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ADVANCED PRESSURIZED WATER REACTOR FOR IMPROVED RESOURCE UTILIZATION

PART I - SURVEY OF POTENTIAL IMPROVEMENTS

Prepared For

U.S. ARMS CONTROL AND DISARMAMENT AGENCY

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TABLE OF CONTENTS

LIST OF ILLUSTRATIONS

1.0 INTRODUCTION

This document is an interim report under ACDA BOA AC9NX707, Task Order 80-03, which covers the evaluation of certain potential improvements in pressurized water reactor designs intended to enhance uranium fuel utilization. The objective of these evaluations is to seek advanced, non-retrofittable improvements that could possibly be commercialized by the end of the century, and, on the basis of a preliminary evaluation, to select compatible improvements for incorproation into a composite advanced pressurized water reactor concept.

The principal areas of investigation include reduced parasitic absorption of neutrons (Task 1), reduced neutron leakage (Task 2), and alternative fuel design concepts (Task 3). To the extent possible, the advanced concept developed in an earlier study (Retrofittable Modifications to Pressurized water Reactors for Improved Resource Utilization, SSA-128, October 1980) is used as a basis in developing the advanced composite concept. The reference design considered typical of present PWR commercial practice is the system described in RESAR-414, Reference Safety Analysis Report, Westinghouse Nuclear Energy Systems, October 1976.

2.0 SUMMARY AND CONCLUSIONS

The investigation of potential improvements in U_3O_8 resource utilization independently covered each of the task areas defined. Results of this investigation revealed that improvements might be achieved by (1) the use of water-displacer rods to adjust the water-to-fuel ratio by a "mechanical spectral shift" during reactor operation, (2) reduced neutron leakage by decreasing the core power density and (3) utilization of spent fuel as a reflector to effectively utilize leakage neutrons. No practical alternative fuel design was identified, although ceramal fuel (mixed-ceramic fuel such as UO_2-BeO) may offer improved reactor safety (higher thermal-conductivity fuel) but with little or no improvement in fuel utilization.

Mechanical spectral shift, with beryllium oxide (BeO) displacing ∿25% of the core water volume, appears to be capable of controlling ∿5.4% Δk (rods-in to rods-out), which is marginally adequate to control the reactivity change in a single fuel cycle in a 10-region core. Additional reactivity control can be achieved by using a displacer material with a higher neutron absorption cross-section. BeO is the preferred displacer material because of its low parasitic absorption cross-section, but the absorption may have to be increased in the composite reactor concept to provide the necessary reactivity control during the fuel burnup cycle.

For a given power level, reduced core power density increases the core volume and hence reduces the neutron losses. From the standpoint of fuel utilization, the maximum improvement (optimum) occurs at 60% of the reference power density, yielding a reduction of slightly over 3% in the 30-year U_3O_8 requirements. However, the economic penalty associated with higher interest charges on the fuel, and the additional capital cost for the larger pressure vessel, would likely make the 60% power density case unattractive. With a lower power density, a higher power peaking factor can be accommodated for the same maximum thermal power limit. Thus, by allowing the power distribution to deviate from the current, relatively-flat distribution, so that the power in the outer fuel assemblies is lowered, the neutron leakage can be reduced without the necessity of reducing the power density all the way to 60% of the present value. The more-peaked power distribution does complicate the fuel-control management program and increase the difficulty of analysis, but the advantage in fuel utilization is of a sufficient magnitude that the composite design will seek to utilize an average core power density of 70-75% of that in current PWR designs.

Another way of reducing radial neutron leakage is through the use of alternative reflector materials. In a comparison study of relative benefits of BeO and spent fuel "blankets," the materials appeared to have comparable effects. Because the cost of spent fuel (available after the first fuel cycle) is less than that of BeO, a spent fuel reflector will be used in the composite advanced PWR. It is recognized that spent fuel loaded into reflector positions could also be considered to represent a modified fuel management scheme with an additional, low-power-density region. Despite this dichotomy in terminology, spent fuel in an outer reflector region will, for convenience of reference, be termed a "blanket."

Advanced fuel element concepts considered included (1) cermet and ceramal fuel matrixes, (2) a duplex pellet concept, and (3) metallic fuel. Neither the cermet fuel or the duplex pellet concept offered any apparent improvement in fuel utilization. Metallic fuel, because of its high density, showed a potentially significant improvement in fuel utilization if the metallic fuel could attain fuel burnups comparable to these of oxide fuel. However, the potential for metal-water reaction under postulated accident contitions, and other safety considerations, render the use of metallic fuel in water reactors highly doubtful. Furthermore, the development of high-burnup metallic fuel would undoubtedly require an extensive and expensive R&D program, with no real assurance of success. Consequently, metallic fuel will not be considered for the composite reactor.

Ceramal fuel — mixed UO_2 and BeO, in this case — does not show any significant improvement over UO_2 in fuel utilization. However, because the thermal conductivity of UO_2-BeO is appreciably higher than that of UO_2 alone, resulting in lower fuel temperatures, ceramal fuel might give significant improvements in reactor safety under accident conditions. For the purpose of the present study, ceramal fuel will not be considered, since no apparent improvement in fuel utilization would result. Small amounts of BeO in the UO_2 fuel matrix could be considered as a means of adjusting fuel specific power, if needed.

As a result of the independent consideration of the potential improvements discussed above, three have been selected for incorporation into the composite advanced PWR concept, as follows:

- Mechanical Spectral Shift;
- reduced power density; and
- spent fuel as blanket material.

A conceptual reference design will be developed encompassing these three features, which offer a potential for reduced, 30-year resource requirements in a once-through reactor.

3.0 POTENTIAL IMPROVEMENTS

3.1 Reduction in Control Poison Losses

The principal method of reducing control poison losses is the use of an inert material to vary the water-to-fuel ratio, a concept first suggested to ACDA in a preliminary assessment report in March, 1980.* Termed "Mechanical spectral shift," the concept uses movable rods of an inert material, such as BeO, to continuously adjust the neutron spectrum and, hence, to vary the level of excess reactivity. Early in a fuel cycle, when excess reactivity is potentially present, the mechanical spectral shift rods would be inserted, displacing water and causing the excess neutrons to be absorbed in the U-238 resonances, producing plutonium. Later in the fuel cycle, the rods would be removed, softening the spectrum and increasing reactivity both from reduced U-238 absorption and from the added plutonium produced earlier in the cycle.

The initial evaluation indicated the possibility that as much as 9% Δk might be achieved with ~25% of the fuel assembly volume containing movable, water-displacing BeO. Further evaluation resulted in a lower range of reactivity control in a practical design (see Appendix A for design considerations), largely because of the rod configurations possible, the mechanical clearances required, and the necessity for water-displacer guide tubes. Figure 1 illustrates a possible fuel assembly arrangement utilizing a hexagonal rod array, and Fig. 2 is a cross-section of the water-displacer rod. With this arrangement, the overall water-to-fuel volume ratio is 2.08 with the rods out and 1.66 with the rods inserted. The reactivity control increment, rods-in to rods-out, is 5.4% Δk, with BeO as the inert material.

Alternative control materials were also investigated, including a void** for comparison purposes. Calculated reactivity control

*Letter, S. E. Turner to M. M. Hoenig, dated March 19, 1980.

**In practice, a void in the control rod would be unacceptable because of the potentially-large reactivity addition that could result from flooding of the void space in the rods.

increments were 5.86% Δk for void, 5.63% Δk for graphite, 6.50% Δk for Al_2O_3 and ∿15% Δk for a mixture of BeO and 5% ThO_2 by weight. The latter composition, BeO and ThO_2, was calculated to investigate the possibility of augmenting the control range of BeO should this prove necessary.

The concept of movable rods of inert material for reactivity control by mechanical spectral shift appears to be feasible and should result in reduced U_3O_8 resource requirements. Therefore, mechanical spectral shift will be incorporated in the composite reactor design.

3.2 Reduced Neutron Leakage

3.2.1 Lower Power Density Cores

Increasing the size of the reactor core at a given power level reduces the core average power density and results in a reduction in the fraction of neutrons lost through leakage. Reduced neutron leakage in turn, allows attaining the design fuel burnup with lower fuel enrichment and thereby improves the resource utilization. However, reduced power density also reduces the fuel specific power and results in an increase in the duration of a fuel cycle (refueling interval). Because of the poorer resource utilization of the startup fuel cycles prior to reaching equilibrium, an increase in refueling interval tends to increase the penalty in resource utilization associated with the startup fuel cycles. Furthermore, an increase in the radial dimension of the reactor core necessitates a larger pressure vessel which will impose an economic penalty due to the capital cost increase. Fuel cycle costs will also be increased due to higher interest charges as a result of the lower fuel specific power.

To investigate the potential improvement in resource requirements resulting from reduced power density, a simple approximation was made that the effective radius of a core with a flattened power distribution was inversely proportional to the square root of the power density. With this approximation, the increase in reactivity due to reduced neutron leakage can be estimated and the (estimated) enrichment required for 50,000 Mwd/mtU burnup derived.

The reduced power density not only reduces the number of neutrons lost in leakage, but also reduces the number of neutrons lost in xenon absorption (plus a small increment from the Doppler effect with reduced fuel temperatures). Figure 3 illustrates the incremental reactivity achievable by reduced power density. Despite the incremental reactivity resulting from reduced power density, the specific power in the fuel is also reduced, which extends the reactor residence time of the fuel elements and results in a larger startup penalty from the initial core loading. For example, at 40% of the reference fuel specific power, the reactor residence time for each fuel assembly is ∿9 years, which is nearly 30% of the reference life (30 years). Figure 4 illustrates the trend in resource requirements with power density (and fuel specific power), showing an optimum at about 60% of the reference power density.

In the assessment of low power density cores, the effect of power distribution introduces some uncertainty into the calculations. Preliminary data discussed above assumes essentially the same power distribution as the reference case. However, when the power density is reduced, a larger power peaking can be tolerated without exceeding performance limitations in the hottest fuel assembly. By adjusting the power distribution to result in low power near the periphery of the core, the reactivity gain from low leakage can be achieved without necessarily reducing the average core power density as far as the optimum indicated by the data in Figs. 3 and 4. In Fig. 4, the increase in resource requirements at the low values of power density results from the increasing importance of the penalty associated with the startup fuel cycle. Detailed evaluation of this effect, together with the fuel management scheme and assembly-wise burnup histories is beyond the scope of the present analysis.

It is evident that an improvement in resource utilization is possible with reduced core power density, so some reduction in power density is deemed desirable in the advanced PWR concept. However, to avoid excessive incremental costs due to increases in pressure vessel size and to interest charges on the fuel, the power density in the advanced concept should not be less than about 70% of the reference value. Furthermore, at 70% power density in the core, the required reactor containment size is not expected to be significantly affected. At much lower power densities, there is the possibility of a larger containment vessel being required, not only because of the larger pressure vessel but also to accommodate the higher blowdown energy resulting from the larger coolant inventory in the primary system.

3.2.2 Alternative Radial Reflector Materials

In current pressurized water reactors, the radial reflector is effectively a mixture of discrete regions of water and stainless steel. Calculations were made with two alternative blanket materials — BeO and spent fuel — to investigate the potential improvement in fuel utilization. By using spent fuel (i.e., fuel that would otherwise be discharged to the storage pool), a dichotomy in terminology arises. The spent fuel loaded into reflector positions could be considered as a "blanket." Yet the spent fuel assemblies obviously resemble core fuel assemblies, contain fissile material, and produce power, although at a low power density. Therefore, a spent fuel "blanket" could also be considered to reflect a modified fuel management scheme in which the outer row of fuel assemblies ("blanket") constitutes an additional zone of the core, effectively reducing the overall core power density. Despite this, the convention adopted here, for convenience, is to refer to the spent fuel in reflector positions as a blanket. Similarly, BeO located in peripheral core positions is referred to as "blanket," with water outside the core being the reflector.

Two methods of evaluating the relative effects of the two different reflector materials were used. In the first method, a two dimensional planar cross-section PDQ7 calculation was made, describing a 3-zone reactor core surrounded by the blanket (in one calculation, water was used as the blanket positions to establish a reference). With a BeO blanket (assumed to be a block of pure BeO of the same size as a fuel assembly), the core reactivity was 3.9% Δk higher than with a water reflector only. With a spent fuel blanket, the core reactivity was only 1.4% Δk higher. Although these data might appear to suggest that BeO is the better blanket, when the additional burnup accumulated in the spent fuel reflector ($\sim$4250 Mwd/mtU) is taken into account, there is very little difference in resource utilization for the two blankets. Furthermore, inspection of the output edit in the PDQ results revealed that the power distribution in the core region was more highly peaked in the more reactive fuel in the core. This means that, in practice, the core power distribution would have to be re-adjusted and the apparent improvement for either reflector would be smaller.

To avoid the difficulty introduced by shifting power distributions, the second method of comparing the relative benefits of BeO and spent fuel blankets was to use a single fuel region of uniform reactivity. In this case, the BeO gave only a 2% Δk increase in reactivity (reduced buckling) compared to a negligible increase with a spent fuel blanket. Increasing the BeO thickness did not result in any significant further increase in reactivity. However, the spent fuel produced about 5% additional power, which approximately compensates for the reactivity increment with a BeO blanket. In addition, a practical BeO blanket, with necessary water cooling, would result in a smaller reactivity increment for the BeO case.

It is concluded that the potential improvements with either a BeO or a spent fuel blanket are comparable. Since BeO would represent an additional expense compared to spent fuel (available at no additional cost after the first fuel cycle), a spent fuel blanket would appear to be the more economical choice. Consequently, the composite PWR concept will utilize spent fuel as a blanket.

3.3 Alternative Fuel Design Concepts

3.3.1 Cermet Fuel

The potential attractiveness of cermet fuel lies in the very high fuel burnups achievable, in terms of megawatt-days per metric ton of contained uranium. Fuel burnups of the order of 200,000-300,000 Mwd/mtU are not inconceivable and have been attained in some early research reactors (e.g., the MTR). The viability of cermet fuel depends upon the ability to achieve very high effective fuel burnups with adequate core reactivity performance. Fuel utilization is directly proportional to the fuel burnup and inversely proportional to the feed-to-product ratio in the enrichment process, as indicated by the following relationship from reference 1:

$$\text{fuel utilization, Mwd/ST } U_3O_8 = 0.76926\ B\ \frac{(0.711 - E_T)}{(E - E_T)}$$

where B is the discharge fuel burnup, E is the fuel enrichment in wt% U-235 and E_T is the tails enrichment in the enrichment process.

It should also be recognized that reducing the UO_2 content of the fuel pins proportionately increases the fuel specific power for the same average pin power. Since the fuel cycle length is inversely proportional to specific power, the increased specific power has to be compensated by correspondingly higher fuel burnups in order to achieve reasonable operating periods. Furthermore, the increased H-to-U ratio would increase and affect adversely the temperature coefficient of reactivity, unless design changes were made to drastically reduce the lattice spacing. This, in turn, would increase the average core power density and pressure drop, necessitating complete core redesign. While these considerations do not necessarily represent insurmountable difficulties, the minimum gain and possible loss in resource utilization does not justify pursuing the concept further.

3.3.2 Mixed UO_2-ThO_2 Fuel

The use of ThO_2 mixed with the UO_2 of the fuel has been investigated to determine if the higher conversion ratio resulting from the higher cross-section in thorium could potentially lead to improved fuel utilization. Results of this investigation led to the conclusion that the higher conversion ratio of thorium does not appear to compensate for the higher enrichment required to yield the necessary reactivity. Consequently, a net loss in resource utilization was found to occur with mixed UO_2-ThO_2 fuel. In part, the loss in resource utilization results from the amount of U-235 discarded in the tails during the enrichment process, which increases with increasing product enrichment. Consequently, it is concluded that there appears to be no advantage in using mixed UO_2-ThO_2 fuel in a once-through reactor concept. ThO_2 rods could conceivably be used to advantage in reactivity and/or power peaking control, with subsequent recovery of neutrons by using the ThO_2 rods as a blanket. This possibility, however, was not pursued in the present evaluations.

3.3.3 Ceramal Fuel

Mixed ceramics — for example, BeO and UO_2 — called ceramals appear to be viable in the range of 10% to 20% by volume BeO. Although some increase in fuel specific power results, the increase does not appear to be excessive (25%) for 10% by volume BeO. Increased specific power reduces the cycle lifetime (refueling interval) for a given discharge fuel burnup. Some experience with UO_2-BeO ceramals has been reported, and other ceramic mixtures could be considered. The greatest potential benefit of ceramal fuel — especially UO_2-BeO — results from the significantly-improved thermal conductivity and the greater resistance to swelling under reactor operating conditions.

Calculations with UO_2 fuel containing approximately 20 vol. % BeO indicate an almost insignificant change in the burnup dependence of reactivity, which, in turn, suggests very little improvement in fuel utilization. Since the primary objective of the present study is to seek methods of improving fuel utilization, the absence of significant improvement in fuel utilization with ceramal fuel reduces their consideration to a very low priority in this study. It should be noted, however, that because of the increased thermal conductivity, ceramal fuel has a potential for improving reactor response under some postulated accident conditions.

3.3.4 Metallic Fuel

The use of metallic fuel has been considered in a prior study[1], which concluded that there was no significant advantage to the use of metal fuel in current reactors as a retrofittable concept. In the present study, changes in lattice spacing were considered to more nearly optimize metallic fuel in pressurized water reactors. For a given fuel enrichment, a somewhat higher optimum reactivity than for UO_2 fuel can be achieved. This reuslts from the higher-density fuel, which increases the probability of neutron capture in fuel rather than in parasitic absorbers, in addition to a small contribution from increased fissions by fast neutrons.

Assuming that any desired discharge fuel burnup could be achieved with metallic fuel, calculations have been made to determine the potential for improved fuel utilization. Figure 5 shows parametrically the 30-year U_3O_8 requirements with metallic fuel, for various combinations of core regionalization and discharge fuel burnup. These data indicate that significant improvement in fuel utilization can potentially be achieved with metallic fuel. Results shown in Fig. 5 for metallic fuel may be compared with the corresponding data for UO_2 fuel given on Fig. 5.5 of reference 1. If resource utilization is compared for metallic and oxide fuels at high discharge burnups (46,000 Mwd/mtU) from a 10-zone core, the metallic fuel shows an improvement of about 10% in 30-year U_3O_8 requirements; in addition, the refueling interval is roughly doubled with the metallic fuel. Achieving discharge fuel burnups of this magnitude with metallic fuel raises serious questions concerning the stability, safety, and performance of the metal fuel. These concerns are addressed in Appendix B. Despite the potential improvement, uncertainty in the feasibility of achieving high burnup, the probable cost of the associated R&D effort, and the serious safety concerns do not justify further investigation of metallic fuel.

3.3.5 Summary of Alternative Fuel Design Concepts

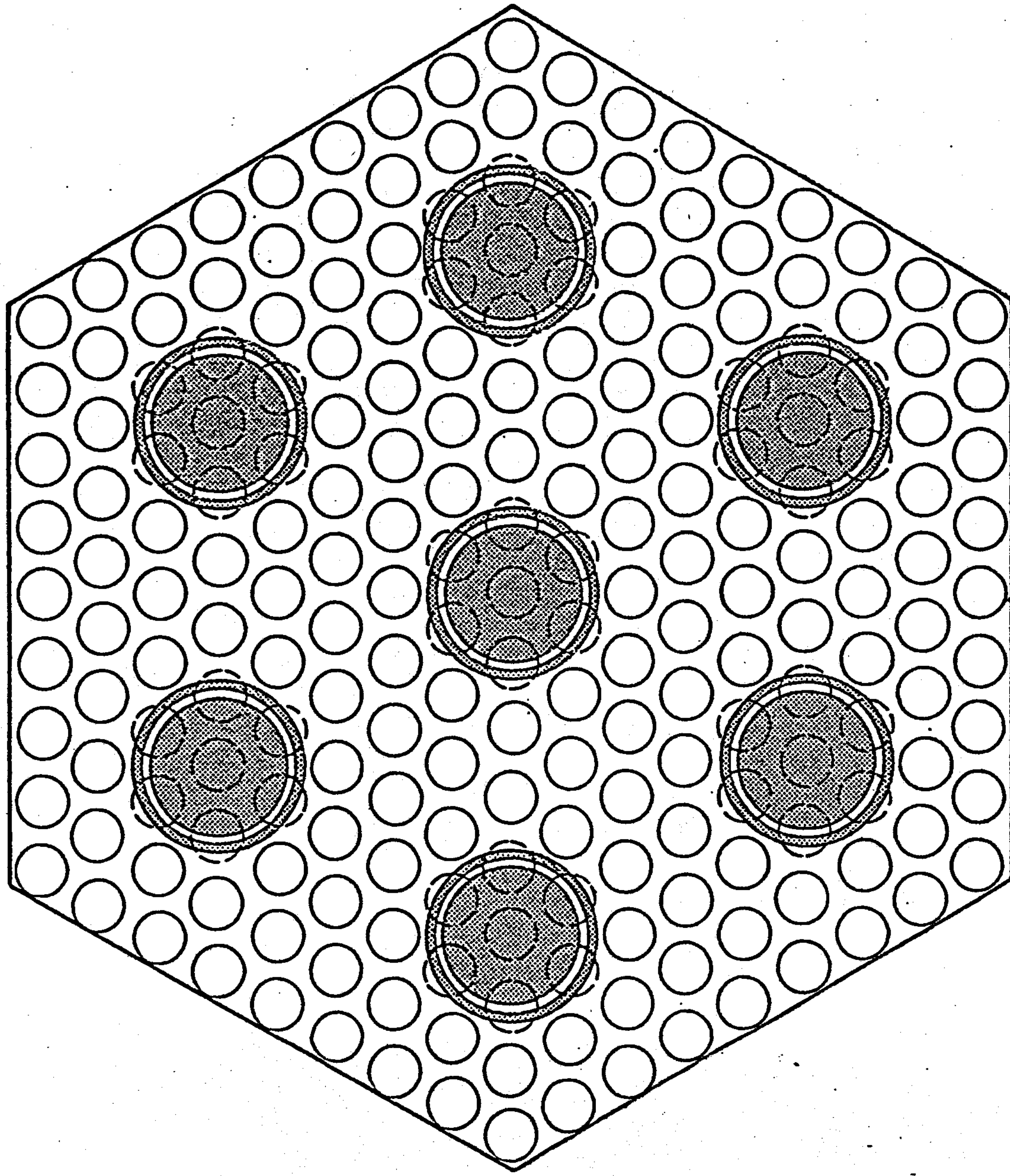
Of the various alternative fuel design concepts considered, only one appears to offer any significant improvement in resource utilization; that one is metallic fuel if high burnups could be achieved. At the present time, there is no reason to believe that the necessary R&D to develop metallic fuel for pressurized water reactors would be successful. In addition, the use of metallic fuel would raise serious safety questions, particularly when considering fuel failure and potential uranium-water reactions. Consequently, metallic fuel will not be considered in the advanced PWR concept.

Cermet fuel and mixed UO_2-ThO_2 fuel do not offer any improvement in resource utilization and will not be considered further. Ceramal fuel could potentially offer a significant improvement in reactor safety. However, since ceramal fuel does not appear to offer improved resource utilization, its use will be restricted to a low-priority option.

REFERENCES

1. S. E. Turner and M. K. Gurley, Retrofittable Modifications to Pressurized Water Reactors for Improved Resource Utilization, Report SSA-128, Southern Science Applications, Inc., October 1980.

2. D. H. Gurinsky and S. Isserow, Nuclear Fuels, in <u>The Technology of Nuclear Reactor Safety</u>, <u>Volume 2</u>, <u>Reactor Materials and Engineering</u>, the M.I.T. Press, Cambridge, 1973.



217 TOTAL POSITIONS

49 replaced by water displacers

168 fuel rods

Fig. 1. Cross-section of hexagonal fuel assembly with water-displacer rods.

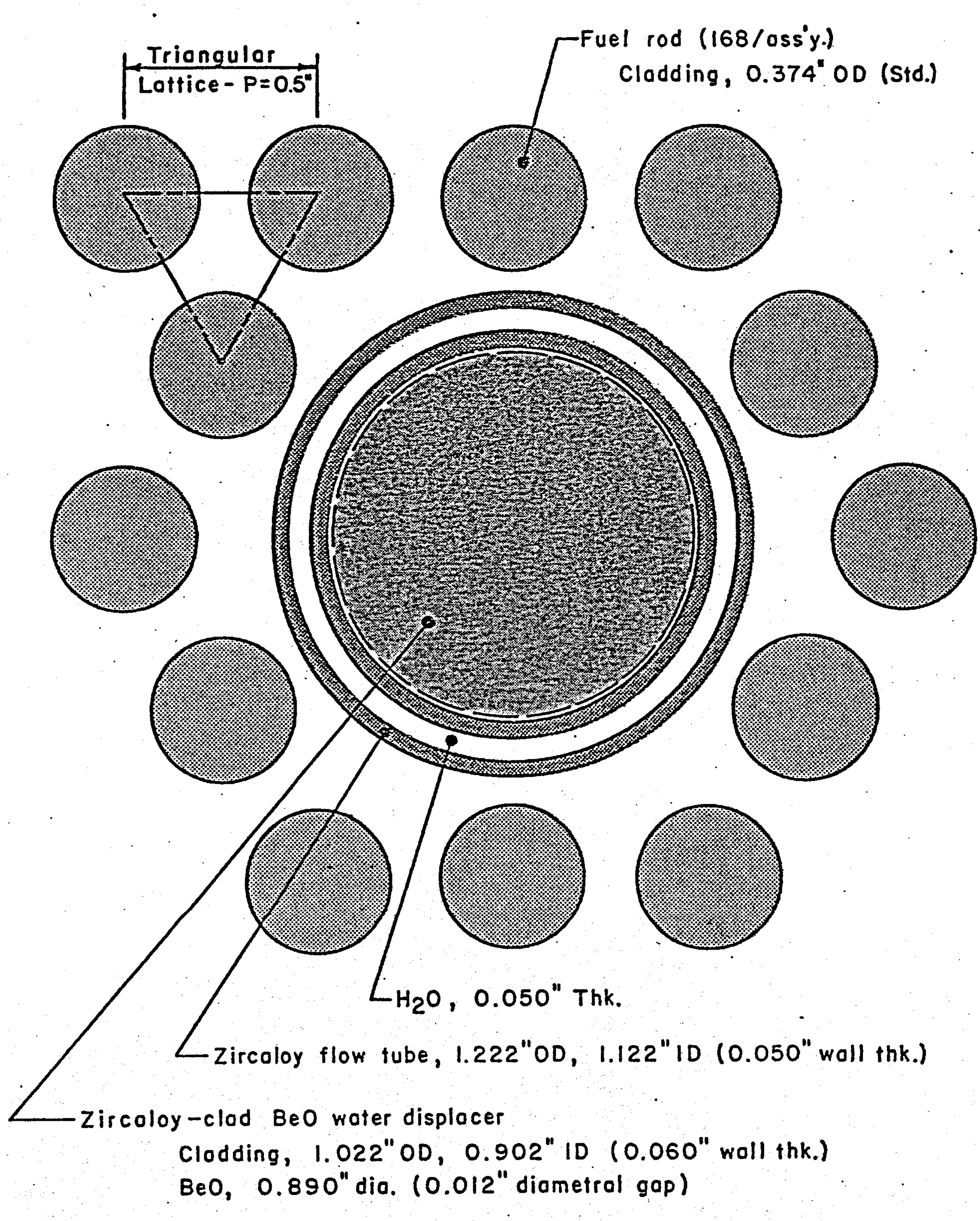


Fig. 2. Cross-section through water-displacer rod.

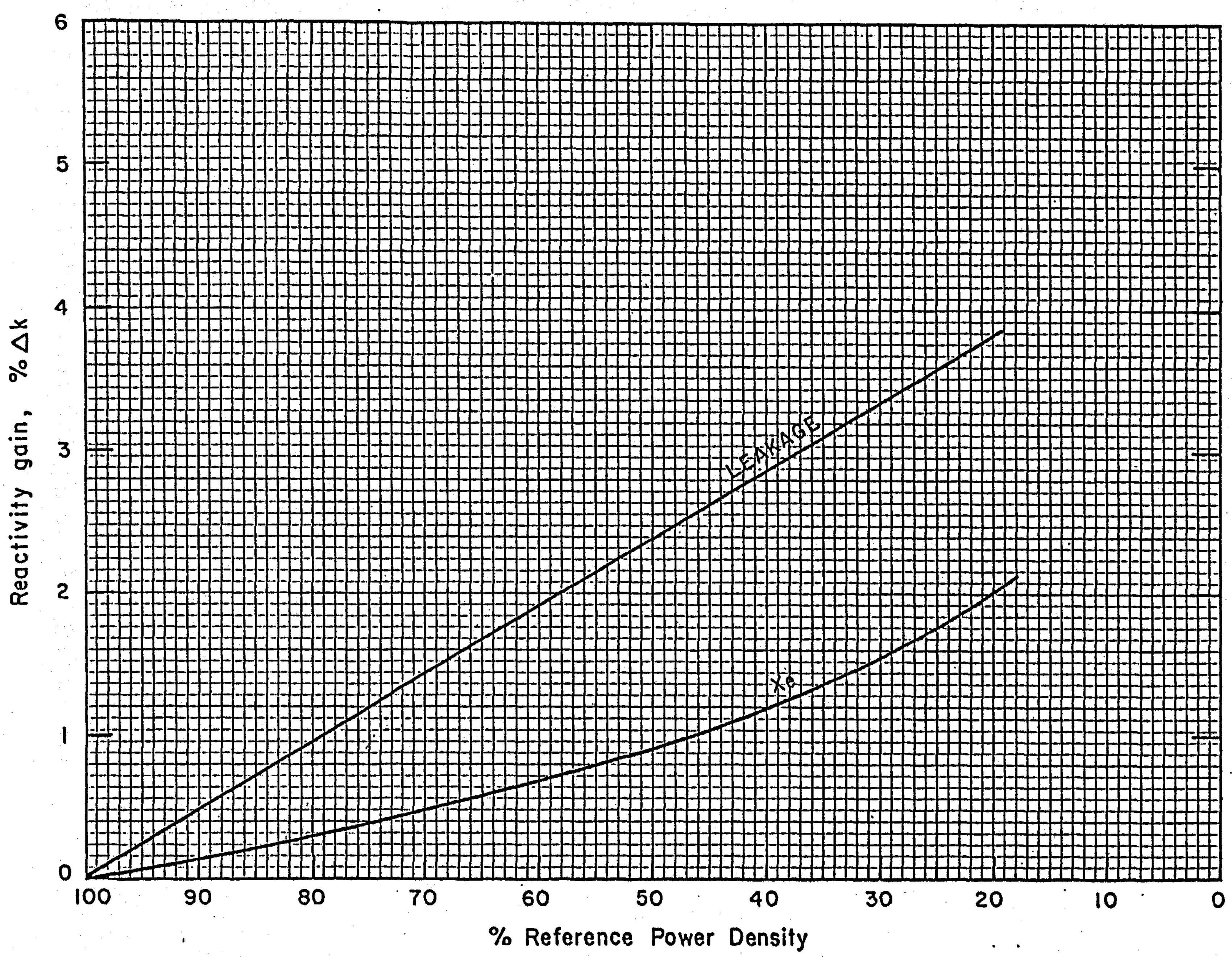


Fig. 3. Reactivity gain as a function of core power density.

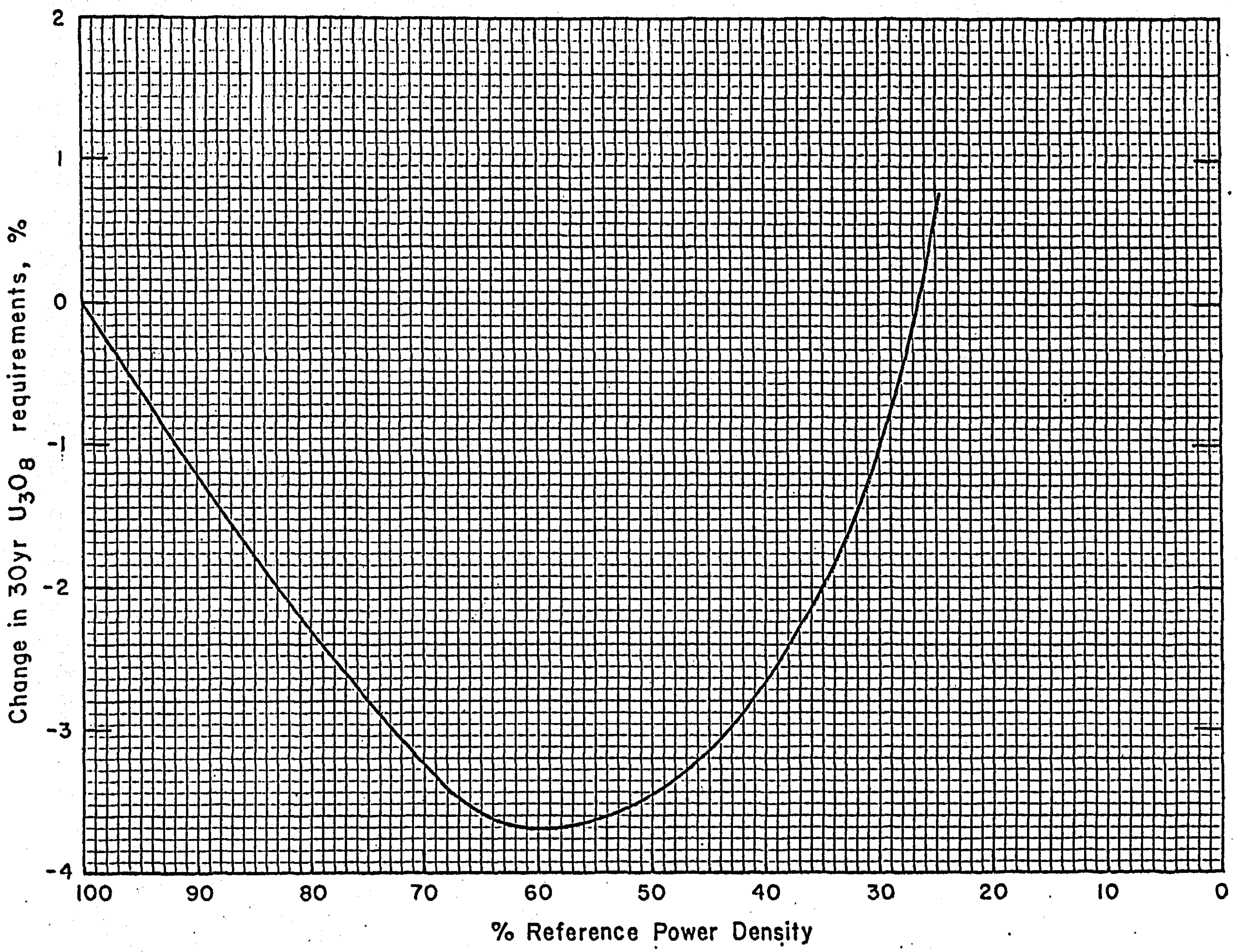


Fig. 4. Change in resource requirements as a function of core power density.

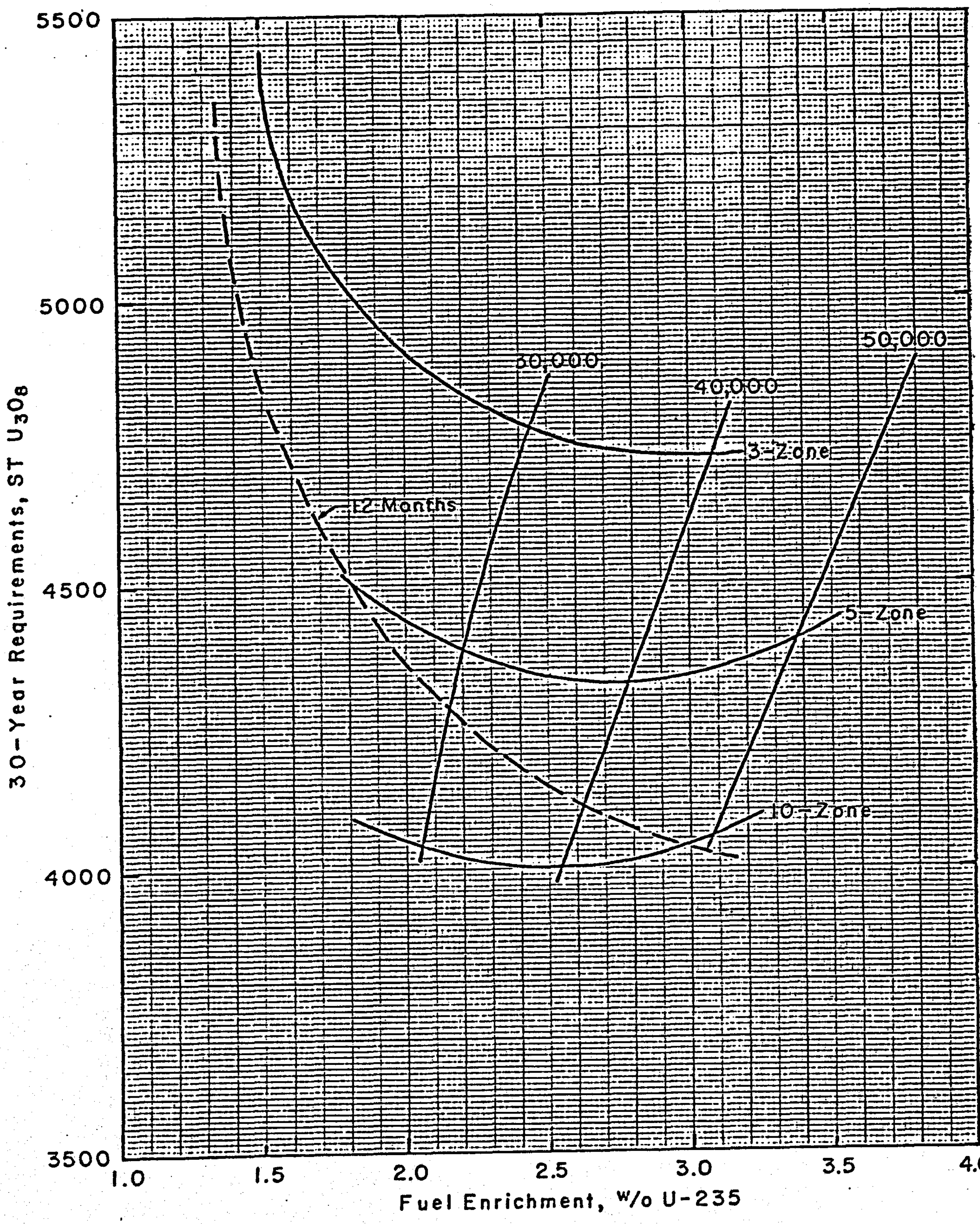


Fig. 5. Resource requirements for metallic uranium fuel.

APPENDIX A. FUEL-ASSEMBLY ARRANGEMENT CONSIDERATIONS

The preliminary effort consisted of an investigation of the basic layout of the principal components of a fuel assembly — the fuel rods and the water-displacing rods that will be inserted to reduce the water/fuel ratio in the assembly and withdrawn to increase the water/fuel ratio, thereby effecting changes in reactivity.

The basic assumptions for the fuel-assembly layout were as follows:

- Each fuel assembly would have movable, water-displacing elements containing beryllium oxide (BeO), much like the rod-cluster-control elements in present PWRs; and
- each fuel assembly would be designed for a low water/fuel ratio in an infinite array of fuel rods, which indicates the use of a triangular rod-lattice (hexagonal array or assembly) to increase the rod-to-rod clearance relative to that of an equally-dry square rod-lattice.

For the degree of "dryness" anticipated to be in the optimum range for the water-displacer concept being investigated, a triangular rod-lattice spacing of 0.50 inches was used. This value, coupled with reasonable values for the amount of water that could be displaced by the BeO elements within the fuel assembly, gave overall water/fuel ratios with BeO elements in and out of the fuel assembly that are in the range of usefulness for spectral-shift control of the reactor.

The standard fuel rod cross-section was assumed to be that specified by Westinghouse in the RESAR-414 document dated October, 1976. This fuel rod design is the standard Westinghouse design that had been used in the earlier RESAR-3S and incorproated in numerous nuclear plant designs. The cross-sectional dimensions of the Westinghouse fuel rod are as follows.

Outside diameter, inches	0.374
Diametral gap, inches	0.0065
Cladding thickness, inches	0.0225

The fuel material is UO_2 and the cladding is Zircaloy-4, materials that would be used in the advanced PWR studied here.

Several configurations were investigated for the water-displacer elements, most based upon an element that would use the space normally occupied by seven fuel rods in the triangular lattice arrangement. (Seven rods form the smallest hexagon within the overall fuel assembly and, with reasonable spacing between the seven-rod hexagons within the overall assembly, also yielded a value of water displacement that appeared to be in the desirable range.) The water-displacer positions within the fuel assembly were assumed to consist of a fixed guide channel containing a movable, clad-BeO element.

In horizontal cross-section, the configurations of the water displacers and their guide channels varied from a simple (clad) rod within a circular tube to elements with a hexagonal cross-section moving within hexagonal flow channels. Some of the water displacers and guide channels considered are shown in Fig. A-1.

As the different water-displacer elements were considered, it was determined that the extreme steps taken with some configurations to reduce the water present in the core were not necessary — that the important factor was the change in the amount of water present, and that the triangular pitch, alone, could yield a sufficiently dry core with any of the control element shapes. Consequently, elaborate configurations such as the fluted guide channel shown in Fig. A-1 were rejected in favor of the simpler rod-in-a-tube arrangement.

With the rod and tube arrangement, at least when it is considered to replace several fuel rods (in this case, seven), an overall fuel assembly arrangement reminiscent of the Combustion Engineering design results: that is, the control elements are substantially larger than fuel elements, and the number of control elements within a fuel assembly is smaller than the number in the Westinghouse design.

Before investigating the arrangement possibilities of several water displacers within a fuel assembly, it was necessary to establish the basic dimensions of the displacer — in this case, the rod-in-tube displacer that replaced seven fuel rods. The design selected for the basic element is shown in Fig. A-2. As the illustration shows, the Zircaloy-clad BeO water-displacer rod moves within a Zircaloy flow tube. The outside diameter of the flow tube was set to give the same clearance between the closest fuel rod and the flow tube as the clearance between guide thimbles and fuel rods in the Westinghouse design (0.068 inch). Tube wall, water gap, and cladding thicknesses were estimated on the basis of values used in the C-E design (relatively large control rod), as was the diametral gap between the BeO and its cladding (with materials properties also considered).

Some consideration was given to the water gap that would exist around the outside of the hexagonal fuel assembly. A check of the C-E and Westinghouse designs shows that the spacing between rod surfaces in adjacent assemblies is 0.203 in. and 0.156 in., respectively. These values exceed the normal, in-assembly, rod-to-rod gaps in the C-E (0.124 in.) and Westinghouse (0.122 in.) designs by 0.079 in. and 0.034 in. These differences are consistent, in that the C-E design is generally a bit huskier and more conservative-appearing. Since the hexagonal fuel assembly in the present study has six surfaces to mate, compared to four in the square C-E and Westinghouse designs, a conservative value was chosen for the increase in water gap above that existing between fuel rods in the triangular-pitch design. The "extra" water gap between fuel assemblies was set at 0.060 in.

Another item that was considered necessary to review briefly was the effect on coolant flow area and, hence, velocity, of using the triangular pin lattice in place of the square one. The Westinghouse RESAR-414 document gives the following information for a current PWR.

Fuel rods, number	50,952
Effective coolant flow rate, lb/hr	143.7×10^6
Effective flow area, ft^2	51.1
Nominal coolant pressure, psia	2250
Average coolant temperature, ^oF	598

The average core coolant velocity in a current PWR can be calculated from the data to be 18.1 ft/sec, which checks present practice. If we assume the advanced PWR will have a thermal power and a coolant flow rate equal to those of the RESAR-414 design, and that it will have the same number of fuel rods, then the reduced flow area per rod associated with the triangular lattice will result in an increase in average coolant velocity in the core to slightly over 24 ft/sec. Although this value exceeds that based on RESAR-414, it is in the acceptable range and leaves the only question an economic one (pressure drop/ pumping power).

Before proceeding to develop the layout(s) of the fuel assembly in terms of how many fuel rods and how many water displacers, it was necessary to think about the grids or other structural components that would be used to insure proper spacing of the fuel rods and displacer guide tubes. Although the guide tube in the design described here is not a multiple of other dimensions in the fuel assembly (as it is, or nearly is, in the C-E and Westinghouse assemblies), the grid spacer designs used in present PWRs should be adaptable to the advanced fuel assembly. There may be a requirement that the guide tubes themselves be a part of the grid spacer, or that short sleeves for the guide tubes be incorporated into the spacers. In any case, low-neutron-absorption spacers are not thought to be a problem, based on U.S. practice to date, as well as the use by the Russians of the "four-story-spacer" in the Novovoronezh PWR. The latter design may be an improvement, in that it may result in a greater longitudinal smearing of the required structural material. In any case, grid spacing was investigated briefly, with the conclusion that an acceptable detailed design could be developed.

With the principal components of the fuel assembly established conceptually, the next effort was centered on devising patterns of fuel rods and water displacers that would yield changes in moderation in what was thought to be the range of interest. Details of the fuel assemblies considered are given in Table I.

As the numbers in the table show, the four concepts give percentages of fuel rods replaced by water displacers ranging from about 23% to 38%. For control purposes, it appears that a fuel-replacement value of around 25% is

Table I. FUEL-ASSEMBLY CONCEPTS USING LARGE WATER-DISPLACER ELEMENTS

Concept No.	Total No. of Lattice Positions in Assembly	No. of Fuel Rods	No. of Displacer Rods	Central Instrumentation Position	Percent of Fuel Rods Replaced by Water Displacers
1*	217	168	7	No	23
2	271	186	12	Yes	31
3	331	240	13	No	27
4	331	204	18	Yes	38

desirable, so there are two or three concepts listed in the table that, upon detailed calculational evaluation, may be in the range, or may be adjustable to the range, of water change that will provide the necessary operational flexibility.

The selection of Concept No. 1 (217 lattice positions, 168 fuel rods, 7 large water displacers) as the initial design for additional study led to several thoughts on the physical/mechanical arrangement of the core and its constituents. In developing the overall core layout, the following conditions were assumed to apply:

- The advanced core consists of fuel rods containing a six-inch length of natural UO_2 at the top and bottom ends of each rod (above and below the low-enriched UO_2 of the active core); and
- the advanced design operates at the same overall power level and has the same total length of active fuel in the core as the RESAR-414 design.

The overall length of a fuel rod was taken to be the same in the advanced design as in the RESAR-414 design, so the advanced core would require more fuel rods

*See Fig. A-3.

than the RESAR-414 core if peaking factors are not allowed to increase (because the advanced core uses a fuel rod whose active length is 12 inches less than the RESAR-414 rod). For this case, the difference in active rod length increases the number of fuel rods required for the advanced core to 54,872. If one assumes that the fuel assembly identified as Concept 1 in Table I, above, is used, then the core of the advanced reactor will require approximately 327 assemblies. With 327 fuel assemblies (more or less, depending on the possible layouts with the hexagonal cross-section), and with each assembly containing movable BeO water displacers, the core obviously will require a large number of water-displacer drive mechanisms if the displacers in each assembly are driven in the manner of ganged finger rods in a present PWR.

The present Westinghouse design uses 57 clusters of finger rods, each cluster individually driven; the older Westinghouse design described in RESAR-3S used a total of 61, of which 53 were full length and eight were part length. If the advanced PWR discussed here could use a system of ganged drives, with each drive accommodating the water displacers associated with seven fuel assemblies (49 displacers), then around 47 drives would be required — a number comparable with the number of control rod drives used in a current plant. Possible arrangements for ganging the water displacers into large groups were investigated, and it was concluded that such ganging may be possible but that the displacers associated with some of the peripheral fuel assemblies in the core would require special treatment.

In view of the fact that there are many water displacers and drive systems, even when the relatively-large displacers described to this point (ones that replace seven fuel rods) are considered, it was decided to study the possible use of smaller water displacers. The principal disadvantage of smaller water displacers is the increase in the number required, while the principal advantage is the more-finely-divided (or more uniform across the assembly) insertion/removal of water.

For the small displacers, the outside diameter of the flow tube was taken as 0.482 in., the same size as the guide thimbles for control rods in current

Westinghouse designs. This resulted in the dimensions shown in Fig. A-4 for the smaller displacers. With the fuel-rod diameter of 0.374 in. mentioned earlier, and the 0.50 in. triangular lattice spacing, the substitution of displacers for fuel rods on a one-to-one basis gives acceptable clearances. The distance between the surface of the displacer flow tube and the nearest fuel pin surface is 0.072 in. in the advanced design, while the Westinghouse design has a clearance of 0.068 in. between the guide thimbles and fuel rods, as stated earlier.

A hexagonal fuel assembly with 217 total lattice positions, corresponding to Concept No. 1 in Table I, earlier, was selected as the base case for investigating the possible locations for small water displacers within an assembly. One arrangement that has possibilities provides (within the 217-position assembly) 162 fuel rods, 54 water-displacer positions, and a central instrumentation position (See Fig. A-5). This arrangement, which yields a relatively-uniform distribution of water displacers across the assembly, results in the replacement of 25% of the fuel rods by displacers.

It is apparent that a number of arrangements can probably satisfy the control requirements of the core (based on early estimates of those requirements). However, the best mechanical design of the drive components for the water displacers is not obvious and, in fact, the design of the displacers and the size of the fuel assemblies themselves will require additional consideration.

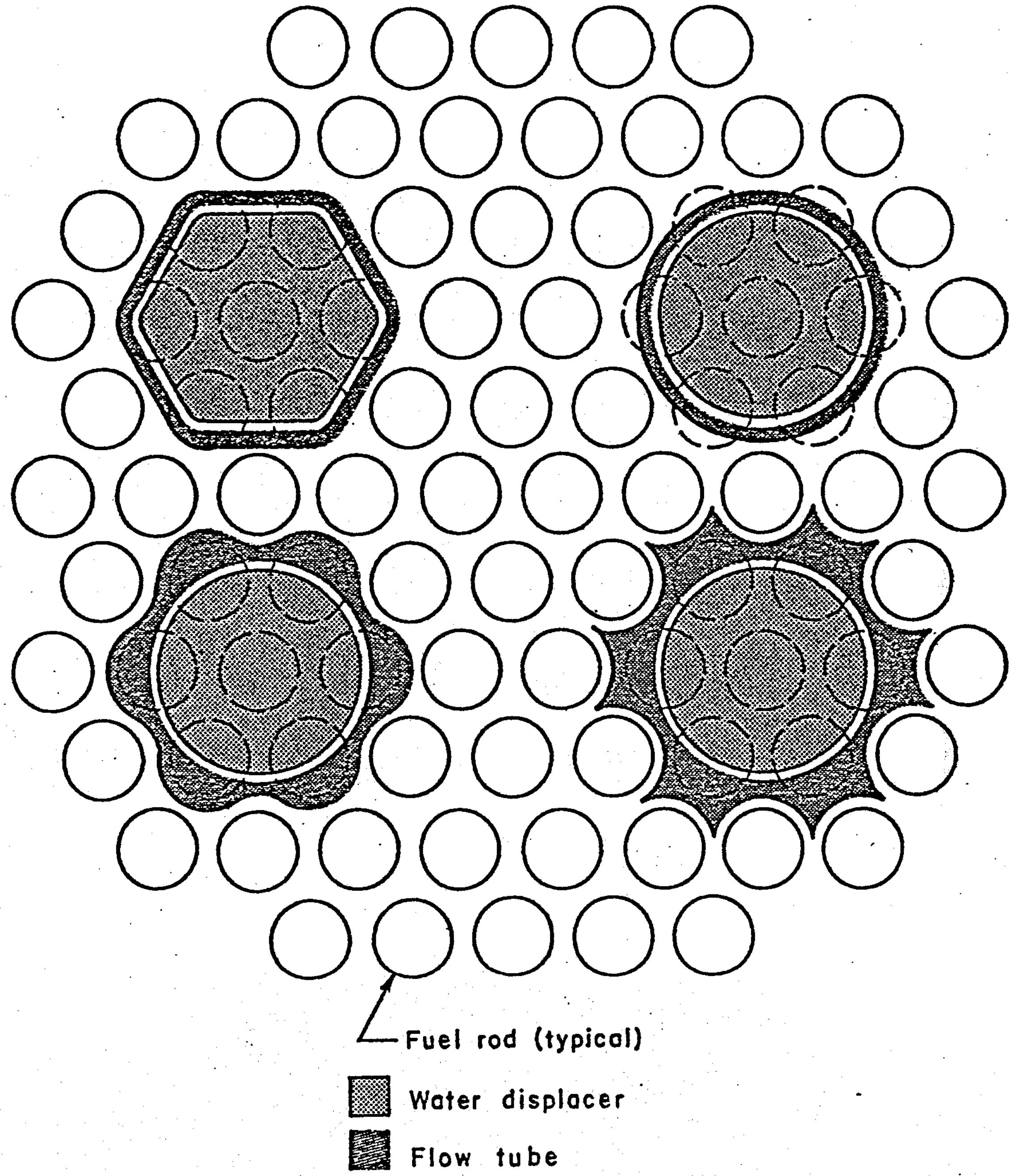


Fig. A-1. Water-displacer concepts (horizontal cross-sections).

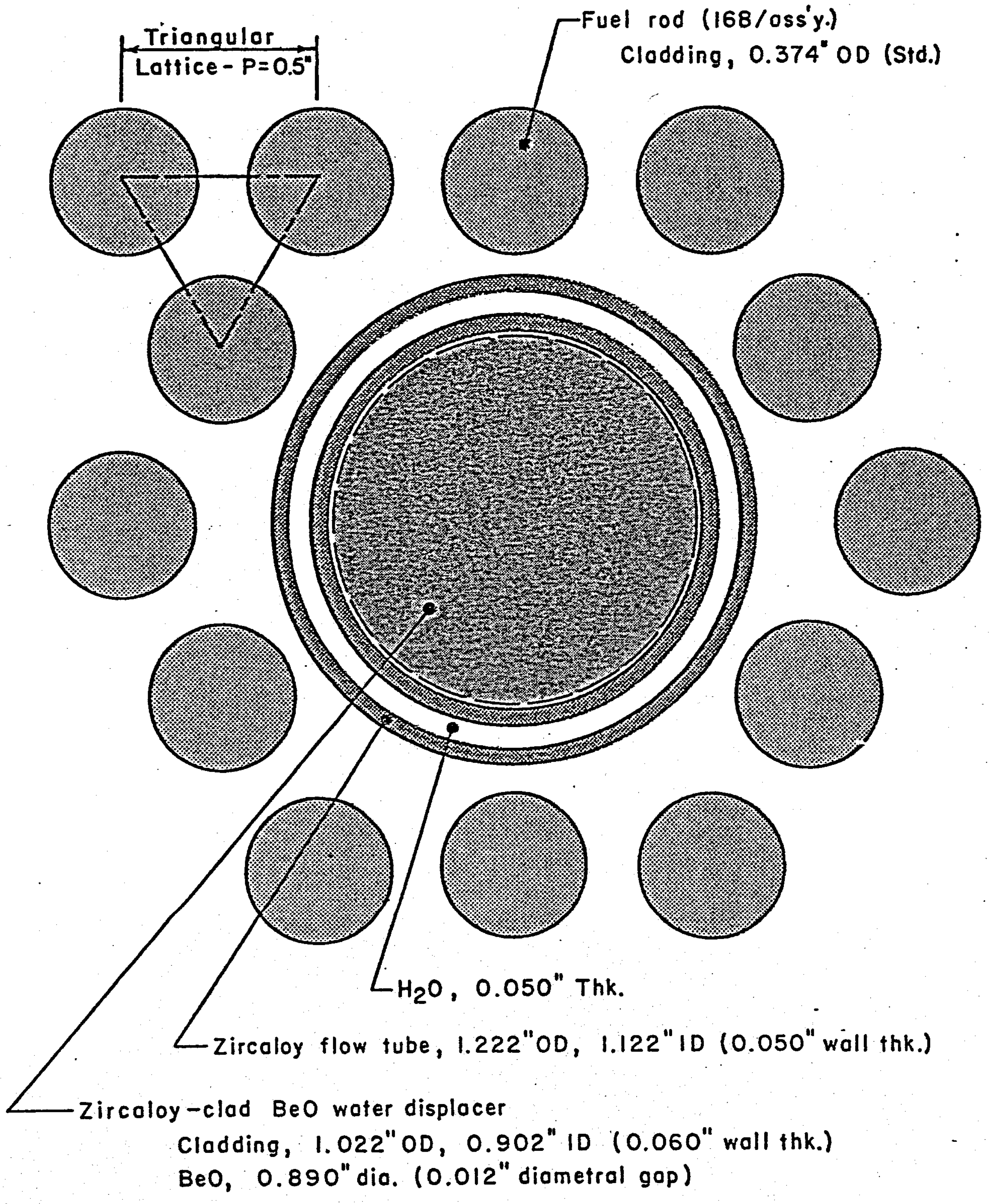
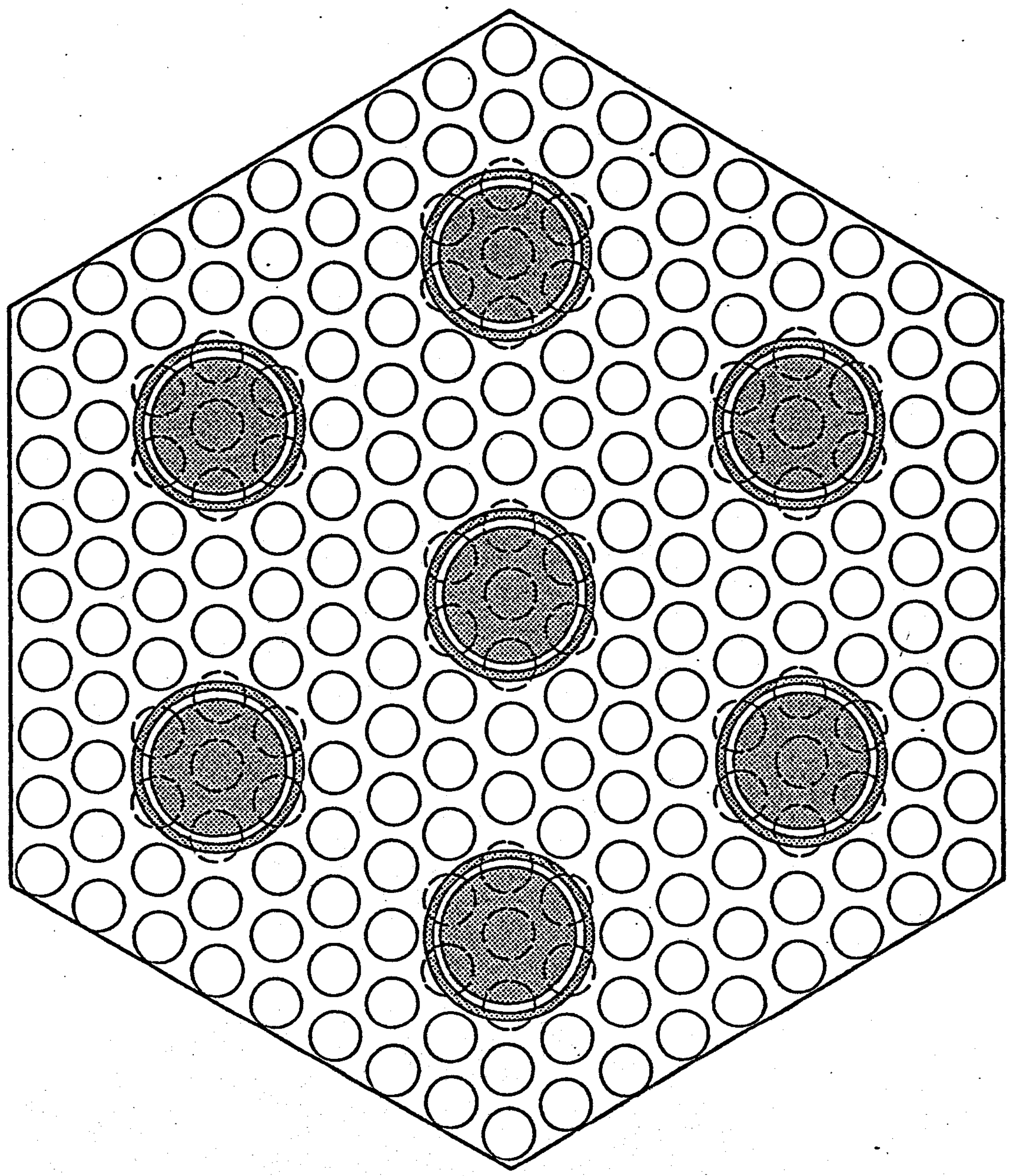


Fig. A-2. Reference water-displacer horizontal cross-section.



217 TOTAL POSITIONS

49 replaced by water displacers
168 fuel rods

Fig. A-3. Horizontal cross-section through fuel assembly containing large water displacers.

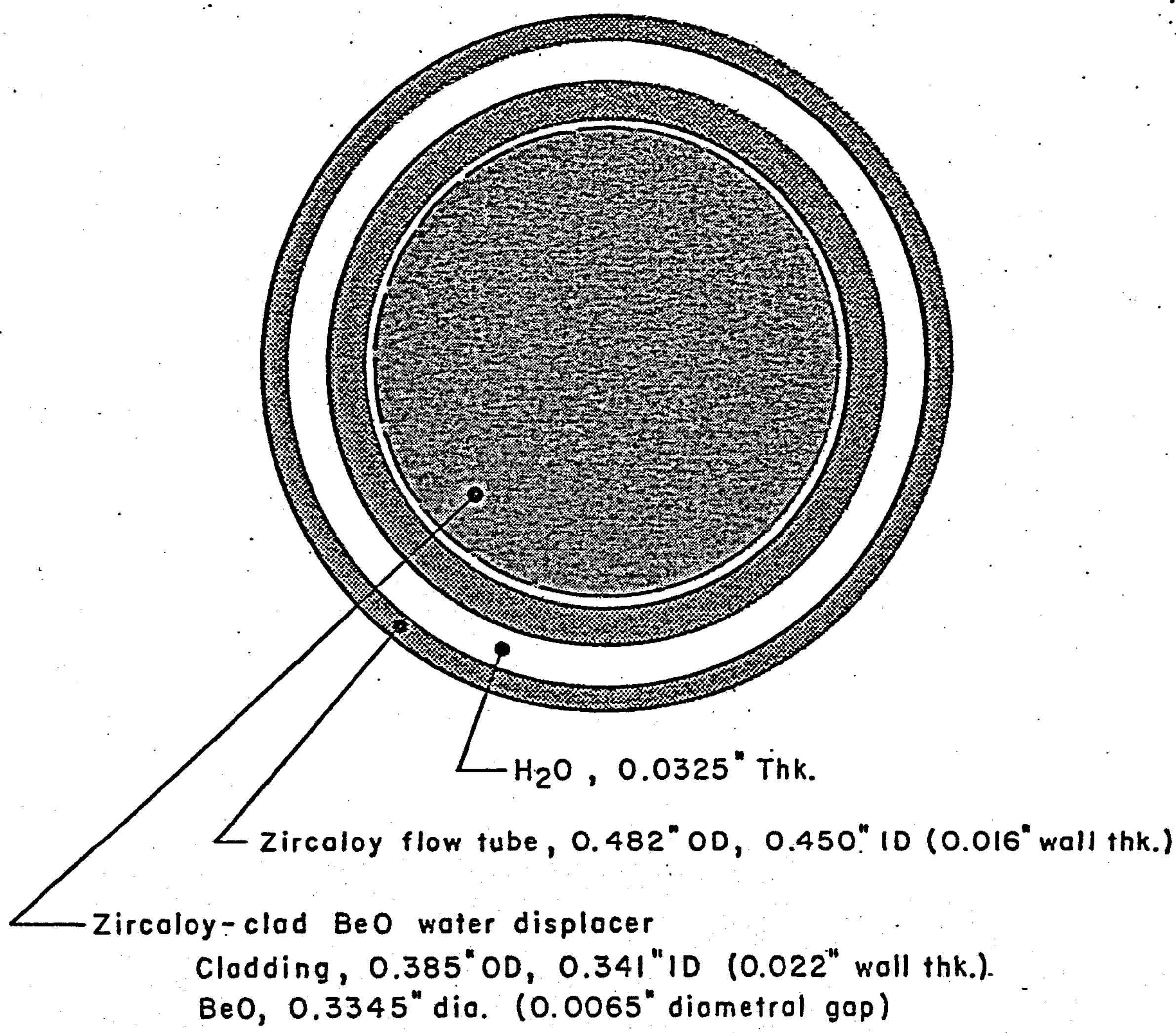
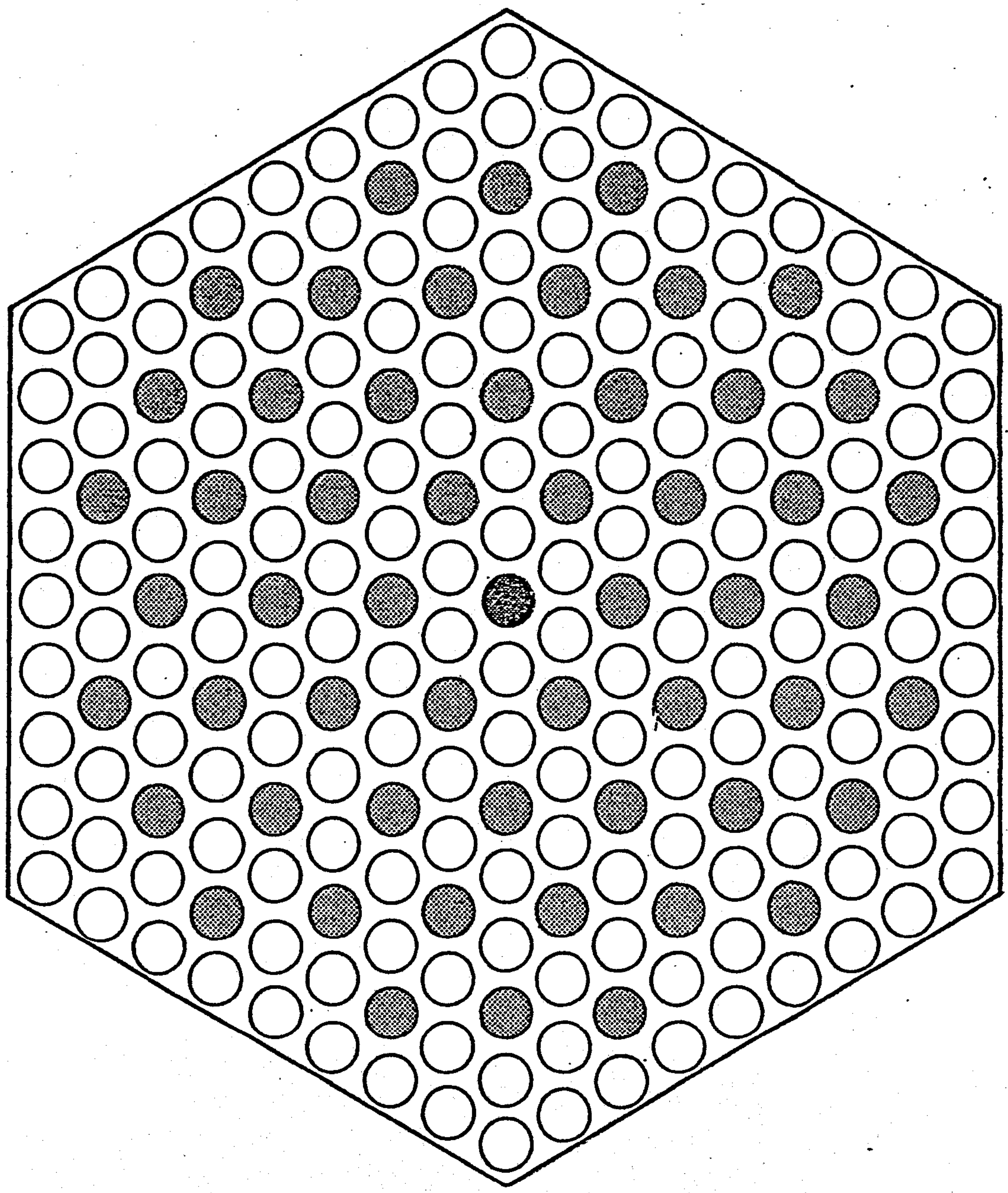


Fig. A-4. Horizontal cross-section through small water displacer.



217 TOTAL POSITIONS

54 replaced by water displacers

1 central instrumentation sheath

162 fuel rods

Fig. A-5. Horizontal cross-section through fuel assembly containing small water displacers.

APPENDIX B. CONSIDERATIONS REGARDING THE USE OF METALLIC FUEL IN NEW LWR DESIGNS

Background

The state-of-the-art for metallic uranium fuel elements has improved markedly in the last ten or fifteen years. The basic fuel is unraium metal containing very small quantities of a few alloying elements (maximum concentration less than 1,000 ppm). Since the density of the fuel is very high, there is significantly less parasitic neutron capture than in less dense fuels. The material is compatible with zirconium alloys, and its thermal properties are very good.

The first consideration in the development of a metallic uranium fuel element is the attainment of a relatively-long fuel lifetime. In the case of metal fuel, lifetime is limited by distortion (swelling) of the fuel, leading to changes in thermal and neutronic performance of fuel assemblies through geometry shifts. Ultimately, fuel swelling results in diametrical growth that causes a circumferential strain in excess of the failure limit of the cladding. The problem, then, is to develop a fuel design capable of obtaining high fuel exposures with minimum risk of failure.

To a first approximation, a metal-fuel assembly contains twice the mass of uranium that the same size ceramic fuel assembly does. Consequently, in the same reactor, and with a specific power of approximately one-half that of the oxide fuel, a metal-fuel assembly operating to the same burnup (in Mwd/mtU) will remain in the core twice as long. Conversely, for operation for a given number of days, the metal fuel burnup in Mwd/mtU will be about one-half that of the ceramic fuel.

Early work on uranium growth had attributed the swelling to cavities formed by the agglomeration of fission gases at unexpectedly-low temperatures. Work at Savannah River showed, however, that fission-gas bubbles contributed only a small fraction of the volume increase. These investigations led, in 1961, to the recognition of cavitational swelling in uranium. Subsequent investigation at

Hanford confirmed the Savannah River observations and revealed another mode of swelling at somewhat higher temperatures — small pores aligned in planar arrays.

Both before and after the determination of the presently-accepted reasons for gross unranium swelling, large amounts of work had been done on alloying and other techniques to control the swelling. Some success in reducing the innate swelling of pure uranium in the temperature range necessary for light water reactor use had been achieved by alloying with small amounts of other metals, as mentioned above. In addition, high-strength cladding and high system pressure were used as external restraints on the volumetric swelling of fuel rods. However, the improvements obtainable through the use of the alloying and restraint techniques were not sufficient to obtain the burnup necessary for economic use in light water reactors. Subsequently, work in the United States and in the Soviet Union was directed toward determining the beneficial effects of a central void to accommodate fuel swelling internally, without increasing the external dimensions of the element. It is the combination of alloying constituents,* external restraint, and a central, longitudinal hole in each fuel element that constitues the basis for the fuel design evaluated here and in a preceding study.[1]

Discussion

Metallic-uranium fuel elements have been used succesfully in water-cooled plutonium production reactors for decades, but the extremely short fuel exposures and low coolant temperatures employed in those reactors do not provide a technical foundation for the use of metallic fuel in commercial LWR power plants. The possible applications of metallic fuels in LWRs have been studied for two cases, with the studies separated in time by a period of around three years. The latest study is reported here.

*The alloying elements are assumed to have no effect on the neutronic behavior of the fuel (because of the properties of the alloying elements themselves and the very small quantities present).

The conclusions given in an earlier report[1] concerning substitution of metallic fuel elements for the ceramic elements in existing power reactors (retrofittable use) were based on reports of work performed in the late 1960s and early 1970s.* For the present study, which considered non-retrofittable PWR designs, it was first necessary to determine whether the information on which the earlier report[1] was based still appeared valid and whether significant work had been reported on metal fuels in the period since the earlier study.

There is a "new" body of information on the use of metallic fuels, albeit the interest of the investigators is in fast-breeder reactor applications. Historically, metallic fuel has been of interest for breeder reactors because it has a significantly lower doubling time than other fuel types. However, consideration of its use was discontinued years ago (late 1960s) in favor of the oxides because the high core coolant outlet temperatures then used in design studies raised questions of compatibility between metal fuel and cladding. One of the recent papers on the performance of metallic fuels[2] points out that interest has been renewed, and lists the following three reasons for this revival of the concept:

1. The required sodium outlet temperatures have dropped within the range where metallic fuels can perform reliably to a high burnup.

2. A desire for proliferation-resistant cores has developed, and metals have been shown, at least at low burnup, to be capable of remote hot reprocessing in a closed fuel cycle. At higher burnups, though, removal of fission products may present a problem.

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*Work performed in the United States at the Engineering Test Reactor and at Battelle Columbus Laboratories led to the selection, in the earlier SSA report, of a fuel rod with a 10% axial void. If 2% strain is allowed in Ziracloy cladding, a rod with a 10% axial void would be expected to survive exposures to a maximum of about 40,000 Mwd/mtU. This is consistent with work in the Soviet Union that shows fuel exposures of 15,000 to 20,000 Mwd/mtU for a 5% void, with the allowable fuel exposure increasing with increases in void volume.

3. The irradiation behavior of the driver fuel in EBR-II has demonstrated that a metallic fuel element is capable of high burnup.

The recent reports on metallic fuel[2-5] are based on work performed in support of a single facility — the Experimental Breeder Reactor II. It is this work on the existing, experimental reactor that may stimulate serious reconsideration of metallic fuel for use in commercial breeders. The EBR-II work was performed because an economic incentive remained to develop high burnup metal driver fuel for the reactor, which had a long-term mission as the primary U.S. fuels and materials irradiation test facility in support of the LMFBR technology base.

The driver fuel of EBR-II was successfully redesigned so that fuel-element lifetime is in excess of 10 atom % burnup for the current Mark II design.[3] Investigations in the 1967-1973 period led to the Mark II fuel, design and irradiation of which was initiated in 1973. The Mark II elements have been the reference driver fuel in EBR-II since that date and have been subjected to extensive testing and study,[2-4] as have the reprocessing techniques required.[5]

Although the entire thrust of the EBR-II fuel work is toward the LMFBR, the results of some of the investigations apply directly to other possible uses of metallic fuel. For example, the use of a void space within metallic fuel to accommodate radial growth[1] has been confirmed as a requirement for extended fuel-element lifetime. The EBR-II design provides this initial void in the form of a large gap between the fuel and the cladding, an acceptable arrangement for the fast reactor design and its sodium-"filled" fuel pins. Since sodium cannot be used to enhance the heat transfer across fuel-to-cladding gap in LWR fuel, because of the sodium-water reaction that would occur in the event of cladding failure, the provision of a large gap in an LWR fuel design would lead to unacceptable fuel temperatures. Consequently, the most important question regarding the use of void space within the active fuel lay in the location of that void. The present opinion is that a fuel smeared density of approximately 75% is necessary to achieve very high fuel exposures (typical of those in LMFBR driver elements), and that the reduction in density to this value can be accomplished by a central hole as well as by the radial fuel-to-cladding gap.[6] Thus, for the fuel exposures necessary for economic operation in LWRs, the provision of a 10% axial void by a central hole in the fuel rod does not appear inconsistent.

Conclusions

The recently-published information appears to confirm the assumptions made in the earlier study of the use of metallic fuel in LWRs. It seems likely that PWR fuel elements with low probabilities of failure could be designed. However, the potential for a uranium-water reaction in the event of cladding failure may exclude the use of fuel with only a low probability of failure, requiring instead a zero probability. Furthermore, an accident in which core cooling is degraded could result in gross cladding failures and the opportunity for substantial reaction between the uranium metal and water. The safety problems — including the generation of large amounts of hydrogen — could be severe. Determining just how reliable metallic fuel must be for use in LWRs, and its behavior under degraded-cooling conditions, would require substantial study, as a mimimum, and would almost surely require an expensive research and development program.

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