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ADVANCED PRESSURIZED WATER REACTOR FOR IMPROVED RESOURCE UTILIZATION

PART II - COMPOSITE ADVANCED PWR CONCEPT

Prepared For

U.S. ARMS CONTROL AND DISARMAMENT AGENCY

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1.0 INTRODUCTION

Work reported in this document was performed for the U.S. Arms Control and Disarmament Agency under BOA AC9NX707 (Task Order 80-03) as part of the Agency's program on improved fuel utilization in light water reactors. The objective of the study was to evaluate the enhanced resource utilization in an advanced pressurized water reactor concept using a composite of selected improvements identified in a companion study. Results of this study are also related to an earlier study on retrofittable improvements to current pressurized water reactors.

The companion study identified three areas where improvement in fuel utilization might be possible in an advanced PWR design without the constraint that improvements should be retrofittable into current reactors.

These areas are:

- Reduced loss of neutrons to control poisons;
- reduced loss of neutrons in leakage from the core; and
- improved blanket/reflector concepts.

The present study incorporates features relating to these three areas of improvement into a single composite advanced PWR and presents a preliminary assessment of resource requirements (U_3O_8), costs and impact on reactor safety.

2.0 SUMMARY

2.1 General Description

In the companion study,[1] the potential improvements identified for inclusion in a composite advanced PWR concept included the following:

- Mechanical spectral shift to control excess reactivity, consisting of movable rods of low neutron absorbing material which displace water and cause a change in the neutron spectrum;
- reduced core power density as a means of increasing core volume and reducing neutron leakage; and
- use of spent fuel assemblies, which would otherwise be discharged to storage, to serve as a neutron reflector around the periphery of the reactor core.

The advanced PWR concept is also intended to incorporate, to the extent possible, improvements considered in the retrofittable reactor evaluation study.[2] Of the retrofittable reactor improvements identified in reference 2, only one – increased moderation – proved to be incompatible with improvements evaluated in reference 1 for inclusion in the advanced PWR concept. Increased moderation (achieved in reference 2 by reduced fuel pin size) reduced the reactivity worth of the water-displacer rods in the mechanical spectral shift concept by an unacceptable amount. Consequently, increased moderation was limited and the advanced PWR uses fuel rods of the same dimensions as the reference design (RESAR) fuel. Overall water-to-fuel ratio in the advanced PWR (2.20 with displacer rods withdrawn) is nearly the same as that in the reference design (1.95).

To provide space for the water-displacer rods, the fuel rod pitch in a fuel assembly was reduced. Two alternative fuel assembly concepts are presented – one being a 17 x 17 array on a 0.477 inch lattice pitch and containing 64 Zr-clad BeO water-displacer rods. The second alternative is a hexagonal array with 217 lattice positions on a 0.50 inch triangular pitch and containing 54 displacer rods in each fuel assembly. The hexagonal fuel array is the preferred concept primarily because of the greater flexibility in core arrangement and because of the more uniform coolant flow area between fuel

rods. In either case, the fuel assemblies may be arranged in a core pattern (385 hexagonal assemblies or 277 rectangular assemblies) within a 17 foot inside diameter 10.5 inch thick pressure vessel. The larger pressure vessel required is estimated to increase the capital cost of the plant by $\$10 \times 10^6$.

The advanced PWR is designed to operate at ∿80% of the power density in the reference PWR. To accommodate the water-displacer rods, the coolant flow area associated with each fuel pin is reduced. In the advanced PWR, the reduction in this coolant flow area is very nearly equal to the reduction in power density and the core thermal-hydraulic will be very nearly that in the current PWRs.

The BeO water-displacer rods have a reactivity worth of ∿5.4%Δk which is adequate to provide the necessary reactivity change during a fuel cycle in the 10-region core. Auxiliary control (e.g., soluble poison) will be necessary for cold shutdown. Although other configurations are possible, the advanced PWR design suggested here uses bottom-entry hydraulic drive systems (similar to current BWRs) with all water-displacer rods in a single fuel assembly connected to an individual drive system. The large number of control rod drive systems required is estimated to increase the capital cost of the plant by $\sim\$7 \times 10^6$.

Decreasing the average core power density not only reduces the leakage loss of neutrons but also will allow the use of a higher radial power peaking factor without exceeding current peak performance limitations. This capability allows the use of low-leakage fuel management strategies to reduce neutron leakage even further. On the basis of several trial-and-error fuel assembly arrangements, it is estimated that radial leakage can be reduced to 1.5%Δk or less. At this low level of radial neutron leakage, the use of spent fuel assemblies as a blanket would have only a marginal effect. Furthermore, the number of fuel assemblies in a reload batch is too small to fill all core peripheral locations. Consequently, the advanced PWR concept presented here uses less than a complete blanket of spent fuel assemblies.

Dismantling and reassembly of spent fuel assemblies could be used to create a thinner blanket surrounding the core periphery. In view of the small radial neutron leakage, reassembling a spent fuel blanket or the use of alternative blanket materials would be a refinement that might be considered in a detailed fuel and control rod management study, with perhaps some small additional improvement.

2.2 Fuel Utilization and Economic Considerations

Although there is some uncertainty in the actual radial leakage that might be achieved in a detailed fuel and control management design study, it is estimated that in the advanced PWR a fuel discharge burnup of 50,000 Mwd/mtU can be achieved in a 10-region core with fuel of 2.7% enrichment.

With this enrichment and discharge burnup, the estimated 30-year resource requirement is 3309 standard tons U_3O_8, compared to 3978 ST U_3O_8 for the composite retrofittable concept or 5470 ST U_3O_8 for the reference design (reference 2). This would result in a net 39.5% improvement over the current reference design. Table 2-1 summarizes the performance and material balance for the advanced PWR and compares these values to the corresponding parameters in reference 2 for the composite retrofittable concept and for the reference design.

The reduced power density (and fuel specific power) in the advanced PWR concept results in increased interest charges during the reactor operation, offset by lower fuel and enrichment costs for the lower enrichment. Table 2-2 compares the levelized fuel cycle costs for the three concepts considered, assuming U_3O_8 ore costs of \$40/lb and \$100/lb, using the ground rules specified in reference 2.

The fuel cycle costs for the advanced PWR are estimated to be 0.55 mills/kwhr and 0.92 mills/kwhr lower than the retrofittable concept (at \$40/lb U_3O_8 and \$100/lb U_3O_8 respectively). Compared to the reference PWR, the corresponding reductions in fuel cycle costs for the advanced PWR are 2.21 mills/kwhr and 3.70 mills/kwhr. However, the higher capital cost due the larger pressure vessel and the increased number of control rod drive systems results in a differential increase in power cost of 0.55 mills/kwhr. Consequently, the net savings in power cost for the advanced PWR becomes 0 mills/kwhr or 0.37 mills/kwhr (\$40/lb U_3O_8 or \$100/lb U_3O_8) compared to the retrofittable concept and 1.66 mills/kwhr or 3.15 mills/kwhr compared to the reference design. At higher U_3O_8 costs, the savings in power cost would be even greater.

2.3 Safety Considerations

In assessing the effect of improvements included in the advanced PWR on reactor safety, no significant adverse effects were

Table 2-1 PERFORMANCE AND MATERIAL BALANCE

	Advanced PWR	Composite Retrofittable PWR	Reference
Discharged fuel burnup (equilibrium), Mwd/mtU	50,000	50,000	30,000
Refueling interval (@70% capacity factor), months	7.63	5.34	14.3
Number of fuel regions in core	10	10	3
Equilibrium cycle feed enrichment, wt%	2.7	3.33	3.1
Fuel specific power, Mw/mtU	31.9	44	36.1
Fuel residence time, years	6.13	4.45	3.58
Fuel Utilization, Mwd/St U_3O_8	7741	6276	4473
Initial loading			
U-235, kg	1521	1343	1970
Uranium, mt	95.3	69.1	84.2
U_3O_8, ST	339	320	470
Total 30 year requirement U_3O_8, ST	3309	3978	5470
30 year enrichment service, SWU (@ 0.2% tails)	1.80×10^6	2.46×10^6	3.31×10^6
30 year fissile Pu production, kg	3378	2464	4459
30 year total Pu production, kg	5391	4154	6215
Discharged fuel composition (equilibrium)*			
U-135, kg/mtU	2.25	2.46	7.93
Pu-239, kg/mtU	5.47	4.28	5.43
Pu-240, kg/mtU	3.01	2.66	2.17
Pu-241, kg mtU	1.78	1.39	1.36
Pu/242, kg/mtU	1.31	1.22	0.51
Total Pu, kg/mtU	11.57	9.54	9.47
Fissile Pu, kg/mtU	7.25	5.66	6.80
$U^{238}_{EOL}/U^{238}_{BOL}$	0.957	0.964	0.976
Equilibrium cycle annual requirements			
U-235, kg	423	518	729.5
U-238, mt	16.3	15.02	22.80
Total Uranium	16.7	15.53	23.53
Enrichment, @ 0.2% tails, SWU	5.68×10^4	7.83×10^4	1.065×10^5
Total U_3O_8, ST	100	124	173.6
Equilibrium cycle annual discharge			
U-235, kg	33.2	35.7	186
Uranium, mt	15.6	14.51	22.43
Discharge enrichment	0.23	0.25	0.79
Fissile Pu, kg	113	82.1	148.6
Total Pu, kg	180	138	207.2

*Calculated for equilibrium feed enrichment of 2.8 wt% U-235.

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Table 2-2 LEVELIZED FUEL CYCLE COSTS

Case	U_3O_8 Cost $/lb	Unit Costs, $/kg Fab	Unit Costs, $/kg Disp	Fuel Cycle Costs, mills/kwhr U_3O_8	Enrich*	Fab	Post Irrad	Incore Interest	Total
Current PWR	40	140	135	2.447	1.970	0.580	0.518	0.638	6.153
	100	140	135	6.118	1.970	0.580	0.518	1.191	10.376
Retrofittable PWR	40	140	135	1.774	1.446	0.383	0.342	0.578	4.492
	100	140	135	4.360	1.446	0.383	0.342	1.065	7.595
Advanced PWR	40	140	135	1.503	1.059	0.383	0.342	0.653	3.940
	100	140	135	3.675	1.059	0.383	0.342	1.213	6.672

* at 0.2% tails

identified. Relatively minor impact on safety is anticipated as a result of lower average fuel temperatures. The only adverse effect identified was the increased Zr inventory in the core, which could increase the problem of combustible gas control under postulated degraded core cooling conditions.

3.0 ADVANCED PWR DESIGN AND EVALUATION

The advanced PWR conceptual design presented here is based upon current practice (RESAR-414) as the reference design and incorporates retrofittable improvements described in a prior study (Ref. 2). With the principal objective being to assess the potential improvement in resource requirements, a detailed design and fuel management evaluation is neither necessary nor possible within the scope of work. In incorporating the further improvements identified in Ref. 1, an intentional effort was made to minimize deviations from current practice (except to the extent necessary) and to avoid design features which might adversely affect reactor safety to any great extent. The principal design features of the advanced PWR, as well as the preliminary assessment of performance, are presented in the following sections, incorporating mechanical spectral-shift control and reduced average core power density.

3.1 Mechanical Design

3.1.1 Design Considerations

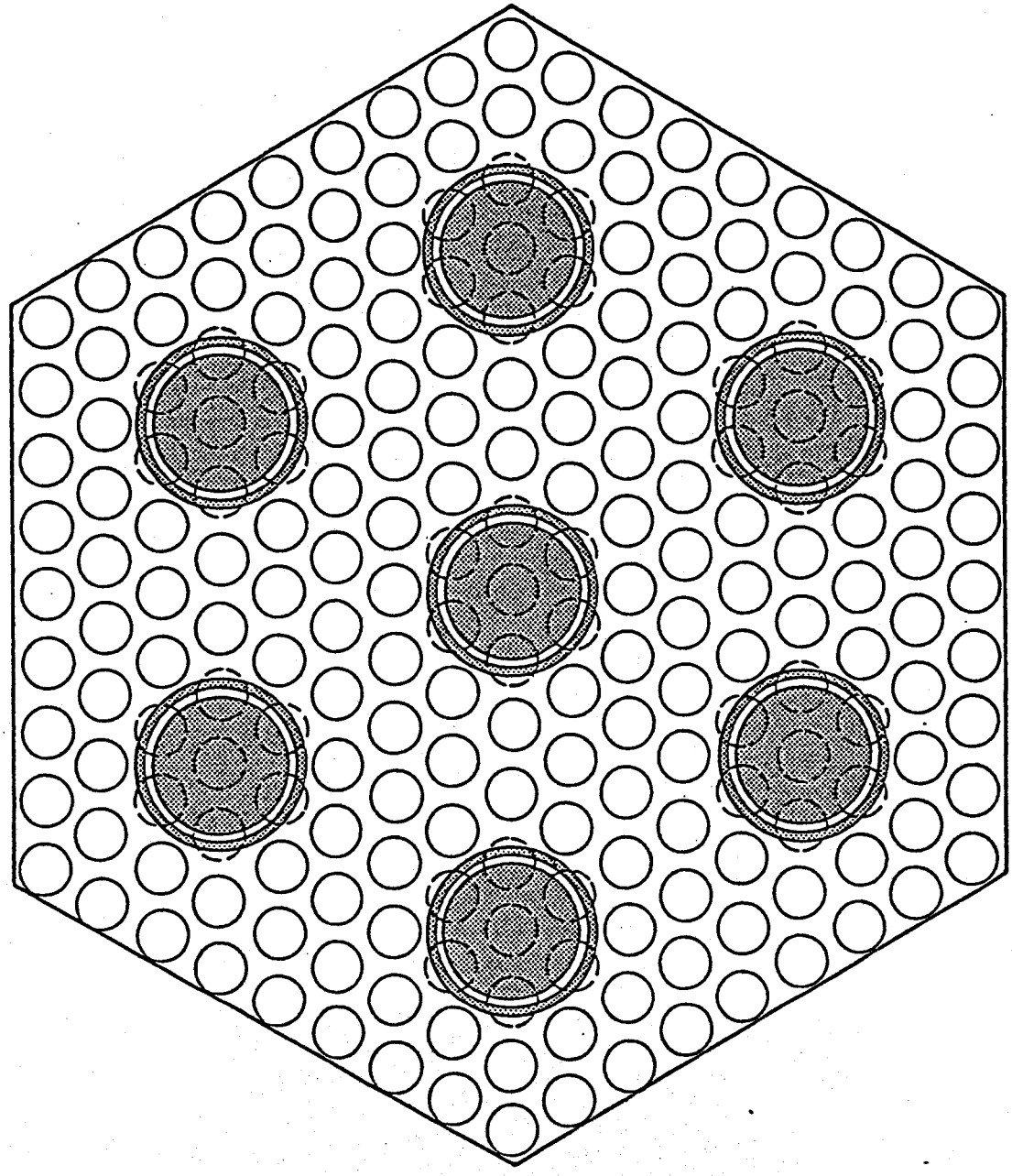
In the present study, mechanical design considerations were limited to those features which differ significantly from current pressurized reactor designs. The major design features affected by the advanced reactor concept are the fuel assembly configuration, core cross-section, pressure vessel and control drive mechanisms. Each of these was investigated only to the extent of developing reasonable confidence of their technical feasibility and to assess the probable economic impact.

3.1.2 Fuel Element Concepts

Within the framework of the mechanical spectral-shift concept, a relatively large number of rods of inert material (e.g., BeO) used to displace water-moderator from the core. Rod insertion decreases the effective water-to-fuel volume ratio and results in a reduction in reactivity. Two configurations of water-displacer rods were considered (large diameter or small diameter), either of which could be used in a triangular or square array. The major advantage of the large-diameter displacers is the less complex rod configuration required. Figures 3-1 and 3-2 illustrate possible fuel assembly configurations using large-diameter water-displacer rods. However, because of the large water holes left when the displacer rods are withdrawn, local power peaking within an assembly is substantially larger than for the small-diameter water-displacer rods (see Section 3.2.3).

With small-diameter displacer rods, the control assembly is more complex due to the larger number of "finger" rods required. Figures 3-3 and 3-4 illustrate possible fuel assembly configurations with the small-diameter water-displacer rods. In current pressurized water reactors (Westinghouse concept) control assemblies with 20 fingers are used, in contrast to 54 "finger" rods in the advanced PWR concept (64 in the rectangular array). The additional complexity of manipulating 54 (or 64) finger rods in a single fuel assembly is more difficult than with 20 "fingers," but is believed to be feasible, particularly when coupled with the hydraulic drive system discussed in Section 3.1.3 below.

Because of the smaller inter-assembly power peaking afforded by the small-diameter displacer rods, this concept is preferred to

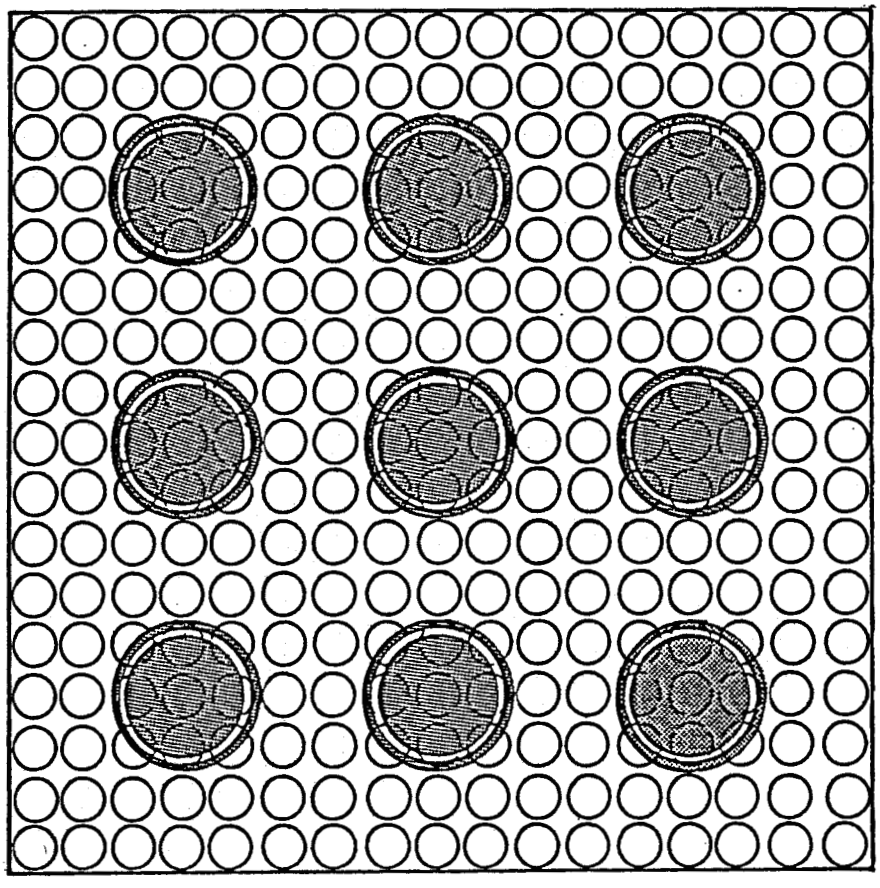


217 TOTAL POSITIONS

49 replaced by water displacers

168 fuel rods

Fig. 3-1 Horizontal cross-section through fuel assembly containing large water displacers.

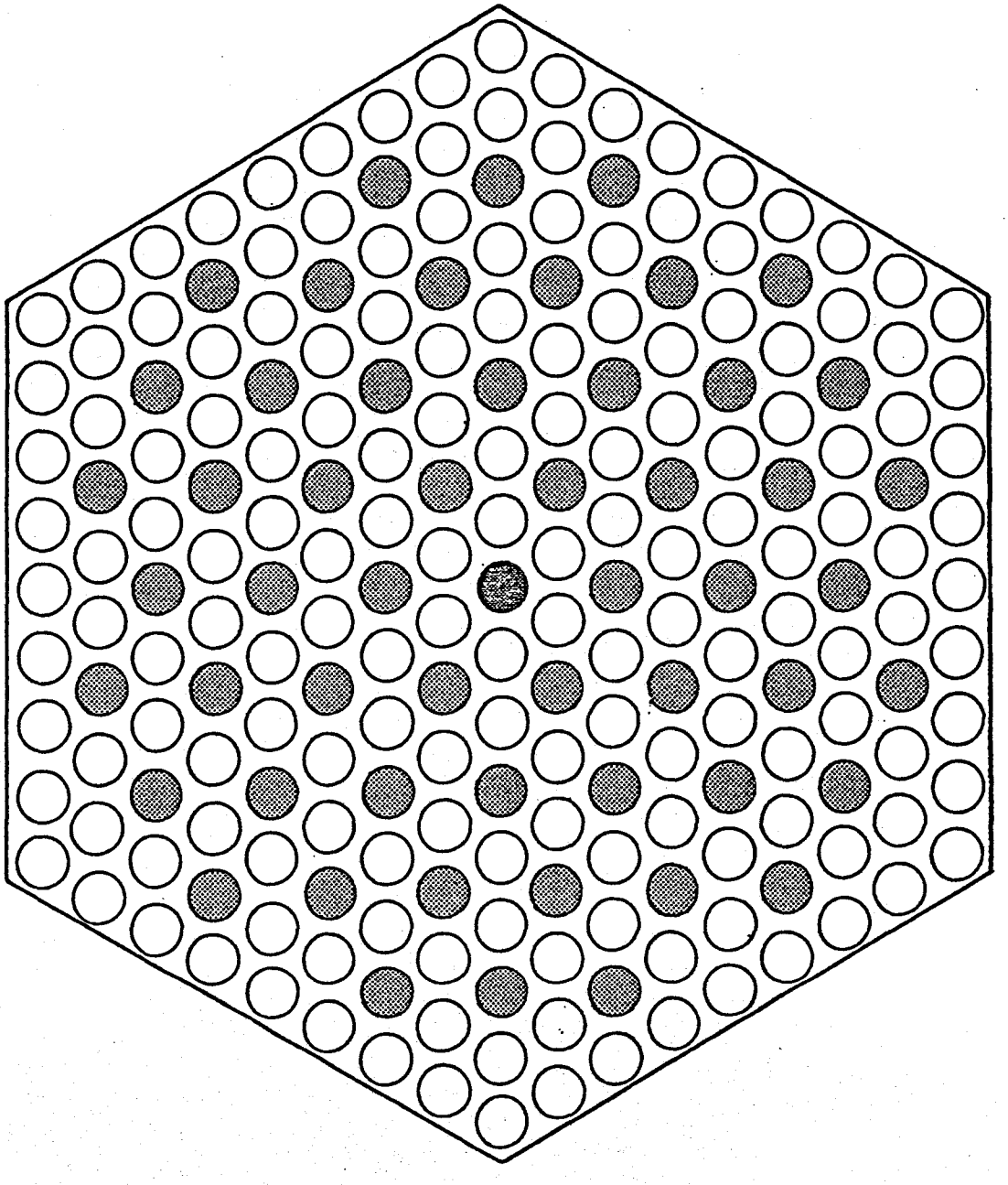


289 TOTAL POSITIONS

9 large BeO rods

208 fuel rods

Fig. 3-2 Horizontal cross-section through square fuel assembly with water-displacer control rods.



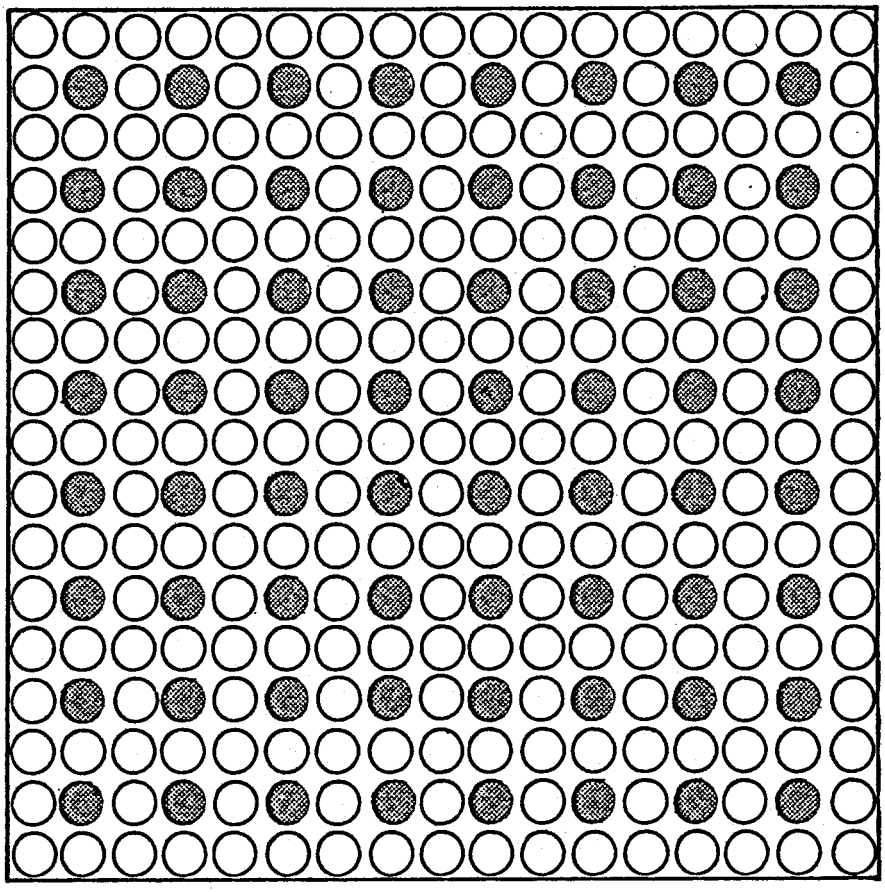
217 TOTAL POSITIONS

54 replaced by water displacers

1 central instrumentation sheath

162 fuel rods

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



289 TOTAL POSITIONS

64 BeO rods

225 fuel rods

Fig. 3-4 Horizontal cross-section through square fuel assembly with small water-displacer control rods.

the larger-diameter rods. However, with the reduced power density core in the advanced PWR concept, higher power peaking could probably be accommodated without exceeding the maximum heat rating of the highest-power pin in current PWR designs. Either the triangular lattice arrangement (Fig. 3-3) or the square lattice array (Fig. 3-4) appears mechanically feasible, although the triangular array is preferred because of the larger separation between fuel pins. Although detailed thermal-hydraulic analyses have not been performed, it seems intuitive that the larger pin-to-pin spacing in the triangular array should result in better performance and lower sensitivity to fuel manufacturing tolerances. Detailed design/thermal-hydraulic analyses (which are beyond the scope of the present study) will be necessary to determine the optimum configuration. Nevertheless, for the purpose of the present study, any of the four possible configurations appears to be mechanically feasible.

Figure 3-5 illustrates the cross-section of a small-diameter water-displacer rod, including the guide tubes which also serve to prevent coolant bypass. The clearance between the Zr-clad water displacer and the inside diameter of the guide tube is approximately the same as that in current PWR designs.

3.1.3 Control Drive Mechanisms

Because control elements are present in every fuel assembly in the advanced PWR concept, the control drive mechanisms required will differ significantly from those in current PWRs. Coupling all water-displacer rods in each assembly to a single drive mechanism would result in a significantly larger number of drive mechanisms at a closer spacing than for current PWRs. However, the number of drive assemblies and spacing would be comparable to the arrangement currently used in boiling water reactors, i.e., hydraulically driven bottom-entry rod drives, as illustrated in Fig. 3-6. During reactor operation, the water-displacer rods would be withdrawn downward to add reactivity and scram upwards under hydraulic pressure to shut down the reactor.

With the large number of water-displacer rods in the advanced PWR concept, alternative rod drive mechanisms could be developed to provide a finer level of reactivity control (and perhaps more control of the power distribution) than is possible with all rods in a given assembly connected to each drive mechanism. For example, small hydraulic lines connected to clusters of water-displacer rods could be used to eject selected clusters of rods. The number of water-displacer rods coupled in each cluster depends upon a compromise between the desired fineness of reactivity control and the complexity and cost of multiple hydraulic lines (and associated ex-reactor valving). Development of a

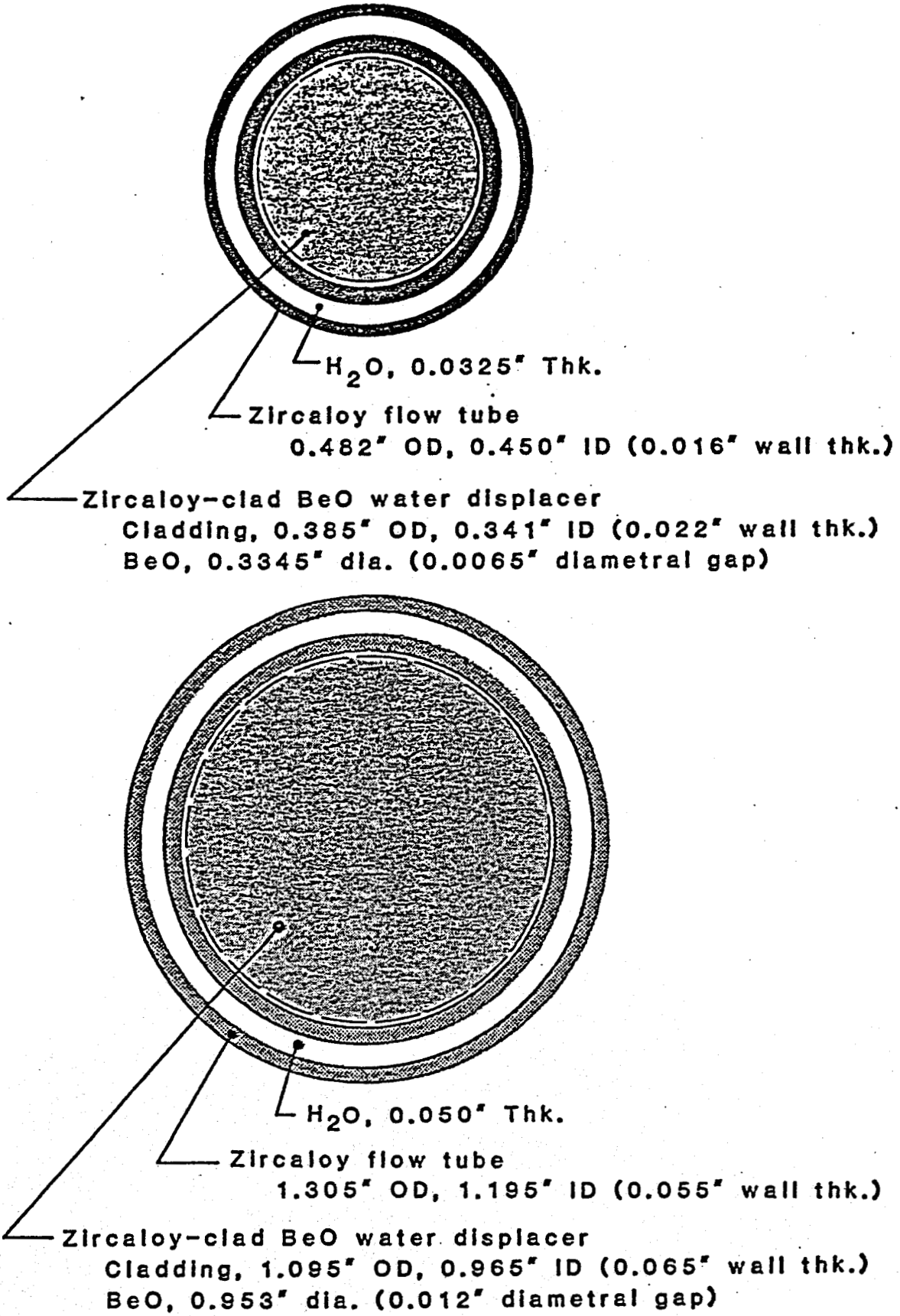


Fig. 3-5 Horizontal cross-section through small and large BeO water displacer rods.

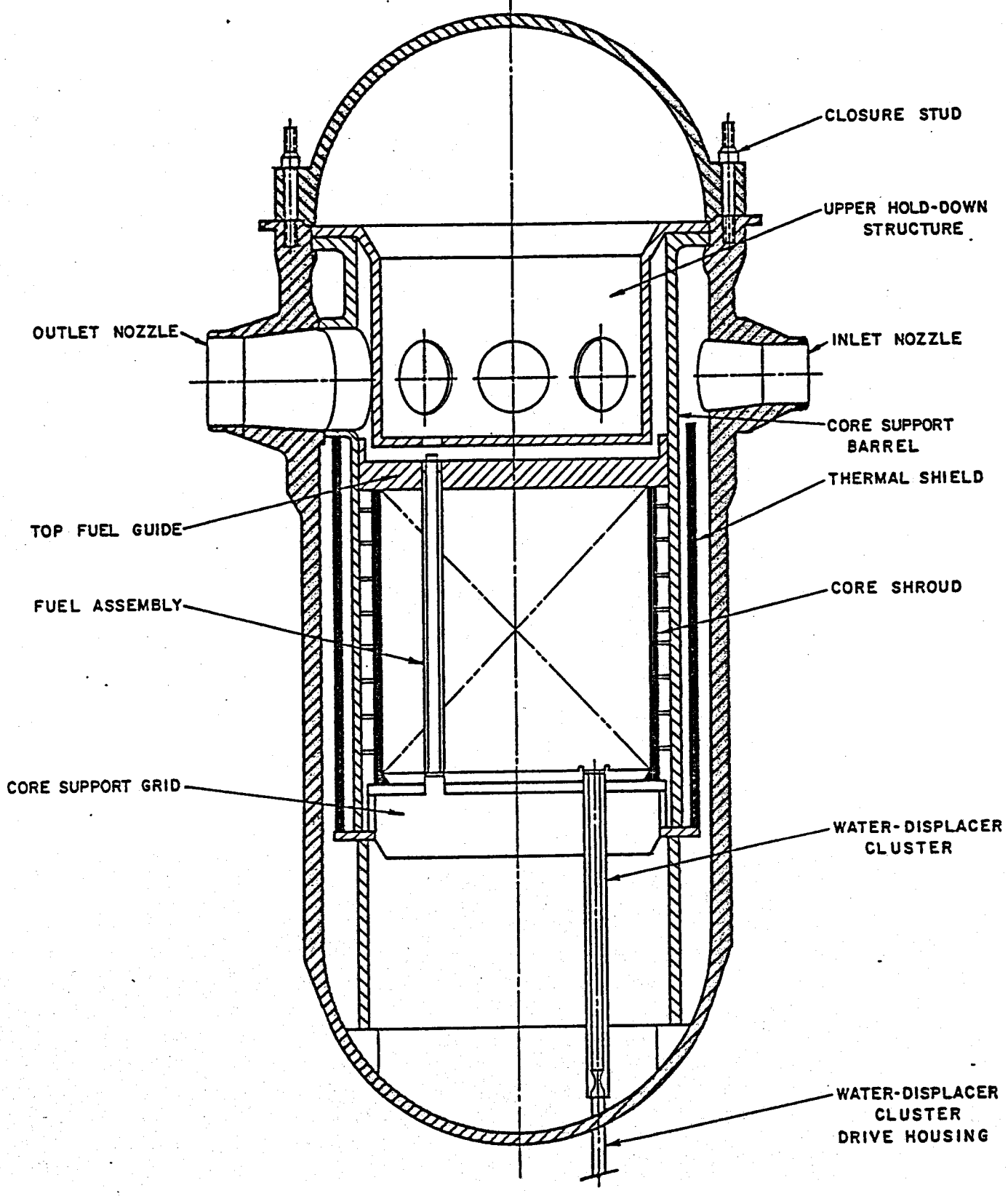


Fig. 3-6 Reactor vessel vertical section.

conceptual design for a fine-level reactivity control system is beyond the scope of the present study, other than to indicate that such a drive system appears technically feasible. A similar system—called the Hy-ball system—was investigated experimentally and analytically by the General Nuclear Engineering Corporation in the early 1960s, although the concept has not been applied in commercial reactor systems.

3.1.4 Core and Reactor Vessel Cross-Section

Both the lower power density core and the use of spent fuel assemblies as a reflector necessitate an increase in pressure vessel diameter above that in current use for large PWRs. Figures 3-7 and 3-8 illustrate the core arrangement for the hexagonal fuel assemblies (Fig. 3-7) or square assemblies (Fig. 3-8) selected as representative of the advanced PWR concept.

With a larger core, the pressure vessel inside diameter is also larger. For approximately the same downcomer flow area as in current PWRs, the required inside diameter is 17 feet. Using simple pipe stress equations and ratioing wall thickness from the 8.75-inch wall thickness at 14 feet I.D. for current designs, the minimum required wall thickness at 17 feet I.D. is 10.5 inches. This compares well with the reactor pressure vessel used for the German Biblis A reactor of 10-inch thickness at 16.4 feet inside diameter. Manufacture of the Biblis A pressure vessel also confirms the technical feasibility of manufacturing the size pressure vessel required for the advanced PWR.

3.1.5 Thermal-Hydraulic Considerations

Thermal-hydraulic analyses have not been performed for the advanced PWR concept. However, assuming the same average coolant exit temperature and the same core coolant temperature rise as in current PWRs, the thermal-hydraulic performance of the advanced PWR is not expected to be greatly different. Reducing the core power density would tend to lead to a proportional decrease in coolant flow rate (and associated pressure drop). However, the coolant flow area associated with each fuel pin in the active core region is also reduced (to accommodate the water-displacer rods). In the advanced PWR concept adopted here, the reduction in coolant flow area associated with each fuel pin is very nearly equal to the reduction in core power density. Consequently, to

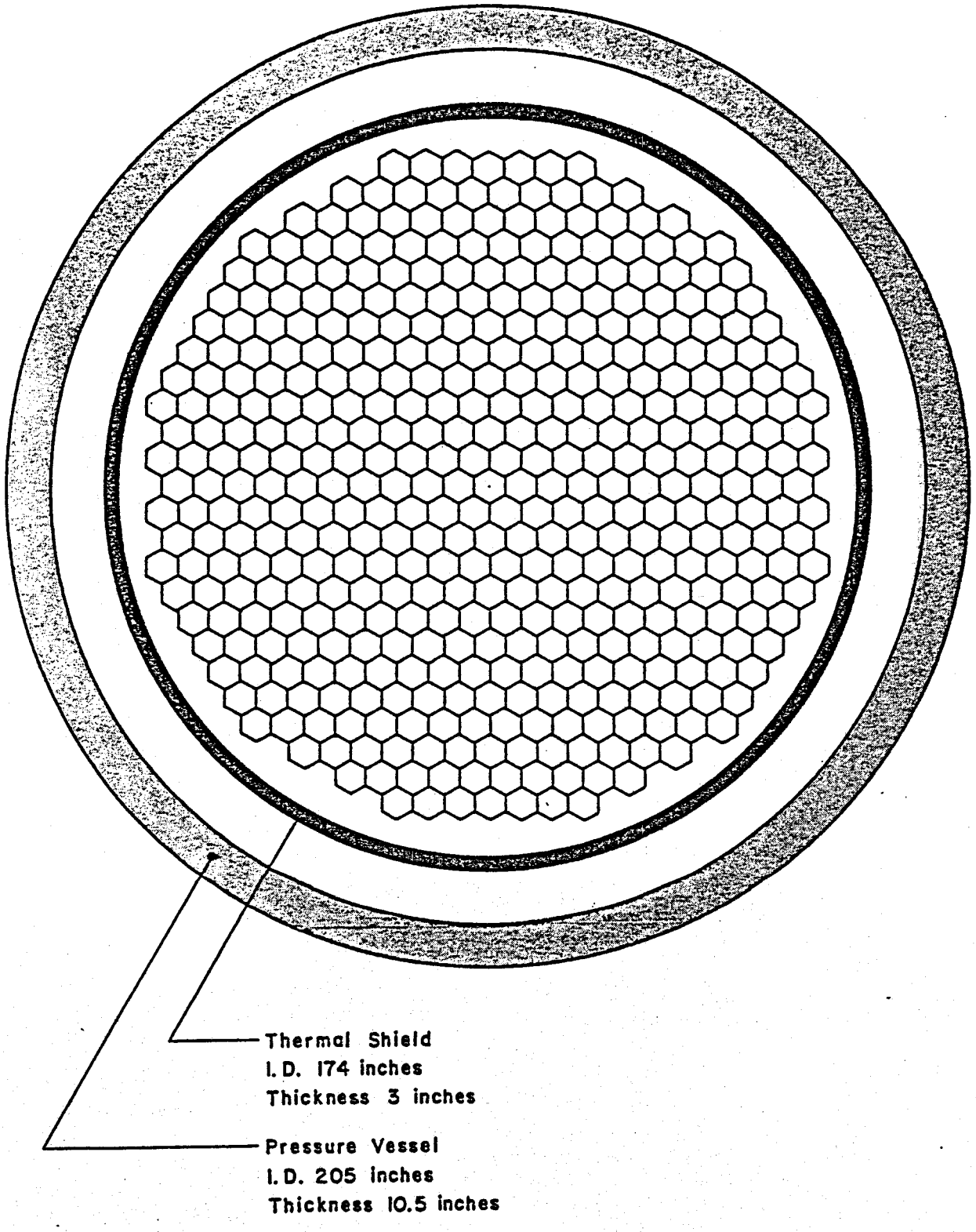


Fig. 3-7 Reactor core and pressure vessel cross-section (hexagonal fuel assemblies).

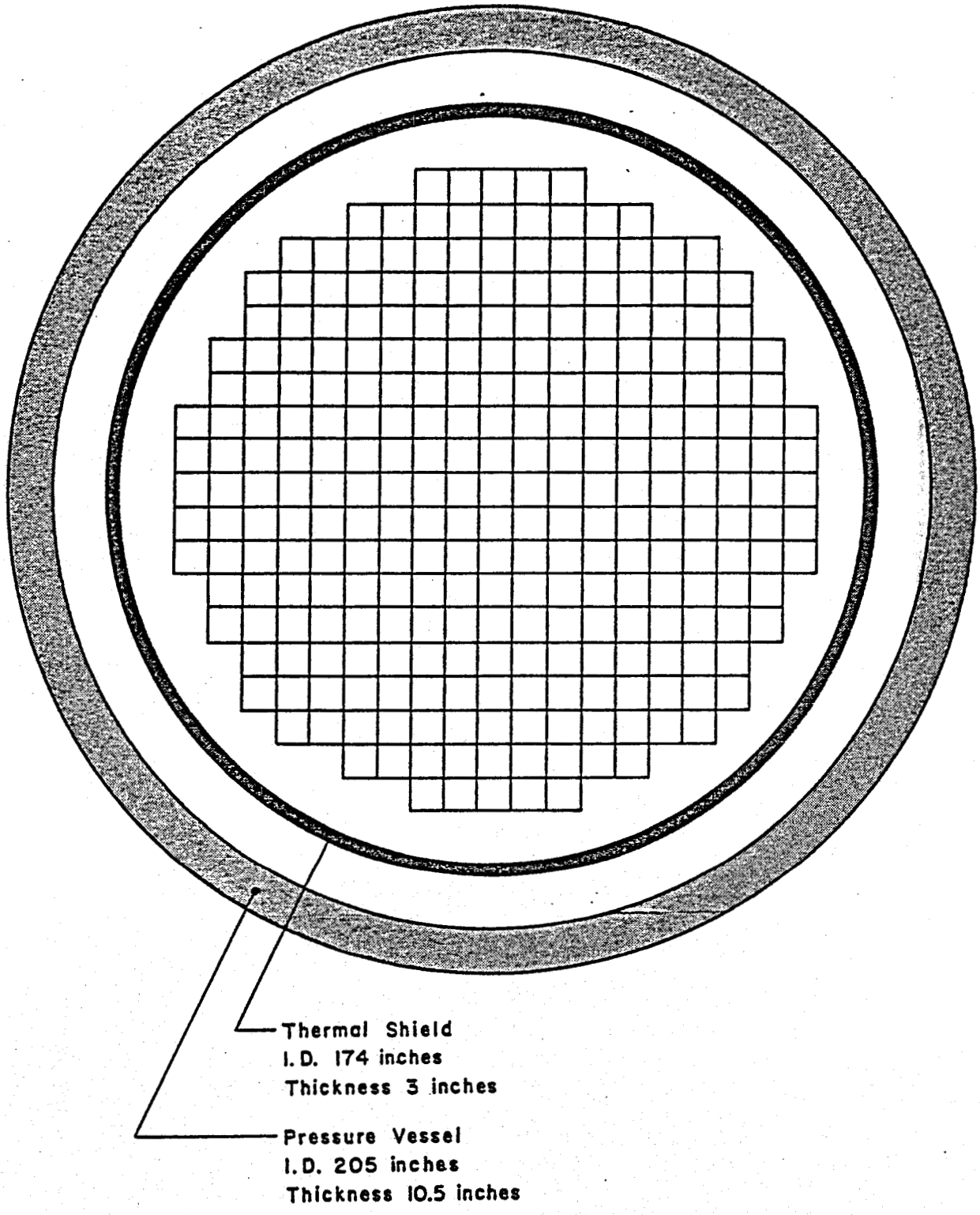


Fig. 3-8 Reactor core and pressure vessel cross-section (square fuel assemblies).

a first approximation, the core thermal-hydraulic performance is not significantly different from that in current PWRs.

3.2 Nuclear Analysis

3.2.1 Nuclear Design Considerations

3.2.1.1 General

The advanced PWR is a composite of those compatible improvements identified in the earlier evaluation[1] and in a previous study.[2] Retrofittable improvements[2] were evaluated in the previous study, assuming, as a design base, the discharge of fuel after 50,000 Mwd/mtU burnup. The previous study also assumed a 10-region core, operating on approximately a 6-month refueling interval.

For the present study, the retrofittable core design developed in the previous study was taken as the starting point. However, since the smaller fuel pins used in the retrofittable concept are not compatible with the mechanical spectral-shift concept, the fuel pin dimensions for the reference design were assumed. Table 3-1 summarizes the reference design fuel characteristics and the control rod dimensions used in the advanced PWR evaluation.

Improvements[1] to the retrofittable concept considered as part of the advanced PWR concept included (1) the use of water-displacer rods for reactivity control, (2) reduced neutron leakage by decreasing core power density, and (3) use of spent fuel as a blanket. Incorporation of these improvements became the purpose of the present study.

3.2.1.2 Water-Displacer Rods

The purpose of the water-displacer rods is to provide a mechanism for mechanically adjusting the effective core water-to-fuel ratio and hence to control reactivity. Also termed "Mechanical Spectral-Shift," the water-displacer rod concept must provide appropriate reactivity control over an operating fuel cycle, with minimum loss of neutrons to parasitic absorption.

Insertion of the water-displacer rods hardens the neutron spectrum and increases the absorption of neutrons in U-238. In the mechanical spectral-shift concept, the excess neutrons are thereby absorbed in U-238, producing plutonium, which, in turn, increases reactivity later in the fuel cycle.

Table 3-1 DESIGN CHARACTERISTICS OF ADVANCED PWR

Reactor Core	
Type of Fuel	Zr clad UO_2
Core Diameter (active)	153 in.
Core Height (active)	168 in. 14'
Enriched fuel	156 in. 13'
Nat. UO_2 fuel	12 in.
Core Power Density	81 Kw/l
Fuel Specific Power	31.9 Mw/mtU
Number of Fuel Assemblies	385 hexagonal or 277 square
Average Linear Heat Rating	4.6 Kw/ft
Fuel Enrichment (equilibrium)	2.8 Wt% U-235
Design Fuel Burnup (equilibrium)	50,000 Mwd/mtU
Average Coolant Temperature	598°F
Average Fuel Temperature	1326°F
Average Clad Temperature	618°F
Fuel Cell	
Rod O.D.	0.374 in.
Fuel Pellet O.D.	0.3225 in.
Gap Thickness	0.0065 in. (diametral)
Clad Thickness	0.0225 in.
Clad Material	Zr-4
UO_2 Density	95% T.D.

Table 3-1 DESIGN CHARACTERISTICS OF ADVANCED PWR (cont'd)

	Small Control Rod	Large Control Rod
Hexagonal Fuel Assembly		
Number of Fuel Rods	162	168
Number of Guide Thimbles	54	7
Instrument Thimbles	1	-
Fuel Rod Pitch (triangular)	0.5 in.	0.5 in.
Water Displacer Rod Pitch	0.866 in.	2.5 in.
Assembly Dimensions across flat	7.361 in.	7.361 in.
Assembly Pitch	7.94 in.	8.15 in.
Rectangular Fuel Assembly		
Number of Fuel Rods	225	208
Number of Guide Thimbles	64	9
Instrument Thimbles	1	-
Fuel Rod Pitch (square)	0.477 in.	0.465 in.
Water Displacer Rod Pitch	0.954 in.	2.325 in.
Assembly Dimensions (17 x 17 array)	8.11 in.	7.91 in.
Assembly Pitch	8.15 in.	7.94 in.
Water Displacer Rods		
Material	BeO	BeO
Clad and Guide Material	Zr	Zr
Clad Dimensions		
OD	0.385 in.	1.095 in.
ID	0.341 in.	0.965 in.
BeO Pellet OD	0.3345 in.	0.953 in.
Guide Tube		
OD	0.482 in.	1.305 in.
ID	0.450 in.	1.195 in.

In any reasonable mechanical spectral-shift concept, the increment in reactivity control is limited. However, the 10-region core concept reduces the incremental reactivity change in a fuel cycle and hence is compatible with the mechanical spectral-shift concept. In fact, the successful application of the spectral-shift concept is dependent upon the use of a multi-zone core arrangement.

The material in the water-displacer rods should have as small a neutron absorption cross-section as possible, while still satisfying the requirement for adequate reactivity control. BeO is the preferred material because of its low absorption cross-section. Small amounts of higher-absorbing material could be added to the BeO to increase the effectiveness of the rods when inserted, although such additives will reduce the improvement in resource utilization that could otherwise be achieved.

It has also been found that the reactivity control increment of the water-displacer rods is reduced as the design water-to-fuel ratio in the core is increased. Thus, the mechanical spectral-shift concept is not compatible with one of the improvements considered in the retrofittable improvement study[2]—increased core moderation (by changes in lattice spacing, fuel rod diameter or annular pellets). For this reason (and because spectral-shift control is expected to result in significantly greater improvement than increased moderation), the design water-to-fuel ratio in the advanced PWR was taken as approximately the same as in current PWR designs.

As in the case of current PWRs, auxiliary reactivity control will be required for shutdown to the cold condition. Detailed evaluation of the reactivity control increment may indicate a requirement for auxiliary control early in a fuel cycle for operating shutdown margin, but such a requirement, if any, has not been assessed in this study.

3.2.1.3 Reduced Power Density

The principal objective of reducing the core average power density is to reduce the loss of neutrons through leakage. In the earlier evaluation, the optimum power density appeared to be about 60% of that in current PWRs, i.e., nearly twice the core cross-sectional area. However, reducing the power density to 60% is expected to result in an excessive economic penalty. Futhermore, since higher peaking in the power distribution can be tolerated in a core of lower power density, it is not necessary to reduce

the power density to 60% to accomplish a significant reduction in neutron leakage. Thermal-hydraulic considerations suggest reducing the power density by an amount comparable to the reduction in coolant flow area around the fuel pins. Thus, the design basis is a power density of approximately 80% of that in current PWRs, which will maintain comparable thermal-hydraulic performance. At this power density, the advanced PWR can accommodate a 25% increase in power peaking factor without exceeding current peak power limitations.

3.2.1.4 Spent Fuel Radial Blanket Elements

The original intent of this concept was to locate spent fuel assemblies around the periphery of the core both to serve as an efficient blanket and to capture neutrons leaking from the core in a productive manner. Earlier evaluations[1] had suggested that spent fuel would be the most effective blanket. Not long after the effort to develop a reasonable core layout had begun and the initial nuclear analysis results became available, two important factors were soon recognized.

- First, the peripheral region of the core was of a size that the number of fuel assemblies required substantially exceeded the number of fuel assemblies in a single discharge batch.

- Secondly, in the nuclear analysis, the spent fuel assemblies designated as the blanket necessarily became part of the fuel management considerations.

To provide a complete blanket of spent fuel assemblies around the entire periphery of the core would require several discharge batches and would effectively introduce an 11th, 12th and part of a 13th fuel management region into what is intended to be a 10-region core. Alternatively, spent fuel could be partitioned into sub-assemblies of one-third to one-fourth of the normal fuel assembly size and located in the periphery to surround the core with a thinner region of spent fuel. This latter alternative was considered beyond the scope of the present study, but could perhaps provide a further refinement in a more detailed design study.

In the advanced PWR, the reduced average core power density allows the possibility of reducing the power generation in the

peripheral fuel assemblies (and accepting more power peaking within the inner fuel assemblies). This in turn reduces the fraction of neutrons leaking from the core and minimizes the contribution of the radial blanket. In developing a representative core layout for the advanced PWR, spent fuel assemblies (i.e., assemblies of 50,000 Mwd/mtU burnup) are used as necessary to obtain a symmetric core arrangement. Credit will be taken for the contribution of these assemblies to reactivity and to total power generation, but not for the additional burnup accumulated when the assemblies are considered blanket. elements.

3.2.2 Methodology

Nuclear analyses for investigation of the advanced PWR design relied primarily on the use of point-depletion calculations and multi-dimensional diffusion theory calculations. The point-depletion calculations were performed utilizing the NULIF code.[3] This code has previously shown excellent correlation with results of a large number of critical experiments and with experimental fuel burnup data in LWRs. This same code has been used in a number of investigations similar to this study. For multi-dimensional diffusion theory calculations the PDQ-7 Code[4] was utilized. This code is well established as a nuclear analysis tool and can utilize input cross-sections obtained from NULIF calculations. To examine flux shape effects at the core ends related to use of an axial blanket, separate analysis using ANSIN,[5] a one-dimensional, discrete-ordinates transport theory code was performed.

A sequence of calculations utilized to investigate the fuel and core performance of the advanced PWR design consisted of both NULIF and PDQ-7 calculations to examine reactivity and power distribution throughout the desired reactor operation. Since a large number of iterative spatial-depletion calculations would be required to examine alternatives within the allowable fuel cycle, depletion calculations and spatial power distribution calculations were generally performed independently to investigate a large number of cases. Attractive configurations were then further examined, for instance, to investigate axial and radial blankets.

Because of the difficulty in modeling the hexagonal core arrangement, the calculations were performed for the square array. Performance of the hexagonal array is not expected to be greatly different, except in details of the fuel management and inter-assembly power distribution.

3.2.3 Fuel Assembly Analysis

3.2.3.1 Cell Reactivity Calculations

Cell calculations concentrated on the determination of the basic fuel cell reactivity as a function of burnup. As described earlier, the basic fuel cell consists of a fuel rod with surrounding water and a beryllium oxide (BeO) water-displacer rod. Mechanical spectral-shift, with BeO displacing about 25% of the core water volume, appears to be capable of controlling about 5.4% Δk rods-in to rods-out. This 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 material because of its low parasitic absorption cross-section, but the absorption could be increased in a detailed design study, if necessary, to provide additional reactivity control during the fuel burnup cycle.

To investigate the reactivity influence of the BeO water-displacer rods, two different cells were examined with different BeO rod sizes. The two different sizes were examined to illustrate the difference between two types of fuel assemblies—one using a "small" BeO rod, the other using a "large" BeO rod. The two different assemblies are illustrated in Figs. 3-2 and 3-4, and fuel assembly specifications are provided in Table 3-1. As described, the difference in the two designs results in a relatively small difference in water-to-fuel variations, but, as will be shown later, power distribution effects are markedly different.

Reactivity profiles for the fuel cells with BeO displacer rods are illustrated in Fig. 3-9. This profile depicts the reactivity effect of moving the BeO rods from the "in" position to the "out" position uniformly with burnup over each 5000 Mwd/mtU cycle of a 10-cycle, 50,000 Mwd/mtU operating cycle. As illustrated by this figure, the reactivity profile can be maintained relatively flat, and even increase, with fuel exposure. This trend should obviously yield increased resource utilization and provides the basic initial performance data from which an optimal advanced PWR design can be developed.

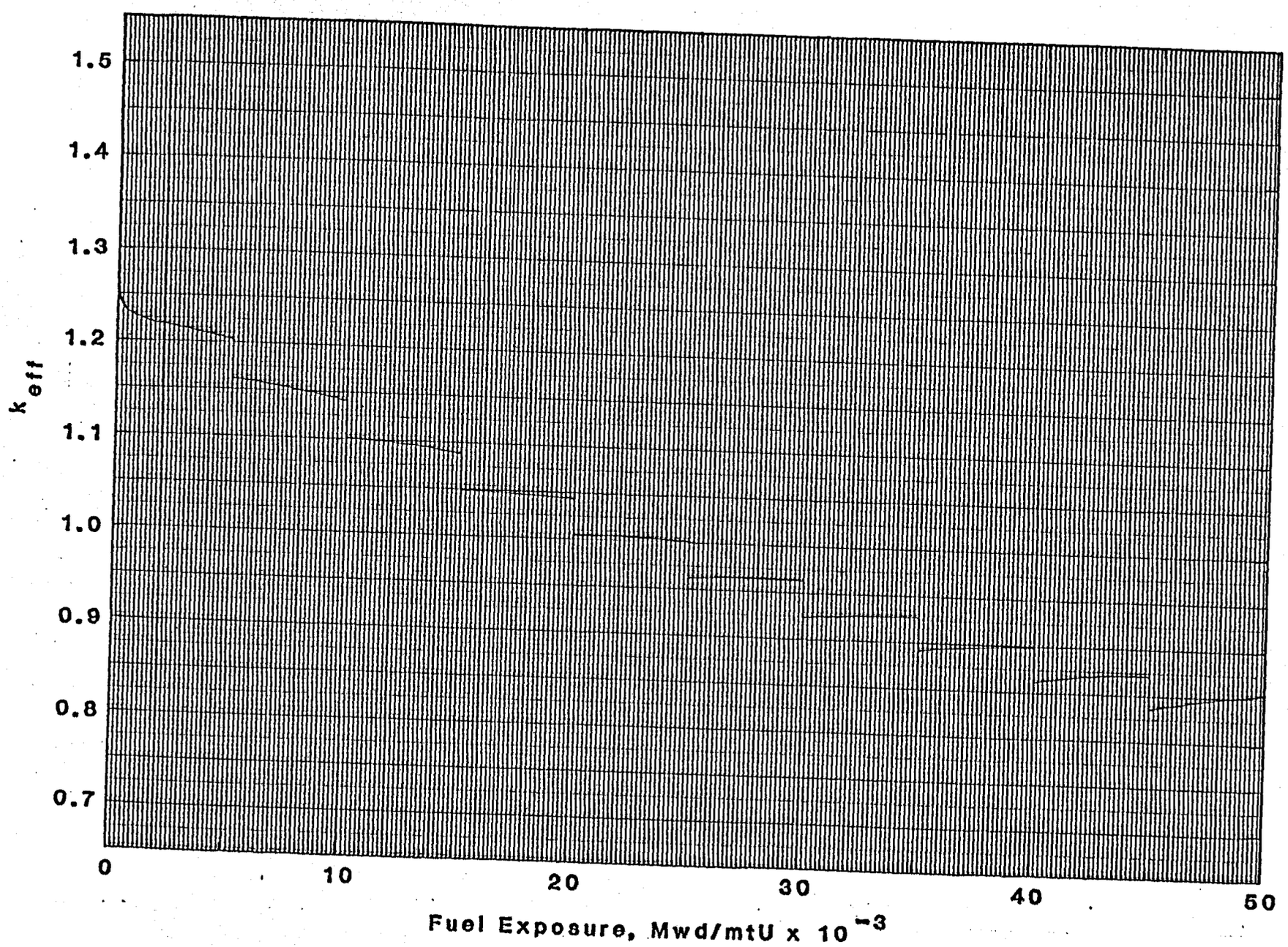


Fig. 3-9 Reactivity dependence on fuel burnup with movable BeO water-displacer rods.

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3.2.3.2 Intra-Assembly Power Distribution

To further investigate the performance of fuel assemblies with BeO water-displacer rods, the power distribution across the fuel assembly was investigated. Figures 3-10 and 3-11 illustrate the power distribution across fuel assemblies with large and small BeO rods, respectively, when the rods are withdrawn. With the BeO rods inserted, the power distribution reveals no significant power peaks. With the rods removed, however, significant peaking near the center of the assembly does become apparent for the assembly with the larger rods.

Different BeO rod sizes, or variation in the rod size across the assembly, would likely reduce peaking effects for relatively large rods. For the purpose of this study, however, the small BeO rods with a 0.482-inch Zircaloy flow tube (which is the same size as flow tubes described in RESAR-414) were selected as the basic element of the mechanical spectral-shift control system for the advanced PWR design.

3.2.3.3 Fuel Assembly Performance and Isotopics

As previously illustrated in Fig. 3-9, the reactivity profile with burnup for the small BeO rods provides ample reactivity over the core life. For illustrative purposes, a uniform average k_{∞} over ten cycles yields 1.021 for beginning-of-equilibrium-cycle and 1.016 for end-of-cycle for the initial enrichment examined. This margin appears adequate depending upon results of detailed fuel management and control management studies.

The use of movable BeO rods to provide "mechanical spectral-shift" results in a continuously changing neutron spectrum and variation in excess reactivity. Early in a fuel cycle, when excess reactivity is potentially present, the BeO rods would be inserted, displacing water, and causing the excess neutrons to be absorbed in the U-238 resonances, providing 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 provided earlier in the cycle. Table 3-2 summarizes the calculated isotopics in the advanced PWR fuel over the anticipated fuel burnup, corresponding to the reactivity history with movable BeO rods shown in Fig. 3-9.

Water		1.1335	1.1268	Water			1.0243	0.9385
		1.1084	1.1019				1.0054	0.9315
		1.0606	1.0544	1.0884	1.0963	1.0484	0.9688	0.9187
			1.0484	1.0824	1.0904	1.0430	0.9642	0.9145
				Water			0.9906	0.9184
							0.9970	0.9148
							0.9617	0.8946
							0.9024	0.8658
								0.8460

Fig. 3-10 Assembly radial power distribution with water-displacer rods withdrawn - Large Control Rods.

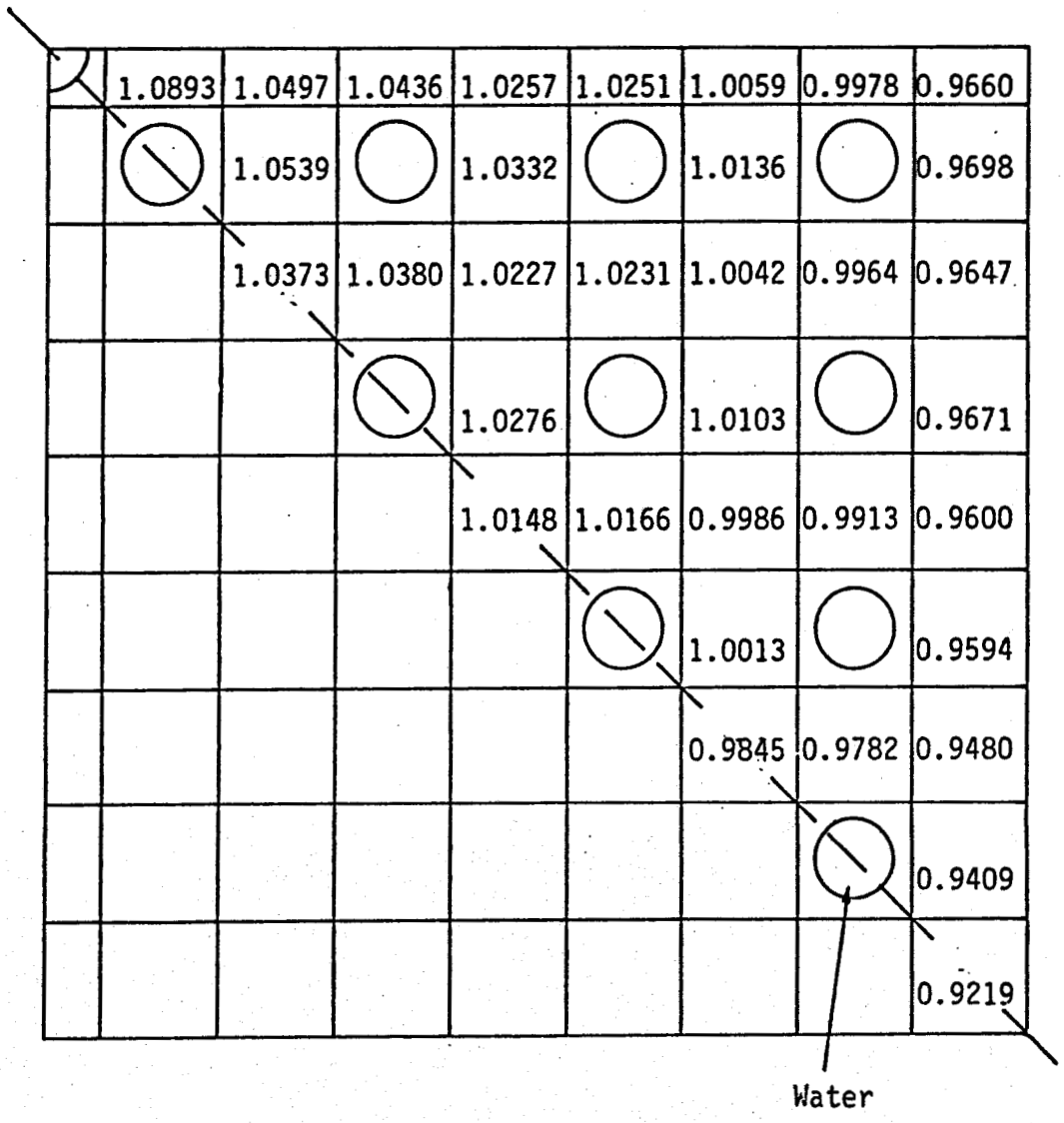

Fig. 3-11. Assembly radial power distribution with water-displacer rods withdrawn - Small Control Rods.

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Table 3-2 ISOTOPIC COMPOSITION OF ADVANCED PWR FUEL

Fuel Burnup, Mwd/mtU	% U-238 Remaining	U-235*	Pu-239	Pu-240	Pu-241	Pu-242	Fissile Pu	Total Pu
0	100	28.00	0	0	0	0	0	0
5,000	99.7	22.84	2.136	0.220	0.057	0.002	2.195	2.416
10,000	99.3	18.53	3.411	0.606	0.267	0.024	3.678	4.310
15,000	98.9	14.90	4.204	1.020	0.545	0.083	4.750	5.852
20,000	98.6	11.84	4.701	1.422	0.826	0.181	5.527	7.129
25,000	98.1	9.288	5.010	1.791	1.078	0.315	6.088	8.194
30,000	97.7	7.184	5.199	2.121	1.289	0.480	6.490	9.090
35,000	97.2	5.481	5.315	2.405	1.460	0.669	6.775	9.849
40,000	96.7	4.120	5.387	2.647	1.595	0.875	6.982	10.504
45,000	96.2	3.059	5.435	2.848	1.699	1.091	7.134	11.073
50,000	95.7	2.252	5.474	3.012	1.779	1.308	7.253	11.573

*For 2.8 wt% U-235 initial enrichment.

3.2.4 Axial Blanket Considerations

Among the additional techniques of improving overall neutron economy and resource utilization is the use of axial blankets. Axial blankets of natural or depleted uranium have been shown[6] to have the potential for improving uranium utilization by decreasing the core leakage, by increasing fertile conversion (in the blanket material) and by increasing the discharge exposure of the enriched fuel region.

From a uranium utilization standpoint, it apparently makes little difference whether the blanket is composed of natural or depleted uranium; however, from the standpoint of separative work and fuel cycle costs, natural uranium is the preferred blanket material.[6, 7] Therefore, investigations into varying lengths of axial blankets containing natural uranium were performed. Three different blanket lengths were examined in which the top and bottom 6, 12, or 18 inches of fuel in a nominal length assembly were replaced by natural uranium.

As was discussed in Ref. 7, the selection of an axial blanket length is influenced by a number of factors. The key factor utilized for selection here is the axial peaking. As shown in Fig. 3-12, beginning-of-life axial peaking profiles show significantly lower power in the end regions with an axial blanket compared to a fully fueled axial length. This reduces axial neutron leakage, but shifts power generation more toward the core interior. The larger the blanket, the greater the power peaking increase, for a given overall fuel height.

The blanket region is also a low reactivity region and changes there yield only small changes in reactivity. For beginning-of-life conditions with BeO rods fully inserted, the effect of blanket size above 6 inches on reactivity is not significant. Beyond a thickness of ∿6 inches, there is little effect on reactivity, although the peak power in the active enriched fuel increases if the active height is reduced. Therefore, from a power-peaking and reactivity standpoint, a relatively short blanket appears appropriate. A 6-inch blanket is therefore utilized as part of the composite advanced PWR.

Based upon a 6-inch axial blanket of natural uranium, one-dimensional calculations indicate an effective axial buckling of 5.65×10^{-5} for the enriched active fuel region (∿0.35% Δk) at beginning of life. This value, coupled with an estimate of the effective radial buckling, was subsequently used in estimating the required equilibrium feed enrichment.

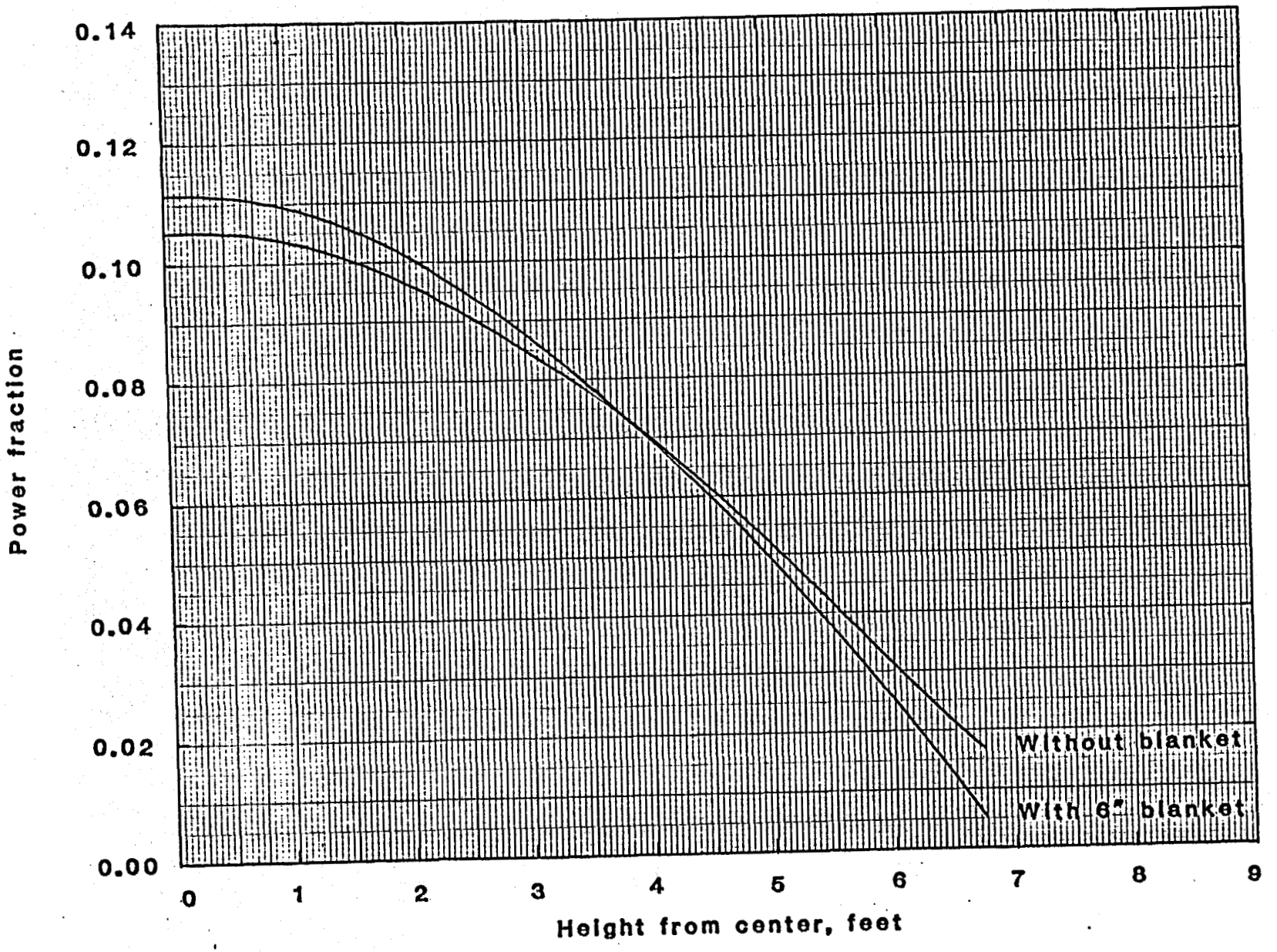

Fig. 3-12 Axial power distribution with and without natural UO_2 blanket.

3.2.5 Radial Blanket Effect and Power Distribution

In conventional PWR fuel management strategy, fresh fuel is introduced in the peripheral region of the core to achieve a comparatively uniform power distribution across the core. In recent years, low-leakage fuel management strategies have been investigated (8-11) as a means of reducing (retrofittable) the loss of neutrons by leakage. Low-leakage fuel management strategies envision dispersing fresh fuel internal to the core and locating partially-burned fuel on the periphery. The net effect of this strategy is to reduce the relative power generation rate (and hence neutrons available for leakage) in the peripheral region with a consequent increase in power generation rate (and power peaking) internal to the core. In some cases, burnable poison has been suggested to minimize the increase in power peaking due to low-leakage fuel management strategies.

In the advanced PWR considered here, the reduced core average power density provides an additional margin to accommodate increased power peaking. In addition, spent fuel assemblies may be loaded on the periphery to form a type of blanket, using fuel which would otherwise have been discharged to the spent fuel storage pool.

Investigation of the power distribution and appropriate arrangements of fresh and partially-burned fuel assemblies throughout the core is essentially a trial-and-error process, complicated by varying burnup histories accumulated in each assembly which, in turn, depends upon its location in the core, the prior operating history and the associated control rod movement strategy.

The development, characterization and optimization of a detailed fuel management scheme (including control) is clearly beyond the scope of the present study. Consequently, the effort was limited to evaluating a few core arrangements which could be considered representative and to estimating the reactivity loss in leakage as a basis for subsequent estimation of an enrichment associated with the discharge burnup of 50,000 Mwd/MtU.

After several attempts to develop a reasonable core pattern, it was realized that the outer periphery of the core contained more locations than could be filled by a single refueling batch (without sub-dividing the fuel). Furthermore, blanket spent fuel assemblies were neutronically indistinguishable from an additional burnup region and a "spent fuel blanket" reduced, analytically, to only one option of a low leakage fuel management strategy.

Core power distributions were examined using a core model consisting of 277 assembly locations on a square array, as illustrated in Fig. 3-8. This array corresponds to 10 regions of 26 fuel assemblies each and 19 "spent fuel" assemblies, where a "spent fuel" assembly was defined as fuel having reached the design discharge burnup of 50,000 Mwd/mtU. Figures 3-13, 3-14 and 3-15 illustrate radial power distributions, calculated in 2-dimensions for three core arrangements, assuming equal distribution in burnup among the fuel assemblies.

Examination of these power distributions reveals that those with higher power peaking result in the lower loss of neutrons through leakage. For the case shown in Fig. 3-13, the leakage loss corresponds to only 0.4% Δk, but results in a relatively large power peaking. Cases 2 and 3 (Figs. 3-14 and 3-15 respectively) illustrate configurations with lower power peaking and with leakage losses corresponding to 1.3% Δk and 1.1% Δk respectively. These latter cases represent reasonably feasible core configurations.

A detailed fuel and control rod management analysis would include individual burnup histories and control element operation of each core region. In the hexagonal core arrangement (Fig. 3-7), somewhat greater flexibility is possible because of the larger number of fuel assemblies and the finer sub-division of control (water-displacer) elements. Considering the high margin in power peaking available with the low power density core, it seems reasonable and conservative to allow $\sim$1.5% Δk for neutron leakage loss, corresponding to a radial buckling of 2.4×10^{-4}. Combined with the axial buckling calculated previously with a 6 inch axial blanket of natural UO_2, the estimated total buckling of $\sim 3 \times 10^{-4}$ serves as a basis for estimating the required enrichment and discharge fuel composition at the design burnup of 50,000 Mwd/mtU. Minor changes in fuel composition of those assemblies constituting the "radial blanket" were not considered.

3.2.6 Enrichment and Fuel Composition

For the estimated total buckling, point depletion calculations indicate that an enrichment of 2.8% U-235 is required to achieve a discharge fuel burnup of 50,000 Mwd/mtU, with uniform reactivity weighting over a 10 region core. At this discharge burnup (neglecting composition changes in radial blanket elements), the composition is shown in Table 3-2.

4.306	0 3.191	2 1.832	6 2.415	0 1.691	2 0.892	6 1.076	0 0.739	2 0.469	1 0.112	9
2.884	4 2.746	3 1.774	5 1.609	4 1.433	3 0.846	5 0.688	4 0.577	3 0.346	3 0.089	10
2.147	8 2.965	1 1.465	7 1.143	8 1.505	1 0.627	8 0.407	9 0.462	2 N/A	5 0.066	9
3.662	0 2.784	2 1.542	6 1.972	0 1.421	2 0.643	7 0.680	0 0.359	3 0.108	10	
2.310	4 2.192	3 1.407	5 1.285	4 1.182	3 0.665	6 0.418	5 0.227	4 0.071	9	
1.544	8 2.135	1 1.054	7 0.834	8 1.226	1 0.942	0 0.289	7 0.099	10		
2.376	0 1.798	2 0.976	6 1.226	0 0.911	2 0.430	6 0.159	8 0.564	9		
1.352	4 1.217	3 0.745	5 0.616	4 0.466	3 0.178	9 0.069	10			
0.964	1 0.773	2 0.443	4 0.213	9 0.129	10					
0.226	10 0.201	9 0.132	10							

— Cycle number

— Relative power density

Fig. 3-13 Relative radial power distribution - configuration No. 1.

0.061	8	6 0.088	8 0.151	7 0.369	6 0.864	4 1.658	1 2.368	0 2.481	2 1.495	10 0.374
0.077	7	8 0.092	7 0.173	6 0.397	3 1.038	0 2.044	5 1.612	4 1.578	1 1.368	9 0.358
0.128	5	7 0.154	6 0.237	8 0.368	7 0.707	3 1.402	4 1.454	5 1.162	2 0.916	8 0.289
0.294	7	6 0.318	7 0.359	3 0.365	6 0.815	5 1.157	0 1.843	1 1.346	9 0.432	
0.744	6	8 0.654	6 0.664	7 0.757	0 1.519	3 1.309	4 1.174	2 0.902	10 0.261	
1.786	3	0 2.030	5 1.200	3 1.260	4 1.310	5 1.110	1 1.092	9 0.383		
2.385	2	4 2.027	3 1.698	5 1.329	0 1.757	1 1.258	2 0.722	10 0.208		
2.067	5	0 2.744	4 1.657	1 1.496	2 1.083	10 0.433	9 0.235			
1.806	2	1 1.872	2 1.270	8 0.578	9 0.334					
0.502	9	10 0.447	9 0.364							

— Cycle number

— Relative power density

Fig. 3-14 Relative radial power distribution - configuration No. 2.

0.413	8 0.549	6 1.029	0 0.753	6 0.785	8 1.403	4 1.648	1 1.142	4 0.973	2 0.276	10
0.551	6 0.588	7 0.628	8 0.915	3 0.896	7 1.797	0 1.040	8 1.021	5 1.004	1 0.281	9
0.701	8 1.229	0 0.872	6 0.890	7 1.094	5 1.263	6 1.384	3 1.092	4 0.811	2 0.255	8
1.026	3 0.930	6 1.305	3 1.767	0 1.174	7 2.049	0 1.346	5 1.280	1 0.428	9	
0.976	5 0.948	7 1.287	4 1.226	7 1.273	6 1.615	3 1.317	4 0.976	2 0.285	10	
1.016	7 1.659	0 1.265	6 2.073	0 1.619	3 1.311	5 1.346	1 0.465	9		
1.250	2 0.910	8 1.331	3 1.329	5 1.308	4 1.338	1 0.879	2 0.263	10		
0.967	4 0.901	5 1.026	4 1.251	1 0.964	2 0.437	10 0.275	9			
0.854	2 0.894	1 0.751	2 0.441	8 0.299	9					
0.256	9 0.236	10 0.219	9							

— Cycle number

— Relative power density

Fig. 3-15 Relative radial power distribution - configuration No. 3.

Isotopic changes resulting from additional burnup accumulated in the "blanket" spent fuel assemblies was generally neglected because of the low power density in these assemblies and the uncertainty in actual power distribution on the core periphery. An estimate of the consequences of this additional burnup indicates the increase in fissile plutonium content is less than 1% for an additional burnup of 1300 Mwd/mtU. The small effect on isotopics is expected since, at the nominal 50,000 Mwd/mtU burnup, the plutonium content has very nearly reached a saturation value.

3.3 Economic Assessment and Fuel Utilization

3.3.1 Capital Cost Increases

The principal capital cost factors associated with the advanced PWR are the larger reactor pressure vessel and the greater number of control rod drive mechanisms. Based upon material and fabrication requirements, the incremental cost of the larger pressure vessel is estimated to be $\$10 \times 10^6$ above the corresponding cost of pressure vessels currently in use in the U.S.

Modern boiling water reactors use bottom-entry hydraulically operated control rod drive mechanisms of a type comparable to the drive mechanism suggested for the advanced PWR. Using cost data from the CONCEPT [12] computer code for estimating capital costs, the incremental cost of the control drive mechanisms in the advanced PWR is estimated to be $\$7 \times 10^6$. This estimate is believed to be conservatively high, since the drive mechanisms in the advanced PWR are expected to be simpler in design (i.e., fully in or out rather than positioned in a step manner), than now required for large BWR's.

The combined incremental cost of the advanced PWR is therefore estimated to be $\$17 \times 10^6$. Assuming an effective 20% fixed charge rate, the higher capital cost of the advanced PWR will contribute 0.55 mills/kwhr to the cost of the electricity produced.

3.3.2 Fuel Cycle Costs

The estimate of fuel cycle costs was based on the economic ground rules established in the previous study.[2] In the

advanced PWR, the higher interest charges due to reduced power density are offset by the lower enrichment required, leading to a net fuel cycle cost lower than either the reference case (current PWR) or the comparable retrofittable case identified in reference 2. Table 3-3 summarizes the fuel cycle costs for the three cases (on the equilibrium cycle). The net fuel cycle costs for the advanced PWR are approximately 0.55 mills/kwhr lower than the retrofittable concept at $40 per lb U_3O_8 or 0.92 mills/kwhr lower at $100 per lb U_3O_8. Including the contribution from the increased capital cost results in a net savings in power costs of 0 mills/kwhr and 0.37 mills/kwhr respectively for the two cases.

3.3.3 Fuel Utilization

Based on the advanced PWR concept presented here and the method of analysis used previously,[2] the 30-year resource requirement for U_3O_8 is estimated to be 3439 standard tons with fuel of 2.8% enrichment, compared to 3978 ST U_3O_8 for the composite retrofittable concept and 5470 ST U_3O_8 for the reference design (current PWR). This represents a 13.5% improvement over the retrofittable concept and a 37% improvement over the reference PWR. Because of uncertainty in actual power distribution, fuel and control management philosophy and radial leakage achievable in a detailed design, the potential benefit of power coastdown is not included in the resource requirement listed above. Power coastdown could potentially reduce the enrichment required to 2.7% and the resource requirements of the advanced PWR to 3309 ST U_3O_8 or a net improvement over the current PWR designs of 39.5%

3.4 Safety Considerations

None of the improvements considered in the advanced PWR should seriously affect reactor safety adversely. Reduced power density results in a proportionally lower fuel average specific power which in turn would reduce the average fuel temperature. Because of higher power peaking, maximum fuel temperatures, however, would likely not be appreciably less than for current PWRs, at least in the hottest fuel assembly. Nevertheless, the lower average fuel temperature should result in lower average release of volatile fission products to the fuel plenum as well as less stored energy in the fuel in the event of a loss-of-coolant accident. Both of these effects should qualitatively result

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Table 3-3 LEVELIZED FUEL CYCLE COSTS

Case	U_3O_8 Cost $/lb	Unit Costs, $/kg Fab	Unit Costs, $/kg Disp	Fuel Cycle Costs, mills/kwhr U_3O_8	Enrich*	Fab	Post Irrad	Incore Interest	Total
Current PWR	40	140	135	2.447	1.970	0.580	0.518	0.638	6.153
	100	140	135	6.118	1.970	0.580	0.518	1.191	10.376
Retrofittable PWR	40	140	135	1.774	1.446	0.383	0.342	0.578	4.492
	100	140	135	4.360	1.446	0.383	0.342	1.065	7.595
Advanced PWR	40	140	135	1.503	1.059	0.383	0.342	0.653	3.940
	100	140	135	3.675	1.059	0.383	0.342	1.213	6.672

* at 0.2% tails

in a small improvement in reactor response to postulated accident conditions, although detailed accident analyses would be required to confirm the anticipated effects and to quantitatively define the magnitude.

Conversely, reduced fuel specific power increases the core inventory of zirconium clad material available for chemical reactions producing hydrogen under postulated degraded core cooling conditions. The greater quantity of hydrogen that could potentially be produced under these conditions would increase the difficulty of post-accident mitigation of hydrogen burning.

Similarly, the reduced core power density results in a larger inventory of coolant in the primary system and increases the blowdown energy released to the containment in the event of a LOCA. However, the increase in primary system coolant inventory is not directly proportional to the reduction in power density because a significant portion of the coolant is contained in the pressurizer, heat exchangers and piping which are not affected. Consequently, it is believed that modifications to the containment will not be required, although detailed evaluation will be necessary to confirm this conclusion.

With the spectral shift control concept in the advanced PWR, soluble poison would not normally be used except for augmented reactivity control during cold shutdown. Consequently, the moderator temperature coefficient of reactivity will be more negative than in current PWRs using soluble poison control. Although the average fuel temperatures will be lower (and less reactivity lost to the Doppler effect), the Doppler coefficient of reactivity should be approximately the same as in current PWRs. Consequently, the overall power coefficient of reactivity should be more negative which should enhance the reactor response to power transients. Reactivity shutdown margin and reserve reactivity control for load following and Xe override are additional factors remaining to be considered in developing a detailed reactor design.

In the advanced PWR concept, there is one feature that has the potential of significantly improving reactor safety under loss-of-coolant accident conditions. Examination of the fuel assembly configurations described in Figs. 3-3 and 3-4 reveals that almost every fuel rod will "see" one or more water-displacer rod assemblies. This feature affords the possibility of developing an internal core spray system to provide emergency cooling directly to the fuel pins under major accident conditions. If the water-displacer rod guide tubes incorporated small appropriately located perforations and were connected to an emergency coolant supply, coolant could be injected through the perforations spraying directly on the fuel pins. This could be an efficient interim method of cooling the core under LOCA conditions pending ECCS operation to re-flood the core.

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