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RETROFITTABLE MODIFICATIONS TO
PRESSURIZED WATER REACTORS FOR
IMPROVED RESOURCE UTILIZATION

Prepared For

U.S. ARMS CONTROL AND DISARMAMENT AGENCY

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1.0 INTRODUCTION

This report summarizes work performed for the U.S. Arms Control and Disarmament Agency under BOA AC9NX707 (Task Order 80-02), as part of the Agency's continuing program on improved fuel utilization in light water reactors. The objective of the study was to investigate improvements in fuel management and design of pressurized water reactors (PWRs) that could potentially increase the utilization of natural uranium resources in a once-through fuel cycle (i.e., without using spent fuel reprocessing and recycle). For the present study, potential improvements were limited to retrofittable concepts, i.e., those which could be introduced in the near-term future without requiring significant modifications to the reactor system or balance of plant. The potential improvements considered were not necessarily restricted to those which might be economical under current uranium ore prices or to those which might be acceptable to the nuclear industry at the present time. A six-month fuel cycle, for example, although technically possible, would be neither economical nor acceptable to the industry at the present time.

Although all potential improvements are not necessarily compatible with each other, the target objective was to seek a composite system of compatible improvements that, if possible, could increase uranium resource utilization by 30% or more. Economic factors, risks involved in the introduction, and potential licensing concerns are also addressed in the report.

2.0 SUMMARY AND COMPOSITE IMPROVED REACTOR CONCEPT

2.1 General

Resource utilization can generally be improved by any method that will increase the available reactivity and allow an increase in discharge fuel burnup without excessive penalty, due to higher enrichment. The relationship between U_3O_8 utilization, fuel burnup (B), and the enrichment (E) necessary to supply the required reactivity can be described in two ways, (1) as follows:

(1) fuel utilization,

$$Mwd/ST_{U_3O_8} = 0.76926 B \left(\frac{0.711 - E_T}{E - E_T} \right),$$

where E_T is the tails enrichment in the diffusion cascade; and

(2) resource utilization, the 30-year cumulative requirement for U_3O_8 for a 1000 Mw(e) plant,

$$ST_{U_3O_8}^{30 \text{ yr}} \approx \frac{6.3098 \times 10^5}{B} \left(\frac{E - E_T}{0.711 - E_T} \right) \left(48 + \frac{N - 1}{N} Y \right),$$

where Y is the equilibrium fuel lifetime in years and N is the number of fuel regions in the core. The constant in the latter relationship is appropriate for an average capacity factor of 70%, a net thermal efficiency of 32.9%, and a discharge enrichment approximately 25% of the initial value. These relationships clearly show that increasing the burnup to enrichment ratio will result in improved resource utilization.

2.2 Improved Reactor Concepts

From among a number of potential improvements, five were considered retrofittable in reactors now operating or under construction. These include the following:

- Increased core regionalization and fuel burnup,
- fuel design modification to increase the water-to-fuel ratio,

- low-leakage fuel management to reduce radial leakage,
- use of natural uranium axial blankets, and
- end-of-cycle power coastdown.

A number of other potential improvements were considered, but they were rejected as being either non-retrofitable, ineffective incompatible, or uneconomic. Table 2.1 summarizes the improvement in resource utilization that can be accomplished through the five methods evaluated.

Of the five improvements considered, the greatest gain is accomplished by increased core regionalization and design fuel burnup. The reference improved cycle utilizes 10 core regions with a design equilibrium fuel burnup of 50,000 Mwd/mtU at discharge. This improved fuel cycle alone is capable of realizing an 18.1% reduction in the 30-year cumulative requirement for U_3O_8 . Although uneconomical at the present time, it represents a logical limiting extension of core regionalization. The remaining improvements can individually realize additional reductions, and the composite concept is capable of achieving a 27% reduction in the 30-year U_3O_8 requirements. At 0.1% tails enrichment, the reduction in 30-year U_3O_8 resource requirements is slightly larger (27.4% for the composite case).

The composite concept utilizes all of the improvements listed above. Performance characteristics and a material balance are given in Table 2.2. In addition to the improvement in resource utilization, fissile plutonium production is reduced from 4459 kg (30-year total) to 2464 kg, nearly a 45% reduction, and the cumulative separative work requirements reduced by approximately 30%. At 0.1% tails enrichment, the 30-year SWU requirement for the composite improved concept (3.24×10^6) is very nearly the same as the reference design at 0.2% tails (3.36×10^6). Table 2.3 summarizes and compares some of the major performance characteristics of the reference design and the composite improved concept.

2.3 Fuel Cycle Economics

Fuel cycle costs for the composite concept are summarized in Table 2.4. In the absence of penalties due to higher fuel fabrication or disposal costs, the fuel cycle costs generally decrease as resource requirements are reduced. To achieve the higher fuel

Table 2.1 POTENTIAL IMPROVEMENTS IN URANIUM UTILIZATION AT
70% CAPACITY FACTOR, 0.2% TAILS ENRICHMENT

<u>Case</u>	<u>Mwd/ST U₃O₈</u> <u>% Increase</u>	<u>30 Year U₃O₈</u> <u>% Decrease</u>
Reference 3-region, 30,000 Mwd/mtU	0	0
Improved Cycle 10-region, 50,000 Mwd/mtU	27.0	18.1 ⁽¹⁾
Increased Moderation	2.4	3.9
Low Leakage Fuel Management	2.6	2.5
Axial Blanket	0.9	0.9
Power Coastdown	3.9	3.5
Composite	40+	27.3 ^{(2) (3)}

-
- (1) 18.5% decrease if both reference and improved cycle at 0.1% tails enrichment.
- (2) 27.4% decrease if both reference and composite at 0.1% tails enrichment.
- (3) Decreasing the tails enrichment from 0.2% to 0.1% reduces the 30-year U₃O₈ requirement by 13.5% for the reference case and 13.7% for the composite case.

Table 2.2 PERFORMANCE AND MATERIAL BALANCE FOR COMPOSITE CONCEPT

	<u>@0.1% Tails</u>	<u>@0.2% Tails</u>
Discharged fuel burnup (equilibrium), Mwd/mtU		50,000
Refueling interval (@70% capacity factor), months		5.34
Number of fuel regions in core		10
Equilibrium cycle feed enrichment, wt%		3.332
Fuel specific power, Mw/mtU		44
Fuel residence time, years		4.45
Fuel Utilization, Mwd/STU ₃ O ₈	7271	6276
Initial loading		
U-235, kg		1343
Uranium, mt		69.1
U ₃ O ₈ , ST	277	320
Total 30 year requirement U ₃ O ₈ , ST	3434	3978
30 year enrichment service, SWU	3.40x10 ⁶	2.46x10 ⁶
30 year fissile Pu production, kg		2464
30 year total Pu production, kg		4154
Discharged fuel composition(equilibrium)		
U-235, kg/mtU		2.460 (12.53) *
Pu-239, kg/mtU		4.275 (3.80)
Pu-240, kg/mtU		2.661 (1.56)
Pu-241, kg/mtU		1.386 (0.92)
Pu-242, kg/mtU		1.221 (0.43)
Total Pu, kg/mtU		9.543 (6.61)
Fissile Pu, kg/mtU		5.661 (4.62)
U ²³⁸ /U ²³⁵ EOL/BOL		0.964
Equilibrium cycle annual requirements		
U-235, kg		518
U-238, mt		15.02
Total Uranium (3.33% enrichment), mt		15.53
Enrichment, SWU	1.081x10 ⁵	7.83x10 ⁴
Total U ₃ O ₈ , ST	107	124
Equilibrium cycle annual discharge		
U-235, kg		3.57
Uranium (@0.246% enrichment), mt		14.51
Fissile Pu, kg		82.1
Total Pu, kg		138

* Parenthetical numbers are the core average values over all 10 regions of the core at the end of the 30-year reactor lifetime.

Table 2.3 COMPARISON OF REFERENCE DESIGN
AND COMPOSITE CONCEPT

	<u>Reference Design</u>	<u>Composite Concept</u>
30-year U_3O_8 requirement, ST U_3O_8		
@ 0.2% tails	4473	3978
@ 0.1% tails	5170	3434
30-year SWU requirement		
@ 0.2% tails	3.31×10^6	2.46×10^6
@ 0.1% tails	4.60×10^6	$3.40 \times 10^6^*$
Annual fuel requirement, mtU/year	23.53	15.53
Equilibrium cycle feed enrichment, %	3.10	3.33
Discharged fuel, mtU/year	22.43	14.51
Discharge U enrichment, %	0.828	0.246
30-year Pu production, kg		
Fissile	4459	2464
Total	6215	4154

* For a constant 30-year SWU requirement of 3.31×10^6 (reference design at 0.2% tails), the corresponding tails enrichment in the composite concept would be 0.106% and the 30-year U_3O_8 requirement would become 3462 ST U_3O_8 .

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Table 2.4 SUMMARY OF LEVELIZED FUEL CYCLE COSTS

Case	U ₃ O ₈ Cost \$/lb	Unit Costs, \$/Kg		Fuel Cycle Costs, mills/Kwhr					
		Fab.	Disp.	U ₃ O ₈	Enrich	Fab.	Post-Irrad	Incore Interest	Total (\$) *
Reference @ 0.2% Tails									
\$40/lb U ₃ O ₈	40	140	135	2,447	1,970	0.580	0.518	0.638	6.153 (1,132)
\$100/lb U ₃ O ₈	100	140	135	6,118	1,970	0.580	0.518	1.191	10.376 (1,909)
Composite Case @ 0.2% Tails									
\$40/lb U ₃ O ₈	40	140	135	1,744	1,446	0.383	0.342	0.578	4.492 (827)
\$100/lb U ₃ O ₈	100	140	135	4,360	1,446	0.383	0.342	1.065	7.595 (1,397)
10% Increased Fabrication	40	154	135	1,744	1,446	0.421	0.342	0.587	4.540 (835)
20% Increased Fabrication	40	168	135	1,744	1,446	0.460	0.342	0.596	4.588 (844)
Increased Fab. & Disposal	40	170	165	1,744	1,446	0.465	0.418	0.574	4.647 (855)
@ \$100/lb U ₃ O ₈	100	170	165	4,360	1,446	0.465	0.418	1.058	7.747 (1,425)
Reference @ 0.1% Tails									
\$40/lb U ₃ O ₈	40	140	135	2,117	2,682	0.580	0.518	0.695	6.592 (1,213)
\$100/lb U ₃ O ₈	100	140	135	5,293	2,682	0.580	0.518	1.115	10.188 (1,875)
Composite Case @ 0.1% Tails									
\$40/lb U ₃ O ₈	40	140	135	1,505	1,961	0.383	0.342	0.630	4.821 (887)
\$100/lb U ₃ O ₈	100	140	135	3,763	1,961	0.383	0.342	1.050	7.498 (1,380)
10% Increased Fabrication	40	154	135	1,505	1,961	0.421	0.342	0.639	4.868 (896)
20% Increased Fabrication	40	168	135	1,505	1,961	0.460	0.342	0.648	4.916 (905)
Increased Fab. & Disposal	40	170	165	1,505	1,961	0.465	0.418	0.625	4.975 (915)
@ \$100/lb U ₃ O ₈	100	170	165	3,763	1,961	0.465	0.418	1.042	7.649 (1,407)

* Parenthetical numbers are the total fuel cycle costs, in millions of dollars, without escalation, for a 1000 Mw(e) plant over a 30-year reactor operating period at 70% capacity factor.

burnup, the fuel fabrication cost could possibly increase due to the necessity of accommodating higher fission gas release and to allow for modifications in fuel element design necessitated by the potential problem of pellet-clad interaction. The effect on fuel cycle economics of 10% and 20% increased fuel fabrication costs are also shown on Table 2.4.

Modification of the fuel element design to utilize smaller diameter fuel pins (increased water-to-fuel ratio) will probably increase the unit costs for fuel fabrication and for spent fuel disposal approximately in inverse proportion to the quantity of fuel contained in each fuel pin. The effect of such increases is also included in Table 2.4.

Low leakage fuel management, the use of axial blankets of natural UO_2 , and end-of-cycle power coastdown will not necessarily result in an increase in unit fuel fabrication costs. In the case of axial blankets, a small increase in unit fabrication costs could possibly result from a requirement for increased quality control/inspection because of the use of fuel pellets of two different enrichments in the same pin. This has been neglected in the present evaluation.

2.4 Feasibility of Composite Concept

All of the potential improvements investigated (except the short refueling interval) would appear to be technically feasible within the next 5-10 years. Detailed evaluation will be necessary to confirm the preliminary values reported here and to establish performance characteristics under accident conditions as a prerequisite for licensing considerations. Achieving the potential improvements indicated herein will involve an increase in power peaking factors and modified reactivity coefficients. In the composite concept, the peak pin power may be as much as 14% higher than in the reference design, in addition to the higher performance (surface heat flux) for the reduced pin diameter.

The composite concept operates with a refueling interval of 5.34 months at the anticipated specific power. With this short refueling interval, it may be difficult to avoid excessive outage time for refueling and maintenance operations. Reducing fuel specific power would reduce reactor power and result in excessive economic penalty. Decreasing the number of regions will increase the refueling interval, accompanied by an increase in resource utilization and fuel cycle costs. A 5-region core

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(11-month refueling interval), for example, would increase fuel cycle costs by ~0.53 mills/kw-hr and increase U_3O_8 requirements by ~10% over the reference composite concept.

3.0 FUNCTIONAL RELATIONSHIPS

Fuel utilization depends primarily on the fuel burnup achieved and the enrichment required to provide the necessary reactivity. External constraints, such as safety considerations, fuel performance restrictions or available reactivity control, limit the practical range of achievable fuel burnup and corresponding fuel enrichments. Improvements in fuel utilization result from modifications in design or operating procedures that provide increased reactivity for a given enrichment and thereby allow a higher fuel burnup to be achieved (or, conversely, allow a reduced enrichment for the same discharge fuel burnup). Methods of achieving higher reactivity and, hence, improved fuel utilization fall into several general categories:

- Improved fuel management,
- modifications in fuel assembly design,
- reduced neutron leakage,
- reduced neutron parasitic absorption, and
- improved plant performance.

Within each of these general categories, there are several methods of achieving higher reactivity, although not all potential improvements are mutually compatible, nor are all potentially retrofittable in current reactors. In prior studies, (2-5) as well as in the present study, a number of potential improvements have been investigated. The improvements reported here are those considered to be the most reasonable within identified constraints.

Two indices of performance have been used—fuel utilization in terms of Mwd/ST U_3O_8 and resource utilization in terms of the accumulated 30-year requirement for U_3O_8 ore. These two indices are discussed in the following paragraphs.

3.1 Fuel Utilization; Mwd/ST U_3O_8

Fuel utilization in terms of Mwd/ST U_3O_8 refers to the thermal energy produced and (for a given tails enrichment) is dependent on the fuel burnup achieved and the required enrichment in the equilibrium fuel cycle, i.e.,

Fuel utilization,

$$\text{Mwd/ST U}_3\text{O}_8 = 0.76926 B \left(\frac{0.711 - E_T}{E - E_T} \right), \quad (3.1)$$

where B is the burnup achieved for an enrichment E, and E_T is the tails enrichment in the diffusion process. This relationship becomes

$$\text{Mwd/ST U}_3\text{O}_8 = 0.3931 \left(\frac{B}{E - 0.2} \right) \quad (3.2)$$

for a tails enrichment of 0.2% U-235 and

$$\text{Mwd/ST U}_3\text{O}_8 = 0.4700 \left(\frac{B}{E - 0.1} \right) \quad (3.3)$$

for a tails enrichment of 0.1% U-235.

3.2 Resource Utilization (30-Year U_3O_8 Requirement)

Resource utilization refers to the accumulated U_3O_8 required over a 30-year operating period, normalized to a plant of 1000 Mw(e) generating capacity. This index of performance includes the net plant thermal efficiency (ξ), the average plant capacity factor (C), and an estimate of the penalty associated with the initial operating cycles before equilibrium is achieved. Reference 1 developed a general expression for the 30-year resource utilization as follows:

$$\text{ST}_{\text{U}_3\text{O}_8}^{30 \text{ yr}} = 4.745 \times 10^5 \left(\frac{C}{B \times \xi} \right) \left(\frac{E - E_T}{0.711 - E_T} \right) \left\{ 30 + \frac{Y}{2N} (N - 1) \left(1 + \frac{E_f}{E} \right) \right\} \quad (3.4)$$

where Y is the fuel operating period at equilibrium in years, N is the number of (refueling) regions in the core, and E_f is the U-235 enrichment in the discharged fuel. The significance of achievable burnup and required enrichment is evident in equation 3.4.

For the present study, a capacity factor of 70% and a net thermal efficiency of 32.9% were assumed. In addition, the final U-235 enrichment term does not represent a large contribution at relatively high burnup and may be approximated as 25% of the initial enrichment. With these assumptions, equation 3.4 reduces to

$$ST_{U_3O_8}^{30 \text{ yr}} = \frac{6.3098 \times 10^5}{B} \left(\frac{E - E_T}{0.711 - E_T} \right) \left(48 + \frac{N-1}{N} Y \right) \quad (3.5)$$

Equation 3.5, used in the assessment of individual improvements, introduces a maximum error of ~1% at 50,000 Mwd/mtU. However, the error would be smaller in the relative comparison between individual improvements. For the final composite concept, the more exact equation 3.4 was used to calculate the 30-year resource utilization.

3.3 Refueling Interval and Fuel Specific Power

Although it is common in the industry to speak of an "annual" fuel cycle (or some multiple thereof), the actual refueling interval at equilibrium depends upon the discharge burnup (B), fuel specific power (S), and average capacity factor (C), and the number of regions (N) in the core. The relationship between these parameters is

$$\text{refueling interval, months} = 0.03288 \frac{B}{SCN} \quad (3.6)$$

For the reference design used in the present study, the fuel specific power of 36.1 Mw/mtU results in a refueling interval (cycle length) of 14.3 months for a discharge burnup of 33,000 Mwd/mtU in a 3-region core at an average capacity factor of 70%. To achieve a fuel cycle of 12 months in a 3-zone core at 70% capacity factor, it would be necessary to reduce the fuel burnup to 27,690 Mwd/mtU or to increase the specific power to 43.05 Mw/mtU (or some appropriate combination), neither of which would be advantageous. Alternatively, a different average capacity could be assumed, with a capacity factor of 83.4% necessary to reduce the refueling interval from 14.3 months to 12 months. For comparison purposes, a constant capacity factor should be used; and, consequently, no adjustment was made to the reference design to achieve an annual fuel cycle.

The cycle length or refueling interval affects the fuel cycle cost as a result of its effect on interest charges. For the reference design, reducing the cycle length to 12 months (by arbitrarily adjusting the fuel specific power) would reduce the fuel cycle cost from 6.153 mills/kw-hr to 6.042 mills/kw-hr (at a uranium cost of \$40/lb U₃O₈) and would reduce the 30-year U₃O₈ requirement by less than 1%.

In practice, the refueling interval and average capacity factor are both affected by differing utility policies and planning

requirements. The desirability of scheduling refueling during periods of minimum power demand will be considered by the operating utility in the context of the particular circumstances faced by the individual utility. However, for comparing potential PWR improvements, it is more important that the evaluation be made under a consistent set of ground rules than whether or not the refueling interval is consistent with currently predominant utility practice.

4.0 REFERENCE DESIGN AND ANALYTICAL METHODS

4.1 Reference Design

Earlier studies by Southern Science ^(2,3) used as a reference the Westinghouse 15 x 15 fuel assembly operating in a reactor of approximately 870 Mw(e) net capacity. For this concept, the design discharge fuel burnup was 33,000 Mwd/mtU, requiring 3.2 wt% U-235 feed enrichment in a 3-zone core for acceptable reactivity performance at equilibrium. Subsequent changes in fuel element design to a 17 x 17 assembly did not materially affect the nuclear performance and the U₃O₈ resource requirements therefore remained very nearly the same. However, increasing reactor size reduces neutron leakage and allows the required enrichment to be reduced. The Westinghouse RESAR ⁽⁶⁾ indicates that, with a 3800 Mw(t) reactor design (~1250 Mw(e)), an enrichment of 3.1% U-235 is adequate to achieve a burnup of 33,000 Mwd/mtU. Since this is more representative of the newer PWRs, it has been adopted as the reference design.

Net thermal efficiency of an operating plant directly affects the 30-year resource requirements for U₃O₈. For the present study, an efficiency of 32.9% was assumed to be representative of most PWR systems now in operation or under construction.

Table 4.1 summarizes the reference design used for the present study and Table 4.2 gives the fuel utilization factors and mass flows for an assumed 70% capacity factor, and for tails enrichments of 0.1% and 0.2%. For the reference design adopted, the accumulated 30-year resource requirement is 4733 and 5470 standard tons of U₃O₈ at 0.1% and 0.2% tails enrichment, respectively.

Previous evaluations of resource requirements have generally assumed a plant capacity factor of 75% in contrast to 70% used for the present study. For comparison, Table 4.3 lists the 30-year resource requirements for the earlier Southern Science and Combustion Engineering ^(4,7) studies at 0.2% tails enrichment. The difference in resource requirements for the reference design in the present study and that of the earlier SSA study is attributable to differences in enrichment required and in the thermal efficiency. Differences with the CE results are most likely an

Table 4.1 REFERENCE DESIGN CORE CHARACTERISTICS

Type of Fuel	Zr clad UO_2
Core Diameter (active)	132.7 in.
Core Height (active)	168 in.
Number Fuel Assemblies	193
Rod Array	17 x 17
Rods per Assembly	264
Fuel Cell	
Rod pitch, square	0.496 in.
Rod O.D.	0.374 in.
Gap thickness	0.0065 in, (diametral)
Clad thickness	0.0225 in.
Clad material	Zr-4
UO_2 density	95% T.D.
Overall Assembly Transverse	
Dimensions	8.426 in. x 8.426 in.
Number Guide Thimbles per Assembly	24
Guide Thimble Dimensions	0.450 in. I.D. x 0.482 in. O.D. (Zr-4)
Fuel Enrichment (equilibrium)	3.1 wt.% U-235
Design Fuel Burnup (equilibrium)	33,000 Mwd/mtU
Reactor Power Level	3800 Mwt 1250 Mw(e) (net)
Average Linear Heat Rating	5.20 kw/ft
Core Power Density	99.8 kw/l
Core Specific Power	36.1 Mw/mtU
Average Core Coolant Temperature	597.5°F
Average Fuel Temperature	1421°F
Average Clad Temperature	617.5°F

Table 4.2 REFERENCE DESIGN FUEL UTILIZATION
1000 Mw(e) PLANT AT 70% CAPACITY FACTOR

	@ 0.1% Tails	@ 0.2% Tails
Discharged fuel burnup (equilibrium, Mwd/mtU)		33,000
Refueling interval (@70% capacity factor), months		14.3
Number of fuel regions in core		3
Equilibrium cycle feed enrichment, wt%		3.1
Fuel specific power, Mw/mtU		36.1
Fuel residence time, years		3.58
MWD/STU ₃ O ₈	5170	4473
Initial loading/GWe		
U-235, kg		1970
Uranium, mt		84.2
U ₃ O ₈ , ST	405	470
Total 30 year requirement U ₃ O ₈ , ST	4733	5470
30 year enrichment service, SWU	4.60x10 ⁶	3.31x10 ⁶
30 year fissile Pu production, kg		4459
30 year total Pu production, kg		6215
Discharged fuel composition (equilibrium)		
U-235, kg/mtU		7.932 (20.46) *
Pu-239, kg/mtU		5.431 (4.78)
Pu-240, kg/mtU		2.171 (1.30)
Pu-241, kg/mtU		1.364 (0.77)
Pu-242, kg/mtU		0.505 (0.21)
Total Pu, kg/mtU		9.471 (7.05)
Fissile Pu, kg/mtU		6.795 (5.54)
U ²³⁸ /U ²³⁵ EOL/BOL		0.976
Equilibrium cycle annual requirements		
U-235, kg		729.5
U-238, mt		22.80
Total Uranium (3.1% enrichment), mt		23.53
Enrichment, SWU	1.476x10 ⁵	1.065x10 ⁵
Total U ₃ O ₈ , ST	150.2	173.6
Equilibrium cycle annual discharge		
U-235, kg		186
Uranium (@0.828% enrichment), mt		22.43
Fissile Pu, kg		148.6
Total Pu, kg		207.2

* Parenthetical numbers are the core average values over all 3 regions of the core at the end of the 30-year reactor lifetime.

Table 4.3 A COMPARISON OF RESOURCE REQUIREMENTS
FROM RECENT STUDIES

	<u>Enrichment % U-235</u>	<u>Burnup Mwd/mtU</u>	<u>ST U₃O₈ 30-Yr Requirement at 0.2% Tails</u>		<u>Net Plant Eff.</u>
			<u>@70% Cap. Factor</u>	<u>@75% Cap. Factor</u>	
Prior SSA Study ⁽⁷⁾	3.2	33,000	5870 (5660)	6270 (6040)	31.70% (@32.90%)
CE Study ⁽⁴⁾	3.0	30,380	5650*	6040	33.24%
Present Study	3.1	33,000	5470	5840	32.90%

* Estimated.

artifact of the analytical methods* used rather than to any real differences between Combustion Engineering and Westinghouse designs. In any event, the relative percent reduction in resource requirements is more significant than small differences in the absolute magnitude over a 30-year operating period.

4.2 Analytical Methods

4.2.1 Nuclear Performance

Nuclear performance characteristics were determined by zero dimensional unit cell calculations using the NULIF code,⁽⁸⁾ a multigroup cell homogenization program with coupled depletion routines. Neutron leakage was included through a geometric buckling factor derived from the nominal equilibrium cycle design parameters. For the special case of axial blankets, a few one-dimensional calculations using PDQ-7⁽⁹⁾ were necessary, using fitted diffusion theory constants derived from the NULIF cell calculations.

4.2.2 Fuel Cycle Cost Analyses

Fuel cycle cost calculations were performed with the CYCO code⁽¹⁰⁾ which calculates levelized fuel costs using the ground rules described in Section 4.2.3 below. Material costs, manufacturing costs and pre- and post-irradiation capital charge costs were amortized over a single batch irradiation at equilibrium, present worth to January 1980. For the reference case described in Section 4.1 above, the levelized fuel cycle costs are given in Table 4.4, assuming the ground rules discussed in Section 4.2.3 below.

* A difference of $\sim 0.6\%$ k in calculated reactivity (and, hence, of required enrichment) would account for the difference in resource requirements accumulated over a 30-year operating period. This is within the expected analytical uncertainty.

Table 4.4 LEVELIZED FUEL CYCLE COSTS
FOR THE REFERENCE DESIGN

	Mills/kw-hr				
	<u>Uranium</u>	<u>Fabrication</u>	<u>Post-Irrad.</u>	<u>Interest</u>	<u>Total</u>
\$40/lb U_3O_8 , 0.2% tails	4.417	0.580	0.518	0.638	6.153
\$40/lb U_3O_8 , 0.1% tails	4.799	0.580	0.518	0.695	6.592
\$100/lb U_3O_8 , 0.2% tails	8.087	0.580	0.518	1.191	10.376

4.2.3 Economic Ground Rules for Fuel Cycle Cost Evaluation

The basic ground rules used for the fuel cycle evaluations are listed in Table 4.5. These ground rules are generally consistent with the NASAP unit costs. Two of the unit costs—fuel fabrication and ore costs—were investigated parametrically since their actual values are uncertain.

Table 4.5 GROUND RULES FOR FUEL CYCLE COST EVALUATIONS

Uranium cost	\$40/lb U_3O_8 (variable)
Conversion	\$4/kg natural U
Enrichment	\$100/SWU
Tailings enrichment	0.1% and 0.2%
Fabrication (includes losses & transportation of fresh fuel)	\$140/kg U (variable)
Spent Fuel Disposal (includes transportation and storage of spent fuel)	\$134/kg U (\$165/kg U for small diameter fuel pin)
Discount rate (present worth factor)	4.525%
Fuel Carrying Charge Rate	8.0%
Capacity Factor	70%
Pre-irradiation period ⁽¹⁾	1 year
Post-irradiation period ⁽²⁾	1 year
Escalation factor	none

(1) Assumes all costs for uranium conversion, enrichment and fabrication paid 1 year prior to installation in reactor.

(2) Assumes spent fuel shipping and disposal charges paid 1 year after the fuel is discharged from the reactor.

5.0 PRESSURIZED WATER REACTOR WITH IMPROVEMENTS

5.1 Nuclear Performance of Reference Design

In assessing fuel utilization in any reactor, the reactivity and its variation with fuel burnup is the single most important factor. Figure 5.1 shows the burnup dependent reactivity (k_{∞}) for the reference design at several assumed values of the initial U-235 enrichment. Figure 5.2 summarizes the initial calculated k_{∞} values as a function of fuel enrichment. Assuming the overall core reactivity can be approximated as the linear average of the reactivity in each region in the core, the achievable fuel burnup (equilibrium cycle with uniform incremental burnup in each region) is shown in Fig. 5.3. The data in Fig. 5.3 is normalized to an end-of-cycle average core reactivity of 1.000 for the reference design fuel burnup of 33,000 Mwd/mtU at 3.1 wt% U-235 initial enrichment. With this normalization, the end-of-cycle average k_{∞} is 1.0522, implying that the effective neutron leakage is 5.22% including the effect of non-uniform burnup and any inadequacy in region reactivity weighting. The data in Figs. 5.1-5.3 and method of averaging constitutes the basis on which subsequent evaluations of potential improvements were made.

5.2 Fuel Burnup and Core Regionalization

5.2.1 General Relationships

Based upon the data for the reference design given in Section 5.1 above, uranium utilization may be calculated for various combinations of equilibrium feed fuel enrichment, average discharge fuel burnup and number of core regions. Results of these calculations are given in Figs. 5.4 and 5.5, which illustrate the improvement in uranium utilization that can be achieved by increasing the discharge fuel burnup and number of core regions.

Increased fuel burnup alone (with necessary increases in feed enrichment to maintain proper reactivity) for a given number of core regions does not greatly improve uranium utilization. However, increasing the number of core regions at a given burnup reduces the cycle length and yields a higher average reactivity over a fuel cycle. This in turn allows a lower feed enrichment to be used and thereby improves uranium utilization significantly. For any given number of core regions, the

Fig. 5.1 Variation in k_{∞} with fuel exposure.

Fig. 5.2 Variation in initial k_{∞} with fuel enrichment.

N 3
 E 4.54
 10 3.62
0.92
 1.385
1.283
 1.023

Fig. 5.3 Achievable fuel burnup as function of equilibrium feed fuel enrichment for various number of regions.

Fig. 5.4 Parametric variation in fuel utilization (Mwd/ST U_3O_8).

Fig. 5.5 Parametric variation in 30-year U_3O_8 resource requirements.

optimum discharge fuel burnup is between 40,000 and 50,000 Mwd/mtU. At a constant refueling interval, increasing both fuel burnup and number of core regions continues to improve uranium utilization without reaching an optimum in the range of the calculations, although there is a suggestion that an optimum is being approached at about 50,000 Mwd/mtU.

The shape of the curves and the resource utilization values in Figs. 5.4 and 5.5 are a sensitive function of the burnup dependent reactivities shown in Figs. 5.1 and 5.3. For example, at 33,000 Mwd/mtU nominal burnup, an increase of 1% in k will increase the achievable burnup by ~2100 Mwd/mtU and result in a 5.7% reduction in 30-year U_3O_8 requirements. Similarly, at 50,000 Mwd/mtU nominal burnup, a 1% increase in k can increase burnup by ~3200 Mwd/mtU and improve resource utilization by 5.5%. If a 1% higher k is used to reduce enrichment, rather than increase burnup, the reduction in resource utilization would range between 5.2% at 33,000 Mwd/mtU to 6.1% at 50,000 Mwd/mtU. Because of the sensitivity to calculated reactivity, especially as a function of fuel burnup, the absolute value of burnup for optimum resource utilization cannot be predicted precisely with the analytical techniques used in this study. In Fig. 5.5, the optimum burnup which appears to be ~46,000 Mwd/mtU could perhaps range from about 41,000 Mwd/mtU to 50,000 Mwd/mtU.

5.2.2 Reference Improved Fuel Cycle

Extrapolation of discharge fuel burnup above the current design level of ~33,000 Mwd/mtU involves some uncertainty due to potential problems that may be encountered. Reference 1 indicates some experience with fuel burnups approaching 40,000 Mwd/mtU, although there is very limited experience at burnups of 50,000 Mwd/mtU or higher. At the present time, there are government programs under way seeking to verify acceptable fuel performance at a discharge burnup of about 50,000 Mwd/mtU. Consequently, it is judged that an average discharge burnup of 50,000 Mwd/mtU is a reasonable objective for near-term improved uranium utilization.

Selection of a reference number of core regions—and, hence, cycle length—is more difficult and subject to greater uncertainty. However, for evaluation purposes, a 10-region core has been adopted as the reference improved cycle, operating to an average discharge burnup of 50,000 Mwd/mtU with a 6.5-month cycle length. Table 5.1 summarizes the operating parameters for the selected reference improved fuel cycle. This improved cycle

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Table 5.1 PERFORMANCE AND MATERIAL BALANCE FOR
REFERENCE IMPROVED FUEL CYCLE

	@ 0.1% Tails	@ 0.2% Tails
Discharged fuel burnup (equilibrium), Mwd/mtU	50,000	
Refueling interval (@70% capacity factor), months	6.5	
Number of fuel regions in core	10	
Equilibrium cycle feed enrichment, wt%	3.66	
Fuel specific power, Mw/mtU	36.1	
Fuel residence time, years	5.42	
Initial loading		
U-235, kg	1887	
Uranium, mt	84.2	
U ₃ O ₈ , ST	390	455
Total 30 year requirement		
U ₃ O ₈ , ST	3856	4481
30 year enrichment service, SWU	3.96x10 ⁶	2.87x10 ⁶
30 year fissile Pu production, kg	3421	
30 year total Pu production, kg	5117	
Discharged fuel composition at equilibrium		
U-235, kg/mtU	5.07	(15.83)*
Pu-239, kg/mtU	6.011	(4.90)
Pu-240, kg/mtU	2.87	(1.62)
Pu-241, kg/mtU	1.87	(1.03)
Pu-242, kg/mtU	1.04	(0.38)
Total Pu, kg/mtU	11.79	(7.94)
Fissile Pu, kg/mtU	7.88	(5.94)
U ²³⁸ /U ²³⁵ EOL/BOL	0.961	
Equilibrium cycle annual requirements		
U-235, kg	568	
U-238, mt	14.96	
Total Uranium (3.66% enrichment), mt	15.53	
Enrichment, SWU	1.232x10 ⁵	8.97x10 ⁴
Total U ₃ O ₈ , ST	118	137
Equilibrium cycle annual discharge		
U-235, kg	73	
Uranium (@0.51% enrichment), mt	14.46	
Fissile Pu, kg	114	
Total Pu, kg	171	

* Parenthetical numbers are the core average values over all 10 regions of the core at the end of the 30-year lifetime.

(at 0.2% tails) reduces the 30-year U_3O_8 requirement to 4481 ST U_3O_8 , an 18.1% improvement over the base reference design.

5.2.3 Fuel Cycle Economics

Fuel cycle costs have also been evaluated for the various options in core regionalization and discharge fuel burnup described in Section 5.2.1 above. Results of this evaluation are summarized in Fig. 5.6, assuming no increase in fuel fabrications costs for fuel or higher burnup and no penalty associated with reduced cycle lengths. To assess the impact of potentially increased fuel fabrication costs, the fuel cycle costs for the reference improved fuel cycle (Section 5.2.2 above) were evaluated for assumed 10% higher fabrication costs. Results, given in Table 5.2, show that the improved fuel cycle is still more economical than the reference cycle, provided the probable penalty resulting from increased outage time is not so large as to eliminate the apparent economic advantage of the improved fuel cycle.

5.2.4 Technical Feasibility and Licensing Concerns

Achieving a fuel burnup of 50,000 Mwd/mtU would appear to be technically feasible within the next 10 years. A few fuel elements have achieved burnups exceeding 50,000 Mwd/mtU and a substantial number have been burned to nearly 40,000 Mwd/mtU. Government-sponsored programs directed toward achieving burnups of 50,000 Mwd/mtU or more have already been initiated, with no obvious reasons why they should not be successful. Two potential problem areas—pellet-clad interaction and higher fission gas release—may possibly require modified fuel element designs with attendant higher fabrication costs to keep failure rates within acceptable bounds. An allowance for potentially higher fabrication costs is included in the economic assessment (parametrically) in Section 5.2.3.

Multiregion cores should present no technical difficulty. However, maintaining short operating periods (e.g., 6 months) with excessive loss in achievable capacity factor due to outage time is more uncertain. At the present time, fuel-moving operations are not the critical path item in refueling. Maintenance, inspection, and other procedures are the critical path items; and, therefore, achieving a short refueling interval, to avoid excessive replacement power costs, poses the most serious

Fig. 5.6 Parametric variation in equilibrium fuel cycle costs.

Table 5.2 LEVELIZED FUEL CYCLE COSTS
WITH INCREASED BURNUP

	mills/kw-hr (@\$40/lb U ₃ O ₈)				
	<u>Uranium</u>	<u>Fabrication</u>	<u>Post-Irrad.</u>	<u>Interest</u>	<u>Total</u>
Reference, 33,000 Mwd/mtU	4.417	0.580	0.518	0.638	6.153
Improved cycle, 50,000 Mwd/mtU Fabrication @\$140/kg U	3.580	0.383	0.342	0.808	5.112
Improved cycle, 50,000 Mwd/mtU Fabrication @\$154/kg U	3.580	0.421	0.342	0.817	5.160
Improved cycle, 50,000 Mwd/mtU Fabrication @\$168/kg U	3.580	0.460	0.342	0.826	5.208

problem. Equipment failure rates could either be increased due to more frequent shutdowns for refueling or reduced due to the opportunity for more frequent inspection and maintenance. Evaluation of these factors is beyond the scope of the present study, other than to recognize their potential significance.

Regulatory (licensing) requirements do not specifically restrict fuel burnup or the refueling interval. However, each refueling is individually reviewed by regulatory authorities, and higher fuel burnup will likely require supporting evidence to demonstrate acceptable fuel integrity under postulated accident conditions and to provide adequate assurance that other safety considerations (e.g., reactivity coefficients, transient and accident response with higher fission product inventories, etc.) are not compromised. The larger inventory of longer-lived fission products could also affect spent fuel storage rack designs and long-term disposal requirements.

An adequate data base for a statistically significant number of fuel elements at the higher burnup will necessitate several years of operation at increasing discharge fuel burnups, accompanied by detailed safety analyses and fuel management calculations. However, the risk of failing to achieve high fuel burnup is judged to be minimal.

5.3 Increased Moderation

5.3.1 Conceptual Design Modifications

Decreasing the amount of moderator in the reactor core (drier lattice) increases resonance capture of neutrons in U-238 and increases the conversion ratio. Despite the greater amount of plutonium produced, however, the enrichment required to maintain reactivity is higher and the result is a net loss in resource utilization. For higher moderation in the core (wetter lattice), the higher reactivity (from reduced U-238 capture) more than compensates for the reduced plutonium production in parasitic capture in water, resulting in a net improvement in resource utilization. Beyond the level of optimum moderation, the loss of neutrons to parasitic capture in the moderator again reduces reactivity.

Licensing considerations limit the practical level of core moderation in order to satisfy requirements for negative temperature coefficient of reactivity. Although detailed evaluation of the temperature coefficient of reactivity throughout lifetime

has not been made, a water-to-fuel ratio of 2.8 (core average) has been adopted to illustrate the effect of increased moderation. At this level of moderation, the temperature coefficient of reactivity (initial unpoisoned case) is $\sim -4.6 \times 10^{-4} \Delta k/^\circ F$ compared to $\sim -6.1 \times 10^{-4} \Delta k/^\circ F$ for the reference case under the same conditions. With soluble boron poison present, the temperature coefficient will be less negative due to the effect of boron absorption. Anticipating that the improved fuel cycle will operate at a lower soluble boron concentration for reactivity control (reduced operating cycle burnup and increased number of core regions), a water-to-fuel ratio of 2.8 is judged to be a reasonable upper limit for the purpose of illustrating potential improvements in resource utilization.

Higher water-to-fuel ratios can be achieved by increasing the fuel pin pitch, by reducing the pin diameter, by utilizing annular fuel pellets or by reducing the fuel density. The latter two design modifications would entail concern over performance and potential licensability, as well as necessitating considerable expense for R&D. Of the first two options, increasing fuel pin pitch would not be a retrofittable option. Reducing the fuel pin diameter from the reference 0.374-inch O.D. to 0.336-inch O.D. would increase the water-to-fuel ratio to 2.8. However, for the same reactor power, reducing the pin diameter to 0.336-inch O.D. reduces the heat transfer surface by $\sim 10\%$ and the amount of UO_2 in the core by $\sim 22\%$ (thereby increasing fuel specific power to 44 Mw/mtU. At the higher specific power, the period between refueling would be reduced from 14.3 months to 11.7 months for the reference 3-zone core at 33,000 Mw/mtU burnup and from 6.5 months to 5.3 months for a 10-zone core at 50,000 Mw/mtU burnup.

In a 10-zone core at 50,000 Mw/mtU, the enrichment required is reduced to 3.58 wt% U-235, which results in a 3.9% improvement in the 30-year U_3O_8 requirement. At this burnup, the improvement in resource requirement (3.9%) is slightly less than that estimated for a 3-zone core at a burnup of 33,000 Mw/mtU (4.5%).

5.3.2 Fuel Cycle Economics

Increasing the water-to-fuel ratio by decreasing fuel pin size does not require any changes to the reactor system. However, the smaller amount of fuel in each fuel assembly increases the unit fuel fabrication and disposal costs and reduces the operating time period over which interest charges accumulate. Since the fuel manufacturing and spent fuel disposal processes must

handle the same number of pieces and will entail approximately the same number of operations as for the reference design, the unit fuel fabrication costs will increase in approximately inverse proportion to the amount of UO_2 fuel contained. (A small savings in materials cost would result, but this is not included in the present evaluation.)

For a fuel pin of 0.336-inch O.D. (UO_2 pellet O.D. of 0.285 inches), the weight of UO_2 contained in each fuel pin is ~22% less than the reference design, which approximately increases the unit fuel fabrication cost to \$170/kg U and the spent fuel disposal charge to \$165/kg U. Table 5.3 gives the fuel cycle costs with these unit costs for the reference design and for the reduced fuel pin size concept. Although there is only a minor effect on the fuel cycle costs at the reference burnup, the fuel cycle cost is reduced by ~2% at the higher fuel burnup.

5.3.3 Technical Feasibility and Licensing Concerns

Modifying the fuel element design to accommodate smaller diameter fuel pins should not involve any technical problems in manufacture or require major modifications to the reactor system. However, because of the smaller surface area, either the reactor must be de-rated or the fuel assembly must operate at a higher average surface heat flux. The first option—de-rating the reactor power level—would substantially increase the cost of the energy produced and would most likely be unacceptable. Operating at a higher surface heat flux will require detailed thermal-hydraulic analysis and possibly testing to determine the consequences and ultimate acceptability of the higher performance rating.

Licensing concerns relate primarily to the temperature coefficient of reactivity and to the higher thermal-hydraulic performance, including the effect on fission gas release, spent fuel pool cooling, and behavior under accident (LOCA) conditions. Detailed evaluation of these parameters will be required to establish the acceptability of the smaller diameter fuel pins.

Table 5.3 FUEL CYCLE COSTS FOR INCREASED
CORE MODERATION

<u>Case</u>	<u>mills/kw-hr</u>				
	<u>Uranium</u>	<u>Fabri- cation*</u>	<u>Post- Irrad.*</u>	<u>Interest</u>	<u>Total</u>
Reference, 33,000 Mwd/mtU	4.417	0.580	0.518	0.638	6.153
@ W/F = 2.8	4.200	0.698	0.628	0.660	6.185
Improved, 50,000 Mwd/mtU	3.580	0.383	0.342	0.808	5.113
@ W/F = 2.8	3.485	0.465	0.418	0.629	4.996

* Fabrication and spent fuel disposal at \$170/kg U and \$165/kg U, respectively.

5.4 Reduced Neutron Leakage

5.4.1 Low-Leakage Fuel Management

Most current pressurized water reactors operate with three core regions on a fuel management program intended to minimize power peaking and to produce reasonably uniform incremental fuel burn-up under equilibrium conditions. Core performance is constrained by limitations on the peak fuel pin performance, which, in turn, requires a considerable degree of power flattening to achieve a reasonable overall average core performance without exceeding limitations on the peak-power fuel assembly.

Usually, the highest reactivity fresh fuel is loaded on or near the core periphery and partially burned fuel is located interior to the core. As a result, a significant number of neutrons are lost through leakage from the outer boundary of the core. To minimize such losses, low-leakage fuel management schemes have been suggested in an effort to reduce the fraction of neutrons lost through leakage, thereby increasing available reactivity in the core and reducing uranium resource requirements. For a given size core, efforts to reduce neutron leakage inherently involves increased power peaking in the internal regions of the core.

Combustion Engineering^(4,5) has suggested a modification to the traditional out-in fuel management method by utilizing lower-reactivity, partially-burned fuel on the outer core boundary. Power peaking effects internal to the core are minimized by the use of boron burnable poison pins located in selected assemblies. Reactivity losses due to residual unburned boron at the end of a fuel cycle tend to reduce the improvement in reactivity that might otherwise be accomplished. (Alternate burnable poison management or other burnable poisons, e.g., gadolinium or a combination might reduce this penalty.) Nevertheless, CE has evaluated several possible fuel management schemes and demonstrated that some improvement in available reactivity can be realized with reasonable increases in peak pin power.

Because evaluation of low-leakage fuel management schemes involves a large number of 2-dimensional analyses with fuel loading patterns and burnable poison concepts determined by trial-and-error, the CE results are assumed applicable to the present study. At a discharge fuel burnup of the order of 50,000 Mwd/mtU, the CE calculations⁽⁵⁾ indicate a potential reduction of 2.5% in U_3O_8 requirements over a 30-year operating

period (at 0.2% tails enrichment) in a 5-zone core. In the CE calculations, peak pin power is increased by ~6.6% to accomplish the lower neutron leakage and improve resource utilization. The 2.5% reduction in resource requirements is assumed applicable to a 10-region core and corresponds to approximately 0.095 reduction in equilibrium cycle feed enrichment at a discharge fuel burnup of 50,000 Mwd/mtU.

5.4.2 Axial Blanket

Axial blankets of natural or depleted uranium oxide have been investigated as a means of reducing neutron losses from leakage in the axial direction, using one-dimensional calculations with the PDQ-7 diffusion theory code. For the reference case (14-foot active core height), Fig. 5.7 shows the calculated axial power distribution at several values of the average fuel burnup. The tendency toward power flattening with burnup is clearly evident in this figure. At the beginning of a fuel cycle, the fraction of neutrons lost by leakage is approximately 0.0027, or a reactivity worth of ~0.3% Δk . At 50,000 Mwd/mtU, the leakage is increased to ~0.8% Δk due to the effect of power flattening.

Using a 6-inch blanket of natural UO_2 at each end of the fuel (reducing the enriched fuel region by ~7% to 13 feet) requires an increase in enrichment from 4% to 4.2% to achieve the same total reactivity lifetime, i.e., an average discharge burnup of ~50,000 Mwd/mtU in a 5-zone core. Figure 5.8 shows the calculated burnup dependent power distribution. At the beginning of a fuel cycle, the power peaking is increased by ~7%. However, in a reactor core, there will be neutron coupling to nearby partially burned fuel and the actual net increase in power peaking should be less than 7%.

With the 6-inch blanket of natural UO_2 , the fuel utilization (Mwd/ST U_3O_8) is increased by approximately 0.9% and the 30-year U_3O_8 resource requirement is also improved by 0.9%. At discharge (49,300 Mwd/mtU calculated average), the enriched fuel would have burned to 51,950 Mwd/mtU and the natural UO_2 blanket would reach 14,854 Mwd/mtU.

Results obtained in the present evaluation are appreciably less than the 4.2% improvement in fuel utilization reported by Combustion Engineering. To investigate this difference, calculations were made using the CE core with 3.3% enrichment, 11.38-foot core height, and a 15-mm blanket at each end.

Fig. 5.7 Axial power distribution in reference design fuel assembly.

Fig. 5.8 Axial power distribution with 6-inch natural UO_2 blanket.

These calculations indicate that the difference arises from a difference in achievable reactivity lifetime. CE data indicate a reactivity limited burnup of 33,940 Mwd/mtU (assembly average) compared to 32,900 Mwd/mtU calculated by the method outlined herein. In the CE calculations, the energy adjusted burnup of the enriched fuel zone is 36,420 Mwd/mtU compared to 35,300 Mwd/mtU in the present calculation, both referenced to 35,100 Mwd/mtU for the uniform core. This difference in reactivity lifetime corresponds to a difference of 0.5% k.

5.4.3 Fuel Cycle Economics

Based upon the calculations described above, the low-leakage fuel management method effectively reduces the required enrichment by 0.095% U-235 at a discharge fuel burnup of 50,000 Mwd/mtU. Similarly, the axial blanket of natural UO₂ (0.9% reduction in 30-year U₃O₈ requirements) corresponds to an equivalent reduction of 0.03% U-235 in the uniform enrichment. With these potential improvements, fuel cycle costs have been calculated and the results are given in Table 5.4.

5.4.4 Technical Feasibility and Licensing Concerns

Neither low-leakage fuel management nor axial blankets pose any technical problems different from those associated with increased discharge fuel burnup. However, both involve increases in the power peaking factor: up to 6.6% in the case of low-leakage fuel management (CE calculations) and something less than 7% for the axial blanket. While either of these increases in peak pin power might be acceptable to regulatory authorities, it remains to be determined if their combination provides adequate margins to be acceptable, particularly if associated with the higher fuel thermal performance required for reduced pin diameter (increased moderation).

Table 5.4 FUEL CYCLE COSTS WITH REDUCED
AXIAL AND RADIAL LEAKAGE

<u>Case</u>	<u>mills/kw-hr</u>				
	<u>Uranium</u>	<u>Fabrication</u>	<u>Post-Irrad.</u>	<u>Interest</u>	<u>Total</u>
10-Region, 50,000 Mwd/mtU	3.580	0.383	0.342	0.808	5.113
w/low-leakage fuel management	3.467	0.383	0.342	0.783	4.974
w/axial blanket	3.544	0.383	0.342	0.800	5.069

5.5 Power Coastdown

5.5.1 Resource Utilization

At the nominal end of core reactivity lifetime, additional reactivity and, hence, additional fuel burnup can be achieved by reducing the reactor power level (xenon effect) and by reducing the average fuel and coolant temperatures (Doppler and moderator temperature coefficients). In an evaluation of the capability of the balance-of-plant to support a reduction in moderator temperature (through feedwater temperature reduction), CE has estimated that the reactor operating period can be extended by ~12-13 days in a 3-zone core and by 9-10 days in a 5-zone core, if the reactor power is allowed to coast down to 85% of rated power. Decreasing the coastdown level to 75% rated power results in an 18-19 day extension for a 3-zone core and 15-16 days for a 5-zone core. Based upon these data, it is estimated that at least a 4% increase in burnup could be achieved in a 10-zone core operating to a nominal 50,000 Mwd/mtU burnup with a coastdown to 75% power in each cycle. This corresponds to a coastdown time of slightly over 6 days in each cycle. Correspondingly, by using power coastdown to 75% power, the enrichment required to achieve 50,000 Mwd/mtU burnup in a 10-region core can be reduced from 3.66 wt% U-235 to 3.53 wt%, a reduction in 30-year U_3O_8 requirement of 3.5%.

5.5.2 Fuel Cycle Economics

Power coastdown at the same discharge fuel burnup reduces the required enrichment and, hence, reduces the fuel cycle costs. Table 5.5 lists the calculated fuel cycle costs for the 10-region, 50,000 Mwd/mtU core both with and without power coastdown to 75% power.

5.5.3 Technical Feasibility and Licensing Concerns

Power coastdown has been utilized in the past and therefore should pose no feasibility or licensing problems. However, to realize the potential benefit from power coastdown requires a commitment on the part of the operating utility to use coastdown on every cycle. To do so means the utility must sacrifice some operating flexibility and must be willing to accept

Table 5.5 FUEL CYCLE COSTS WITH
POWER COASTDOWN

<u>Case</u>	<u>mills/kw-hr</u>				
	<u>Uranium</u>	<u>Fabrication</u>	<u>Post-Irrad.</u>	<u>Interest</u>	<u>Total</u>
Reference 10-region 50,000 Mwd/mtU	3.580	0.383	0.342	0.808	5.113
w/coastdown to 75%	3.425	0.383	0.342	0.773	4.923

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the reduced power output over the coastdown time period. Acceptance by a utility will be influenced by each utility's operating philosophy, cost of replacement power, and peak demand periods.

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