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1. INTRODUCTION

The Department of Energy (DOE) is sponsoring a program on Advanced Reactor Design Studies, the objective of which is to identify, assess, and develop non-backfittable concepts for improving uranium utilization in light water reactors (LWRs). Non-backfittable concepts are those that cannot be introduced economically into currently operating plants or plants under construction. The program is being conducted for DOE by Battelle/Pacific Northwest Laboratories (PNL).

PNL performed an initial survey of non-backfitable concepts, which resulted in the list attached as Table 1-1. The Babcock & Wilcox Company (B&W) was then requested to participate with other reactor vendors in an industrial assessment of the concepts with the objective of selecting the most promising candidates for more detailed evaluation.

The assessment began with a review of background material provided by PNL in preparation for an initial workshop attended by all participants. Through a joint effort at the initial workshop, the participants selected the concepts and assessment criteria for the subsequent evaluation. Key questions related to each concept were identified, and a uniform format was selected for rating the concepts against the criteria and ranking the concepts against each other.

The initial workshop was followed by a brief period of in-house assessment during which the concepts were evaluated independently by each participant. Following this assessment period, a second workshop was sponsored by PNL to review the conclusions, recommendations, and technical rationale provided by the participants. This report documents the results of our assessment as provided to PNL in support of their overall evaluation of non-backfittable improvements to LWRs.

After consideration of the initial survey results provided by PNL, the workshop participants selected the following concepts and assessment criteria:

Concepts	Criteria
Rapid/frequent refueling	Uranium utilization
Lower power density	Economics
Coastdown	Technical
Radial blanket	Operational
Small fuel assembly	Other considerations
Core periphery modifications	
Higher temperatures and pressures	

The Composite Improved PWR described in Volume 9 of the NASAP program report was defined as the reference design to which these concepts would be applied.[1] The composite includes the following backfittable improvements:

- Extended exposure, annual cycle
- Low-leakage fuel management
- Lattice optimization
- Axial blankets
- Preplanned coastdown
- Full use of early startup core batches

Non-backfittable concepts include some significant departures from current technology. These concepts were assessed primarily on the basis of industrial experience since neither time nor funding would permit detailed engineering analysis. Although the reference design is a Combustion Engineering plant, in certain cases conclusions are necessarily drawn from experience with B&W systems. All fuel utilization and fuel cost savings, however, are based on the NASAP Composite Improved PWR.

Table 1-1. Non-Backfittable Design Modifications

Category	Frequent refueling	Increased system efficiency	Non-backfittable nuclear fuel and core designs
I	PWR rapid refueling system	Higher temperatures and pressures	Coastdown[a]
	BWR rapid refueling system		BWR flow control[a]
	Hot standby refueling		Variable lattices
			Lower power density reactors
			Fissile material control
			Vented fuel
			Blankets[a]
			Reflectors[a]
			Soluble boron for BWR cold shutdown
II		Integral nuclear superheat	Seed blankets
		Add-on nuclear superheat	Spectral shift (without D_2O)
III	Unit core refueling	Add-on fossil fueled superheat	PWR flow control
	On-line refueling		Tubular fuel
		Supercritical pressures	High peaking factor reactor
			Advanced control rod systems

[a] Beyond the limits of backfittability.

2. SUMMARY

Seven non-backfittable improvements to light water reactors were assessed for Battelle/Pacific Northwest Laboratories in support of the Department of Energy's program on Advanced Reactor Studies. The objective was to provide industrial perspective as to which concepts have the best potential for development to improve fuel utilization.

The work reported here comprised several phases. First, within the framework described in Appendix A, the concepts were rated against the assessment criteria while considering the key questions identified for each concept, and recommendations were made for further action on unresolved key questions. Second, the concepts were subjectively ranked against each other in terms of relative investment potential. The ranking considered all criteria but, for example, weighted fuel utilization savings more heavily than development costs. Finally, conclusions and recommendations for future action were determined.

The reference design for this study was the NASAP Composite Improved PWR as described in Volume 9 of reference 1 and in Appendix A of this report. Capital and development costs were estimated in July 1980 dollars. Levelized fuel cycle costs were determined under the assumptions provided in Table 2-1 and normalized to the July 1980 values provided in Appendix A for the NASAP Composite Improved PWR.

The principal results (see also Table 2-2) and conclusions of this study are as follows:

- Substantial uranium savings are available from the non-backfittable concepts, although considerable technical difficulty or costs are involved in some cases.

- The low power density, radial blanket, and core periphery modification concepts should be considered together because all three compete for pressure vessel volume, are interdependent, and yield a combined uranium savings that is less than the sum of their individual contributions.
- Frequent refueling and a low-power-density/blanket/core periphery combination offer the greatest potential for savings. Achieving the quick refueling techniques required for 6-month refueling without increasing annual outage time by more than 10 days will be difficult, however, and will require substantial demonstration. The acceptability of the low-power-density/blanket/core periphery combination can be more readily predicted from a design study.
- Substantially higher pressures and temperatures offer moderate savings, but the technical obstacles appear to outweigh the gains.
- A small additional increase in coastdown capability beyond that already included in the NASAP Composite appears to be achievable at relatively low cost. Further increases will probably require turbine redesign which, while beyond the scope of this study, may be worthy of further consideration.
- The value of a smaller (x-y cross section) fuel assembly remains uncertain. Small fuel utilization gains can result from the greater flexibility provided in fuel management but at a potential penalty in increased handling time. However, the small assembly can facilitate introduction of other concepts, both backfittable and non-backfittable, such as the radial blanket or low-leakage shuffle scheme, and is probably better considered in this aspect.

The following concepts appear to merit further consideration under the Advanced Reactor Design Studies program:

- Rapid/frequent refueling
- Low-power-density/radial blanket/core periphery modifications
- Increased coastdown capability
- Small fuel assembly

Table 2-1. Fuel Cost Assumptions

Cost of U_3O_8, \$/lb	40, 100
Cost of U_3O_8 to UF_6 conversion, \$/kgU	4
Cost of separative work, \$/SWU	100
Tails assay, wt %	0.20
Fabrication cost, \$/kgU	
Small fuel assembly, blanket assembly	168
High-pressure case at 3100 psia	164.70
High-pressure case at 2450 psia	161.10
All other cases	160
Lump sum post-irradiation services, \$/kgHM	250
Carrying charge rate, %/yr	
Before and after irradiation	3.8375
During irradiation	7.675
Discount rate for levelizing costs, %/yr	3.8375
Electrical efficiency of plant, %	33.42

Table 2-2. Summary of Concept Ratings

Criteria	Concept I: High temperatures and pressures 3100 psi +50F	Concept I: 2450 psi +15F	Concept I: 2250 psi +15F	Concept II: Freq refueling if extra outage days =0	Concept II: = 11	Concept III: Coastdown beyond NASAP	Concept IV: Core periphery Steel/Zr	Concept IV: Be/graphite	Concept V: Radial blanket	Concept VI: Lower power density 15%	Concept VI: 30%	Concept VI: 45%	Concept VII: Small fuel assembly
Uranium Utilization/SWUs[a]													
30-yr rqmt, STU_3O_8/GWe	4608	4700	4680	4425	4273	4691[h]	4693	4597	4645	4617	4455	4321	4693-4741
Uranium savings, %	3.8	1.9	2.3	7.6	NA[g]	2.0	2.0	4.0	3.0	3.6	7.0	9.8	1.0-2.0
30-yr rqmt, 10^3 SWU/GWe	3212	3219	3206	2936	2829	3177[h]	3181	3096	3182	3087	2911	2748	3216-3242
SWU savings, %	1.7	1.5	1.9	10.1	NA[g]	2.8	2.6	5.2	2.6	5.5	10.9	15.9	0.8-2.0
Economics[b]													
Development cost, 10^6 $	35-50[f]	25-40	15-25	6-12	6-12	2-4	5-15	10-15	5-10	5-10	5-10	5-10	6-12
Capital cost, 10^6 $	35-50	15-25	3-5	1-5	1-5	2-5	3-6	9-14	3-5	7-10	15-20	25-30	2-5
Fuel cycle cost, m/kWe-h[c]													
@ $40/lb U_3O_8	6.59	6.68	6.65	6.24	6.24	6.74	6.67	6.56	6.67	6.74	6.68	6.63	6.70-6.76
@ $100/lb U_3O_8	10.35	10.50	10.46	9.83	9.80	10.60	10.49	10.28	10.47	10.61	10.52	10.44	10.53-10.63
Fuel cycle cost savings, %	3.2	1.8	2.2	8.0	8.3	0.9	1.9	3.7	2.0	0.9	1.7	2.5	0.6-1.5
Capacity factor, %[d]	Lower at higher pressures			75	72.3	74.26	Slightly lower		Slightly lower	Slightly higher			Sl. lower
Impact on construction, yr	1	0	0	0-1	0-1	0	0-1	0-1	0	0-1	0-1	0-1	0
Technical													
Feasibility	High pressure - very difficult			<11 d very difficult		Turbine-limited	Available technology		Avail. tech.	Available technology			Avail. tech.
Safety	Reduced margins			Minor impact		Minor impact	Minor impact		Minor impact	Improved margins			Minor impact
Operational													
Reliability	Less	Less	Less	Less	Less	Same	Slightly less		Sl. less	Higher	Higher	Higher	Less
Availability	Lower	Lower	Lower	Lower	Lower	Same	Slightly lower		Sl. lower	Slightly lower			Lower
Operability	Less	Less	Less	Less	Less	Same	Same	Same	Sl. less	Same	Same	Same	Less
Other Considerations													
Utility acceptance	Poor	Poor	Poor	Moderate		Good	Moderate	Moderate	Moderate	Moderate to good			Moderate
First commercial operation[e]	2002	1999	1996	1999-2002		1994[i]	1995-2000	2004	1999	1995	1995	1995	1999
Non-proliferation aspects	No impact			No impact		No impact	No impact	No impact	No impact	No impact			No impact
Compatibility with recycle	Possibly worse			Improved		Same	Same	Same	Same	Improved			Improved
Compatibility w/ other concepts: Backfittable[j]	Poss. conflict - high burnup			All	All	All	All	All	All	Poss. confl - hi burnup			All
Non-backfittable	All but some VI			All but VI,VII		All	All	All	All	All but II, some I			All but II
Retrofit potential	Poor	Poor	Poor	Good	Good	Good	Good for Zr/SS		Good with VI, VII	Poor	Poor	Poor	Good

[a] Values for NASAP Composite Improved PWR are 4789 STU_3O_8/GWe and 3267 × 10^3 SWU/GWe.
[b] July 1980 dollars.
[c] Values for NASAP Composite Improved PWR are 6.8 m/kWe-h at $40/lb and 10.7 m/kWe-h at $100/lb.
[d] Reference value is 75%.
[e] Reference date is 1994 if no delays for development or construction.
[f] Excluding development costs of an improved Zircaloy if required.
[g] Total energy extraction less than NASAP case; effective savings same as column to left.
[h] Total energy extraction less than NASAP case; savings due only to coastdown are 1.1% U_3O_8/1.5% SWUs.
[i] Without turbine redesign.
[j] NASAP Composite Improved PWR.

3. CONCEPT RATINGS

3.1. Case I — High Pressure and Temperature

3.1.1. Objective

The objective of this case is to improve fuel utilization through higher system efficiency obtained by increasing system operating pressure and temperature. Three cases were considered: Case A was for operation of a reactor at 3100 psia (which is approximately 100 psia below critical pressure) with primary coolant temperature increased by 50F. This results in a maximum cladding surface temperature of 700F. Case B was for operation at 2450 psia/ +15F, yielding a maximum cladding surface temperature of 665F. This pressure/temperature was selected because it appears to be a reasonable upper bound for acceptable corrosion and creep behavior of the Zircaloy materials currently in use. Case C is an alternate to Case B in which the system pressure remains at 2250 psia, but the outlet reactor temperature was allowed to increase 15F. This bounds what may be achievable through flow control, reduced peak-to-average power distribution in the core, or improved methods of thermal analysis.

3.1.2. Criterion A — Uranium Utilization

For a given fuel assembly lattice, increased temperatures require higher enrichments because of the reduced H/U ratio. Increased pressures require higher enrichments because of increased fuel cladding thickness. Thus, to improve uranium utilization by raising temperature and pressure, the plant's efficiency must increase faster than the uranium requirements.

Changes in fuel utilization and separative work relative to the NASAP Composite were estimated for the three cases described above under two assumptions:

use of the optimized fuel lattice for current design temperatures and use of a reoptimized lattice in which additional fuel reduction restores the H/U ratio to a level approximating that of the optimized lattice.

Case	30-Year Cumulative Requirements st U_3O_8/GWe	SWU/GWe, 10^3
NASAP Composite	4789	3267
Optimized lattice		
A +50F, 3100 psia	4807 (+0.4%)	3344 (+2.4%)
B +15F, 2450 psia	4759 (-0.6%)	3267 (0.0%)
C +15F, 2250 psia	4755 (-0.7%)	3263 (-0.1%)
Re-optimized lattice		
A +50F, 3100 psia	4608 (-3.8%)	3212 (-1.7%)
B +15F, 2450 psia	4700 (-1.9%)	3219 (-1.5%)
C +15F, 2250 psia	4680 (-2.3%)	3206 (-1.9%)

Essentially no improvement in fuel utilization is forecast for the cases using the lattice optimized for current temperatures. Gains of up to 4% are estimated for the cases in which further lattice optimization is considered. The latter estimates involve more uncertainty because two significant parameters are varying (temperature, geometry).

3.1.3. Criterion B — Economics

3.1.3.1. Capital

The wall thickness of the reactor vessel and primary system components will increase by the ratio of the pressures and mean diameters adjusted for allowable design stress at the various temperatures. To a first approximation, the weight will increase by the same factor for all primary components subjected to the higher pressures. No shop fabrication or material limits are anticipated.

The construction period is assumed to increase because of the larger amount of weld material that will be required during fabrication and erection of the unit, and second, because of the greater difficulty in handling the heavier components. This may also require increased crane capacity for the field fabricators.

Higher primary temperatures also permit higher steam pressures and, consequently, higher turbine efficiencies. Estimates of the increased efficiency for each case along with the estimated capital cost changes and impact on construction time follow:

Case A — Plant efficiency increased from 33.42 to 35.55. The weight of the reactor and primary system components is assumed to increase by approximately 50%, and it was also assumed that the cost of the components is proportional to their increased weight. This will result in an additional cost of some $35 to $50 million above current reactor systems. Turbine modifications that might be required at the higher system pressure and temperature have not been considered. It is anticipated that construction time will increase by one year.

Case B — Plant efficiency increased from 33.42 to 34.35. The weight multiplying factor, 1.15, will result in a component cost increase of $15 to $25 million. No increased fabrication span is anticipated.

Case C — Increased plant efficiency will be about the same as Case B unless a significant increase in pumping power is required. In this case, the components are relatively unchanged, but flow control, reduced power peaking in the core, or improved thermal analysis will be required. It is estimated that increased moderator temperature at the same system pressure would require an increase of approximately $3 to $5 million in reactor vessel and internals costs.

In all of the cases above, first-of-a-kind engineering costs will be high because of component redesign, licensing, and the need to establish new operational limits. Engineering costs have been included in the estimates above.

3.1.3.2. Fuel Assembly

Development of a re-optimized lattice fuel assembly will cost $5 to $10 million.

3.1.3.3. Fuel Cycle Costs

Levelized 30-year fuel costs were estimated for comparison with those of the NASAP Composite PWR, in July 1980 dollars.

	Fuel Costs, m/kWe-h @ U_3O_8 Price, \$/lb	
	\$40.00	\$100.00
NASAP Composite	6.8	10.7
Optimized lattice		
Case A	6.81	10.73
Case B	6.75	10.62
Case C	6.74	10.61
Re-optimized lattice		
Case A	6.59	10.35
Case B	6.68	10.50
Case C	6.65	10.46

In estimating Case A (+50F, 3100 psia), the cladding thickness was increased 35% to compensate for the higher pressure, but no allowance was made for degradation of creep and corrosion properties; it was assumed that an improved Zircaloy material could be developed to withstand the higher conditions. A 9% allowance in cladding thickness was used in Case B and none in Case C. Substitution of steel or Inconel would have such an adverse effect on uranium utilization that it was not considered.

3.1.3.4. Capacity Factor

To a first approximation, failures and maintenance costs might be assumed to be proportional to the pressure increase. In addition to increased maintenance problems, one may also anticipate that the availability factor will decrease during the early years of operation, probably as some fraction of the pressure ratio. Cladding failures may also be more prevalent, especially during early operation of the core until material properties and the fuel rod design have been optimized.

3.1.3.5. Development

One of the major impacts on cost and elapsed time to commercialization in Case A may be caused by the development of cladding material and fuel rods. It cannot be assumed that increased cladding thickness alone will be an acceptable solution; therefore, a new material development program may be required.

A major restriction in the material development is that the current practice of using commercial reactors as test beds for new fuel will not be possible at the higher temperatures and pressures, and there are too few high-pressure/high-temperature loops in operation today to provide meaningful statistical information on cladding operation.

A second major problem in the development of this concept is that many facilities for environmental tests may not be capable of operating at the 3100 psia and 700F range; this may involve major facility additions for CHF and loss-of-flow tests. If facilities must be designed and built before testing can commence, we would expect that delays of 5 to 8 years may be encountered before prototype designs would even begin. Development costs, excluding a new Zircaloy, could be $30 to $40 million depending on the adequacy of the facilities, principally those involved in CHF testing and loss-of-coolant accidents.

Design of a nuclear steam supply reactor system at the intermediate pressure of 2450 psia would be somewhat easier, but not necessarily much less expensive, depending on the availability of test facilities. Virtually all the first-of-a-kind engineering would have to be repeated, although in this instance it may be possible to extrapolate or use available test facilities. Licensing would be easier than for the higher pressure case, but it would nonetheless require complete reanalysis. Development costs may be in the range of $20 to $30 million. Flow control, fuel assembly lift, pump development, and fuel lattice optimization may cost from $10 to $15 million for Case C.

3.1.4. Criterion C — Technical

3.1.4.1. General

High pressure and high temperature require significant extension of the present PWR technology for component design, manufacture, and operation. All primary and secondary system components will require thicker walls, and their additional size and weight will make fabrication and handling more difficult. Higher temperatures and pressures will require, as a minimum, increased fuel cladding thickness and probably major development programs as well.

Higher system temperatures and pressures for Cases A and B will result in reduced system inventory and the need for additional major changes in component design. A larger pressurizer will be required to accommodate coolant shrinkage during cooldown. Higher pressures within the primary and secondary systems will require higher head pumps for coolant makeup and steam generator feedwater, changes in the pressure relief system design, and extensive modifications to handle higher pressure steam. These modifications may also require requalification of process variable measurement systems and the indicated detection range for their instrumentation.

The design complexity of the plant may be further increased because the temperature differential between the hot fluids of the plant primary and secondary systems may require water preheating to prevent excessive thermal stresses during cold water injection.

Increased stored energy in the reactor coolant may require a larger or stronger reactor cavity and reactor containment building design to withstand a possible 8% increase in containment pressure. These designs will need strengthened supports and restraints for RCS components. Consideration must be given to design basis accidents, especially the loss-of-coolant accident (LOCA), in determining overall system interactions and consequences. Important factors include steam generator heat removal, the effectiveness of RC pump coastdown on removal of stored energy from the fuel, strengthening of the reactor internals, and the adequacy of the core cooling system designs.

Case C would be simpler to implement than Cases A or B and would have its major impact on the balance-of-plant (BOP) design. Reactor coolant flow, however, might increase as much as 40% unless improved power peaking, flow control, or improved thermal analysis are used to obtain the higher performance. Increases in flow beyond about 7% would require a major extension of the current state-of-the-art reactor coolant (RC) pump design. A major design and test effort would be needed to requalify the pump, the pump seal, and the primary system loop resistance requirements. If flow control is required, orificing may be preferred over canned fuel assemblies. B&W used cans until 1967 but found them to be mechanically undesirable. Orificing will work but extensive testing will be required.

Re-optimization of the fuel assembly lattice to compensate for the lower water density will reduce heat transfer surface area. In view of the higher temperatures, an increase in the number of pins may be required, i.e., 18 by 18 or 19 by 19.

3.1.4.2. Safety

Operation and transients at the higher temperatures and pressures of Cases A and B will result in more severe inservice conditions for primary and secondary system components, particularly higher thermal stresses at points of cold water injection. The normal operating and accident environment for these components and their instrumentation will be more severe.

Primary barriers to radioactivity release, principally fuel cladding and steam generator tubes, will become more important at higher temperatures and pressures. Relief valves will be required to function under more severe operating environments. A greater burden is placed on the reactor containment building to accept an increase of up to 8% in stored energy rejection to the coolant.

Accidents at higher pressures and temperatures will probably become more closely tied to the overall plant design basis. In present plant design practice, the LOCA tends to have the most impact on plant design and provides ample margin to handle the system and the consequential needs of other events. The overcooling events, including the steam line break, are typical of how this design approach may change. Higher operating temperatures will provide more negative moderator temperature coefficients toward the end of a cycle than the reference design. The more negative coefficient would make overcooling events a more prevalent factor in increasing the frequency of reactor trips and reductions in core shutdown margin.

Operation at the slightly higher operating temperatures for Case C also tends to be less safe but much safer than Cases A and B since a smaller overall systems design impact is involved. For this case, the stored energy in the reactor coolant is increased less than 2% and should thus have little influence on transient and accident consequences. A larger differential may exist between cold leg temperatures at operation and shutdown, which may merit further consideration of steam generator and pressurizer operating conditions.

3.1.4.3. Potential for Retrofit

Cases A and B require such a substantial change in system component design that it would not be economically advantageous to attempt to design into the plant the necessary margins for possible future upgrading of operating conditions. Case C would lend itself more readily to retrofit consideration. This case would still require some major changes in system component design, including the pressurizer size and the RC pump flow capacity.

3.1.5. Criterion D — Operational

Operation at higher temperatures and pressures will degrade plant reliability, availability, and operability because of the more severe environment for the primary system components. Component failure rates may increase, particularly in such vulnerable areas as RC pump seals, valve internals, rotating parts, fuel cladding, and steam generator tubes. Failure rates may also be accelerated because of thermal stresses, which take on greater significance at points of cold water injection into the primary and secondary systems.

Higher operating temperatures and pressures may also increase leakage rates, resulting in higher radiation levels within the reactor containment building and designated maintenance areas.

The formation and deposition of corrosion products within reactor coolant components may be increased by operation at higher temperatures and pressures and lead to increases in radiation levels inside and outside the primary system. Consequently, the number of staff personnel involved in maintenance activities would be increased.

The principal plant control and pressure relief systems must be adapted to higher temperature and pressure operation. Maneuverability may require additional control measures to compensate for what could be significantly more negative moderator temperature coefficients toward the end of the cycle. The complexity of these control measures will depend on the stability of the plant during anticipated transient conditions and the need to reduce the frequency of reactor trip and subsequent reductions in plant availability.

Case C will have the least impact on system design and operation since only reactor temperature is increased; however, primary coolant flow may increase. If it is required, the increased coolant flow will require major development of RC pump, seal, and loop components. The flow increase will also affect the holddown forces required for the fuel assemblies. System transients will be affected because of enhancement of energy transfer to the steam generators, the fluid transport rate to primary system breaks, and the severity of pump cavitation effects.

3.1.6. Criterion E — Other Considerations

3.1.6.1. Utility Acceptance

Utility acceptance is likely to be very poor because:

1. The high capital costs and anticipated higher maintenance and outage costs are barely justified by the savings achievable through improved fuel utilization and fuel cost unless very high relative uranium costs are assumed.

2. Licensing requirements are proliferating with current systems and would begin anew if major changes were made in component design or operation. A major cause for concern would be a new cladding material.
3. Utilities encountered much difficulty in "shaking down" systems regarded as standards today and will be reluctant to undertake similar risks.
4. Safety margins will be perceived as reduced at elevated temperatures and higher pressures.
5. Increased maintenance and the prospect of reduced availability are a probable result of higher pressures.
6. The need for extensive development will delay commercialization for too long to be attractive.

3.1.6.2. Commercial Operation

Case A - 2002

Case B - 1999

Case C - 1996

3.1.6.3. Non-Proliferation Compatibility

High temperatures and pressures present no added non-proliferation concerns.

3.1.6.4. Viability With Recycle

Plutonium recycle tends to increase power peaking. This could present a problem in a high-temperature design based on minimized power peaking.

3.1.6.5. Interaction With Other Concepts

As discussed previously, high temperatures may require re-optimization of the fuel lattice in order to achieve gains in fuel utilization.

Coastdown capability may be improved because of a more negative moderator coefficient. Re-optimization of the lattice will diminish this improvement.

Lower power density will tend to complement higher temperatures incore, but will result in substantial increases in the cost of the pressure vessel.

High temperatures and pressures will increase the difficulty of maintaining fuel cladding integrity at the higher exposures included in the NASAP Composite Improved PWR.

3.2. Case II — Frequent Refueling

3.2.1. Objective

The objective of this concept is to improve uranium utilization through more frequent refueling with smaller batches of fuel. Classical fuel cycle theory indicates that higher values of Z, the number of refueling batches, yield greater uranium utilizations for the same discharge burnup. One concept envisions a six-month cycle with a very short refueling-only outage (10 days or less) alternated with an annual normal refueling and maintenance outage. Another considers two medium-length outages at six-month intervals. In either case, loss of more than 10 additional days per year is expected to involve replacement power costs that exceed the fuel cycle gains. Both concepts are based on cold shutdown refueling.

3.2.2. Criterion A — Uranium Utilization

The NASAP Composite PWR is based on annual refueling with replacement of one fifth of the core each cycle (Z = 5). Development of quick refueling would permit a six-month cycle with replacement of one tenth of the core per cycle (Z = 10). Estimated changes are compared for two six-month cases: one in which the capacity factor remains at 75% (no increased outage time) and the other in which 11 additional refueling days* are required per year and the capacity factor falls to 72.3%. Total energy production in the last case is 3.6% lower than the two cases at 75% capacity factor.

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*Actual estimated minimum theoretical refueling time is 11-12 days.

Case	GWe-yrs	Capacity factor	Cumulative 30-yr requirements	
			st U_3O_8/GWe	SWU/GWe, 10^3
NASAP Composite, annual cycle	22.5	0.750	4789	3267
6-mo. cycle, no outage increase	22.5	0.750	4425 (-7.6%)	2936 (-10.1%)
6-mo. cycle, 11 extra outage days	21.7	0.723	4273 (-10.8%)	2829 (-13.4%)

A substantial savings in uranium utilization is associated with frequent refueling. The challenge is to develop a quick refueling system that minimizes capacity factor loss enough that replacement power costs do not consume the gains.

3.2.3. Criterion B — Economics

3.2.3.1. Capital

Incorporating near-term procedures and equipment could conceivably reduce the theoretical outage from approximately 22 to 11 or 12 days with the reference fuel cycle (Z = 10). Costs for capital equipment, training, etc. would be in the neighborhood of $1 to $2 million.

The achievement of 10-day refueling on a consistent and confident basis would be a major redesign task involving completely new concepts. The costs for this would probably be in the range of $5 million.

3.2.3.2. Fuel Cycle Costs

Levelized 30-year fuel cycle costs were estimated for the two six-month cycles and compared to those of the NASAP Composite PWR in July 1980 dollars. The break-even replacement power cost was also estimated for the case in which 11 additional refueling days per year are required.

Case	Fuel cost, M/kWe-h @ U_3O_8 price, $/lb @ $40	@ $100
NASAP Composite PWR	6.8	10.7
6-mo. cycle, no outage increase	6.26	9.83
6-mo. cycle, 11 extra outage days	6.24	9.80
Break-even replacement power cost[a]	21.8	34.8

[a]Excluding additional capital costs for rapid refueling.

Eleven additional outage days present a marginal situation where break-even replacement power costs are in the range of actual replacement power costs.

3.2.3.3. Capacity Factor

The addition of an 11-day refueling outage will reduce the capacity factor from 75 to 72.3% unless the main refueling outage can be shortened or maintenance outages outside the main outage can be scheduled coincidentally. As discussed in the operational section, our service records are not sufficiently detailed to resolve this issue. Comparisons of current actual refueling times to theoretical times strongly suggest that multiple refuelings in the same year will reduce capacity factor. Fuel utilization and fuel cost comparisons were made for both 75 and 72.3% capacity factors.

3.2.3.4. Development Costs

Redesign of major components, modifications to the containment building, and demonstration of equipment necessary to achieve 10-day refueling might extend the date of initial commercialization by 3 to 5 years and cost from $6 to $12 million. Subsequent units should not have longer construction times.

3.2.4. Criterion C — Technical

3.2.4.1. Feasibility

Limiting cold-shutdown refueling outage periods to 10 days will be difficult and will require extensive NSS and fuel handling equipment redesign. Multiple fuel handling capability may also be required. The complexity of the anticipated design changes will probably increase the frequency of component failures during handling of NSS equipment and fuel. The shorter outage period will accordingly require improvements in activity planning, equipment maintenance, and the availability of a wider variety of spare parts onsite. More attention must be given to maintaining a well-trained and efficient refueling team.

Refueling outages of 14-15 days duration appear to be achievable for a three- to five-batch core shuffle. This could probably be reduced to 11-12 days with the adoption of a "salt and pepper" (minimum shuffle) replacement of 1/10 of the core per outage as required by the NASAP Composite PWR. These estimates assume that refueling activities, irrespective of reactor shutdown and heatup, can be accomplished more expediently with the improvements listed below. (These items relate specifically to the B&W design; similar improvements would apply to other PWR designs.)

3.2.4.2. Design

1. Consolidation of RV insulation pieces.
2. Use of bellows-type permanent seal plate.
3. Improved method of parking lead screws.
4. Lift rigging attached to top of service structure.
5. Improved guide studs.
6. Use of O-rings and powered wrenches on transfer tube cover removal/replacement.
7. Improved indexing fixture.
8. Use of a stand to permit rigging to remain assembled for plenum lift.
9. Use of a dual-function refueling mast or a two-mast bridge and fuel transfer tube.
10. Improved CRA handling tool.

3.2.4.3. Additions

1. Use of three stud tensioners and additional work crew.
2. Use of premeasured chromate containers.

3.2.4.4. Procedures

1. Indexing fixture removal.
2. APSR coupling.

3.2.4.5. Training

1. Plenum removal/storage.
2. Indexing fixture removal.
3. RV head repositioning.
4. Guide stud removal.

The improvement in refueling activities has been established through changes in the operation and design of present fuel handling equipment rather than attempting to qualify uncertainties regarding the impact of major design changes in the NSS hardware. The recommended design improvements will make the present equipment slightly more complex. These changes are within the current technology for materials and manufacture and should require no demonstration.

Other more complex design changes, which further development and demonstration may prove supportive of the more frequent refueling effort, include (1) rotating bridge cable tray, (2) multiple stud removal/closure, and (3) an integral reactor vessel head configuration.

3.2.4.6. Safety

Frequent refueling should have no impact on the safety of plant operation unless modifications are undertaken to reactor vessel internals, fuel, or control elements. Core performance parameters should be bounded within the reference core design.

An increase would be anticipated in the number of required refueling operations and the complexity of fuel handling techniques and equipment employed. Statistically, this should lessen the overall safety of the plant since additional opportunities will be available for personnel injury and schedule delays leading to increased radiation exposure. Most equipment modifications should be within the current technology and capable of development without major impact on the safety of mechanical operation. The more complex major design changes will require demonstration of reliability.

The increased frequency of fuel and control element handling will increase the probability of fuel damage resulting in the release of fission product activity. The consequence of such an activity release will be bounded by those for the fuel handling accident for the reference core. If multiple fuel handling becomes a consideration, then the activity release and radiation exposure consequence will be bounded in proportion to the number of assemblies being handled.

3.2.4.7. Potential for Retrofit

Many fuel handling components can be redesigned for retrofit after a plant is built. Special consideration may be necessary for space allowance requirements if equipment size increases significantly or multiple fuel handling capabilities become desirable. Certain components that might require major changes, including the reactor vessel head, equipment attached to the RV head, and fuel handling equipment that interfaces with the RV head and its appurtenances, are normally permanently installed features of the plant. It would be difficult, if not impractical, to allow for retrofit of these components.

3.2.5. Criterion D — Operational

3.2.5.1. Plant Reliability

Equipment durability is the most important reliability concern during refueling outages. Frequent refueling could well double the work load placed on fuel handling equipment as well as greatly increase the number of individual operations that must be performed. The reactor vessel head and other associated hardware would also be subject to more frequent disassembly, creating additional opportunities for operational and mechanical problems. Some simplification of this process may be possible through the use of multiple fuel handling capability, integration of the reactor vessel head and associated hardware, and further modification of present fuel handling equipment and procedures. This will, however, contribute to the overall complexity of the operation and requires that each change be examined for its potential contribution to plant maintenance and inspection.

3.2.5.2. Plant Availability

Semi-annual refueling will significantly lower the plant availability and capacity factor if refueling represents the critical path for the outages. If maintenance and other BOP concerns control the critical path, as it appears they now do for the annual outage, then frequent refueling may be possible with little impact on plant availability through better planning to capitalize on anticipated long unscheduled outages or to allow for multiple planned maintenance periods.

Consideration was given to the potential for refueling in conjunction with more frequent scheduled maintenance periods. The service records for several PWR plants were reviewed. This review supports the expectancy that utilities are quite consistent in scheduling maintenance and repair operations, especially those of long duration, coincident with annual refueling activities. Scheduled outages between refueling outages are rare, regardless of duration. Forced outages longer than 10 days tend to fall in the category of one-time major repair or compliance with regulatory requirements. Maintenance operations during refueling normally extend overall outage duration, but the service records provide little detail as to the frequency, variety, and extent of individual maintenance programs. Utility maintenance planning records may prove to be a better source of establishing the potential for separating the scheduled maintenance programs into two groups, each of which can accommodate a 10-day refueling period without significantly increasing total annual outage time.

A typical critical path refueling outage is now estimated to take about 21.5 days as described below:

Critical path item	Outage duration, days Typical	Improved
1. Reactor shutdown, preparation of containment for entry	1.3	1.3
2. Equipment disassembly, preparation of core for fuel shuffle	3.5	1.6
3. Fuel assembly, control element repositioning	6.0	1.2-4.1
4. Equipment assembly, reactor vessel latchup	5.5	2.3
5. Reactor cavity area cleanup	0.5	0.2
6. Systems alignment, checkout	3.0	3.0
7. RCS refill, heatup	1.7	1.7
Total	21.5	11.3-14.2

This critical path effort can be reduced by 7 to 10 days (depending on the size of the fuel batch) through minor improvements in fuel handling techniques and equipment. Critical path items 1, 6, and 7 are governed by the plant Technical Specifications and are thus more difficult to influence. Major modifications to NSS hardware that integrate reactor vessel head functions are also possible and may provide further improvement in the outage schedule. Such changes are too difficult to assess due to their general complexity and the need for additional investigation of the design and reliability of these components.

3.2.5.3. Operability

The operability of frequent refueling is most affected by the extent of personnel radiation exposure compared to the typical annual refueling period of 21.5 days. Based on the projected optimum outage duration with some modification in equipment and fuel handling techniques, semi-annual refueling is anticipated to require at least 23-28 days, i.e., 1-7 additional days of radiation exposure for the plant staff. The need for equipment maintenance during the performance of refueling tasks will contribute further to the accumulated exposure. Emphasis should be placed not only on mechanical and operational improvements but their associated impact on equipment reliability.

Since radiation exposure is primarily dependent on the duration of the refueling outage, further optimization in outage schedule is possible as demonstrated by refueling experience through the following:

- Detailed front-end planning of refueling activities and fuel handling equipment checkout.
- Task performance by well-trained and highly motivated service crews.
- Dedication of engineering support services to equipment problems during refueling.

3.2.6. Criterion E — Other Considerations

3.2.6.1. Utility Acceptance

Utilities are converting to 18-month fuel cycles, even though the fuel costs are higher than for shorter cycles, for two principal reasons: increased plant availability and decreased licensing problems. Frequent refueling is in opposition to both of these. Consequently, it may not find ready acceptance without demonstrated reliability and repeatability plus assurance that licensing for interim refueling will be simplified. (NRC cooperation in licensing is anticipated.) Increased radiation exposure of personnel is also an inherent penalty and a serious one for utilities.

Frequent refueling and fast refueling are inseparable, and it can be anticipated that utilities will want demonstrated performance before investing in the equipment, and they may not be willing to attempt interim refueling until after several normal cycles in their own plants. If this time is added to the development time, 5 to 8 years may elapse before the practice becomes operational on the first unit.

3.2.6.2. Commercial Operation

First commercial operation is projected for 1999-2002.

3.2.6.3. Non-Proliferation Compatibility

Frequent refueling adds no new non-proliferation concerns.

3.2.6.4. Compatibility With Recycle

Frequent refueling is compatible with recycle and is perhaps advantageous in that recycled material may enter the core in 18 months instead of 2 years after initial discharge.

3.2.6.5. Compatibility With Other Concepts

Frequent refueling is compatible with all of the backfittable and non-backfittable concepts except possibly the small fuel assembly and a substantial power density reduction. Unless an essentially shuffle-free fuel management plan is developed, increased fuel handling time will present a problem in both cases. Optimization of burnup should be enhanced by frequent refueling because of the smaller burnup increment received in each cycle.

3.3. Case III — Coastdown

3.3.1. Objective

The objective of coastdown is to improve fuel utilization by continuing operation at declining power beyond the depletion of reactivity, utilizing reactivity available from the Doppler, xenon, and moderator temperature reduction. Moderator temperature reduction can more than double the amount of additional energy produced during coastdown to a given terminal power. Techniques used to maximize coastdown capability will differ somewhat among the various PWR designs, but they can be divided into two general categories: those utilizing current turbine designs and those requiring larger or redesigned turbines. The discussion that follows is based on the B&W standard design, which includes once-through steam generators producing superheated steam and the integrated control system.

Potential coastdown capability with the current turbine design was estimated and the associated uranium savings compared to the savings resulting from the preplanned coastdown included in the NASAP Composite PWR. The difference approximates the remaining improvement available for this system beyond that included in the NASAP design. The use of a larger or redesigned turbine to accommodate greater steam flow at reduced temperatures and pressures may produce additional gains, but estimation of these savings was beyond the scope of this study.

3.3.2. Criterion A — Fuel Utilization

Coastdown of a single cycle to 75% power (electrical), using moderator temperature, Doppler, and xenon reactivities, is estimated to produce 50 EFPD (electrical) over a period of 57-58 days. Performed on a repetitive basis, the net gain is proportional to 2/(Z+1) where Z is the number of batches. Thus, with the NASAP Composite (Z=5), each cycle would receive a net increase of approximately 16-17 EFPD. The utility has several options, which include accepting the longer cycles or reducing the feed enrichment enough to retain the annual cycle. In any case, the coastdown will result in both uranium utilization and separative work savings at the expense of a reduced capacity factor. In order to remain consistent with the NASAP Composite, which uses annual refueling, uranium and separative work savings were first estimated for the option in which enrichment is reduced to retain the annual cycle. Since this case produces less GWe-years in 30 years than the base case, data are also provided for a case in which total energy production is identical to that of the NASAP Composite, although it is produced over a period of 30.2 years.

Case	GWe-years	Capacity factor	Cumulative 30-yr requirements: st U_3O_8/GWe	SWU/GWe, 10^3
NASAP Composite PWR (annual cycle)	22.5	0.7500	4789	3267
Annual cycle using maximum coastdown	22.3	0.7426	4691	3177
Non-annual cycle, energy = NASAP Composite	22.5	0.7445	4736	3219

For identical energy production, maximum coastdown to 75% power is estimated to yield additional uranium savings of 1.1% beyond the 4.8% included in the NASAP Composite PWR, for a total of 5.9%.

Comparisons of the annual cycle with the NASAP Composite cycle should reflect the purchase of replacement power for 2.7 additional days each year due to the reduced capacity factor.

3.3.3. Criterion B — Economics

Capital and development costs discussed in this section are total costs to achieve the 5.9% savings in uranium, not the incremental cost to achieve the 1.1% improvement over the NASAP base.

3.3.3.1. Capital

Primary system instrumentation and control component cost increases should be in the range of $2-5 million, depending on the desired temperature reduction capability. A larger pressurizer will be required and, at significantly lower temperatures, LOCA forces on the internals may dictate a more rugged design.

3.3.3.2. Fuel Cycle Costs

Levelized 30-year fuel cycle costs were calculated for the annual cycle with maximum coastdown capability and compared with those of the NASAP Composite PWR, in July 1980 dollars. Since the annual coastdown cycle requires purchase of replacement power for 2.7 EFPD/year, the breakeven replacement power cost was also calculated for each uranium value. The optimum coast-down period may vary substantially among utilities.

	Fuel cost, m/kWe-h @ U_3O_8 price, $/lb	
	@ $40	@ $100
NASAP Composite	6.8	10.7
Annual cycle with maximum coastdown	6.74	10.60
Break-even replacement power cost	12.8	20.7

Excluding replacement power costs, fuel cost savings of approximately 1% result from the increased coastdown capability.

3.3.3.3. Capacity Factor

The capacity factor would be expected to drop from 75 to 74.26% because of additional operation at reduced power for approximately 12 days per cycle.

3.3.3.4. Development Costs

Development costs should be low, probably $2 to $4 million, depending on the amount of FOAK engineering and testing of primary components, controls, and instrumentation required to justify the lower moderator temperatures.

3.3.3.5. Impact on Construction Times

Once the concept has been developed, no impact on construction time is anticipated.

3.3.4. Criterion C — Technical

Most reactors are loaded with sufficient reactivity to provide the desired energy extraction under full-power conditions. Continued operation beyond depletion of reactivity (DOR) is possible through reductions in power, which makes available part of the reactivity held by the Doppler and xenon and the moderator temperature, which utilizes the PWR's negative moderator coefficient. Maximum power during a coastdown on Doppler and xenon is reactivity-limited, whereas when moderator temperature reduction is used, it is generally turbine-limited. Development of advanced coastdown capability would center on the changes necessary to accommodate the required moderator temperature reduction and to maximize the plant's electrical output at reduced moderator temperatures. The concept is technically feasible and has been demonstrated. The coastdown process following depletion of reactivity is expected to pass through three stages, during which moderator temperature is being reduced gradually to provide additional reactivity:

- Reduction of superheat to the minimum value required to prevent moisture carryover while maintaining reactor power and steam pressure.
- Reduction of steam pressure until turbine valves are wide open (VWO) while maintaining reactor power and minimum superheat. Feedwater flow and temperature are adjusted for optimum performance.
- Reduction of reactor power and steam pressure at VWO conditions while maintaining minimum superheat. Feedwater flow and temperature are continually adjusted for optimum conditions.

3.3.4.1. Feasibility

The length of operating cycles can be effectively extended beyond DOR through reduction in moderator temperature. Typically, about 0.3% in moderator temperature reactivity feedback is available for each degree F decrease in moderator temperature. A more negative EOC moderator temperature coefficient could be achieved by raising T_{avg} at full power. This would provide additional reactivity for use in coastdown but would also cause a loss of safe shutdown margin following such overcooling events as a steam line break. Other constraints on the extent of coolant temperature reduction must also be considered.

Maintenance of full-power operation while reducing moderator temperature is currently turbine-limited. A decrease in moderator temperature will require a corresponding decrease in secondary steam pressure to maintain superheat. This requires a steam volume flow increase in order to retain full-power output from the turbine. The steam flow capacity of current turbines in operation limits moderator temperature reduction to approximately 6F without a reduction in power. The technology is available, including fossil-fired plant turbine applications, to permit oversizing of turbines for greater steam capacities at low pressures. A more extensive redesign of the turbine may also permit greater steam bypass control to the low-pressure turbine as steam pressure decreases. Turbines with dual admission setpoint capability permit a split of about 15% in steam control. This could be used to approximately triple the VWO capacity of the turbine at low pressure. It is important that turbine efficiency at full power not be compromised significantly to gain coastdown capability.

Special control measures may be necessary to avoid steam generator moisture carryover to the turbine. The amount of reduction in secondary steam pressure to prevent moisture carryover is a function of the reactor power, coolant temperature, and steam generator fouling.

The present range of temperature instrumentation is 520-620F. This will limit coverage of decreases in average coolant temperature to about 525F. Cold leg temperatures may change more rapidly in a coastdown operation than with present average coolant temperature control. This may require changes in nuclear instrumentation power calibration frequency and technique.

Decreasing feedwater temperature provides an additional means of reducing coolant temperature. Feedwater temperature reductions to 300F could be used during coastdown. Below this temperature, thermal stress limits on the current feedwater nozzle design are approached.

3.3.4.2. Safety

The decreasing coolant temperature and reactor power conditions associated with coastdown would be expected to make the reference plant safer for most transient and accident conditions since DNBR and linear heat rate margins are improved. LOCA loads on the reactor vessel internals, however, may limit the extent of the cold leg temperature decrease allowed in present designs. Significant coastdown capability will probably require redesign of the reactor vessel internals. LOCA loads will increase roughly in proportion to the saturation pressure decrease corresponding to the required decrease in cold leg temperature. A system temperature decrease of about 40F would be required to provide enough reactivity for the coastdown to 75% FP. This reduction in cold leg temperature would result in an increase of about 25% in LOCA loads on the internals beyond the present design.

3.3.4.3. Potential for Retrofit

With consideration of turbine capacity and reactor vessel internals LOCA loads, coastdown operation could be retrofitted into plants. Major hardware changes will be necessary to optimize the extent of coolant temperature reduction and the power operation desired from the operating cycle.

3.3.5. Criterion D — Operational

3.3.5.1. Plant Reliability

The turbine and feedwater control systems and their associated instrumentation should withstand the extended service required for coastdown. These are highly active systems during normal operation. Present maintenance and inspection procedures for these systems will probably ensure that their response during coastdown is just as reliable. Further optimization of plant coastdown capability may require major turbine and RV internals modification. The complexity of these design changes could reduce their reliability.

Steam velocity and quality place the most severe conditions on plant reliability during coastdown. Special control measures may be necessary to prevent steam generator overfill and excessive low quality steam flow to the turbine as steam generator pressure is reduced during coastdown. This condition will be further influenced by a decrease in feedwater temperature, which would tend to reduce the requirement for feedwater inventory within the steam generator.

3.3.5.2. Plant Availability

Preplanned coastdown does not increase availability. Coastdown beyond the scheduled end of cycle provides an increase in availability at the expense of reduced power capability and capacity factor. Inability to readily recover from a trip during coastdown may adversely affect availability.

Turbine optimization without major redesign may permit operation for up to 15 EFPD beyond EOC without power reduction. Further improvements in turbine capacities at lower steam pressures will be limited by the design capabilities of the turbine manufacturer and the associated cost-benefit relationship.

3.3.5.3. Operability

Coastdown introduces additional complexities in plant control and operational procedures. A special unit tracking situation would soon be required where the turbine throttle valves follow steam pressure and the operator controls the average coolant temperature and feedwater conditions. The operator must also monitor steam generator level and use other means of assessing steam quality to prevent damage to the turbine and other secondary plant equipment. This mode of operation would be amenable to automation and would probably require it to be administered effectively. Analysis will also be needed to assess the consequences of upset and anticipated transient conditions should they occur during coastdown.

3.3.6. Criterion E — Other Considerations

3.3.6.1. Utility Acceptance

Coastdown capability provides the potential for fuel cycle flexibility and power cost savings. Thus, it should be well received by utilities but not necessarily used for the repetitive preplanned coastdowns that produce maximum uranium utilization gains, particularly if replacement power costs are high. In addition, the increased flexibility is lost if preplanned coastdowns are used.

The cost of implementing coastdown capability should also be well received by utility rate commissions since it produces power cost savings as well as the ability to sustain power production during periods of power shortage.

3.3.6.2. Non-proliferation Compatibility

Coastdown presents no added non-proliferation concerns.

3.3.6.3. Compatibility With Recycle

Coastdown is fully compatible with recycle if national policy should change.

3.3.6.4. Compatibility With Other Concepts

Coastdown is compatible with all backfittable and non-backfittable concepts although savings will be 45% smaller with 6-month refueling than with annual refueling because of the higher Z. It may be enhanced by the more negative moderator coefficient associated with higher temperatures and pressures, although reoptimization of the fuel lattice and turbine would tend to offset this effect. The EOC moderator coefficient for the NASAP Composite PWR may also be more negative than that of current designs, thus providing increased capability.

3.3.6.5. Commercial Operation Date

Excluding major turbine redesign, commercial operation is projected for 1994.

3.4. Case IV — Core Periphery Modifications

3.4.1. Objective

The objective of this concept is to enhance uranium utilization by making certain changes around the periphery of a PWR core. These changes, which may involve the configuration and/or materials, basically aim at reducing the radial leakage of neutrons from the core. Specific variations of this concept are (1) changing the thickness of the peripheral water gap and/or the core baffle (shroud), (2) replacing the present shroud material with a zirconium-base alloy, and (3) adding another reflector material such as beryllium or graphite. These concepts are discussed in the following sections.

3.4.2. Criterion A — Uranium Utilization

Improved radial reflection enhances uranium utilization by reducing the number of neutrons lost by leakage from the core. Increasing the water reflector thickness or changing the shroud material from stainless steel to Zircaloy primarily reduces the leakage of thermal neutrons. On the other hand, increasing the stainless steel shroud thickness primarily reduces the leakage of fast neutrons.

An even more drastic change would be the addition of a beryllium or a graphite reflector. Both of these materials are good reflectors, and they also have considerably smaller neutron absorption cross sections than the conventional borated water/steel reflector. Parasitic absorption in the reflector would be reduced, resulting in more neutrons being reflected back into the core improving neutron economy. Greater thicknesses are required for effective reflectors made of beryllium or graphite.

Babcock & Wilcox has not calculated the uranium utilization improvements achievable with such reflector region modifications. However, others have recently performed survey calculations assessing these effects and concluded that modest improvements were possible by implementing these kinds of changes. For example, Decher and Shapiro report that increasing the thickness of the peripheral water gap from 0.15 to 1 inch increases end-of-cycle reactivity by 0.14%, which is equivalent to an improvement of about 0.4% in uranium utilization.[2] The same report indicates that increasing the stainless steel shroud thickness from 1 to 4 inches (while maintaining the peripheral water gap thickness at 0.15 inch) leads to an increase of 0.8% in EOC reactivity and a uranium utilization improvement of about 2%.

Fujita, Driscoll, and Lanning conducted studies on a beryllium oxide (BeO) reflector.[3] They conclude that the use of such a reflector can reduce uranium ore requirements by as much as 5%.

Dabby has reported analyses that show that changing the core shroud material from stainless steel to Zircaloy improves uranium utilization by 1 to 2%.[4] A private communication from D. Newman of Battelle-PNL confirms this estimate.

The calculated effects of improved reflectors depend heavily on the models and methods used for the nuclear analysis. The benefits of improved reflectors are also dependent on the incore fuel management scheme assumed. For example, the uranium utilization improvement of a modified reflector would be expected to be somewhat smaller in a reactor employing a low-leakage shuffle scheme compared to one using a conventional out-in scheme. Similarly, the uranium utilization improvement gained from an improved reflector would depend on the presence or absence of a radial blanket. With a blanket the improvement would be smaller by about a factor of 2 than without a blanket.

The scope of this study does not permit a detailed quantitative assessment of these factors. However, it can be generally concluded that modest uranium utilization improvements (i.e., 2%) are probably achievable with some combination of Zircaloy, steel, and water; and that more substantial gains (i.e., 4-5%) may be possible with graphite or beryllium. The effect of such gains on cumulative requirements would be as follows:

	30-yr cumulative requirements	
	st U_3O_8/GWe	SWU/GWe, 10^3
NASAP Composite PWR	4789	3267
ZR/steel/water refl	4693 (-2.0%)	3181 (-2.6%)
Be/BeO_2, graphite refl	4597 (-4.0%)	3096 (-5.2%)

3.4.3. Criterion B — Economics

3.4.3.1. Capital

Capital cost increases will be encountered because of a larger reactor vessel, a more complicated internals design, the reflector itself, and additional handling equipment. These will be a function of the reflector material, desired effectiveness, and geometry chosen. The pressure vessel OD may increase by 4 to 8 inches for a Zr/steel/water reflector and by 12 to 30 inches for a beryllium or graphite reflector.

Capital cost increases for the various types of reflectors are estimated to be in the following ranges:

	Capital cost, $ million	
	Initial	Replacement
Zr/steel/water	3-6	0.5-1
Graphite	9-11	0.5-1
Beryllium	11-14	6-8

The impact on construction time should be minimal because reflector components will be shop-fabricated.

3.4.3.2. Fuel Costs

Levelized 30-year fuel costs were estimated on the assumptions that reactivity gains equivalent to a 2% uranium utilization improvement can be achieved with a combination of Zircaloy, steel and water; and that gains equivalent to 4% may be achievable with beryllium or graphite reflectors.

	Fuel cost, m/kWe-h @ U_3O_8 price, $/lb	
	@ $40.00	@ $100.00
NASAP Composite PWR	6.8	10.7
Zr/steel/water refl	6.67	10.49
Be/BeO_2, graphite refl	6.56	10.28

3.4.3.3. Capacity Factor

A more complex design or the use of exotic materials will probably require more inservice inspection initially. This could be a significant factor with regard to capacity factor in the early years.

3.4.3.4. Development

Development will be required to determine material performance as a function of lifetime, cladding necessity and method, reflector geometry, and cooling requirements. An increase in reflector importance and the possible use of two moderators in the same core will require considerable physics model development and physics calculations. Some experimental work may be required to establish form factors and confirm the adequacy of thermal and physics analytical methods. Out-of-core instrumentation will also be affected because of increased reflection.

Development costs are expected to be in the range of $5 to $15 million depending on the number of combinations and variables investigated. Approximately 5 years will be required before the final design can be initiated.

3.4.4. Criterion C — Technical

Two of the several concepts for improving the reflector are considered most practical and promising: (1) increasing the thickness of the stainless steel core baffle and (2) replacing the present stainless steel baffle material with Zircaloy. The first of these concepts enhances uranium utilization by improving the reflection of fast neutrons, whereas the second concept improves the reflection of thermal neutrons. Other concepts reviewed but not considered practical are discussed briefly below.

Beryllium and beryllium oxide, which have also been suggested as possible reflector materials, are expensive, difficult to fabricate, and highly toxic. An effective reflector constructed of either of these materials would have to be considerably thicker than the currently employed steel/water region, requiring an increase of perhaps 12 to 15 inches in pressure vessel OD. Although beryllium reflectors were installed in some early research reactors (i.e., MTR and ATR), its application in power reactors would be severely restricted because of its high sensitivity to radiation damage (which causes cracking), its tendency to oxidize at 600-650F, its galvanic action with other metals, and its low resistance to thermal shock. In addition, its toxicity, especially during machining, is such as to have a severe impact on fabrication costs. Beryllium oxide is probably the preferred form because of its stability in the operating environment. Cladding may be required, in which case concentric-clad cylinders or sandwiched slabs are conceivable. Pellets enclosed in rods may also be acceptable and will facilitate cooling if the additional thickness can be tolerated.

Graphite is non-toxic, easier to fabricate than beryllium, and should perform satisfactorily in water at PWR temperatures. Tests have shown that graphite can be used under these conditions without cladding. Graphite, however, is not as good a moderator as water or beryllium, and thus would require an even thicker reflector region and a correspondingly larger increase of perhaps 24 to 30 inches in pressure vessel OD. At elevated temperatures, e.g., 1500F, which might occur during severe accidents, CO and CO_2 would be generated, and degradation of the material could create greater safety concerns.

A reflector of either graphite or BeO should be installed to facilitate removal and inspection. Ideally, reflector life should be equal to plant life, but this is highly unlikely for graphite. An additional factor for both is the possibility for extended outage for removal, inspection, and replacement.

Clearly, a significant R&D effort would be required to determine the feasibility and potential problems associated with the behavior of graphite and BeO under long-term PWR neutron irradiation. The ramifications of leaching, washout, cladding deterioration, and failure would need to be considered as well.

Heavy water and certain hydrocarbons have also been proposed as possible candidates for improved reflectors. However, these would require a separate liquid system (makeup, circulation, letdown, purification, etc.). The incremental uranium utilization benefits are not large enough to warrant the added cost, complexity, and proliferation of concerns that would accompany the use of these materials in LWRs.

Of the two concepts considered most practical, the one that presents the least problem from a technical standpoint is increasing the thickness of the stainless steel core baffle. There is a limit, however, to how much baffle thickness can be increased and still be accommodated within the existing core support barrel. Excessive gamma heating can also become a problem as thickness increases. A larger pressure vessel (4-8 inches increase in OD) may be required to derive the full potential benefit in uranium utilization.

The remaining concept, which involves replacing the stainless steel baffle with Zircaloy, appears to show the most promise. Implementation without changing pressure vessel size may be possible, but the cost tradeoffs of further uranium savings versus pressure vessel cost increases should be considered in detail before ruling out the larger vessel. This concept presents several problems related to materials compatibility and fabrication technology that must be addressed in reflector redesign.

The use of Zircaloy for the core baffle causes several unique material compatibility problems. The most significant arise from the low coefficient of thermal expansion of Zircaloy relative to that of stainless steel and the growth of the material upon exposure to neutron fluence. There is a potential impact on safety analysis during a LOCA because of the oxidation behavior of Zircaloy. If baffles and formers are not supporting members, the impact would be minimized.

The low coefficient of thermal expansion of Zircaloy and irradiation-induced growth are physical properties of the material which limit the means of attachment to stainless steel to mechanical techniques. Under fast neutron irradiation (E>1 MeV) in the absence of stress, anisotropic dimensional changes occur in Zircaloy which are proportional to neutron fluence. As a consequence, Zircaloy grows in one direction and shrinks in another during irradiation. Irradiation growth in Zircaloy is not a strong function of temperature, but it tends to maximize near 300C, which is in the range of PWR temperatures. Growth characteristics are different for cold-worked and annealed Zircaloy, particularly under the effects of irradiation-induced stress relief. Some control over irradiation growth can be obtained by controlling the crystalline structure of the material during processing. Considerable data in the literature demonstrate that these effects can be minimized and controlled to yield predictable design behavior. Additional concerns arise because of nonuniform axial and azimuthal neutron irradiation, which results in nonuniform radiation growth and, subsequently, a wide variation in the total deformation in the various components. The degree to which these effects will influence the final design can only be determined after the fabrication procedure for the material is established, material is produced, and irradiation data are obtained.

One of the most important factors influencing the design will be the ability to consistently control manufacturing procedures to ensure predictable irradiation effects. Manufacturing approaches have been developed and are currently being used to produce Zircaloy fuel cladding, and there is no reason to doubt successful development of the plate material essential to fabricate the core baffle. Research will be necessary, however, to establish the most acceptable fabrication process and then to verify the results with irradiation studies under expected conditions. Future retrofit of a Zircaloy baffle is probably feasible.

3.4.5. Criterion D — Operational

3.4.5.1. Reliability

Maintenance requirements are dependent on the reflector material, reflector design, and the influence of the reflector on the radiation environment of the reactor vessel and its internals. Optimization of these effects will require further study and testing of the specific reflector material and geometry requirement.

Reflector materials must be compatible with the inservice reactor environment over their expected lifetime. Radiation damage and corrosion place the most severe restrictions on reflector material and size. Ideally, a reflector should be selected that has a service life consistent with the critical path for reactor vessel and internals maintenance and inspection. Under present technical specifications, this would be 10 years. Some reflectors will require the use of a cladding material to ensure the design life and to limit corrosion. These cladding materials must be compatible with both the reflector material and the reactor environment. Of the concepts considered, increasing the thickness of the steel baffle will probably have the least effect on reliability and may, in fact, increase it.

3.4.5.2. Availability

The initial application of new reflector design and associated reactor internals design may require frequent inspection until performance and service life are demonstrated. Testing would be anticipated intermittently throughout plant life to establish that cooling, vibration, flow, and other operational effects are accommodated by the design.

The degree to which this affects availability will depend to a great extent on the amount of testing done before full core installation. With the possible exception of the thicker steel baffle, all of these concepts have the potential for causing significant inservice problems despite a substantial development effort.

3.4.5.3. Operability

No significant impact on staff requirements, plant maneuverability, or plant operation is expected from the use of an improved reflector. A requirement for more frequent inspections will result in somewhat higher personnel radiation exposure.

3.4.6. Criterion E — Other Considerations

3.4.6.1. Utility Acceptance

Utility acceptance will depend on the degree to which R&D programs demonstrate and establish the benefits, costs, and the licenseability of the alternate reflector designs. Improved economics and comprehensive preoperational demonstration will be required.

3.4.6.2. Commercial Operation

Commercial operation dates for the various designs are projected as follows:

Steel only	1995
Zircaloy	2000
Graphite or beryllium	2004

3.4.6.3. Compatibility With Non-Proliferation

Core periphery modifications such as those considered here have no impact on proliferation issues.

3.4.6.4. Recycle Compatibility

The use of an improved reflector does not preclude future plutonium recycle.

3.4.6.5. Compatibility With Other Concepts

The gains in uranium utilization will be lower with the use of a low-leakage shuffle scheme, a radial blanket, or low power density. Otherwise, an improved reflector is compatible with both the backfittable and non-backfittable concepts. If a radial blanket is under consideration, it may be beneficial to optimize the blanket and reflector together for maximum fuel utilization.

3.5. Case V — Radial Blanket

3.5.1. Objective

The objective of this concept is to improve fuel utilization by either surrounding the core and/or replacing the fuel around the core periphery with a fertile material. The radial blanket material may be natural uranium, tails uranium, thorium, or spent fuel. The overall effect is to place fissionable material in a relatively more important core location, reduce the unproductive leakage of neutrons, and increase the conversion ratio. Consequently, uranium requirements are reduced.

3.5.2. Criterion A — Uranium Utilization

The blanket material surrounding the core enhances uranium utilization by (1) reducing unproductive leakage of neutrons from the core by capturing neutrons in fertile material, (2) increasing reflection of neutrons back into the core, and (3) providing in situ power production in the fissile material produced in the blanket. The degree to which a radial blanket enhances uranium utilization depends on the blanket material, the metal-to-water ratio in the blanket (the optimum of which varies with the blanket material), the blanket thickness, the fuel management scheme employed in the reactor, and the service life of the blanket.

Over this broad range of parameters, it has been estimated that the use of a radial blanket could improve PWR uranium utilization by 3 to 4%. Assuming that a uranium blanket occupying approximately 10% of the core volume yields uranium savings of 3%, the 30-year cumulative requirements are estimated as follows:

	30-yr cumulative requirements	
	st U_3O_8/GWe	SWU/GWe, 10^3
NASAP Composite PWR	4789	3267
Radial blanket	4645 (-3.0%)	3182 (-2.6%)

3.5.3. Criterion B — Economics

3.5.3.1. Capital Costs

Addition of a truly effective radial blanket to the base design PWR would require a larger pressure vessel (and possibly pressurizer) and internals to accommodate the additional volume required by the blanket. This is considered a requirement, because if existing peripheral core locations were converted to blanket locations, the linear heat rate would increase and DNBR margins would be degraded unacceptably. Therefore, one of the major costs associated with effective radial blanket implementation is the cost of a larger vessel and internals. Capital costs required to accommodate a 4- to 8-inch radial blanket are estimated at $3 to $5 million.

3.5.3.2. Fuel Cycle Costs

Levelized 30-year fuel cycle costs were estimated for a radial blanket that improved uranium utilization by 3%. The use of both small fuel and blanket assemblies (at 5% fabrication cost premium) was assumed, as was a blanket service life of eight years. A fuel cost savings of about 2% relative to the NASAP Composite PWR is predicted.

	Fuel cost, m/kWe-h @ U_3O_8 price, $/lb	
	@ $40.00	@ $100.00
NASAP Composite PWR	6.8	10.7
Radial blanket, 3% savings	6.67	10.47

3.5.3.3. Capacity Factor

The radial blanket is expected to have a long service life and thus will have minimal effect on refueling times and the capacity factor. However, if the small fuel assembly is required to have a viable blanket, a substantial increase in refueling time may occur unless more advanced refueling techniques or a minimum shuffle scheme are developed. Determination of the probable effect on capacity factor should be included in a blanket design study.

3.5.3.4. Development Cost

A radial blanket would require development and testing of the blanket assemblies; redesign of the internals, including flow testing; a larger pressure vessel; and possibly a larger pressurizer. Costs are estimated at $5 to $10 million and would take from 3 to 5 years.

3.5.4. Criterion C — Technical

3.5.4.1. General

The addition of a radial blanket adds to the complexity of a PWR in several ways. For example, power distributions are changed, flow requirements are altered, in-core fuel management strategies must be reoptimized, changes will most likely be required in the shape of the core periphery, refueling operations will be affected, and additional "spent blanket" storage and disposal concerns are created. In addition, the lower neutron leakage will reduce the signal to the out-of-core detectors, particularly if an improved reflector is included.

Resolution of these concerns is well within the bounds of current technology. Radial blankets have been studied extensively, especially in the early days of the nuclear industry. They have also been studied extensively in recent years in connection with the breeder reactor programs. Finally, radial blankets have been studied recently for incorporation in modern boiling water reactors (BWRs).

Thus, the use of radial blankets in future PWRs is considered technically feasible. Although challenging design work is required in several areas to develop the practical details of this concept, no fundamental technological barriers are forseen to prevent its implementation. The radial blanket represents a straightforward extension of existing technology and is considered, from the technical standpoint, to have a relatively high probability of success.

Radial blankets may be designed in various configurations, including rodded or plate-type assemblies. In either case, the design would provide a higher metal/water ratio for the blanket than for the core in order to reduce neutron thermalization and enhance resonance neutron capture in the blanket.

Although there is no concern about cooling the blanket early in life because power production is low, the low coolant fraction raises concerns about adequate heat removal later on when the blanket contributes a significant fraction of core power. Flow orificing is a possible solution even though it adds complexity to the design. Limiting the service life of the blanket is also an option, but this may affect uranium savings. Resolution of the cooling issue would be an essential part of a blanket design study.

Optimum blanket thickness will vary, depending on the fertile material used, the M/W ratio, the reflector design, and the characteristics of the reactor. However, it is anticipated that the optimum thickness for a typical PWR application would be approximately half the thickness of a conventional PWR fuel assembly, or 4 to 5 inches. Thus, specially designed blanket assemblies would be placed around the core, either as components of peripheral fuel assemblies or as independent small assemblies between the fuel and the reflector.

The choice of fertile materials includes natural uranium, tails uranium, spent fuel, reconstituted spent fuel, and thorium. The use of reconstituted spent fuel is not recommended — first because of the problems associated with disassembly, inspection, and reassembly of the fuel rods (into a tighter lattice) and second because of the uncertainty of subsequent service life. Reinsertion of spent fuel assemblies as a blanket, however, is a concept worthy of further consideration.

Thorium presents other difficulties, including lack of suppliers or fabrication facilities, radiation hazards resulting from ^{232}U production, and possible proliferation concerns because of the high fissile content of the uranium produced. Thus, it is concluded that the most desirable materials are natural or tails uranium oxide and spent fuel assemblies. The scope of this assessment did not permit calculations to evaluate the relative uranium utilization benefits of these materials.

3.5.4.2. Safety

No major safety problems are forseen for a PWR with a radial blanket based on virgin materials (natural uranium, tails uranium, or thorium) or spent fuel assemblies. The use of reconstituted spent fuel would raise safety questions related to fuel integrity.

3.5.4.3. Potential for Retrofit

The potential for retrofit is high if a reactor is designed initially with the small fuel assembly since fuel and blanket assemblies could be interchangeable. The ramifications of changing flow requirements would have to be considered in the original design.

3.5.5. Criterion D — Operational

3.5.5.1. Reliability

Overall reliability will be somewhat less with a radial blanket because it will increase the number of components inside the pressure vessel. These components will be subjected to radiation damage, thermal stresses, and flow-induced vibration. Consequently, frequent inspection and maintenance are anticipated along with periodic replacement. Maintenance requirements will depend on the blanket material and design as well as the design of the vessel internals. Blanket assemblies may require periodic rotation to achieve a more uniform burnup.

Effective implementation of the radial blanket may result in the use of the small fuel assembly and thus present the same additional operating, refueling, and maintenance considerations anticipated for the small assembly. These include requirements to either develop high-speed refueling methods or a minimum shuffle scheme.

Reduced power production in the peripheral blanket assemblies may result in reduced nvt for the pressure vessel and internals as well as reduced gamma heating of the internals. Although these components are designed for the life of the plant, the increased margins would have a positive impact on both reliability and service life.

3.5.5.2. Availability

Availability will probably be a little lower with the radial blanket, depending on how much additional fuel handling, inspection, or maintenance is required and whether or not it becomes a critical path item. Blanket service life will be relatively long and is not expected to affect availability significantly.

3.5.5.3. Operability

Power operation and maneuverability should be relatively unaffected by the use of a radial blanket. The major impact on operability would result from increases in complexity and the duration of refueling outages to accommodate the small assembly.

Personnel radiation exposures would increase with additional fuel handling operations and unanticipated maintenance problems. Additional, more highly trained staff personnel will probably be required.

3.5.6. Criterion E — Other Considerations

3.5.6.1. Utility Acceptance

The utilities have tended to accept the axial blanket because it is an integral part of the fuel assembly and, aside from the acceptable effect on thermal margins, it creates no significant problems for them that offset the gains. A radial blanket (other than a one-for-one substitution for fuel) will add critical components inside the pressure vessel, decrease margins, and possibly increase refueling time. We would expect the utilities to take a hard look at the cost/benefit relationship for a radial blanket.

Blanket assemblies will have to meet the same standards as fuel assemblies. No unusual licensing problems are anticipated.

3.5.6.2. Non-Proliferation Compatibility

Some adverse non-proliferation characteristics may be associated with the ^{233}U content of a thorium blanket or with the higher fissile plutonium fractions that occur at low burnups for a natural or tails uranium blanket, but these are expected to be minor.

3.5.6.3. Compatibility With Other Concepts

Radial blankets are generally compatible with most other concepts, both backfittable and non-backfittable. Gains, however, will probably be diminished if the low-leakage shuffle scheme is used, and frequent refueling may be less attractive depending on the fuel assembly design used to accommodate the blanket. The small fuel assembly is complementary because it permits a larger blanket in the same space. Low power density and core peripheral modifications compete with the radial blanket for pressure vessel volume. These concepts are compatible, but the composite savings will be less than the sum of the individual contributions.

3.5.6.4. Commercial Operation Date

Commercial operation date for the radial blanket is projected to be 1999.

3.6. Case VI — Low Power Density

3.6.1. Objective

The objective here is to improve fuel utilization through improved neutron economy at lower power density. Lower power density provides reactivity gains from reduced Doppler, xenon, and leakage reactivities and from the use of high-Z (number of batches) shuffle schemes. It was assumed that lower power density would be achieved by increasing the number of fuel assemblies. In an optimization study, changes in active length would also be considered.

Reductions of up to 45% were considered in the uranium utilization and fuel cost evaluations. A value of 30% was used in assessments against the other criteria.

3.6.2. Criterion A — Uranium Utilization

Reducing power density results in equilibrium uranium utilization and separative work savings because of reduced Doppler, xenon, and leakage reactivities and because more refueling batches (higher Z) can be used with the same cycle energy extraction and fuel burnup limit. The initial core requires more uranium (and less separative work) with reduced power density, offsetting some of the equilibrium gains. The fraction of equilibrium uranium savings offset by Core 1 loading increases with reduction in power density.

Fuel utilization and separative work savings were estimated for 15, 30, and 45% reductions in power density. Since average discharge burnup is conserved in all cases, a longer incore residence period is required as power density is reduced. Enriched uranium requirements decline substantially with power density reduction.

	30-yr cumulative requirements	
Case	st U_3O_8/GWe	SWU/GWe, 10^3
NASAP Composite	4789	3267
15% reduction	4617 (-3.6%)	3087 (-5.5%)
30% reduction	4455 (-7.0%)	2911 (-10.9%)
45% reduction	4321 (-9.8%)	2748 (-15.9%)

3.6.3. Criterion B — Economics

3.6.3.1. Capital

Capital costs will be most affected by changes in the reactor vessel and internals, increased containment volume, and increased number of control rods.

For a 30% reduction, pressure vessel ID will increase by 15 to 16% and weight will increase by approximately 33%. No shop fabrication limitations are envisioned over the range of reductions considered, although shipment may create problems for certain sites. Forces on the internals during LOCAs may require special design consideration but should be within current technological capability. The containment volume or design pressure must also increase to accommodate the increase of approximately 22 to 25% in reactor coolant system volume. The increased capital cost for the containment, vessel and internals, and additional control rod devices, plus modifications to supports, handling equipment, storage space, etc. is expected to be in the range of $15 to $20 million. Equivalent values for power density reductions of 15 to 45% are $7 to $10 million and $25 to $30 million, respectively.

3.6.3.2. Fuel Cycle Costs

As power density is reduced, the costs of loading the initial core increase while the direct costs (excluding capital charges) of loading subsequent cycles decrease, reflecting lower uranium and SWU requirements. Since the incore residence time required to achieve the same discharge burnup is inversely proportional to power density, capital charges on the fuel tend to increase with reduced power density.

Levelized 30-year fuel costs were estimated for the 15, 30, and 45% power density reduction cases and compared to those of the NASAP Composite PWR in July 1980 dollars.

	Fuel cost, m/kWe-h @ U_3O_8 price, $/lb	
	@ $40.00	@ $100.00
NASAP Composite	6.8	10.7
15% power density reduction	6.74	10.61
30% power density reduction	6.68	10.52
45% power density reduction	6.63	10.44

Fuel costs are decreasing with reduced power density because the increased inventory charges due to longer residence time are less than the direct cost savings in uranium and separative work. The calculation is relatively sensitive to the capital charge rate used in the calculation, which varies among public and private utilities. A rigorous evaluation of the effect of reduced power density on fuel costs requires accurate modeling of a utility's financial structure.

The present values of 30 years of fuel cost savings were compared to the estimated capital cost increases to determine whether an optimum power density was indicated. As the following data show, the fuel cost savings are essentially equal to the increased capital costs in all cases using $40/lb U_3O_8 and show an increase with power density reduction when $100/lb U_3O_8 is used. Thus, for the carrying charge rates and the range of uranium prices used, no optimum power density was found. The calculation does suggest, however, that the uranium savings associated with reduced power density may be obtained without a significant overall cost effect.

Power density reduction, %	Capital cost increase, $ million	Present value of 30-yr fuel cost savings, $ million	
		U_3O_8 @ $40/lb	U_3O_8 @ $100/lb
15	7-10	9.5	14.0
30	15-20	18.6	27.4
45	25-30	26.7	39.3

3.6.3.3. Capacity Factor

A low power density design should equal or exceed the reference capacity factor since greater operational and safety margins are inherent, but we have not attempted to assign a numerical value to the increase.

3.6.3.4. Development

No new technology is required, but extensive fuel management optimization will be required to take advantage of the low power density. Although the power density is reduced, first-of-a-kind engineering will be extensive, and a new standardization effort will be required. The large core size may also require vessel model flow distribution tests and field verification. Development costs should be in the range of $5 to $10 million for power density reductions of 15 to 45%.

An advantage to this concept is that the improvements in fuel utilization could be obtained by using existing component sizes; e.g., a 205-fuel assembly core and reactor with system components sized for 145- and 177-fuel assembly cores. Reductions in power density would be 29 and 14%, respectively.

3.6.3.5. Impact on Construction Time

Handling and installing the significantly larger components will increase construction time, but only by something less than one year.

3.6.4. Criterion C — Technical

3.6.4.1. Technical Feasibility

The changes required in the primary system and some BOP component designs to accommodate the increase in the number of fuel assemblies should be feasible and within current technological capability. The immediate effect on primary system design includes a scale-up in reactor vessel and internals and an increased number of control rod assemblies. The increase in coolant inventory will require approximately 10% more pressurizer volume in order to ensure proper response during transients in which coolant density changes. Some increase in the coolant volume handling capacity of the makeup and letdown systems may also be necessary to maintain the same effectiveness in boron control.

The increased coolant volume will also result in more stored energy which must be accommodated during loss-of-coolant leakage to the containment. Close coupling between LOCAs and containment design requires that the containment volume or design pressure be increased to account for an additional 22 to 25% stored energy.

3.6.4.2. Safety

Reduced power density should increase safety margins, particularly in DNB, linear heat rate limits, and fission gas generation. Special consideration should be given to transient and accident analysis of the reactor vessel internals, pressurizer, and containment.

Increasing size will tend to decouple the core, possibly making it more susceptible to power tilts and xenon oscillations. The lower flux will compensate to some extent. Stuck rod worths may be adversely affected.

3.6.5. Criterion D — Operational

3.6.5.1. Plant Reliability

It is not anticipated that scale-up of the reactor vessel, the reactor vessel internals, or the pressurizer will significantly alter the present plant reliability. The increased reactor coolant volume will result in a proportional increase in feed-and-bleed processing time during maneuvering and plant startup or shutdown. The additional use of BOP equipment in the performance of these operations could lessen reliability if these system processing rates are not scaled up accordingly.

3.6.5.2. Plant Availability

The increase in the number of fuel assemblies required for the low power density core will result in lost plant availability. Refueling may take 3 to 4 days longer unless a minimum-shuffle scheme is employed.

3.6.5.3. Operability

Plant operation should not be significantly altered from that for the reference core, except during refueling operations. The increased number of fuel assemblies that must be handled will tend to proportionally extend the refueling period. It may be anticipated that this will lead to greater radiation exposure for the station personnel or the need to increase the number of personnel involved in the outage. Total added exposure, assuming complete shuffle of 43% more assemblies, is estimated at 2 Rem or less.

3.6.6. Criterion E — Other Considerations

3.6.6.1. Utility Acceptance

If power generation costs relative to the standard plant are attractive, utilities should find this concept acceptable. No new technology is required, licensing should be no more difficult than for a standard plant, and the image of safety through conservatism is enhanced.

3.6.6.2. Commercial Operation

Commercial operation could begin by 1995.

3.6.6.3. Non-Proliferation Compatibility

Adoption of lower power density should have no impact on non-proliferation issues.

3.6.6.4. Compatibility With Recycle

Lower power density would be an asset with plutonium recycle because of the increased thermal margins. Power peaking is more difficult to control with plutonium recycle.

3.6.6.5. Compatibility With Other Concepts

Lower power density is compatible with most backfittable and non-backfittable concepts. It probably facilitates introduction of a radial blanket and reflector because of lower leakage. Low power density conflicts with very high temperatures and pressures because of the impact on the pressure vessel and the difficulty of ensuring fuel cladding integrity during the increased core residence period required to achieve the increased exposure of the NASAP Composite PWR. (A 45% reduction in power density results in a nine-year core residence to reach the 57,000 MWd/mtU burnup achieved in five years with the NASAP design.)

3.6.6.6. Retrofit Potential

The large vessel size and greater operational flexibility plus increased safety margins make this concept attractive for implementation of future design innovations. A possible exception is the "small fuel assembly" concept which, without the benefit of "salt and pepper" shuffling, would result in excessive outage extension.

3.7. Case VII — Small Fuel Assembly

3.7.1. Objective

The objective of this concept is to increase uranium utilization by using a smaller fuel assembly, with perhaps one-fourth of the XY cross section of current PWR assemblies. In principle, smaller assemblies will permit better mixing of fresh and depleted fuel, require less lumped burnable poison to control peaking, facilitate greater use of low-leakage shuffle schemes, and permit somewhat higher average discharge burnups for the same maximum burnup. Small fuel assemblies will also mitigate the effects of fuel failure and provide greater flexibility in accommodating radial blankets. Two concepts of the small assembly are considered: an independent assembly similar to the standard assembly and one in which several small assemblies are joined together with common end fittings.

3.7.2. Criterion A — Uranium Utilization

Uranium utilization savings for the small fuel assembly are estimated to be in the range of 1 to 2%. The impact of better fuel mixing, lower residual burnable poison reactivity, and slightly higher average burnup is estimated at 1%. The potential improvement in the low-leakage shuffle scheme could not be readily determined in the time permitted for the assessment. On this basis, the 30-year requirements compared to the NASAP Composite PWR would be as follows:

	30-yr cumulative requirements	
	st U_3O_8/GWe	SWU/GWe, 10^3
NASAP Composite PWR	4789	3267
Small fuel assembly	4693-4741	3216-3242

3.7.3. Criterion B — Economics

3.7.3.1. Capital Costs

The independent small assembly will require more complex core internals and possibly higher-speed multiple handling or more automated refueling equipment. Small assemblies that share common end fittings will require minimal changes in core internals but will require sophisticated out-of-core disassembly, inspection, and reassembly equipment. The increase in capital costs to accommodate these designs is estimated to be \$2 to \$5 million.

3.7.3.2. Fuel Cycle Costs

Levelized 30-year fuel costs were estimated for the small fuel assembly and compared to those of the NASAP Composite PWR in July 1980 dollars. Fabrication cost is expected to increase approximately 5% because of the increased number of operations and components, plus the possibility that more grids per assembly may be required.

	Fuel cost, m/kWe-h @ U_3O_8 price, \$/lb	
	@ \$40.00	@ \$100.00
NASAP Composite PWR	6.8	10.7
Small fuel assembly	6.70-6.76	10.53-10.63

The estimated enriched uranium savings exceed the antiticpated fabrication cost increase, resulting in a modest fuel cost reduction.

3.7.3.3. Capacity Factor

Increased fuel handling time, either incore with the independent small assembly or out-of-core with the small assembly combination, will tend to reduce availability and hence capacity factor. Development of a minimum-shuffle scheme and/or improved refueling equipment, however, can offset this effect considerably. Consequently, we do not anticipate a significant change in capacity factor if the small assembly is used.

3.7.3.4. Development Costs

Development costs, including those for designing and testing prototype assemblies; modifications to reactor internals, control rods, and incore instrumentation; fuel handling, disassembly and reassembly equipment; and the development of fuel management schemes that produce the savings in fuel utilization, are expected to be in the range of $6-$12 million.

Four to five years for fuel assembly design and testing are anticipated. The independent assembly can probably be developed sooner than a small assembly combination.

3.7.3.5. Impact on Construction Time

Although minimal impact on construction time is anticipated for the small fuel assembly, initial core loading will take longer.

3.7.4. Criterion C — Technical

3.7.4.1. Feasibility

A major concern with the use of small, independent, and free-standing assemblies is the impact of the higher slenderness ratio on the assembly's ability to withstand vibration, LOCA, and seismic forces. This could conceivably be corrected by using more and stronger grids, more rigid engagement of grids to guide tubes and fuel rods, more rigid attachment of end fittings, and longer engagement of the end fittings into the upper and lower core support structures.

The reactor internals would also require modification: more holes would be required in the support plates, causing a loss of strength; tolerances on assembly positioning would be tighter to minimize water gaps between assemblies; and control rods and guidance of the rods in the internals would have to accommodate rods capable of spanning multiple assemblies.

Handling four times as many assemblies creates a problem, and development of a minimum shuffle scheme would be desirable; however, a multi-head handling mast could conceivably handle four or more independent assemblies simultaneously with some savings in outage time. A minimum-shuffle scheme would have the advantageous side effect of reducing handling of partially burned assemblies.

The design of the internals and control rod drive train could be simplified by using a modular concept in which four quarter-size modules were assembled as a unit but could be separated and later reassembled to retain the desired fine structure shuffle scheme. The modular concept would also conceivably enable much of the bundle rigidity to be regained. Difficulties expected with the modular concept are differential expansion during operation, dissimilar irradiation-induced growth as a function of burnup, mechanical attachment, and the problems of incorporating rapid disassembly design features. A modular design will also be more expensive to test and fabricate.

The savings in vessel internals achieved with the modular design may well be offset by the amount and increased complexity of handling equipment required to perform the modular manipulations.

3.7.4.2. Safety

The small fuel assembly raises no significant new safety concerns, but the potential additional fuel handling (perhaps multiple handling) and more complex refueling equipment will statistically increase the risk of personnel injury and radiation exposure. Design improvements can be made to minimize these effects.

3.7.5. Criterion D — Operational

3.7.5.1. Plant Reliability

The use of a smaller fuel assembly will greatly increase the number of assemblies and associated mechanical components beyond the reference design.

This will amplify the requirements for operational and refueling maintenance activities. Existing technology must permit design and construction of the smaller assembly to the same operational service standards as the larger assemblies. This will require optimization of material properties and strengthening of the fuel rod and assembly design for inservice conditions.

The reduced size and larger number of assemblies may also increase the probability of a fuel handling mishap. Refueling operations will require development of special tools and a smaller fuel handling mast. If the small assemblies are grouped into larger ones and handled as a unit, then additional tools will be needed for disassembly of the unitized bundles. The increased work load on fuel handling equipment would be expected to accelerate wear, and consequently, maintenance and inspection will increase.

3.7.5.2. Plant Availability

If conventional shuffle schemes and refueling equipment are used, the small fuel assembly could increase normal refueling time by over 600 hours or 25 days, significantly affecting both plant availability and capacity factor. Consideration must be given to additional measures to enhance refueling speed, such as the use of multiple-assembly handling tools or limiting the number of assemblies to be shuffled at each refueling. "Salt and pepper" replacement of one-fifth of the core would reduce the refueling time to less than that currently required for a full shuffle and is the preferable method if it can be achieved; this will require a significant study beyond the scope of this assessment. An alternate approach is to limit the small assembly to core peripheral applications which provide the advantages of blankets and/or reflector assemblies without affecting refueling in the remainder of the core.

3.7.5.3. Operability

The small fuel assembly may increase the complexity and duration of refueling outages. It would be anticipated that personnel exposure would increase and involvement of more staff personnel would be required to offset the number of total operations that must be performed and unanticipated fuel handling problems. An increase in accumulated radiation exposure for the refueling team is very likely with either small assembly concept.

3.7.6. Criterion E — Other Considerations

3.7.6.1. Utility Acceptance

The small fuel assembly offers both advantages and disadvantages to the utility. Advantages include a uranium savings, mitigation of fuel failure consequences, and potentially greater thermal margins. Disadvantages include a probable cost increase (per kgU); increased logistical requirements for fuel management (i.e., more assemblies) and, most significantly, the potential for reduced availability due to increased refueling time. Utility acceptance will depend on the development of refueling techniques and equipment that ensure repetitive acceptable refueling times as well as a clear definition of the advantages to be gained.

3.7.6.2. Commercial Operation

First commercial operation is projected for 1999.

3.7.6.3. Compatibility With Non-Proliferation

The small fuel assembly introduces no new non-proliferation concerns.

3.7.6.4. Recycle Compatibility

A smaller fuel assembly would enhance plutonium recycle because of the improved power peaking control.

3.7.6.5. Compatibility With Other Concepts

The small fuel assembly is compatible with all of the backfittable concepts represented by the NASAP Composite PWR and most of the non-backfittable concepts. It would enhance the higher temperature-higher pressure or radial blanket concept but may not be compatible with frequent refueling or substantially reduced power density (excessive refueling time).

3.7.6.6. Retrofit Potential

The concept where several small assemblies share common end fittings has good retrofit potential.

4. UNRESOLVED KEY QUESTIONS

Key questions relevant to the assessment of each concept were identified at the initial workshop and are included in Attachment A of Appendix A. The responses to most key questions have been incorporated into the technical rationale supporting the ratings, but some were beyond the scope of this study. Unresolved key questions and recommendations for their resolution follow, listed under case numbers.

I. High Temperature and Pressure

- What is the economic optimum design?

  A sophisticated economic analysis that assessed the expected monetary value of costs resulting from lower reliability and availability as well as the more easily estimated capital cost increases and fuel cost savings would be necessary to resolve this question. Current designs are probably close to the economic optimum when these added costs are considered.

II. Rapid/Frequent Refueling

- Can utility operations capitalize on unscheduled shutdowns?

  Our service records do not provide enough data to resolve this question. A detailed study of utility operations and experience would be required.

IV. Core Periphery Modifications

- Are any local hotspot problems anticipated?

  This is not expected to be a problem area, particularly with the low-leakage shuffle scheme, but we do not now have calculational support for this position. Upcoming studies will provide some data for Zr/steel/water reflectors. A special study would be required for graphite or beryllium reflectors.

V. Radial Blankets

- Does the blanket require core orificing and flow tailoring?

  The need for flow control is not established at this point. Cooling requirements should be included as part of any radial blanket study.

- What are the effects on power profiles?
- What are the effects of reflector-blanket interactions?
- Should there be spectral transition zones?
- How do the blanket, reflector, and low-power density interactions affect composite uranium savings?

  These questions are all appropriately resolved by a systematic study and optimization of the core, blanket, and reflector relationship. For any given vessel size, there is an optimum allocation of the space to each of these components that maximizes uranium utilization.

5. CONCLUSIONS AND RECOMMENDATIONS

5.1. Basis for Ranking

This section provides a subjective ranking of the concepts, the rationale for the ranking, and recommendations for further action. In setting the ranking, potential fuel utilization gains, technical feasibility, operational characteristics, and estimated capital and fuel costs were given greater weighting than development costs and other considerations. The seven concepts were divided into three groups.

5.2. Group 1 — Greatest Potential

• Quick/Frequent Refueling

Frequent refueling (i.e., 6-month cycles) offers high uranium and fuel cost savings for a moderate investment. It currently has little commercial appeal because the increased annual outage time with present refueling methods would require purchase of enough additional replacement power to more than offset the gains. Thus, the challenge is to develop refueling and maintenance techniques capable of accommodating two refuelings per year while limiting the increased outage time to a length acceptable to the utilities. Based on current replacement power costs, this is estimated at 7 to 10 days.

We concluded that, in the absence of a major technological breakthrough, minimum theoretical cold shutdown refueling time will be 11 to 12 days. Since refueling is rarely on the critical path during the main annual outage, the result of a second outage would be an unacceptable net increase in total annual outage time. However, the concept of shifting critical path items from the main annual outage to the "quick" outage may alleviate the problem by increasing the maximum allowable refueling time.

Thus, instead of typically alternating outages of 45 and 8 days, we would plan for 38 and 15 days. Our service records do not contain sufficient detail to determine whether or not the maintenance work can indeed be divided in this manner, but the gains of frequent refueling justify additional investigation.

Further Action

Programs to reduce refueling time to the minimum practical limit should be encouraged. Significant departures from current technology may be required to achieve acceptable results. Studies should be initiated to evaluate utility maintenance practices and requirements to determine the feasibility of separating the main annual outage into two parts, neither of which is dependent on unattainably short refueling times. Once it has been established that 6-month refueling is theoretically achievable, an overall plan to demonstrate its practicality to the utilities should be formulated.

- Low Power Density, Radial Blanket, Core Periphery Modification

These concepts all provide worthwhile potential savings in uranium utilization but at relatively higher costs or with increased technical complexity. All three require larger pressure vessels for maximum gains and in this regard are competitive concepts that should be optimized together. These concepts are achievable with available technology although substantial development programs may be required for the blanket and core periphery modifications. A well-developed combination of any of these concepts has potential commercial appeal.

Future Action

Designs combining low power density, radial blankets, and core periphery modifications should be studied comprehensively and systematically to determine the optimum allocation of pressure vessel volume from the viewpoint of maximizing uranium utilization. Such a study should cover a range of power densities and a variety of blanket and core periphery designs. Subsequently, programs to develop and demonstrate the recommended improvements should be supported. These may include, for example, irradiation tests of new core peripheral materials or blanket assemblies.

5.3. Group 2 — Less Potential But Worthy of Further Consideration

• Coastdown

Coastdown capability provides a proven mechanism for substantial uranium savings with few negative aspects. The primary question remaining is how to optimize this capability beyond that included in the NASAP Composite PWR considering that certain modifications to improve coastdown may reduce performance at full power. Improved coastdown capability will have commercial appeal and should be either incorporated in standard designs or offered as an option.

Further Action

Joint studies involving reactor vendors, turbine manufacturers, and possibly AEs are recommended with the objectives of optimizing coastdown capability with existing turbine designs and developing greater capability in this area based on modified turbine designs.

• Small Fuel Assembly

This concept alone produces a small uranium savings at the cost of much increased complexity; consequently, it lacks commercial appeal. However, the small assembly may play a synergistic role in obtaining or increasing the benefits of certain backfittable and non-backfittable concepts, including the low-leakage shuffle scheme and the radial blanket. Thus, the small assembly concept should be retained for further consideration.

Further Action

The potential contribution of a small fuel assembly to uranium savings should be considered in design studies involving other concepts. Additional studies to evaluate the advantage of the small assembly on its own merits should also be considered since little technical support actually exists for the estimated savings.

5.4. Group 3 – Least Potential

• High Temperatures and Pressures

The high costs and technical obstacles associated with significant increases in temperature and pressure do not justify the estimated gains. Losses in reliability, availability, and safety margins may be substantial under such conditions. Designs of this type may require significant innovation simply to retain the conservatism provided in current design practice. Commercial appeal of this concept is considered to be relatively low.

Future Action

Further increases in moderator temperature while maintaining current pressure levels may be advantageous in some designs, and programs to achieve these gains should be supported. Development of designs with significant pressure and temperature elevations is not recommended.

6. REFERENCES

[1] Nuclear Proliferation and Civilian Nuclear Power, Report of the Nonproliferation Alternative Systems Assessment Program, DOE/NE-0001/9 and DOE/NE-0001/5, U. S. Department of Energy, Washington, D. C., June 1980.

[2] U. Decher and N. L. Shapiro, Improvements in Once-Through PWR Fuel Cycles — Interim Progress Report for Fiscal Year 1978, CEND-376, Combustion Engineering, Inc., January 1979.

[3] E. K. Fujita, M. J. Driscoll, and D. D. Lanning, Design and Fuel Management of PWR Cores to Optimize the Once-Through Cycle, MIT-EL 78-017, Massachusetts Institute of Technology, August 1978.

[4] D. Dabby, Fuel Utilization Improvements in a Once-Through PWR Fuel Cycle, Final Report on Task 6, ORNL/SUB-7494/3, Westinghouse Electric Corp., June 1979.

APPENDIX

Battelle
Pacific Northwest Laboratories
P.O. Box 999
Richland, Washington U.S.A. 99352
Telephone (509)
Telex 15-2874

July 15, 1980

Mr. S. Wayne Spetz
Babcock & Wilcox Company
P.O. Box 1260
Lynchburg, VA 24505

Dear Mr. Spetz:

SUBJECT: Reference Case Data for Assessment of Nonbackfittable Concepts

At the workshop to initiate the assessment of nonbackfittable LWR concepts for improving uranium utilization on July 1 and 2, 1980, PNL was assigned responsibility for providing reference case data to each participating industrial organization, to provide a common basis for the information to be furnished to PNL in the draft working papers by August 15, 1980. The purpose of this letter is to provide you with the reference case data. The nonbackfittable concepts should be assessed relative to LWR systems which have incorporated the backfittable improvements that are included in reference case data.

The basis for the reference case data is the Report of Nonproliferation Alternative Systems Assessment Program, DOE/NE-0001/9, "Volume IX: Reactor and Fuel Cycle Description" June 1980. The reference case includes most of the retrofittable fuel design and fuel management improvements that are considered technically and economically feasible in currently operating plants or plants under construction. The uranium utilization improvement potential of the reference case over current design practice is in the range of 25% (21% for PWRs and 28% for BWRs). The composite improvements included in the reference case are considered potentially retrofittable options which might be deployed before the year 2000.

Reference Reactor and Fuel Cycle Data for PWR:

General PWR performance specifications are listed in Table 1. The fuel consists of low-enriched UO_2 pellets encapsulated in a Zircaloy-4 cladding. Control rods travel into the fuel assembly from the top. A stainless steel baffle (or shroud) is used around the core periphery.

Retrofittable improvements incorporated into the reference PWR which provide a composite saving of about 21% when employed with a 12-month cycle include:

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- Extended exposure,
- Low-leakage fuel management,
- Lattice optimization,
- Axial blankets,
- Preplanned costdown,
- Fuel use of early batches of startup core.

The parameters of the reference once-through fuel cycle for the PWR are listed in Table 2. Comparisons of requirements for U_3O_8 and separative work should be based on the annual equilibrium values, assuming that the final core and the initial core have similar scrap values.

Reference Cost Data

The costs and other economic-related information presented in this reference cost data was derived from Appendix C of the NASAP report. Since the costs in the NASAP report are given in terms of January 1978 dollars, a factor of 1.34 was used to correct for inflation to July 1980 dollars. Reference capital investment costs for LWRs are listed in Table 3.

TABLE 3. Capital Costs for LWRs (July 1980 \$/KWe)

	Capital Costs[a]		
LWR Capacity	600 MWe	1000 MWe	1300 MWe (Reference)
Excluding interest during construction	1179	891	771
Including interest during construction	1420	1072	925

[a]Includes 7% owner's cost during construction.

Operation and maintenance costs for the reference LWR is \$17/KWe per year for fixed costs plus \$1.3/KWe x capacity factor per year for variable costs in July 1980 dollars. The reference lead time for design, licensing and construction of an LWR is ten years. Adding two years for planning and one year for startup tests brings the total time from the beginning of an LWR project to its production of commercial power to 13 years. Fuel cycle costs for the reference LWR are 6.8 mills/KWe-hr at \$40/lb U_3O_8, and 10.7 mills/KWe-hr at \$100/lb U_3O_8 in July 1980 dollars. The total power cost for the reference LWR is 25.4 mills/KWe-hr at \$40/lb U_3O_8 in July 1980 dollars.

Sincerely,

Darrell F. Newman

Darrell F. Newman, Principal Investigator
Advanced Reactor Design Program

DFN/kt

cc: R. M. Fleischman
S. Goldsmith
A. W. Prichard

TABLE 1. General Reactor Performance Specifications - PWR

Power plant performance	
Core thermal power, MW	3800
Electrical power, MW	
Gross	1344
Net	1270
Thermal efficiency, %	33.4
Reactor parameters	
Core volume, liters	40,050
Equivalent core diameter, m	3.66
Core height, m	3.81
Core power density, MW/liter	0.095
Coolant flow rate, Mg/sec	20.66
Coolant inlet temperature, °C	296
Coolant outlet temperature, °C	327
Primary system pressure, MPa (psia)	15.5 (2250)
Fuel parameters	
Average Fuel temperature, °C	688
Maximum fuel temperature, °C	1882
Cladding temperature, °C	342

TABLE 2. Fuel Management Information - PWR, Once-Through Cycle

	Reference PWR With Composite Backfittable Improvements
Average capacity factor, %	75
Fraction of core replaced per refueling	0.2[a]
Refueling interval, years	1
Fissile fabrication loss fraction	0.015
U_3O_8 requirements, st/GWe[b]	
Initial core	376
Annual equilibrium	150
30-year cumulative	4789
Separative work requirements 10^3	
Separative work units (SWU)/GWe[b]	
Initial core	196
Annual equilibrium reload	106
30-year cumulative	3267
Equilibrium cycle enrichment, %	4.70[c]
Plutonium in spent fuel	
Annual equilibrium discharge	
Plutonium fissile, kg/GWe	95
Plutonium total, kg/GWe	140
30-year cumulative discharge	
Plutonium fissile, kg/GWe	2575
Plutonium total, kg/GWe	3807
Residual U-235 in spent fuel	
U-235, wt %	0.48
Annual equilibrium discharge, kg/GWe	69
30-year cumulative discharge, kg/GWe	1854
Batch-average discharge burnup, MWd/t	57,130
Peak discharge exposure, MWd/t	69,450

[a]This is a nominal value. The use of fuel reinsertion from early batches lowers this value by about 2.4% on the average.

[b]Tails composition 0.2 wt %.

[c]Active fuel region; blanket enrichment, 0.711 wt %.

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July 18, 1980

Mr. S. Wayne Spetz
Babcock & Wilcox Company
P.O. Box 1260
Lynchburg, VA 24505

Dear Mr. Spetz:

SUBJECT: Selected Concepts and Criteria for Assessments

At the workshop to initiate the assessment of nonbackfittable LWR concepts for improving uranium utilization on July 1 and 2, 1980 the industrial participants selected the concepts that will be included in the assessment, and the evaluation criteria to be used in rating the concepts. Enclosure "A" lists the seven PWR concepts that were selected and the key questions that must be considered in assessing each concept. Enclosure "B" lists the evaluation criteria in five separate categories: uranium utilization, economic, technical, operation, and other considerations. This material is being sent to each participating industrial organization to standardize the framework for the assessment effort so that the work can be performed in a consistent and comparable manner by all participants.

I am looking forward to receiving your draft working papers on the assessments of these concepts by August 15, 1980. As I indicated at the initial workshop it is important that we adhere to this schedule, since PNL is only allowed two weeks to consolidate these working papers is shown in Attachment "C". If you have any questions about this material or the preparation of the working papers, please call me on (509) 376-4663.

Sincerely,

Darrell F. Newman

Darrell F. Newman, Principal Investigator
Advanced Reactor Design Study

DFN/kt

Enclosure

cc: P. M. Long (DOE-H) B. I. Spinrad (OSU)
A. Mowery (DOE-H) J. C. Cleveland (ORNL)
R. Speier (ACDA)

Attachment A

PWR CONCEPTS TO BE ASSESSED

I. HIGHER TEMPERATURES AND PRESSURES

Consider design concepts for improving the thermodynamic efficiency of a PWR by increasing the temperature and pressure of the primary coolant system. Such a concept may involve:

- Primary system pressures up to 1000 psi higher than current designs.
- Use of PWR flow control.

Key Questions to be assessed include:

. Is there a material compatibility problem, or material service limitation, especially with Zircaloy cladding?

. Does the manufacturing capability exist for increased efficiency designs?

. What are the major primary components that will be affected?

. What is the economic optimum design?

. Is the desired PWR flow control achievable?

II. RADID - FREQUENT REFUELING

Consider design concepts for decreasing the total outage time for refueling-only to ten days or less in order to allow economic use of frequent refueling.

Key Questions to be assessed include:

. Is it possible to achieve a ten day refueling outage, considering Actual versus Theoretical time schedules?

. Will the Nuclear Regulatory Commission permit sub-cycle refueling, without increased regulatory exposure?

. Can utility operations capitalize on unscheduled shutdowns?

. What is the effect of frequent refueling on plant personnel exposure?

. What refueling operations are on the critical path?

III. COASTDOWN

Consider design concepts which increase the capabilities of a PWR to coastdown from design electrical output at the end of cycle, allowing either the useful length of operating cycles to be extended without changing fuel composition, or the fuel enrichment to be reduced without changing the cycle lengths. Such PWR concepts may involve:

- Decreasing feed water temperature
- Steam turbine redesign
- More negative moderator temperature coefficient

<u>Key Questions</u> to be assessed include:

. What are the necessary turbine capabilities?

. What is the impact of extended coastdown on Loss-Of-Coolant-Accident heat loads?

. What other Nuclear Steam Supply System limits are affected?

IV. <u>CORE PERIPHERY MODIFICATIONS</u>

Consider design concepts for improving uranium utilization by modifying the periphery of an LWR core. Such concepts may involve:

- Improved reflector design
- Material substitutions
- Configuration modifications

<u>Key Questions</u> to be assessed include:

. What are the material compatibility problems regarding service life and material properties?

. Can the designs be fabricated?

. What is the effect of these designs on other reactor components, including the reactor vessel?

. Are any local hotspot problems anticipated?

V. <u>RADIAL BLANKETS</u>

Consider design concepts for improving uranium utilization by adding a radial blanket around an LWR core. Fertile material for these blanket concepts may be either:

- Natural Uranium
- Depleted Uranium (0.2% ^{235}U tails from storage)
- Thorium
- Spent Fuel

Key Questions to be assessed include:

- What is the service life of the different blankets?
- Does the blanket require core orificing and flow tailoring?
- What are the effects on power profiles?
- What are the effects of reflector-blanket interactions?
- Should there be spectral transition zones?
- What fertile material should be used?
- What core periphery changes are needed?
- How do the blanket, reflector, and low-power-density interactions affect composite uranium savings?

VI. LOW POWER DENSITY CORES

Consider design concepts for improving uranium utilization by substantial reduction in the average power density of LWR cores. Such designs should not be restricted by current reactor vessel size. Uranium savings from low power density cores are attributable to:

- Increasing the number of fuel batches in the core.
- Reduction of parasitic neuton capture in Xenon.
- Lower leakage fuel management, which is enabled by accomodating higher power peaking.

Key Questions to be assessed include:

- What is the physical size of the reactor?
- What are the transportation and fabrication limits? i.e. Does the reactor vessel have to be field-fabricated?
- Will PWRs have to use fuel channels to distribute coolant flow? What is the penalty on uranium utilization for their use?
- What is the effect of low power density cores on the containment structure?
- What is the economic optimum core power density?

VII. SMALL PWR ASSEMBLIES

Consider design concepts which utilize small fuel assemblies, with about a quarter of the fuel pins as current PWR assemblies. Such designs may improve uranium utilization in PWRs by providing:

- Increased fuel management flexibility.
- Means to implement radial blankets.

Key Questions to be assessed include:

- Are there rigidity problems with small assemblies?
- What is the impact of small assemblies on refueling operations?
- What structural material changes are required?

Attachment B

EVALUATION CRITERIA

Evaluation criteria selected at the workshop on July 1 and 2, 1980, in Washington D.C. to initiate the industrial assessment effort are listed below. The criteria were grouped into five categories:

- Uranium utilization
- Economics
- Technical
- Operational
- Other considerations

Specific items to be used in rating each concept are presented under each of the five categories.

A. Uranium Utilization

. Cumulative requirements for 30 years of operation of the plant at an average capacity factor of 75%, (Short tones U_3O_8/GWe).

. Percent improvement over base case,* (%).

. Cumulative requirements for separative work for 30 years of operation of the plant at an average capacity factor of 75%, (10^3 SWU/GWe).

B. Economic

- Relative Power, Cost:
  . Change in capital cost of plant from base case* ($/KWe)
  . Fuel Cycle Costs (Levelized over 30 years of operation) for two uranium prices: $40/lb U_3O_8 and $100/lb U_3O_8, (mills/KWe-hr).
  . Estimate of percent change in Fuel Cycle Cost from base case*, (%).
  . Estimate of average capacity factor, if changed from the 75% value used in the base case*, (%).
- Development Cost:
  . Development cost per vendor, up to making a firm bid proposal on a system using the concept, (July 1980 $).
- Impact on Construction Time:
  . Change in time for design, licensing and construction of the plant, from ten years assumed in the base case*, (years).

* Letter from D. F. Newman (PNL) to Workshop Participants (July 14, 1980) "Reference Case Data for Assessment of Nonbackfittable Concepts".

C. Technical

- Technical feasibility (subjective analysis) i.e., yes...; yes, but...; no, unless...; or no... (provide rationale for your judgment)
- Safety (subjective analysis) i.e., more safe..., just as safe...; or less safe... (provide rationale for your judgment)

Consider the following aspects for the Technical Feasibility and Safety Evaluations:

- Complexity
- Degree of Demonstration
- State-of-the-Art
- Manufacturability
- Impact on the Balance of the System

- Potential for Retrofit (subjective analysis)
i.e., Would it be possible to make minor changes when the plant is built that would accomodate a later retrofit, if it became desirable?

D. Operational

- Plant Reliability (subjective analysis)
i.e., more reliable...; just as reliable...; less reliable...

  Consider the following aspects for Plant Reliablity Evaluations:

  - Maintainability
  - Inspectability
  - Equipment Lifetime

- Planned Plant Availability and Capacity Factor
i.e., higher..., just the same..., lower...

  Include estimate of change in plant availability and capacity factor, if possible.

- Operability (subjective analysis)
i.e., more operable..., just as operable..., less operable...

- Consider the following aspects for Plant Operability Evaluations:

- Number of Staff
- Quality of Staff
- Radiation Exposure of Staff
- Plant Maneuverability
- Plant Simplicity

E. <u>Other Considerations</u>

. Utility Acceptability (subjective analysis)

Provide Comments considering the following aspects:

- Quality Image
- Licensability
- Adaptability for Future Need (e.g., Load Following)

. Date of First Commercial Operation

Provide estimated date, assuming start of first step in FY 1981.

. Nonproliferation Compatibility (subjective analysis)

i.e., To what extent is there access to sensitive material and/or would safeguard changes be required.

. Viability with Recycle (subjective analysis)

Comment on whether the plant could be converted to Pu recycle if national policy were to change mid-life.

. Interaction with Other Concepts (subjective analysis)

Provide narrative consideration of interaction this concept has with other concepts (both backfittable and nonbackfittable concepts).

Attachment C

ADVANCED REACTOR DESIGN STUDY

NONBACKFITTABLE CONCEPT ASSESSMENT EFFORT

REPORT FORMAT:

- INTRODUCTION

  SUMMARY
- ASSESSMENT OF CONCEPTS
  - CONSIDERATION OF KEY QUESTIONS
  - RATING (CHART)
  - RATIONALE FOR RATING
- UNRESOLVED KEY QUESTIONS AND RECOMMENDED ACTIONS FOR RESOLUTION
- CONCLUSIONS AND RECOMMENDATIONS
  - RANKING OF CONCEPT
  - RATIONALE FOR RANKING
  - FUTURE ACTIONS