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INDUSTRIAL ASSESSMENT OF NONBACKFITTABLE PWR DESIGN MODIFICATIONS

**FINAL REPORT
NOVEMBER 1980**

**Work performed for
BATTELLE PACIFIC NORTHWEST LABORATORIES**

**under contract
B-A1964-A-H**

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INDUSTRIAL ASSESSMENT OF NONBACKFITTABLE
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FINAL REPORT

NOVEMBER 1980

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R. S. Daleas
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Prepared Under

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With

Pacific Northwest Laboratory
Battelle Memorial Institute

For

U.S. Department of Energy

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ABSTRACT

As part of the U.S. Department of Energy's Advanced Reactor Design Study, various nonbackfittable PWR design modifications were evaluated to determine their potential for improved uranium utilization and commercial viability. Combustion Engineering, Inc. contributed to this effort through participation in the Battelle Pacific Northwest Laboratory industrial assessment of such design modifications. Seven modifications, including the use of higher primary system temperatures and pressures, rapid-frequent refueling, end-of-cycle stretchout, core periphery modifications, radial blankets, low power density cores, and small PWR assemblies, were evaluated with respect to uranium utilization, economics, technical and operational complexity, and several other subjective considerations. Rapid-frequent refueling was judged to have the highest potential although it would probably not be economical for the majority of reactors with the design assumptions used in this assessment. The combined concept that includes low power density, radial blankets, and core periphery modifications was ranked closely behind rapid frequent refueling. The design concept possessing the smallest potential for successful implementation was the use of higher temperatures and pressures. It represents the most severe departure from current technology and therefore requires the most extensive development effort.

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Section 1

INTRODUCTION

1.1 BACKGROUND

The majority of nuclear power reactors in operation and under construction in the United States and worldwide are light water reactors (LWRs). These reactors have developed to an advanced state as a result of many years of experience and commercial competition. From their inception, it was envisioned that light water reactors would initially operate on the once-through uranium cycle and, after sufficient research and development, would shift to a recycle mode in which bred plutonium and residual uranium would be recovered through reprocessing, and recycled to reduce the requirements for mined uranium and separative work. This strategy was felt to be a natural precursor to the development and eventual deployment of the fast breeder reactor. The experience gained in reprocessing and refabricating LWR fuel would lead the way to a commercially viable fast breeder reactor. In addition, an initial source of breeder fuel would be readily available in the form of plutonium recovered from LWRs.

With the indefinite deferral of reprocessing and deployment of the fast breeder reactor, interest has focused on the adequacy of the uranium resource base and on methods of improving the fueling efficiency of light water reactors. This interest has manifested itself in the form of the U.S. Department of Energy's Uranium Utilization Improvement Program. Uranium utilization improvements fall into two broad categories: backfittable improvements which can be incorporated into existing plants and nonbackfittable improvements which involve major plant design changes and therefore are practical only in new plants. The focus of this report is on the latter of these two categories; the major elements in the backfittable program are reported elsewhere.

The forum for the assessment of nonbackfittable LWR uranium utilization improvements is the U.S. Department of Energy's Advanced Reactor Design Study. The study is being conducted under the overall management of the Battelle Pacific Northwest Laboratories (PNL) with the participation of a number of U.S. reactor vendors and consultants. The objective of the initial phase of this study is to provide the information needed to determine whether, and how, to proceed further with design effort and/or hardware development and demonstration for the most promising nonbackfittable improvements.

1.2 SCOPE OF ASSESSMENT

The Advanced Reactor Design Study concentrated on three main areas which were felt to possess the potential to substantially improve the uranium utilization in light water reactors. The three assessment areas were frequent refueling, increased system efficiency, and nonbackfittable nuclear fuel and core design changes. Table 1-1 is a list of nonbackfittable design modifications which fall into these three areas and which were offered by Battelle PNL as possible concepts for

assessment. As can be seen in this table, the three assessment areas have been further divided into three categories. The categories are roughly indicative of the implementation potential of the various design modifications. The Category 1 modifications are easily implemented but, in general, offer the lowest potential improvement in uranium utilization, whereas the Category 3 modifications are much more difficult to implement but may have higher potential uranium utilization improvement. It should be noted that some Category 3 modifications offer low potential improvements, particularly when combined with Category 1 modifications.. From this table of potential design modifications, several were selected for "in-depth" evaluation. In addition, a limited number of modifications were selected for a brief scoping evaluation as part of the Industrial Assessment of Nonbackfittable Design Modifications.

The Industrial Assessment consisted of an initial workshop, a review and assessment period, and a final workshop. In the initial workshop, the framework for the assessment effort was defined so that the work would be performed in a consistent and comparable manner by all participants. To this end, the design modifications to be assessed were selected and the criteria for assessing these modifications were established. In addition, key questions relating to each modification were identified and a methodology for rating each modification and for ranking the modifications relative to each other was established. During the review and assessment period, the selected design modifications were evaluated based on the criteria established during the first workshop. An attempt was made to answer the key questions relating to each design modification and to develop recommendations as to what if any additional effort should be expended on developing the selected design modifications. During the final workshop, each of the participants reviewed the results of the individual assessments to determine areas of agreement and disagreement. A general consensus was reached among the various participants as to what results and conclusions should be presented in the final report by Battelle PNL to DOE. In addition, recommendations were made in terms of further design effort and/or hardware development and demonstration. The contents of this report reflect the work accomplished by Combustion Engineering, Inc. during the review and assessment phase of this Industrial Assessment of Nonbackfittable Design Modifications. The overall results of the final workshop which include the assessments performed by the other participants will be contained in the final Battelle PNL report (1).

The LWR design modifications included in the industrial assessment along with the key questions relating to each modification are contained in Appendix A. These design modifications include the use of

- . higher temperatures and pressures,
- . rapid-frequent refueling,
- . end-of-cycle stretchout,
- . core peripheral modifications,
- . radial blankets,
- . low power density cores, and
- . small PWR assemblies.

These design modifications were selected on the basis that they would provide the greatest improvement in uranium utilization, that they had the highest probability of successful implementation, or that they were felt to have the lowest cost benefit ratio.

The criteria against which the selected design modifications were evaluated are given in Appendix B. The evaluation criteria were grouped into five categories: uranium utilization, economics, technical, operational, and other considerations which tended to be more subjective in nature. These criteria were used to determine the rating of each of the design modifications with respect to their overall potential for successful implementation in a new reactor design. The ratings were then inter-compared to establish an overall ranking of the different design modifications. These ratings and rankings will be discussed in Section 2 of this report.

1.3 REFERENCE FUEL CYCLE AND ECONOMIC GROUND RULES

The reference reactor and fuel cycle used in this assessment were the Combustion Engineering standardized System 80 3800 Mwt PWR operating on the composite improved once-through fuel cycle. This reactor and fuel cycle were used in the Nonproliferation Alternative Systems Assessment Program (NASAP)(2) to represent an estimate of the total savings of U_3O_8 consumption that can be achieved in current-design PWRs by implementing improvements which can be developed and demonstrated in the near term. The composite improved once-through cycle consists of the following uranium utilization improvement options:

- . extended burnup with modified fuel management,
- . low-leakage fuel management,
- . modified lattice design (in the form of reduced pellet diameter),
- . end-of-cycle coastdown,
- . full use of early batches of the initial core, and
- . axial blankets.

Extended burnup with modified fuel management involves the use of high-burnup fuel (50,650 MWd/t batch average)* together with an increased number of in-core batches (five). This combination of fuel management parameters results in an annual refueling schedule at a 75% capacity factor. Low-leakage fuel management entails loading fresh fuel in the core interior using a burnable poison to suppress the power in these assemblies during the first cycle of irradiation. Reduced core radial leakage is then achieved by loading a combination of once- and twice-burned assemblies in the peripheral core locations. A modified fuel design employing a lattice with an increased water-to-fuel volume ratio was adopted for the composite improved core. This modified design consists of reducing the fuel pellet diameter from 0.325 in. to 0.315 in., and reducing the clad thickness from 25 mils to 22.5 mils. The fuel rod pitch was kept constant at 0.506 in. This corresponds to an increase in the water-to-fuel ratio from 2.04 for the standard design to 2.29 for

* This discharge exposure corresponds to the unshimmed standard fuel design. Modifications such as the use of a shimmed fuel assembly design and a reduced fuel rod diameter will increase this value.

the modified design. Full use of early batches of the initial core involves reinsertion of assemblies from the first core for an additional cycle of exposure. Eight assemblies are reinserted each year for a total of 24 years; assemblies from Batches A, B, C and D of the initial core are thus irradiated for one additional cycle. Axial blankets are employed at the top and bottom of the core in order to reduce neutron leakage. In this improvement option, the upper and lower six inches of the 150 in. long fuel rods are loaded with natural uranium pellets. End-of-cycle stretchout, using feedwater-pressure augmentation (i.e., reduced feedwater temperature) followed by power coastdown, was employed as a means of operating the reactor beyond its nominal cycle length. Power levels at the end of cycle were reduced to approximately 70-80% of the plant net rated electrical output.

A summary of the most important reactor and fuel parameters for the reference PWR is given in Table 1-2. The fuel resource requirements for this reference PWR are given in Table 1-3. The reference fuel cycle represented by the data in Table 1-3 exhibits a reduction in U_3O_8 requirements of about 21% averaged over a 30-year reactor lifetime. The nonbackfittable design modifications assessed in this report have been considered relative to this composite improved cycle. Only improvements beyond those of the composite cycle are attributed to the nonbackfittable design modifications.

The economic analyses performed in this assessment use the cost assumptions summarized in Table 1-4. These assumptions are based on those used in the Nonproliferation Alternative Systems Assessment Program, but with the various costs updated to reflect July 1980 dollars (3).

1.4 ORGANIZATION OF FINAL REPORT

Using the evaluation criteria discussed in Section 1.2 and more fully defined in Appendix B, an assessment was performed of the potential of certain selected nonbackfittable design modifications to contribute to the long-term improvement in LWR uranium utilization. The results of this assessment are summarized in Section 2 in the form of ratings of each of the design modifications against the various evaluation criteria. The individual modifications are then ranked against one another so as to define which modifications should receive further attention. The rationale used in establishing this ranking is also summarized to provide Battelle PNL a basis for intercomparing the rankings of the various industrial assessment participants against one another.

Section 3 is the main body of the assessment. In this section, each design modification is discussed in terms of the individual assessment criteria. An attempt is made to address each of the KEY QUESTIONS as posed in Appendix A. Where possible, numerical values are given for fuel resource requirements, economic analyses, R&D needs, etc. to provide a quantitative figure of merit to use in the concept ratings.

The conclusions and recommendations for further effort are given in Section 4. In addition, any unresolved KEY QUESTIONS are delineated

together with recommendations for future actions that will supply answers to these questions. In some cases, no such recommendations are made because the ranking of the applicable design modification indicates that further work to resolve a KEY QUESTION is not warranted.

Following this last section of the report are two appendices. The first of these describes the design modifications selected for assessment and the KEY QUESTIONS to be resolved. In the second appendix, the evaluation criteria are specified and the level of detail desired for each is stated.

TABLE 1-1

PRELIMINARY LISTING OF NONBACKFITTABLE DESIGN MODIFICATIONS

<u>CATEGORY</u>	<u>FREQUENT REFUELING</u>	<u>INCREASED SYSTEM EFFICIENCY</u>	<u>NONBACKFITTABLE NUCLEAR FUEL AND CORE DESIGNS</u>
1	PWR Rapid Refueling System BWR Rapid Refueling System Hot Standby Refueling	Higher Temperatures and Pressures	Coast Down ^a BWR Flow Control ^a Variable Lattices Lower Power Density Reactors Fissile Material Control Vented Fuel Blankets ^a Reflectors ^a Soluble Boron for BWR Cold Shutdown
2		Integral Nuclear Superheat Add-On Nuclear Superheat	Seed Blankets Spectral Shift (Without D ₂ O)
3	Unit Core Refueling On-Line Refueling	Add-On Fossil-Fueled Superheat Supercritical Pressures	PWR Flow Control Tubular Fuel High Peaking Factor Reactor Advanced Control Rod Systems

a. Beyond the limits of backfittability

TABLE 1-2

REFERENCE REACTOR AND COMPOSITE FUEL PARAMETERS

Core Power, MW(th)	3800
Electrical Output, MW(e)	
Gross	1344
Net	1270
Fuel Assemblies	241
Specific Power, W/cm ³	94.9
Average Linear Heat Rate, (kW/ft)	5.34

OPERATING PARAMETERS

Primary System Pressure, psia	2250
Reactor Coolant Average Temperature °F	594
Nominal Fuel Temperature, °F	1270
Cladding Temperature, °F	648

FUEL ASSEMBLIES (Dimensions at 68°F)

Fuel Rods Per Assembly	236
Control Rod Channels Per Assembly	4
Instrumentation Channels Per Assembly	1
Active Fuel Length, in.	138
Blanket Length (at each end), in.	6
Fuel Rod Array	16x16
Assembly Pitch, in.	8.18
Fuel Cell Pitch, in.	0.5063
Fuel Pellet Diameter, in.	0.315
Zr Clad Thickness, in.	0.0225
Gap Thickness, in.	0.00350
Coolant-to-Fuel-Pellet Volume Ratio	2.287
Fuel Stack Density, gm/cm ³	10.054
Fuel Discharge Burnup, MWd/t	57,130

TABLE 1-3

RESOURCE REQUIREMENTS FOR REFERENCE PWR
WITH COMPOSITE BACKFITTABLE URANIUM UTILIZATION 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 ₃ O ₈ requirements, short tons/GWe ^b	
Initial core	376
Annual equilibrium reload	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

-
- 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%.

TABLE 1-4
ECONOMIC ASSUMPTIONS
(Constant July 1980 Dollars)

Uranium Price, \$/lb U ₃ O ₈	40, 100, 160
Conversion to UF ₆ , \$/kg HM	4.5
Separative Work, \$/kg SWU	114
Fabrication, \$/kg HM	160
Shipping, \$/kg HM	17
Disposal, \$/kg HM	137
Discount Rate, %	4.5
Carrying Charge Rate, %	8
Tails Assay, wt%	0.2
Plant Heat Rate, BTU/kWh	10212
Uranium Losses, %	1.5
Nominal Capacity Factor, %	75
Fixed Charge Rate on Capital, %	10.1

Section 2

SUMMARY OF THE INDUSTRIAL ASSESSMENT OF NONBACKFITTABLE DESIGN MODIFICATIONS

2.1 RATINGS OF INDIVIDUAL DESIGN MODIFICATIONS

The ratings of the individual design modifications are summarized in Table 2-1. The order in which these modifications are listed is arbitrary; later discussion of the relative ranking of the design modifications lists them according to their overall potential. The ratings shown in Table 2-1 fall into five general categories as defined in Appendix B. With the exception of uranium utilization, a more specific evaluation criterion is shown in brackets below each general evaluation category. These specific criteria were felt by the authors to be the most appropriate against which to judge the individual design modifications. By selecting a single criterion in each of these general categories, a more specific rating could be given, since in each of these general categories, a number of diverse items were addressed. It would have been difficult to make a single rating statement for each of the general categories without choosing one specific criterion because of the competing considerations of the various items. Although shown as separate concepts in Table 2-1, core periphery modifications, radial blankets and low power density cores should probably be considered as a design package because they all compete for space within the reactor vessel and they all capitalize to some extent on reductions in core neutron leakage.

2.1.1 Higher Temperatures and Pressures

The use of higher temperatures and pressures is, in principle, appealing because it results in a more efficient power generation system. Such increases in plant efficiency impact the total power generation cost and therefore have considerably more leverage than design modifications that impact only the fuel cycle cost. The improvement in uranium utilization resulting from the use of higher temperatures and pressures is estimated to be 5 to 9%, depending on how high the primary system temperature and pressure are increased. These uranium utilization improvements are based on the assumption that at end of cycle, when the soluble boron has been removed, the moderator temperature is reduced back to its reference value. Otherwise, the reactivity tied up in the higher Doppler absorption and in the poorer moderation of the higher temperature coolant will result in a cycle length penalty. Thus, this concept (as envisioned) includes a form of end-of-cycle coastdown to regain part of the uranium utilization benefit that would otherwise be lost because of the harder neutron spectrum.

It appears that design modifications that would increase the temperature and pressure of the primary system would be economical for the range of operating parameters considered in this evaluation. The R&D costs associated with such design changes would be substantial however because of the requirement to totally redesign the major NSSS and balance of plant systems. Testing facilities for such components as reactor coolant pumps would need to be radically modified or new facilities constructed to simulate the more adverse operating conditions associated with this design change.

Perhaps the most adverse impact of using higher primary temperatures and pressures would be on the fuel design. It may be necessary, for example, to use higher fill gas pressures and thicker cladding under these operating conditions to meet current licensing guidelines. Furthermore, radically new fuel designs may be necessary, including new cladding materials, because of the accelerated water-side corrosion rate of Zircaloy when the reactor coolant temperature is raised substantially. Because of the large development costs required to design and license PWRs for higher temperature and pressure operation, it is doubtful that the nuclear industry would adopt this design modification unless it was totally supported by government funding.

2.1.2 Rapid-Frequent Refueling

Of the design modifications considered in this assessment, rapid-frequent refueling offers the highest potential improvement in uranium utilization. At the same time, this design modification presents the greatest challenge to obtaining favorable economics. The improvement in uranium utilization is about 7-10%, depending on the fuel cycle and fuel management scheme used. Furthermore, this improvement in uranium utilization results in a reduction in fuel cycle costs of about the same percentage. However, the use of frequent refueling decreases the plant availability and necessitates the purchase of replacement power while the reactor is down for refueling. The total energy costs, which include both the fuel cycle cost and the cost of replacement power, are found to be higher than the standard annual refueling cycle for refueling outage times in excess of 7 to ten days. The actual break-even outage time is dependent on the cost of replacement power which for each utility is unique.

Although there will be a hesitancy on the part of the utilities to refuel more frequently because of the additional headaches and complications associated with bringing a large power plant back on line, the rapid refueling features of this design modification will help facilitate a quicker return to power from each refueling outage. Furthermore, some of the design changes needed for rapid-frequent refueling would shorten the recovery time for all outages, including those that were unplanned. It will be economical, therefore, for a utility to install the additional equipment required for rapid refueling during the construction of the reactor plant. The choice of what fuel cycle is utilized can then be deferred until the specific economic situation for that utility is known and an evaluation of the merits of performing frequent refueling is made.

2.1.3 End-of-Cycle Stretchout

End-of-cycle stretchout can be performed in currently operating LWRs. The uranium utilization improvement techniques that were addressed in this assessment were those that are considered nonbackfittable and that enhance the ability to perform end-of-cycle stretchout beyond current capabilities. Because very little, if anything, can be done in the area of core design to enhance end-of-cycle stretchout without deterring from other design modifications that enhance uranium utilization, this

assessment concentrated on evaluating the potential for extending the current capability to transfer power from the reactor to the turbine plant. It is estimated that the additional uranium utilization benefit from such nonbackfittable design modifications is approximately 1% beyond that which can be achieved with current reactors. This additional benefit is derived from extending the capability to practice feedwater temperature reduction as a means of improving the heat transfer from primary to secondary plants. The economic benefits derived from making the appropriate design modifications to achieve this goal are about 1% on cycle energy costs if the additional capital costs to implement the concept are small. The exact economic benefit is a function of the costs of U_3O_8 and replacement power, and on the amount of feedwater temperature reduction that is practiced.

The ability to accomplish the required design modifications to enhance end-of-cycle coastdown is strongly dependent on the flexibility designed into turbine generators and on turbine generator warranty considerations. Currently, turbine output is limited when feedwater heaters are taken out of service to reduce feedwater temperature because of the redistribution of steam flow within the turbine and the variation of component pressures from their design values. Since extraction steam from the turbine stages is used to preheat the feedwater, interruption of this extraction can cause changes in stage pressures, stage pressure drops, and steam flow throughout the turbine. As a consequence of these changes, some of the turbine bucket, diaphragm, and thrust loadings are increased beyond their design values. To prevent overstressing the turbine, therefore, the throttle valve opening must be reduced until the loadings are equal to or less than their design values. Thus, the feasibility of implementing the various nonbackfittable design modifications in the area of end-of-cycle stretchout lies, to a large extent, in the ability to design the turbine generator to operate under off-normal conditions.

2.1.4 Core Periphery Modifications

Improvements in uranium utilization can be obtained through the reduction of neutron leakage from the radial surface of the core. Such a reduction in neutron leakage can be accomplished by increasing the number of fast neutrons that are reflected back to the core and by reducing the absorption (both fast and thermal) in the ex-core regions so that more neutrons can diffuse back into the core. Several techniques can be used to accomplish this objective. These techniques include increasing the existing water gap between the peripheral assemblies and the core shroud, increasing the thickness of the current core shroud, replacing the present stainless steel shroud with a Zircaloy shroud, and substituting alternate material (notably beryllium, beryllium oxide, or graphite) for the existing reflector materials. All of the methods are subject to certain complications and/or development requirements that must be addressed before this design modification can be implemented. The improvement in uranium utilization that can be achieved by making core periphery modifications ranges from 0.4% to about 5%. As might be expected, the lower portion of this range can be achieved through implementation of design changes that are only an extension of current designs, e.g., the use of thicker peripheral water gaps or thicker core shrouds. In contrast, design modifications that result in uranium

utilization improvements at the top of this range require more radical departures from current designs, e.g., the use of BeO reflectors.

The additional capital costs associated with core periphery modifications are mainly a consequence of the requirement to increase the reactor vessel diameter to accommodate these design modifications (except for the use of a Zircaloy shroud and a BeO reflector). The cost of these alternate reflectors will be large because of the cost of the materials and the high cost of fabrication.

Each of the proposed core periphery modifications has design and development requirements that would delay implementation and necessitate government-funded R&D. The major problem with increasing the peripheral water gap between the core and the shroud would be the resulting unacceptably high core bypass flow and the increased potential for assembly bowing. The use of a very thick shroud would pose a difficult fabrication problem which would be costly. In addition, high internal heat generation due to neutron and gamma heating would cause high thermal stresses and eventually distortion that would cause high loads on the shroud attachments. The use of Zircaloy as shroud material poses a number of structural and manufacturing difficulties. Zircaloy grows and creeps significantly under high temperature conditions and in a neutron flux, and in time would lead to severe distortion. Furthermore, Zircaloy has poor wear resistance to fuel assembly rubbing which might occur during operation. Finally, welding of a Zircaloy shroud would have to be performed in an inert environment which would require new techniques because of the size of the structure. There has been considerable experience with the use of graphite, beryllium and BeO in test reactors. However, these materials have not been used in power reactors at the high temperatures and pressures encountered in commercial PWRs. A substantial development effort would be required to qualify these two reflector materials for commercial application.

2.1.5 Radial Blankets

The addition of a radial blanket composed of fuel assemblies loaded with fertile material improves the uranium utilization in a commercial PWR. This improved utilization is due to a reduction in core neutron leakage, a productive use of the neutrons that do leak from the core, and a shift of the flux to the periphery of the active core where the higher reactivity fresh fuel is loaded. Four different blanket materials were evaluated in this industrial assessment: spent fuel, natural uranium, depleted uranium, and thorium. The improvement in uranium utilization ranges from 5% for spent fuel to approximately 2% for thorium. The fuel cycle cost savings associated with the use of such blanket materials follows closely the percentage savings in uranium for the various blanket materials. The increased capital cost required to implement the radial blanket concept is estimated to be about 10-12 million dollars with the majority of the cost associated with the manufacture of a larger reactor vessel to accommodate the extra row of blanket assemblies and the other modifications which result when the reactor vessel is made larger.

The only major technical question in addition to the design and construction of an NSSS with a large reactor vessel is whether or not the blanket assemblies should have orificed flow (flow tailoring) and how to

effect such flow control. If the flow to the blanket assemblies is orificed, then some type of flow separation skirt would probably be needed to prevent large cross flows from the active core out to the blanket region. An alternative to orificing the flow would be to increase the overall coolant flow rate so that sufficient flow was provided to the higher power active core assemblies while maintaining full flow to the blanket assemblies.

2.1.6 Low Power Density Cores

Low power density cores offer the potential for improved uranium utilization through a reduction in core neutron leakage, through less Doppler absorption in U-238, through a reduction in xenon poisoning, and through the use of more in-core fuel batches. In addition to an improvement in uranium utilization, the low power density core also offers the potential for increased thermal margin which would tend to make the reactor design more conservative and therefore safer in the event of postulated accidents. Current manufacturing capabilities are adequate to fabricate a reactor vessel which can accommodate an additional row of fuel assemblies on the core periphery. This design change represents a possible 22.5% increase in core area and therefore a 22.5% reduction in power density. This increase in core size and reduction in core power density result in an improvement in uranium utilization of about 5%. As a consequence of this improved uranium utilization, the fuel cycle costs for the low power density design are about 4% lower than those for the reference composite PWR. These lower fuel cycle costs will have to compensate for the higher capital costs associated with the lower power density design. The NSSS modifications required for this design include an enlarged reactor vessel and core internals, a larger pressurizer, a larger safety injection system, more control rods and drive mechanisms, more in-core instrumentation, and a modified containment structure. The majority of this cost increase is associated with the manufacture of the larger reactor vessel and the construction of the redesigned containment building. It appears that the higher capital cost associated with the lower power density design could be balanced by the savings in fuel cycle costs so that there would be a small positive economic incentive to adopt the low power density design.

The low power density design discussed in this report is considered to be state-of-the-art technology. No demonstration of the concept is felt to be required; only design and licensing is needed before the concept could be offered as a new product. Customer acceptance of the low power density design should be high because of the potentially improved safety margin in this design. This feature, together with improved fuel utilization, should enhance the public image of this reactor concept. Again, because of the increased thermal margin of the low power density core, additional licensing problems should be minimized. Thus, given the proper development funding, a market should emerge for a reactor design that utilizes the low power density concept.

2.1.7 Small PWR Assemblies

The use of smaller PWR assemblies will result in better uranium utilization because of lower burnable poison shim residuals and the

ability to implement a radial core blanket in a standard PWR (within a standard core volume). Lower shim residuals can be obtained because of the smaller quantities of poison that are required to suppress local reactivity when the core is more homogeneous. A radial core blanket can be implemented because the loss in active core volume is much less (less than half) than that if full assemblies were used as a blanket. Indirect uranium utilization improvement may also be obtained through the use of smaller PWR assemblies by providing a mechanism to gain fuel management flexibility and therefore to more fully implement other improvement options, including some that are considered backfittable. The total direct improvement in uranium utilization is approximately 3.5% with about 1% of this improvement resulting from reduced shim residuals and 2.5% from the implementation of radial core blankets.

Two methods of implementing a small PWR assembly design can be envisioned: one in which the assemblies are maintained independent of one another and the second in which groups of assemblies are locked together for purposes of facilitating fuel shuffling and handling. The additional capital costs to implement this design modification are considered small in the context of new plant designs. These additional costs are associated with fuel handling equipment, fuel storage racks, and possibly reactor internals. It appears that the economics of using small PWR assemblies are marginally favorable.

There are a number of potential design problems associated with the small assembly design; nevertheless, this concept is considered technically feasible, presenting no significant safety concerns. The concept is well within the state of the art and primarily requires design analyses and limited testing of the smaller rod arrays. Also, fuel handling equipment and procedures would need development to ensure the safe and efficient movement of the more flexible small assemblies. Finally, fuel alignment and hold-down may need redesign to accommodate the smaller cross section assemblies. All of these potential design problems are felt to be easily accommodated in a new reactor design.

In this brief summary, the major features of each of the nonbackfittable design modifications has been highlighted. A more detailed discussion, including other aspects relating to the desirability of implementing these design modifications are discussed in Section 3 of this report. In the section that follows, the individual design modifications are ranked relative to one another and the rationale for this ranking is presented.

2.2 RANKING OF NONBACKFITTABLE DESIGN MODIFICATIONS

The relative ranking of the individual nonbackfittable design modifications is shown in Table 2-2. The highest ranked design modification is rapid-frequent refueling followed closely by the combined concept of the lower power density core, radial blankets, and core periphery modifications. These three concepts are viewed as a combined package since they all compete for reactor vessel space and since they try to capitalize on the same mechanism-core neutron leakage. Considerably lower on the ranking scale are small PWR assemblies and end-of-cycle stretchout. The lowest ranked concept is the use of higher temperatures and pressures.

The rationale for the ranking shown in Table 2-2 is indicated by the remarks in this table under each of the design modifications. It should be remembered that this rationale must necessarily be subjective and is based on a rather limited assessment effort. It is felt that the present ranking reflects the current understanding and knowledge of the proposed nonbackfittable design modifications and represents a reasonable assessment from which future programs can be initiated.

2.2.1 Nonbackfittable Design Modifications With the Highest Potential

Rapid-frequent refueling has been ranked number one because it offers the highest potential uranium utilization improvement, and the rapid refueling feature of this design modification would have great utility appeal and is consistent with current R&D development programs. The major reservation with this design modification is whether frequent refueling can be economically adopted. Replacement power costs place a large economic penalty on additional outage time, making it difficult for the fuel cycle cost savings to make the overall economics favorable. The combined concept that includes low power density, radial blankets, and core periphery modifications was ranked second because of the potential for good uranium utilization improvement and because no fundamental development work is required to implement these design modifications. The improvement in core thermal margin as a result of lower power density can make this design modification appealing from the viewpoints of both customer acceptance and licensing complexity. In the economic analyses performed as part of this assessment, the design modifications of this combined concept were found to be attractive. With other economic assumptions, however, these design modifications (particularly the low power density design) may be unattractive, especially at high carrying charge rates.

2.2.2 Nonbackfittable Design Modifications With Moderate Potential

The use of small PWR assemblies and the nonbackfittable aspects of end-of-cycle stretchout fall into the category of possessing moderate potential. The uranium utilization improvement directly attributable to these concepts is in the range of 1 to 2% (if in the case of small PWR assemblies the benefit of using radial core blankets is not taken credit for). However, both concepts have utility in their ability to facilitate or promote other aspects of uranium utilization improvement. For example, the use of small PWR assemblies can mitigate to a large extent the higher power peaking exhibited by certain high burnup fuel management schemes. Similarly, adoption of the nonbackfittable features of end-of-cycle stretchout can make the backfittable capabilities of this design modification more appealing by offering the utility more flexibility in its management of cycle energies. In addition, neither design modification requires any fundamental development work to allow implementation of the concept.

2.2.3 Nonbackfittable Design Modifications With Smallest Potential

Although the use of higher temperatures and pressures offers the potential for good uranium utilization improvement, this design modification is felt to have the least potential for successful implementation. It represents the most severe departure from current technology and therefore requires extensive development work with broad implications. The most dramatic development effort would probably be

required in the area of fuel design. The accelerated corrosion rate of Zircaloy at the higher temperatures considered in this evaluation may require radically different fuel designs including alternate cladding material.

The ranking shown in this section is based on the individual assessments which are presented in Section 3 of this report. The individual assessments of the various nonbackfittable design modifications are followed in Section 4 with the major conclusions and recommendations of this evaluation.

Table 2-1

PNL INDUSTRIAL ASSESSMENT
RATING OF NONBACKFITTABLE DESIGN MODIFICATIONS

CONCEPTS	URANIUM UTILIZATION	ECONOMIC (Power Generation Cost)	TECHNICAL (Feasibility)	OPERATIONAL (Plant Availability)	OTHER CONSIDERATIONS (Utility Acceptability)
HIGHER TEMPERATURES AND PRESSURES (1)	5-9%	LOWER	YES, BUT (2)	SAME	ACCEPTABLE SUBJECT TO (2)
RAPID-FREQUENT REFUELING	7-10%	HIGHER (3) ON THE AVERAGE	YES	LOWER	ACCEPTABLE SUBJECT TO (3)
END-OF-CYCLE STRETCHOUT	1-1.5% (4)	SLIGHTLY LOWER	YES	LOWER	ACCEPTABLE SUBJECT TO (5)
CORE PERIPHERY MODIFICATIONS	0.4-5%	UNKNOWN	YES, BUT (7)	SAME	ACCEPTABLE SUBJECT TO (8)
RADIAL BLANKETS	2-5%	SLIGHTLY LOWER	YES	SAME	ACCEPTABLE
LOW POWER DENSITY CORES (6)	5%	SLIGHTLY LOWER	YES	SAME	ACCEPTABLE
SMALL PWR ASSEMBLIES	3.5% (9)	SLIGHTLY LOWER	YES	SLIGHTLY LOWER	ACCEPTABLE

NOTES:

- (1) 500 psig and 1000 psig increases in primary pressure accompanied by 25° F and 50° F increases in primary temperature.
- (2) Depends on how radical the fuel design must be to sustain the higher primary temperatures.
- (3) Economics of frequent refueling strongly dependent on replacement power costs.
- (4) Relative to normal power coastdown.
- (5) Turbine vendor commercial considerations.
- (6) 22.5% reduction in power density and 6.13 batch fuel management.
- (7) Zircaloy shroud and enhanced reflectors (graphite & BeO) require considerable development; it is not clear whether all problems can be solved.
- (8) Requires in-reactor demonstration.
- (9) Of this savings, 2.5% from implementation of radial blanket and 1.0% from reduction in EOC shim residual.

TABLE 2-2

PNL INDUSTRIAL ASSESSMENT
RANKING OF NONBACKFITTABLE DESIGN MODIFICATIONS

HIGHEST POTENTIAL

Rapid-Frequent Refueling

Highest potential uranium utilization improvement
High utility acceptability for rapid refueling feature
Required R&D effort consistent with current outage
improvement development programs

Lower Power Density Core*

More thermal margin
Good uranium utilization improvement
No fundamental development work

Radial Blankets*

Moderate to high uranium utilization benefit depending on blanket
material
No fundamental development work required

Core Periphery Modifications*

Significant departure from current designs
Benefits reduced substantially by use of radial blanket

MODERATE POTENTIAL

Small PWR Assemblies

Moderate improvement in uranium utilization
No fundamental development work
Would help facilitate some backfittable improvement concepts

End-of-Cycle Stretchout

Small potential uranium utilization benefit relative to
backfittable capacity
No significant development work if turbine not modified
Possibly high capital cost-to-benefit ratio

SMALLEST POTENTIAL

Higher Temperature and Pressure

Extensive development work with broad implications
Severe fuel corrosion problems

*These three concepts are viewed as a package in that they all compete for vessel space and try to capitalize on core neutron leakage.

Section 3

ASSESSMENT OF INDIVIDUAL DESIGN MODIFICATIONS

3.1 HIGHER TEMPERATURES AND PRESSURES

One basic method of increasing uranium utilization is to improve the conversion efficiency of thermal energy into electrical energy so that more electrical power generation can be obtained from a given amount of uranium consumed. This may be accomplished by increasing the turbine throttle pressure and temperature fluid conditions. The present assessment was limited to consideration of higher design and operating pressure and temperature conditions in a large current-design pressurized water reactor (PWR) to increase the turbine inlet saturated steam conditions.

To make a significant improvement in the efficiency of a turbine cycle using saturated steam inlet conditions requires a correspondingly significant increase in the Reactor Coolant System (RCS) fluid average temperature. For purposes of evaluation, two cases were selected: a 25°F and a 50°F increase in the reference plant RCS fluid average temperature conditions. This would permit an approximate increase of 275 psi and 555 psi in saturated steam pressure, respectively. These changes in pressure and temperature would improve the overall plant net cycle efficiency by approximately 0.9% and 1.5%, respectively.

The increase in cycle efficiency and hence the increase in electrical output is not linear with the increase in turbine throttle pressure. At elevated pressures, the increase in cycle efficiency achievable with increasing saturated steam pressure is increasingly less significant. In terms of maximizing the cycle efficiency, some of the potential increase in steam pressure should, in practice, be converted into superheat by appropriate steam-generating equipment. An additional several tenths of one percent improvement in cycle efficiency should be realizable.

To preserve core thermal margins during steady-state and transient operations, the over-pressure margin to saturation in the reference plant was considered in selecting the design pressure of the RCS components for the two cases under investigation. The RCS primary side design pressures which result are 3000 psia and 3500 psia, respectively. This represents an increase of 500 psi and 1000 psi to the current 2500 psia design condition.

The increased RCS fluid operating temperature and the increased RCS component design pressure constitute the focal points of the technical and economic evaluation. To facilitate the economic evaluation of this concept, the cost estimate was based upon upgrading the design pressure and temperature conditions of a current 3800 Mwt reactor while keeping RCS component internal dimensions and volumes essentially constant. Potential improvements in turbine throttle conditions by decreasing the temperature difference between the primary and secondary

systems were not considered. Furthermore, it was assumed that fuel of current design would be used with the higher temperature and pressure plant designs; however, the higher operating temperatures may require radically different fuel designs. In the sections that follow, the general technical feasibility and the potential impact on the costs of generating electricity, including the capital and fuel cycle costs of implementing this concept, are addressed.

3.1.1 Uranium Utilization

Improvement in uranium utilization as a result of increasing the moderator temperature (and consequently the secondary pressure) is derived from increasing the reactor plant overall thermodynamic efficiency and from increasing the core conversion ratio. With a higher thermodynamic efficiency, more energy is converted into useful work (i.e., electrical energy) and less is rejected as waste heat. This results in more electrical energy produced per ton of uranium mined. Also, at the higher moderator temperatures, the neutron spectrum is hardened, resulting in greater neutron resonance capture in U-238. The greater neutron capture, in addition to producing more plutonium, also lowers the core reactivity during most of the operating cycle; this lower reactivity, in turn, is compensated by reduced absorptions in the control poisons. At end-of-cycle, when the soluble boron is essentially completely removed, the moderator temperature must be reduced back to its reference value (e.g., 594°F for System 80TM); otherwise, the reactivity tied up in this higher Doppler absorption and in the poorer moderation of the higher temperature coolant will result in a cycle length penalty that would partially counteract the higher thermodynamic efficiency. Thus, this concept (as envisioned) includes a form of end-of-cycle coastdown to regain part of the uranium utilization benefit that would otherwise be lost because of the harder neutron spectrum. Alternately, the fuel lattice can be opened up or wettened, so that at the higher moderator temperatures and pressures, the neutron spectrum is similar to that in current designs. This approach, however, would reduce the potential of actual coastdown because of the trend of increasing (becoming less negative) moderator temperature coefficient with increased water-to-fuel ratio.

In the discussions that follow, it was assumed that fuel of essentially current design would be used in the higher temperature and pressure plant. As will be mentioned later, it may be necessary that radically different fuel designs be used. For example, it may be necessary that a more corrosion-resistant (at high temperatures) cladding material be developed and that thicker cladding be used to prevent collapse onto the fuel column early in life. Such modifications to present fuel designs could adversely affect both uranium utilization and fuel cycle costs.

For a 25°F increase in moderator temperature (500 psig increase in primary pressure), uranium utilization will increase approximately 5.2%. For a 50°F increase (1000 psig increase), uranium utilization will increase about 9.5%. This lower incremental increase in utilization reflects the fact that turbine electrical output is not linear with the increase in throttle pressure. Contributions to the total uranium utilization improvement from increasing moderator temperature and secondary steam pressure are split nearly equally between the two

mechanisms (increased nuclear conversion and higher thermal efficiency). At the lower temperature increment (25°F), thermal efficiency contributes slightly more (about 2.7%), while at the higher temperature increment (50°F), increased conversion contributes more (about 5.0%). Similar reductions in separative work requirements are obtained for this improvement option. With a 25°F rise in moderator temperature, separative work requirements show a 5.8% savings, while for a 50°F rise the separative work requirements are 10.5% lower than those of the reference composite cycle.

If the moderator temperature is not reduced back to its reference value, then there will be little if any improvement in uranium utilization. A greater end-of-cycle fissile inventory will be required in this situation and thus more fissile material will be discharged with the spent fuel. The fuel can be reoptimized to a higher water-to-fuel volume ratio so that at the elevated moderator temperature essentially the same neutron spectrum exists as that in the reference plant design. This choice of fuel designs would not, however, provide a uranium utilization benefit from increased conversion. Under this assumption, the improvement in uranium utilization would be only about half of that mentioned above.

3.1.2 Economic

The incremental capital costs of the higher temperature and pressure reactor design are approximately \$20 million for the 3500 psig design, not including turbine plant changes. The individual contributions to this total are given in Section 3.1.3 along with some of the required development costs. Since the cost of turbine plant equipment is about 85% of that for reactor plant equipment and since the turbine design would be substantially different from current PWR turbines, it is possible that significant additional capital costs may be required. Assuming that these additional costs are also \$20 million (regarded as a conservative value), then approximately \$4 million savings as a result of better fuel cycle costs and higher thermal efficiency would be required to recover the additional capital costs of this improvement option at a fixed charge rate of 10.1%.

Because of the better uranium utilization and greater thermal efficiency of the higher temperature and pressure plant, the fuel cycle costs are lower than the reference composite PWR. For the plant with a 25°F rise in moderator temperature, the fuel cycle costs are about 5% lower over a range of uranium prices from \$40/lb U_3O_8 to \$160/lb. For the plant with a 50°F rise, the fuel cycle costs are approximately 9% lower. It should be remembered that improvements in thermal efficiency, in addition to reducing fuel cycle costs, also reduce capital costs. For a given core thermal power, more electrical output is obtained from the higher temperature and pressure plant. For a 3800 MWt System 80™ PWR, the net electrical output is approximately 1270 MWe. For the 25°F higher temperature system, the net electrical output would be 1304 MWe and for the 50°F higher temperature system it would be 1327 MWe. In terms of mills/kWh, the total power costs are reduced by 2.7% and 4.5% for the 25°F higher temperature plant and the 50°F higher temperature plant, respectively. Since the fuel cycle cost is only about 25% of the total power cost at current U_3O_8 prices (\$40/lb), changes in plant design which increase thermal efficiency have four times the leverage as changes

in plant design that lower only the fuel cycle cost. For the 500°F higher temperature design, this improvement in thermal efficiency represents an annual savings in power generation costs of approximately \$10 million. Improved neutron economy due to higher conversion increases this savings about \$1 million for current U₃O₈ prices. This \$11 million per year savings in power generation costs more than compensates for the added capital costs to install the higher temperature and pressure system. For the 250°F higher temperature design, the savings in power generation cost is about \$6.5 million per year.

The development costs to design, test, and license components to operate at significantly higher temperatures and pressures would be substantial, particularly with respect to a fuel design than can withstand the higher primary temperatures. No attempt was made to quantify these costs.

A study on plant construction time was recently published by Mooz (4). Regression analysis of the historical data base for plant construction times showed that only certain variables appeared statistically significant. These include the date of construction permit issuance, the plant size, the number of plants that had been previously built by the architect-engineer, and whether the plant is a duplicate unit of another plant on the site.

The physical construction time should not be affected by erecting components which will operate at higher pressure and temperature conditions. However, construction time was defined by Mooz to be the elapsed time between the date that the construction permit is issued and the date that the operating license is granted. In this light, if there was a significant departure from previously licensed designs such as a radically modified fuel design, delays at the operating license stage should be expected. Direct and indirect costs, possibly significant, might accrue to the first such units.

3.1.3 Technical

The use of higher primary system pressures and temperatures generally will require components with greater wall thicknesses which will increase significantly the weight (and to a lesser extent the size) of such components. In some instances, these increases in weight and size will exceed current manufacturing limits, while, in others, facilities to test the larger components at the more extreme operating conditions are not presently available.

1. RCS Components

The information on the manufacturability and differential cost estimate for 3500 psia RCS components was developed by C-E's engineering and heavy equipment manufacturing departments. The practical limitations for shop fabrication are plate size and thickness, flange size, welding capabilities, non-destructive testing capabilities including x-ray and post-weld heat treatment facilities.

a. Reactor Vessel

At 3500 psia design pressure, the reactor vessel flange and nozzle shell become the controlling areas. The gross weight of this flange approaches maximum mill/manufacturing capabilities worldwide.

The nozzle shell plate finished thickness (or minimum thickness) would exceed by several inches the current thickness for which plate suppliers guarantee mechanical properties. Plant suppliers would incur development costs to guarantee current material specification requirements for SA-533 plate at the required thickness.

At the 3500 psia design condition, the primary nozzle ligaments for the primary nozzle locations are marginally acceptable. Also, the design is approaching limits regarding existing press capabilities.

b. Steam Generator

At 3500 psia design pressure, the steam generator tubes and tube sheet become the controlling areas. The tubesheet diameter and thickness approach the maximum mill capabilities worldwide. Also, the thickness of the tubesheet would approach the maximum thickness capability for existing tube hole drilling equipment.

Due to the increased primary-to-secondary differential pressure requirements, the tube wall thickness would be increased for a 3500 psia design pressure. To compensate for the reduced effectiveness of the heat transfer surface area, the tube length would have to be increased to a point where the maximum tube length capability of suppliers is reached or exceeded. Development and additional tooling by tube suppliers would be required. The differential cost estimate prepared for the steam generators did not account for any required increase in heat transfer surface area.

The maximum finished weight of an RCS component for shop handling is that of the steam generator. The weight of a steam generator designed for 3500 psia would be marginally within existing lifting capabilities.

c. Pressurizer/RCS Piping

The manufacture of these components for 3500 psia does not approach any existing shop or supplier limitations. For the 3500 psia design, a design temperature of 700°F was selected for the RCS components except for the pressurizer. This is the ASME Boiler and Pressure Vessel Code, Section III limit for ferritic materials which are used extensively throughout the RCS. For the pressurizer, a design temperature of 750°F was selected. To use ferritic steels this would require a Code change which, if possible, is an expensive and time-consuming process. More expensive austenitic steels (currently qualified to 800°F) could be substituted, but additional material would be necessary due to their lower strength characteristics. Neither cost was included in the differential cost estimate prepared for the pressurizer.

d. Reactor Coolant Pumps (RCP)

It is considered technically feasible to design, fabricate and test a RCP with a 3500 psia design condition. The major problem area is considered to be the shaft seals. A two-year seal development program costing two million dollars is envisioned. Bearings, heat exchangers and shaft would have to be resized.

Present manufacturing facilities are considered adequate to handle a 3500 psia design; but testing facilities are not. RCPs are full-flow tested at pressure and temperature. Expansion of existing test facilities would cost 2.5 to 5.0 million dollars depending upon approach.

e. Transportation of RCS Components

Water transportation would be more expensive for 3500 psia design RCS components. Barge stability considerations and river depth limitations at the manufacturing facility would require that components be shipped on two or more barges. The extra cost would be a function of the trip length but a representative increment is a half-million dollars per plant.

Road transportation would remain within the current state of the art for hydraulic platform transporters. More platform modules and an increase in prime mover capacity over current equipment would be required. The cost of a unit delivery overland would increase by an estimated half-million dollars.

Rail transportation is constrained by the load-bearing capability of the rail beds. Additionally, the maximum capability of existing schnabel cars could handle the reactor vessel but not the steam generator for a two-loop 3500 psia design plant. Sites inaccessible overland or by water would mandate significant additional transportation costs and probably more steam generators of smaller size.

f. Field Fabrication of RCS Components

Practical limitations for field fabrication would generally be the same as established for shop fabrication. The significant exception is post weld heat treatment capability. This is a major if not insurmountable problem for field fabrication as currently envisioned. Current site rigging and lifting capability exceed current in-shop capability by about a factor of two.

g. High Energy Line Break Impact

Nuclear power plant components, systems and structures must be designed to withstand the consequences of the assumed rupture of lines bearing high energy fluids. System blowdown has a direct bearing on the reactor vessel and steam generator internals design as well as the supports for the major RCS components. Differential cost increases for the higher pressure design are unknown.

h. Increased Cost of 3500 psia RCS Components

Subject to the preceding comments, the differential cost for 3500 psia design RCS components is estimated to be 16 million dollars. Development costs, first-time engineering design and analysis costs, and the cost of increasing manufacturing capabilities are not included. Therefore, the differential cost estimate reflects a mature industry receiving multiple orders over which first-time costs have been amortized.

For the purposes of this study, the differential cost for 3000 psia design RCS components may be considered to be half of that for the 3500 psia design RCS components.

2. Balance-of-Plant (BOP) Components Supplied By a Reactor Vendor

Portions of the Chemical, Volume and Control System (CVCS), Safety Injection System (SIS), Shutdown Cooling System (SCS) and Primary Pressure Relief System (PPRS) are designed for exposure to reactor coolant at RCS operating temperature and pressure. Piping, valves, instrumentation and certain components of these systems subject to the proposed increased RCS pressure and temperature would have to be upgraded. Specific items include:

- RCS Relief Valves
- Regenerative Heat Exchanger
- Charging Pumps
- Seal Injection Filters and Heat Exchanger
- Letdown Control Valves
- Chemical Addition Pumps (Primary Side/Secondary Side)
- High Pressure Safety Injection Pumps and Relief Valves
- Primary Sample System Heat Exchanger and Vessels
- Secondary Sample System Heat Exchanger
- Miscellaneous Valves, Piping and Instrumentation

The differential cost for these components to be designed consistent with a 3500 psia RCS design is estimated to be one to two million dollars. Development costs, if any, and first-time engineering, design and analysis costs are not included.

3. Containment Building, Structures and Systems

The compartments or cavities surrounding the major RCS components are designed to withstand the consequences of the assumed rupture of main coolant piping, main steam lines and main feedwater lines. Differential cost increases for design of these subcompartments for the increased blowdown loads from the higher pressure design are unknown.

The compressed air supply for operating valves for the higher pressure design would have to be considered for upgrading. The containment air cooling system would have to be evaluated for higher heat loads expected with a RCS operating at elevated temperature conditions.

The environmental qualification requirements for equipment required to function in the post-accident environment would have to be investigated to determine if equipment currently qualified would be acceptable with the higher pressure design. If not, there would be costs incurred by the industry to re-qualify necessary electrical and mechanical components with harsher environmental conditions.

The containment building is sized in relation to the fluid-stored energy within the RCS. The basis for this study involved keeping RCS component internal dimensions and volumes essentially constant. Therefore, the RCS fluid-stored energy in the 3500 psia design when operating at full power would increase by approximately three percent. Assuming a directly proportional increase in containment building costs would increase the plant capital cost by about 1.7 million dollars. A detailed design of 3500 psia components might permit a reduction of internal volumes to negate this effect if it were a design basis.

4. Turbine Plant

Portions of the turbine plant are designed for exposure to the steam generator operating pressure and temperature conditions. Areas of the plant which would be affected include:

- Main Steam Piping, Safety Valves, Isolation Valves,
- Turbine Stop Valves, Throttle Valves, High Pressure Turbine,
- Turbine Bypass System (possible impact on condenser inlet region),
- Portions of Main Feedwater Piping, High Pressure Heaters, Control and Isolation Valves, Main Feedwater Pumps,
- Portions of Auxiliary Feedwater Piping, Control and Isolation Valves, Auxiliary Feedwater Pumps,
- Pump Drivers, Emergency Diesels, Component Cooling Water, and Turbine Plant Instrumentation.

The impact on the turbine is expected to be significant but within the state-of-the-art. A downgraded 2400 psig front-end of a fossil turbine might be added to a low-pressure saturated nuclear unit or the front end of a conventional nuclear turbine might be upgraded.

The turbine generator and associated balance-of-plant equipment are approximately 1.5 percent lower in capital cost for a standard fossil supercritical cycle (3500 psig/1000°F/1000°F) compared to a standard fossil subcritical cycle (2500 psig/1000°F/1000°F). Presumably the 50 percent increase in operating pressure is offset by the 40 percent reduction in steam specific volume, thereby permitting a smaller-sized unit with attendant cost savings. The change in steam conditions associated with the 50°F increase in RCS fluid average temperature of this study are similar.

Whether the superheat fossil turbine cost data trend is representative of what to expect for saturated nuclear units is problematical. Any further development of the higher pressure/temperature concept for plant thermal cycle improvement should include significant turbine-vendor involvement.

5. Reactor Core

Fuel rod design changes would be needed due to the proposed increase in RCS operating pressure. Depending upon the nature of these modifications, uranium utilization and/or operating margins may be impacted. A higher helium gas fill pressure in possible combination with a thicker cladding wall would be necessary to meet the current licensing criterion which prohibits the occurrence of cladding collapse. Even if higher fill pressures are used to maintain the same operating differential pressure across the cladding wall, collapse is expected to occur in about two-thirds the time expected at current (lower) operating temperatures. This is counterproductive to the concept of improving uranium utilization through the use of high-burnup fuel.

The propensity for clad ballooning during a LOCA would be increased due to the higher clad operating temperature and the higher pressure differential across the cladding following RCS depressurization. A thicker cladding to preclude ballooning would have an adverse impact on uranium utilization.

A 50°F increase in RCS fluid average temperature results in an even larger increase in fuel pellet centerline temperature due to reduced thermal conductivity of uranium oxide at elevated temperatures. PWR fuel operating with a linear heat generation rate (LHGR) necessary to reach the pellet-clad interaction (PCI) failure threshold would have the failure probability increased to about 10 percent due to enhanced gas release and interaction induced stresses and strains at the elevated coolant temperature. Therefore, a reduction in power capability of approximately 0.5 kW/ft would be required to provide equivalent margins for the PCI phenomenon as well as to provide equivalent margins to centerline melt, if necessary.

A 50°F increase in RCS fluid average temperature will produce a dramatic increase in the waterside corrosion rate of Zircaloy. Generally, the corrosion weight and thickness gain have been expressed as an exponentially increasing function of $-1/T$, where T is the absolute temperature. Two rates have been recognized: one for pretransition and one for post transition. The latter is significantly greater than the former and represents an undesirable PWR operating condition. The transition point at 703°F is reached in 40 percent of the time required at 653°F. Furthermore, about 2.5 times as much corrosion product will be added to the corrosion layer per unit time at the elevated temperature following transition to the faster weight gain. Due to the significant insulating effect of thick oxide layers, the cladding and fuel pellets will operate significantly hotter as the corrosion layer builds up.

In order to offset these basic adverse effects without impairing uranium utilization, it may be necessary to radically change current PWR fuel designs and operating characteristics. Such concepts would include a fuel which permits cladding collapse during normal operation as well as the ability to operate modern PWR cores essentially corrosion free. Design, development, licensing and demonstration costs are unknown.

3.1.4 Operational

The operability of a PWR whose nominal temperature and pressure are significantly higher than current plants will need to be fully assessed. For example, the 50°F increase in RCS fluid average temperature would result in RCS operation close to the critical point, particularly in the pressurizer. If there are any thermal/hydraulic problems, steady-state or dynamic, when approaching the critical point, it would be detrimental to operate with such a large increase over current operating conditions. Experience in other fields such as fossil fuel technology should be reviewed on this point if the concept is pursued.

Plant availability should be affected marginally, if at all, by the higher operating conditions. Increasing RCS operating pressure might lead to enhanced RCP seal problems and pressurizer safety valve leakage. However, problems experienced of this nature have been attributed to installation and maintenance practices. The major portion of a nuclear power plant's unavailability is attributed to the non-nuclear portions of the plant. Mechanical and electrical component failures and systems interaction problems dominate nuclear power plant unavailability. Increasing fluid system operating conditions should not by itself make the plant inherently less reliable. Therefore, the plant capacity factor should remain essentially the same.

The number and quality of staff should not be impacted significantly as a result of higher temperature and pressure operation unless these fluid conditions cause increased maintenance problems (which they should not, as stated in the preceding paragraph, as long as the plant has been designed for such conditions). Consequently, radiation exposure to the staff should not be impacted by implementation of this design concept.

3.1.5 Other Considerations

Utility Acceptability - One of the positive aspects of increasing operating pressure and temperature conditions as a method of improving uranium utilization is that utility acceptance presumably would be high as long as the plant appeared "conventional" and the "complexity" of the plant did not increase. A radical departure from current PWR fuel designs, if required, would be detrimental in this regard.

Date of First Commercial Operation - As mentioned in Section 3.1.3, there are many components that would require redesign and testing, not the least of which is the fuel itself, before a reactor plant employing significantly higher temperatures and pressures could be offered in a commercial bid. It is estimated that such design, development and testing would take on the order of ten years to complete, making the earliest date of first commercial operation after the year 2000 (possibly about 2005).

Nonproliferation Compatibility - There would be essentially no difference in the proliferation resistance between a plant using higher temperatures and pressures and a current-design PWR. Somewhat higher plutonium concentrations may be available in the discharge fuel because of

increased conversion if the fuel is not redesigned to soften the neutron spectrum. However, since the access to such plutonium would have the same restrictions as exist currently, and since the increases in plutonium concentration would be relatively small, the proliferation resistance is judged to be similar to that of present plants.

Viability With Recycle - The use of a higher temperature and pressure design is compatible with recycle. As mentioned previously, somewhat higher plutonium concentrations may result if the neutron spectrum is not softened by lattice reoptimization. This higher plutonium production would reduce the uranium requirements of the recycle mode.

Interaction With Other Concepts - The higher temperature and pressure concept is compatible with the other improvement concepts. It will, however, reduce somewhat the benefits of the low-power density core. The reactivity benefit from reduced Doppler absorption will be diminished or negated. Also, the added thermal margin of reduced power density operation will be countered by increased moderator temperature.

3.2 RAPID-FREQUENT REFUELING

One well-known method of achieving improved uranium utilization in advanced reactor designs is to increase the number of fuel batches resident in the core, thereby reducing core average exposure and absorptions in fission products. Associated with this increase in the number of fuel batches is an increase in the frequency of refueling when the fuel discharge exposure is fixed. An additional benefit is obtained in this instance from the reduction in neutron absorptions in control poisons since less reactivity and therefore less soluble boron is required for the shorter cycles when frequent refueling is practiced. Such freedom to reduce the excess reactivity required to drive the system to the end of cycle will enable the reactor designer to use advanced fuel management with reduced enrichment requirements.

Frequent refueling is impractical with today's PWR designs since the additional one-to-two-month outage span each year would be economically unacceptable to utilities. Therefore the objective of this preliminary assessment was to evaluate the technical and economical feasibility of designing future plants for rapid-frequent refueling. Rapid-frequent refueling is considered feasible for new plant designs but only after considerable costly development effort is directed toward reducing those critical path activities which must still be accomplished during a refueling/service outage of the nuclear plant.

For study purposes, we have considered one normal annual refueling/service outage in order to accomplish long-term repairs along with preventive and normal maintenance operations. An additional refueling-only outage will be accomplished with a goal of 11-13 days outage span. Outages will be Spring and Fall, respectively, allowing base load operation during traditional peak generation periods.

Extensive development, equipment and construction costs will be required to design and build plants having this rapid-frequent refueling capability. A more comprehensive economic analysis should precede development of this concept in order to weigh the incurred costs against the long-term political and economic benefits of improved uranium utilization.

3.2.1. Uranium Utilization

As discussed previously, the use of more frequent refueling with more in-core batches results in a lower core average exposure and, consequently, less parasitic neutron absorptions in fission products. In addition, less excess reactivity is held down by control poisons when the cycle length is six months compared to 12 months. This also results in lower neutron capture in parasitics (in this case control poisons) and therefore better neutron economy. The net result of these effects is that less fissile material is consumed and/or discharged in the spent fuel per unit of energy produced.

For the case of semiannual refueling, the reference five-batch composite cycle becomes a ten-batch composite cycle. The U_3O_8 requirements for the semiannual cycle are about 7.8% less than those of the annual

cycle. This results in a 30-year cumulative uranium demand of about 4415 short tons/GWe. The separative work savings resulting from frequent refueling is even larger, approximately 11%, giving a 30-year cumulative requirement of about 2900 MT SWU/GWe. These fuel resource requirements were determined for equal energy out of both the annual and semiannual cycles, that is, for the same average annual capacity factor. In reality, since the semiannual cycle will require an additional refueling outage, the average capacity factor for a plant on this cycle will be lower than that of a plant on an annual cycle. Nevertheless, the appropriate way to judge the fuel resource benefits of increasing the frequency of refueling is on an equal energy basis.

3.2.2 Economic Assessment

The incremental capital costs associated with the rapid refueling equipment are estimated to be about \$24.3 million or about \$20/kWe. This investment in plant equipment can be compared with a potential maximum annual savings in fuel cycle costs of about \$4 million at current U_3O_8 prices. It should be remembered, however, that this equipment will help reduce the time for each refueling outage (i.e., whether annual or semiannual refueling is adopted). Since current replacement power costs of a large PWR (1270 MWe) are in the range of \$0.64 million per day, the equipment additions can be easily justified if they result in as little as 4 day's savings in outage time per year (\$24.3 million $\times$ 10.1% fixed charge rate = \$2.5 million per year). Thus, the decision whether or not to install the rapid-frequent refueling equipment can be divorced almost completely from whether frequent refueling will be practiced.

The decision on whether to practice frequent refueling must be made on the basis of fuel-cycle cost savings, the time it takes to accomplish the extra refueling, and the actual replacement power costs of the plant being considered for frequent refueling. Figure 3-1 shows the energy costs (fuel cycle costs plus replacement power costs) for semiannual refueling and \$40/lb U_3O_8 . This figure shows the impact on energy costs of additional outage time as a function of replacement power costs. It can be seen that with excessive outage time, the energy costs can be easily doubled when high replacement power costs exist.

The intersections of the different energy cost curves for the various replacement power costs with the reference composite fuel cycle costs give the breakeven extra outage time as a function of replacement power costs. Several such curves are shown in Figure 3-2 for different assumed U_3O_8 prices. In a simplified way, a utility can decide whether or not to practice semiannual refueling merely by selecting the replacement power cost and U_3O_8 price that best describes its specific economic situation and then using these selected values with Figure 3-2 to determine the extra refueling outage time that gives breakeven energy costs with annual refueling. If the extra refueling outage could be accomplished within this time, then it would be economical to practice semiannual refueling. If the extra refueling outage would take longer than this breakeven time, then an annual refueling cycle should be chosen.

3.2.3 Technical

The refueling/maintenance outage has been demonstrated by many previous studies to be the largest single contributor to nuclear plant unavailability. The outage critical path is made up of many lengthy operations including the fuel transfer operation. Some of these operations can be reduced in existing plants, but major reductions in time and manpower can only be obtained by addressing the overall plant design and will therefore apply to advanced reactor designs only.

Referring to Figure 3-2, it is reasonable to assume that the maximum additional refueling-only outage required for improved uranium utilization should not exceed that time which gives break-even energy costs. Preliminary studies indicate that achievement of a 11-13 day span for refueling can be considered feasible with a revised plant design, but only if the following developments are accomplished:

1. An advanced 10-batch, semiannual refueling strategy is adopted with approximately 24 assemblies loaded, 24 assemblies discharged, and 85 assemblies shuffled in an outage critical path time period not exceeding 22 hours. See optimum critical path schedule Figure 3-3. This minimum shuffle fuel management scheme would be accomplished each refueling cycle utilizing cantilever-supported dual fuel handling machines. Development of the dual handling concept, a major departure from the standard bridge and trolley machine, has yet to be accomplished.

Shuffling schemes that leave a large fraction of the fuel untouched in existing core locations may be difficult to realize because of fuel distortions such as assembly bowing. Presently, some utilities move a substantial portion of the fuel to create a relatively large empty space in the core to facilitate shuffling of fuel. One such technique termed a "modified moving slab" removed two entire rows of assemblies from one edge of the core. The proper assembly for the first location is then inserted, then another assembly is removed. Subsequent assemblies are replaced and removed while maintaining the two-row gap between the old and new fuel assembly locations. This gap appears to move through the entire core until shuffling has been completed. If distortions in fuel geometry (such as bowing) persist, then the savings in time attributed to shuffling fewer assemblies may be unachievable.

Fuel inspection activities involving non-discharged assemblies, including those made necessary by licensing requirements, must be limited to the single long outage each year. Ratchets in licensing requirements for fuel inspection as well as for increased assurances of proper fuel loading could potentially frustrate the implementation of rapid-frequent refueling.

2. All other critical path activity spans must be compressed into 234 hours which constitutes the balance of an 11-day-outage span beginning with reactor shutdown and ending with power ascension.

Many activities may be shortened through procedural improvement, code and regulatory relief, and design changes. Other activities will be eliminated by balance-of-plant design changes.

Major activity blocks comprising the rapid-frequent refueling outage critical path, as shown on Figure 3-3, will not differ markedly from present-generation activities. However, some activities, such as cavity seal installation and removal will be eliminated through development of permanent seals and most operations will be compressed by development and design of radical new equipment and procedures.

The following critical path activity/time blocks are identified as development goals for the semiannual "short" refueling-only outage and will (in addition) provide major reductions in the refueling/service outage:

<u>Activity Block</u>	<u>Time (Hours)</u>
Reactor Shutdown-Vent	36
Head Removal	38
Cavity Clean/Fill	18
Upper Guide Structure Storage Set	12
Fuel Transfer-Mapping	22
Cavity Drain/Decontamination	6
Reassembly Head Area	66
Vent/Testing/Heatup	10
Physics Tests-Power Ascension	48
Total Hours	256

The longer refueling/service outage critical path, which is shown on Figure 3-4, will take advantage of all equipment and procedures developed for the alternating short outage, but must allow additional time for the following annual inspection and preventive maintenance activities:

<u>Activity Block</u>	<u>Time (Hours)</u>
Ex-core Detector Replacement & Fuel Handling Machine Maintenance	36
Engineered Safeguards Testing	10
Inspection & Maintenance of Reactor Coolant Pumps	24
Inservice Inspection of Steam Generators & Reactor Coolant System Components	24
Turbine Inspection & Maintenance	126
Additional C/P Hours	220

Nuclear steam supply system and balance-of-plant system improvements which will contribute to the rapid-frequent refueling concept will include, but will not be limited to, development and/or design improvements in the following areas:

- Dual Handling Machine, Cantilever Supported
- Improved Startup Physics Test Procedures
- Integrated Head/Lift Arrangement
- Reactor Coolant Pump Seal Improvement
- Equipment Diagnostic Systems
- In-Service Inspection Improvement
- Fuel-Shuffling Computer Program
- Leaktight H-X Tube Sleeves
- H-X Design/Material Improvement
- Refueling Machine Full Automation
- Special Tooling
- Refueling Cavity Decontamination Equipment
- Containment Design Optimization
- Improved Training and Procedures
- Permanent Cavity Seal
- Outage Management Tools
- Supplemental and Improved Containment Rigging
- Increased R/A Material Storage
- Multiple RV Stud Tensioning
- Improved Containment Integrated Leak Testing
- System Valve Design/Maintenance Improvement
- Improved T/G Design and Maintenance Procedures

Development and application of advanced design concepts leading to these improvements is expected to significantly reduce refueling time, labor, and radiation exposure to plant personnel. However, the development cost and additional construction costs must still be accurately defined and weighed against the potential for significant economic benefit discussed in Section 3.2.2.

A preliminary assessment of required R&D costs for NSSS and balance-of-plant hardware changes results in an estimate of \$17.7 million (July 1980 \$) to develop equipment to a firm proposal stage for the rapid-frequent refueling concept.

Plant construction time is not expected to increase beyond the ten-year base case. However, the incremental capital cost would be estimated at \$24.3 million per unit or approximately 1.5 percent above the average cost of units which will attain commercial operation during the 1980-81 period.

Although some of the individual outage improvements are backfittable, the overall rapid-frequent refueling concept is not felt to be backfittable. Excessive down-time and capital investment would be required to modify an existing reactor to the point where it would be practical to practice frequent refueling.

3.2.4. Operational

The operational impact of developing and incorporating the rapid-refueling concept should be positive and readily acceptable to utility operations since its desirable benefits are obvious, even without considering uranium utilization improvement. Maintenance management at many nuclear stations has already been adopted and is capitalizing on

maintenance planning and scheduling policy for optimum utilization of unscheduled or forced outages. By incorporating on-line diagnostic monitoring of sensitive plant equipment along with carefully planned preventive maintenance programs many of the outage time-consuming equipment problems can be eliminated.

The key is advance planning and scheduling so that mobilization of proper personnel and spares can be accomplished rapidly during any outage. With continuing development of improved predictive methods and planning techniques, utilities can expect fewer surprises during the normal refueling outage.

Reactor disassembly and fuel handling operations contribute only 6 to 8% of the yearly integrated radiation exposure in a typical LWR plant. Normal inspection, maintenance and operational activities contribute the balance of plant occupational exposure. For this reason, the effect of frequent (or semiannual) refueling is expected to increase total exposure only slightly. Improved planning, which would be expected to accompany a frequent refueling concept, should offset most of the differential. The net increase would be minimal.

Although it is difficult to predict the regulatory climate during the era in which advanced design reactor plants will be operational, nothing has been found in NRC regulations or in rulemaking intent that would specifically impact sub-cycle (semiannual) refueling. Increasing regulatory exposure is not expected to present additional future problems to either the Vendor or to the Licensee should a rapid-frequent refueling concept be developed and adapted to new plants.

3.2.5 Other Considerations

Utility Acceptability- The development program for rapid-frequent refueling would be expected to span 3 to 5 years. Improvements would be incorporated into system proposals shortly thereafter. Prompt utility acceptance of plant improvements leading to shortened refueling outages is assumed since (as stated in Section 3.2.2) the decision could be justified by utility management on availability basis alone.

Date of First Commercial Operation- Earliest commercial operation can be in the range of 10 to 12 years after system proposal. This would place earliest commercial operation of this concept within the 1994 to 1998 span.

Nonproliferation Compatability- The use of rapid-frequent refueling has essentially no impact on the proliferation resistance of the PWR. The same number of fuel assemblies are loaded and discharged each year as in the reference annual cycle and approximately the same quantity of plutonium is bred.

Viability With Recycle- The use of rapid-frequent refueling would enhance somewhat the fuel resource utilization and economics of the recycle mode. The improved neutron economy that results from reducing absorptions in fission products and control poisons gives a uranium utilization improvement of about 5%. This fuel resource benefit together

with the capability to recycle the bred and residual fuel in a shorter time after discharge from the reactor (e.g., 1 1/2 years versus 2 years) will provide some additional economic incentive in the recycle mode.

Interaction With Other Concepts- This design concept is compatible with all the backfittable concepts and most of the non-backfittable concepts. Depending on how they were implemented, the low power density concept and the small assembly concept could require more fuel handling and therefore more outage time, which would be detrimental to the rapid-frequent refueling concept.

Figure 3-1
EXTENDED BURNUP
SEMI-ANNUAL REFUELING
\$40/LB U_3O_8

Figure 3-2

ALLOWABLE EXTRA OUTAGE TIME FOR SEMI-ANNUAL REFUELING
TO OBTAIN BREAKEVEN ENERGY COSTS WITH ANNUAL REFUELING
(EXTENDED-BURNUP CASE)

Figure 3-3
SHORT "REFUELING-ONLY" OUTAGE CRITICAL PATH

Figure 3-4
REFUELING/SERVICE OUTAGE CRITICAL PATH

3.3 END OF CYCLE STRETCHOUT

End of cycle stretchout may be accomplished by several different methods. The most common method, which has been practiced by a number of utilities primarily to obtain cycle length flexibility when it was advantageous to delay a refueling outage, is normal power coastdown. Power coastdown is accomplished by reducing thermal power and secondary pressure so that the lower fuel and moderator temperatures and lower xenon levels provide excess reactivity to allow the reactor to operate beyond its nominal end-of-cycle life and thereby accumulate additional burnup on the fuel. At the point where the core is depleted of excess reactivity (i.e., when soluble boron is at its minimum value), a mismatch between reactor power and secondary steam demand occurs. To balance the primary and secondary plants, the average moderator temperature decreases, providing the excess reactivity needed to match reactor power with steam demand. In order to transfer the same amount of heat to the secondary side with a lower primary temperature, the secondary steam pressure (and temperature) must decrease. With the reduced secondary pressure, the ability of the turbine admission valve to pass the rated steam flow to the turbine becomes the limiting factor. Some compensation is provided by opening the admission valve to its wide-open position. However, after reaching the valve-wide-open (VWO) position, the power level of the plant decreases because the turbine can no longer draw the rated power.

Depending to some extent on the plant design, the core thermal power can be kept constant at the VWO position by decreasing the feedwater temperature. In this operational mode, called feedwater-pressure augmentation (FWPA), extraction steam flow to the feedwater heaters is sequentially decreased, and the throttle valve is maintained in the fully open position. With a lower feedwater temperature, the secondary fluid can experience a greater enthalpy rise for a given steam generator outlet pressure. In addition, the decrease in extraction steam flow for the feedwater heaters permits lower pressure operation before the volumetric flow limit on the turbine admission valve is reached. This mode of operation permits extended operation at rated thermal power after the VWO position is reached but at a lower thermal efficiency. Once rated thermal power can no longer be transferred to the turbine, power coastdown can be initiated to proceed further with cycle extension. The point at which power coastdown must be initiated may be dictated by operational limits on the turbine (i.e., thrust loadings, moisture content, etc.) rather than by the ability to transfer thermal power to the turbine. This depends on the specific turbine design and the limitations adopted by the turbine manufacturer for warranty purposes.

Both power coastdown and, to a limited extent, feedwater-pressure augmentation can be practiced in current design PWRs. The potential operation beyond present capabilities is the subject of this assessment. Such operation can be accomplished by enhancing the mechanisms that allow continued operation past the normal reactivity lifetime and/or by extending the capability of the turbine plant to transfer more power from the reactor at lower secondary temperature and pressure conditions.

Examination of the nuclear characteristics of the core reveal little latitude to enhance the mechanisms that provide additional reactivity to the core. The reactivity contribution due to the negative moderator temperature coefficient (MTC) is by far the most significant factor. Unfortunately, the design modifications that result in a more negative MTC reduce uranium utilization. In particular, the MTC can be made more negative by reducing the water-to-fuel ratio, but this change is in the opposite direction as that incorporated into the reference composite PWR to reduce U_3O_8 requirements. Another mechanism that contributes to a more negative MTC is increased core average exposure. If the fuel discharge exposure is extended substantially beyond that of the reference composite PWR (i.e., beyond 57,130 MWd/t) then a more negative MTC will result. This mechanism was not examined in the present assessment because it is a natural evolution of fuel development that will depend on the satisfactory performance of fuel at lesser burnups. The potential for a more negative MTC is indicated in Figure 3-5 which shows the change in reactivity with decreasing moderator temperature for a standard-burnup core (20,000 MWd/t core average) and an extended burnup core (30,000 MWd/t core average). Similar increases may be possible with the next incremental extension of burnup. It should be remembered, however, that limitations are usually placed on how negative the MTC can become as a precaution against severe cold water accidents (e.g., steam line break). Such limitations may ultimately restrict the potential gains from this mechanism.

Very little, if anything, can be done to the Doppler effect or the xenon worth to enhance their contributions to reactivity gain during coastdown. Techniques that change these mechanisms do so for the entire cycle and have no increased effect during the coastdown evolution. Such changes have been examined in other assessments of both backfittable and non-backfittable improvement options.

The area that will therefore be evaluated in this assessment is the potential for extending the capability of the turbine plant to transfer more power from the reactor. This capability is determined by the secondary plant characteristics. The sizing of the steam system, e.g., the turbine admission valve, and the margins allowed in the turbine warranties determine how far the secondary pressure can be lowered before power must be reduced. The most severe restriction appears to be on warranted turbine operation. The discussion that follows strictly applies to a specific turbine and turbine manufacturer. However, it is felt to be representative of the turbine operating limitations that currently exist for large-capacity PWR turbines.

Turbine output is limited when feedwater heaters are taken out of service because of the redistribution of steam flow within the turbine and the variation of component pressures from their design values. When the feedwater heater taken out of service is the highest pressure heater, steam which was formerly extracted from the turbine to heat the feedwater now passes through the turbine. The turbine output increases in this situation if the admission valve is held at a constant opening. When the feedwater heater taken out of service is not the highest pressure heater, the extraction steam redistributes to the next higher pressure heater.

In this situation, the turbine output decreases if the admission valve position is held constant. In either event, the turbine-generator cycle efficiency decreases.

The net results of these effects are changes in stage pressures, stage pressure drops, and steam flows throughout the turbine. As a consequence of these changes, some of the turbine bucket, diaphragm, and thrust loadings are increased beyond their design values. To prevent overstressing the turbine, therefore, the throttle valve opening must be reduced until the loadings are equal to or less than their design values.

The above-mentioned restriction means that power level must be reduced at some point when the feedwater temperature is lowered during FWPA. Based on discussions with the engineering staff of a major utility, a 25°F feedwater temperature drop is clearly allowable without reducing turbine load. This temperature reduction is obtained by isolating one string of the highest pressure heaters (two heaters total). A feedwater temperature reduction of 50°F (isolating both strings of high pressure heaters) appears possible, but this would have to be evaluated and approved by the turbine manufacturer. A further reduction in feedwater temperature is possible by taking more heaters/strings out of service, but this would require a reduction in turbine load.

It is in this last area, taking more feedwater heaters out of service, that constitutes the principal area of potential improvement for end-of-cycle stretchout. Subsequent remarks assume that this can be accomplished by one of several techniques (to be discussed in Section 3.3.3).

3.3.1 Uranium Utilization

The impact of end-of-cycle stretchout on U_3O_8 requirements is shown in Table 3-1. The values presented in this table were obtained by holding the fuel enrichment fixed and determining the equilibrium cycle length extension that could be achieved by repeated use of EOC stretchout. For purposes of this assessment, the reference EOC stretchout is that obtained with normal coastdown. This reference was chosen since existing plants could employ this technique and gain the indicated U_3O_8 savings if they so desired. They could not, however, practice FWPA to the extent indicated in the table. Thus the additional benefit of employing this non-backfittable improvement option is about 1.0 - 1.5% depending on the minimum plant electrical output to which the reactor is operated. An identical savings in separative work would be achieved using this method of assessing the benefits of EOC stretchout. If the cycle length were held fixed and the fuel enrichment reduced, a slightly larger separative work benefit would be calculated because of the nonlinearity of separative work with fuel enrichment.

3.3.2 Economics

Because of the reduced requirements for U_3O_8 and separative work, the fuel cycle costs when end-of-cycle stretchout is employed are lower than when the reactor is shutdown at its nominal end-of-reactivity lifetime. To account for the fact that the reactor is operating at less than its

rated electrical output, replacement power costs must be combined with the fuel cycle costs to get the actual energy costs for the cycle. The capital costs of the reactor with and without the use of stretchout would also have to be included if the additional capital costs to implement feedwater-pressure augmentation (FWPA) turned out to be significant. These costs should not be large unless a radically different turbine design were used to facilitate FWPA. Otherwise, plant changes would be restricted to some additional secondary plant pipes, valves, and instrumentation to allow extraction steam from the turbine to be diverted from the feedwater heaters and used (or dumped) elsewhere in the steam plant. It is estimated that such secondary plant changes would cost 3-5 million dollars, excluding major turbine redesign.

Figure 3-6 shows the relative energy cost of the composite improved cycle as a function of the number of days of cycle stretchout for normal coastdown and for FWPA with several different feedwater temperature reductions. The results have been normalized to the "no stretchout" energy cost. Also shown is the sensitivity of the results to the magnitude of the replacement energy cost.

As shown in this figure, the use of FWPA is more economical than normal coastdown, with the additional savings increasing as a function of the feedwater temperature reduction. This trend is quantified in Figure 3-7, where the maximum savings in energy cost is shown versus the feedwater temperature reduction. The additional economic savings obtainable by the use of FWPA (compared to normal power coastdown) ranges from 1.1% at 20 mills/kWh replacement power cost to 0.7% at 80 mills/kWh replacement power cost. This represents an annual cost savings of between \$360,000 and \$570,000 for replacement power costs of 80 mills/kWh and 20 mills/kWh, respectively. It should be remembered that currently, the national average replacement power cost is 21 mills/kWh, with a range of 3 mills/kWh to 45 mills/kWh, depending primarily on whether coal or oil is used to generate the replacement power. If a utility employed end-of-cycle stretchout during low load conditions when it could get the replacement power from its own generating units at only the differential fuel cost (difference between nuclear and fossil fuel costs), then the replacement power costs would be well below the values used in this analysis and the economic savings would be greater.

The effect of higher U_3O_8 prices is to extend the optimum stretchout time and to increase the savings in relative energy cost. Particularly at the higher replacement power costs, a larger differential savings in energy cost is calculated between FWPA and normal power coastdown, on the order of 1.2% at 80 mills/kWh replacement power cost.

The impact on construction time should be negligible because the plant changes would be restricted to only a few systems and consist primarily of piping rearrangements or additions and some added instrumentation and control systems. The development costs would be significant only if a turbine were designed specifically to enhance or facilitate FWPA. Most likely, this approach would not be economical because of the potentially high first-of-a-kind engineering costs. It is estimated that the development costs for secondary plant modification to enhance end of cycle stretchout, excluding turbine development, would be less than 1 million dollars.

3.3.3 Technical

Several discussions with turbine users provided the following on the turbine limitations associated with removing feedwater heaters from service. When heaters are removed from service, the steam which was formerly extracted now passes through the turbine. At some point the volumetric flow limits on internal nozzles are reached. This may occur before the volumetric flow limit on the turbine admission valve is reached. Secondly, the extra flow produces changes to stage pressures and stage pressure drops. Restrictions are placed on operating in this mode to keep bucket, diaphragm and thrust loads within design limits.

Conceptually, there appear to be three approaches to provide extra capability for FWPA cycle coastdown from that currently available. First, a turbine vendor could provide a turbine built to some specific FWPA performance specifications. This probably would entail development and first-time engineering costs. Second, an existing design rated at higher flow (power) conditions could be purchased and effectively de-rated during normal operation. This is a variation on the first option. The purchaser obtains whatever FWPA capability is possible in current designs but without absorbing non-standard costs. In both cases, the cycle efficiency during normal operation should be degraded since the turbine is effectively operating at less than full load. This could be a very serious economic penalty. The final approach entails removing the extraction steam from the turbine in normal fashion but bypassing the feedwater heaters. The steam could be used as process steam or the heat rejected to the environment through the condenser. Additional piping, control valves, instrumentation and control systems would be required. The potential impact on the condenser and feedwater pumping requirements would have to be investigated. There might be some utility interest in this third approach if the specific economic benefits to the utility (e.g., because of low replacement power costs) warranted it.

This third approach could be backfitted in plants under construction by making the relatively minor piping and instrumentation changes discussed above. It is doubtful that such changes could be justified on an operating plant because of the potential for substantial additional outage time to effect the changes. Some plants may be well configured for such design changes already, but this would have to be assessed on a case by case basis.

There are some potentially adverse effects on the safety analysis associated with employing end-of-cycle stretchout. These are primarily a consequence of operating at lower primary coolant temperatures. Most notably is the potential for higher stress loads on the various support members and core internals during a postulated loss of coolant accident. Such potential adverse effects are not expected to be significantly worse for FWPA than for normal power coastdown which is the reference operational mode for this assessment.

The reduction in feedwater temperature will also affect the plant response to certain operational transients and postulated accidents. For example, changing the valve lineup for the feedwater heaters (e.g., isolating certain heaters) might exacerbate the feedwater malfunction

events, e.g., excess feedwater flow, feedpump overspeed, etc. The consequences of such feedwater temperature reduction can only be assessed through detailed analyses, which are beyond the scope of this assessment.

3.3.4 Operational

The operability of the reactor plant should be essentially the same as that for a plant employing normal power coastdown. There will be some minor valve realignments when feedwater heaters are cut out during FWPA and the turbine extraction steam will be redirected elsewhere in the plant, but this should not impact plant reliability to a noticeable degree. Of course, anytime additional valves, components, and/or instrumentation is installed, there is the potential for more failures, but considering the present complexity and numbers of components in an existing PWR, failures of such FWPA equipment would not noticeably affect plant availability.

One situation where operability of the plant may be less with FWPA than with normal coastdown is during the recovery from an inadvertent reactor trip. The reactivity to overcome peak xenon will have to be supplied by dropping further the moderator temperature. Since with FWPA the core may have slightly higher burnup, greater moderator temperature reductions will be required during this recovery.

Plant capacity factor will be reduced relative to the no-stretchout case when EOC stretchout is employed. The maximum reduction is about 0.5%. However, compared to the use of normal power coastdown the capacity factor increases for the same additional energy output or the same end point electrical output. This increase in capacity factor is very slight, about 0.1%.

The use of preplanned EOC stretchout reduces the capability of the reactor plant to operate beyond its scheduled cycle length in the event that some unforeseen circumstance (e.g., the forced outage of another power plant in the utility's system) necessitates the continued operation of the plant. However, relative to normal power coastdown, the reactor employing FWPA can economically operate longer in such a situation. FWPA operations show a breakeven energy cost (i.e., the energy cost which equals that of the no-stretchout case) occurring about one week later than when normal power coastdown is used. Thus, in those situations where continued operation is desired, FWPA operation can proceed longer with no economic penalty.

3.3.5 Other Considerations

Utility Acceptability - Utilities may not be interested in pursuing FWPA or for that matter, normal power coastdown, because of the reduced flexibility it affords to alter the cycle length when the need arises. In particular, the small potential incremental benefits of FWPA over normal power coastdown are probably insufficient to motivate further development of this operational mode. Furthermore, the higher costs of licensing the plants to operate at reduced feedwater temperatures will be another deterrent to the development of this concept.

Date of First Commercial Operation - Very little design and licensing effort would be required to implement this concept. These efforts can easily be accomplished within one year.

Nonproliferation Compatibility - There is no incremental proliferation resistance or susceptibility relative to either the no-stretchout case or the normal power coastdown case.

Viability With Recycle - In the recycle mode, there is a negative fuel resource conservation incentive because of the reduced thermal efficiency of the generating plant during the stretchout operation. During stretchout, less electrical energy is produced for a given consumption of fissile material. Since with recycle all residual fissile material in the spent fuel is recovered and subsequently reinserted into the reactor, operation in a less efficient mode results in a net loss of fissile material and therefore a higher uranium makeup requirement. It is estimated that, if end-of-cycle stretchout were employed in the recycle mode, an increase in uranium requirements of about 0.5 - 1.0% would occur relative to the no-stretchout case. Thus, in the recycle mode, planned end-of-cycle stretchout would not be employed to improve uranium utilization but still may be attractive from an economic standpoint. Stretchout may also be used effectively in the recycle mode to gain cycle length flexibility when the need arises.

Interaction With Other Concepts - EOC stretchout is compatible with all the backfittable uranium utilization improvement options. Its effectiveness is reduced somewhat when implemented in conjunction with the low power density concept because of the lower Doppler absorption and xenon poisoning of the concept. Also, with the frequent refueling concept, it may be desirable, primarily from an economic standpoint, to shut down from full load and refuel as quickly and as frequently as possible rather than delay each refueling while the reactor is operated at less than rated output.

TABLE 3-1

THE IMPACT OF EQUILIBRIUM CYCLE STRETCHOUT
ON U_3O_8 REQUIREMENTS

49	Minimum Plant Net Electrical Output (% of Rated)	Cycle Stretchout Time (Days)				Normal Coastdown	U ₃ O ₈ Savings (%)			
		Normal Coastdown	Feedwater Pressure Augmentation ^a				Normal Coastdown	Feedwater Pressure Augmentation		
			50 ^o F	100 ^o F	150 ^o F			50 ^o F	100 ^o F	150 ^o F
	95	3.3	5.1	6.9	7.7	1.2	1.8	2.5	2.7	
	90	6.0	7.7	9.6	11.1	2.1	2.7	3.4	3.9	
	85	9.2	10.4	12.2	13.3	3.1	3.6	4.2	4.6	
	80	12.1	13.2	14.8	15.5	4.0	4.4	5.0	5.3	
	75	14.8	16.0	17.2	17.8	4.8	5.2	5.7	5.9	
	70	17.6	18.7	19.7	20.2	5.5	5.9	6.3	6.5	
	65	20.4	21.7	22.2	22.6	6.2	6.6	7.0	7.1	
	60	22.9	24.2	24.7	24.8	6.8	7.2	7.5	7.6	
	55	25.4	27.0	--	--	7.3	7.8	--	--	

a. The different temperatures shown indicate the reduction in feedwater temperature from its normal value.

Figure 3-5
REACTIVITY CHANGE DUE TO VARIATION IN MODERATOR TEMPERATURE

Figure 3-6
EQUILIBRIUM CYCLE STRETCHOUT
RELATIVE ENERGY COST vs STRETCHOUT TIME
(\$40/LB U_3O_8)

Figure 3-7
MAXIMUM SAVINGS IN ENERGY COST vs FEEDWATER TEMPERATURE REDUCTION
(\$40/LB U_3O_8)

3.4 CORE PERIPHERY MODIFICATIONS

Uranium utilization improvement can be obtained through the reduction in neutron leakage from the radial surface of the core. Presently, neutrons that leave the outer face of a peripheral assembly intersect a small water gap (~ 0.15 in. thick), a stainless steel core shroud (~ 1.0 in. thick), another water gap (~ 7 in. thick on the average) and then the core barrel (~ 2.6 in. thick). The majority of neutrons that leave the core are fast neutrons. Some are reflected back toward the core as fast neutrons, while others are absorbed there or are reflected back to the core as thermal neutrons. Improvements in uranium utilization can thus be made by increasing the number of fast neutrons that are reflected back to the core and by reducing the absorption (both fast and thermal) in these ex-core regions so that more neutrons can diffuse back into the core.

Several techniques have been suggested to accomplish this increased reflection. These techniques include increasing the existing water gap between the peripheral assemblies and the core shroud, increasing the thickness of the current core shroud, replacing the present stainless steel shroud with a Zircaloy shroud, and substituting alternate materials (e.g., Be, BeO or graphite) for the existing reflector materials. All of these methods are subject to certain complications and/or development requirements; these will be discussed in Section 3.4.3 of this assessment.

3.4.1 Uranium Utilization

Variation in the thickness of the core shroud and the water gap between the core and shroud can result in increased reactivity and hence improved uranium utilization. This concept is considered non-backfittable since a thinner shroud leads to manufacturing problems, whereas a thicker shroud, in general, cannot be accommodated within the core support barrel. Additional reflecting material, however, could be placed next to corners of the shroud plates which do not touch the support barrel. Such fillers would need to be cooled adequately such that the temperature in the shroud and in the fillers do not exceed current design limits. Fillers of this type would be considered backfittable and are therefore not the subject of this assessment.

The effect of moving the shroud further away from the core, and thus increasing the gap, is shown in Figure 3-8. The uranium utilization improvement results primarily from a reduction in thermal leakage. The end-of-cycle reactivity is increased by $0.14\% \Delta\rho$ when the gap thickness is increased from its reference value of 0.15 in. to 1.0 in.; this reactivity gain would improve uranium utilization by about 0.4%. Additional increases in the gap thickness result in a negligible further benefit.

The effect of increasing the shroud thickness while maintaining a constant peripheral gap of 0.15 in. is shown in Figure 3-9. Increasing the shroud thickness from 1.0 in. to 4.0 in. yields an increase in reactivity of about $0.8\% \Delta\rho$, which represents a uranium utilization improvement of about 2%. This uranium utilization improvement results primarily from the reflection of fast neutrons back toward the reactor

core, which reduces the fast neutron leakage. The improvements due to increases in the gap and shroud thickness are mutually exclusive. The benefits cannot be added since increasing the gap also moves the shroud away from the core and therefore decreases the fast neutron reflection potential of the shroud. The shroud in current design PWRs consists of steel plates approximately 1 in. thick which are welded together such that their faces are adjacent to the faces of the peripheral fuel assemblies. The principal purpose of the shroud is to minimize lateral motion of the fuel assemblies; the shroud also limits the core bypass flow to an acceptable value. As mentioned above, it also serves as a fast neutron reflector. Improved neutron economy could be obtained if Zircaloy were substituted in place of the present stainless steel shroud because of its reduced neutron absorption; the scattering properties of Zircaloy are not that dissimilar from those of stainless steel. Uranium savings are estimated to be about 1% for such a substitution of shroud materials. As will be discussed in Section 3.4.3, the major problem with such a substitution is the structural behavior of Zircaloy.

By radically reconfiguring the reflector region of the core, more substantial gains in uranium utilization could be obtained. In particular, the soluble boron that accompanies the water in the various gaps (i.e., between the core and shroud, and between the shroud and barrel) has relatively high neutron absorption. If this absorption could be removed so that more leakage neutrons could be reflected back to the core, then significantly improved performance would result. Two materials that have long been regarded and used as reflectors are graphite and beryllium oxide (BeO). The effect of using these materials as reflectors can be inferred from Figure 3-10. In this figure, various albedo values are shown as a function of reflector thickness for both graphite and BeO. The improvements in uranium utilization as a result of using graphite and BeO as reflector materials are estimated to be 2% and 5%, respectively.

Similar, but somewhat larger, savings in separative work can be obtained by using the various techniques described above to improve the core reflector. These potential savings in separative work are 0.6% for an increased water gap, 3.3% for an increased shroud thickness, 1.4% for the use of a Zircaloy shroud instead of a stainless steel shroud, 2.6% for the use of a graphite reflector, and 6.4% for the use of a BeO reflector.

3.4.2 Economics

The additional capital costs associated with modifications to the core periphery are very similar to those for the low power density core design because of the necessity to use a larger diameter reactor vessel (except for the Zircaloy shroud design). An estimate of \$10 million was developed for the low power density design. This is an upper limit for the wider water gap and thicker shroud designs (assuming that a thick shroud is not excessively costly), since the maximum diameter vessel used with the low power design would not be required for the present concepts. In particular, for the wider water gap, only a 2 in. larger diameter vessel is needed. For the graphite and BeO reflector concepts, the maximum diameter possible would be used to achieve the best reflector characteristics. The cost of these alternate reflectors will also be

large, particularly for the Zircaloy and BeO concepts because of both the high costs of the materials and the high costs of fabrication. No estimate of these costs was obtained.

The percentage savings in fuel cycle costs for the various modifications are essentially constant as a function of U_3O_8 price and are approximately the same as the percentage savings in U_3O_8 . These savings are 0.4% for the wider water gap, 2.5% for the thicker shroud, 1.1% for the Zircaloy shroud, 2.0% for the graphite reflector, and 4.9% for the BeO reflector.

The development costs are very difficult to estimate for the concepts that require reactor demonstration. Because of previous irradiation experience with most of these materials, it may not be necessary to conduct large demonstration programs. This could, however, involve a large risk; one that probably would not be acceptable to a reactor manufacturer. If a reactor demonstration were deemed necessary, it would be quite expensive because of the costs of in-pile irradiation demonstration.

Construction time will not be significantly impacted by core periphery modifications. These modifications are exclusively within the scope of the NSSS supplier with the exception of the impact of a larger reactor vessel on the remainder of the plant. This item is covered in Section 3.6 under the low power density core design.

3.4.3 Technical

The primary technical problem with increasing the peripheral water gap between the core and shroud would be the resulting unacceptably high core flow leakage. If local horizontal plates were attached to the shroud, e.g., at the fuel assembly grid locations, the leakage would be reduced. However, fuel rod fretting problems could likely develop because of local cross flow.

The use of a very thick shroud (e.g., up to 5 in.) would pose a very difficult fabrication problem and would be very costly. There is a good chance that such a shroud might become distorted with time and cause high loads on its attachments to the core plate. In addition, high internal heat generation due to neutron and gamma ray heating would cause high thermal stresses.

The use of a Zircaloy shroud is not recommended because of a number of structural and manufacturing considerations. First, Zircaloy is a poor structural material for this application. It grows and creeps significantly which in time would lead to severe distortion. Furthermore, it has poor wear resistance to fuel assembly rubbing which might occur during operation. Secondly, because of the extreme difference in its thermal coefficient of expansion compared with that of the surrounding stainless steel structures, a rigid structure could not be designed. Segmented or flexible structures would be subjected to vibration and wear problems as well as high core flow leakage. Finally, the attachment design of the Zircaloy shroud to the core barrel and/or the core plate would be extremely difficult and a real engineering challenge.

A Zircaloy shroud could be fabricated, but it would be very difficult and extremely expensive. All welding would have to be performed in an inert environment which would require new techniques because of the size of the structure.

Considerable experience exists outside the commercial LWR industry in the design and in-reactor demonstration of reflector shrouds and of outer reflector assemblies. The two commonly employed moderator/reflector materials are beryllium and carbon. In a water-moderated reactor, the metal beryllium remains in contact with the primary coolant, while the carbon is canned in either stainless steel or Zircaloy to prevent carbon-water reactions. The long-term corrosion resistance of the metal beryllium, however, is not as good in this application as can be attained in the canned design using stainless steel or Zircaloy. This might limit the life of such a reflector. It would not be a limiting problem if replaceable outer reflector assemblies are employed (i.e., typical of the R-2 test reactor design). The poor availability and expense of beryllium, moreover, present potential problems which are unique to the application of this reflector material. Alternately, beryllium oxide (BeO) and beryllium carbide (Be_2C), which are refractory compounds of beryllium, could be considered along with graphite as candidate moderator materials suitable for use in a canned design. Special handling of BeO is needed due to its high level of toxicity.

EG&G Idaho has developed considerable expertise in the design and operation of beryllium reflectors for the ETR and ATR cores. Corrosion is not a limiting design consideration for these applications since these reactors operate at relatively low coolant temperatures (200°F at 400 psi system pressure) and maintain the demineralized coolant water at high purity and a pH in the range of 5.5-6.5. (These reflectors corrode at a rate of about 1-2 mils oxide thickness per year). The reflectors in both reactors, however, are limited by fast neutron flux embrittlement and subsequent brittle cracking and material spalling which occur due to differential, neutron-flux induced growth. The 14-18 in. thick ATR reflector currently is replaced about every 6-7 years after a fast fluence exposure of about $2 \times 10^{22} \text{ n/cm}^2$ ($E \geq 1 \text{ Mev}$). At ETR, the 4 in. thick reflector experiences considerably less differential stress loading, has a higher fast fluence exposure capability, and has been replaced only once in the last 18 years. The intent of the limits for each reactor is not to preclude in-reactor brittle cracking from occurring, but to preclude spalling of the beryllium into the coolant flow.

From a nuclear design perspective, Zircaloy would be the better canning material for the BeO and graphite reflectors than stainless steel. Due to the higher strength properties of stainless steel, especially for the shroud application, this material has been favored as the canning material in various test reactors. Long-term residency, however, may suggest that the more ductile Zircaloy be used. The Sodium Reactor Experiment (SRE) at Hallam, Nebraska did experience stainless steel canning failures, apparently attributed to high local strains. A redesign employed Zircaloy as the canning material. The SRE reactor was in operation for too short a period (i.e., 1957-1963), however, to ascertain the long-term benefits of the newer Zircaloy design. In a

water environment such as a PWR, the long residence of stainless steel may be limited by intergranular corrosion cracking that is stress assisted.

Only the Zircaloy shroud concept is regarded as possibly backfittable. But even this concept may not be backfittable because of the interface problems with the core barrel and core plate.

Considerable development effort would be required with each of the concepts discussed above. The details of this effort have not been determined, nor has the time or cost. Except for the use of a wider water gap or a thicker stainless steel shroud, the other core periphery modifications would need to be demonstrated before they could be licensed. This would impact significantly the time and cost of the development effort.

3.4.4 Operational

Once installed in a reactor, the choice of reflectors should be largely transparent to the reactor operators. The one exception to this impression might be that more attention in the design to prevent rubbing wear during operation may be needed to protect the less resistant materials, e.g., the Zircaloy shroud. It seems certain that a reflector design would not be selected that would adversely affect the plant availability or capacity factor. Thus, the operability of a reactor using this concept would be essentially the same as the reference composite PWR. There might be an increased desire for inspection, particularly for the Zircaloy shroud and the graphite or BeO reflectors, since the performance of these materials is not well characterized in an operating PWR environment. There should be no impact on plant staffing as a result of incorporating the improved reflectors. Furthermore, radiation exposure of the staff would be increased only if increased refueling time or increased inspection requirements were imposed with the implementation of these core periphery modifications.

3.4.5 Other Considerations

Utility Acceptability - The three more conventional concepts (i.e., wider water gap, thicker shroud, and Zircaloy shroud) would be perceived by utilities as small variations to current designs. As long as they are first demonstrated and licensed, then they will be adopted by utilities because of their better economics. Adding reflectors (graphite, Be, and BeO), on the other hand, is a major departure from proven technology and would be perceived as radical changes. These modifications also provide the greatest potential benefit. Therefore, they would be adopted with a much more cautious attitude. If these concepts are adequately demonstrated and if acceptable performance is obtained then utilities will be willing to consider them.

Date of First Commercial Operation - Because of the need to demonstrate a number of these concepts in an actual operating PWR environment, before a utility would be willing to purchase them, a fairly long development time would be required. Considering the time it takes to design, license, fabricate, and irradiate these concepts, 10-15 years before

commercial offering appears reasonable. For those concepts that do not need demonstration, a much shorter time, on the order of 2-5 years, would be sufficient to develop them to the point where they could be offered commercially.

Nonproliferation Capability - Core periphery modifications do not impact proliferation resistance.

Viability With Recycle - The use of improved reflectors will enhance the core conversion ratio somewhat. Since the effects of using this concept are similar to those when low-leakage fuel management schemes are employed, similar performance is expected. About half the benefit of using improved reflectors in the once-through fuel cycle will carry over to the recycle mode.

Interaction With Other Concepts - The use of all the different core periphery modifications other than the Zircaloy shroud require more space in the reactor vessel than currently exists. This means that a larger vessel or a smaller core must be used in conjunction with this concept. Thus, the low-power density core design competes with this concept for the use of additional vessel volume. Also, the use of core blankets and improved reflectors, although not mutually exclusive, reduces the desirability or need for each other, i.e., each reduces neutron leakage.

Figure 3-8
CHANGE IN REACTIVITY DUE TO CHANGES IN THE PERIPHERAL GAP DIMENSION

Figure 3-9
REACTIVITY CHANGES DUE TO VARIATIONS IN THE SHROUD THICKNESS
(0.15 MIL GAP)

Figure 3-10
SLAB ALBEDOS
(FROM TID - 6303)

a_f (FAST ALBEDO) - PROBABILITY THAT A FAST NEUTRON ENTERING THE REFLECTOR WILL RETURN TO THE CORE AS A FAST NEUTRON
 a_s (SLOW ALBEDO) - PROBABILITY THAT A SLOW NEUTRON ENTERING THE REFLECTOR WILL RETURN TO THE CORE AS A SLOW NEUTRON
 a_c (CONVERSION ALBEDO) - PROBABILITY THAT A FAST NEUTRON ENTERING THE REFLECTOR WILL RETURN TO THE CORE AS A SLOW NEUTRON

3.5 RADIAL BLANKETS

By the addition of a radial blanket composed of fuel assemblies containing fertile material, it is possible to increase fuel utilization in PWRs. The increased utilization is due to three factors. First, the blanket shifts the flux to the periphery of the active core where the higher reactivity fresh fuel is loaded (when conventional OUT-IN fuel management is used). Second, the radial blanket changes the albedo of the active core, reducing core leakage and thus increasing core reactivity. Finally, neutrons which leak from the active core are available in the blanket for fertile conversion. This conversion leads to an increased production of fissile material, some of which is used for in-situ power production, the remainder of which is available for possible later reprocessing and recycle. Four sources of fertile material were considered: natural uranium, depleted uranium (0.2 wt%), spent fuel, and thorium. As a potential blanket material, natural uranium offers a small improvement in fuel performance over depleted uranium (0.711 wt% U-235 vs. 0.2 wt%), while the cost of natural uranium is significantly higher. Spent fuel has a shorter blanket life because of its prior irradiation, but it is readily available at no extra cost and has the highest fissile content, which will contribute to a greater reactivity gain. Both thorium and depleted uranium offer a readily available, low-cost, easily-managed (both during fabrication and installation) blanket material. The use of each of these materials will be assessed in the sections which follow.

For the purposes of this assessment, it is assumed that the blanket assemblies are added as an additional row on the present core periphery. Use of existing core locations would require substantially higher linear heat ratings of the remaining active core fuel (the fuel interior to the blanket region) because of the large fraction of total assemblies that reside on the core periphery. As discussed in Section 3.6 - LOW POWER DENSITY CORES, manufacturing techniques currently exist to fabricate sufficiently large pressure vessels to accommodate such an extra row of blanket assemblies.

3.5.1 Uranium Utilization

The improvement in uranium utilization as a result of installing a core blanket on the periphery of an existing PWR core is dependent on the blanket material. The largest benefit can be obtained by using spent fuel because of its significantly higher fissile loading. Using such fuel is analogous to implementing a low power density core. By incorporating a radial blanket, the core becomes larger, and since the blanket produces appreciable power, the power density of the active core can decrease. Since spent fuel is readily available each year, the maximum benefit would be obtained by replacing the blanket each year with "new" spent fuel. This would ensure the maximum fissile content at the beginning of each cycle. It is estimated that approximately a 5% improvement in uranium utilization would result from such a core-blanket arrangement.

Use of a depleted or natural uranium blanket would result in a smaller benefit than that obtained with spent fuel because of the lower initial

fissile content of this blanket material. With these materials, the blanket assemblies would be changed infrequently, possibly even lasting the entire life of the plant. A potential U_3O_8 savings of 3.0-4.5% is estimated to be possible with this type blanket with the natural uranium blanket having somewhat better fuel utilization than the depleted uranium blanket.

The thorium blanket will probably show the poorest performance of the four blanket materials because of its initial zero fissile content, its low fast fission cross section, and its relatively high thermal absorption cross section. The high absorption cross section of thorium decreases the number of neutrons that reenter the active core from the blanket. It is estimated that the reduction in U_3O_8 requirements is about 2-3% compared to the reference composite cycle. Despite its more modest potential benefit, it may be desirable to use a thorium blanket for recovery of U-233 in the event that reprocessing is authorized. The superior performance of U-233 in a thermal reactor could eventually result in better long-term utilization of uranium resources.

The separative work requirements of a core surrounded by a radial blanket are lower than those of the reference core. When spent fuel is used as the blanket assemblies, a savings of approximately 8% occurs. With other blanket materials, the separative work savings is less: about 6% with natural uranium, 4% with depleted uranium, and 3% with thorium.

3.5.2 Economics

The change in capital costs for the radial blanket designs is estimated to be about \$10 million (\$8/kWe), the same as for the low power density core design. This higher capital cost results from the necessity to manufacture a larger reactor pressure vessel to accommodate the extra row of blanket assemblies. In addition to the larger pressure vessel, other modifications required to incorporate the blanket design would include modified core internals, a larger pressurizer, a larger safety injection system, more in-core instrumentation, possibly more control rods and drive mechanisms, and a modified containment structure. The major portion of this higher capital cost is associated with the larger pressure vessel and containment structure. Depending on what blanket material is selected and whether flow orificing (flow tailoring) is employed, it may be necessary to increase the reactor coolant flow rate. If this occurs, then additional capital costs (perhaps \$2/kWe) could be incurred to provide larger reactor coolant pumps.

The savings in fuel cycle costs show the same trend as the uranium savings. The use of spent fuel as the blanket material results in the largest benefit, approximately 5% lower fuel cycle costs. The reduction in fuel cycle costs for the other blanket materials are estimated to be 4.5% for the natural uranium blanket, 3% for the depleted uranium blanket, and 2.5% for the thorium blanket.

The use of radial blankets requires no fundamental development work. The larger pressure vessel to incorporate one additional row of fuel assemblies is within the state-of-the-art. To be able to offer this design in a firm bid proposal would nevertheless require considerable

engineering design and licensing effort. This effort is estimated to cost about \$10-15 million for the NSSS portion and extend 2-3 years. A similar effort may be required by the architect-engineer (A-E) for the balance of plant, but no cost or time estimate is available.

Because the incorporation of the radial blanket is an extension of current designs with the major changes being component size, there should be no significant impact on the plant construction time.

3.5.3 Technical

Design problems are not expected for the use of blanket assemblies containing fresh fuel rods fabricated from Zircaloy cladding and fuel pellets of natural uranium oxide (0.711 wt%) or depleted uranium oxide (0.20 wt%) or of thoria. For a PWR blanket comprising approximately 20% of the outer core volume, the local flux will average about 20-30% of the core average flux, when natural uranium is the blanket fuel. For blankets of depleted uranium or thoria, the local flux will be lower, in the range of $\leq 10\%$ of the core-average flux. The power in these latter two blankets will gradually increase as more fissile material is bred. Eventually, both UO_2 blankets will develop the same power with the thoria blanket at a somewhat lower power because of its higher absorption cross section. Uniform depletion in a blanket fuel assembly is attainable through the steep radial flux profile of the blanket region by the rotation of fuel assemblies during normal refueling of the active core. Considerable industry irradiation experience has shown that higher burnups can be achieved while minimizing fuel duty through long-term operation at reduced heat ratings. Waterside corrosion of Zircaloy cladding is a potential life-limiting problem for high power, high temperature operation. For the blanket application, however, waterside corrosion is not expected to be a problem, although verification will be required. For this reason, no adverse effects of low-power operation for up to about 30 years are expected for blanket fuel rods fabricated solely for this application.

Spent fuel may also be used in the blanket region. For very long residency times as blanket fuel, it may be necessary to remove selected fuel rods from discharged assemblies, clean these rods of crud deposits, and inspect and place into new assembly cages, all at the spent fuel pool. The inspection might include an assessment of the existing waterside oxide thickness layer on the Zircaloy cladding to assure that continued operation would not be limited by the potential for accelerated corrosion. The lifetime of spent fuel in the blanket region is less certain than that of fresh blanket fuel due to its prior, more severe duty in the active core and because its heavy metal fissile concentration still remains significantly higher than that of the fresh fuel concepts discussed above. This higher fissile content can result in considerably higher power density in the spent fuel blanket than in the other blanket concepts. Furthermore, the lifetime of individual rods in the blanket may be strongly dependent on their relative duties experienced while residing in the active core. For this reason, the lowest power fuel rods discharged from the active core would be the preferred candidates for reconstitution into a blanket assembly. Since spent fuel is readily available at no extra cost, it may be desirable to changeout these blanket assemblies each year with "new" spent assemblies rather than subject them to additional multiple cycles of irradiation.

The blanket assemblies should have orificed flow (flow tailoring) because of the relatively low power these assemblies will be producing. Only the spent fuel blanket (with a changeout each year) will produce relatively high power. If the blanket assemblies were not orificed, they would cause the remainder of the core (the high power active core) to be under-cooled, resulting in considerable thermal margin loss. If the flow to the blanket assemblies was orificed, then some type of flow separation skirt would probably be needed to prevent large cross flow from the active core out to the blanket region. In addition, as the bred fissile material content of the blanket increases, the blanket will produce more power (i.e., will change with burnup) possibly requiring more flow as a function of burnup. An alternative to orificing the flow would be to increase the overall coolant flow rate so that sufficient flow was provided to the higher power active core assemblies while maintaining full flow to the blanket assemblies. This option may be more expensive than orificing because of the requirement for larger reactor coolant pumps and/or more pumping power, but may entail less design development work and testing.

The use of radial blankets will tend to shift the gross radial power distribution to the outer part of the active core and lower the power in the central part of the core. This shift in power can be as much as 20% if the same fuel loading pattern is used for both the blanketed and unblanketed cores. In reality, however, a new fuel loading pattern will be used if a blanket is installed so that the gross radial power distribution is similar for the two designs. Nevertheless, the outer row of active core assemblies will have a more even power distribution (and therefore fuel burnup) across the width of the assembly and will not exhibit as large an exposure gradient as seen in unblanketed cores.

The use of radial blankets as discussed in the previous sections does not lend itself to being retrofit unless a larger reactor vessel and associated plant modifications are provided during plant construction. This would necessitate that a low power density core be adopted in the interim to capitalize on the extra capital costs. Once low power density operation was initiated, it is doubtful that it would be altered to include radial blankets.

3.5.4 Operational

The operability of the reactor plant should be essentially the same as for the reference composite design. If spent fuel is used as the blanket material, then more fuel shuffling would probably be required each refueling outage because of the resource advantage of using "new" spent fuel each cycle. If the other blanket materials are used, then only very infrequently, if ever, would the blanket assemblies be moved. As mentioned in Section 3.5.3, it may be desirable to rotate the blanket assemblies to even out the fluence and exposure. However, such rotations would be infrequent. The impact on plant availability or capacity factor should therefore be either small or negligible.

The number and quality of staff and the radiation exposure of the staff would be impacted significantly only if fuel reconstitution is practiced, e.g., in conjunction with the use of a spent fuel blanket. Otherwise, a small increase in staff radiation exposure may be experienced if a spent

fuel blanket were used because of the increased fuel handling and shuffling.

3.5.5 Other Considerations

Utility Acceptability - This depends on the method chosen to handle core flow and the final economic analysis. If the core looks radically different, for example, with flow skirts or separators, then this concept would be adopted more cautiously. If higher capacity coolant pumps are used, then the economics might not be as favorable, but the core would appear more like current designs. In general, the concept should be acceptable to utilities as long as satisfactory performance can be demonstrated and any licensing hurdles are overcome.

Date of First Commercial Operation - The date when this concept can be offered commercially will depend primarily on the time required to complete the design engineering of the larger vessel and associated components and to determine an appropriate method of controlling flow to the active core and blanket regions. Undoubtedly, some flow testing will be required if orificing is selected. This could delay somewhat the time required to make this concept commercially available. It is estimated that 2-4 years would be needed to complete the design and testing effort.

Nonproliferation Compatibility - Proliferation resistance will be impacted by the use of radial blankets if fresh uranium (either natural or depleted) or thorium is used because of the relatively low exposure of the blanket assemblies at intermediate plant lifetimes. With lower exposure, the bred plutonium will be composed of a higher percentage of the fissile isotopes, consisting primarily of Pu-239. It is preferable to manufacture nuclear weapons from plutonium of this quality. The situation is worse for a thorium blanket because almost pure U-233 will be produced. The performance of U-233 as a weapons usable material is nearly as good as Pu-239. The access to this blanket fuel will be rather limited, however, especially if the blanket assemblies remain in core for the life of the plant. Then only during refueling operations would it be possible to divert this material.

Viability With Recycle - The use of radial blankets will enhance the performance of the recycle mode. A large fraction of the neutrons that would otherwise be lost from the core due to leakage will be captured in the blanket producing additional fissile material through conversion. Some of this additional fissile material will fission during operation, but the remainder will be recovered during reprocessing to help supplement the fuel bred in the active core. If the blanket is replaced frequently, e.g., every three years, then as much as 10% additional bred fissile material would be available for recycle.

Interaction With Other Concepts - The use of radial blankets, as mentioned previously, is analogous to the use of a low power density core, particularly if spent fuel is chosen as the blanket material. Furthermore, the use of radial blankets reduces considerably the potential for core periphery modifications. The leakage neutrons are reduced by the blanket so that much fewer leak into the reflector.

3.6 LOW POWER DENSITY CORES

Low power density cores offer the potential for improved uranium utilization through less Doppler absorption in U-238, through a reduction in xenon poisoning, through the use of more in-core fuel batches (as a result of a greater fuel loading for a fixed fuel discharge exposure), and through the reduction in core neutron leakage. In addition to these direct savings in U_3O_8 , the potential exists for further savings as a result of increased thermal margin. For example, increased core power peaking is usually accompanied by higher core reactivity due to the increased flux weighting of the high reactivity fuel (i.e., the high power density fuel). Also, lower BOC burnable poison concentrations can be used in low leakage fuel management schemes if constraints on power peaking can be relaxed. Accompanying these lower BOC concentrations will be lower EOC concentrations (i.e., lower shim residuals) which will increase the EOC reactivity and provide additional life for a given fissile fuel loading.

The economic savings resulting from this better uranium utilization will have to compete against the higher capital costs of the reactor plant (for a given rated electrical output). Conceptually, only the reactor vessel, reactor internals, containment structure, pressurizer, and certain minor NSSS components would require modification to adopt the low power density design. Most of the NSSS and all of the balance-of-plant could maintain their current designs. Most likely, however, a number of additional systems and components would be redesigned to be compatible with the larger size core and reactor vessel.

The analysis contained in this section assumes that the current System 80TM plant electrical output is maintained and the size of the reactor core is increased to the maximum diameter consistent with reactor pressure vessel manufacturing capabilities. Based on a quick survey analysis, it appears that the current fabrication limit of a 2500 psia vessel is 196 in. inside diameter. This compares to the current System 80TM reactor vessel inside diameter of 182.25 in. A more detailed design study may indicate that a slightly larger vessel could be manufactured using current techniques; this depends upon the availability of steel plates of sufficient thickness and with guaranteed mechanical properties. An alternative approach would be to reduce the plant electrical output so as to obtain the desired reduction in core power density. This has the disadvantage of requiring a complete redesign of the entire plant, both NSSS and balance-of-plant. Due to time and budget constraints only the former approach was adopted in this analysis.

A schematic diagram of the System 80TM reactor core cross section, showing the relevant dimensions of the core, core support barrel, and reactor vessel, is presented in Figure 3-11. As can be seen in this figure, a spacing of 10 in. is maintained between the barrel OD and the vessel ID. This spacing is based on establishing a maximum coolant flow velocity between the vessel and the barrel and on limiting the vessel fluence. By maintaining the flow area between the vessel and barrel constant, the flow velocity will remain approximately constant. The allowable core diameter with the larger reactor vessel (i.e., 196.0 in. ID), assuming a constant flow area for the barrel-vessel annulus, is

158.30 in. This represents a possible 22.5% increase in core area (the addition of one additional row of fuel assemblies around the core periphery) and therefore a 22.5% reduction in power density. For this power density reduction and with the reduced neutron leakage of the larger diameter core, a lower vessel fluence will result despite the reduced barrel-vessel annulus thickness (7.8% less). This 22.5% power density reduction will be used in the remainder of this section as the reference low power density design. With the same core height and primary coolant flow rate, the same core inlet and outlet temperatures will be maintained and thus the same thermodynamic efficiency of the plant.

3.6.1 Uranium Utilization

As mentioned previously, improvement in uranium utilization results from several mechanisms. By far, the largest improvement is from increasing the number of in-core batches. By increasing the core heavy metal loading and decreasing the core power density, 6.13 in-core batches are required to maintain the same batch-average discharge exposure as the reference composite PWR. Relative to the reference composite fuel cycle, the use of more batches results in a 3% reduction of U_3O_8 requirements. Additional improvements in uranium utilization result from a reduction in Doppler absorption (0.5%), lower xenon poisoning (0.7%), and lower core neutron leakage (0.8%). The variation in core reactivity of these three mechanisms is shown in Figures 3-12 through 3-14. The reactivity contributions shown in these figures are translated into a change in required fuel enrichment by the derivative $0.144 \text{ wt\%/}\Delta\rho$.

The total reduction in U_3O_8 requirements from all of these mechanisms is estimated to be 5%, giving a 30-year cumulative uranium requirement of 4550 short tons/GWe. A similar reduction in separative work requirements is achieved (about 7%), with a 30-year cumulative requirement estimated to be about 3035 MT/GWe.

3.6.2 Economic

The change in capital costs for the low power density plant (22.5% lower) is estimated to be about \$8/kWe higher than the reference costs for a 1300 MWe LWR. This estimate is based on an essentially identical balance-of-plant and only a few modifications to the NSSS. These NSSS modifications include an enlarged reactor vessel and core internals, a larger pressurizer, a larger safety injection system, more control rods and drive mechanisms, more in-core instrumentation, and a modified containment structure. About two-thirds of this cost is associated with the larger reactor vessel and containment structure.

The fuel cycle costs for the low power density design are 4.3% lower than those of the reference composite PWR for U_3O_8 prices in the range of \$40/lb to \$160/lb. For \$40/lb U_3O_8 , this reduction in fuel cycle costs represents an annual savings of about \$2.0 million. This savings, which will be discounted more and more with each succeeding cycle, must offset the additional capital cost of the plant; estimated to be about \$10 million. Using an uninflated fixed charge rate on capital of 10.1% gives an annual charge of \$1.01 million to amortize this increased

capital cost. It therefore appears economical, disregarding first of a kind engineering costs, to deploy the low power density design.

Because of time and budget limitations, no attempt was made to determine the economic optimum core power density. It appears that reduction beyond 22.5% can be justified as long as the capital cost increases do not become too nonlinear with increased core size. In particular, it may be economical to reduce the plant output, while keeping the reactor vessel and core at the maximum size to achieve the most economical low power density design.

The low power density design requires no fundamental development work. Changes from current designs are generally in the direction of being more conservative. In particular, the lower power density means more kW/ft margin and therefore a safer design with respect to some postulated accidents. The thermal margin (DNB margin) would be the same as for the reference design because the core flow is held constant as is the overall core power. To be able to offer this design in a firm bid proposal would nevertheless require considerable design and licensing effort. This effort is estimated to cost \$10-15 million for the NSSS portion and take 2-3 years. Similar effort will be required for the balance-of-plant, but no estimate of the cost is available at this time. Because the low power density design is an extension of current designs with the major changes being component size, it is felt that there would be essentially no impact on construction time.

3.6.3 Technical

The low power density design described in the previous sections is considered state-of-the-art technology. No demonstration of the concept is required; only design and licensing (e.g., preparation of a standard safety analysis report) is needed before the concept could be offered as a new product. The limiting component is the reactor pressure vessel. However, the vessel considered in the present assessment is felt to be within current technology for a 2500 psia reactor coolant system (RCS). Thus, manufacturability is possible with existing equipment and under current code specifications. For vessel sizes substantially beyond the design described above (196 in. diameter), considerable development effort would be required.

The major impact on the balance-of-plant of the low power density design is the sizing of the containment structure. This building is sized in relation to the fluid stored energy within the RCS. The fluid stored energy in the low power density design is about 7% greater than System 80TM. This increase in available energy during a postulated loss of coolant accident would necessitate either a larger building or a building designed to contain a greater released energy. Most likely the latter option would be the most cost effective in this situation. In any event, either option is within current technology. Assuming a directly proportional increase in containment building costs, the plant capital costs would increase about \$4 million.

The low power density core is not felt to be backfittable because of the modifications required of the containment building and the larger NSSS

components which would have to be fabricated and accommodated within the containment structure layout.

Field fabrication of larger reactor vessels than used in this assessment is not regarded as advantageous. Practical limitations for field fabrication would generally be the same as established for shop fabrication. The significant exception is post weld heat treatment. This is a major, if not insurmountable, problem for field fabrication as currently envisioned.

Transportation of the larger pressure vessel would not be a limiting factor for vessels in the diameter range being considered. The System 80™ steam generator maximum diameter is already approximately 40 in. greater than the low density design described above.

3.6.4 Operational

The low power density design should not materially affect the operational characteristics of the plant. To a large extent, this design change will be transparent to plant operators and staff. The lower neutron fluence may have some positive effect and allow more flexibility in plant operations (e.g., the rate of plant heatup and cooldown), but this would have to be justified by additional development work.

With the larger core, there will be more of a tendency to be unstable to radial and azimuthal xenon-induced spatial oscillations. However, current PWR reactors are designed to be inherently stable to such oscillations. Reactors can be unstable to axial oscillations; this mode of oscillation will not be affected by the larger core diameter. Furthermore, the lower power density will contribute to a more stable condition to oscillations in all modes. Only through detailed analyses can the actual stability of the low power density core be established.

3.6.5 Other Considerations

Utility Acceptability - Utility acceptance of the low power density design should be high because of the potentially improved safety margin in this design. This together with improved fuel utilization would enhance the public image of their operation. Again because of the increased thermal margin of the low power density core, additional licensing problems should be minimized. Possibly, this type plant would get somewhat preferential treatment because of its potential for superior safety performance.

Date of First Commercial Operation - As mentioned previously, it would take about 2-3 years to design and prepare a standard safety analysis report for a low power density plant. At that time, it could be offered commercially. The time till first commercial operation would then depend on bid evaluation, vendor selection and negotiation, and plant construction. Realistically, this time sequence would take 15-20 years.

Nonproliferation Compatibility - From a nonproliferation standpoint, the low power density core has essentially the same characteristics as the reference composite design. Slightly lower plutonium production will

result from the lower fuel temperatures (reduced Doppler absorption). On the other hand, the greater thermal margin would allow more fuel management flexibility, which could be used in the diversion of fuel for non-peaceful purposes. However, access to weapons-usable material is the same as for the current once-through fuel cycle.

Viability With Recycle - The low power density design is compatible with plutonium recycle in LWRs. The somewhat lower plutonium production and the longer in-core residency time will result in a resource utilization penalty relative to the reference composite recycle design; this penalty is estimated to be about 3% greater U_3O_8 requirements.

Interaction With Other Concepts - The low power density design is compatible with both the backfittable and the nonbackfittable improvement options. Many of the backfittable options have higher power peaking than present fuel cycles in operating PWRs. The low power density design, because of its increased thermal margin, will be able to accommodate this higher peaking more easily.

The only significant drawback of the low power density design is the greater number of fuel assemblies that must be handled. This may conflict with the rapid refueling concept which requires that fuel handling during a refueling-only outage be minimized. On the other hand, there is more flexibility in the low power density design to employ innovative fuel management schemes that minimize fuel shuffling.

The use of low power density will also reduce somewhat the effectiveness of end-of-cycle stretchout (coastdown) because of the lower Doppler absorption and lower xenon poisoning which contribute to the reactivity gain when reducing plant power.

Figure 3-11
 REACTOR CORE CROSS SECTION
 241 FUEL ASSEMBLIES

Figure 3-12
FUEL TEMPERATURE DEFECT

Figure 3-13
EQUILIBRIUM XENON WORTH

Figure 3-14
EFFECT OF CORE SIZE ON RADIAL LEAKAGE REACTIVITY
(SYSTEM 80™ REFERENCE)

3.7 SMALL PWR ASSEMBLIES

The use of smaller PWR assemblies will result in better uranium utilization by providing a mechanism to gain fuel management flexibility and thereby to more fully implement other improvement options, including some that are considered backfittable. For example, higher power peaking has been identified as a potential drawback to extended-burnup, annual cycles. This higher power peaking is a result of poor power sharing between fresh and partially burned fuel, particularly with three and four times burned fuel. By loading fuel in smaller assemblies, power mismatches can be reduced significantly since the core would appear more homogeneous with smaller localized units of high and low reactivity fuel. It is anticipated that of the possible 10-15% higher power peaking experienced in the extended burnup cycle, over half could be mitigated by this technique.

Direct uranium utilization improvements can also result from implementation of this design change through the following mechanisms:

Reduced burnable poison requirements and lower shim residual. With the fresh fuel loaded in smaller units, burnable poison requirements to help control local power are lower. In particular, in low leakage fuel management schemes where fresh fuel is loaded in the interior of the core and exposed fuel is loaded on the core periphery, the concentration of burnable poison can be lower, resulting in a more complete burnout of the poison material by the end of the first operating cycle of this fuel.

Radial core blankets. The large size of PWR assemblies precludes the dedication of the peripheral core locations for use as a blanket region. Such a dedication would necessitate an increase in the linear heat rate of the active core region in order to keep the same overall plant rating. With the use of small assemblies, a core blanket of one-half assembly thickness and composed of highly burned fuel could be easily constructed. The resulting increase in linear heat rate of the remainder of the core would be much less than when the entire peripheral row of assemblies (of current size) is used as a blanket. Figure 3-15 shows the placement of such a radial blanket in a typical current PWR core configuration.

Fuel reinsertion. Reinsertion of previously discharged fuel (e.g., the initial batches of the first core which received less than the equilibrium discharge exposure) will be facilitated by this concept since power mismatches between the reinserted fuel and the already loaded fuel with low burnup would be reduced because of the more homogeneous mixture of fuel provided by the smaller assembly design.

3.7.1 Uranium Utilization

The use of small PWR fuel assemblies has the potential to directly increase uranium utilization by 4-5% as a result of reducing the end-of-cycle poison (shim) residual - about 1.5%, incorporating a radial core blanket into the fuel management pattern - about 2-3%, and reinserting fuel that has not been irradiated for its full lifetime - about 1%. This design concept may also lead to indirect improvements in uranium utilization by helping to mitigate (by reducing power peaking)

the potential loss in thermal margin that would occur if certain improvement options (e.g., the use of higher temperatures and pressures) were incorporated. An estimate of the 30-year cumulative U_3O_8 requirement for a PWR on the composite improved fuel cycle using small fuel assemblies is 4620 short tons/GWe. This includes 2.5% improvement for the use of a radial blanket and 1% for the reduction in EOC residual poison; no credit was taken for fuel reinsertion since initial core fuel is already assumed to be used in this manner in the composite cycle and other fuel (e.g., reconstituted fuel) would be used in this manner only on a case-by-case basis when it was justified by some operational consideration (e.g., failed fuel). A similar savings of about 4.5% in separative work can be realized by implementing this improvement option, resulting in a 30-year cumulative SWU requirement of 3120 MT/GWe.

3.7.2 Economic Assessment

The development costs for the small assembly design are relatively small. It is estimated that approximately \$1 million would be sufficient to design, analyze, and test the concept to the point where it could be offered in a firm bid proposal. Further development costs may be incurred in the design and demonstration of fuel handling equipment and core internals. These costs would depend on the specific method chosen to implement the small assembly concept. This does not include any costs that the reactor plant might incur in implementing this concept. Additional plant costs might be incurred for fuel handling equipment, for fuel storage racks, and for reactor internals. These added costs should be small (perhaps 2 or 3 million dollars) compared to the total plant cost when considering new plant designs.

Because of the reduction in U_3O_8 and separative work requirements, the fuel cycle costs of the small assembly design will be lower than those of the standard design. Compared to the reference composite improved cycle, the fuel cycle costs of the small assembly design will be about 3.5% less over a range of uranium prices from \$40/lb U_3O_8 to \$160/lb U_3O_8 . These savings assume no added cost for the fabrication of the small assemblies. Since the fabrication cost is only about 6.5% of the total fuel cycle cost, this process unit cost could increase more than 50% and still have an economic advantage relative to the standard fuel assembly design.

3.7.3 Technical

There are a number of potential design problems associated with the small assembly design; nevertheless, this concept is considered technically feasible, presenting no significant safety concerns. The concept is well within the state-of-the-art and primarily requires design analyses and limited testing (e.g., 8 x 8 rods).

Small cross section fuel assemblies are considerably more flexible than full size assemblies. However, this is not necessarily a detrimental feature. The natural frequency would be greatly reduced and, in general, this places it in a region where the seismic loading is very low. At least one manufacturer has conducted flow tests on long small cross-section-area bundles (4x4 - 10-foot long) which have indicated no problem with fuel rod fretting. Small cross section assemblies are not

more difficult to handle than full size assemblies when appropriate procedures are used; in fact, the lateral flexibility might be an advantage in certain instances (e.g., when bowed fuel assemblies are present). On balance, the use of small assemblies does not suggest any new problems. It might be desirable to use anti-bow grids to reduce grid-to-grid clearance.

The impact of the small assembly design on other components depends on the specific design chosen. Potentially, fuel alignment and hold-down would need redesign to accommodate the smaller cross section assemblies. Also, fuel handling equipment and fuel storage racks would need modification. If an alternative fuel design which allows the locking of four small assemblies together (see Section 3.7.4) were developed, then the impact on other plant components would be minimized. In addition, such a design would probably be backfittable into existing plants or plants under construction.

3.7.4 Operational

The operational impact of implementing the small cross section assembly design lies mainly in the areas of (1) longer fuel handling times during refueling outages, (2) greater potential for fuel handling accidents, and (3) increased quality assurance and analysis requirements. Since refueling outages are rarely (if ever) controlled by the time required to load and shuffle fuel, this concern may not be substantive. Typically, it takes about two weeks to handle fuel during an outage, but outage durations are averaging eight to ten weeks. Thus, the increased time to perform fuel shuffling may be accommodated within the normal outage.

If, on the other hand, refueling outage times approach their theoretical reference values or if frequent refueling were implemented as a means of improving uranium utilization, then the potentially longer outage time associated with the use of small PWR assemblies could be a significant operational disadvantage.

One method of reducing this impact would be to develop a fuel assembly design that would allow fuel handling of groups of these small assemblies. One such design might consist of a standard PWR fuel assembly that can be quickly disassociated into four quarter assemblies (see Figure 3-16). With this approach, most fuel handling would be with groups of four assemblies. The individual small subassemblies would be handled only when a new combination (when a new grouping of subassemblies) was desired. This handling could be performed in a special handling rack specifically designed for rapid disassociation and reassembly. Such a quartered fuel assembly design might reduce the potential for fuel handling accidents if efficient locking mechanisms, handling racks, and handling procedures were developed. Additional design considerations for this quartered fuel assembly include: differential growth of individual subassemblies within the quartered assembly, bowing, and subassembly lateral stiffness.

No matter which small assembly design is adopted, because of the greater number of individual units (small assemblies) that will be loaded and shuffled, the possibility of incorrectly loading one or more of the units

is increased. This will require more attention to the quality assurance aspects of the loading evolution. This is an administrative complication to the implementation of the concept; it can be minimized by establishing more stringent procedures and performing more prerefueling training.

Since more fuel handling will be required with the use of the small assembly design, a higher radiation exposure of plant personnel will most likely result. The extent of this increased exposure will depend on the specific fuel design developed and on the fuel handling procedures established. There is also a strong likelihood that the number of staff personnel may need to be increased somewhat to reduce the radiation exposure to key personnel (e.g., the refueling machine operators) and/or to expedite the refueling operation.

3.7.5 Other Considerations

Utility Acceptability - Except for the present general hesitancy on the part of utilities to depart significantly from proven technology, the small assembly concept should be acceptable once its performance is demonstrated. If the quartered assembly design (where four subassemblies can be locked together and handled as one assembly) is adopted, then this concept would appear quite similar to existing fuel designs and therefore may be perceived as less of an innovation.

Once loaded into the reactor, small assemblies should behave much like standard assemblies with respect to thermal hydraulic performance and fuel performance. Thus, the adaptability of this design concept to future needs of the utility (e.g., the requirement to load follow) should be essentially the same as for current fuel designs.

Certain development efforts would be required to qualify the small assembly design and support the licensing of this design. These efforts would probably include thermal hydraulic tests, seismic tests, and the engineering analyses required to verify the general acceptability, from a licensing standpoint, of the small assembly design.

Date of First Commercial Operation - It is estimated that the small assembly concept could be designed, analyzed and tested in 1.5 to 2.0 years. At this time, it could be offered in a commercial fuel bid. Initial loading would depend upon what reactor modifications are required to allow loading of this fuel. Such modifications would probably be more significant for the single small assembly design (e.g., 8 x 8 fuel rods) than for the quartered fuel assembly design where the assembly envelope is essentially identical to current fuel designs.

Nonproliferation Compatibility - The proliferation resistance of the small assembly design would be approximately equivalent to that of the current once-through fuel cycle. For clandestine diversion by a subnational group, the lesser weight of each small assembly would be an advantage, while the less rigid design might be a slight disadvantage. Overall, however, a fuel cycle employing small assemblies would exhibit approximately equivalent proliferation resistance to that of the current cycle.

Viability With Recycle - The use of small assemblies would facilitate the recycle of plutonium in PWRs. Because of the large cross section differences between uranium and plutonium isotopes, zoned enrichment (in the form of different plutonium concentrations) of the mixed oxide assemblies has traditionally been envisioned. The use of small assemblies will reduce the need for such zoning because of the more homogeneous array of fuel in the core when these smaller units of fuel are loaded.

Interaction With Other Concepts - As mentioned previously, the use of a smaller assembly design will facilitate the deployment of many of the backfittable uranium improvement options. In fact, the small assembly concept could be made backfittable itself if the quartered assembly design were adopted. The only nonbackfittable concept that may be adversely affected by this fuel design is the frequent refueling concept. The larger number of assemblies (as many as four times the standard design) will result in greater fuel handling time which is critical in the case of frequent refueling where the additional outage time must generally be kept below 7 to 10 days (see Section II). The other nonbackfittable concepts are generally favored by the use of smaller assemblies.

89 CEDM's
89 CEA's
241 FUEL ASSEMBLIES

S INDICATES ADDITIONAL
CEA LOCATIONS AVAILABLE

BLANKET SUBASSEMBLIES

Figure 3-15
CORE CONFIGURATION WITH BLANKET
CONSTRUCTED OF QUARTERED ASSEMBLIES

FIGURE 3-16 QUARTERED FUEL ASSEMBLY

Section 4

CONCLUSIONS AND RECOMMENDATIONS

4.1. HIGHER TEMPERATURES AND PRESSURES

As discussed in Section 2.2.3, the use of higher temperatures and pressures in a PWR has the least potential for successful implementation. This conclusion is based on the fact that this design modification represents the most severe departure from current technology and therefore requires the most extensive development work of any of the proposed concepts. Nevertheless, there appear to be two areas where further development work may be fruitful. First, the optimization of coolant temperature at existing primary pressures may offer a moderate improvement in uranium utilization for little additional cost. In such an investigation, it may be advantageous to explore the use of PWR flow control (flow tailoring) as a means of compensating for the potential loss of thermal margin as the primary coolant temperature increases. This type of design modification is much more modest than that discussed in Section 3 of this report. At the same time, the cost of development and the potential benefit in terms of uranium utilization are substantially less than those contemplated in this report. A second area deserving additional development work because of the large potential improvement in uranium utilization is the in-depth assessment of the feasibility of going to the relatively high temperatures and pressures considered in the present assessment. Because of the limited time and resources devoted to this industrial assessment, many adverse features were identified with the use of higher temperatures and pressures, but solutions to or methods for circumventing these adverse features could not be made in the time allotted. Such an attempt should be made, however, as part of a more comprehensive follow-on Advanced Reactor Design Study since the potential improvement in uranium resource utilization is substantial. In particular, attention should be focused on improvements in the operating characteristics of Zircaloy cladding for PWR fuel. Materials properties cause fundamental limitations to many industrial applications and the LWR industry is no exception. Development of improved cladding materials should be considered in any significant R&D program aimed at improving uranium utilization in future PWR plants through increasing operating temperatures and pressures.

4.2 RAPID-FREQUENT REFUELING

This particular design modification is the most challenging of those evaluated in this industrial assessment. It offers the greatest potential improvement in uranium utilization of the design modifications considered, but as presently evaluated would not be economically viable. Nevertheless, it is strongly recommended that this design modification be pursued to determine what additional changes could be made to make frequent refueling economically attractive. Starting with the design changes recommended in the in-depth assessment of this concept (a part of the overall Advanced Reactor Design Study), further plant modifications should be investigated that will eliminate or substantially reduce the time required to perform various critical path items in a refueling-only

outage. Such additional plant modifications should be geared to obtaining a refueling-only outage of about 7 days to provide a positive economic incentive for the adoption of frequent refueling. Furthermore, it is recommended that DOE, in conjunction with a participating utility, should demonstrate the practicality of implementing the rapid-frequent refueling design changes developed in the Advanced Reactor Design Study and the proposed follow-on study. As a near-term effort, it may be advantageous to commission the Houston Lighting and Power Company to demonstrate the Westinghouse rapid-refueling system which was an option in their South Texas projects. Such a demonstration might indicate which design changes are still needed to circumvent the practical problems of rapid refueling.

4.3 END-OF-CYCLE STRETCHOUT

Although the additional uranium utilization improvement that results from implementing the nonbackfittable aspects of end-of-cycle stretchout is not large, this design modification has appeal in that it can help promote the backfittable potential of end-of-cycle stretchout by providing more flexibility to a utility in the event that an unscheduled outage of another power plant might require that the nuclear unit extend its cycle beyond that which was planned. Since the feasibility of implementing the nonbackfittable features of end-of-cycle stretchout lies, to a large extent, in the ability of the turbine generator to operate under off-normal conditions, it is recommended that a turbine manufacturer be commissioned to assess the full capabilities of his existing turbine generator and to estimate the cost and design changes that would be required to extend the turbine capability to perform end-of-cycle stretchout. Another area related to end-of-cycle stretchout that should be investigated at an early date is the safety and licensing aspects of operating at the off-normal conditions encountered during the stretchout period. As the capabilities of this design modification are extended, the operating conditions of the reactor plant diverge more from those included in the original safety analysis. It would be beneficial, therefore, to assess the impact of these extended operating conditions on the safety analysis so that when this uranium utilization improvement concept is implemented, licensing impediments are not encountered. Finally, it is recommended that a demonstration of this design modification be undertaken in cooperation with a nuclear utility. This demonstration should consist of two phases. The first phase should be a design study of a specific reactor plant to identify the potential hardware changes that would enhance its capability of performing end-of-cycle stretchout. The second phase would be the implementation of these design changes in that reactor plant and the demonstration of its enhanced end-of-cycle stretchout capability.

4.4 LOW POWER DENSITY CORES, RADIAL BLANKETS AND CORE PERIPHERY MODIFICATIONS

These three design modifications should be considered as one concept as discussed previously in this report. In the context of new reactor designs, this combined concept is viewed as having both good uranium utilization and a high probability of successful implementation. Because these three individual design modifications all compete for reactor

vessel space and all capitalize, to a large extent, on core neutron leakage, it is recommended that a parametric analysis be performed with reactor vessel size as the principal variable. In this parametric analysis, the design, size and composition of the three individual design modifications would be varied to determine the economic and resource utilization optima for this combined concept. Such a cost benefit analysis could then be used to develop a conceptual design of a low power density reactor that includes radial blankets and improved neutron reflectors. At the same time, consideration should be given to irradiating the various reflector materials in the environment of an actual operating PWR. These reflector materials could be located outside the reactor core shroud where space is available to accommodate material samples. These three research and development efforts would answer the KEY QUESTIONS regarding this combined design concept and should provide the necessary evaluation information to allow a commercial vendor to consider these design changes in his next plant design.

4.5 SMALL PWR ASSEMBLIES

Although offering only a small improvement in uranium utilization, the use of small PWR fuel assemblies requires no fundamental development work and would help to promote certain other uranium utilization improvements because of the more uniform properties exhibited by a core composed of small assemblies. The major question regarding this design modification is the interaction of small PWR assemblies with other design concepts and, in particular, the effect of this design modification on frequent refueling. A careful analysis of these interactions would be sufficient to assess the practicality of using small PWR assemblies. At the same time, some thought should be given to the design of fuel handling equipment for such small assemblies since this equipment could mitigate, to a large extent, any adverse effects on the other design modifications (e.g., rapid-frequent refueling) of using the smaller PWR fuel assemblies.

REFERENCES

1. D. F. Newman, et al., PNL-3654, "Advanced Reactor Design Study: Assessing Nonbackfittable Concepts for Improving Uranium Utilization in Light Water Reactors," Final Report, November 1980.
2. DOE/NE-0001, "Nuclear Proliferation and Civilian Nuclear Power," Report of the Nonproliferation Alternative Systems Assessment Program, June 1980, Volume IX.
3. DOE/NE-0001, "Nuclear Proliferation and Civilian Nuclear Power," Report of the Nonproliferation Alternative Systems Assessment Program, June 1980, Volume V.
4. W. E. Mooz, "A Second Cost Analysis of Light Water Reactor Power Plants," R-2504-RC, December 1979.

APPENDIX A

PWR DESIGN MODIFICATIONS TO BE ASSESSED

The design modifications selected at the Advanced Reactor Design Study on July 1 and 2, 1980 in Washington, D. C. to initiate the Industrial Assessment of Nonbackfittable Uranium Utilization Improvements are listed below.

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. RAPID - 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

Key Questions 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. CORE PERIPHERY MODIFICATIONS

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

Key Questions 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. RADIAL BLANKETS

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%²³⁵U 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 neutron capture in xenon.
- . Lower leakage fuel management, which is enabled by accommodating 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?

APPENDIX B

EVALUATION CRITERIA

Evaluation criteria selected at the Advanced Reactor Design Study 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, and
- . 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 tons 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/kWh).
 - . 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 \$).

*Base case is composite improved PWR design as reported in Report of Nonproliferation Alternative Systems Assessment Program, DOE/NE-001/9, "Volume IX: Reactor and Fuel Cycle Description," June 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).

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, and
- . 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 accommodate a later retrofit, if it becomes desirable?

D. Operational

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

Consider the following aspects for Plant Reliability Evaluations:

- . Maintainability,
- . Inspectability, and
- . 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, and
- . Plant Simplicity.

E. Other Considerations

. Utility Acceptability (subjective analysis)

Provide comments considering the following aspects:

- . Quality Image,
- . Licensability, and
- . 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

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).