D_2O coolant and natural uranium, one can say that reactors of these two types having the same net electrical output, overall thermal efficiency, reflected core volume and fuel lattice have roughly the same capital cost. In these circumstances, the fuel burn-up obtainable has a significant influence on the relative economics. Comparisons of burn-up values made on this basis are presented in this report and the influence on the results of certain design assumptions are discussed. One of the comparisons included is based on the dimensions and ratings proposed for CANDU. Moderator temperature coefficients are compared and differences in kinetic behaviour which generally result in different design philosophies for the two types are mentioned. A comparison of different methods of obtaining flux flattening is presented.

The influence of slight enrichment and other coolants, (boiling D_2O and gases) on the comparison between pressure tube and pressure vessel designs is discussed and illustrated with comparative designs for 400 MW electrical output.

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Pressure vessel and pressure tube designs; certain comparisons

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1. Introduction

The primary goal for work in Sweden on nuclear reactors has been the attainment of economic power production with a system which could operate on natural uranium, so as to give a certain degree of independence of fuel supplies and a favourable trade balance. This and some subsidiary aims of the Swedish programme such as an interest in nuclear district heating have led to a concentration on D_2O as moderator. Most work has been done on the pressure vessel type of design for this moderator as adapted for instance in the Ågesta reactor (previously called R3/Adam),[1), 2)] the current designs for Marviken [3)] (previously called R4/Eve) and certain preliminary designs for larger stations of the pressurized D_2O type. Boiling D_2O with pressure vessel design is regarded as a logical future development for this reactor type.

The basic aims of economic power production with a system which could operate with natural uranium could also be fulfilled with D_2O moderated reactors of other types than those described above, in particular designs with pressure tubes using pressurized D_2O or various other coolants. Evaluation of various factors for and against pressure vessels as compared to pressure tubes have tended to change during the past five or six years partly as a result of new information on the subjects of reactor physics and materials, partly as a result of the development of certain design philosophies in different organisations. It is still not possible to prove which system is the best from the aspect of behaviour of materials, prevention of leakage, facility of repairs of fuel changing machines and other engineering components, but it is possible to make certain generalized comparisons from the aspect of reactor physics and kinetic behaviour, and to draw certain conclusions concerning the influence of these factors on economics. The present paper attempts to make such comparisons with a view to testing the logic of proceeding with the Swedish development line as compared with alternative types.

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2. Designs chosen for comparison

The largest part of this paper concerns comparisons of pressurized D_2O reactors of the pressure tube and pressure vessel types. For the pressure tube type it is convenient to select the basic CANDU design[4] illustrated in fig. 1 as a basis for discussion. This design has horizontal pressure tubes of Zircaloy insulated by a gas gap from the Zircaloy calandria tubes, a cold moderator, the heat of which is discarded, and bi-axial on-load fuel changing. The irradiation exposure in the central channels can be increased compared to the peripheral channels to obtain the desired degree of flux flattening. Because of the bi-axial system of fuel element changing, there is no axial reflector, whilst the current design envisages a relatively thick radial reflector.

For the pressure vessel type a variety of different designs have been proposed and the one chosen for this discussion has the virtue that it has a relatively large number of features in common with the pressure tube design thus facilitating the comparison. The design is illustrated in fig. 2 and can be regarded as a development of the concepts for Ågesta and Marviken. The pressure vessel has a domed lid filled with a steam cushion used for pressurizing the reactor, the top layer of water being insulated from the coolant by normally stagnant layers of water. The lid has only one central fuelling port, through which a manipulator loads and removes fuel whilst the reactor is on load. Each channel contains two fuel elements which can be axially reversed to obtain reasonably uniform axial irradiation exposure, and the elements can be transported radially using the manipulator, without leaving the tank. The shroud tubes stay in the pressure vessel during fuel element changing operations, but can be exchanged if damaged by removing the lid. The D_2O circulates first downwards through the moderator space and then upwards through the fuel channels, and the resulting overpressure on the shroud tubes is assumed to be met by slightly corrugating the tubes. The design would normally allow for a uniform axial and radial reflector thickness.

Since the moderator and coolant channels are connected in series, the design can utilize a negative moderator temperature coefficient to some extent for self regulation with load changes, pump failures and power surges, though stronger self regulating properties can be obtained in other pressure vessel designs, e.g. that described in another paper before this conference.[5)]

It should be noted that essential components of the above pressure vessel design such as the fuel manipulator are at the drawing board stage whereas similar components for the pressure tube design are already under manufacture for the prototype reactor NPD-2. From this point of view the pressure tube design has an advantage which will be disregarded in the subsequent discussion.

3. First idealised comparison of burn-up for similar core and reflector dimensions.

As a first, somewhat academic, comparison we may assume that the pressure vessel design is dimensioned in such a way that it has exactly the same fuel element cluster (19 x 14, 5 mm rods), and the same spacing between rods and between elements as the pressure tube design (CANDU). The moderator temperature for the pressure vessel design can be chosen to give the same overall thermal efficiency as for the pressure tube design (which results in a moderator temperature of about 225° C) and the ideal burn-up[*)] of the two systems may then be compared. As discussed in more detail later, pressure vessel and pressure tube designs having the same core and reflector dimensions, net electrical output and overall thermal efficiency may be expected to have roughly similar capital costs, so that a comparison of the ideal burn-up obtained is significant for the overall economy. The moderator temperature of about 225°C required by this comparison does not result in excessive pressure vessel dimensions, as shown for instance by column 2a in table 1 though the dimensions could be made more suitable for pressure vessel technology by using a greater height/diameter ratio for the core.

The calculations presented in table 1 columns 1 and 2a and described in greater detail in Appendices 1 and 2 indicate that the pressure vessel design in these circumstances has a higher ideal burn-up than the pressure tube design, the difference being about 1100 MWd/ton. This is illustrated also by the reactivity burn-up curves fig. 3. It should be mentioned, however, that Swedish designers consider that the value of 0, 6 chosen in CANDU for the crossectional channel area ratio of coolant to

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*) The expression "ideal burn-up" signifies in this paper the burn-up calculated for a hypothetical continuous fuel element changing system where all degrees of irradiation exist at every point of the lattice. Flux flattening is disregarded. It is expressed in MW days per metric ton of U in this paper.

fuel, A_c/A_f, is very low, particularly for pressure vessel designs, where it is desired to select a higher value for the following reasons:

a) to avoid obtaining a large positive void coefficient according to the calculations presented in fig. 4 and to obtain a stronger negative moderator temperature coefficient of reactivity, according to the calculations presented in fig. 5. A strong positive void coefficient is considered a disadvantage from the aspect of safety and a negative moderator temperature coefficient is an advantage from the aspect of self regulation in pressure vessel designs where the coolant and moderator are closely coupled.

b) to reduce the power rating of the back-up supplies to the pumps necessary to guard against supply failures (since an increase in the coolant to fuel area ratio greatly reduced the pumping power requirement), and to improve cooling with natural circulation in vertical coolant channel designs.

c) to reduce the risk of damage due to bowing of rods when using relatively long rods.

In addition, this reduces of course pumping power requirements.

To satisfy these requirements according to current data, values of A_c/A_f of about 2, 0 are visualised for a reactor having the channel ratings selected for CANDU. The influence of this on burn-up is illustrated by curves 1 and 2 in fig. 6 which show that the advantage in burn-up of the pressure vessel design increases with increasing values of A_c/A_f (since this increases the amount of structural material in the pressure and calandria tubes), reaching the value 3700 MWd/ton for an area ratio of 2, 0.

The above discussion indicates that the advantages of the pressure tube reactors regarding a lower moderator temperature is not sufficient to compensate for the disadvantage of more structural material and extra gas gaps in the core, resulting in a net disadvantage regarding ideal burn-up of 1100 to 3700 MWd/ton, according to the value chosen for A_c/A_f in the range 0, 6 to 2, 0.

It must, however, be emphasized that current methods of physics calculations still contain a considerable degree of uncertainty especially for high values of burn-up, so that the results cited above must be treated with caution. The numerical values depend also to some extent on design assumptions such as the permissable stress for the pressure tubes (based in this case on Canadian values) and the minimum practicable thickness of shroud tubes.

4. Practical burn-up values

a) without flux flattening

If the bi-axial system of continuous on-load fuel element changing used for the CANDU design is employed without radial flux flattening, it gives the same radial flux distribution as for our postulated ideal system, and a slightly more peaked axial flux distribution. This results in a slightly (2 to 3%) higher burn-up than for the ideal system. The same is true for the fuel element changing system assumed for the pressure vessel design, where the elements are axially inverted after half the burn-up is reached, provided this takes place continuously i. e. one element at a time. This is illustrated by fig. 7 curves 2 and 4.

For both types of systems, the flux will vary slightly in adjacent channels due to differences in irradiation, and this will slightly reduce the burn-up. The above discussion suggests that, in the absence of radial flux flattening, both systems give a burn-up which is about equal to the ideal burn-up.

b) with radial flux flattening

With the bi-axial system of fuel loading, it is currently envisaged to achieve flux flattening by allowing the fuel elements in the central channels to achieve a higher exposure than in the periphery. For the pressure vessel system on the other hand it is proposed to achieve flux flattening largely by transposing elements from the peripheral zone towards the central region after a given burn-up has been reached which can be done rather conveniently in that design without removing the fuel elements from the reactor vessel. As illustrated in fig. 8 (see also Appendix 1) the latter system reduces by a factor of about two the reduction in burn-up caused due to flattening the flux by a given factor. From this point of view the proposed pressure vessel design has an advantage compared to the proposed pressure tube design, though it is possible that the fuelling machine: the pressure tube reactor can also be adapted to perform radial position of fuel elements at the cost of some increase in complic

The differences in the burn-up reduction due to flattening betwe two designs for given total volume of the reflected core are acc if the pressure tube design has no axial reflector (which is conv for bi-axial fuel element changing) and a very thick radial refle failing to use the thermal flux in a uniform manner, whilst the p vessel design uses an approximately uniform axial and radial re thickness. In these circumstances a higher relative radius is ne for the flattened zone in the pressure tube design for given latti meters than in the pressure vessel design.

5. Design margins

In Sweden, more conservative values are used at present regar maximum allowable heat rating per cm fuel rod, the temperatur against bulk boiling at the outlet of the fuel channels, and the ra working pressure to design pressure than the corresponding val for CANDU. The results for typical current Swedish design pra illustrated in column 3 of table 1, and shown graphically by poin fig. 6, retaining, however, the reflected core volume, fuel eler diameter overall thermal efficiency and net electrical output for The change in design practice which is summarized in item 5 to table does not significantly affect the comparisons between the t actor types for the coolant to fuel area ratios up to about 1, 5 i. e maximum value for which the pressure tube design can achieve lity with the assumed volume. The actual value of A_c/A_f used fc lumn 3 in the table is, however, 2. 0 from fig. 6. It is of intere: note that the pressure vessel design according to current Swedi: tice, V_s would still give a practical burn-up not very much infe: that obtained for the CANDU practice (T_c) though the latter invo siderable reductions in a number of design margins.

6. Lattice limitations

For pressure tube designs, the theoretical cost minimum is oft ned at a lattice spacing too low to be practicable from an engine aspect, and this tends to increase costs slightly, especially for reactors with very large heat ratings, where one obviously desires to reach high heat rates per unit core volume.

For pressure vessel designs of the type described, this particular design limitation does not exist, as a more close lattice pitch is possible from the engineering aspect, but there is, instead, a desire to adopt low volume ratios of moderator to fuel (i. e. low lattice pitches) and high values of A_c/A_f as mentioned earlier in order to obtain negative reactivity coefficients. Even in this case some departure from the theoretical point for minimum cost is thus desirable.

In both cases, the cost increase compared to the theoretical optimum is small, i. e. a few percent.

7. Output limitation

In a pressure tube design an increase in output can obviously be obtained by increasing the number of pressure tubes without obtaining fundamentally new problems in design or manufacture, though in practice it would generally be desirable to let an increase in output be accompanied by some increase in the channel rating. For pressure vessel designs it is not possible to increase the volume of the vessel in proportion to the reactor output indefinitely, and this has resulted in a fear that pressure vessel designs for natural uranium have a definite limit in the possible output.

Some design studies having a bearing on this subject have recently been made in Sweden. Table 2, design 6, gives data for a preliminary design for 400 MW electricity. Fig. 9, curve 1, shows the calculated optimum diameter plotted against net electrical output and indicates that an increase in output by a factor 2 corresponds to an increase in the optimum diameter by only about 17%. Curve 2 shows furthermore that a reduction in diameter by about 10% (and a reduction in the reactor volume by nearly 30%) increase the overall cost per kWh by only about 3% compared to the minimum cost. The curves indicate that diameters of 6 to 6.5 m corresponding to wall thicknesses of 85 to 95 mm with low alloy steel are of interest for reactors with an output of 400 to 500 MW electricity per reactor. Swedish manufacturers believe that vessels having such dimensions can be manufactured with the equipment envisaged towards the end of the nineteensixties. For most Swedish sites it should be possible to transport the vessels by water in one piece, but even partial fabrication on site may be considered practicable at a later date when additional experience is available. There seems accordingly no reason to fear that pressure vessels will limit the output of pressurized D_2O reactors by the time large units require to be built.

The fact that the cost of pressure vessel designs might be increased by a few percent due to a departure from the apparent optimum volume for the largest outputs does not make pressure vessel designs less favourable than pressure tube designs at large outputs since pressure tube designs for very large outputs require to increase the value of A_c/A_f if optimum heat ratings are used, which also results in slightly, increased costs.

8. Power costs

As mentioned earlier, it is believed that pressure vessel and pressure tube designs having the same total volume of core plus reflector, the same net electrical output and the same overall thermal efficiency have approximately the same capital cost. Actually the turbo generator will probably be a little cheaper for the pressure tube design because of the somewhat higher steam conditions*) and the building may also be cheaper because of the reduced mass of D_2O at high temperature, which reduced the possible pressure build-up in accidents. On the other hand the cost of the pressure tubes (19 tons of Zircaloy) for CANDU and the calandria is estimated to be somewhat higher than the cost of a pressure vessel, and the fuelling machines for bi-axial fuelling are more complicated and therefore probably more expensive than the machine for changing fuel in the pressure vessel design. Also the D_2O investment for a given volume is greater in the pressure tube design than the pressure vessel design because of the lower moderator temperature. These factors probably cancel roughly so that one cannot expect to find significant differences in the capital cost.

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*) The influence of higher steam conditions on the thermal efficiency is offset by the heat discarded by the moderator cooling circuit.

The differences in burn-up mentioned in para. 4 correspond to a cost difference of 0.15 to 0.5 mills/kWh according to the basic cost data used in this paper. On the other hand, there are considerable uncertainties in the determination of the burn-up, and the above statement concerning capital costs is of course very approximate. The calculated difference in fuel costs cannot, therefore, be taken as a proof of economic superiority of the pressure vessel design for natural uranium, though it does give a hopeful indication concerning possible future economy.

9. Slight enrichment

A slight degree of enrichment can be used to increase the burn-up by very considerable amounts for D_2O moderated reactors as well as to reduce the reactor volume, at the price of an increase in the cost of fuel per kg. To preserve negative void and moderator temperature coefficients at the high burn-up values, resonance absorption should however be increased, which brings about a reduction in the desired moderator to fuel volume ratio and an increase in the desired coolant to fuel area ratio. From what has been said previously in paragraphs 3 and 6, it will appear that pressure vessel designs can more easily be adapted to these conditions than pressure tube designs.

An analysis of the economics of slight enrichment requires knowledge of the influence of enrichment of the cost per kg of delivered fuel. For the cost of nuclear material price lists are published by the A.E.C. and generally used for such calculations. For the cost of fabrication and conversion on the other hand, no generally established figures exist. Fig. 10 curve 1 shows the cost data used for this item in this paper for natural uranium elements, whilst the shaded band of figures, curve 2, shows corresponding costs used for slightly enriched elements. The difference in level between these curves is caused largely by the omission of the UF_6 conversion process for natural uranium elements, as well as reduction of interest charges and uranium losses during fabrication, and lastly a slight cost reduction due to a reduction in precautions necessary during fabrication. On the same figure are shown on the one hand the Canadian estimate for the cost for natural uranium fuel elements and a cost figure cited by the U.S.A. in connection with the EURATOM agreement for enriched element. It will be seen that these figures differ by very large amounts from the costs assumed in this paper, the Canadian costs being much lower*) and the U.S.A. costs much higher. It would be of great value if the specialists in the various countries could jointly clarify the reasons for these discrepancies. Table 2 shows technical data and approximate cost estimates for a number of 400 MW reactor designs based on the technical status considered practicable for 1970. Design 6 represents pressurized D_2O design for natural uranium, whilst design 7 shows a pressurized D_2O reactor of pressure vessel design for 1 % enriched uranium. The enrichment was used to increase the burn-up from 8550 to a mean value of 14 200 MWd/ton and to reduce the reflected volume by 36 %, and which is estimated to achieve a net reduction in costs by about 5 %. This design has not however been optimized so that a slightly larger reduction is probably obtainable. The fuel cycle costs are shown in greater detail in table 3 which also indicates that graded enrichment employing a certain proportion of natural uranium elements is more advantageous than uniform enrichment, since it avoids submitting all the fuel through extra processes such as the UF_6 conversion process. It should be mentioned that the calculations for graded enrichment were based on the simplifying assumption that the rate of plutonium production is independent of the distribution of enrichment.

It might be argued that the selection of slight enrichment is a fundamental departure from the policy of independency of fuel supplies and a favourable trade balance. On the other hand it is possible to design the slightly enriched system in such a way that the fuel elements can be replaced in an emergency by natural uranium elements with a higher moderator to fuel volume ratio and somewhat lower rod diameter, giving the same electrical output, though at a much lower burn-up and somewhat higher fuel cost per kWh. The cost for this emergency operation would indeed be slightly higher than for a system optimized from the start for natural uranium, but would be quite satisfactory

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*) One reason for the lower Canadian cost may be an assumption that the raw material prices for uranium will drop, since the Canadian cost figure in fig. 10 was calculated as a difference between the total cost of fuel element in ref. 4 and the current U.S.A. price for natural uranium.

as an emergency measure. As regards the trade balance, this would still be more favourable with 1 % enriched uranium than with say 2 % to 3 % enriched uranium necessary for H_2O moderated reactors.

Table 3, design 10, shows typical fuel cycle costs for a large H_2O moderated reactor. It appears that the slightly enriched D_2O moderated reactor can achieve a saving in fuel costs of about 1.3 mills/kWh, and would incur extra capital charges and leakage costs of about 1.1 mills/kWh for Swedish accountancy practice according to the discussion in Appendix III. The latter figure is obtained by considering only the differences in the installation of pressurized reactors of the pressure vessel type for D_2O and H_2O and should be reasonably accurate since these two types of station have so many components in common. These figures suggest that slightly enriched pressurized water reactors for D_2O can economically compete with reactors for H_2O when the comparison is based on large units and Swedish accountancy practice. The comparison, however, assumes that both types have reached the same stage of technological development, whereas at the present time H_2O reactors have an advantage in this respect.

10. Boiling D_2O

Direct cycle boiling reactors have the advantage of higher efficiency for given design pressure than pressurized water reactors as well as the advantage of omitting the main heat exchangers. They have the disadvantages of

a) more acute leakage problems, especially in the condenser

b) limitations in the core imposed by voids.

Whilst D_2O moderator accentuates problem a) compared with H_2O moderator, it makes b) less severe, since it is possible with D_2O to keep the void coefficient very slightly negative by choosing suitable coolant to fuel area ratios, thus avoiding the very strongly negative void coefficients present with H_2O, which limit the power rating for large H_2O boiling reactors. This advantage of D_2O exists however primarily with pressure vessel designs, since pressure tube designs can hardly afford the large coolant to fuel area ratios which would be necessary. Fig. 11 shows the design assumed for the boiling D_2O reactor data in this paper. It is based on natural circulation.

The advantages of the pressure vessel design from this point of view are accentuated if the reactor is designed for natural circulation, as this requires still larger coolant to fuel area ratios, as illustrated by Fig. 12, where it is assumed thatthe largest permissible volume fraction of voids from the aspect of burn-out and for hydrodynamic instability (with the assumed degree of throttling at the channel inlet) is 81 %. The method of calculation used for this figure is described in ref. 6.

Table 2, design 8, shows a design for a direct cycle boiling water reactor indicating that one might expect a saving of 0 - 16 % of the cost per kWh compared to the cost for pressurized D_2O, depending primarily on the amount by which the costs of leakage are estimated to exceed those for pressurized D_2O, and the somewhat uncertain estimates concerning the costs of extra measures required to minimize leakage for the direct cycle.

11. Internal superheating

Internal superheating appears to offer only marginal economic with D_2O moderated systems as long as stainless steel has to be used as the canning material but would offer significant improvements in economy if zirconium alloys suitable for superheated steam were developed.

Internal superheating can be used without appreciable complication to the core in pressure tube designs of the pressurized D_2O or boiling D_2O types though there would be certain additional problems in control.

With pressure vessel designs of the pressurized water type, internal superheating would introduce complications which make its merits very doubtful, whereas with pressure vessel designs of the boiling water type, internal superheating could be used without introducing such a high degree of complication, as illustrated for instance by the designs presented in ref. 7. The problem is rather similar to that for H_2O moderated boiling reactors, where designs incorporating internal superheating are also being seriously considered. An alternative method which avoids the use of headers for the superheated steam inside the pressure vessel is illustrated by the broken lines in fig. 11. It is assumed in this case that the superheater elements are uniformly distributed over the entire core except a narrow region near the reflector, thereby producing a large pitch of the holes in the bottom of the pressure vessel.

Summarizing one can say that internal superheating will probably at some future date be used with advantage in both pressure tube and pressure vessel reactor types, though in the latter case it will be used only in conjunction with the boiling type of reactor.

12. Other coolants

The pressure tube design lends itself for obvious reasons better than the pressure vessel design to the use of other coolants than D_2O. The only one amongst these other coolants which has been studied in reasonable detail in Sweden is CO_2 gas, which offers the attainment of very high efficiencies on the assumption that satisfactory beryllium cans are developed at a reasonable price, and satisfactory solutions to the internal insulation of pressure tubes are developed.

The results from one of the approximate design studies made for this reactor type in Sweden[8] are summarized as design 9 in table 2. Just as the other designs in this table, it has been optimized for 400 MW electricity, and in common with the designs 6 and 8 it uses natural uranium as fuel. It appears however that the advantage of a superior thermal efficiency compared to pressurized D_2O and boiling D_2O are offset by the influence of the lower optimum heat rating of the fuel as dictated by considerations of heat transfer from the can to the gas, the relatively large and expensive gas circulators, and the greater cost of beryllium cans as compared to Zircaloy cans. Assuming for instance that beryllium canned fuel elements cost 30 % more than Zircaloy canned elements having the same rod diameter (13.5 mm), the overall economy of the CO_2 cooled reactor appears to be intermediate between that of pressurized D_2O and boiling D_2O designs - the limits of error of the estimates however exceeding the calculated differences in costs.

In Canada somewhat similar economic comparisons have been carried out between reactors cooled by pressurized D_2O on the one hand and H_2O steam[9] on the other hand leading to the conclusion that pressurized D_2O appeared to offer the better economy, even if zirconium alloy cans suitable for 480°C steam were developed. The advantage of steam from the aspect of coolant inventory costs and thermal efficiency appear to be more than offset by the reduction in burn-up caused by the large area ratio, A_c/A_f, resulting from the relatively small coolant temperature range which can be used with high pressure steam, whilst the advantages of organic liquids from the aspect of pressure containment and fuel inventory are counteracted by the parasitic neutron absorption and inferior heat transfer.

Whilst future developments might change the picture, the preliminary comparisons made suggest that there has not appeared as yet a design for other coolants than D_2O which would give pressure tube reactors a significant advantage over pressure vessel designs.

13. Concluding remarks

The discussion in this paper appears to lead to the following main conclusions.

1. Pressure vessel designs offer somewhat higher burn-up values than pressure tube designs when both types are designed for the same volume of core plus reflector, the same fuel geometry and the same net electrical output and overall thermal efficiency, and accordingly, very roughly, the same capital cost. This may give the pressure vessel type a slight overall cost advantage, though the differences are not significant in relation to possible errors in burn-up and cost predictions. The above applies when both types of designs use similar design philosophies regarding margins against a variety of occurances.

2. Pressure vessel designs have better possibilities of utilizing the moderator to achieve some degree of self-regulation and of dimensioning the coolant channels to given negative void coefficients and good self-regulation. The assessment of the importance of this factor varies however between different design organisations.

3. With the equipment which manufacturers consider it reasonable to order within a few years in Sweden, it is visualized that pressure vessels can be made for the largest electrical outputs per reactor which are considered to be of interest, e.g. well over 400 MW electricity per reactor with natural uranium.

4. Slight enrichment appears to give somewhat lower costs than natural uranium for D_2O moderated reactors even for very large electrical outputs, particularly with pressure vessel designs which have greater freedom to use low moderator to fuel moderator ratios and higher coolant to fuel area ratios than pressure tube designs. Slight enrichment also increases the economic power output from a pressure vessel of given dimensions. With slight enrichment, the D_2O moderated and cooled designs for large electrical outputs appear also to be able to compete economically with the more heavily enriched H_2O moderated reactors.

5. Direct cycle boiling reactors for D_2O appear to offer certain possibilities of making important cost reductions compared with pressurized D_2O reactors. Pressure vessel designs are more suitable for boiling reactors than pressure tube designs because of the possibility of using larger coolant to fuel area ratios which is an advantage from the aspect of void coefficients and natural circulation.

6. Both types of design lend themselves to the use of internal superheating although this can be performed with less complication to the core in pressure tube designs.

7. Pressure tube designs ledn themselves better than pressure vessel designs to the use of other coolants than D_2O, but comparable design studies for large reactors have not as yet indicated significant potential gains due to the use of other coolants such as CO_2 gas, H_2O steam, or organic liquids.

No doubt many other points comparing pressure tube and pressure vessel designs can be found, but their evaluation is currently a matter of opinion. The points indicated here have at least served to strengthen the belief in Sweden in the future of the pressure vessel type of reactor using pressurized D_2O and ultimately boiling D_2O as coolant.

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REFERENCES

1. P.H. Margen, H. Carruthers, B. Hargö, G. Lindberg and B. Pershagen: R3. - A Natural Uranium Fuel Heavy Water Moderated Reactor for Combined Electricity Production and District Heating. Proc. Geneva Conf. P/135 (1958)

2. E.G. Malmlöw, O. Mileikowsky, S. Ryman and I. Wivstad: Nuclear Heat and Power for the City of Stockholm. The Joint Swedish Project. World Power Conference, Madrid. (1960)

3. E.G. Malmlöw: Heavy Water Cooled Reactors With Pressure Vessels. EAES Mallorca Conference (1960)

4. W.B. Lewis: Pressure-Tube Heavy-Water-Cooled Reactors. EAES Mallorca Conference (1960)

5. B. Hargö, A. Dahlgren, G. Andersson: Utilizing the Larger Moderator Volume over Heavy Water Power Reactor. EAES Mallorca Conference (1960)

6. H. Lange, O. Steinman: Calculation of Two Phase Pressure Drop in Tubes and Fuel Element Bundles (in Swedish) AB Atomenergi IPV-19 19. 5. 60

7. M. Treshow, D. Shaftman, L. Templin, M. Petrick, B. Hoglund and L. Link: A Study of Heavy Water Central Station Boiling Reactors (CSBR) ANL-5881, Sept. 1958

8. P.H. Margen: Preliminary Cost Comparison Between Four D_2O Moderated Reactors for 400 MW Electricity. AB Atomenergi IP-1 7. 6. 1960.

9. Partial Economic Study of Steam Cooled Heavy Water Moderated Reactors. AECL-1018 (1960)

10. B. Pershagen, G. Andersson and I. Carlvik: Calculation of Lattice Parameters for Uranium Rod Clusters in Heavy Water and Correlation with Experiments. Proc. Geneva Conf. P/151 (1958)

11. G. Andersson: Formulae for Calculation of Physical Properties of Heavy Water Moderated Reactors with Cylindrical Lattice Cells and Uniform Burn-up. AB Atomenergi RFR-47. 20. 11. 1959.

12. Heavy Water Lattices. IAEA Symposium, Vienna (1960)

13. P.E. Ahlström: Burn-up Calculations for a Thermal Reactor. AB Atomenergi RFR-10 (In Swedish) 15. 9. 1958

14. P. Weissglas: Theoretical Calculation of the Effect on Lattice Parameters of Emptying the Coolant Channels in a D_2O Moderated and Cooled Natural Uranium Reactor. AB Atomenergi RFR-67 12. 5. 60

15. P.E. Ahlström: Reactivity Changes by Emptying the Coolant Channels for Different Coolant Areas and Lattice Pitches in the R4/Eve Reactor. AB Atomenergi RFR-66/R4-101 (In Swedish) 30. 5. 60

16. P.E. Ahlström, P. Weissglas: Spatial Variations of the Isotopic Concentrations in a Fuel Rod. AB Atomenergi RFR-79 (6. 9. 1960)

Burn-up and flux flattening in a two-zone reactor

1. Introduction

Consider a reactor of given dimensions, fuel elements and lattice spacing. Let the burn-up without flux flattening for a continuous charge and discharge system, where all points in the reactor contain fuel of all degrees of burn-up, be u (= "ideal burn-up"). If the radial flux is flattened in a central zone, the average burn-up will be reduced by a certain amout depending on the method of obtaining the flattening. Three different methods are considered, namely

a) All fuel elements are discharged when they have reached the same burn-up, u_a . Flux flattening is achieved by inserting absorbers in the central zone.

b) Fuel elements in the peripheral zone are discharged at a given burn-up, u_2, which is lower than the average, u_b, for the reactor, whilst the fuel elements in the central zone are discharged at a given burn-up, u_1, higher than the average. Both zones are fed with new fuel elements.

c) Fuel elements in the peripheral zone are removed at a given burn-up, u_2, and transferred to the central zone, where they are allowed to reach the burn-up u_c. The peripheral zone is fed with new fuel elements. The central zone is fed partly with fuel elements transferred from the peripheral zone and partly with new fuel elements which are also taken to the burn-up u_c.

2. One-group theory for a two-zone reactor

For the purpose of comparing the methods of obtaining flux flattening one-group theory is considered adequate.

Introduce the following notation:

R_1 = radius of central (flattened) zone
R_2 = radius of core
R_x = extrapolated radius
H_x = extrapolated height

B_1^2 = buckling of central zone

B_2^2 = buckling of peripheral zone

β^2 = radial buckling of peripheral zone
j = 2.4048 = first zero of $J_O(x)$
F_{ro} = radial form factor (= ratio of maximum to average radial flux) without flux flattening
F_r = radial form factor with flux flattening

The following relations are easily derived:

$$J_O(\beta R_x)\, Y_1\, (\beta R_1) - Y_O\, (\beta R_x)\, J_1\, (\beta R_1) = O$$

which is a criticality equation determining .

$$F_{ro} = \frac{j\, \frac{R_2}{R_x}}{2\, J_1 \left(j\, \frac{R_2}{R_x} \right)}$$

$$\frac{1}{F_r} = \left(\frac{R_1}{R_2}\right)^2 + \pi \frac{R_1}{R_2} \left[J_1(\beta R_1)\, Y_1(\beta R_2) - Y_1(\beta R_1)\, J_1(\beta R_2) \right]$$

We also have

$$B_1^2 = \left(\frac{\pi}{H_x}\right)^2 \quad ; \quad B_2^2 = \left(\frac{\pi}{H_x}\right)^2 + \beta^2$$

3. Comparison of different methods of obtaining flattening

We assume continuous charging, discharging and transposition of fuel elements and use the following definitions:

u_1 = ideal burn-up corresponding to B_1^2

u_2 = -"- B_2^2

u_a = burn-up for method a)

u_b = -"- b)

u_c = -"- c)

$$n = \left(\frac{R_1}{R_2}\right)^2 F_r = \frac{\text{Sum of (fuel element positions x flux) for the central zone}}{\text{Sum of (fuel element positions x flux) for the entire reactor}}$$

In case (a) the attainable burn-up is given by

$$u_a = u_1$$

In case (b) fuel elements are removed at different burn-up, u_1 and u_2 in the central and peripheral zones respectively. The average burn-up from the aspect of fuel cost is

$$u_b = \frac{1}{\frac{n}{u_1} + \frac{1-n}{u_2}}$$

In case (c) it can be shown that the attainable burn-up, u_c, corresponds to a buckling, B_c^2, given by

$$B_c^2 = n\, B_1^2 + (1 - n)\; B_2^2$$

The fraction, z, of fuel element positions in the central zone occupied by elements which have been transferred from the peripheral zone is

$$z = \frac{u_c - u_2}{u_2} \qquad \frac{1 - n}{n}$$

The condition that $0 \leq z \leq 1$ implies that

$$1 - n \leq \frac{u_2}{u_c} \leq 1$$

which is always fulfilled in practical cases.

4. Numerical results

Numerical calculations have been made for a 200 eMW pressure vessel reactor, case 2a (cf. Table 1). The results expressed as the reduction in burn-up as a function of the degree of flux flattening for the three methods of obtaining the flattening are shown in fig. 8. It is seen that method (c) gives a considerable advantage over other methods, method (a) being, of course, least advantageous.

The relative reduction in burn-up are found to be approximately independent of the degree of flattening. If u is the ideal burn-up without flattening the following ratios are obtained

$$(u - u_a) : (u - u_b) : (u - u_c) = 1 : 0,44 : 0,23$$

The reason why method (c) gives so much better results than method (b) is largely the very large difference between the burn-up values at which elements are removed from the reactor with method (b) in the flattened and unflattened zones, whereas all elements are removed at the same burn-up with method (c).

Reactor physics calculations

1. Lattice parameters

The lattice parameters were calculated by the methods described by Pershagen et al (ref. 10). The full set of formulae were given at the IAEA symposium on Heavy Water Lattices in Vienna (1959) (ref. 12).

The method used for calculation of reactivity changes with irradiation is described in ref. 13.

2. Void coefficients

The calculations of the void coefficients were made by the methods developed by Weissglas (ref. 14). The calculations were made for a fuel element with 27 rods and 13, 5 mm rod diameter in a hexagonal (open) lattice. The details are given in ref. 15 and a future RFR-report. The uncertainty in the absolute values of the void effect must be considered very large. However, the curves are supposed to give a correct picture of the changes in the effect, when changing the coolant area or the lattice pitch.

The largest changes in reactivity when emptying the coolant channels come from the changes in U 238 resonance absorption. At small lattice pitches and large coolant areas the influence of the changes in the non-leakage probability is dominant. This effect is elsewhere of the same magnitude as the change of the fast fission factor, but of opposite sign. The thermal utilization is not very sensitive to the void effect, at least not with fresh fuel in the reactor. The effect of changes in the neutron spectrum have not been taken into account. As was shown by Weissglas they are small with fresh fuel in the reactor. The influence of burn-up, absolute temperature etc. has not been investigated.

It should be observed that the calculations give the change in reactivity at total and momentaneous loss of coolant in all channels

of the reactor. The changes of the temperatures, the geometrical dimensions etc. are not included in the calculations.

3. Moderator temperature coefficient of reactivity

The moderator temperature coefficient of reactivity has been calculated by a new programme for the Ferranti Mercury computer. The programme determines the lattice parameters at different temperatures as a function of the average irradiation of the fuel. When calculating the changes of isotopic concentrations with burn-up the non-uniform distribution of flux over a fuel element is included (ref. 16). The lattice parameters are calculated by almost the same formulae as given in ref. 11. A complete description of the programme will by given in an RFR.-report, which is now being prepared.

The temperature coefficients are calculated as $\frac{\delta \rho}{\delta T_m}$ where

$\rho = \frac{k_{eff} - 1}{k_{eff}}$ is the reactivity and T_m is the average moderator temperature. The non-leakage probability has been determined by one-group-theory. The coefficients are "point-coefficients", which means that the macroscopic distribution of the fuel isotopes are not included. To find the effective coefficiert one has to calculate a weighted average value over the reactor considering that the irradiation is different in different parts of the core.

Cost Comparison between pressurized water reactors for D_2O and H_2O

Fuel Cycle Costs

Table 3 compares the fuel cycle costs between a 1% enriched D_2O reactor (Design 7a) and a 2% enriched H_2O reactor (Design 10), both of the pressurized water type and designed for 400 MW electricity. Both use Zircaloy canned fuel elements, and give a mean burn-up of 14 200 MWd/ton.

The D_2O moderated reactor is seen to make a fuel cycle cost saving of 1, 31 mills/kWh, which is due mainly to the following reasons:

a) Lower enrichment which reduces interest charges for nuclear material,
b) The 1% enrichment is obtained by mixing natural uranium and enriched uranium elements. The natural uranium elements avoid conversion costs, which reduces the mean cost of fabrication
c) The continuous fuel element shuffling and changing system possible for the D_2O moderated reactor makes it possible to reduce changes in the flux pattern caused by variations in irradiation. In addition the reflector saving with D_2O improves the flux form factors. The net results is that the product, (flux form factor) x (flux peaking factor) is lower for the D_2O moderated design, which makes it possible to use a larger mean fuel rod diameter. This also reduces the cost of fabrication per kg.

It should be mentioned that factor c) also results in a higher maximum burn-up for the H_2O reactor than the D_2O reactor, for given mean burn-up. Should irradiation damage limit fuel life, then the D_2O design would obtain a further advantage not allowed for in the table.

The calculations in table 3 are based on 6% interest rates for all materials, and the refore do not take account of the subsidy currently given by the A. E. C. to enriched designs in the form a low 4% interest rate for fuel but not D_2O. On the other hand, fabrication costs are based on fig. 10, curves 1 and 2, which favours enriched H_2O designs compared to current U. S. A. fabrication prices.

Capital charges maintenance and leakage costs

The H_2O reactor achieves cost savings regarding the following points:

a) The D_2O inventory and leakage costs are avoided,

b) the size of the pressure vessel is reduced, which also affects building dimensions,

c) the standard of design and construction from the aspect of leakage prevention can be relaxed slightly, which reduces costs.

The significance of points a) and b) can be reduced by increasing the reactor rating as this makes possible a low D_2O investment and low tank volume per MW electricity. For design 7 in table 2, for instance point a) amounts to 46 $/kW or 0, 76 mills/kWh and a charge of 10% per year for inventory and leakage. Point b) is estimated to amount roughly to 15 $/kW or 0, 25 mills/kWh, since the dimensions of the pressure vessel have a relatively small influence on total costs for large reactors with a high specific heat rating. Point c) is difficult to estimate, but has been assumed to be 7 $/kW or 0, 12 mills/kWh. The combined effect of the three points is thus 68 $/kW or 1, 13 mills/kWh. This is slightly less than the extra costs on the fuel cycle discussed earlier, (1, 31 mills/kWh). It appears from this discussion that large D_2O moderated reactors can compete economically with large H_2O moderated reactors for Swedish accountancy practice, assuming that both types reach the same state of technological development.

TABLE 1 Burn-up Comparisons between Pressure Vessel and Pressure tube Reactors for 200 MW electrical output

			1*	2a	2b	3
			Canadian	design	basis	Swedish design basis
			TUBE	VESSEL (same reflector thicknesses as CANDU)	VESSEL (uniform reflector thicknesses)	VESSEL
I.	Given data, common to tube & vessel reactors					
1.	Net electrical output	MW	~200			
2.	Total heat rate (fuel + moderator)	MW	720			
3.	Reflected core volume	m^3	141			
4.	Fuel element (number & dia. of rods)		19 x 14,5 mm			
5.	Maximum permissable heat loading of rods, allowing for axial, radial & internal form factors & rod end effect	W/cm	562			450
6.	Cross-sectional area ratio, coolant: fuel	-	0,6			2,0
7.	Ratio, design pressure/working pressure		0,9			0,85
8.	Mean temperature margin against bulk boiling at channel exit	°C	12			17
II.	Other data					
A.	Temperatures, pressures, efficiencies					
9.	Moderator temperature	°C	49	225	225	230
10.	Design pressure	bar	102	69	74	65
11.	Working pressure	"	92	62	66	55
12.	Corresponding saturation temperature	°C	305	278	282	270
13.	Coolant outlet temperature	"	293	266	270	253
14.	Coolant temperature range	"	44	44	48	25
15.	Coolant inlet temperature	"	249	222	222	228
16.	Temperature of saturated secondary steam	"	251**	221	221	222
17.	Steam cycle efficiency (practical)	%	32,8	30,2	30,2	30,3
18.	Losses due to auxiliaries	%	8	8	8	7
19.	Heat rejected by moderator circuit	%	8	-	-	-
20.	Overall thermal efficiency, based on items 16 to 19	%	27,8	27,8	27,8	28,2
21.	Net electrical output	MW	200	200	200	203
B.	Overall dimensions & heat loadings					
22.	Length or height of core***) (17%)	m	5,00		4,96	4,96
23.	Diameter of core	m	4,58		4,96	4,96
24.	Volume of core	m^3	82,5		96	96
25.	Volume ratio: core:fuel	-	17,1		17,1	16,1
26.	Volume of fuel	m^3	4,82		5,62	5,97
27.	Axial reflector thickness	m	0		0,34	0,34
28.	Radial reflector thickness	m	0,70		0,34	0,34
29.	Extrapolated core volume	m^3	149		145	145
30.	Form factors:					
	a) axial+		1,466		1,384	1,41
	b) radial (partly flattened)++		1,394		1,720	1,43
	c) internal (approx.)		1,10		1,10	1,10
	d) end peaking (approx.)		1,05		1,05	1,05
	e) overall		2,36		2,75	2,35
31.	Require radius of flattened zone/extrapolated radius, R_1/R_x (one group theory)		0,201		0,127	0,311
32.	Total fuel heat rate	MW	676		676	676
33.	Mean heat loading per cm	W/cm	238		204	191
34.	Max. heat loading per cm = 33 x 30	"	562		562	450
C.	Fuel channel & pressure vessel data					
35.	Maximum flow velocity (nominal)	m/sec	8,3	8,2	8,6	3,8
36.	Inside diameter of shroud or pressure tube	mm	82,6	82,6	82,6	111
37.	Design stress of pressure tubes	bars	10,4	-	-	-
38.	Mean cross sectional area ratios:					
	a) cans:fuel		0,13	0,13	0,13	0,13
	b) pressure tube or shroud tube:fuel		0,39	0,10	0,10	0,13
	c) calandria tube: fuel		0,13	-	-	-
	d) total Zr:fuel +++		0,65	0,23	0,23	0,26
	e) gas gap: fuel		0,63	-	-	-
39.	design stress of pressure vessel assuming low alloy steel	bar	-	2690	2690	2690
40.	Inside diameter of pressure vessel	m	-	6,28	5,94	5,94
41.	Wall thickness of pressure vessel	mm		91	92	80
D.	Results of physics calculations					
42.	Burn-up: for idealized, continuous fuel loading system, neglecting flux flattening	kWd/kg	9950	11110	11100	8800
43.	Adjusted burn-up	kWd/kg	9780	10950	11025	8350

* The latest data for CANDU are given in ref. 4. The data shown in column 1 were obtained in 1959 and differ in a few minor respects from the latest data - e.g. moderator temperature was then 49° C is now 43° C. A higher internal turbine efficiency is used in ref. 4, but this change effects pressure tube/pressure vessel designs alike.

** Item 16 is two degrees lower than item 15 for CANDU, whilst it is taken to be 1°C higher for the corresponding pressure vessel designs. This advantage credited to the pressure tube designs is due to the assumption that a higher coolant inlet temperature can be used in the peripheral coolant channels for a pressure tube reactor since a separate inlet header can be used for these channels.

*** Effective length of UO_2 column in core is taken to be 97 % of core length

\+ Neglecting the slight axial flux peaking resulting from the bi-axial or axial inversion fuelling systems compared to the ideal distribution

++ One group theory. Somewhat lower values obtained by two group theory for the burn-up values cited in item 43.

TABLE 2 DATA FOR FOUR D_2O MODERATED REACTORS WITH 400 MW NET ELECTRICAL OUTPUT

			6	7	8	9
	TYPE		Pressure vessel			Pressure tube
	COOLANT		pressurized D_2O		boiling D_2O	CO_2
	ENRICHMENT (mean)		natural	1 %	natural	natural
1.	LEADING DIMENSIONS					
1.1	Core height & diameter	m	6,29 x 5,39	5,38 x [illegible],61	5,39 x 5,72	4,64 x 6,60
1.2	Axial and radial reflector thickness	m	0,40 x 0,36	0,35 x 0,32	0,36 x 0,38	0,45 x 0,32
1.3	Reflected and unreflected core volumes	m^3	210/144	[illegible]/90	201/138	232/158
1.4	Fuel in core	tons U	75	61,5	69,6	76,5
1.5	Inside diameter of pressure vessel	m	6,41	5,55	6,78	
1.6	Wall thickness of vessel	mm	93,0	81,0	100,0	
2.	LATTICE DIMENSIONS & BURN-UP					
2.1	Number & diameter of UO_2 fuel rods	mm	27 x 12,7	27 x 11,8	19 x 13,5	19 x 13,5
2.2	Canning thickness & material	mm	0,5 Zr	0,5 Zr	0,5 Zr	0,8 Be
2.3	Coolant to fuel area ratio		2,5	2,5	2,7	1,11
2.4	Inside diameter of coolant channels	mm	126,2	117,6	115,5	90,3
2.5	Moderator (excluding coolant) to fuel volume ratio		13,6	9,15	13,0	15,2
2.6	Design stress of pressure tubes	bars	-	-	-	923
2.7	Inside diameter of pressure tubes	mm	-	-	-	96,2
2.8	Sectional area of UO_2 per channel	cm^2	34,2	29,5	27,2	27,2
2.9	Area ratios between structural material and UO_2					
	a) fuel cans		0,176	0,190	0,166	0,265
	b) shroud tubes or pressure tubes		0,129	0,139	0,148	0,578
	c) internal insulation (equivalent Zircaloy)		-	-	-	0,105
2.10	Initial conversion ratio		0,901	0,829	0,905	0,840
2.11	Burn-up:					
	a) ideal, without flux flattening	MWd/ton	9220	15200	8500	9750
	b) practical, with flux flattening	"	8550	14200	8270	9200
3.	FLUX & HEAT RATING					
3.1	Reactor heat rate	MW	1420	[illegible]	1230	1211
3.2	Fuel heat rate	MW	1333	1333	1155	1162
3.3	Form factors					
	a) axial		1,414	1,405	1,382	1,386
	b) radial (flattened)		1,329	1,270	1,357	1,279
	c) internal (approx.)		1,12	1,11	1,09	1,09
	d) ends "		1,05	1,05	1,05	1,04
	e) overall = a) x b) x c) x d)		2,21	2,08	2,15	2,01
3.4	Max. heat rating per cm of rod	W/cm	450	450	450	378
3.5	Max. can surface heat flux	W/cm^2	105	112	99	83
4.	TEMPERATURES, PRESSURES, VELOCITIES, EFFICIENCIES					
4.1	Mean temperature of moderator	°C	230	230	220	100
4.2	Coolant inlet temp. (reactor)	°C	228	228		243
4.3	Coolant outlet temp.	°C	258	258	276	483
4.4	Saturation temperature of coolant	°C	275	275	276	-
4.5	Coolant working pressure	bar	59,5	59,5	60,4	75
4.6	Design pressure	"	70	70	71	85,5
4.7	Max. allowable can temperature	°C	-	-	-	600
4.8	Coolant velocity, central channel	m/sec				41
4.9	Heat transfer coefficient	$W/cm^2 °C$				0,565*
4.10	Hot spot factor (can/gas)		heat transfer not critical			
4.11	Hot channel factor					1,25
4.12	Steam content at outlet of central channel	vol %	-	-	81	-
4.13	Steam pressure	bar				dual pressure cycle**
4.14	Steam temperature	°C	220	220	276	***
4.15	Steam cycle efficiency	%	30,3	30,3	34,4	39,6
4.16	Circulator power consumption	%	3	3	-	10

			6	7	8	9
	TYPE		Pressure vessel			Pressure tube
	COOLANT		pressurized D_2O		boiling D_2O	CO_2
	ENRICHMENT (mean)		natural	1 %	natural	natural
4.17	Other auxiliaries	%	5	5	5	5
4.18	Heat loss to moderator	%	-	-	-	8
4.19	Overall thermal efficiency	%	28,2	28,2	32,5	33,0+
4.20	Net electrical output	MW	400	400	400	400
4.21	Generator rating	MW	428	428	420	470
4.22	External heat transfer surface of heat exchangers	m^2	20000	20000	-	36000
5.	CAPITAL COSTS					
	A. NUCLEAR PART					
5.1	D_2O at 67,5 $/kg:					
	a) within tank	$/kW	37	23	35	40
	b) *external system*	"	19	19	11	4
			56	42	46	44
5.2	Engineering plant					
	a) reactor vessel equipment inside vessel, fuel handling	"	18	13	18	19
	b) main heat exchangers circulators and main piping	"	35	35	8	38
	c) auxiliary circuits	"	13	12	15	15
	d) instrumentation, control equipment, electrical station supplies	"	9	9	8	13
	e) miscellaneous	"	15	14	15	15
5.3	Building, shielding, ventilation etc.	"	33	31	32	34
5.4	Design & development ++	"	17	17	17	17
5.5	Total, nuclear part	"	196	173	159	195
	B. CONVENTIONAL PART					
5.6	Turbo-generator, turbine house, generator switchgear etc.	"	65	65	74	62
5.7	Total for station (5.5 + 5.6)	"	261	38	233	257
5.8	Interest during construction	"	31	29	28	31
5.9	Commissioning cost +++	"	8	8	8	8
5.10	Grand Total	"	300	275	269	296
6.	FUEL CONSUMPTION COST (see table 3)					
6.1	Cost of fuel	$/kg U	88.9	159.2	84.8	105.5
6.2	Net credit for spent fuel	"	16.8	36.0	15.2	17.2
6.3	Net cost of fuel	"	72.1	123.2	69.6	88.3
6.4	Fuel consumption cost ϕ	mills/kWh	1.20 (1.49)	1.28 [1.19] (1.66) [(1.56)]	1.08 (1.32)	1.21 (1.45)
7.	*SUMMARY OF COSTS*					
	A. FIXED COSTS					
7.1	D_2O leakage costs:					
	a) percent per year of D_2O investment	%	3	3	5	1
	b) $/kW & year		1.68	1.26	2.30	0.44
7.2	Capital charges and maintenance					
	a) plant buildings and overheads (10%/year)	$/kW & year	24.40	23.30	22.30	25.20
	b) D_2O (7 %/year)	"	3.92	2.94	3.22	3.08
7.3	Fuel inventory costs (table 2)	"	0.73	1.10	0.65	0.88
7.4	Total fixed costs = 7.1b + 7.2 + 7.3	"	30.73	28.60	28.37	29.60
7.5	Fixed costs per kWh for 6000 full load hours/year	mills/kWh	5.12	4.76	4.73	4.94
	B. RUNNING COSTS					
7.6	Fuel consumption cost (item 6.4)	"	1.20	1.28 [1.19]	1.08	1.21
7.7	Operation (except maintenance)	"	0.35	0.35	0.35	0.35
7.8	Total running cost	"	1.55	1.63 [1.54]	1.43	1.56
7.9	Total cost for 6000 h/year = (7.5 + (7.8)	"	6.67	6.39 [6.30]	6.16	6.50

*) Allows for 10 % increase in heat transfer coefficient and 20 % increase in friction factor due to surface roughening of cans

**) Logarithmic mean temperature difference in heat exchangers = 46° C; effective mean temperature of heat intake by steam = 578° K

***) Without recovery of heat from moderator for feed heating

+) With recovery of heat from moderator for feed heating

++) Assuming that this is second generation station

+++) Not included in the costs cited in ref. 8

ϕ Costs in square brackets refer to design with graded enrichment (see table 3). Costs in round brackets do not take account of credit for spent fuel.

TABLE 3 FUEL CYCLE COSTS (6 % interest rate throughout)

			6	8	9	7	7a		
			D_2O moderated reactors						Pressurized H_2O reactor
			Natural uranium designs			Slightly enriched design (pressurized D_2O)			
			pressurized D_2O	boiling D_2O	CO_2 cooled	uniformly enriched	2 grades of enrichment: N.U. elements (45% by weight)	enriched element (55% by weight)	
A.	General data								
1.	Degree of enrichment	%	0,7	0,7	0,7	1,0	0,7	1,25	2,0
2.	Fuel rod diameter (UO_2)	mm	12,7	13,5	13,5	11,8	14,1	10,6	10,6
3.	Burn-up (mean)	MWd/t	8850	8270	9200	14200	9900	17700	14200
4.	Pu in depleted fuel	g/kg U	4,17	4,03	4,20	5,86	5,86**	5,86**	6,0
5.	U 235 in depleted fuel	"	not recovered			not recovered			1,03
6.	Thermal efficiency	%	28,2	32,5	33,0	28,2	28,2	28,2	28,2
7.	Mean fuel heat rating	MW/t	17,8	16,6	14,9	21,7	14,4	26,4	20,0
B.	Fuel consumption cost								
8.	Cost of nuclear material	$/kg U	40,5	40,5	40,5	75,8	40,5	110,5	220,0
9.	Conversion UF_6/UO_2 inc. losses	"	-	-	-	20,6	-	21,0	22,9
10.	Fabrication (function of diam.)***	"	44,8	40,8	61,2	54,7	38,7	63,3	63,3
11.	Transport & interest costs up to time of loading into reactor	"	3,6	3,5	3,8	8,1	3,5	10,5	18,2
12.	Total cost new fuel (8 to 11)	"	88,9	84,8	105,5	159,2	82,7	205,3	324,4
13.	Financial value after irradiation:								
	a) Pu	"	50,0	48,4	50,4	70,3	70,3	70,3	72,0
	b) U 235	"	-	-	-	-	-	-	79,9
	c) total	"	50,0	48,4	50,5	70,3	70,3	70,3	151,9
14.	Cost of separation & clean-up including interest, and transport, 1 % Pu loss & 1 % U 235 loss	"	33,2	33,2	33,2	34,3	34,3	34,3	41,4
15.	Net fuel credit = 13 - 14	"	16,8	15,2	17,2	36,0	36,0	36,0	110,5
16.	Net fuel cost = 12 - 15	"	72,1	69,6	88,3	123,2	46,7	169,3	213,9
							1,19		
17.	Fuel consumption cost per kWh+	mills/kWh	1,20	1,08	1,21	1,28	0,70	1,42	2,23
			(1,49)	(1,32)	(1,45)	(1,66)	(1,24)	(1,72)	(3,38)
							(1,56)		
C.	Fuel inventory costs								
18.	Interest on fuel in reactor*	$/kW year	0,63	0,56	0,76	0,96	0,89	0,96	2,30
							0,93		
19.	Interest on 10 % standby fuel	"	0,10	0,09	0,12	0,14	0,13		0,35
20.	Total	"	0,73	0,65	0,88	1,10	1,06		2,65
21.	Total per kWh for 6000 full load hours/year	mills/kWh	0,12	0,11	0,15	0,18	0,18		0,44
22.	Total fuel cost (6000 h/year) = = (17) +(21) +	"	1,32	1,18	1,36	1,46	1,36		2,67
			(1,61)	(1,42)	(1,59)	(1,84)	(1,74)		(3,82)

* based on linear depreciation in fuel value from value for new fuel, item 12 to value after irradiation, item 15 (neglecting reduced burn-up or increased enrichment for first and last fuel charges)

** Neglecting variations in neutron flux between enriched and natural uranium elements (assuming uniform distribution of enriched and natural uranium elements)

*** Addition of $ 6/kg for enriched elements

\+ Figures in brackets, items 17 and 22, show costs without taking into account fuel credit.

Schematic presentation of pressure tube and pressure vessel designs

Figs 1 and 2

Fig. 1

Assumed pressure tube design

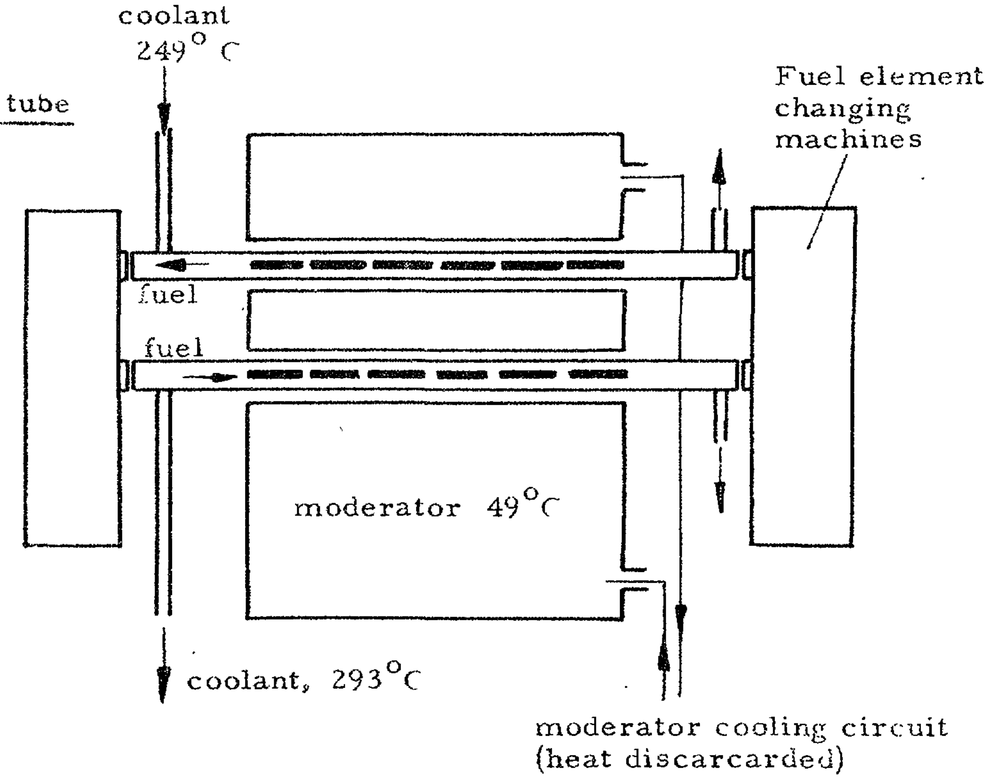
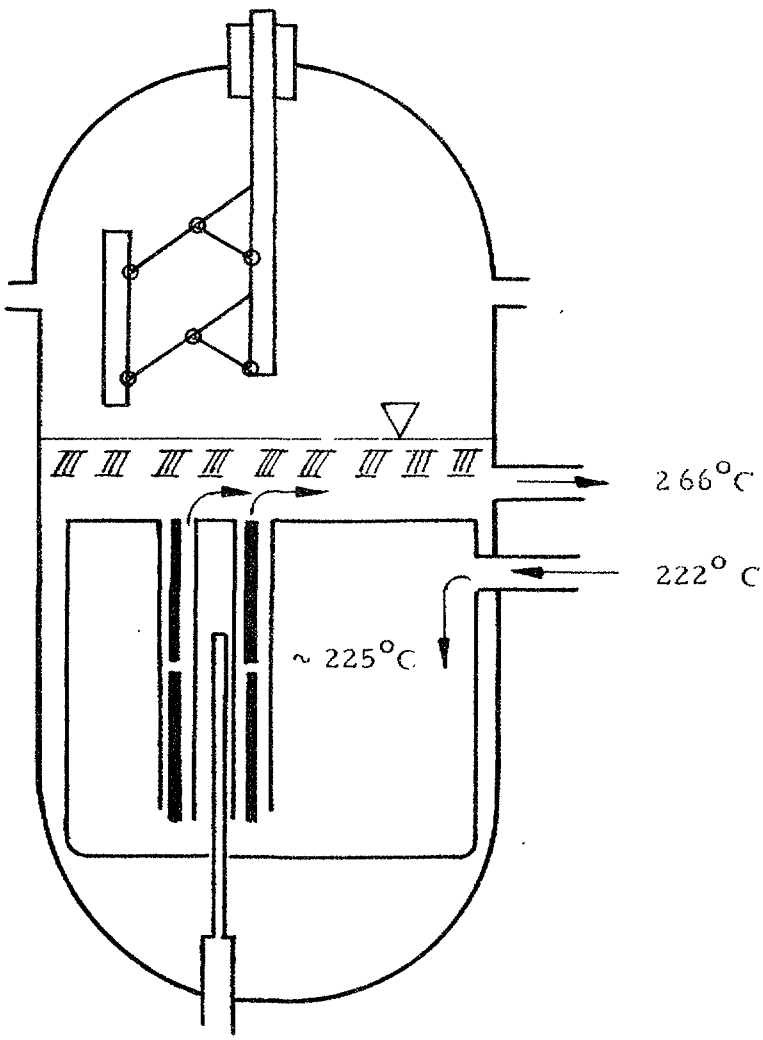


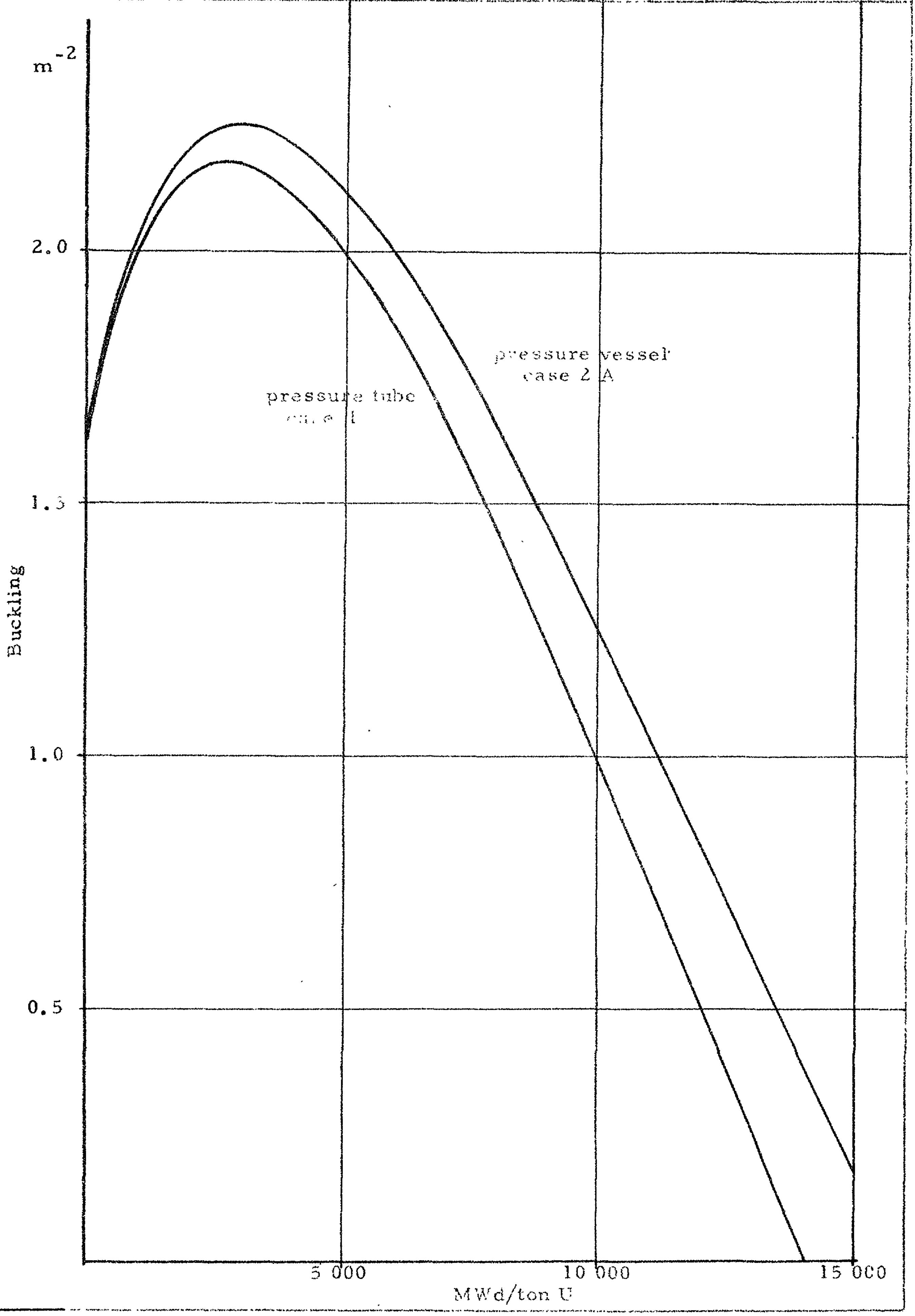
Fig. 2

Assumed pressure vessel design



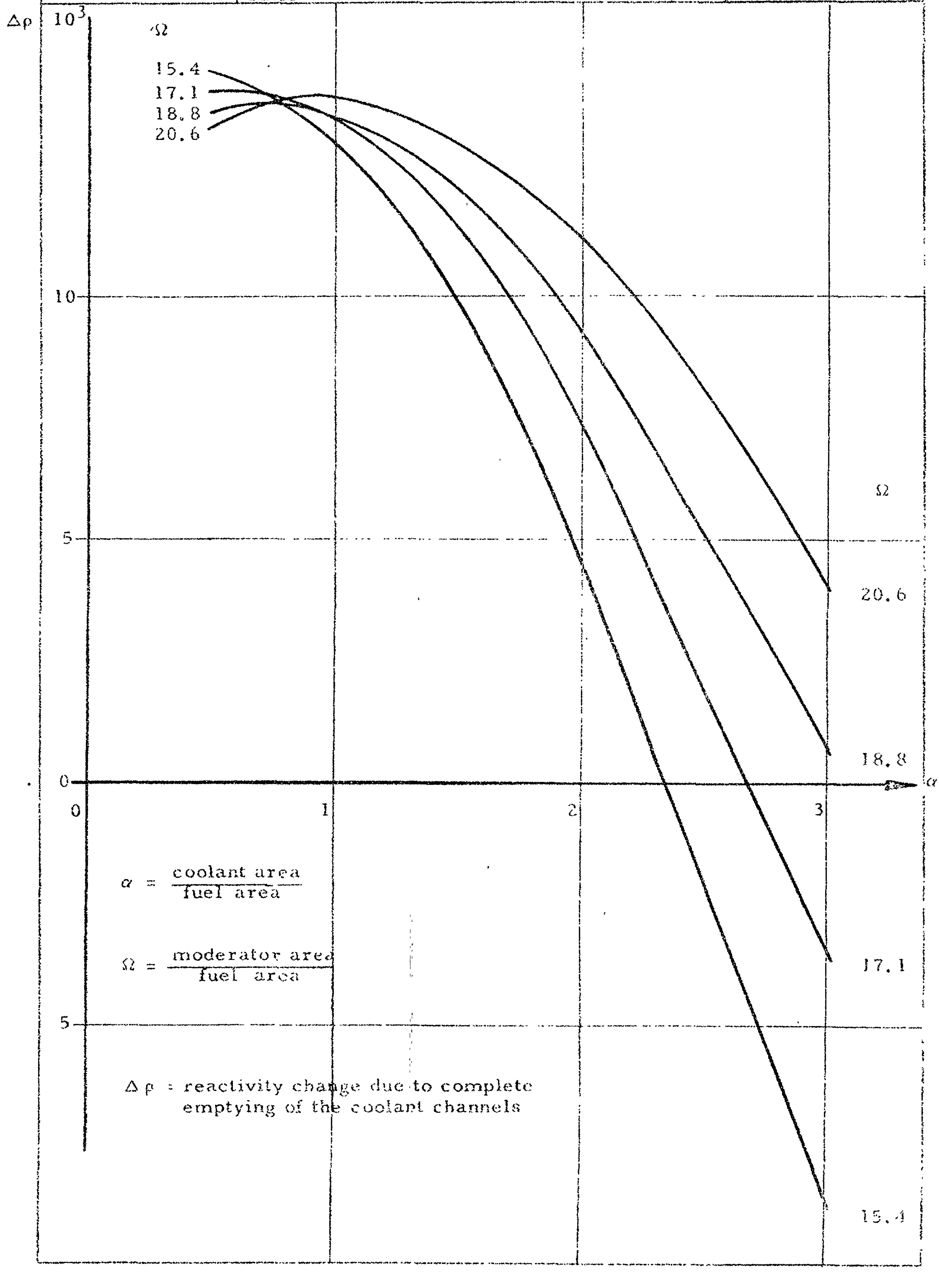
Material buckling versus ideal burn-up for pressure tube and pressure vessel reactors with the same fuel element and lattice spacing

Fig. 3



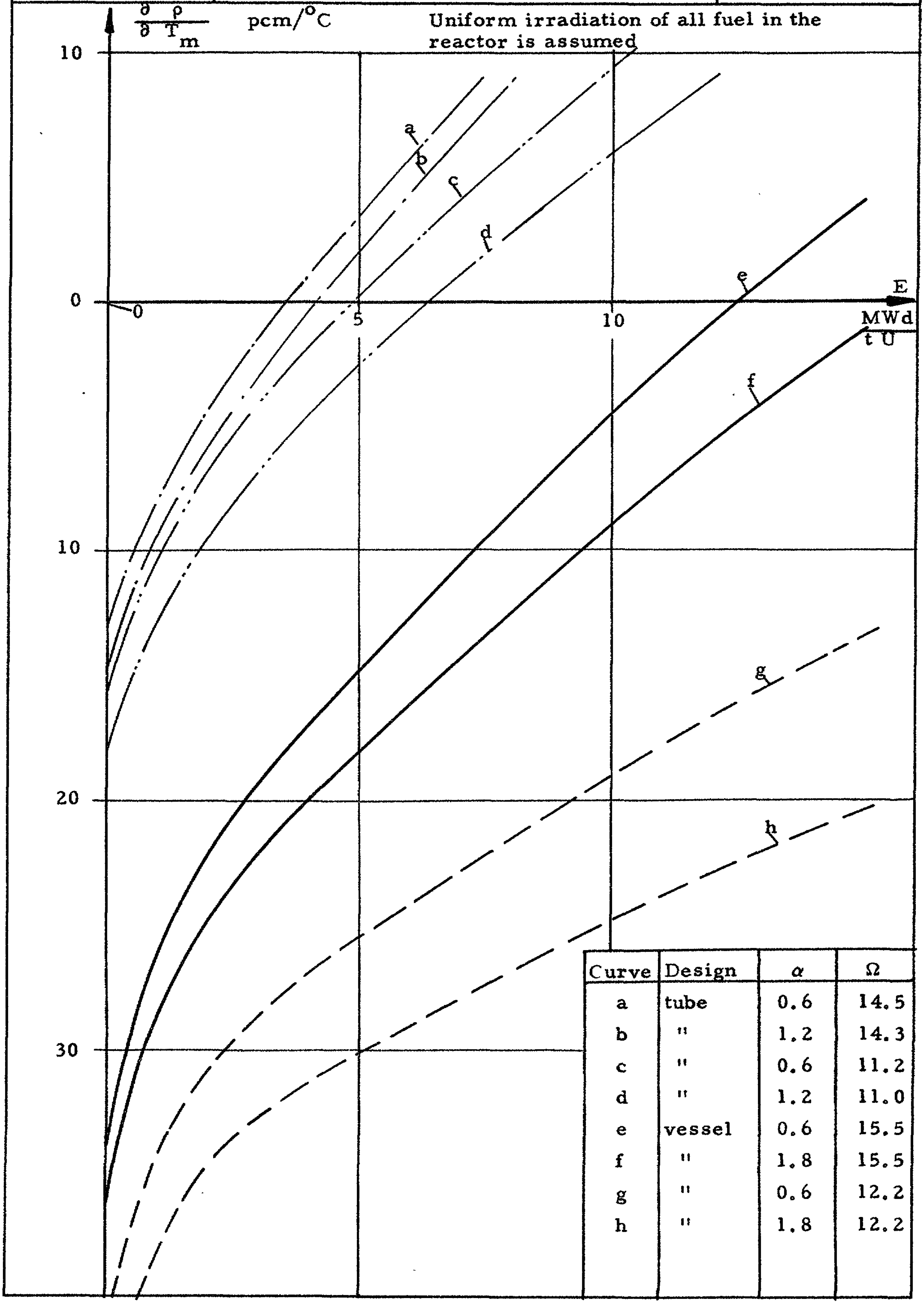
Void coefficients as a function of coolant to fuel area ratio and moderator to fuel area ratio

Fig. 4



Moderator temperature coefficient $\left(\frac{\partial \rho}{\partial T_m}\right)$ as a function of burn-up (E) for different coolant to fuel (α) and moderator to fuel (Ω) area ratios and different designs.

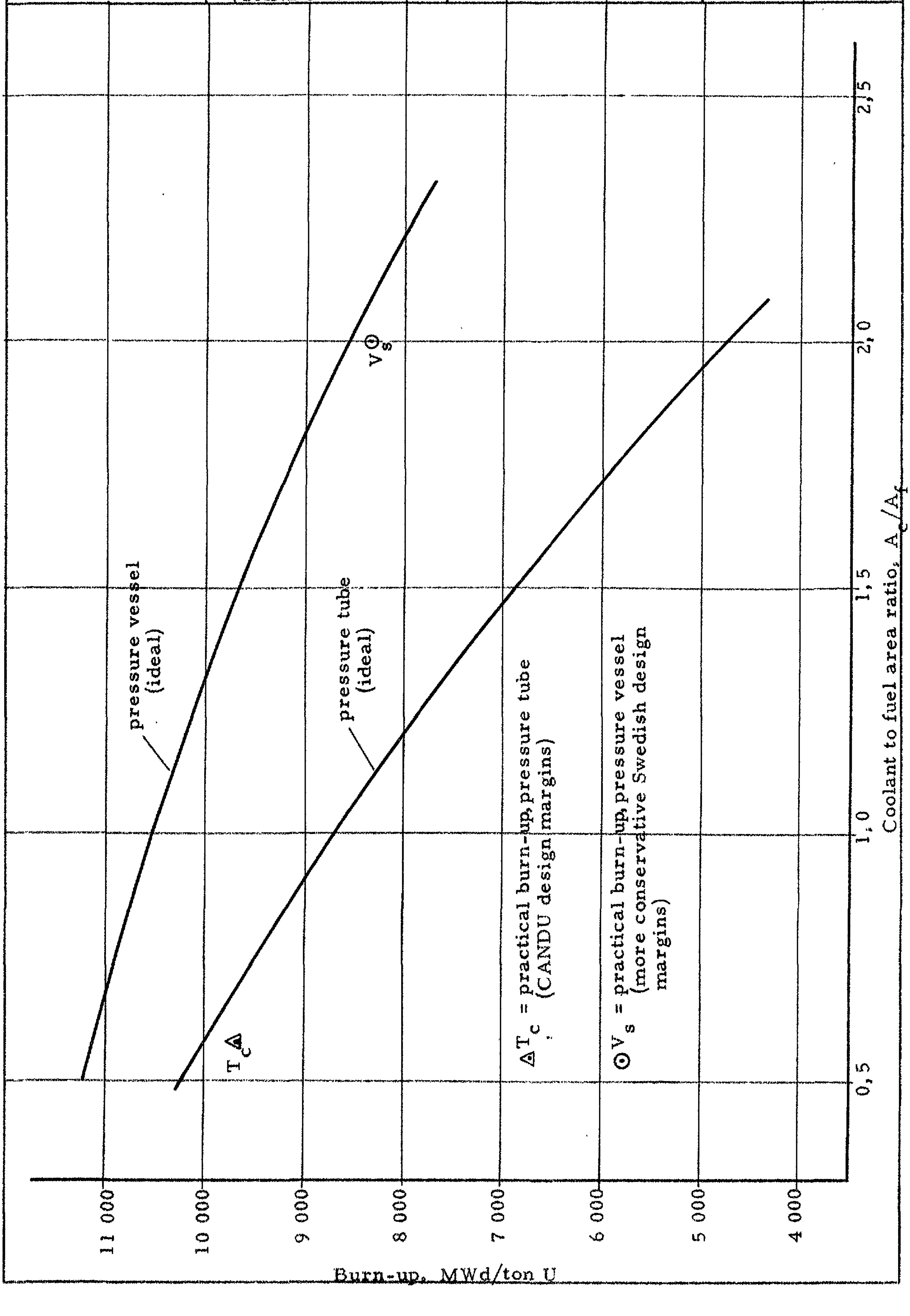
Fig. 5



Curve	Design	α	Ω
a	tube	0.6	14.5
b	"	1.2	14.3
c	"	0.6	11.2
d	"	1.2	11.0
e	vessel	0.6	15.5
f	"	1.8	15.5
g	"	0.6	12.2
h	"	1.8	12.2

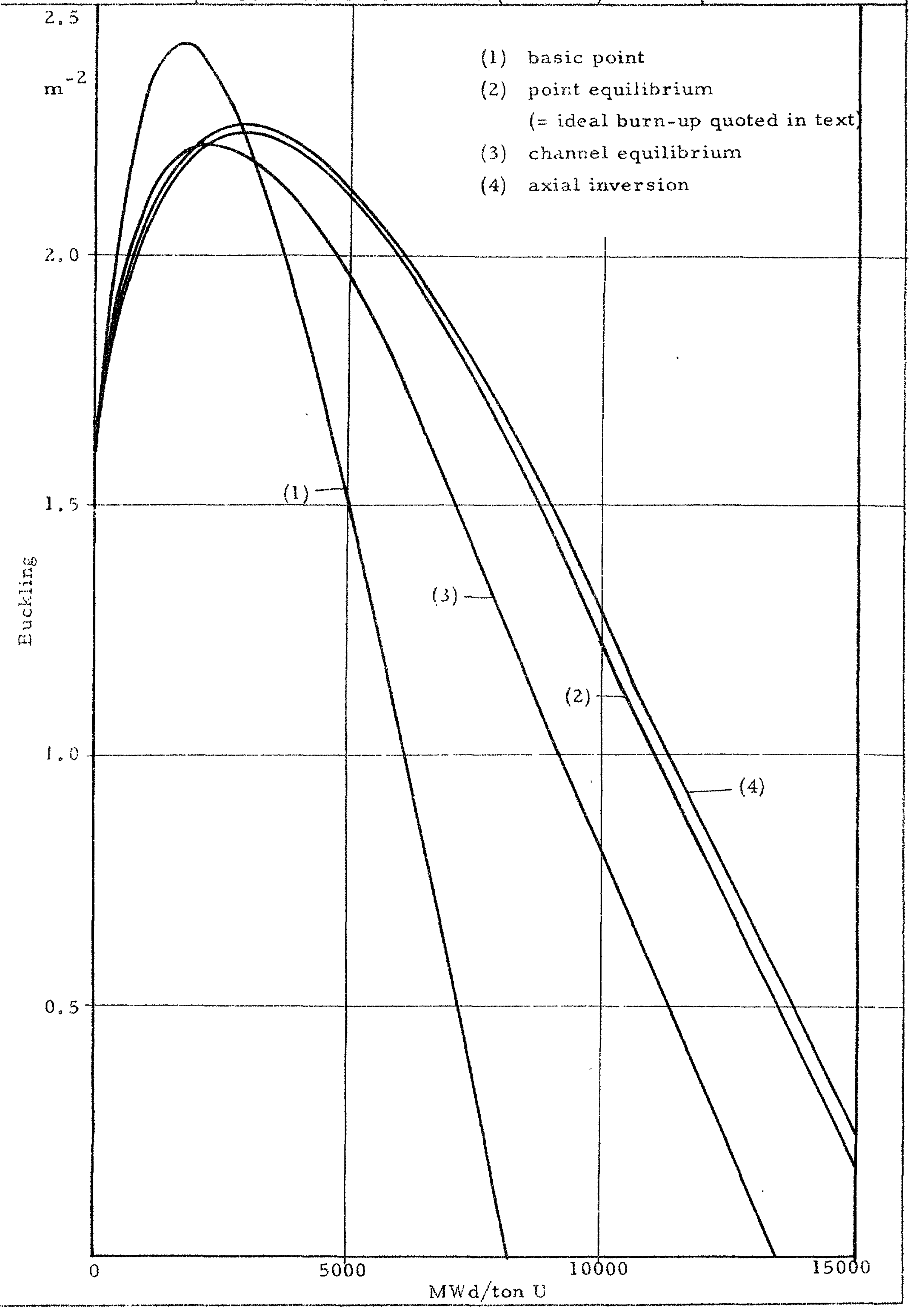
Comparison of burn-up values for pressure tube and pressure vessel reactors (CANDU dimensions)

Fig. 6



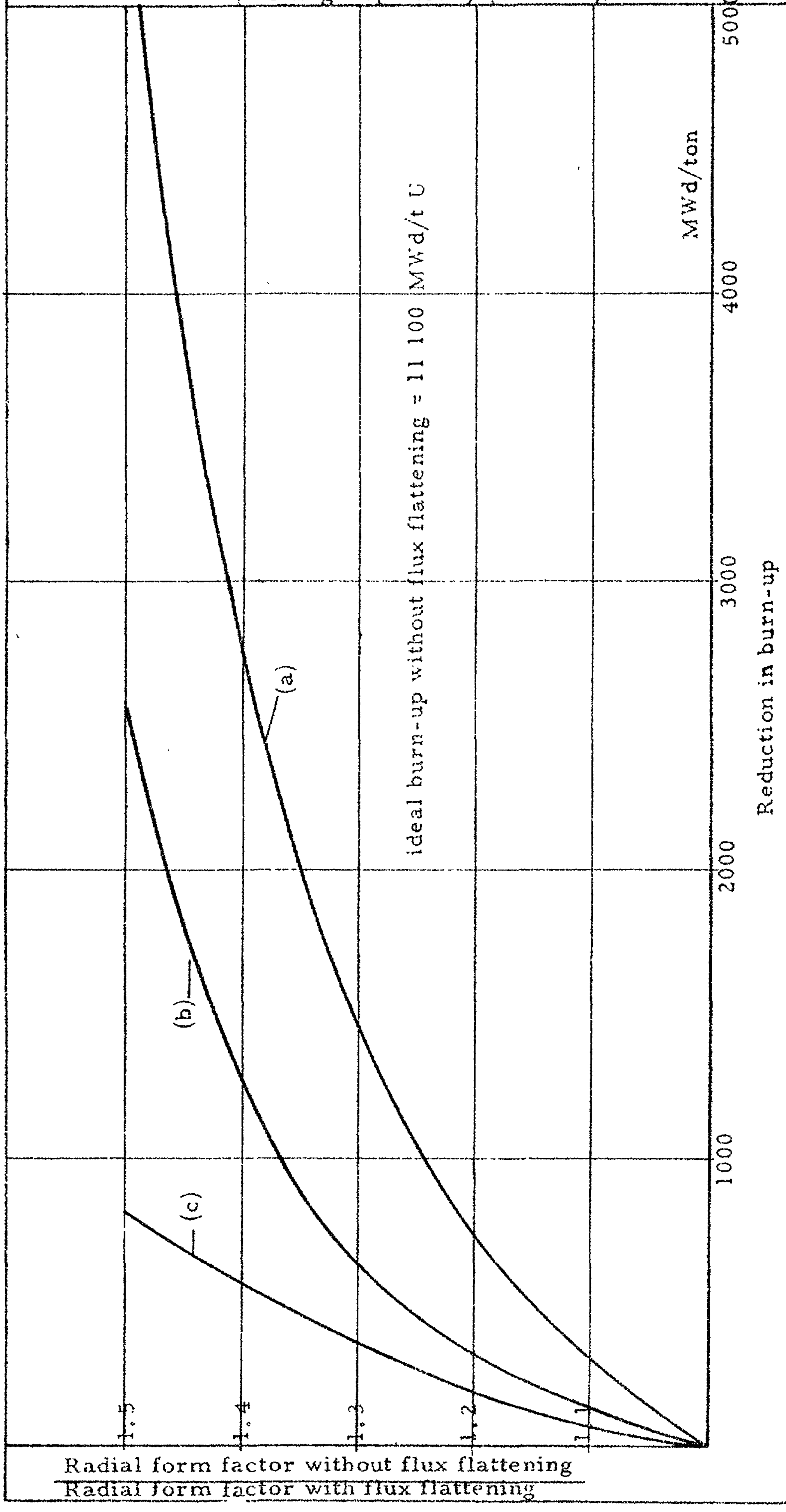
Material buckling versus burn-up for various charge and discharge systems. Pressure ve...el reactor (case 2 A)

Fig. 7



Influence of radial flux flattening on burn-up in a two-zone reactor
One group theory (case 2A)

Fig. 8

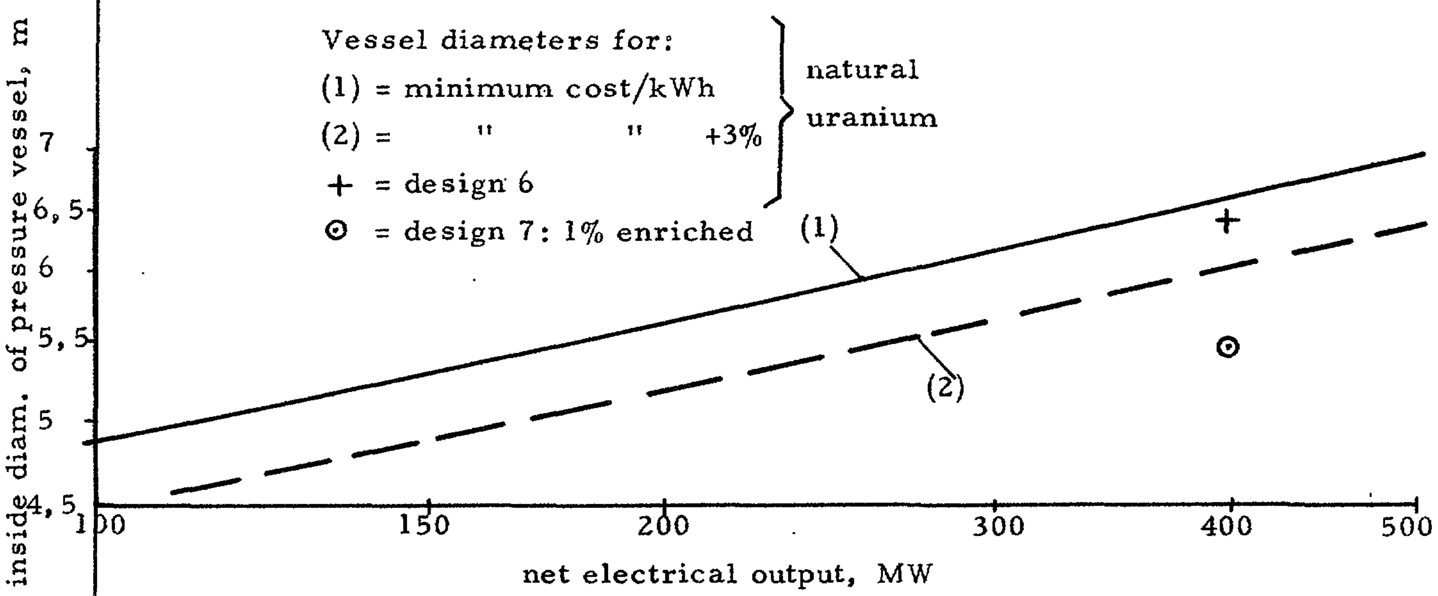


(a) = flux flattening with absorbers in central zone
(b) = " " discharge at different burn-up in central and peripheral zone
(c) = " " fuel element transposition from peripheral to central zone and all fuel elements discharged at the same burn-up

Diameter/output relations

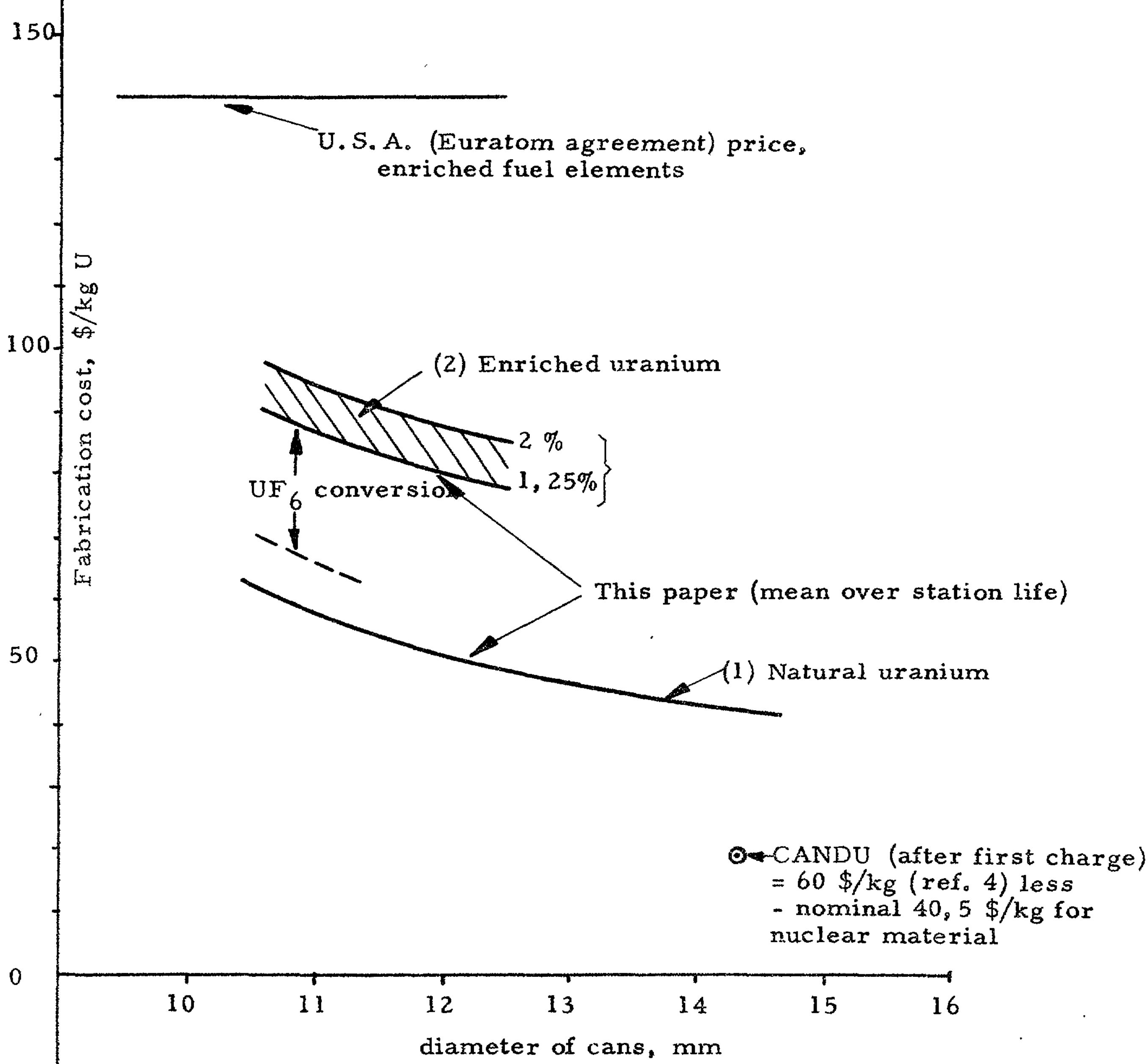
(Assumptions: H/D = 1,17,
Credit for spelt fuel = 0)

Fig. 9



(logarithmic scales)

	Cost of fuel fabrication, including UF_6 conversion (for enriched elements), interest during fabrication and losses (Zr-2 cans)	Fig. 10



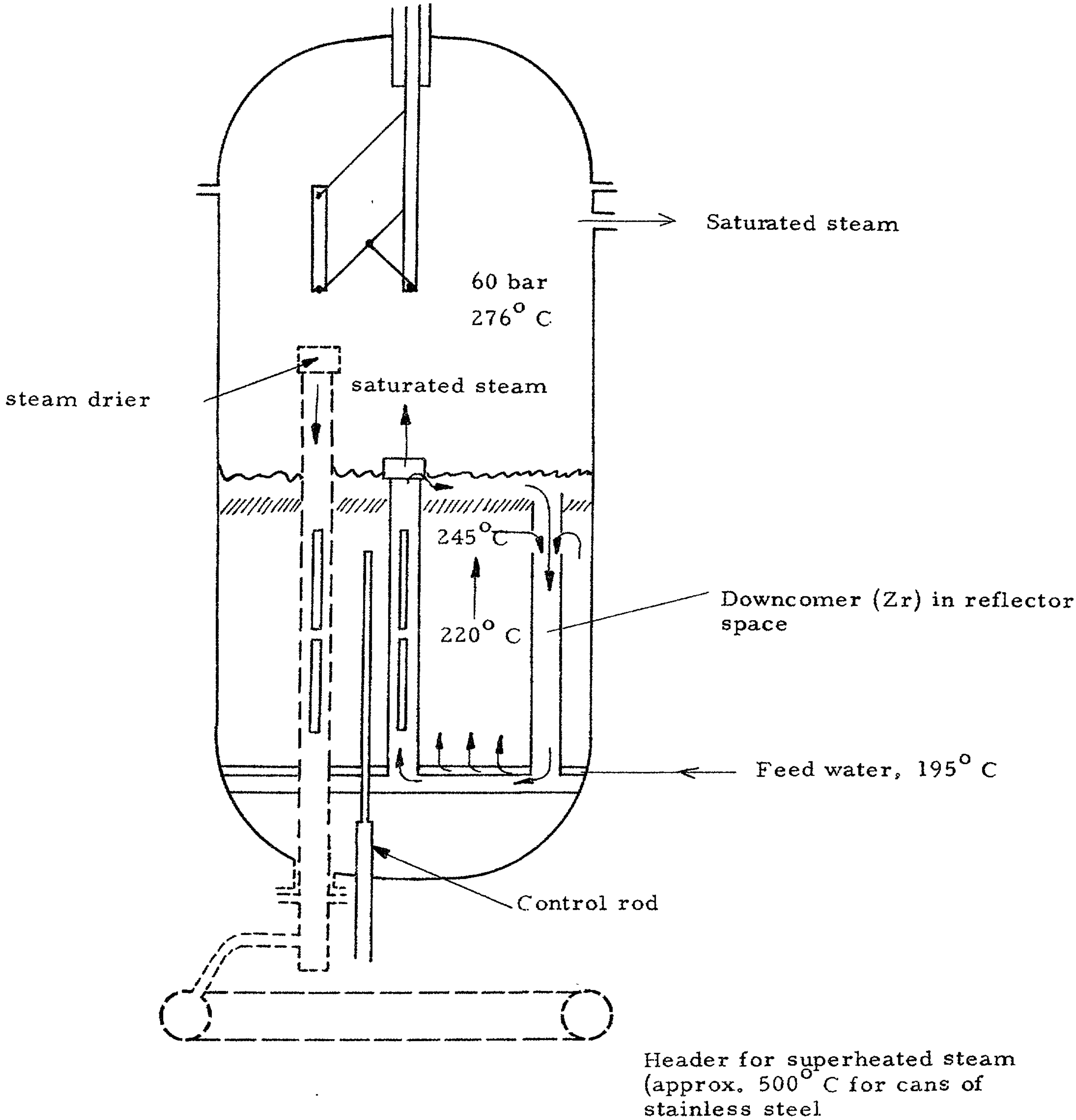
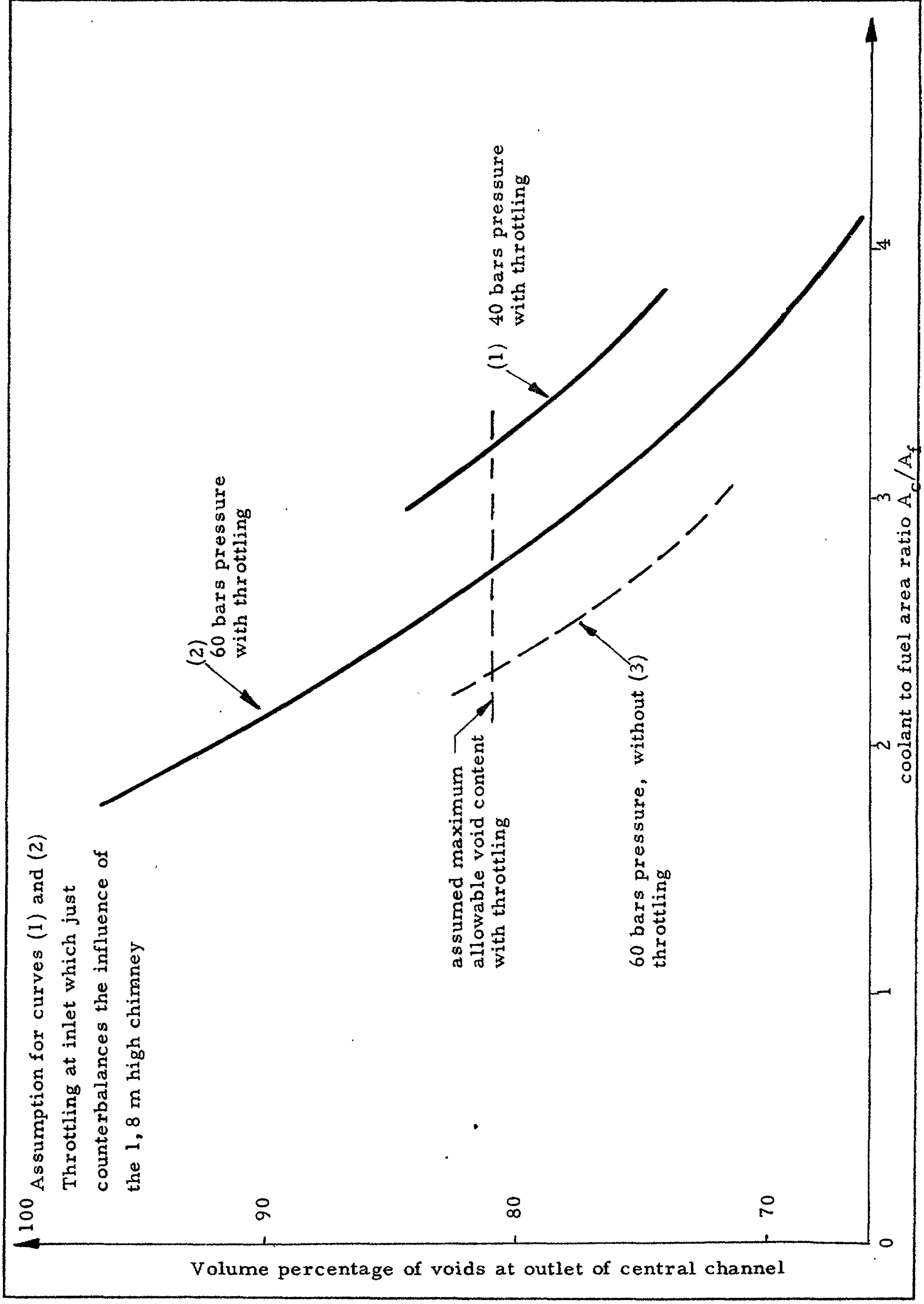


FIG. 11 Schematic diagram for D_2O boiling reactor

(The broken lines show a possible way of adding superheat channels uniformly spaced over the major part of the core to this type of reactor)

Natural circulation for D_2O boiler as function of coolant to fuel area ratio

Fig. 12



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