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**URANIUM RESOURCE
UTILIZATION IMPROVEMENTS
in the
ONCE-THROUGH PWR FUEL CYCLE**

MASTER

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**POWER
SYSTEMS**
COMBUSTION ENGINEERING, INC.

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ONCE-THROUGH PWR FUEL CYCLE

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ABSTRACT

In support of the Nonproliferation Alternative Systems Assessment Program (NASAP), Combustion Engineering, Inc. performed a comprehensive analytical study of potential uranium utilization improvement options that can be backfit into existing PWRs operating on the once-through uranium fuel cycle. A large number of potential improvement options were examined as part of a preliminary survey of candidate options. The most attractive of these, from the standpoint of uranium utilization improvement, economic viability, and ease of implementation, were then selected for detailed analysis and were included in a single composite improvement case. This composite case represents an estimate of the total savings in 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 improvement options which were evaluated in detail and included in the composite case were a new five-batch, extended-burnup fuel management scheme, low-leakage fuel management, modified lattice designs, axial blankets, reinsertion of initial core batches, and end-of-cycle stretchout. The thirty-year lifetime uranium and separative work requirements of this composite improved PWR fuel cycle were calculated to be 21% and 9% lower, respectively, than those of the reference PWR cycle. Accompanying these lower fuel resource requirements are fuel cycle costs that are about 15% lower than the reference fuel cycle. Fuel performance, safety and licensing, and environmental assessments were also performed for this composite improved cycle, and RD&D requirements and time schedules were estimated based upon an integrated development program leading to commercial introduction. Lastly, an evaluation was made of the resource utilization and economics of the composite improved PWR in the recycle mode, in order to establish how the implementation of these once-through cycle improvements affects the incentive for recycle.

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Section 1

EXECUTIVE SUMMARY

1.1 INTRODUCTION

The great majority of nuclear power plants, either in operation or planned worldwide, are light water reactors (LWRs) of either the pressurized water reactor (PWR) or boiling water reactor (BWR) type. Seventy such light water reactors, constituting about 50 GWe, are currently licensed to operate in the United States and an additional 96 LWRs, constituting about 105 GWe are under construction. A similar number of LWRs are operable, under construction, or on order in other nations throughout the world. Thus, if uranium demand is to be significantly reduced in the near and intermediate time frame, the uranium utilization efficiency of the LWR must be improved. Furthermore, techniques devised to accomplish this improved utilization must be backfittable into current reactor designs; otherwise the majority of generating capacity operating in the next several decades would do so in a less resource-efficient manner, and significant reductions in uranium demand could not be realized until well into the next century.

In recognition of this need to improve the fuel resource efficiency of the LWR, the U.S. Department of Energy has initiated a uranium utilization improvement program for light water reactors. This program is aimed at identifying modifications to current LWR designs and practices which will significantly reduce uranium demand and which have the potential for improved power generation costs. Such modifications include those that can be implemented in the near term and are, therefore, backfittable into existing plants as well as those that can be implemented only in the longer term and which may require new reactor configurations. The initial emphasis of this program is demonstration of the capability of current fuel designs to achieve extended burnup in the range of 50-55 MWd/kg batch-average exposure. Extending the discharge exposure to this level while maintaining approximately annual refueling is the single most effective method of improving the uranium utilization of current-design LWRs. Other mechanisms that improve the fueling efficiency of the LWR and which are being examined under DOE's overall program include the use of low-leakage fuel management schemes, modified fuel designs, axial blankets, fuel reinsertion, planned end-of-cycle stretchout, and a number of nonbackfittable

improvement options. The overall program is designed for completion by the mid-1990s, with the mid-1980s as the target date for demonstrating the extended-exposure capability of fuel employing essentially current designs. Demonstration of advanced fuel designs and some of the more exotic improvement options will proceed through the late 1980s and early 1990s.

The work presented in this report constitutes a comprehensive analytical study of potential PWR backfittable uranium utilization improvement options. A large number of potential improvement options were examined as part of the preliminary survey of candidate options. The most attractive of these options were then selected for detailed analysis and were included in a composite improvement case. This composite case represents an estimate of the savings in 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 (within about 10 years). The following table lists the options included in the composite design along with their potential for reducing U_3O_8 demand.

Table 1.1-1
POTENTIAL NEAR-TERM IMPROVEMENT OPTIONS IN

	<u>Reduction in 30-Year Cumulative U_3O_8 Requirements, %</u>
1. Extended-Burnup, Modified Fuel Management	11.0
2. Low-Leakage Fuel Management	2.6
3. Lattice Modifications	3.0
4. Axial Blankets	2.5
5. Full Use of Early Batches of Initial Core	0.9
6. End-of-Cycle Stretchout	3.0
Total *	21.0

* Percentage reductions in U_3O_8 requirements are not directly additive.

The essential features of each of the individual improvement options, together with the major difficulties and obstacles in implementing them, are discussed in the sections which follow.

1.2 EXTENDED-BURNUP, MODIFIED FUEL MANAGEMENT

The extended-burnup, modified fuel management option consists of increasing the discharge exposure of the fuel to about 50,000 Mwd/t batch average (compared to about 30,000 Mwd/t in current designs) and employing a five-batch fuel management scheme (compared to the currently-used three-batch scheme). The use of more in-core batches lowers the core-average burnup and therefore the end-of-cycle fissile inventory needed to compensate for the buildup of fission products. The use of higher discharge exposures allows annual refueling to be maintained with the lower fuel reload fraction of the five-batch scheme (one-fifth vs. one-third for the three-batch scheme). The benefit in terms of both uranium utilization and fuel cycle costs is maximized by combining these two mechanisms. Additional benefit can be realized if the burnup is increased still further and more in-core fuel batches are employed; however, the rate of improvement decreases beyond the values mentioned above.

The major technical concerns with this improvement option are fuel performance at high burnup and the potential increase in power peaking. The specific fuel performance effects which must be considered are pellet-clad interaction, fuel-rod internal pressure resulting from fission gas release, fuel assembly dimensional changes, and external clad corrosion. These potential problem areas are currently being evaluated as part of the extended burnup irradiation programs. Once data at high burnups are obtained from these programs, the necessity for fuel design changes can be assessed.

The second major difficulty with implementing this option is the added complexity it introduces to the fuel management. Power peaking tends to be higher (by about 10%) than that with conventional three-batch, standard-burnup fuel management. This higher peaking results from the high enrichments (4.3 wt%) and the relatively small reload fraction. The large reactivity differences between the fresh and previously burned fuel (particularly four-times burned fuel) cause poor power sharing between batches and large local flux and power gradients. More extensive experience with extended-burnup fuel management schemes may result in lower power peaking; however, at the present time it appears that it will be necessary to use enrichment zoning of assemblies and/or burnable poison to control these

potentially higher peaks within acceptable limits. Nevertheless, deployment of the extended-burnup option has a high probability of success. RD&D programs have already been initiated by DOE, EPRI and the fuel manufacturers with a number of different utilities actively participating in these programs. Furthermore, both national goals (the improvement of uranium utilization) and commercial incentives (the reduction of power generation costs) favor implementation of this option.

1.3 LOW-LEAKAGE FUEL MANAGEMENT

Present PWR fuel management schemes place fresh, high-reactivity fuel on the core periphery, while fuel with low reactivity (because of its low enrichment or prior burnup) is loaded in the core interior. This OUT-IN fuel management was developed in order to achieve flat radial power distributions in large, chemically shimmed PWR cores. However, when the radial power distribution is flattened in this manner, large flux gradients occur at the core boundary and the net radial leakage of neutrons from the core is increased relative to a fuel assembly arrangement in which lower reactivity fuel is placed on the core periphery. The OUT-IN fuel management scheme can be modified to reduce radial neutron leakage by loading partially burned fuel in most of the peripheral locations and the majority of the fresh fuel, mixed with the remaining partially burned fuel, in the core interior. Power peaks in the fresh assemblies are controlled by use of burnable poison shim rods which are either loaded in water hole locations or fuel rod locations, depending on the specific fuel design. This low-leakage fuel management scheme results in a reduction of the fissile feed requirements because of the lower radial neutron leakage and because of the higher importance weighting at the end of cycle of the high-reactivity fuel placed in the core interior.

Low-leakage fuel management schemes are presently being offered for commercial deployment by several reactor manufacturers. It has the disadvantage over conventional OUT-IN fuel management of requiring the use of burnable poison with each reload batch, of possessing a different and as yet unverified fuel duty cycle (i.e., high-low-high power density versus exposure), and of complicating analyses since the maximum power peaking factors occur late in the cycle rather than at or near the beginning of cycle. In the context of extended-burnup, modified fuel management, the

situation is aggravated by the high enrichments required. Considerably more burnable poison is required to achieve acceptable power distributions with this high enrichment fuel; these higher poison loadings can, in turn, result in a larger residual poison concentration at the end of cycle, which reduces the potential benefit of this option. To minimize the reactivity tied up in this residual, it is desirable to load more poison rods each with a lower initial concentration. However, for fuel designs where poison shims displace fuel rods, the linear heat rates of the remaining fuel rods must increase to maintain the same core power rating. Similarly, for fuel designs where poison shims are placed in unused control rod locations, the availability of such locations limits the reactivity that can be suppressed by this method. Thus, if low-leakage schemes are to be used with the extended-burnup, modified-fuel-management option, it appears from the analyses performed to date that more sophisticated methods of controlling local power variations, such as assembly enrichment zoning or the incorporation of gadolinium burnable poison admixed with the fuel, are necessary to achieve power distributions and thermal margins comparable with those of current PWR cores.

1.4 LATTICE MODIFICATIONS

Modifications to the standard lattice design can be made to increase the overall reactivity of the core. The two lattice modifications which are backfittable into existing reactor designs are decreasing the fuel rod diameter and reducing the fuel density. Both of these lattice changes soften the neutron spectrum by increasing the moderating ratio, thus causing an increase in core reactivity and, therefore, reducing the fissile inventory needed to maintain criticality at end of cycle. Increasing the lattice pitch is often mentioned as a means of softening the neutron spectrum; however, this lattice modification requires an increase in core diameter (and therefore vessel size) to maintain the same plant rating. Decreasing the fuel rod diameter was selected over reducing the fuel density because it gave the greatest uranium utilization improvement of the two remaining options and because it was felt to be more easily implemented from a RD&D standpoint.

At least one reactor vendor is already offering fuel with a somewhat wetter lattice than is currently being employed. An initial core loading with this fuel is currently scheduled for 1982. The question remains, however, how this option can be implemented in a core presently operating with standard design fuel. The safety and licensability of such a mixed core needs to be assessed before this improvement option can be implemented in other than new reactors. Two other potential problem areas with this option are its lower thermal margin, as a result of reducing the heat transfer area of the fuel, and its higher internal fuel rod fission gas pressure as a result of the loss of void space when the clad ID is decreased. Depending on the particular core involved, the former difficulty may or may not be a significant problem. In existing reactors, it may be possible to recover this potential margin loss by increasing assembly cross flow through the use of flow mixing grids. Once more data is available on the release of fission gas at extended burnups, an assessment can be made of the acceptability of the internal rod pressure in reduced diameter fuel. Present gas release models, which are based on semi-empirical formulations, have not been benchmarked at high burnups because of the lack of experimental data and therefore cannot be relied upon at this time to make accurate predictions.

1.5 AXIAL BLANKETS

Axial blankets composed of natural or depleted uranium have the potential for improving uranium utilization by decreasing the core leakage, by increasing fertile conversion (in the blanket material) and by increasing the discharge exposure of the enriched fuel region. By replacing the enriched fuel pellets at the ends of the rods (e.g., 15.2 cm (6 inches) at both the top and bottom) with blanket material, lower power is produced at the ends of the rods and therefore fewer neutrons are available to leak into the reflector where they can be lost through parasitic capture. Neutrons which would otherwise be lost are absorbed instead in the blanket material, primarily in U-238, producing additional fissile material. From a uranium utilization standpoint, it makes very little difference whether the blanket is composed of natural or depleted uranium; however, from the standpoint of separative work and fuel cycle costs, natural uranium is the preferred blanket material.

Since the power density in the blanket region is lower than it would be at these locations in an unblanketed core, the power density of the enriched fuel region (active core) must increase. This higher power density results in lower core thermal margins and higher peak axial discharge exposures. The acceptability of the lower core thermal margins resulting from the use of axial blankets depends on the specific core design (i.e., which uranium improvement options are included) and its impact on the core safety analysis (e.g., the LOCA analysis). The effect of the incremental increase in peak axial exposure can be determined only after the results of ongoing extended-burnup demonstrations have been completed. In addition, the higher active core power density and discharge burnup will aggravate the already potentially troublesome problem of fission gas internal rod pressure.

1.6 FULL USE OF EARLY BATCHES OF INITIAL CORE

Investigations of the reactivity and fissile content at discharge of the early batches of the initial core (i.e., those batches which are discarded after fewer irradiation cycles than a typical reload batch) reveals that their potential for additional exposure is very similar to that of typical reload fuel before its last irradiation cycle. This initial core fuel, therefore, has the potential for improving uranium utilization by reinserting it for an additional cycle of exposure and thereby more fully consuming the fissile material remaining in such fuel. Since use of this option increases the core average exposure by approximately 350 MWd/t, slightly higher enrichments are required for the fresh reload fuel. These higher enrichments compensate somewhat for the fewer number of assemblies loaded over the plant's lifetime (approximately 2.4% less) when reinsertion is practiced. Thus, the saving in U_3O_8 is only about 0.9% when the four low-exposure batches of the initial core are reinserted eight assemblies at a time over the next 24 years.

The full use of early batches option, although offering a relatively small benefit, is attractive because of the minimal requirements for

RD&D. The general feasibility of this option has been established both by analyses of current design cores and by utility practice of this option with existing three-batch fuel management. In the present application, however, this option is faced with the same difficulty with respect to power sharing and local power variations as the extended-burnup, modified-fuel-management and low-leakage options discussed earlier. These issues can only be resolved on a case-by-case basis and will probably require techniques similar to those mentioned earlier to obtain acceptable power distributions. In addition, the acceptability of reinserting fuel after long periods of storage in a spent fuel pool must be established. The unusual duty cycle of such fuel together with the increased corrosion from many years of storage will require further assessment to ensure the licensability of this option.

1.7 END-OF-CYCLE STRETCHOUT

End-of-cycle stretchout is an attractive method of increasing uranium utilization in that it has already been practiced at one time or another by a number of utilities. However, it has normally been used to gain cycle length flexibility, for example, by delaying the refueling of a plant to a period of lower power demand or to compensate for some unforeseen event, such as the forced outage of another unit. Planned stretchout has not been used extensively because its use reduces the ability to fully apply this technique when the need arises and because of the increased fuel management complexity which can occur in later cycles. Nevertheless, a considerable amount of cycle length flexibility still exists at no extra cost even when planned cycle stretchout is employed. Furthermore, as the cost of uranium increases, the additional cycle length that can be obtained at no extra cost also increases, providing more flexibility.

End-of-cycle stretchout can be accomplished by simple power coastdown whereby additional reactivity to extend the cycle is obtained through the reduction of the power level and coolant temperature, or it can be accomplished by employing feedwater-pressure augmentation whereby the feedwater temperature is first reduced by taking feedwater heaters out of service which

allows a period of continued operation at full-rated thermal power (but at a reduced thermal efficiency). Once full thermal power can no longer be transferred to the turbine or a turbine limitation is encountered, then power coastdown is initiated. Feedwater pressure augmentation increases the energy extractable from the core by about 0.5% over that obtainable by standard power coastdown; this translates directly to a 0.5% extra uranium utilization improvement. The ability to employ feedwater-pressure augmentation (i.e., maintain rated core thermal output) is limited primarily by restrictions on turbine operation with feedwater heaters taken out of service. Considerably more potential exists than is indicated by the 0.5% quoted above if turbines can be designed to operate over a larger range of secondary plant conditions (e.g., with more feedwater heaters out of service).

1.8 COMPOSITE IMPROVEMENT

An analysis of the composite effect of combining all the previously mentioned uranium utilization improvement options was performed as part of the present study. This analysis consisted of detailed multi-dimensional fuel management calculations using various core feedback options and functionalized cross sections. Starting with the initial core, cycle-by-cycle depletion calculations were performed to determine fuel enrichments, assembly loading patterns, and core power distributions. At the end of each cycle, EOC stretchout was employed to a minimum electrical output of approximately 70% rated power.

The equilibrium U_3O_8 and separative work requirements of the composite improved PWR were determined to be 22.5% and 9.7% lower than those of the reference PWR. Averaged over a thirty-year lifetime, the cumulative savings in U_3O_8 and separative work are 20.7% and 9.2%, respectively. The slightly lower reduction in fuel resource savings when averaged over thirty years is a result of the long approach to equilibrium (8 to 10 cycles) which is necessary to increase the feed enrichment from the initial core value of 2.31 wt% to the equilibrium cycle value of 4.70 wt% while maintaining reasonable power distributions.

As discussed previously for the individual improvement options, power peaking in the composite improved core is greater than that of the reference PWR core by about 18%. Most of this increase in power peaking is due to the five-batch, extended-burnup fuel management which has relatively large reactivity differences between fresh and partially burnt fuel.

It should be recognized that not all of the options included in the composite case will be adopted by all reactor operators; therefore, only part of the total improvement will be realized within the near term. Improvements will most likely be made incrementally, particularly with respect to extending fuel discharge exposure. Also, some of the options investigated may, in the long term, receive less widespread acceptance because of their adverse effect on core thermal margins.

There are a number of nonbackfittable improvement options which were not evaluated as part of this study but which can reduce U_{308} demand in a PWR substantially more than the composite value. These options include the use of more frequent refuelings, the use of variable lattice fuel assembly designs which require changes to the water-to-fuel ratio without refueling shutdowns, the extensive use of blankets, bundle reconstitution and reinsertion, and improved core reflectors. Evaluations are being conducted elsewhere to assess the full potential of these options and the likelihood of their success.

1.9 EFFECT ON THE RECYCLE MODE

The question has arisen as to the impact of the above once-through improvement options on the performance of the recycle mode of operation; that is, whether the improvement in uranium utilization from each of these options also carries over to recycle or whether the incentive for recycle is diminished as a result of the implementation of these once-through improvement options. This question is directly related to the mechanism by which each improvement option reduces U_{308} demand. In the once-through mode, uranium utilization is solely dependent on the fissile material charged per unit of

energy production, while for the recycle mode it is dependent almost exclusively on the conversion ratio. The fissile material charged can be reduced by increasing the conversion ratio or by decreasing the end-of-cycle fissile inventory required for criticality. However, the means of achieving these two effects usually conflict. Most often, optimal fuel utilization in the once-through fuel cycle tends to be achieved at the expense of conversion ratio by minimizing the end-of-cycle inventory requirements.

It should be noted that the considerations involved in the improvement of fuel cycle costs are somewhat different from those with respect to resource utilization. This is particularly true in the recycle mode. Fuel cycle costs are incurred both due to the depletion of fuel and from carrying charges on the fuel. Thus, although only the depletion component determines the resource utilization in the recycle mode, an economic assessment involves a balance between depletion charges and carrying charges on fuel inventories. This competition between depletion charges and fuel inventory carrying charges tends to result in designs which minimize the fissile inventory component at the expense of conversion ratio and resource utilization, i.e., designs which, from a resource utilization standpoint, are more nearly optimal for once-through fuel cycles.

The impact of each of the near-term, once-through improvement options on the recycle mode is shown in Table 1.9-1. This table is not intended to represent a strategy for the recycle mode, but rather to indicate the impact on the uranium requirements of the recycle mode when each of the once-through improvement options is employed. Some of the once-through improvement options are economical in the recycle mode even though they exhibit a uranium utilization penalty in this mode. Since the decision to deploy a fueling strategy is usually based on economics rather than resource conservation, it is likely that these economic options will be adopted if and when recycle is practiced despite their uranium utilization penalty. Furthermore, since some options are not associated with design changes (e.g., the full use of early batches of the initial core and end-of-cycle stretchout) they would be employed only when the operational mode and/or existing economics were favorable.

Table 1.9-1
EFFECT OF ONCE THROUGH IMPROVEMENT OPTIONS ON
THE URANIUM UTILIZATION OF THE RECYCLE MODE

	Reduction in 30-Year Cumulative U ₃ O ₈ Requirements, %
1. Extended Burnup, Modified Fuel Management	-4.0
2. Low-Leakage Fuel Management	1.2
3. Lattice Modifications	-2.5
4. Axial Blankets	1.2
5. Full Use of Early Batches of Initial Core	-0.2
6. End-of-Cycle Stretchout	-0.5
	-4.5*

*Percentage reductions in U₃O₈ requirements are not directly additive.

The percentage changes shown in Figure 1.9-1 apply (but not in a directly additive manner) to the reference benefit of recycle in the PWR which, for a 75% capacity factor and 0.2 wt% tails assay, amounts to a 32% cumulative U₃O₈ savings (38% net*) over a 30-year plant lifetime. When all the once-through improvement options are included in the recycle mode, these percentage benefits due to recycle are reduced to 10% cumulative (16% net). This reduction in the benefits is due primarily to the substantial improvement (21%) in resource efficiency in the once-through mode, and secondarily to the degradation in resource efficiency in the recycle mode as a result of extended burnup and lattice modifications. Consequently, it is concluded that the uranium utilization improvement options identified for near-term demonstration can obtain a substantial fraction of the uranium utilization improvement possible with recycle. It should be noted, however, that there are improvements which benefit the resource utilization in the recycle mode (e.g., drier lattices), but which do not benefit resource utilization in the once-through mode. These recycle mode improvements are not, in general, cost effective under the present economic assumptions.

*The net requirement is determined by taking the 30-year cumulative requirement less a credit for the ex-core fuel batches and fuel remaining in core after reactor's final year of operation. Such fuel can be used to reduce the requirements of newer reactors still operating.

SECTION 2

INTRODUCTION

2.1 BACKGROUND

Moved by concern for potential nuclear-weapons proliferation, President Carter announced on April 7, 1977 a new administration policy deferring the commercial use of plutonium as power reactor fuel in the United States. Subsequently, he urged other countries to join in reformulating a nuclear power policy and proposed, as a means of aiding this effort, that an International Nuclear Fuel Cycle Evaluation (INFCE) be undertaken to examine the ways in which the risk of weapons proliferation could be reduced while maintaining the benefits of nuclear power worldwide. Specifically, he stated that commercial reprocessing and recycling of plutonium in thermal reactors would be deferred indefinitely and that the U.S. fast breeder reactor program would be restructured to delay the date when breeder reactors would be put into commercial use and to place greater priority on the development of alternate nuclear fuel cycles and breeder designs which would exhibit reduced proliferation risk.

The Nonproliferation Alternative Systems Assessment Program (NASAP) was initiated by the U.S. Department of Energy to evaluate and provide recommendations for developing more proliferation-resistant civilian nuclear power systems. This program has provided a technical basis for the evolution of domestic and international nuclear policy, and has also been the source of many of the U.S. submittals to INFCE. A wide range of systems were evaluated by NASAP and from these studies have emerged several concepts to improve uranium utilization without the widespread use of separated plutonium fuel.

The NASAP evaluations also underscored the importance of developing improvements to light water reactors. A great majority of the nuclear power plants, either in operation or planned worldwide, are light water reactors of either the pressurized water or boiling water reactor type. Seventy such water reactors, constituting about 50 GWe, are currently licensed to operate in the United States and an additional 96 LWRs, constituting about 105 GWe, are under construction. A similar number of LWRs are in operation, under

construction, or on order in other nations throughout the world. If uranium demand is to be significantly reduced in the near and intermediate time frame, the uranium utilization efficiency of these LWRs must be improved; otherwise, the majority of the generating capacity operating in the next several decades would do so in a less resource-efficient manner, and significant reductions in uranium demand could not be realized until well into the next century.

The work presented in this report was initiated by the U.S. Department of Energy in recognition of this need to improve the fuel resource efficiency of the LWR. The program constituted a comprehensive analytical evaluation of methods of improving uranium utilization in operating pressurized water reactors. Attention was limited to options applicable to operating reactors, since only such "backfittable" design changes can improve the uranium utilization efficiency of the substantial number of light water reactors currently in operation or under construction. Although the improvements in uranium utilization realized by these backfittable modifications may not appear as dramatic as that which can be claimed for advanced reactor designs, the potential for rapid and widespread implementation of backfittable improvements in the LWR generally results in more substantial reductions in uranium demand than can be obtained with other alternatives until well into the next century.

2.2 PROGRAM OBJECTIVES AND SCOPE

The objective of the program summarized in this report was to perform an evaluation of the improvement in uranium utilization which can be obtained by incorporating various design and fuel cycle modifications in the current generation of pressurized water reactors operating in the once-through fueling mode. The focus of this work was on backfittable improvement options; however, a brief scoping assessment of some nonbackfittable options was also performed. The emphasis on improvements which are backfittable into operating PWR plants and which can be implemented in the relatively near future applies somewhat restrictive criteria for judging the acceptability of suggested improvements to PWR design, fuel management, or operating procedures. In particular, candidate improvements must be both technically and economically sound if they are to attain widespread acceptance

and implementation in commercial power stations, and capable of being developed and demonstrated consistent with the goal of attaining near and intermediate term reductions in uranium demand.

The work presented in this report consisted of three main tasks. In the first task, a large number of potential improvement options were surveyed with the objective of quantifying the improvement in uranium utilization anticipated for each alternative and of identifying areas where further development would be fruitful. Since the list of items which were evaluated was quite lengthy, simplified models were utilized in performing this survey. Many of the items in this survey of potential improvements were eliminated from further evaluation on the basis that they did not afford significant savings in uranium utilization, evoked disproportionately large technical problems or safety concerns, appeared uneconomic, or were either nonretrofitable or necessitated unreasonably long development and demonstration times.

The list of potential improvement options was eventually culled to a more manageable number of potentially attractive improvements which were subject to more detailed evaluation in the second task of the program. These items were judged as having the greatest potential for reducing uranium requirements and of having the potential for near-term implementation. This phase of the program concentrated on a more accurate assessment of the reduction in uranium requirements achievable with each option and upon potential technical, safety and licensing, and environmental issues which would have to be addressed as part of RD&D programs leading to commercial implementation.

The last phase of the program consisted of the evaluation of the uranium resource and economic performance of an improved PWR incorporating all of the improvements which were judged to be technically and economically sound. Options which were included in this composite case met the following criteria:

- The option must be retrofittable.
- The option must be economically viable so as to enhance the likelihood of utility adoption (or alternately the conditions under which the option became economically viable was identified).
- The option must pose no fundamental design, operability, safety or

licensing issues which are not resolvable through a defined R&D program.

- The inclusion of an improvement option did not preclude the utilization of other options which had a more significant uranium utilization improvement.

In order to accurately evaluate the improvement in uranium utilization which could be achieved by combining attractive options, detailed fuel management calculations incorporating, to the greatest extent possible, all of the improvement types were performed. Fuel performance, safety and licensing, and economic evaluations were also performed for this composite improved PWR case, and RD&D requirements and time schedules were developed based upon an integrated RD&D project leading to commercial introduction. Lastly, an evaluation was made of the resource utilization and economics of the improved composite PWR in the recycle mode, in order to establish how the implementation of these once-through cycle improvements affects the incentive for recycle.

2.3 GENERAL COMMENTS ON IMPROVING URANIUM UTILIZATION

In the present PWR once-through mode of fueling, uranium fuel enriched to about 3 wt% U-235 is typically employed. Energy is produced both by the U-235 initially loaded (about 65%) and by the fissile plutonium produced and burned in place (about 35%). The total energy produced is about 1 MWd/g U-235, roughly 20% greater than would be obtained by burning all of the U-235 initially loaded. The total fissile content of the spent fuel is typically about half as large as the initial U-235 loading. The fissile content of the fuel cannot be fully consumed during the operating cycle since fissile depletion is limited by reactor criticality constraints, i.e., there must be sufficient fissile content in the reactor at end of cycle to compensate for neutron losses to structure, core leakage, U-238 capture, etc. The initial U-235 loading of the fuel must consequently be sufficient both to provide for consumption during the operating cycle and to provide the required fissile inventory at end of cycle needed to compensate for neutron losses and maintain criticality.

When plutonium and uranium recycle are contemplated, only the quantity of fissile material consumed during the operating cycle is of direct significance from the standpoint of uranium utilization. The quantity of fissile material which remains unutilized in the spent fuel does not adversely affect uranium utilization since this material is recovered during reprocessing and utilized through recycle in subsequent operating cycles. Resource utilization in the recycle mode is improved by reducing the net consumption of fissile material, during the operating cycle through design changes which improve neutron economy. Since the energy produced per fission (per unit fissile material consumed) is essentially constant, reductions in the net fissile consumption can be achieved only by producing additional quantities of plutonium; improvements in neutron economy make additional neutrons available for plutonium production through additional captures in U-238. Uranium utilization in the recycle mode is often characterized by the conversion ratio, i.e., by the ratio of fissile atoms produced (plutonium) to fissile atoms consumed (plutonium plus U-235). Resource utilization in the recycle mode is enhanced by increasing the conversion ratio, so that additional quantities of plutonium are produced per unit of fissile material consumed.

The most frequent method of improving uranium utilization in recycle systems is to reduce the amount of moderation so as to obtain a "fast" neutron spectrum. With a fast neutron spectrum, losses of neutrons to the coolant and structural materials are reduced and plutonium production is enhanced through additional neutron captures in U-238, thus improving the conversion ratio. The fast breeder reactor is an example of a recycle system where captures in U-238 have been enhanced sufficiently that the conversion ratio exceeds unity, so that more fissile plutonium is produced during the operating cycle than is consumed to produce power. As a result of the increased neutron captures in U-238, the required initial fissile enrichment of the fresh fuel is high, as is the fissile content of the spent fuel; thus, such systems are attractive only in the recycle mode where the large fissile values of the spent fuel can be recovered by reprocessing and utilized through recycle in subsequent operating cycles.

In contrast to the recycle case, only the quantity of U-235 (or more precisely U_3O_8) which must be loaded into the reactor to produce a unit of energy is of importance to fuel resource utilization in the once-through fueling mode, since the fissile content of the spent fuel, by definition, is not recovered. Uranium requirements for the once-through fueling mode can consequently be reduced, in principle, by either of two methods: by reducing the amount of U-235 consumed per unit energy produced (i.e., by increasing the conversion ratio as in the recycle case), or by more completely utilizing the fissile content of the fuel so that spent fuel is discharged with lower residual quantities of fissile material.

As mentioned previously, the conversion ratio can be increased by design modifications which improve neutron economy to make available additional neutrons for plutonium production. This can be done by design changes which reduce parasitic neutron losses to structure or leakage, or by modifying the neutron spectrum so that neutrons are preferentially absorbed in U-238. However, since additional captures in U-238 decrease reactivity, the fissile content of the spent fuel often increases as a result of these design changes. Whether or not uranium utilization is improved as a result of design changes which enhance the conversion ratio consequently depends on whether the reduction in the quantity of U-235 consumed during the operating cycle is sufficiently large to outweigh the higher quantities of fissile material which may be unproductively discharged in the spent fuel. Although in principle design changes to enhance conversion ratio can improve the fuel utilization of the once-through mode, the detailed analyses performed as part of this evaluation indicate that the detrimental effects of increased spent fuel discharge inventory generally either exceeds the benefits of increased conversion ratio or results in small net gains, so that little improvement in uranium utilization in the once-through mode appears attributable to conversion ratio increases, at least for the range of design changes explored. This is in contrast to the recycle case where the fissile content of spent fuel is recovered, and hence where improvements in conversion ratio can markedly improve uranium utilization.

The second method of improving uranium utilization in the once-through fueling mode is to more fully consume fissile material present in the fuel, so that less fissile material is discharged (lost) in spent fuel. The second method, in practice, proved to be the primary means of improving uranium utilization with once-through fueling. The fissile content of the spent fuel can be reduced by improving neutron economy so that additional reactivity is made available for continued operation to further deplete the fissile content; alternately, improving neutron economy would allow the same period of operation to be obtained with a slightly lower initial enrichment. Improved neutron economy can be obtained by such means as reducing neutron losses to leakage from the core or parasitic captures in structural material, employing a larger number of refueling batches (which reduce neutron losses to fission products), or by increasing the water-to-fuel ratio to decrease the loss of neutrons to U-238. The latter is an example of options which decrease conversion ratio; improved uranium utilization is obtained in the once-through cycle in spite of this decrease in conversion ratio, since the ability to more fully utilize the fissile content of the spent fuel proves to outweigh the loss in plutonium production.

The observation that the great majority of the improvement in uranium utilization for once-through fueling is derived from the more complete utilization of the fissile content of the fuel, rather than by improvements in conversion ratio, has a number of ramifications. To the extent that the uranium utilization of the once-through cycle is improved by more complete utilization of fissile material, rather than by increases in conversion ratio, these gains in uranium utilization will not carry over into the recycle case; only those gains which result from improved conversion ratio can also be realized when recycle is performed. As a corollary, the maximum improvement in uranium utilization for the once-through fuel cycle corresponds to the complete utilization of the total fissile content of the fuel assembly. Since such complete utilization of fissile material is most nearly achieved when recycle is performed (i.e., the fissile values of spent fuel are recovered with reprocessing and utilized through recycle except for fuel cycle process losses), such improvements to the once-through

fuel cycle can do no better than approach those savings possible with recycle. In practice, it is never possible to fully utilize the fissile content in the once-through fueling mode, so that improvements to the once-through cycle are capable of obtaining only a part of the savings in uranium possible with recycle. If further savings in the once-through mode are to be realized, methods of increasing conversion ratio (without increasing the fissile content of spent fuel) must be developed; however, this study largely failed to identify such methods in the context of practical, retrofittable options.

With these basic considerations in mind, a substantial list of potential methods for improving uranium utilization in the once-through fueling mode was investigated. The results of this evaluation are summarized in Section 3, Summary, Conclusions, and Recommendations. Section 4 reviews the many suggested once-through improvement options which were addressed as part of the preliminary survey, while Section 5 reports the more detailed analyses performed for the selected improvement options which were judged to be practical while offering significant improvements in uranium utilization. Section 6 describes the composite improvement case which was analyzed to establish the uranium requirements, safety and environmental impact, economics, and RD&D requirements for a PWR incorporating attractive improvement options, while Section 7 discusses the effect of these improvements on the recycle mode.

Section 3

SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

3.1 SUMMARY AND CONCLUSIONS

The purpose of this study was to evaluate a variety of possible methods of improving the uranium utilization in a current-design, pressurized water reactor (PWR) operating on the once-through uranium fuel cycle. Only options which could be backfit into operating reactors were pursued. Initially, a survey of approximately twenty potential improvement options was conducted; each option was evaluated in terms of uranium utilization and its ability to be backfit into existing PWR designs. From this list of potential options, six were selected for detailed evaluation and incorporation into a composite improvement design. The selection criteria included the magnitude of the U_3O_8 savings achievable, the economic viability of the option, and the ease with which the option could be deployed (retrofittability).

The selected options were generally evaluated through multi-dimensional fuel management calculations using methods analogous to those employed in standard reload designs. In addition to the determination of fuel resource requirements and fuel cycle economics, the impact on safety of deploying the various options was assessed in a qualitative manner. Further, the research and development (R&D) required to verify the licensability of the options was estimated in conjunction with the likelihood of utility adoption. Once the individual options were examined and their requirements assessed separately, a composite improvement case, which includes all of the selected options, was evaluated. Finally, the impact on the recycle mode of incorporating the various once-through mode improvement options was determined.

In the remainder of this section, the major conclusions of the study are briefly discussed. The reader is referred to the appropriate sections in the body of the text for a comprehensive discussion of these items. Following the conclusions are the recommendations for further work. These recommendations are aimed at resolving some of the problem areas identified in the present study and at exploring methods of enhancing the benefits of certain of the selected improvement options.

3.1.1 Survey of Once-Through Improvement Options

The potential uranium improvement options evaluated during the survey phase of this study are shown in Table 3.1-1; these options have been grouped according to type for ease of discussion. Uranium utilization was used as the initial screening figure of merit because improvement of this parameter was the motivation behind the study, and because it is well defined and easily determined. Once a particular option demonstrated sufficient incentive in terms of improving uranium utilization, it was assessed according to its economic attractiveness and ease of deployment. Those options that satisfied all three criteria were then analyzed in greater detail during the second phase of the study as described in Section 5. The major results and conclusions from the survey of once-through PWR improvement options are presented in the following paragraphs.

3.1.1.1 Increased Burnup

Extending the discharge exposure while at the same time increasing the number of in-core fuel batches is the single most effective method of improving the uranium utilization of current-design PWRs. The major benefit is derived from the increased number of in-core batches. By increasing the number of in-core batches, the end-of-cycle average core exposure is reduced and therefore the fissile inventory required to compensate for fission product poisoning is lower. The use of higher discharge exposure allows annual refueling to be maintained with the lower fuel reload fraction which is associated with the use of more in-core fuel batches. Increasing the fuel discharge exposure without increasing the number of in-core fuel batches results in a savings in uranium of only about one-third that attainable when the two techniques are used together. For the majority of this study, a fuel management scheme employing five in-core batches and 50,000 MWd/t discharge exposure was used. Based on the point reactor calculations performed as part of the survey of once-through improvement options, this fuel management scheme has the potential for an equilibrium cycle savings in U_{308} of about 15% compared to the standard three-batch 30,000 MWd/t fuel management scheme.*

*The more accurate multi-dimensional fuel cycle analysis performed in the second phase of the program indicates a 12% savings in U_{308} .

Table 3.1-1

POTENTIAL URANIUM IMPROVEMENT OPTIONS FOR THE ONCE-THROUGH PWR

Improvement Option	Estimated Savings in U_3O_8 , %
1. Increased Burnup	15
2. Lattice Changes	
Increasing fuel rod pitch	6.5
Decreasing fuel rod diameter	5.0
Decreasing fuel density (or annular fuel)	3.0
Clad thickness reduction	1.0
3. Enrichment Zoning/Blankets	
Assembly enrichment zoning	0.1
Reduced diameter rods around water holes	0.4
DUPLEX pellets	0.1
Radial blankets	1.5
Axial blankets	3.0
Shroud and peripheral water gap thickness	2.0
4. Full Use of Early Batches	1.0
5. Improved Fuel Management and Control	
Low-leakage fuel management	3.0
Insertion of blanket rods in unoccupied control rod locations	0
Isotopically purified zirconium	1.5
Axial shim cutback	0.2
Axial relocation of fuel prior to discharge	0.6
Six-month cycle length	7.8
6. End-of-Cycle Stretchout	4.0
7. Variable Lattice/Spectrum Shift	
Removal of high-burnup rods prior to fourth cycle	0
Increasing moderator temperature (for each $10^\circ F$)	0.2

3.1.1.2 Lattice Changes

PWR fuel can be made more reactive by modifying the lattice in such a way as to increase the water-to-fuel ratio. Such modifications have two competing effects. First, by increasing the water-to-fuel ratio, the neutron spectrum becomes more thermalized and less neutrons are available for epithermal capture in U-238. With less captures in U-238, plutonium production is decreased resulting in a reduction in the fuel's conversion ratio. The second effect of increasing the water-to-fuel ratio is a reduction in the fissile inventory required to maintain criticality. Typically, this second effect outweighs the first effect in once-through PWR fuel cycles so that the fuel resource requirements decrease as the water-to-fuel ratio increases.

There are three primary methods for modifying the water-to-fuel ratio; these are increasing the lattice pitch, decreasing the fuel-rod diameter, and reducing the effective fuel density. The most effective of these three methods for increasing uranium utilization in a PWR is increasing the lattice pitch, while the least effective method is reducing the fuel density. The maximum possible increase in uranium utilization from increasing the lattice pitch is ~6.5%, from reducing the fuel rod diameter is ~5%, and from reducing fuel density is ~3%. However, increasing the lattice pitch requires an increase in core diameter (and therefore vessel size) to maintain the same plant power rating. This lattice modification is, therefore, not considered backfittable into existing PWRs. Furthermore, thermal-hydraulic and MTC considerations set practical limits for lattice variations which are only about half these values. It therefore appears that about a 3% increase in uranium utilization can be achieved due to lattice modifications without compromising the fuel performance or safety of a PWR.

One additional lattice change that results in improved uranium utilization is a reduction in clad thickness. By decreasing the clad thickness, parasitic captures in zirconium are reduced thereby requiring less fissile inventory to maintain criticality. Consequently, in the once-through cycle, less fissile material will be discharged in the spent fuel at end of cycle. This translates directly into a smaller fissile loading requirement. A small additional benefit is derived from an increase in water-to-fuel ratio when the clad thickness is decreased (assuming the pellet and gap dimensions are maintained). For a

5 mil clad thickness reduction, a 2.5% to 3.0% increase in uranium utilization can be achieved. To implement this option, however, requires assurance that clad integrity will not be adversely affected by reducing the clad thickness. This assurance will require an extensive irradiation testing program.

3.1.1.3 Enrichment Zoning/Blankets

Enrichment zoning of fuel rods within an assembly was investigated to determine the potential for reducing the peak power and peak burnup. A reduction in peak power can improve core operating margins while a reduction in peak burnup can improve uranium utilization by reducing the spread between limiting fuel rods and the remaining population of discharge rods. A 6% reduction in the assembly peaking factor was obtained through enrichment zoning; however, the margin improvement for the whole core will most likely be less than this value because fresh assemblies are often not limiting and because reductions in the peak-to-average assembly power do not usually translate directly into a margin improvement for realistic fuel-management schemes. The peak-to-average burnup is lower by about 4% as a result of enrichment zoning. This flatter burnup distribution has the potential of improving uranium utilization by about 0.1% if the maximum rod burnup is limiting. If the maximum rod burnup is not limiting, the flatter burnup distribution simply increases the fuel performance margins without increasing uranium utilization.

A technique similar to enrichment zoning was investigated. This technique consisted of decreasing the diameter of fuel pellets in the fuel rod locations around control rod guide tubes. This technique has the potential for improving uranium utilization by increasing the assembly effective water-to-fuel ratio, increasing the assembly average burnup, and flattening the burnup distribution across the assembly. The conclusion of this analysis is that a small uranium utilization benefit on the order of 0.3-0.5% is achievable by reducing the pellet diameter of fuel rods around water holes. However, the result of such a lattice design change would be either a severe reduction of permissible maximum linear heat rate from LOCA considerations (if the gap thickness were increased), or a fuel assembly design with multiple rod outside diameters. Neither of these changes appears acceptable in view of the small additional uranium utilization gain.

The use of a zoned pellet (DUPLEX pellet), which has a central core of natural uranium surrounded by an annulus of enriched fuel, has the potential for improving uranium utilization by reducing the average fuel temperature and thus neutron absorptions in the U-238 resonances. It was determined that this technique could reduce uranium requirements by only about 0.1%. Such a utilization improvement is small and does not compensate for the added separative work costs and the added manufacturing cost associated with the use of DUPLEX pellets. There may be, however, other reasons for changing to the DUPLEX design, such as LOCA or fuel performance behaviors, which are worthwhile. If the cost of changing to a DUPLEX pellet are justifiable based on these other considerations, the small additional benefit in the uranium utilization would then be an added bonus.

The use of radial blankets in existing PWRs is both infeasible and uneconomical because of the large size of PWR assemblies and the large fraction of assemblies which lie on the core periphery. Another option that might be possible is the use of triangular fuel assemblies to fill in corners around the periphery of the core. It is estimated that the use of such assemblies could result in an improvement in uranium utilization of about 1.5%. However, it is doubtful that such a small improvement would justify the cost of core design changes required to accommodate the addition of triangular assemblies on the core periphery.

The use of axial blankets shows considerably more promise than the use of radial blankets in a PWR as a means of improving uranium utilization. By using either natural uranium or depleted uranium at both ends of a standard fuel rod in place of some of the enriched uranium, a uranium savings on the order to 2-4% can be obtained. This improvement in uranium utilization is achieved at the expense of increased linear heat rate and burnup in the central portion of the core. These may lead to more limiting fuel performance conditions, such as fission gas release and/or a loss of thermal margin.

Improvement in uranium utilization can be obtained by reducing the number of neutrons leaking from the core. Thermal neutron leakage can be reduced by

increasing the thickness of the peripheral water gap surrounding the core. Similarly, fast neutron leakage can be reduced by increasing the thickness of the stainless steel shroud. The former technique has the potential of improving uranium utilization by about 0.4%, while the latter technique can improve uranium utilization by about 2%. The benefits derived from these two options are mutually exclusive, however, since increasing the gap also moves the shroud away from the core and therefore decreases the fast neutron reflection potential of the shroud. Because of limitations on available space, it is doubtful whether these options could be retrofit into operating plants; major design changes would be required for either option.

In summary, it appears that the use of axial blankets to replace the under-utilized fuel at the ends of the rods is a practical method of improving uranium utilization, but that the other options discussed under this general improvement category are not. The use of DUPLEX pellets may have a fuel performance benefit because of its lower average fuel temperature. The resource utilization improvement (which is estimated to be only about 0.1%) is an added benefit.

3.1.1.4 Full Use of Early Batches

The reinsertion of the early batches of an initial core for one additional cycle of exposure has the potential of improving uranium utilization by about 1% over a 30-year plant lifetime. This option is attractive since it has already been practiced on a limited scale and introduces no significant problems requiring extensive research and development.

3.1.1.5 Improved Fuel Management and Control

The radial leakage of neutrons from a conventional PWR can be reduced by the use of low-leakage fuel management schemes. In these schemes, the fresh reload fuel is loaded on the core interior with burnable poison used to control the power distribution to a more centrally peaked shape, thereby reducing the power and leakage of neutrons from the peripheral assemblies. The use of low-leakage fuel management improves uranium utilization by about 2-4% compared to the more conventional OUT-IN fuel management scheme.

The insertion of blanket-like rods in the unoccupied control rod locations was investigated as a means of increasing the conversion ratio of a PWR. The benefit of using such blanket rods is either negligible or negative. This conclusion is a result of the added absorption at end of cycle due to captures in U-238, the reduction of the assembly average water-to-fuel ratio due to the displacement of water in the control rod guide tubes, and the additional uranium requirements to fabricate the blanket rods. The use of such blanket rods in low-leakage fuel management schemes as an alternative to using burnable poison shims was investigated. It was determined that the negative reactivity effect of using such blanket rods is not sufficient to produce acceptable core-wide power distributions. Furthermore, the depletion of U-235 results in increased poison worth during the fuel's initial cycle of exposure which is a trend opposite to that desired for burnable poison shims.

By the use of isotopically purified zirconium (e.g., zirconium enriched in the isotope Zr-90), uranium utilization can be increased about 1.5% as the result of the decrease in thermal and epithermal captures in the zirconium metal. Such an enrichment process would require the development of advanced isotope separation techniques; the cost associated with such techniques has not yet been established and it is doubtful that the cost will be low enough to justify implementation of this process.

A small (about 0.2%) improvement in uranium utilization can be obtained by increasing the shim cutback (the distance from the ends of the core where the burnable poison concentration in shim rods is greater than zero). In current-design fuel, the shims have a 6 in. (15.2 cm) cutback to reduce shim residual at end of cycle. Substantially larger shim cutbacks, although increasing uranium utilization, would result in axial power distributions which are peaked at the top of the core and consequently have lower DNB margins.

A potential exists for improving uranium utilization by axially relocating the fuel inventory prior to its last cycle of irradiation. This technique involves splitting four-times burned fuel at the axial mid-plane and subsequently inverting the

two halves so that the low burnup ends of the assemblies are relocated into the center of the core. A simple inversion of assemblies without splitting has virtually no impact on uranium utilization because the burnup distribution in PWRs is nearly axially symmetric. Splitting then inverting can, on the other hand, increase uranium utilization by about 0.6%. This technique, however, results in unacceptable axial peaking factors at the beginning of cycle. The peaks burn down relatively quickly so that at about 3000 MWd/t into the cycle a potentially acceptable axial power distribution is obtained. The modest uranium utilization improvement of this option is not likely to offset the costs of axially relocating the fuel or the potential loss of thermal margin due to higher axial peaking factors.

The use of more frequent refueling can improve uranium utilization by reducing the core average exposure and therefore the fission product poisoning at end of cycle. By maintaining the fuel discharge exposure at 50,000 MWd/t and increasing the number of in-core batches from five to ten, a six-month refueling interval will result. This fuel-management strategy has the potential for improving uranium utilization by about 8.5% relative to the twelve-month/five-batch strategy. The viability of a six-month cycle length depends heavily upon the ability to reduce the refueling outage time. To make the six-month refueling schedule appealing to utilities, refueling times on the order of 7-10 days would be needed. This requires the use of some type of rapid refueling technique which, in general, necessitates substantial hardware changes. This improvement option was therefore deemed impractical for backfit into existing PWRs and was consequently not evaluated further in this study.

3.1.1.6 End-of-Cycle Stretchout

Current-design PWRs are inherently capable of operating for a substantial length of time beyond the nominal end of fuel cycle. This inherent capability is the result of a large negative moderator temperature coefficient, the Doppler effect and the xenon worth. The most common method of performing end-of-cycle stretchout consists of standard power coastdown where the achievable power is limited by the ability of the turbine to draw thermal power at a lower steam pressure. A variation of this method, called feedwater-pressure augmentation,

is to initially reduce feedwater temperature by taking out of service feedwater heaters. By securing extraction steam to the feedwater heaters and therefore lowering feedwater temperature, more heat can be transferred across the steam generator for a given steam pressure and more steam can pass through the turbine to produce electrical energy. This procedure allows extended operation at full core thermal power before power must be reduced as in the standard coastdown mode of operation. By employing end-of-cycle stretchout, 3 to 5% better uranium utilization can be achieved than without the use of end-of-cycle stretchout. For a given electrical power reduction, FWPA increases the energy extractable from the core by about 0.5% over that obtainable by standard power coastdown. This extra electrical energy extraction translates directly to a 0.5% uranium utilization improvement.

3.1.1.7 Variable Lattice/Spectrum Shift

The variable lattice concept was assessed briefly by removing a number of fuel rods after the fourth cycle of operation of a five-batch fuel management strategy and then reinserting the assembly for its fifth cycle of exposure. The fuel rods removed from such assemblies were those with the highest exposures; analyses were not performed for the reuse of these removed fuel rods. This method of employing the variable lattice design showed no uranium benefit compared to the standard fuel management. Two factors contributed to this result. First, the lattice design employed in this study was that of the standard System 80TM PWR. Subsequent studies of this concept by others ⁽¹⁾ have started with a fuel lattice design which is tighter than that of the standard PWR. This results in an increased conversion during the early cycles of residency of the fuel and, in the later cycles when the fuel rods are removed, affords the opportunity to consume the added fissile material bred earlier. The second reason that no benefit was obtained by use of this variable lattice technique is the fact that the removed fuel rods are assumed not to be used elsewhere. Thus, while there may still be some long-term incentive for the variable lattice concept, near-term applications which could be readily retrofitted into operating PWRs do not result in a significant uranium savings.

One potential way of improving uranium utilization in PWRs consists of increasing the moderator temperature, which has the effect of increasing the conversion ratio through a harder neutron spectrum. Toward the end of cycle, the moderator temperature is reduced back to its reference value allowing the reactor to operate beyond its nominal cycle length. An added benefit is also obtained from the higher electrical efficiency due to higher secondary steam pressures. The total increase in uranium utilization as a result of increasing moderator temperature is about 0.25% for every 1°F increase. Offsetting this benefit is the decrease in thermal margin which results from the higher moderator temperature. The overall result is that for every 1% improvement in uranium utilization accomplished by increasing the primary coolant temperature, a 2% loss in thermal margin or power capability results. Reactors with no extra thermal margin are therefore not able to utilize the benefit of higher moderator temperature without decreasing power level.

3.1.2 Detailed Analysis of Selected Once-Through Improvement Options

In the previous section, a large number of potential uranium utilization improvement options were summarized. Some options were shown to possess significant potential for reducing the U_3O_8 requirements of PWRs operating on the once-through cycle. Those options which, in addition to providing a fuel resource incentive, were also determined to be backfittable and economically attractive were subsequently evaluated in greater detail. The options selected for further analysis were:

- . Increased burnup with modified fuel management
- . Low-leakage fuel management
- . Lattice modifications
- . Axial blankets
- . Planned end-of-cycle stretchout
- . Full use of early batches

After evaluating these options individually in terms of their fuel resource requirements, their impact on safety, their economic viability, and their technical status, they were combined in a single composite improvement case. The results of this composite case represent a reasonable estimate of the behavior of a PWR optimized in the near-term for improved uranium utilization. In the sections

that follow, each of the selected improvement options, together with the composite improved case, will be summarized and the major conclusions regarding each option will be stated.

3.1.2.1 Extended Burnup With Improved Fuel Management

Five-Batch, OUT-IN Fuel Management - The five-batch, extended-burnup OUT-IN fuel management scheme was analyzed using two-dimensional, coarse-mesh PDQ and ROCS calculations. The equilibrium cycle enrichment predicted by these spatial calculations was 4.3 wt%; this compared to 4.1 wt% predicted by the point reactor calculations performed during the survey of once-through improvement options (cf. Section 4.2.1). This higher fuel enrichment (and associated higher uranium requirement) is the result of an increased core leakage and a higher core-average, end-of-cycle exposure as compared to those assumed in the point reactor calculations performed in the survey of once-through improvement options. The increased core leakage is a result of the relatively high power produced in the fresh reload assemblies placed on the core periphery. The higher core-average, end-of-cycle exposure is the result of a shift in cycle-dependent burnup ratios toward the earlier cycles, i.e., the fuel receives relatively more burnup during its first few cycles of exposure than it does during its last few cycles of exposure. The net result of these effects is a uranium savings of approximately 12% in the equilibrium cycle, as compared to 15% predicted by the survey of once-through improvement options. Because of the lower uranium utilization of the initial core and transition cycles, the 30-year cumulative uranium savings is 11% compared to the reference three-batch fuel management.

As a result of the large reactivity difference between fresh fuel and fuel with three and four cycles of exposure, the power sharing between batches in a five-batch, high-burnup core is poorer than that in a standard three-batch core. This poor power sharing results in higher assembly powers and higher rod-to-assembly peaking factors. The overall result is that the maximum one-pin peaks in the five-batch, extended-burnup core are about 10 to 12% higher than those of the reference core.

Five-Batch, Low-Leakage Fuel Management - In an effort to decrease the higher core leakage experienced with the five-batch, extended-burnup fuel cycle, a low-leakage fuel management strategy was analyzed. In this strategy, fresh fuel containing burnable poison shims (B_4C in an Al_2O_3 matrix) is loaded in the core interior while once-burned fuel is shuffled to the core periphery. By a judicious choice of fuel enrichments, burnable poison concentrations, and fuel locations, a more centrally-peaked core radial power distribution can be obtained than with the more conventional OUT-IN fuel management. In addition to the reduction in radial core leakage, the low-leakage fuel management scheme derives benefit from the higher importance weighting placed on the fresh fuel when it is in the core interior. The higher core reactivity resulting from these two mechanisms is diminished somewhat by an end-of-cycle shim residual. Because of the high enrichments needed in the five-batch, extended-burnup fuel cycle and the high burnable poison loadings which are required to control the power distribution, a portion of the burnable poison remains at end of cycle. The negative reactivity tied up in this residual burnable poison represents a potential savings in U_3O_8 on the order of 1 to 1½%.

Averaged over a 30-year reactor lifetime, the low-leakage fuel management results in a reduction in U_3O_8 requirements of about 2.5% as compared to the five-batch, extended-burnup, OUT-IN fuel-management scheme.

Fuel Performance at High Burnup - Before a commitment is made to extend burnup, its impact on fuel performance must be carefully assessed. The four main areas which need evaluation are pellet-clad interaction, fuel rod internal pressure resulting from fission gas release, fuel assembly dimensional changes, and external clad corrosion. Although fuel exposure is an important ingredient in PCI failures because of clad creep-down and the increased availability of certain fission products which promote stress corrosion cracking, the effects of fuel exposure appear to saturate before the burnups of conventional fuel cycles are reached. Hence, further increases in discharge exposure may not increase the probability of fuel failure due to pellet clad interaction. Test programs sponsored by DOE and others which will extend the data to higher burnups

are already under way and are expected to confirm this position prior to the full-scale deployment of extended-burnup fuel cycles.

Extending fuel burnups gives rise to two concerns in the area of fuel assembly dimensional changes: bowing and radiation-induced axial growth. Excessive bowing could affect power peaking, DNB, and assembly handling. Axial growth is well characterized and can be accommodated by proper design allowance between the assembly upper-end fitting and the top of the fuel rods. Although engineering evaluations and tests are needed to determine if current fuel assemblies have adequate margin at extended burnups, it appears likely that the effects of fuel assembly dimensional changes can be accommodated.

Although current PWRs have not experienced excessive corrosion on the outside surface of fuel rods, a thicker layer of oxide and crud deposits may develop because of the increased residency times of the fuel in extended-burnup fuel cycles. The concern is that the increased insulation caused by these thicker layers may raise the oxide-cladding interface temperature sufficiently to further accelerate the corrosion of the zircaloy cladding. This is the subject of a joint EPRI/C-E/KWU program initiated in 1978; corrosion data for exposures up to 50,000 MWd/t will be available from this program at the end of 1980.

Extending the fuel burnup increases the amount of fission gas which is available for release in the event of power ramping during maneuvering or various postulated plant occurrences. The consequences of fission-gas release are more severe at high burnups because of the larger fission-gas inventory within the fuel, increases in the fission gas release rate which occurs at high burnups, and a decrease in the void space available for accommodating the fission gas as a result of clad creepdown and fission product swelling of the pellets. Furthermore, the release of the gaseous fission products, xenon and krypton, can reduce the heat conductance through the gap between the fuel and cladding, resulting in an increase in the fuel temperatures and the subsequent release of additional quantities of fission gas. Prediction of fission gas release is extremely difficult at the present time because of uncertainties in fission gas release enhancement at high burnups. The lack of data inhibits the development of accurate gas-release models for high burnup. Fission gas release is the subject of a

number of ongoing fuel-development programs; as data becomes available it will be incorporated into fission-gas release models.

Impact on Core Physics Characteristics and Safety Evaluation - The boron worth of the five-batch, extended-burnup fuel cycles is lower than that of the standard three-batch cycle. This safety-related parameter is important in the context of the over-dilution accident. This accident is expected to be less severe for the five-batch, extended-burnup cores than for the standard three-batch core because more time will be available to detect and terminate the dilution operation before the shutdown reactivity margin is reduced to zero.

The moderator temperature coefficients for the five-batch, extended-burnup cores are more negative at both beginning and end of cycle than the standard three-batch core. The more negative MTCs of the five-batch cores mitigate the consequences of the warm-water accidents but make potentially more severe the consequences of cold-water accidents. For the steam-line-break accident, in particular, the results are further aggravated by the reduced shutdown worth of the control rods (discussed below).

The fuel-temperature coefficients for the five-batch, extended-burnup cores are slightly more negative at both beginning and end of cycle than the standard three-batch core. The small differences in FTCs should not significantly alter the consequences of anticipated operational occurrences or other postulated accidents.

Calculations of the shutdown worth for the five-batch, extended-burnup cores indicate a loss in total and net rod worth of about 10% compared to the three-batch core. This reduction in worth of control rods appears to result from a neutron spectrum change due to the higher feed enrichments and the high core average exposures. The resulting higher fast-to-thermal flux ratio reduces the effectiveness of the control rods which are primarily thermal absorbers. In addition to the loss in margin for the steam-line-break accident, the reduced rod worth lessens the severity of a rod-ejection accident.

In general, the safety concerns of the five-batch, extended-burnup PWR are not substantially different from those of the three-batch reference PWR. Those areas that may affect the safety evaluation are the higher power peaking of the five-batch core and the fission gas internal pressure of extended-burnup fuel. It appears that it may be possible to alleviate these concerns by design changes to the fuel and by the use of more advanced fuel management schemes.

Economic Assessment - The five-batch, extended-burnup cycles are more economical than the reference three-batch cycle. In the near term, when the price of uranium is \$40/lb, the five-batch, extended-burnup cycles have fuel cycle costs approximately 10% lower than the reference three-batch cycle. Similar cost savings are obtained in the longer term when the price of uranium is assumed to increase to \$100/lb.

3.1.2.2 Lattice Modifications

Available Options - Uranium utilization in PWR cores can be improved by changing the fuel lattice to obtain a more thermalized spectrum. This can be accomplished by three general techniques: increasing the rod pitch, reducing the fuel rod diameter, or decreasing the effective fuel density (alternately, using annular fuel). The largest benefit can be achieved by increasing the fuel-rod pitch; a maximum benefit of 6.5% can be obtained at a rod pitch of .56 in., which corresponds to a water-to-fuel ratio of about 2.9. The next most effective lattice modification from the standpoint of increasing uranium utilization is reducing the fuel rod diameter. A maximum benefit of about 5% is obtained by reducing the rod diameter to .35 in. which corresponds to a water-to-fuel ratio of about 2.8. The least effective method of improving uranium utilization is through reductions in the fuel density. A maximum of 3% improvement in uranium utilization results if the fuel density is reduced from 10.0 to 7.9 gms/cc. Although increasing the lattice pitch is the most effective method of improving uranium utilization, it requires an increase in core diameter (and therefore vessel size) to maintain the same plant rating. This lattice modification is therefore not considered to be backfittable into existing PWRs. There are several major design constraints which limit the potential of the remaining two lattice modification improvement options to less than their maximum benefits. These design constraints

include the requirement to maintain a negative moderator temperature coefficient and an acceptable core thermal hydraulic performance in terms of DNBR. Considering these practical constraints, it appears that a 3% improvement in uranium utilization as a result of lattice modifications is a practical value.

An additional method of improving uranium utilization is through reduction in the clad thickness. The benefit from this technique results from a slight increase in the water-to-fuel ratio and from a reduction in parasitic absorptions in the cladding material. A 10% reduction in clad thickness for the System 80TM PWR results in approximately a 1% improvement in uranium utilization. The combination of reduced pellet diameter and reduced clad thickness were taken in this study as the modified lattice which is subsequently analyzed in the composite improvement design. These modifications were chosen based on a balanced consideration of resource utilization benefit and fuel performance (discussed below). The total benefit of these two lattice changes is about 3 to 4% in terms of U_3O_8 requirements.

Impact on Core Characteristics and Fuel Performance - An increase in water-to-fuel ratio leads to a positive increase in moderator temperature coefficient; consequently, the allowable increase in water-to-fuel ratio is limited. Operational considerations dictate that the moderator temperature coefficient remain negative; this is reflected in tech spec limits on operating reactors. Further limitations on the value of moderator temperature coefficient may be imposed if the anticipated transient without scram (ATWS) event is included in the design basis. The fuel pellet and clad thickness reductions selected for the modified lattice result in a water-to-fuel ratio change of about .25. This change in water-to-fuel ratio results in a MTC which is about $.2 \times 10^{-4} \Delta\rho/^{\circ}F$ less negative than that of a core employing the standard fuel design. This change in MTC is acceptable for a five-batch, extended-burnup core because the MTC of this core with the standard lattice design is significantly more negative than that of the reference three-batch core. Therefore, with a lattice change of this magnitude, an acceptable MTC will be maintained.

Fuel rod internal gas pressure as a result of fission gas release may be a limiting fuel performance criteria for extended-burnup designs, and this potential problem is made somewhat more severe by reductions in fuel rod diameter because of the concomitant reduction in fission gas plenum volume. Present gas release models, which are based on semi-empirical formulations, have not been benchmarked at high burnups because of the lack of experimental data and therefore cannot be relied upon at this time to make accurate predictions. Unacceptable internal rod pressures are obtained if the NRC model with its burnup-dependent enhancement factor is used. C-E is in the process of developing a fission-gas release model for extended-burnup fuel. This model is based on recent gas release data from various ramp tests from ongoing fuel performance verification programs. The C-E developmental model predicts gas pressures which are considerably below those predicted by the NRC model. With the C-E developmental model, acceptable internal gas pressures can be obtained if the no-lift-off criterion* is utilized.

Although the use of annular pellets was not chosen for the modified lattice design of this study, it is an attractive alternative for applications with extended burnup because of its lower fission gas internal rod pressures and practically unchanged thermohydraulic performance. It may be prudent to develop this fuel type as a contingency in the event that the modified lattice design is unacceptable from the standpoint of internal fission gas pressure and/or thermohydraulic performance.

The thermal hydraulic performance of the modified lattice design was compared with that of the standard System 80TM core design. A 3-4% loss in overpower margin results from the reduced heat transfer area and lower coolant flow velocity of the modified lattice design. This loss in overpower margin is based on maintaining the same DNBR as the standard System 80TM design. Because of the conservatism in overpower margin included in the standard plant design, a reduction in margin of this magnitude will have only a small impact on the overall thermohydraulic performance of the core. Nevertheless, a detailed analysis is necessary to confirm the feasibility of using the modified lattice design in the five-batch, extended-burnup fuel cycle.

*This criterion is that fuel rod internal pressure will not increase to such an extent that the cladding expands away from the fuel into the coolant channel, thereby impeding the flow of coolant and degrading the clad/fuel gap conductance. This criterion allows internal pressures that are greater than the primary system pressure because the strength of the cladding material and its ability to resist outward expansion is considered.

Reduction of the fuel rod diameter and clad thicknesses makes the modified lattice design more susceptible to fuel-rod bowing. The major impact of this fuel-rod bowing is a potential for channel closure. The estimated increase of 11% in channel closure is, however, offset by the 11% increase in channel spacing achieved by the use of smaller fuel-rod diameters.

Economic Assessment of Modified Lattice Designs - A fuel cycle cost savings on the order of 3-4% compared to the standard lattice design can be achieved by using the modified lattice which incorporates a reduced fuel pellet and clad thickness design. This savings in fuel cycle cost is exclusively a result of lower uranium and separative work requirements. The charges for fabrication and spent fuel disposal were assumed to be on a per-assembly basis and therefore constant with changes in the lattice design. There would, in fact, be a small additional savings due to the reduction in requirements for zircaloy clad.

3.1.2.3 Axial Blankets

Available Options and Their Impact on Fuel Resource Requirements - Two types of axial geometries were examined as part of this study. The first option, which would be backfittable into existing PWRs, preserved the overall length of the fuel assembly by replacing enriched fuel pellets at the top and bottom of the fuel rod with blanket pellets. The second option, which would not necessarily be backfittable into current-design PWRs, kept the enriched fuel length constant and concatenated blanket material at either end of the fuel rod. For the first option, the blanket length was varied from 4 inches to 12 inches. Two different blanket materials were studied in this evaluation: natural uranium and depleted uranium with a U-235 enrichment of 0.2 wt%. The axial blanket concept was evaluated in the context of both a standard three-batch fuel management and the five-batch, extended-burnup fuel management.

For the standard three-batch fuel management, the use of a shorter active fuel length results in a net increase in uranium utilization of about 4.2%, while the use of a longer fuel rod results in a uranium utilization improvement of only 0.6%. This difference in uranium utilization for the two different rod geometries reflects the fact that the axial leakage in current-design PWRs is relatively small and that the major benefit from using an axial blanket is the replacement of the under-utilized enriched fuel pellets at the ends of the rods. For the five-batch, extended-burnup fuel-management scheme, a more extensive evaluation was performed. For both natural uranium and depleted uranium blankets, the uranium savings as a function of blanket length is nearly linear over the range 0 to 20 cm blankets. Beyond about 20 cm, the benefit from employing additional blanket material saturates, resulting in a maximum benefit of about 3.3% at a blanket length of about 25 to 30 cm. The uranium savings is nearly independent of the blanket material (natural uranium or depleted uranium) with depleted uranium showing only a slight advantage. In contrast, the separative work savings is much more for the natural uranium blanket than for the depleted uranium blanket because neither blanket material has a separative work requirement, while in the case of the depleted uranium blanket the active core enrichment is higher. The maximum separative work benefit for natural uranium blankets is about 1.8% and occurs at a blanket length of 20 cm. For depleted uranium, the maximum separative work benefit is only about 0.5% and occurs at a blanket length of about 16 cm.

Impact on Power Peaking and Peak Pellet Burnup - The potential resource benefits resulting from the use of axial blankets (with constant fuel-rod length and power level) may be limited by core power peaking and peak burnup considerations. As an example, for the five-batch, extended-burnup core with a six-inch blanket, the initial core axial power peaks may increase by as much as 10% for a natural uranium blanket and 12% for a depleted uranium blanket. The increase in axial power peaking as a consequence of employing axial blankets is only about half these initial core values in equilibrium. These power peaking factors are a function of blanket length and increase nonlinearly as the blanket length increases. Due to a more centrally peaked power distribution, blanketed cores have a higher burnup in the active core region than do unblanketed cores. The higher active core power together with the increased

discharge burnup will aggravate the already potentially troublesome problem of fission gas internal rod pressure.

Economic Assessment of Axial Blankets - The use of natural or depleted uranium blankets is an economical improvement option for the once-through fuel cycle. A slightly greater fuel-cycle cost savings is obtained for the natural uranium blanket with an optimum cost savings occurring at a blanket length of approximately 25 cm. A fuel-cycle cost savings of about 2% is obtained by employing a blanket of this length.

3.1.2.4 Full Use of Early Batches of Initial Core

Available Options and Their Impact on Fuel Resource Requirements - Of the many ways of reinserting fuel into operating reactors, two have been studied as part of this evaluation: reinserting assemblies into the same reactor in a later cycle, and fuel sharing between two different reactors. In the first method, the assemblies from the initial four batches of a five-batch, extended-burnup core were reinserted eight at a time into subsequent cycles. Over 30 years of reactor operation, approximately 2.4% fewer assemblies are loaded with this reinsertion scheme. Since reinserting initial core assemblies results in a higher core-average exposure, slightly higher feed enrichments are required compared to the case without reinsertion. As a result, the 30-year resource savings due to fuel reinsertion are approximately 1% and 0.4% for uranium and separative work, respectively. The second method of implementing this improvement option, i.e., fuel sharing, shows resource savings of similar magnitude.

The Impact on Core Power Distributions - A number of earlier evaluations of fuel reinsertion have established the general feasibility of this improvement option. The maximum planar pin peaking value is usually no larger than that obtained without reinsertion. These earlier evaluations were performed for more standard fuel-management schemes (e.g., for three-batch, standard-burnup schemes). For the five-batch, extended-burnup, fuel-management scheme, it is expected that the maximum pin-peaking value will remain approximately the same with and without reinsertion if the number of fresh assemblies loaded remains the same. Every fifth year, however, a smaller number of fresh assemblies is loaded into the core employing fuel reinsertion. At these times, the maximum pin-peaking factor will probably increase slightly since there will be a fewer number of fresh assemblies over which to distribute the power. The overall impact of

this higher power peaking will have to be established on a case-by-case basis for the particular core employing the fuel reinsertion strategy.

Economic Assessment of Fuel Reinsertion - Because of the fewer number of assemblies manufactured when fuel reinsertion is practiced, fabrication charges and spent fuel disposal charges are lower. These fuel cycle service cost savings together with the reductions in uranium and separative work requirements result in a net reduction in fuel cycle cost of approximately 0.5% levelized over 30 years.

3.1.2.5 End-of-Cycle Stretchout

Available Options and Their Impact on Fuel Resource Requirements - Two methods of accomplishing planned end-of-cycle stretchout have been evaluated as part of this study. These two methods are power coastdown and feedwater-pressure augmentation followed by power coastdown. The two methods differ in that with feedwater-pressure augmentation feedwater temperature is decreased by securing extraction steam to the feedwater heaters. The decrease in extraction steam flow to the feedwater heaters permits lower pressure operation before the volumetric flow limit on the turbine admission valve is reached. This method of stretchout permits extended operation at rated thermal power but at a lower thermal efficiency. Once rated thermal power can no longer be transferred to the turbine, standard power coastdown can be initiated to proceed further with cycle extension.

The ability to operate a reactor beyond its nominal end-of-cycle condition by employing end-of-cycle stretchout, enables the use of a lower feed enrichment to obtain a given cycle length. This technique results, therefore, in a savings of both uranium and separative work compared to the case where stretchout is not employed. The magnitude of these fuel resource savings is dependent on the minimum plant output which is allowed during the stretchout evolution. The lower the plant output, the greater the resource savings. Because feedwater-pressure augmentation allows operation for a longer period at full core rated thermal output, the resource savings are somewhat higher with this mode of operation. Uranium savings on the order of 3 to 5% can be obtained for practical values of the minimum plant power output. Feedwater-pressure augmentation gives about 0.5% greater savings than does normal power coastdown.

The secondary plant characteristics control the extent to which the coastdown operation and the feedwater-pressure augmentation operation can be employed. Feedwater-pressure augmentation is limited by the sizing of the steam piping system (e.g., the turbine admission valve) and the margins allowed in the turbine warranties which determine how far the secondary pressure can be lowered before power must be reduced. In the present study, a maximum feedwater temperature reduction of 28°C was used; this corresponds to isolating two feedwater heater strings. This limitation on feedwater pressure is based on operating instructions for turbines with feedwater heaters cut out. If a greater reduction in feedwater temperature were allowed, a substantially higher differential benefit could be obtained by the use of feedwater-pressure augmentation. It is felt that a much greater benefit could be obtained by this mode of operation if turbines were analyzed or redesigned with this operation specifically in mind.

The Effect on Core Characteristics of End-of-Cycle Stretchout - No significant licensing concern was identified for planned end-of-cycle stretchout. The greatest impact of the end-of-cycle stretchout operation appears to be on moderator temperature coefficient. Depending on various fuel cycle parameters, for example, cycle length, fuel discharge exposure, etc., the moderator temperature coefficient will be about the same or slightly more negative than the no-stretchout case if the resulting core average exposure is increased beyond that of the no-stretchout case. The primary reactor accident which has potentially more severe consequences is the steam-line-break accident. However, based on present safety analyses, there should be sufficient margin in the rod worth to cover the extra reactivity incurred in this accident as a result of a more negative moderator temperature coefficient.

Economic Assessment of End-of-Cycle Stretchout - With current uranium prices, planned end-of-cycle stretchout shows economic optima for stretchout times between about 8 and 16 days depending on the mode of operation and the cost of replacement power. An energy cost savings of about 2% is obtained when planned end-of-cycle stretchout is employed, with feedwater-pressure augmentation showing about a 0.5% benefit relative to normal power coastdown. Feedwater-pressure augmentation is more economic than normal power coastdown because less replacement energy needs to be purchased during the stretchout. The effect of higher replacement energy cost is to shift the optimum stretchout period to a fewer

number of days and to increase the relative savings of feedwater-pressure augmentation compared to normal power coastdown. The effect of higher uranium prices is to increase the optimum stretchout time and to increase the cost savings due to planned stretchout.

3.1.2.6 Technical Status and RD&D Requirements

Extended Burnup - There are two main issues which need resolution prior to the fullscale use of the five-batch, extended-burnup fuel cycle. The first issue is fuel performance for exposures significantly greater than 30,000 Mwd/t. Under this general area of fuel performance are the individual issues of fission gas release, pellet clad interaction, fuel dimensional changes and clad external corrosion. These issues are currently under investigation as part of DOE's integrated high-burnup demonstration program. The second issue which must be addressed for the five-batch, extended-burnup fuel cycle is the potential loss of thermal margin due to higher fuel pin powers. The primary source of the high pin powers is intrinsic in five-batch fuel management. With the smaller batch size, power sharing between assemblies becomes more difficult and higher power densities occur in the fewer fresh assemblies. In the low-leakage fuel management scheme, these higher values are further aggravated by the displacement of fuel pins by burnable poison shim rods. This second issue will be addressed to some extent in the later part of the ongoing DOE programs. However, more extensive fuel management development must be performed to optimize loading patterns in terms of their pin power peaking. Analyses should be conducted to determine quantitatively the effects of the higher pin power values on thermal margin and the DNBR. If the margin loss is unacceptable, methods to reduce the pin power values, such as zone-enriched fuel assemblies or more effective shim design (e.g., the use of gadolinium) should be investigated.

Lattice Modifications - Fuel rods with smaller pellet diameters and/or thinner clad thicknesses than those used in the reference design have been manufactured by other reactor vendors and are currently being irradiated in operating reactors under various fuel demonstration programs. In addition, ongoing DOE-sponsored programs have a number of alternate fuel designs, including both solid and

annular fuel pellets, earmarked for irradiation to high burnup values. In addition to such irradiation demonstrations, detailed nuclear and thermal hydraulic analyses must be conducted for specific fuel designs. Also, hardware experiments should be conducted which include hydraulic flow testing and critical heat flux testing for designs which depart significantly from those currently employed. In addition, static and dynamic structural tests should be performed to determine the assembly responses under transient and accident conditions. Finally, it may be prudent to initiate a program which includes an in-depth assessment of annular fuel pellets to make this option available if the thermal hydraulic performance and fission gas pressure impose significant limitations on the use of a modified lattice design which employs reduced diameter fuel rods.

Axial Blankets - The use of axial blankets is beset with the same fuel performance concerns as associated with extended burnup. These concerns may, however, be influenced somewhat differently by the more severe axial gradients that occur with blanketed fuel. The basic principle of axial blankets will be demonstrated in the ongoing DOE program at Rancho Seco. This program, although in the context of standard fuel management schemes, will provide valuable data to assess the performance of blanketed fuel assemblies. It is felt, however, that a similar program will be needed in conjunction with an extended burnup fuel demonstration to qualify this improvement option for implementation at higher discharge burnups.

Full Use of Early Batches - The RD&D requirements for the full use of early batches are minimal. The remaining work in the area of core fuel management design for this improvement option lies in developing explicit fuel loading patterns for specific reactors in conjunction with fuel reinsertion for five-batch, extended-burnup, low-leakage fuel management. Another area requiring development work is the effect on fuel performance of many years of storage in a spent-fuel storage pool. The major questions that need to be resolved are the effect of long storage times on clad and overall assembly corrosion and on the susceptibility of the fuel to PCI failure due to the unusual fuel duty cycle which accompanies this strategy.

End-of-Cycle Stretchout - End-of-cycle stretchout has been practiced by various utilities for a number of years. In the majority of cases, however, utilities have reserved this practice to gain flexibility in their refueling schedule rather than as a preplanned method of improving uranium utilization. Based on the already wide usage of end-of-cycle stretchout, it is felt that little, if any, research and development is required, particularly for standard power coastdown. It may be beneficial, however, to perform a systematic study of the capabilities and limitations of the feedwater-pressure augmentation mode of operation. In particular, restrictions on turbine operation should be established with a view toward implementing feedwater pressure augmentation.

3.1.2.7 Likelihood of Utility Adoption

Extended Burnup - Since the cost savings for the five-batch, extended-burnup fuel cycle are substantial, the motivation for utility adoption is high. The primary concerns which must be addressed for utility adoption are fuel performance and the licensability of extended-burnup fuel. When the issue of fuel performance has been resolved and the licensability of extended-burnup fuel has been demonstrated, utilities will most likely elect to employ this improvement option.

Modified Lattice - The incentive for adopting a modified fuel lattice design is not as great as that to adopt extended fuel burnup. Nevertheless, it is anticipated that a modified lattice design which offers better uranium utilization and lower fuel cycle costs will eventually be adopted by utilities, but only after the required RD&D programs have been successfully completed. To initiate these RD&D programs, some governmental support is considered necessary, although market pressure will also influence fuel manufacturers to shift to advanced fuel designs which show better fuel utilization and fuel economy. The DOE programs which have recently been initiated with a number of utility/fuel manufacturer teams may provide sufficient governmental support to satisfy these requirements.

Axial Blankets - Utilities will adopt the axial blanket improvement option if the impact on core thermal margins and fuel performance is not significantly

adverse. The small savings in uranium and fuel costs cannot be justified if fuel performance or core power output is compromised.

Full use of early batches - Although the savings due to the full use of early batches is relatively small, it requires very little effort in terms of RD&D and therefore is attractive from the standpoint of a utility. A number of utilities have already employed fuel reinsertion in a limited form. Perhaps, as an integrated program along with some of the other improvement options, the full use of early batches might receive more widespread usage.

End-of-Cycle Stretchout - Utilities have already adopted cycle stretchout as a means of gaining cycle length flexibility; in at least one instance, planned cycle stretchout has already been practiced. It appears that the likelihood of the utility adopting planned cycle stretchout lies closely to their management's perception of the benefits versus the risks. Reduced energy costs and lower fuel resource requirements must be weighed against the somewhat reduced flexibility on cycle length. Emphasis on the plant's operating flexibility that remains when planned cycle stretchout is practiced may be sufficient to convince utility management to adopt this uranium utilization improvement option.

3.1.3 Composite Improvement Case

3.1.3.1 Improvement Options Considered in the Composite Analysis

The improvement options which were incorporated in the composite case were extended burnup (50,650 MWd/t) with a five-batch, low-leakage fuel management scheme, a modified lattice (in the form of reduced fuel rod diameter and clad thickness), planned end-of-cycle stretchout (in the feedwater-pressure augmentation mode), the use of a 6 inch natural uranium blanket at both the top and bottom of the core, and the reinsertion of assemblies from the early batches of the initial core. All but the last two of these options were explicitly analyzed through two-dimensional fuel-management calculations. The last two of these improvement options were not explicitly included in the two-dimensional, fuel management calculations since the savings in uranium attributable to these

options were small and well-established in the individual assessments and because the composite analysis would have been complicated considerably by their inclusion. Nevertheless, a limited number of three-dimensional fuel depletion calculations were performed to verify the previous axial blanket assessment.

3.1.3.2 Fuel Management and Resource Requirements

Fuel Management Calculations - Fuel management for the composite core proved to be similar to the five-batch, low-leakage, fuel-management schemes discussed in Section 5.3. As is the case for all of these schemes, the cycle length of the initial cycle is longer than that of the equilibrium cycle while the subsequent transition cycles are somewhat shorter than the equilibrium cycle. This cycle energy distribution is required to obtain reasonable power peaking factors during the transition cycles when the feed enrichment is being increased from the relatively small value of the initial core to the relatively high value of the equilibrium cycle. Again, power sharing between the fresh fuel and the fuel with three or four cycles of exposure is a problem, resulting in relatively high core power peaking factors. Because of the short transition cycles and the need to limit the enrichment increase of the feed batch on any given cycle so as to obtain reasonable power peaking factors, a relatively long approach to equilibrium conditions occurs (on the order of 8 to 10 cycles).

Because of the high enrichments associated with the five-batch, extended-burnup fuel cycle, a relatively large number of burnable poison shim rods were required to control the power in the fresh fuel assemblies placed in the core interior. In the initial cycle, a number of different shim loading patterns using from 16 to 24 shims per assembly were used. For subsequent cycles, only 16 shims per assembly were required. The relatively high burnable poison shim loadings used in the composite design resulted in a shim residual at the end of the equilibrium cycle of about 0.8% $\Delta\rho$. Tied up in this burnable poison shim residual is the equivalent of about 1.5% in uranium utilization.

End-of-cycle stretchout was implemented in each cycle for the composite improvement case. The minimum plant output at the end of the coastdown operation ranged from 71 to 80% of the net electrical rating of the plant. The higher minimum plant outputs were used during the transition cycles to limit the

burnup increment added as a result of the stretchout operation. A near economic optimum value of 72.5% of rated output was used for the equilibrium cycle.

Three-Dimensional Core Analyses - As mentioned previously, three-dimensional core depletion calculations were performed on the initial cycle to verify results obtained in the axial blanket assessment. The core-average axial power distributions obtained from these three-dimensional analyses in general verify the results of the one-dimensional axial assessment calculations discussed in Section 5.5. The major differences between the two calculations resulted from the fact that the earlier one-dimensional study did not use feedbacks in the calculations, whereas the three-dimensional core calculations used thermal hydraulic, fuel temperature, and xenon feedbacks. Including the feedbacks in the three-dimensional calculations resulted in axially asymmetric power shapes, whereas the previous one-dimensional analyses resulted in axially symmetric power shapes. The asymmetric power shapes in the three-dimensional analyses have somewhat higher axial peaking factors than were obtained previously.

Comparisons between two-dimensional planar power distributions and axially integrated three-dimensional (3-D) power distributions show relatively good agreement in the X-Y power distribution. In addition, comparisons of axially integrated 3-D power distributions for the composite core with and without axial blankets shows excellent agreement. These comparisons indicate that two-dimensional calculations are sufficiently accurate to assess the composite core with an axial blanket provided that the axial leakages which are input into the two-dimensional calculation are consistent with those which would be obtained from a blanketed core. Therefore, two-dimensional calculations were used in all subsequent composite cycles.

Assembly Enrichment Zoning - It was noted previously that five-batch, extended-burnup fuel cycles have approximately 10 to 12% higher power peaking factors than the reference three-batch PWR. One technique to reduce this potential loss in core thermal margin is by enrichment zoning of the fuel rods within an assembly. By lowering the enrichment of the fuel rods located around the water holes and increasing the enrichment of the other fuel rods within the assembly, 5 to 8% lower peaking factors can be obtained with the zoned assembly compared to the uniform enrichment assembly.

3.1.3.3 Resource Requirements

The fuel resource utilization of the composite improved PWR is considerably better than that of the reference PWR. Averaged over a 30-year reactor lifetime, the uranium requirements of the composite improved PWR are 21% less than those of the reference PWR, while the separative work requirements are 9% less than those of the reference PWR. Accompanying this better fuel resource utilization are 40% lower spent fuel storage and disposal requirements. In addition, the composite improved PWR has about 40% less fissile plutonium in its spent fuel.

3.1.3.4 Core Characteristics and Their Impact on Safety

The core characteristic that will have the most significant impact on the safety evaluation of the composite design is the higher power peaking of this core compared to that of the reference core. This problem, however, is generic to five-batch, extended-burnup cores and is not unique to the composite core design. The other core parameters which have a potential impact on plant safety and which were expected to deviate the most from those of the reference PWR were analyzed for the composite improved case. These safety-related core parameters include the boron worth, the moderator temperature coefficient, and the fuel temperature coefficient. Based on the analysis performed for the composite case, the impact of these core parameters on the safety evaluation of the composite design is expected to be no more limiting than that of the reference core design.

3.1.3.5 Economic Assessment of the Composite Design

The 30-year levelized fuel cycle costs for the composite PWR design are 14-16% less than those of the reference core design, the specific value depending on

the replacement power costs assumed during the end-of-cycle stretchout operation. With the exception of the indirect charges resulting from the carrying charges on the fuel during its in-core residency, each phase of the fuel cycle is more cost effective for the composite design compared to the reference design.

3.1.3.6 Environmental Assessment of the Composite Design

The majority of the environmental impact of the nuclear fuel cycle is associated with uranium mining and milling, UF_6 production, and separative work. Since the composite improved PWR has 21% lower uranium requirements and 9% lower separative work requirements than the reference PWR, utilization of the composite fuel cycle will clearly reduce the environmental impact in the above-mentioned areas. Since the reload batch size for the composite design is 40% smaller than that used in the conventional PWR design, fabrication, fuel transportation, and spent fuel storage requirements would decrease by a like amount.

In light of the higher discharge burnup of the composite improved fuel cycle, two aspects of spent fuel storage can potentially affect the environmental impact of the composite design: the decay heat and the criticality of stored fuel. Neither of these potential concerns is expected to present a significant problem. However, the spent fuel storage racks will probably need to be redesigned for the possible intentional or inadvertent storage of low-burnup, high-reactivity, fresh fuel.

Other areas which potentially affect the environmental impact are associated with fuel performance and reactor operation of the composite improved PWR. Concerns over fuel performance are mainly centered on the extension of burnup from current values to values in the range of 50,000 MWd/t. These issues are currently being examined under various DOE, EPRI and industry-sponsored research and development programs. Data from these programs will be available in the early to mid-1980s. The impact of the other improvement options, i.e., low-leakage fuel management, axial blankets, modified lattice, end-of-cycle stretch-out, and the reinsertion of early core batches on the fuel performance of extended-burnup fuel appears to be minimal. No significant new problems or considerations are introduced with these other options which would result in basically different performance than that of the high-burnup option. Therefore, it appears that no significant additional impact on the environment results

from the development of the composite improved PWR in comparison to that which will result from the use of the extended-burnup option.

3.1.4 Effect of Once-Through Improvement Options on the Recycle Mode

The question has arisen as to the impact of the previously-discussed once-through improvement options on the performance of the recycle mode of operation, that is, whether the improvement in uranium utilization from each of these options also carries over to recycle, or whether the near-term incentive for recycle is diminished as a result of the implementation of these once-through improvement options. Consequently, each of the once-through uranium improvement options has been assessed in the recycle mode and the results of this assessment are summarized here.

3.1.4.1 Extended Burnup

Use of the five-batch, extended-burnup fuel cycle in the recycle mode leads to a lower conversion ratio and thus to a decrease in uranium utilization. Because of the higher fission product absorption in the extended burnup-cycle, the conversion ratio of this cycle is about 2.5% lower than that of the reference cycle. Evaluations of the five-batch, extended-burnup cycle in the recycle mode indicate that the 30-year cumulative uranium requirements are about 4% higher for this fuel cycle than for the standard fuel cycle.

Uranium utilization in the recycle mode could be improved if the discharge burnup were decreased below current values, but economic considerations limit the practicality of this approach. Reducing the discharge exposure results in the more frequent need for fuel reprocessing in mixed-oxide fabrication and since both of these fuel cycle process steps are expected to be expensive, economic considerations generally preclude such reduced discharge exposures. For this reason, evaluations indicate that 50,000 MWd/t burnup is more economic than current burnup values in the recycle mode for uranium prices up to about \$100/lb U_3O_8 .

Consequently, it can be concluded that the once-through fueling mode improvement in uranium utilization derived from extended burnup does not result in a similar improvement in the recycle mode. Furthermore, independent optimization of uranium

requirements by changing the burnup in the recycle mode does not appear attractive since this is achieved by decreasing discharge exposure which would result in economic penalties due to increased reprocessing and mixed-oxide fabrication requirements. In fact, the use of extended burnup in the recycle mode appears to be more economical than the use of standard burnups since the savings from reduced reprocessing and refabrication throughput exceeds the cost of the additional U_3O_8 prices of \$100/lb or less.

3.1.4.2 Low-Leakage Fuel Management

Evaluations of low-leakage fuel management schemes indicate that the once-through uranium utilization benefit is derived almost solely from a reduction in core leakage which impacts the end-of-cycle fissile inventory required to sustain criticality. There is a small increase in the conversion ratio as a result of the better neutron economy; it is estimated that low-leakage configurations will therefore reduce uranium requirements by about 1.2% in the recycle mode.

3.1.4.3 Lattice Modifications

The improvement of the uranium utilization in the once-through cycle attributable to lattice modifications stems from design changes which increase the water-to-fuel volume ratio. Evaluations of modified lattice geometries indicate that such increases in water-to-fuel ratio would decrease the conversion ratio of the recycle core by about 1.5 to 2%. This reduced conversion would increase uranium requirements in the recycle mode of operation by about 3 to 4%.

In principle, uranium requirements in the recycle mode can be reduced by design changes which reduce the water-to-fuel volume ratio. Although it is clear that such reductions in water-to-fuel ratio are technically feasible in the context of new reactor designs, it is doubtful whether these design changes can be backfitted into existing or committed reactors without reducing power output and thus incurring considerable cost penalties. A further deterrent to modifying the fuel lattice in the direction of lower water-to-fuel ratio is poor economics. As a result of the lower water-to-fuel ratio, core fissile material inventories will increase, compromising the fuel cycle economics through increased carrying charges. In fact, the economics of plutonium recycle presently favor lattices with higher water-to-fuel ratios than are used for current-design PWRs. At present uranium prices, the economic optimum for a PWR employing recycle is at a water-to-fuel ratio of approximately 2.3.

Consequently, it appears doubtful that reductions in water-to-fuel ratio could be retrofitted into existing reactors without reduced power output and considerable cost penalties, and further that fuel cycle economics will not favor this fueling mode for some time until uranium prices have risen significantly.

3.1.4.4 Axial Blankets

By employing axial blankets in the improved PWR design, neutron leakage from the core is decreased, fertile conversion due to captures in the blanket material is increased, and the exposure of the enriched active core fuel is increased. In the recycle mode, only the second of these effects is important from a fuel resource standpoint. The first and third effects allow the end-of-cycle fissile inventory to be lower, which is important from an economic point of view, but not from the standpoint of resource utilization. The conversion ratio of a core with six-inch natural uranium axial blankets increases only slightly above that of the unblanketed core. This small increase in conversion ratio represents an improvement of about 1.2% in uranium utilization for the recycle mode, which is only about half that obtained for the once-through fueling mode.

3.1.4.5 Full Use of Early Batches

The full use of early batches of the start-up core will not result in a reduction in U_3O_8 requirements in the recycle mode. Two factors contribute to this consequence. First, reinserting this partially-burned fuel accumulates additional exposure and therefore this once-through improvement option suffers the effect on the recycle mode of increased burnup. Secondly, inserting these early core batches in later cycles delays the recycle of bred plutonium. This delay in reaping the resource benefits of recycle increases slightly the 30-year cumulative uranium requirements; it has little, if any, impact on the net requirements.

3.1.4.6 End-of-Cycle Stretchout

In the once-through mode, there is both a fuel resource conservation incentive and an economic incentive to practice planned end-of-cycle stretchout. In the recycle mode, however, there is a negative fuel resource conservation incentive because of the reduced thermal efficiency in the generating plant during the stretchout operation. During stretchout, less electrical energy is produced

for a given consumption of fissile material. Since with recycle, 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% would occur relative to the no-stretchout case.

3.1.4.7 Effect of Composite Improvements on the Recycle Mode

The effect on the recycle mode of developing the composite improved PWR was assessed by combining the individual effects of each improvement option on the recycle mode. Several cases were considered in this evaluation: one case assumes that all once-through improvement options are included in the recycle mode, the second case assumes that those once-through options which degrade the resource utilization performance in the recycle mode are not included, and the third case assumes that only those options which are economic and therefore commercially desirable are included.

In the near term, it appears that the maximum composite improvement in uranium utilization for the recycle mode is about 2%. This is obtained when only the resource-efficient options for the recycle mode are included in the composite design. If economic considerations are included in the selection process, then the recycle mode would suffer about a 4% reduction in uranium utilization compared to the reference recycle case as a result of incorporating the economically attractive improvement options.

In the longer term, the range of potential composite improvement values for the recycle mode is larger than it is in the near term, reflecting the impact of including non-backfittable options. The non-backfittable options considered for the long term are specified in Section 7.4 and include reduced water-to-fuel ratio, longer fuel to accommodate axial blankets without reducing the active fuel region, increased frequency of refueling, movable fuel, and a variable lattice design which requires changes to the water-to-fuel ratio without a refueling shutdown. Inclusion of some of these non-backfittable design changes (those that improve the uranium utilization of the once-through cycle) increase the potential

uranium composite improvement to about 40% for the once-through mode compared to the reference once-through cycle. Including all of these non-backfittable design changes in the recycle mode increases uranium utilization in the mode by about 36% compared to the reference recycle case. These estimates are based on scoping analyses only. Furthermore, although detailed economic evaluations have not been performed for the long-term uranium utilization improvement options, evaluations performed thus far indicate that, in general, the price of U_3O_8 would have to increase substantially before there would be sufficient economic incentive to implement most of these options.

In summary, the 32% cumulative improvement (38% net improvement) due to recycle in the reference case is reduced to a 15% (22% net) advantage when near-term improvements are implemented in both the once-through and the recycle refueling modes. If all near-term, once-through improvements are included in the recycle mode, this value drops to 10% (16% net) because of the degradation in resource utilization efficiency of some of these options when recycle is employed. When only those options which are economic in the recycle mode are included, the advantage in uranium utilization of recycle over the once-through cycle is only 10% (17% net). In the longer term, U_3O_8 requirements (based on scoping calculations) for the most optimistic recycle case might be about 27% (34% net) lower than those of the most optimistic once-through fueling case.

Consequently, it is concluded that the improvements identified for near-term demonstration can obtain a substantial fraction of the uranium utilization improvement possible with recycle. These near-term improvements will reduce the difference in uranium utilization between the once-through and recycle modes from about 32% to about 10-15%. From a resource improvement standpoint, little of the R&D required to implement these near-term improvements will be directly applicable to the recycle mode. However, from an economic standpoint, much of the once-through R&D effort will benefit the recycle mode.

3.2 RECOMMENDATIONS

3.2.1 General

A relatively large number of design and operating modifications which could potentially improve the uranium utilization of operating reactors were evaluated as part of this program. The evaluation identified a number of options which have the potential for significant U_3O_8 savings, are economically sound, and appear achievable through the successful completion of prerequisite R&D programs. When employed separately they improve uranium utilization, and all could be employed simultaneously in a single reactor to achieve quite significant reductions in uranium demand. Consequently, it is recommended that the development of each of these items be continued and pursued through in-reactor demonstrations. Improvements recommended for further development are listed below:

- . Extended burnup with improved fuel management
- . Lattice modifications to increase water-to-fuel ratio
- . Axial blankets
- . Full use of early batches of the initial core
- . End-of-cycle stretchout

Each of these items is discussed in detail in Chapter 5, where development problems and RD&D programs for their resolution are presented. Key recommendations for the further development of each of these improvement options are summarized below. In certain instances, recommendations for further work have been made in areas where DOE development programs have already been initiated or contemplated; recommendations are made in these areas both for completeness and to underscore the need for these programs as evident from this assessment.

3.2.2 Extended Burnup With Improved Fuel Management

Extending the discharge exposure, coupled with modifications to the fuel management to increase the number of fuel batches resident in the core and reduce core leakage, is the single most significant method for improving uranium utilization. Because of the importance of this option, development programs to remove the significant technical impediments and uncertainties

which presently preclude commercial deployment should be vigorously pursued.

An essential prerequisite to the commercial exploitation of this option is demonstration programs to establish fuel performance at extended-discharge exposures. These programs are necessary to demonstrate that such concerns as fuel assembly dimensional and structural changes, fission gas release and internal fuel rod pressure, cladding corrosion, and pellet clad interactions do not lead to a significant increase in the probability of fuel failures for the increased fuel burnups and in-core residency times which characterize this option. Such demonstration programs have been initiated by DOE and industry groups, and should be pursued to the point where a statistically significant number of fuel rods have demonstrated successful performance. In addition, a number of research programs have been initiated to develop additional data and insight on such concerns as pellet clad interaction, fission gas release and clad corrosion. These programs should be continued and extended to include high burnup samples as they become available from irradiation demonstration programs.

The design and licensing of standard burnup fuel is supported by an extensive data base of commercial operating experience and special experiments, the latter designed to provide information on basic fuel properties and on fuel performance under atypical operational conditions, such as may occur during operational transients. It is not evident at the present time whether existing information and the results of irradiation demonstration programs will be fully sufficient to meet design and licensing data needs, or whether supplementary experimental and analytical effort will be required. Consequently, a licensing assessment would be useful, in which the data needs normally required for design and licensing are compared with the expected availability of information from ongoing test programs so as to identify areas where further experimental or analytical effort may be required. Such a licensing assessment has been initiated by DOE for pressurized water reactors; a parallel effort on boiling water reactors would appear to be warranted.

The qualitative licensing assessment performed as part of this evaluation indicated that the core characteristics and expected consequences of various anticipated occurrences and other postulated accidents would be similar to those exhibited by cores employing standard burnup and fuel management practices; nevertheless, certain differences were noted which warrant further attention. The most significant problem area appears to be the increase in power peaking which was observed in the extended-burnup, modified fuel management schemes. This increase in power peaking results from the large variation in reactivity between fresh fuel and more highly irradiated fuel batches. The power peaking observed in this study would result in unacceptable reductions in thermal over-power margin and margins-to-peak-linear-heat-rate limits (the limits applicable to the loss-of-coolant accident analysis); consequently, power peaking must be reduced before such fuel-management schemes could be commercially employed. There are several approaches which might be considered to involve commercially-acceptable fuel-management designs. These approaches are summarized below; in practice, a number of these approaches may actually be required in order to achieve acceptable peaking.

Perhaps the most straightforward method for facilitating the commercial introduction of this improvement option would be to establish somewhat less ambitious near-term goals than the 50,000 MWd/t discharge burnup, five-batch fuel management scheme assumed in this evaluation. It is expected that more reasonable power peaking could be obtained with a four-batch fuel management scheme employing a discharge exposure of about 40,000 MWd/t. While uranium utilization improvements, of course, would not be as large, such an interim target could probably be introduced commercially at an earlier date, simultaneously with experimental and fuel-management development programs to support still higher burnup. Consequently, an evaluation of a four-batch fuel management scheme having discharge exposures of approximately 40,000 MWd/t is recommended to establish whether power peaking and other fuel performance concerns are sufficiently alleviated to warrant the development of this fueling mode as an interim target for commercial deployment while the technology for still higher discharge burnups is being developed.

The power-peaking problem itself, of course, must be addressed if the larger uranium savings associated with employing five-batch fuel management schemes with 50,000 MWd/t discharge exposure (or perhaps still greater discharge exposures) are to be realized. The fuel management development performed as part of this evaluation appears to indicate that power peaking is inherently higher in extended-burnup cores and that achieving acceptable loading patterns will be inherently more difficult than for conventional fuel management schemes. However, it is recognized that relatively little effort has been devoted to developing acceptable extended-burnup fuel management as compared to the years of design experience with conventional fueling schemes. It is recommended that additional effort be devoted to extended-burnup fuel management, and it is expected that this will be at least partially successful in reducing margin penalties. The potential benefits of fuel assembly zoning should also be explored as part of the extended-burnup fuel management effort. The fuel management development performed as part of this evaluation indicates that a major cause of the increased power peaking is the large flux (and hence power) gradients between fuel assemblies of greatly varying burnup, and suggests that significant reductions in power peaking (perhaps recovering about half of the margin loss) might be obtained with zoning.

Lastly, the use of gadolinia additions to fuel as burnable poison shims should be explored. The improved fuel management assumed in this evaluation reduces core neutron leakage (and hence reduces uranium requirements) by locating fresh fuel toward the core interior and partially irradiated fuel along the core periphery. Burnable poison shims, of some type, are required in the fresh interior fuel in order to limit and control the power density in these assemblies. In the present analyses, boron shims were assumed, similar in design to those currently used in initial cores. The use of such boron shims, however, does pose certain limitations since these shims do not fully deplete during the operating cycle (i.e., they have a residual worth at end of cycle). Since a high residual worth would negate the uranium utilization benefits of the low-leakage fuel management configuration, the concentration of depletable boron in the shims must be limited so as to obtain acceptably small end-of-cycle residuals. Furthermore, the number of shims must be limited in fuel designs which employ

lattice shims (i.e., where shims replace fuel rods) to minimize the impact on the linear heat rate of the remaining rods. It should be mentioned that fuel designs that use water-hole shims are also subject to limitations like these because of the availability of shim locations (i.e., the number of water holes). Thus, these limitations decrease the flexibility of using boron shims for power distribution control. The use of gadolinia as a burnable poison in place of the boron shims might have several advantages, which derive largely from the fact that gadolinia shims will deplete almost completely during the operating cycle and from the fact that the gadolinia can be mixed homogeneously with the fuel. By eliminating shim residual, gadolinia will improve uranium utilization (the estimated improvement is 1-2%). From the standpoint of power peaking, gadolinia burnable poison shims may also prove attractive since it should be possible to employ a larger number of shimmed fuel pins for the explicit purpose of suppressing power peaking. The degree to which gadolinia burnable poison shims can recover the margin loss observed for extended-burnup fuel management schemes is difficult to foretell; this will be known only when such shims are applied in fuel-management analyses. Consequently, it is recommended that a gadolinia burnable poison shim evaluation and demonstration program be initiated as an extension of high-burnup demonstration programs. Included in such a gadolinium program should be a reload design analysis aimed at demonstrating that power margin improvements can be obtained and at establishing the required shim characteristics (gadolinia loading, number of shims, etc.) and an irradiation demonstration to establish the performance of gadolinia-bearing fuel, and to provide confirmation of analytical predictions of gadolinia burnup and the evolution of power distributions with irradiation. Both a demonstration of gadolinia-bearing fuel performance and of the adequacy of analytical models are thought necessary for the licensing of commercial reload cores employing this shim concept.

The qualitative evaluation of safety and licensing also identified a number of other parameters which are altered somewhat from that observed with conventional fuel management schemes. The most notable of these is a reduction in control rod shutdown margin which results from a reduction in control rod worth and of more negative moderator temperature coefficients of reactivity at end of cycle; a number of other parameters important to safety also change somewhat,

although to a lesser degree. The safety and licensing evaluation performed for the reference reactor design, Combustion Engineering's System 80TM plant, indicates the likelihood that these slight changes in safety-related parameters can be accommodated by the margins inherent in the design or are encompassed by conservatisms present in the safety analysis. However, detailed safety analyses will have to be performed to confirm this tentative conclusion, as well as for other designs in which extended burnup with improved fuel management is anticipated. Thus there appears to be a need to extend the scope of existing demonstration programs, which are presently aimed primarily at establishing the irradiation performance of prototypical assemblies, to encompass the detailed reload design and safety analysis for the eventual full-core use of this improvement option. An early exploration of the potential design and licensing problems associated with the end use of extended-burnup fuel will help to ensure that such design and licensing problems can be resolved in a timely fashion and facilitate commercial acceptance and introduction.

3.2.3 Lattice Modifications

The evaluation of the composite improved PWR core assumed the use of reduced diameter fuel rods so as to obtain a more thermalized neutron spectrum through an increase in water-to-fuel ratio, with a concomitant reduction in U_3O_8 requirements. Reduced rod diameter was selected over the alternative retrofittable option of annular fuel pellets since U_3O_8 savings were somewhat larger and development costs and schedules were minimized. However, there are two disadvantages to the use of reduced diameter fuel in conjunction with other methods of improving uranium utilization which are largely eliminated with the alternative of annular fuel.

First, a reduction in fuel rod diameter will reduce thermal margin because of the decrease in heat transfer area. Although this reduction in thermal margin is not large for the modest decreases in rod diameter assumed in this evaluation, it does add further to the margin loss which results from extended discharge exposure. This margin loss would not be present for the alternative of annular fuel, since rod diameter and hence heat transfer area remain unaltered. The second area of potential concern is internal rod pressure which will increase

with burnup due to increased fission gas release; fission gas release is considered to be one of the major fuel performance considerations which may limit discharge exposure. The problem of internal rod pressure becomes somewhat more severe when the fuel rod diameter is reduced, since the volume of the fission gas plenum also becomes smaller, while discharge exposure must be increased somewhat to compensate for the lower fuel loading. The problem of fission gas release and excessive internal rod pressure are significantly reduced with annular fuel since the central void region provides additional free space to accommodate fission gases, while the lower operating temperature of annular fuel will decrease the amount of fission gas release. Annular fuel may also be less prone to pellet clad interaction type failures and the reduced stored energy of such fuel, resulting from the lower operating temperatures, may also provide increased margin to peak linear heat rate limits.

It is not clear at this time whether the use of annular fuel will be required or whether concerns on thermal margin and internal rod pressure can be successfully resolved using solid pellets. However, it is recommended that fuel rods containing annular fuel pellets be included in extended-burnup demonstration programs, so that irradiation performance information will be available should the use of this fuel prove necessary or advantageous.

3.2.4 Methods to Enhance End-of-Cycle Stretchout

This assessment has indicated that significant U_3O_8 savings can be achieved through planned end-of-cycle stretchout, and further that uranium utilization and economic benefits can be enhanced through the use of feedwater-pressure augmentation. In the latter, extraction steam to the feedwater heaters is reduced, allowing the reactor to operate for a period of time at full thermal power (but at a slightly lower net electrical efficiency). After the limits of feedwater-pressure augmentation are reached, normal coastdown is initiated in which both reactor thermal and electrical output are reduced.

The benefits of end-of-cycle stretchout would be increased if feedwater-pressure augmentation could be employed to a greater extent. In this evaluation, feedwater-pressure augmentation was limited to about a 50°F reduction in

feedwater temperature, based upon turbine warranties limitations. It is anticipated that a detailed evaluation of turbine performance would indicate that feedwater-pressure augmentation could be performed for a longer period of time without exceeding turbine limits. Consequently, it is recommended that the limits on feedwater pressure augmentation imposed by turbine design be evaluated in greater detail.

There are several other possible methods of enhancing end-of-cycle stretchout which warrant further attention. In particular, attention should be given to the area of operating procedures for coastdown; by doing so it may be possible to optimize power output during stretchout. For example, it may be possible to employ feedwater pressure augmentation until turbine limits are reached, proceed with normal coastdown until these turbine limits are relieved, and then to institute further feedwater pressure augmentation. A second area is the prospect of enhancing end-of-cycle stretchout by altering core conditions during the operating cycle prior to the initiation of the stretchout mode of operation. In particular, it may be possible to increase moderator temperature slightly during periods of the operating cycle where adequate reactivity and thermal margin is available. This would increase the conversion ratio during operation and would therefore extend normal full-power operation before stretchout begins. The stretchout requirements would be less in this situation; furthermore, the stretchout operation would be more efficient because of the more negative MTC (as a result of the higher core average burnup) and because of the somewhat higher core average temperatures during the stretchout.

3.2.5 Composite Improvement Case

Each of the improvements incorporated into the composite case contribute to the alteration of safety-related core properties. The most significant of these changes appears to be the decrease in thermal overpower margin and margin-to-peak linear heat rates. This margin loss results largely from extended burnup/improved fuel management, but lattice modifications (reduced fuel rod diameter) and the use of axial blankets are also contributing factors. Other core parameters change to a smaller degree, but these changes may be significant depending upon the margins and design conservatism present in the reactor into which these improvements are to be incorporated.

In order to more definitively establish the design and licensing problems which might be encountered in fully implementing all of these improvement options in a single core, a more detailed thermal margin and safety assessment of the composite case should be performed.

REFERENCES

1. T. R. Robbins, "Preliminary Evaluation of a Variable Lattice Fuel Assembly and Reactor Design Concept," ORNL/SUB-79/13571/1, Pickard, Lowe & Garrick, Inc., February 1979.

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Section 4

SURVEY OF ONCE-THROUGH IMPROVEMENT OPTIONS

4.1 INTRODUCTION

There are a variety of changes which can be made to the core design, fuel management, and plant operation of a PWR which have the potential for improving uranium utilization. These changes are grouped together by type in Table 4.1-1 where initial estimates of their potential for reducing uranium requirements are shown. As Table 4.1-1 indicates, a rather substantial reduction in uranium requirements appears possible through the use of increased burnup. Significant further improvements appear possible by combining many of the other potential improvement types with increased burnup.

The purpose of this section is to report the results of the survey of potential improvement options. The large number of options included in this survey were examined with the intent that the most promising options would be explored in greater depth in subsequent sections of this report. The methods used in this survey were therefore more approximate than would normally be used in design calculations. They were, however, of sufficient accuracy to establish the approximate fuel resource benefits of employing the improvement options. Those options which were found to exhibit a sufficiently large benefit were then examined using standard design methods and procedures.

Table 4.1-2 contains a listing of the individual improvement options which were examined along with the estimated improvement in uranium utilization and the equivalent reduction in uranium requirements. The list of potential improvement options is fairly lengthy, and it is recognized that not all items can be employed in combination (i.e., a number of the items are mutually exclusive of one another). Furthermore, not all of the options on this list proved backfittable into existing PWR designs, while others probably could not be implemented in the near term or their implementation would necessitate an extensive RD&D program.

Table 4.1-1

INITIAL EVALUATION OF POTENTIAL IMPROVEMENTS
ONCE-THROUGH FUEL CYCLE FOR PWR

<u>Improvement Type</u>	<u>Potential for Reducing^a Uranium Requirements</u>
1. Increased Burnup	High
2. Lattice Changes (pitch, rod diameter, annular or diluted pellets)	Medium
3. Variable Lattice/Spectrum Shift (mechanical, coolant temperature)	Medium/High
4. Enrichment Zoning/Blankets (axial or radial; natural or depleted uranium)	Low/Medium
5. Full Use of Early Batches of Initial Core (including fuel exchange between reactors)	Low
6. Improved Fuel Management and Control Designs (low-leakage schemes, use of more in-core batches, substitution of fertile material for poisons, more uniform batch burnup, low cross section Zr, reduced EOC soluble boron concentration, etc.)	Medium
7. EOC Stretchout (including feedwater pressure augmentation and power coastdown)	Low
8. Reconstitution of Fuel	Medium

a. High means >10%
Medium means 2 - 10%
Low means < 2%

Table 4.1-2

Item

b. EOC electrical power endpoint

c. For each 10°F increase in coolant temperature

Many of the options shown in Table 4.1-2 have previously been examined during the development of commercial reactor designs. However, in general, the uranium resource savings associated with the various options either have not been quantified or have not been evaluated in conjunction with increased discharge burnups where, for many options, somewhat different benefits might be anticipated. Consequently, this survey was undertaken with the objective of quantifying the potential for improvements in uranium utilization for each option in conjunction with increased discharge burnup and estimating which of the various options can be combined so as to obtain an estimate of the overall improvement. In the process of performing this survey, it was possible to eliminate certain options whose potential benefits were too small to warrant the required development program or which introduced significant questions of licensability or practical feasibility.

4.2 SURVEY OF POTENTIAL URANIUM UTILIZATION IMPROVEMENT OPTIONS

4.2.1 Increased Burnup

One of the most straightforward means of improving uranium utilization is to increase the fuel's discharge burnup and/or decrease the number of fuel assemblies replaced at each refueling. Using a simple point reactor model, a study (1) was performed to derive approximate quantitative relationships among discharge burnup, batch size and refueling interval. A brief summary of the results obtained from this study is given below.

The impact of the refueling batch size on uranium utilization can be understood through the following approximate equation:

$$E_a = \frac{n+1}{2n} E_d, \quad (4-1)$$

where E_a = core-average end-of-cycle exposure

E_d = batch-average discharge exposure, and

n = number of in-core refueling batches.

As Equation (4-1) indicates, the core-average exposure is minimized by increasing the value of n for any given discharge exposure. Since the fission

product burden is proportional to the core-average exposure, smaller quantities of fissile material will be required at end of cycle to maintain criticality if the core-average burnup is reduced by increasing the number of in-core batches. Since the required end-of-cycle fissile inventory is reduced, the initial inventory can also be reduced as the number of batches is increased, thereby reducing U_3O_8 requirements.

Equation (4-1) has two limits: single-batch refueling ($n = 1$) and on-line refueling ($n = \infty$), where the core-average burnup is equal, respectively to the discharge burnup or one-half the discharge burnup. Since, for a given initial enrichment, cores employing either single-batch or on-line refueling would operate to the same core-average burnup, the discharge burnup achievable with on-line refueling would be a factor of two larger than that for the single-batch refueling option; consequently, the amount of energy which can be produced from a ton of fuel will be twice as large if on-line refueling, rather than single-batch refueling, is employed (i.e., U_3O_8 requirements are 50% of those for single-batch fuel management). Since PWRs typically employ a three-batch fuel management, their U_3O_8 requirements are about 33% lower than if single-batch refueling were employed. Thus, the three-batch fuel management now employed in most PWRs achieves part of the gain in U_3O_8 utilization which can be obtained as the number of batches is increased to approach that of on-line refueling.

Figure 4.2-1 illustrates the impact of discharge burnup, cycle length and the number of refueling batches on uranium utilization, which is defined to be the number of megawatt-days produced per short ton of U_3O_8 . At present, most PWRs are refueled annually, use three-batch ($n = 3$) fuel management, and have a batch-average discharge exposure of $\sim 30,000$ MWd/t; this set of conditions corresponds to the circled point at the lower left-hand corner of Figure 4.2-1. If the batch-average discharge exposure could be increased, the reactor operator may choose to maintain three-batch fuel management and extend the cycle length, which would correspond to Path A shown on Figure 4.2-1. For example, if the batch-average discharge burnup were increased 50% from 30,000 to 45,000 MWd/t, the operator could increase the cycle length from 12 to 18 months and concurrently could improve uranium utilization by 4%. This alternative may be appealing to many utilities who want to reduce

Figure 4.2-1
URANIUM UTILIZATION AS FUNCTION OF ENRICHMENT
AND REFUELING INTERVAL

the number of refueling outages and thereby improve plant capacity factor. However, as illustrated by the trajectory marked "B" on Figure 4.2-1, the improvement in uranium utilization is significantly greater if an annual refueling interval is maintained and n , the number of refueling batches, is increased. For example, if an annual refueling interval is maintained and the discharge burnup is increased to 50,000 MWd/t, an 18% improvement in uranium utilization* is realized by increasing n from three to five. Even greater improvements in uranium utilization would be possible if the discharge exposure and number of batches were increased further; however, Figure 4.2-1 indicates that the rate of improvement is beginning to decrease.

As shown in Figure 4.2-1, the improvement in resource utilization which results from the use of a five-batch fuel management scheme having a discharge burnup of 50,000 MWd/t results partially from the increase in discharge exposure from 30,000 to 50,000 MWd/t, but primarily from the increase in the number of refueling batches from three to five. The improvements in uranium resource utilization realized by employing five-batch fuel management while retaining the 30,000 MWd/t discharge exposure are only slightly smaller. However, as Figure 4.2-1 also indicates, increasing the number of refueling batches while holding discharge burnup constant results in a decrease in the interval between refuelings, and hence a decrease in plant availability because of more frequent refueling outages. Thus, extending discharge burnup serves two functions: first, increasing the discharge exposure from 30,000 to 50,000 MWd/t results in a modest improvement in uranium utilization, about 5%, due to the higher burnup itself; and second, extending the burnup of discharge fuel allows the number of refueling batches to be increased while preserving an annual refueling interval, and it is this increase in the number of refueling batches from which the majority of the improvement in uranium resource utilization is realized. The benefits of more frequent refuelings (a six-month cycle length), which allows a larger number of refueling batches to be employed for a given discharge burnup, is discussed in Section 4.2.17.

* An 18% increase in uranium utilization (MWd/STU₃₀₈) corresponds to a 15% reduction in uranium requirements (STU₃₀₈).

To verify the above estimate of uranium utilization improvement and to uncover any unknown problems which may be associated with five-batch, extended-burnup fuel management, two-dimensional core depletion calculations were performed. The results of these spatial calculations are presented in Section 5.3 along with the methods and procedures used.

4.2.2 Enrichment Zoning of Fuel Rods Within An Assembly

Enrichment zoning of fuel rods within an assembly was investigated to determine the potential for reducing the peak power and peak burnup. A peak power reduction has the potential for improving core operating margins, while a peak burnup reduction has the potential for improving uranium utilization by reducing the spread between limiting fuel rods and the remaining population of discharge rods.

The fuel assembly geometry used in these calculations is shown in Figure 4.2-2 along with the chosen enrichments. Depletion calculations were performed using an integral transport theory code (2) for this assembly and the reference assembly containing uniformly enriched fuel (with an average enrichment of 4.2 wt%). This enrichment was chosen from Figure 4.2-1 from the intersection of the $n = 5$ and annual refueling curves. The fuel discharge exposure at this intersection point is 50,650 MWd/t. Results show virtually no reactivity differences during the depletion of this assembly compared to the reference assembly, which implies that no direct increase in uranium utilization is obtained by enrichment zoning.

Figure 4.2-3 compares fuel assembly power peaking factors for the reference unzoned case and for the enrichment zoned case having a high-to-average enrichment ratio of 1.075. This figure shows the evolution of the relative power density in the highest power fuel rod during irradiation. For this enrichment-zoned assembly, the maximum power peak shifts midway during the irradiation away from a pin next to the water hole to an assembly corner location. A further reduction in the assembly peaking factor can be obtained by increasing the peak-to-average enrichment split to 9% as shown in Figure 4.2-4.

Figure 4.2-2
ENRICHMENT ZONED FUEL ASSEMBLY

Figure 4.2-3
ASSEMBLY PEAKING FACTORS OF AN ENRICHMENT ZONED ASSEMBLY
COMPARED TO THOSE OF REFERENCE UNZONED ASSEMBLY

Figure 4.2-4
FUEL PIN POWER VERSUS ENRICHMENT SPLIT *

Additional increase in the enrichment split would simply increase the peaking factor in the assembly corner pin which then becomes more limiting. Beginning-of-life assembly power distributions for the uniform enrichment and split enrichment cases are shown in Figures 4.2-5 and 4.2-6, respectively.

The reduction in the assembly peaking factor may improve the core operating margins. The margin improvement for the whole core, however, may be less than the 6% indicated in Figure 4.2-4 because fresh assemblies are often not limiting and reductions in the peak-to-average assembly power do not usually translate directly into a margin improvement for realistic fuel management calculations. Therefore, the potential for increased operating margins exists, but detailed core calculations are necessary to establish the magnitude of these increases.

The burnup census at discharge is shown in Figures 4.2-7 and 4.2-8 for the reference unzoned fuel assembly and the zoned fuel assembly, respectively. The peak-to-average burnup is lowered from 1.09 to 1.05 by enrichment zoning. This flatter burnup census has the potential of improving the uranium utilization if the maximum rod burnup is limiting, since the flatter burnup distribution allows more fuel pins to be discharged with higher burnups. This improvement in uranium utilization is, however, less than 0.1%. If the maximum pin burnup is not limiting, the flatter burnup distribution simply increases the fuel performance margins without increasing the uranium utilization.

It might be noted that the relatively small uranium utilization improvement estimated for this option is due partially to the rather uniform power distribution within the PWR fuel assembly, and partially to the extended discharge exposure. With respect to the latter, power peaking decreases with burnup (as is evident in Figure 4.2-3) so that the burnup distribution becomes progressively more uniform as discharge exposure is increased. Thus, there is less of a uranium utilization incentive for enrichment zoning in conjunction with extended-burnup fuel management.

4.2.3 Pellet Diameter Reduction in the Fuel Rods Around the Control Rod Guide Tubes

Decreasing the diameter of the fuel pellets in fuel rods situated next to the water-filled control rod guide tubes has the potential for improving

Figure 4.2-5
POWER DISTRIBUTION WITHIN THE FUEL ASSEMBLY AT BOC
(NO ENRICHMENT ZONING)

						34 1.021	35 1.097
					31 1.039	32 0.966	33 0.962
					28 1.136	29 0.969	30 0.934
					24 1.135	25 0.964	26 0.925
		16 1.051	17 1.148	18 1.145	19 1.037	20 0.946	21 0.922
	9 0.978	10 0.980	11 0.995	12 0.991	13 0.966	14 0.943	15 0.932
1 1.031	2 1.000	3 0.987	4 0.986	5 0.982	6 0.973	7 0.963	8 0.958

FUEL CELL

CEA WATERHOLE

Figure 4.2-6
POWER DISTRIBUTION WITHIN THE FUEL ASSEMBLY AT BOC,
WITH PEAK TO AVERAGE ENRICHMENT OF 7.5%

						34 0.982	35 1.052
					31 0.999	32 0.930	33 0.921
					28 1.087	29 0.926	30 1.002
					24 1.082	25 0.916	26 0.982
		16 0.991	17 1.090	18 1.088	19 0.982	20 0.889	21 0.971
	9 1.023	10 1.036	11 0.937	12 0.934	13 1.024	14 0.990	15 0.971
1 1.070	2 1.040	3 1.032	4 1.038	5 1.024	6 1.017	7 1.002	8 0.994

FUEL CELL

CEA WATERHOLE

Figure 4.2-7

BURNUP DISTRIBUTION AT DISCHARGE FOR AN ASSEMBLY WHICH HAS ACCUMULATED 50,000 MWd/t
(NO ENRICHMENT ZONING)

Figure 4.2-8

BURNUP DISTRIBUTION DISCHARGE FOR AN ASSEMBLY WHICH HAS ACCUMULATED 50,000 MWd/t (ENRICHMENT ZONING)

uranium utilization by increasing the assembly-effective water-to-fuel (W/F) ratio, increasing the assembly-average burnup and flattening the burnup distribution across the assembly, thereby allowing more fuel pins to be discharged with higher burnups when the maximum rod burnup is limiting. Increasing the W/F ratio results in a reactivity increase (as discussed in Section 4.2.4 and 5.4) and hence a uranium savings through a reduction in the required fissile material for a given energy extraction. Similarly, increasing the burnup improves uranium utilization as discussed in Section 4.2.1. If the clad thickness is increased to maintain the same pellet-clad gap and clad outside diameter, then this potential savings would be offset somewhat.

The fuel assembly geometry used in these calculations is shown in Figure 4.2-9. Calculations were performed using a multipgroup integral transport code (2). The 40 fuel pins around the water holes are of reduced pellet diameter and increased clad thickness to maintain the same gap and clad OD reference fuel rod. The fuel enrichment is maintained at the reference value of 4.2 wt%.

Power peaking results are shown in Figure 4.2-10 for the reference case and two cases having smaller pellet diameters. It can be seen that reducing the pellet in the 40 fuel pins around the water holes reduces the power peaking in these pins, but it increases the power peaking in the fuel pins on the water hole diagonals. The maximum peaking factor reduction is about 7%.

The reactivity change due to the reduced fuel pellet diameter is shown in Figure 4.2-11 for an assembly with and without increased clad thickness. This figure shows that reducing only the pellet diameter without increasing the clad thickness has only a very small effect on the beginning-of-life (BOL) reactivity. The reason for this apparent lack of sensitivity to W/F ratio changes is due to the fact that the neutron spectrum in the fuel pins around the water holes is already fully thermalized. If the W/F ratio change were distributed uniformly throughout the assembly, the reactivity change would be about 0.15% $\Delta\rho$.

For an equal output, the burnup of the assembly containing thin fuel pellets is somewhat higher than the reference assembly due to the decreased loading.

Figure 4.2-9
GEOMETRY OF THE "THIN PELLET" CALCULATION

THIN PELLET LOCATIONS
 CEA WATERHOLE
 INSTRUMENT

	REFERENCE PIN	"THIN PELLET" PIN
PELLET OD	0.325 INCHES	0.3087 INCHES
CLAD THICKNESS	25 MILS	33.1 MILS
CLAD OD	0.382 INCHES	0.382 INCHES

Figure 4.2-10
ASSEMBLY PEAKING FACTOR COMPARISONS
FOR THIN FUEL PELLETS DISTRIBUTED
AROUND WATER HOLES

Figure 4.2-11

REACTIVITY DIFFERENCE BETWEEN AN ASSEMBLY HAVING 40 FUEL PINS AROUND THE WATERHOLES
WITH A THIN PELLET AND THICK CLAD AND AN ASSEMBLY WITH A UNIFORM PELLET AND CLAD DIMENSIONS

The decrease in reactivity with burnup relative to the reference assembly is due to this fact and, for the case of thicker clad, due to the added absorptions in zircaloy. Increasing the clad thickness to maintain the same clad/pellet gap dimension results in about a 0.2% reduction in uranium utilization. The reduced uranium loading resulting from the use of smaller diameter pellets around the water holes more than offsets the added absorptions in the thicker clad so that the net effect is a slight increase in the uranium utilization of about 0.3%.

The BOL assembly power distribution is shown in Figure 4.2-12 which may be compared to the reference case with uniform pellet diameters (Figure 4.2-5).

The burnup census distribution presented in Figure 4.2-13 shows that the thin fuel pins are discharged at higher burnups relative to the rest of the fuel pin population even though the peaking factors are reduced when compared to a case with uniform fuel pellet diameters (Figure 4.2-7). This increased burnup is due to the volume reduction of the thin pellet (10%) which is larger than the peaking factor reduction (~6%).

The conclusion of this analysis is that a small uranium utilization benefit is achievable about 0.3%) by reducing the pellet diameter of fuel pins around water holes. This benefit could be increased to about 0.5% if it were possible to achieve this pellet diameter reduction without increasing the clad thickness. The result of such a lattice design change would be either a severe reduction of the permissible maximum linear heat rate from LOCA considerations (by increases in the gap thickness), or a fuel assembly design with multiple rod outside diameters. Neither of these changes is acceptable in view of the small additional uranium utilization gain.

4.2.4 More Reactive Lattice Geometries

Modifications to the water-to-fuel (W/F) ratio affect both the conversion ratio and the end-of-cycle (EOC) fissile inventory requirements of a reactor by altering the neutron spectrum. Typically, the conversion ratio can be increased by reducing the water-to-fuel ratio, since the less thermalized neutron spectrum will result in increased conversion of the fertile material (U-238) into fissile plutonium, a fraction of which will be consumed in situ

Figure 4.2-12
POWER DISTRIBUTION AT BOC FOR AN ASSEMBLY
CONTAINING FUEL PINS AROUND WATERHOLES
WHICH HAVE THIN FUEL PELLETS

						34 1.051	35 1.031
					31 1.070	32 0.986	33 0.982
					28 1.067	29 0.989	30 0.948
					24 1.065	25 0.983	26 0.937
		16 1.078	17 1.076	18 1.074	19 1.064	20 0.959	21 0.931
	9 0.988	10 0.993	11 1.013	12 1.009	13 0.979	14 0.952	15 0.939
1 1.037	2 1.007	3 0.996	4 0.997	5 0.993	6 0.982	7 0.970	8 0.964

FUEL CELL

CEA WATERHOLE

THIN PELLETS DIMENSIONS
PELLET OD = 0.3087 INCHES
CLAD = 33.1 MILS

Figure 4.2-13

BURNUP DISTRIBUTION IN A FUEL ASSEMBLY WHICH HAS ACCUMULATED 50,000 MWD/T BURNUP,
WITH 40 FUEL PINS AROUND WATER HOLES CONTAINING THIN FUEL PELLETS
(0.325 → 0.3087" OD) AND THICKER CLAD (25 → 33.1 MILS)

to produce energy. This improvement in conversion ratio is generally accompanied by an increase in end-of-cycle inventory requirements, because of the reduced cross sections of fissile material in the harder neutron spectrum relative to the U-238 capture cross section. On the other hand, end-of-cycle fissile inventories are typically reduced in a well-thermalized neutron spectrum which can be attained by providing a higher water-to-fuel ratio. Accompanying such a reduction in end-of-cycle inventory is a reduction in the conversion ratio.

Water-to-fuel ratio optimization involves determining the effect of these two competing factors: conversion ratio (i.e., the quantity of fissile material consumed during the operating cycle) and end-of-cycle inventory. For low water-to-fuel ratios, the high end-of-cycle inventory will increase uranium requirements, although the conversion ratio may be quite high. For high water-to-fuel ratios, end-of-cycle inventory requirements are minimized, but the lower conversion ratio may result in increased uranium requirements. Between these extremes lies a value of water-to-fuel ratio at which conversion and end-of-cycle inventory effects combine so as to minimize enrichment requirements.

Modifications to the water-to-fuel ratio can be achieved by a number of methods:

1. Changing the amount of water present in the lattice by varying the spacing between the fuel rods (changing the lattice pitch).
2. Reducing the amount of fuel present in the core through decreasing the effective density of the fuel, and
3. Modifying both the amount of water and fuel loading as would occur when the fuel rod diameter is changed or when both lattice pitch and fuel rod diameters are altered.

Changes in the water-to-fuel ratio achieved by varying lattice pitch are straightforward, but are generally only applicable to new reactor designs since the change in overall fuel assembly dimension will generally preclude backfitting. Decreasing the effective fuel density has the advantage of limiting design changes to fuel pellet design. This modification could be

readily backfit into existing plants since neither the assembly geometry nor thermal hydraulic performance of the reactor is altered. Reductions in the effective density of the fuel could be achieved in a number of fashions: by the use of fuel with an inherently low density, by the introduction of low-cross-section diluent materials, or by the use of annular fuel pellets (fuel with a hollow central region). The disadvantage of this alternative is the concomitant necessity of developing and demonstrating a new fuel design. Modifying the water-to-fuel ratio by changing fuel rod diameter is a second option which could potentially be backfitted into existing plants since the overall dimensions of the fuel assembly remain unchanged; however, care must be exercised to ensure that thermal margins are not degraded as a result of a change in fuel rod diameter.

The use of any of the above methods to increase the W/F ratio, however, will likely lead to a more positive moderator temperature coefficient (MTC) which must be analyzed before the practicality of any of these methods can be evaluated. The positive increase in the MTC arises primarily because of a more thermal spectrum which increases the worth of the soluble poisons in the moderator and reduces changes in neutron thermalization due to changes in temperature. The change in MTC for a given change in W/F ratio depends to some extent, on the method which is employed to accomplish the W/F ratio change (i.e., changes in pitch, rod diameter, UO_2 density, etc.). However, to a first approximation, this dependence can be ignored.

Preliminary calculations performed for the equilibrium cycle of a current-design pressurized water reactor employing a five-batch, extended-burnup fuel management scheme indicate that a change of about 0.6-0.8 in W/F ratio will result in an increase in the MTC by $1.0 \times 10^{-4} \Delta\rho/^{\circ}\text{F}$ ($1.8 \times 10^{-4} \Delta\rho/^{\circ}\text{C}$). Since the beginning-of-cycle (BOC) MTC of a five-batch, extended-burnup core is about $1.0 \times 10^{-4} \Delta\rho/^{\circ}\text{F}$ ($1.8 \times 10^{-4} \Delta\rho/^{\circ}\text{C}$) more negative than that of a core employing conventional fuel management, the above given change in W/F ratio represents an upper limit on the amount which the W/F ratio can be increased and still maintain an acceptable MTC. This change in W/F ratio is an upper limit because during the early transition cycles, while the core-average burnup is still low, the MTC is less negative than that of the equilibrium cycle. Therefore, a change in W/F ratio less than the 0.6-0.8 value would

be the maximum allowable from MTC considerations. Changes in W/F ratio are also limited by thermal-hydraulic considerations and the need to maintain sufficient thermal margins. Thus, an acceptable change in W/F ratio somewhat less than the 0.6-0.8 value would be expected. To better define the range of acceptable values requires more detailed analyses; these have been performed as reported in Section 5.4.

The uranium utilization analysis of this section was performed with a point reactor model using the CEPAC (3) cross section generating code.

The benefit of increasing the W/F ratio is illustrated in Figure 4.2-14 for several types of lattice design changes. Curve A of this figure shows the benefit resulting from reducing the UO_2 density in the fuel (or alternately using annular fuel) while retaining the same U-235 loading. A 3.0% improvement in the uranium utilization results if the UO_2 density is reduced from 10.054 to 7.9 g/cc yielding a W/F ratio of 2.6. The constant fissile loading increases the U-235 enrichment from 4.2 to 5.6 wt%. The burnup increase concomitant with this density reduction is quite large, increasing the batch average from 50,000 to about 70,000 MWd/t. This large burnup results from the reduced heavy metal loading in the fuel rod (25% reduction) and a 4.3% increase in the achievable energy. The 3.0% improvement in uranium utilization is then due to the 4.3% increase in achievable energy less the 1.3% due to the greater losses in the diffusion plant tails as a result of using higher feed enrichments. The uranium utilization benefit obtained by UO_2 density reductions is lower than that obtained by other methods of increasing the W/F ratio (to be discussed below), such as changing the lattice pitch or changing the rod diameter. One reason is that although captures in U-238 decrease as a result of lowering the U-238 concentration, the resonances of U-238 become less self-shielded at the same time, thereby partially compensating for the lower concentration. Another reason for this lower benefit is that higher enrichments are required to maintain the constant fissile loading. More uranium is therefore lost in the diffusion plant tails stream.

Figure 4.2-15 shows the uranium utilization variation due to UO_2 density changes for a smaller range of W/F ratio, i.e., for the approximate practical

Figure 4.2-14

CHANGE IN URANIUM UTILIZATION FOR VARIOUS CHANGES IN LATTICE
PARAMETERS WHICH INCREASE THE WATER - TO - FUEL RATIO

Figure 4.2-15
CHANGE IN URANIUM UTILIZATION OBTAINED BY UO_2 DENSITY REDUCTIONS

range constrained by MTC, and fuel design and performance considerations. This figure shows the increase in uranium utilization for various constant U-235 loadings in the fuel pin (solid lines). The change in uranium utilization at a constant batch-average discharge burnup is indicated by the dashed line in the figure. Figure 4.2-16 shows the change in achievable energy as a function of UO₂ density and U-235 content.

The relationship between uranium utilization (U) and achievable energy (E) may be expressed as follows:

$$\frac{U}{U_0} = \left(\frac{E}{E_0} \right) \left(\frac{L_0}{L} \right) \left(\frac{\epsilon_0 - \epsilon_t}{\epsilon - \epsilon_t} \right), \quad (4-2)$$

where L is the fuel loading,
 ϵ is the U-235 enrichment,
 and the subscripts o and t denote the reference value and the tails assay, respectively.

Once the maximum allowable value of W/F ratio is established, then Figure 4.2-15 can be used to estimate the maximum benefit that can be achieved by changing the fuel density.

Curve B of Figure 4.2-14 shows the change in uranium utilization obtained by increasing the lattice pitch. A maximum benefit of 6.5% is indicated in this figure; however, if the change in the W/F ratio is limited as previously discussed, then the achievable benefit would be about half this value. To maintain the same overall core power when the lattice pitch is increased requires increasing the core volume. Alternatively, the pitch could be increased by removing rods from the 16 x 16 assembly (and make it a 14 x 14 for example). The change in the pitch required to accomplish this is large (from 1.286 cm to 1.470 cm for a 14 x 14 core). This change is much larger than that which can practically be made.

Neither an increase in core volume nor the removal of fuel pins is a backfittable method for increasing the pitch by the desired amount; hence, changing the pitch is not a practical method of increasing the uranium utilization in operating plants.

Figure 4.2-16
CHANGE IN THE ACHIEVABLE ENERGY OBTAINED BY UO_2 DENSITY REDUCTIONS

Curve C of Figure 4.2-14 shows the uranium utilization benefit obtained by reducing the pellet and clad outside diameter while maintaining a constant clad and gap thickness. A maximum benefit of about 5.0% is calculated (at a W/F ratio of 2.8, which corresponds to a fuel pellet OD of 0.749 cm); however, if a maximum W/F ratio change is imposed based on moderator temperature coefficients and thermal-hydraulic constraints, the achievable benefit will be on the order of 3.0-4.0%. The latter benefit is obtained by reducing the pellet diameter from 0.826 cm to 0.787 cm. An additional benefit in terms of uranium utilization can be obtained by reducing the clad thickness. For example, maintaining the pellet and gap dimensions constant while reducing the clad thickness by 0.127 mm (from 0.635 mm to 0.508 mm) results in a change in the W/F ratio of 0.07. In addition to a benefit from softening the neutron spectrum by increasing the W/F ratio; reducing the fuel clad thickness also reduces the parasitic neutron absorption in cladding material. These two mechanisms taken together would result in a uranium utilization improvement on the order of 2.0-2.5% for the 0.127 mm thickness change mentioned above.

The results of further analyses in the area of more reactive lattice geometries is given in Section 5.4. Here many of the practical considerations involved with modifying the standard lattice design are discussed. Such issues as the impact on core physics characteristics and the impact on fuel performance of making certain lattice modifications are discussed in the context of extended burnup fuel cycles.

4.2.5 Duplex Pellet

The use of a zoned pellet, which has a central core of natural uranium surrounded by an annulus of enriched fuel, has the potential for improving uranium utilization by reducing the average fuel temperature; a lower temperature increases lattice reactivity by reducing absorption in the U-238 resonances.

Calculations were performed with the DIT integral transport theory code (2) using a unit cell representation of the core. The duplex pellet consists of two concentric fuel regions such that the inner region represents 25% of the

fuel pellet volume. The inner region consists of natural uranium and the outer region of uranium enriched to 4.16 wt%; the volume-average enrichment is 3.3 wt%. For comparison purposes, a uniformly-enriched (3.3 wt%) "reference pellet" was also analyzed.

The duplex pellet and the reference pellet were depleted to an exposure of about 30,000 MWd/t. The variations in the relative power densities of the inner and outer regions of the duplex pellet and the reference pellet are shown in Figure 4.2-17. This figure shows the power reduction in the central pellet region due to the low enrichment. The relative power in this region increases with burnup due to a reduction in the relative fissile concentration difference between the inner and the outer fuel regions. The lower power production in the inner region of the reference fuel pellet is also shown in Figure 4.2-17.

The impact of these power distribution changes on fuel temperature is illustrated in Figure 4.2-18. The temperature distribution of a reference pellet is compared with a duplex pellet at BOL and discharge, respectively. The temperatures shown in Figure 4.2-18 are based on a core-average linear heat rate of 5.35 kw/ft, and assumed region-wise uniform power distributions. In addition, pellet and clad are assumed to be in contact, so that the gap conductance is the same value for all cases ($11.36 \text{ kw/m}^2 \cdot ^\circ\text{k}$). These assumptions eliminate any difference between BOL and EOL for the reference pellet. Differences in temperature for the duplex pellet are then due only to the differences in the region-wise power level split.

The EOC temperature distribution of interest for the duplex pellet for life-time determination lies somewhere between the BOL and the EOL curves shown in Figure 4.2-18. A 17°C to 22°C reduction in temperature is then expected,* which increases the EOC reactivity by about 0.04% $\Delta\rho$ and increases the uranium utilization by about 0.1%. Such a utilization improvement is small and does not compensate for the added separative work costs and the added manufacturing cost associated with the use of duplex pellets. There are, however, other reasons for changing to the duplex design, such as LOCA or

* It is noted that the small temperature difference between the standard and the duplex pellet is representative of the core average fuel temperature. The difference in the centerline temperature of the peak power rod is much larger.

Figure 4.2-17
RELATIVE POWER DENSITY IN THE CENTER AND PERIPHERY REGIONS OF A DUPLEX PELLET
AND A UNIFORMLY ENRICHED PELLET

Figure 4.2-18

TEMPERATURE DISTRIBUTION IN A SYSTEM 80 FUEL PELLET OF A CORE AVERAGE
LINEAR HEAT RATE. (DUPLEX PELLET COMPARED TO A REFERENCE
PELLET OF UNIFORM ENRICHMENT)

(ALL CASES ASSUME THE SAME PELLET SURFACE
SURFACE TEMPERATURE AND UNIFORM
POWER DENSITIES TO EACH ENRICHMENT ZONE)

fuel performance behaviors, which may well be worthwhile. If the costs of changing to a duplex pellet are justifiable based on these other considerations, the small additional benefit in the uranium utilization is then simply an added bonus.

4.2.5 Low-Leakage Fuel Management

In conventional three-batch fuel management schemes, initial cores are arranged with the highest enrichment batch loaded on the core periphery, while the batches having the low and intermediate enrichments are mixed in a checkerboard pattern in the core interior. When the core is refueled, the low-enrichment batch is discharged, the high-enrichment batch is moved to the core interior and mixed with the intermediate-enrichment batch, and the fresh batch is loaded on the core periphery. Because of the movement of the high-enrichment fuel from the core periphery to the interior, this type of fuel management is called an OUT-IN strategy.

OUT-IN fuel management was developed in order to achieve flat radial power distributions in large, multi-batch chemically-shimmed PWR cores. However, when the radial power distribution is flattened by placing the high-enrichment fuel on the core periphery, the net radial leakage of neutrons from the core is increased relative to a fuel assembly arrangement in which lower enrichment fuel is placed on the periphery. Since higher leakages require higher enrichments to sustain criticality, use of OUT-IN fuel management to minimize peaking factors increases uranium and separative work requirements.

Fuel-management studies have been performed which demonstrate that the fuel loading configuration can be modified so as to reduce the radial leakage without degrading peaking factor margins. For a three-batch reload core, this is achieved by placing once-burned assemblies in most of the peripheral locations; most of the fresh assemblies are mixed with the remaining once-burned and twice-burned assemblies in the core interior. Power peaks in the fresh assemblies located near the core interior are controlled with burnable poison rods which replace about 2% of the fuel rods in these assemblies.

Fuel management studies performed for several core sizes and cycle length show that the low-leakage scheme can be implemented in the first cycle or in any subsequent cycle. These studies indicate that use of the low-leakage scheme in a three-batch fuel management scheme results in a 2-4% improvement in uranium utilization relative to the reference three-batch OUT-IN fuel management. It was originally thought that this improvement might be larger for the five-batch fuel management scheme because of the increased leakage of a five-batch OUT-IN fuel management scheme compared to a three-batch OUT-IN fuel management scheme. This, however, did not prove to be true when detailed calculations of the five-batch, low-leakage scheme were performed because of the relatively high shim residual at end of cycle. In the five-batch, extended-burnup, low-leakage scheme, much higher shim concentrations are required to control the power of the fresh fuel loaded initially on the core interior than are required in three-batch, standard-burnup, low-leakage schemes. This is a consequence of the high enrichment (~ 4.3 wt%) of the five-batch scheme. The exposure received by the fresh fuel during its first cycle of irradiation is not sufficient to exhaust all of the burnable poison. A shim residual worth about $-0.7\% \Delta\rho$, which is two to three times that obtained in three-batch, low-leakage schemes, remains at end of cycle. This amount of reactivity is equivalent to approximately a 1.5% increase in uranium utilization if fully recovered. Section 5.3.3 gives the details of these calculations.

4.2.7 Assembly Reinsertion After Removing High-Burnup Rods

An investigation has been conducted to determine the potential for improving uranium utilization by removing the high-burnup rods from an assembly prior to its last irradiation cycle; water-filled cladding tubes are inserted into the vacated lattice positions. Removal of the high-burnup rods reduces the fission product burden and, since the reference fuel lattice is somewhat undermoderated, the increase in the water-to-fuel ratio increases the reactivity of the assembly.

Calculations were performed with an integral transport theory code (2) in which a 16 x 16 fuel assembly is modeled assuming one-eighth assembly

symmetry. The assembly was depleted for the equivalent of four exposure cycles after which (prior to the fifth exposure cycle) some of the 40 rods which surround water holes were removed and were replaced by rods containing water. These water-filled rods were placed in the fuel lattice away from the water holes in order to avoid large power peaks and to obtain the reactivity benefit from increased thermalization. This scheme, therefore, utilizes a fuel-pin-shuffle scheme within assemblies. Since the high-burnup fuel pins are normally located next to water holes, a simple removal of high-burnup fuel pins without a pin-shuffle scheme results in larger water holes with unacceptable power-peaking increases. .

Figures 4.2-19 through 4.2-21 show the power distributions at the beginning of the residence cycle for various numbers of fuel rods replaced by rods containing water. A corresponding power distribution at the end of the fourth cycle without removal of any rods is shown in Figure 4.2-22. It can be seen that the power peak increases when fuel rods are removed from the assembly. The peak tends to occur in a low-burnup rod which is placed next to a water hole. However, the magnitudes of the peaks are not as high as the BOL peak of 1.15. Furthermore, the power in these assemblies would be much lower than in other assemblies which have accumulated less burnup, so that the assembly power peaks are expected to be not limiting.

The change in the assembly reactivity during the last residence cycle is shown in Figure 4.2-23. This figure shows that the removal of fuel rods leads to an increased reactivity at BOC, but the lower conversion ratio which results from softening the neutron spectrum reduces the benefit at the end of cycle. This lower conversion ratio is illustrated in Figure 4.2-24 which shows the assembly average PU-239 concentration versus burnup. The figure shows that the Pu-239 tends toward a lower equilibrium concentration as the neutron spectrum is "softened" by the removal of fuel rods.

The uranium utilization results are actually even less attractive than indicated by Figure 4.2-24 because the remaining fuel rods must be more highly burned to achieve the same energy extraction as the reference fuel assembly. This effect is illustrated in Figure 4.2-25 which shows that, on an equal energy basis, the benefit disappears midway through the last cycle of residence. It is concluded, therefore, that the procedure discussed above will not yield

Figure 4.2-19
POWER DISTRIBUTION AT THE BEGINNING OF THE LAST RESIDENCE
CYCLE, WITH 8 PINS CONTAINING WATER

FUEL CELL

WATER FILLED
FUEL ROD

CEA WATERHOLE

PEAK ROD 1.105

Figure 4.2-20
POWER DISTRIBUTION AT THE BEGINNING OF THE LAST RESIDENCE
CYCLE, WITH 16 PINS CONTAINING WATER

FUEL CELL

WATER FILLED
FUEL ROD

CEA WATERHOLE

PEAK ROD 1.089

Figure 4.2-21
POWER DISTRIBUTION AT THE BEGINNING OF THE LAST RESIDENCE
CYCLE, WITH 24 PINS CONTAINING WATER

FUEL CELL

WATER FILLED
FUEL ROD

CEA WATERHOLE

PEAK ROD 1.103

Figure 4.2-22
ASSEMBLY POWER DISTRIBUTION AT THE BEGINNING OF
THE LAST RESIDENCE CYCLE, WITHOUT REMOVING ANY
FUEL PINS

FUEL CELL

PEAK ROD 1.060

CEA WATERHOLE

Figure 4.2-23
 REACTIVITY CHANGE DUE TO THE REMOVAL OF SEVERAL HIGH BURNUP RODS
 DURING THE FIFTH RESIDENCE CYCLE

Figure 4.2-24
 ASSEMBLY AVERAGE Pu-239 CONCENTRATION VERSUS BURNUP *

Figure 4.2-25
EIGENVALUE VARIATION DURING AN ASSEMBLY DEPLETION CALCULATION

any uranium utilization benefit. A possible alternate procedure would be to remove the fuel pins at discharge to increase reactivity and to operate the reactor for a short extra cycle after each refueling. The extra cycle could increase the fuel utilization by about 1%; however, since the number of refueling outages would be doubled, this procedure is not practical.

4.2.8 Changes in the Thickness of the Shroud and Peripheral Water Gap

The purpose of the shroud is to minimize lateral motion of the fuel assemblies. For a C-E reactor, the shroud consists of steel plates approximately 1.0 in. thick which are welded together such that their faces are adjacent to the faces of peripheral fuel assemblies. Variation in the thickness of the core shroud and the water gap between the core and shroud were investigated to determine if such a design modification would increase reactivity and hence improve uranium utilization. In principle, changes in the shroud thickness are retrofittable in current-design PWRs. As a practical matter, however, a thinner shroud leads to manufacturing problems, whereas a thicker shroud cannot, in general, be accommodated within the reactor support barrel. Additional steel, however, could be placed next to corners of the shroud plates which do not touch the support barrel. Such steel fillers would need to be cooled adequately such that the temperatures in the shroud and in the fillers do not exceed current design limits.

Calculations were performed for various shroud and water gap thicknesses using a radial representation of the reactor and the one-dimensional transport theory code, ANISN (4). The effect of moving the shroud further away from the core, and thus increasing the gap, is shown in Figure 4.2-26. 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 4.2-27. Increasing the shroud thickness from 1.0 in. to 4.0 in. yields an increase in reactivity of about

Figure 4.2-26
CHANGE IN REACTIVITY DUE TO CHANGES IN THE PERIPHERAL GAP DIMENSION

Figure 4.2-27
REACTIVITY CHANGES DUE TO VARIATIONS IN THE SHROUD THICKNESS
(0.15 MIL GAP)

0.8%. If a feed enrichment of 4.2% is assumed (for 50 Gwd/t discharge burnup), a uranium utilization improvement of about 2% is realized. 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. Because of limitations on available space, as mentioned above, it is doubtful whether increases in gap or shroud thickness could be retrofit into operating plants; major design changes would be required for either option.

4.2.9 Insertion of Natural Uranium Rods in Unoccupied Control Rod Guide Tubes to Increase the Conversion Ratio or to Act as Burnable Poison Shims

The insertion of blanket-like fuel rods as a means of increasing the conversion ratio has been investigated. Such rods could be inserted into unoccupied control rod guide tubes or instrumentation tubes. Initially, the power in these rods will be quite low since natural uranium is used, but, after several cycles, the buildup of plutonium will increase the power production. Moreover, the presence of those rods in the guide tubes will reduce the power peaking in the surrounding rods.

Calculations were performed with an integral transport theory code (2) using an assembly representation of a fuel batch. Rods of two sizes were considered: a large rod the size of a System 80TM control rod, and a small rod the size of a System 80TM fuel rod.

To simulate the use of blanket rods in available water hole locations, four rods were inserted in an assembly and the assembly was depleted. (The fifth water hole is generally reserved for an in-core instrument). This procedure simulates blanket rods in 80% of the CEA guide tube locations (in practice, only about 30% would be available). Uranium utilization was calculated and

compared to that of a reference case without blanket rods. To determine the effect of recycling the blanket rods after one exposure history the blanket rods were reinserted in the water holes of a fresh assembly for a second exposure history. Utilization was then calculated over both exposure histories and compared with the reference case.

Results obtained by inserting four standard fuel rods in each assembly (which could not be done in practice due to the number of CEA's committed to guide tube locations) show that the discharge burnup of the rods in lattice positions is decreased by about 1%. The energy extracted in the blanket rods (1.2%) compensates for this burnup decrease. The extra 0.2% in energy, however, is just compensated by the 0.2% extra natural uranium required to load the blanket rods. The result is no gain in uranium utilization.

Results also show that loading discharged blanket rods into fresh assemblies, or using larger blanket rods results in a net uranium utilization decrease. The blanket rods result in a loss of reactivity at EOC due to increased neutron absorption in the blanket rods. This loss of reactivity limits the burnup achievable in the lattice fuel rods. The loss of burnup in the lattice rods proves to exceed the energy benefit obtained from the blanket rods themselves. The utility of using blanket rods as a way of increasing uranium utilization is, therefore, doubtful.

The potential of using natural uranium rods in water holes as shims, in a low-leakage (SAVFUEL) cycle for example, could be attractive since the rods could be removed after the first irradiation cycle, thereby reducing the residual poison. Results show, however, that the reactivity reduction of shims in water holes is not enough to produce acceptable power distributions in a SAVFUEL cycle. This result is shown in Figures 4.2-28 and 4.2-29. The former presents the negative reactivity worth typically required during the first cycle of a three-batch SAVFUEL cycle, and the latter presents the negative reactivity worth resulting from various sizes of natural uranium rods (four in one assembly) placed in water holes. Even if depleted uranium is used, it is clear that the negative reactivity is not enough to produce an acceptable core-wide power distribution. The large natural uranium rods are worth only 1.5% $\Delta\rho$ at BOL. Approximately 6% $\Delta\rho$ is needed on an assembly basis

Figure 4.2-28

TYPICAL REACTIVITY WORTH * OF BURNABLE POISON SHIMS
DURING THE FIRST RESIDENCE CYCLE OF A FUEL ASSEMBLY

Figure 4.2-29

REACTIVITY WORTH* OF BLANKET RODS LOCATED IN CONTROL ROD GUIDE TUBES
(FOUR BLANKET RODS PER ASSEMBLY)

(or about 2% Δp core-wide). Furthermore, the depletion of U-235 results in increased poison worth during the first 10,000 MWd/t of burnup. This is opposite to the trend desired for burnable poison shims.

The use of blanket rods as a way of reducing the power peak near the water hole is also doubtful. First, large rods are required to affect the local power peak (greater than 2.5 cm pellet OD) which reduces uranium utilization. Secondly, it is important to load blanket rods in all the guide tubes of an assembly in order to reduce power peaking. Otherwise, the peak power would simply shift to near an empty or an instrumented guide-tube location. Due to the fact that about 70% of the guide tubes are committed to instruments or to control rods, this method of reducing power peaking is of doubtful usefulness.

4.2.10 Radial Blanket

The addition of triangular fuel assemblies, to fill in corners around the periphery of the core and hence reduce the neutron leakage, has been considered. The triangular assemblies are composed of fuel rods from discharged assemblies which can be reused because of the low power density on the core periphery.

Uranium utilization is increased due to reduced leakage and the extra burnup accumulated in the peripheral fuel rods. It is estimated that these two effects will combine to yield an improvement in uranium utilization of about 1.5%.

It is very doubtful that such a small improvement would justify the cost of the core design changes required to accommodate the addition of triangular assemblies on the core periphery. A possible alternative would be to maintain the same core geometry and use natural uranium (or exposed fuel) in some or all of the peripheral core locations. Any such change which would yield a significant uranium savings for a PWR would undoubtedly have a major impact on the radial power distribution and would degrade linear heat rate and DNBR margins.

4.2.11 Axial Blankets

The potential uranium savings associated with the use of uranium blankets at the top and bottom of the core has been investigated. If a standard length fuel rod is used, the active core length is reduced by replacing enriched pellets at the top and bottom with pellets containing blanket material (either natural or depleted uranium). If the length of the fuel rod is increased, the reflector can be added without reducing the length of the active core. The latter concept, of course, requires a major (and expensive) reactor design change.

The potential benefit of employing axial blankets was investigated using a one-dimensional (axial) model of the core. With this model, fuel was loaded and depleted over a time interval which simulates its full in-core residency. Both three-batch and five-batch management schemes were analyzed in this way. Uranium utilization was determined by comparing the cases with axial blankets with a reference unblanketed case. The average end-of-cycle reactivity of the blanketed cases was compared to the reference case to determine the appropriate core reactivity. In each case, the core environment was simulated by inputting radial neutron leakages and soluble boron concentrations determined for the reference case from detailed two-dimensional core calculations.

Depending on the blanket length, the blanket material and the fuel management scheme employed, uranium utilization increases to a maximum benefit of 4%. The uranium utilization improvement is less for the five-batch, extended-burnup fuel management than for the conventional fuel management. This result is a consequence of the fact that the uranium utilization is more fully optimized at the higher burnup and therefore obtains less benefit from increases in the burnup of the active core.

Accompanying the addition of an axial blanket is a change in peak linear heat rate for a fixed power rating of the core. For the long fuel rod option, the peak linear heat rate will decrease by about 2% due to the addition of the blanket, whereas the short active fuel option causes an increase in peak linear heat rate on the order of 5%. This higher axial power peaking may have some adverse effect on the safety analysis for those events which are strongly dependent on local power density. It will not, however, translate into a 5%

loss of thermal margin, but rather something less than this amount.

The conclusion of this analysis is that a significant uranium utilization improvement could be achieved by the use of axial blankets. This improvement is achieved at the expense of increased linear heat rate and burnup in the central portion of the core, which may lead to more limiting fuel performance conditions such as fission gas release. The benefits and the potential problems are quantified in more detail in Section 5.5.

4.2.12 Axial Shim Cutback

In any fuel management scheme which employs burnable poison shim rods, the poison residual at end of cycle can be reduced by using shims which are shorter than the fuel rods. Presently, shims are only needed in initial cores to reduce the soluble boron concentration at beginning of cycle and thereby prevent a positive moderator temperature coefficient (MTC). However, if cycle lengths are extended, as contemplated by many utilities, shims may also be needed in reload cores to control the MTC. Moreover, shims will be required in any case if the low-leakage fuel management scheme previously described is employed.

Assuming a shim worth of $2.0\% \Delta\rho$ at beginning of cycle, the burnable poison residual at end of cycle is about $0.3\% \Delta\rho$ if the shims are not cut back axially. Typical C-E cores utilizing shims have a 15.2 cm cutback (top and bottom), which reduces the poison residual by $0.05\% \Delta\rho$. Increasing the cutback to 38.1 cm would reduce the poison residual an additional $0.07\% \Delta\rho$. Thus, for cores which have a BOC shim loading of $2.0\% \Delta\rho$, increasing the cutback from 15.2 cm (current design) to 38.1 cm would decrease the EOC poison residual by $0.07\% \Delta\rho$, which is equivalent to a uranium utilization improvement of about 0.2%. Large shim cutback lengths, however, lead to undesirable power distribution at BOC (i.e., peaked at the top of the core) which have lower DNB margins.

4.2.13 Axial Relocation of Fuel Inventory Prior to Discharge

The potential of improved PWR uranium utilization by axially rearranging the fuel inventory was investigated. The axial rearrangement would be performed

in fuel assemblies prior to their final cycle of irradiation. Two types of rearrangements were investigated: inverted fuel assemblies and split fuel assemblies which are subsequently inverted (to get the low-burnup ends of the assemblies into the center of the core). These two arrangements are referred to as the inverted method and the split-inverted method, respectively.

Investigations of axially rearranging fuel were performed using a reference three-batch, 217-assembly class reactor using three-dimensional methodology. A set of standard three-cycle depletion calculations were compared to a modified Cycle 3 in which Batch 3 assemblies were either inverted or split and inverted.

Results show that a simple inversion of assemblies has virtually no impact on uranium utilization, or on the power distribution. This results from the fact that the burnup distribution in a PWR is virtually axially symmetric. The split-inverted method, however, places considerable amounts of low-burnup fuel material in the center of the core as shown in Figure 4.2-30.* This method adds reactivity to the core during Cycle 3 (Figure 4.2-31), and has a significant effect on the axial power distribution at BOC as shown in Figure 4.2-32. The 2.1 axial power peaking factor at the beginning of Cycle 3 is clearly not acceptable. At end of cycle, however, the axial peak is acceptable (Figure 4.2-33). The rate at which the axial power peak burns down (Figure 4.2-34) shows that the core must be depleted to at least 3000 MWd/t in Cycle 3 before the axial peak is below a potentially acceptable value of 1.5.

The results, which are based on a three-batch fuel management scheme, may be extrapolated to a five-batch scheme. The end-of-cycle reactivity benefit should be approximately three-fifths times the three-batch reactivity ($3/5 \times 0.4\% \Delta\rho = 0.24\% \Delta\rho$). The fuel assemblies which are to be depleted for one more cycle represent one-fifth instead of one-third of the total fuel assemblies. This reactivity benefit represents an enrichment savings of about 0.24 wt% and an increase of 0.6% in the uranium utilization.

* Note that only one-third of the fuel assemblies are split and inverted; the planar average burnup shown in this figure is a flux-weighted average of normal and split-inverted fuel assemblies.

Figure 4.2-30

CORE AVERAGE BURNUP DISTRIBUTION AT BOC3 RESULTING FROM AXIALLY RELOCATING FUEL INVENTORY IN ASSEMBLIES WHICH ARE TO BE DISCHARGED AT EOC3
(APPROXIMATELY 1/3 OF THE ASSEMBLIES IN THE CORE)

Figure 4.2-31
 REACTIVITY BENEFIT DURING CYCLE 3 ASSOCIATED WITH SPLITTING THE FUEL ASSEMBLIES
 AND PLACING THE LOW BURNUP PELLET TOWARD THE CENTER OF THE CORE

Figure 4.2-32
 AXIAL POWER DISTRIBUTION AT BOC3

Figure 4.2-33
AXIAL POWER DISTRIBUTION AT EOC3

Figure 4.2-34
AXIAL POWER PEAK COMPARISON DURING CYCLE 3 FOR THE SPLIT-INVERT METHOD
COMPARED TO THE REFERENCE CASE

The small uranium utilization benefit (0.6%), however, is not likely to offset the costs of axially relocating the fuel. This method of increasing uranium utilization is therefore not practical.

4.2.14 Isotopically Purified Zirconium

Since the zirconium used in the cladding, grids and control rod guide tubes consists of several isotopes having different absorption cross sections, core reactivity could be increased by increasing the percentage of those isotopes which have low absorption cross sections. For example, by increasing the percentage of the low cross section isotope Zr-90 from 50% to 80%, through advanced isotope separation techniques, the core reactivity can be increased by 0.6% $\Delta\rho$, which is equivalent to a 1.5% improvement in uranium utilization.

The benefits of this method are summarized in Table 4.2-1 which shows the relative abundance and the cross sections for the zirconium isotopes. This table shows that enriching Zr-90 from 51% to 80% reduces the average thermal cross section of zirconium to 68% of its original value and the resonance integral to 52% of its original value. The thermal absorption fraction (0.27%) and the fast absorption fraction (1.05%) of zirconium in a fuel assembly are thus reduced by these factors. The result is a total reduction of 0.6% in the relative neutron absorptions, which corresponds to a 0.06 wt% decrease in the required enrichment and a 2% increase in the uranium utilization (for a 3.0 wt% feed enrichment). Somewhat smaller results are expected when high burnup fuel is considered ($\sim$ 1.5% improvement).

The cost associated with commercial enrichment of zirconium has not yet been established. It is doubtful, however, that the costs will be low enough to justify the implementation of this process.

4.2.15 Full Use of Early Batches of an Initial Core

Initial PWR cores typically contain multiple fuel batches, each having a different enrichment. At the first refueling, the batch of lowest enrichment (Batch A) is discarded and fresh fuel is loaded on the periphery of the core

Table 4.2-1
REDUCED ABSORPTION IN ZIRCONIUM
BY THE ENRICHMENT OF
Zr-90

Isotope	Natural Abundance, ^a wt%	Enriched Mixture, wt%	2200 m/sec Cross Section, ^a barns	Resonance Integral, ^a barns
Zr-90	0.5146	0.800	0.100 ± .070	0.20 ± .03
Zr-91	0.1123	0.046	1.030 ± .520	6.50 ± .80
Zr-92	0.1711	0.070	0.260 ± .080	0.54 ± .12
Zr-94	0.1740	0.072	0.056 ± .004	0.30 ± .03
Zr-96	0.0280	0.012	0.017 ± .003	5.00 ± .40
Zr (weighted by natural abundance)			0.222 ± .070 ^b	1.12 ± .090
Zr (weighted by enriched mixture)			0.150 ± .060	0.58 ± .045
Zr ^{nat} (measured)			0.185 ± .003	1.10 ± .150

	Thermal Absorption Fraction, %	Fast Absorption Fraction, %	Total, %
System 80 Absorption in Zr ^{nat}	0.27	1.05	1.32
System 80 Absorption in Zr ^{enriched}	0.18 ^c	0.54 ^d	0.72
Reduced Absorption.....			0.60

a. S. G. Mughabghab/D. I. Garbes, "Neutron Cross Sections, Volume 1 Resonance Parameters," BNL-325 Third Edition, Vol. 1, June 1973.

b. It is noted that the 2200 M/S cross section of natural zirconium computed from the individual isotopes results in a value which has a larger uncertainty and is significantly different than the measured natural zirconium cross section.

c. $\frac{.1502}{.2218} \times .27\% = .18\%$

d. $\frac{0.578}{1.117} \times 1.05\% = .54\%$

for conventional OUT-IN fuel management). At the end of the second cycle, the next lowest enrichment batch (Batch B) is discarded, and so on. Since these early batches are discarded after fewer irradiation cycles than a typical reload batch, they achieve lower burnups than reload batches. Investigation of the reactivity and fissile content of these initial core batches at discharge reveals that their potential for additional exposure is very similar to that of typical reload fuel before its last irradiation cycle. Thus, this initial core fuel has the potential for improving uranium utilization by reinserting it for an additional cycle of exposure and thereby more fully consuming the fissile material remaining in such fuel.

Preliminary evaluations performed to date for conventional PWR fuel management schemes indicate the general feasibility of the reuse of these low-burnup, initial-core fuel assemblies. This can be accomplished by reinserting these assemblies into later cycles or, alternately, they can be used to provide some of the fuel for initial cores in later plants. The uranium savings which result from this reuse are modest, typically about 1% over a 30-year plant lifetime. Nevertheless, this option is attractive since it introduces no technical problems requiring extensive RD&D effort, and could be utilized in the near term. Examples of the two methods of fuel reinsertion/fuel sharing are given in Section 5.6.

4.2.16 End-of-Cycle Stretchout

Several methods exist for extending the cycle length beyond the point at which excess reactivity is exhausted at full power. The most common method consists of a simple power and secondary pressure reduction so that the lower fuel and moderator temperatures provide excess reactivity to allow additional burnup to be accumulated. A variation of this basic coastdown mode called "feedwater pressure augmentation" is to initially reduce the feedwater temperature by taking out of service feedwater heaters which allows more heat to be transferred across the steam generator for a given steam pressure and therefore extend operation at full core thermal power. By turning off the feedwater heaters, less turbine extraction steam is used, allowing more steam to pass through the turbine and produce electrical energy. Decreasing feedwater preheating in this manner also makes the steam cycle less thermodynamically

efficient, i.e., lowers the plant thermal efficiency.

The extra electrical energy output resulting from feedwater pressure augmentation is presented in Figure 4.2-35 as derived from three-batch fuel management calculations. The figure shows feedwater-pressure augmentation followed by a normal coastdown in thermal power. Also shown is a normal coastdown in power without feedwater-pressure augmentation. In a normal coastdown, the electrical power reduction to 80% is obtained in about 15 days, while for feedwater-pressure augmentation (followed by a normal power coastdown) the same power level is reached about three days later. The data given in Figure 4.2-35 show the benefit obtained in an equilibrium cycle. It therefore represents both the cycle stretchout at reduced power and the reduced cycle length at full power due to the cycle-to-cycle carryover of extended-burnup fuel. This figure clearly shows the benefits of feedwater-pressure augmentation. For a given electrical power reduction, FWPA increases the energy extractable from the core by about 0.5% over that obtainable by a standard coastdown for a three-batch fuel management scheme. This extra electrical energy extraction translates directly to a 0.5% uranium utilization improvement.

The benefit in using power coastdown and feedwater-pressure augmentation is somewhat smaller for a five-batch fuel management scheme than for a three-batch fuel management scheme as indicated above. The reason for this reduction is that a larger fraction of the burnup increase is carried over to the next cycle, therefore reducing the possibility of extending the cycle length.* In the limit of on-line refueling, the cycle extension equals zero, which means that coastdown would yield only a one-time benefit. After the first cycle, all subsequent coastdowns would be required just to maintain the original cycle length.

It should be noted that the above-quoted benefits for the use of feedwater-pressure augmentation are dependent on the assumption that the reduction in feedwater temperature is no more than 50°F. According to one turbine manufacturer (5), turbine capacity is limited when feedwater heaters are taken out of service because some turbine component loadings might otherwise increase beyond their design values. The 50°F value used here is representative

*This burnup carryover offsets the more negative moderator temperature coefficient of the five-batch core.

Figure 4.2-35

**EXTRA ELECTRICAL OUTPUT RESULTING FROM FEEDWATER
PRESSURE AUGMENTATION FOLLOWED BY
POWER COASTDOWN
(THREE-BATCH FUEL MANAGEMENT)**

*50°F MAXIMUM FEEDWATER
TEMPERATURE REDUCTION

of the feedwater temperature drop which occurs when these limitations are invoked. The potential benefits of feedwater-pressure augmentation are, therefore, dependent on the extent to which operations are restricted when feedwater heaters are taken out of service.

From this brief assessment, it was concluded that the benefits of employing end-of-cycle stretchout are indeed significant. This uranium utilization improvement option is, therefore, examined in greater depth in Section 5.7. In addition to a more thorough determination of fuel resource savings, the economics of stretchout are examined for both modes of feedwater system operation, the major contributing factors to the capability of a reactor to operate beyond its normal end of cycle are discussed, and a comparison of the benefits of stretchout is made between cores employing three-batch, standard-burnup fuel management and cores employing five-batch, extended-burnup fuel management.

4.2.17 Six-Month Cycle Length

It was noted in a previous section that PWR uranium utilization is improved 18% if the discharge burnup is extended from 30,000 to 50,000 MWd/t, and the number of refueling batches is increased from three to five while retaining an annual cycle length. If the 50,000 MWd/t burnup is maintained and the number of reload batches is increased to the theoretical limit, which represents on-line refueling, an additional uranium utilization improvement of 18% is theoretically available. Since on-line refueling is not possible with current PWR designs, and since the substantial plant changes needed to accommodate on-line refueling are economically unattractive, a more practical goal would be to lower the cycle length from one year to six months, and to increase the number of refueling batches from five to ten. If the batch discharge burnup was maintained at 50,000 MWd/t, use of the six-month/ten-batch scheme would improve uranium utilization by about 8.5% relative to the twelve-month/five-batch case. Since the plant capacity is a primary economic parameter, the viability of a six-month cycle length depends heavily upon the ability to reduce the refueling outage time.

Results are shown in Figure 4.2-36 which extend the data presented in Figure 4.2-1 to short cycles. This data assumes that the capacity factor for short cycles can be maintained the same as the reference yearly cycle. A more realistic assumption would be that the refueling outage remains a constant (6 weeks), which would reduce the capacity factor and increase the cycle length of the short cycles. These resulting cycle lengths are also shown in Figure 4.2-36 in parenthesis.

It is concluded that the use of six-month cycles has the potential for significant improvement in uranium utilization. To make such an operation schedule appealing to utilities, refueling times on the order of 7-10 days would be needed. This requires the use of some type of rapid refueling technique which, in general, necessitates substantial hardware changes. This improvement option was, therefore, deemed impractical for backfit into existing PWRs and, consequently, was not evaluated further in this study.

4.2.18 Increasing Moderator Temperature

A potential way of improving uranium utilization in PWRs consists of increasing the moderator temperature which has the effect of increasing the conversion ratio through a harder neutron spectrum. This method results in lower core reactivity which must be compensated by reduced absorptions in the control poisons. Reducing the moderator temperature back to its reference value at EOC results in an increased cycle length which increases uranium utilization. An added benefit is also obtained from the higher electrical efficiency due to higher secondary system pressures. Offsetting these benefits is the decrease in thermal margin which results from higher moderator temperature.

It should be recalled that sufficient thermal margin must be provided to accommodate not only steady-state conditions, but also various transients (e.g., maneuvering and control rod motion) and operational occurrences (e.g., partial loss of flow). This margin, normally expressed in terms of overpower margin, is design specific but is generally in the range of 120% to 130% of rated output. Some cores, because of over conservatism in their original analyses, have been designed with additional overpower margin beyond that

Figure 4.2-36
URANIUM UTILIZATION AS A FUNCTION OF ENRICHMENT
AND CYCLE LENGTH

normally needed. In some cases, the output of these plants has been upgraded (i.e., they have obtained a stretch-power rating), while in others increased flexibility in maneuvering, fuel management, etc. is available. Only that additional margin, above the overpower margin necessary to accommodate normal and abnormal operating occurrences, represents "extra" margin which can be employed for operating flexibility or to allow the implementation of fuel management or design options which degrade thermal margin. If no extra margin exists and the moderator temperature is increased, then reactor power must be lowered to maintain the same thermal margin.

Results show that for every 1°F increase in the primary coolant temperature, the increased conversion ratio yields a 0.1% improvement in uranium utilization. In addition, every 1°F increase in the primary coolant temperature would allow a 7-8 psia (0.48-0.55 bar) increase in the secondary pressure due to the change in the saturated steam pressure at the higher temperature, which, in turn, increases electrical efficiency by about 0.05% (i.e., from 34.00% to 34.05%). This increase in efficiency represents about a 0.15% increase in uranium utilization. The total increase in uranium utilization is therefore about 0.25% for every 1°F increase in the primary coolant temperature.

It is estimated that every 1°F increase in the primary coolant temperature will result in about a 0.5% decrease in thermal margin. The overall result is then that every 1% improvement in uranium utilization through increased primary coolant temperature results in 2% loss in thermal margin or power capability. Reactors with no extra thermal margin are therefore not able to utilize the benefit of higher moderator temperature without decreasing power level.

4.3 OPTIONS SELECTED FOR MORE DETAILED ANALYSIS

In the previous section, a rather substantial list of design and fuel management modifications, which have potential for reducing the uranium requirements of PWRs operating on the UO_2 once-through cycle, was examined. During this survey of potential improvement options, effort was concentrated on estimating the uranium savings achievable through the implementation of each option, and

on assessing the practicality of backfitting each option into an existing PWR. This survey of improvement options identified the following items for further evaluation:

1. Extended burnup with modified fuel management (a higher number of in-core batches),
2. Low-leakage fuel management,
3. Axial blankets,
4. Lattice modifications, and
5. Planned end-of-cycle stretchout.

In addition, the full use of early batches of the initial core, though having the potential for less significant reductions in uranium requirements, appears attractive because of the minimal requirements for RD&D necessary for implementation.

The improvement options identified in the above list form the basis for the composite improvement design. These options have been investigated in detail, both individually and in conjunction with the composite design to more accurately assess their impact on uranium requirements and power cost economics. In addition, an estimate of the RD&D effort required to realize each option is given along with any significant impact on the safety-related core physics parameters. Each of the above-identified improvement options is discussed individually in Section 5 and in composite in Section 6. Finally, the impact of employing these once-through improvement options on the recycle mode is discussed in Section 7.

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Section 5

DETAILED ANALYSIS OF SELECTED IMPROVEMENT OPTIONS

5.1 INTRODUCTION

The uranium utilization performance of a pressurized water reactor (PWR) operating on the uranium oxide once-through fuel cycle can be improved through any one of several techniques which increase the conversion ratio of the fuel and/or decrease the required end-of-cycle fissile inventory of the core. Techniques which accomplish these results do so by utilizing the neutron population more efficiently (e.g., by reducing neutron leakage) or by consuming the fissile material loaded more completely (e.g., by driving the fuel to higher discharge exposures). The improvement options explored in detail for this study are: an improved PWR fuel cycle which employs five in-core batches and extended burnup (50,650 MWd/t), a low-leakage fuel management scheme, lattice modifications, preplanned end-of-cycle stretchout, axial fuel blankets and the full use of early batches from the initial core. Each of these improvement methods can be incorporated into current-design PWRs without necessitating substantial hardware modifications and therefore could be introduced more rapidly than other improvement options, resulting in a greater impact on overall uranium resource usage.

In Section 5.2, the reference PWR design which was used as the basis for evaluating potential improvements is described. The individual improvement options, along with the economic and safety implications of employing these potential improvements in a PWR, are discussed in Sections 5.3 through 5.7.

5.2 DESCRIPTION OF THE REFERENCE PWR DESIGN

In order to estimate the potential uranium savings from the various improvement options, a reference reactor design and fuel cycle were established. For the purposes of this study, the Combustion Engineering, Inc. standardized PWR, System 80TM, was chosen as the reference design since it is representative of current, large-capacity PWRs. A cross-sectional view of the core and fuel assemblies is shown in Figure 5.2-1. The core design parameters for the System 80TM

Figure 5.2-1
REACTOR CORE CROSS SECTION
241 FUEL ASSEMBLIES

PWR are summarized in Table 5.2-1. System 80TM is a 3800 MWth plant with a nominal electrical output of 1270 MWe. The plant operates at a primary system pressure of 2250 psia (155 bar) and an average coolant temperature of 312°C. The core is composed of 241 assemblies with approximately one-third of the core (80 assemblies) replaced each year during refueling. A 16 x 16 fuel assembly lattice with a water-to-fuel volume ratio of 2.041 is employed in the standard design. The 0.826 cm O.D. fuel pellet is composed of approximately 3 wt% enriched uranium dioxide with a 0.635 mm thick Zirc-4 cladding. The active fuel length is 381 cm and the stack density is 10.054 g/cm³.

The reference fuel cycle used to assess improvements in uranium utilization was the UO₂ once-through fuel cycle with a three-batch, OUT-IN fuel management scheme and annual refueling. This fuel management strategy is employed almost exclusively in existing commercial nuclear power plants and therefore is the most appropriate fuel cycle for comparison. In the OUT-IN fuel management scheme, the most reactive fuel (either the highest enrichment fuel in the case of the first cycle core or the fresh feed fuel in the case of later cycles) is placed on the core periphery for its initial cycle of exposure. Subsequently, this fuel is moved into the core interior in a checkerboard pattern with other previously exposed assemblies. By situating the most reactive fuel on the core periphery, a flatter radial power distribution is achieved in the core. For a three-batch core, one-third of the fuel is discharged annually with equilibrium fuel batches receiving 30,380 MWd/t exposure. The equilibrium cycle length of 10,128 MWd/t corresponds to annual refueling at 75% capacity factor.

The discharged fuel batches from Cycles 1 and 2 receive exposures equal to approximately one-third and two-thirds of this nominal value, respectively. However, the cycle length during these initial cycles of operation differ somewhat from the equilibrium cycle value. Modification of these initial cycle lengths is advantageous since the core average enrichment for the initial core is substantially lower than the equilibrium enrichment (2.21 wt% versus 3.00 wt%) and consequently, the reactivity difference between the remaining batches of the initial core and the fresh feed fuel in Cycles 2 and 3 is quite large. In order to reduce power peaking problems as a result of this reactivity difference, the first cycle is extended from 10,128 MWd/t to 12,124 MWd/t allowing higher enrichment fuel to be loaded in the first cycle.

Table 5.2-1

SYSTEM 80TM CORE PARAMETERS

Core Power (MW(th))	3800
Nominal Electrical Output (MW(e))	1270
Fuel Assemblies	241
Fuel Rods Per Assembly	236
Control Rod Channels Per Assembly	4
Instrumentation Channels Per Assembly	1
Active Fuel Length (cm)	381
Specific Power (W/cm ³)	94.9
Average Linear Heat Rate (kW/cm)	0.175
<u>FUEL MANAGEMENT</u>	
Fraction of Core Replaced Per Year	1/3
<u>OPERATING PARAMETERS</u>	
Primary System Pressure (psia/bar)	2250/155
Reactor Coolant Average Temperature (°C)	312
Nominal Fuel Temperature (°C)	688
<u>FUEL ASSEMBLIES</u> (Dimensions at 20°C)	
Fuel Rod Array	16 x 16
Assembly Pitch (cm)	20.78
Fuel Cell Pitch (cm)	1.2860
Fuel Pellet Diameter (cm)	0.826
Zr Clad Thickness (cm)	0.064
Gap Thickness (cm)	0.00889
Water Channel Between Assemblies (half width) (cm)	0.1001
CEA/Instrumentation Channel Zr Guide-Tube: ID/OD (cm)	2.289/2.469
Coolant-to-Fuel-Pellet Volume Ratio	
Fuel Cell Ratio	1.714
Assembly Ratio	2.041
<u>FUEL STACK DENSITY</u> (g/cm ³)	
UO ₂	10.054

Mass flow data for the initial and equilibrium cycles is given in Table 5.2-2. More detailed information concerning core loading patterns and power distributions are given in Figures 5.2-2 and 5.2-3. In the initial core fuel management pattern, the most reactive fuel (Batch C) is loaded on the core periphery. The less reactive fuel assemblies (those belonging to Batches A and B) are placed in the core interior in a checkerboard pattern. The maximum relative box power is 1.384 and the maximum pin peak is 1.462. This pin peak value is typical of the maximum value attained in three-batch fuel management. The power distribution for the first cycle is somewhat centrally peaked. The same general fuel management scheme was employed in the equilibrium cycle (i.e., OUT-IN fuel management). The resultant power distribution for the equilibrium cycle is fairly flat. The maximum relative box power density is 1.244 with a pin peak of 1.304. For subsequent analyses of modified fuel management strategies, box power and pin peaking maxima typical of three-batch fuel management were used as target values to determine acceptability of loading patterns. The uranium and separative work requirements for the reference three-batch core, along with other fuel management information, are summarized in Table 5.2-3.

Figure 5.2-2
POWER DISTRIBUTION EARLY IN CYCLE 1 FOR A
CONVENTIONAL THREE-BATCH FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	SHIMS PINS/ASSY	ENRICHMENT WT %	SHIM LOADINGS GRAMS B-10 PER INCH
A	81	0	1.70	
B+	52	16	2.20	0.0119
B*	36	16	2.20	0.0104
C	40	0	2.70	
C+	8	16	2.70	0.0119
C*	24	16	2.70	0.0081

T	Q
XX	
YY	

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

						C 1	C 2	C 3	C 4
						0.489 0.843	0.701 0.993	0.787 1.051	0.812 1.075
				C 5	C* 6	C* 7	B* 8	C+ 9	B* 10
				0.456 0.810	0.648 0.967	0.857 1.097	0.910 1.068	1.080 1.233	1.010 1.139
		C 11	C* 12	A 13	B* 14	A 15	B* 16	A 17	
		0.535 0.904	0.850 1.142	0.825 0.916	0.982 1.141	0.980 1.041	1.111 1.235	1.045 1.096	
	18		19	A 20	B* 21	A 22	B+ 23	A 24	B+ 25
				0.877 0.951	1.047 1.179	1.017 1.074	1.155 1.275	1.101 1.151	1.207 1.301
	26		27		28	A 29	B+ 30	A 31	B+ 32
						1.036 1.090	1.182 1.301	1.132 1.183	1.258 1.362
	34	35	36	37	38	A 39	B+ 40	A 41	B+ 42
						1.144 1.195	1.285 1.391	1.208 1.256	1.321 1.411
	43	44	45	46	47		48	A 49	B+ 50
								1.222 1.269	1.351 1.444
	52	53	54	55	56	57	58	A 59	B+ 60
								1.267 1.311	1.384 1.462
	61	62	63	64	65	66	67		68
									A 69
									1.282 1.325

Figure 5.2.3
POWER DISTRIBUTION EARLY IN THE EQUILIBRIUM CYCLE
FOR A CONVENTIONAL THREE-BATCH FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	ENRICHMENT WT %
E	1	3.00
F	72	3.03
F*	8	2.74
G	72	3.03
G*	8	2.74
H	72	3.03
H*	8	2.74

T	Q
XX	
YY	

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

						H 1	H 2	H 3	H 4	
						0.547 0.899	0.697 0.971	0.767 1.051	0.753 0.995	
					H 5	H 6	H 7	F 8	H 9	F* 10
					0.596 0.999	0.853 1.124	1.043 1.179	0.826 0.922	1.129 1.204	0.818 0.895
			H 11	H 12	F 13	G 14	F 15	G 16	G* 17	
			0.702 1.136	1.088 1.235	0.979 1.091	1.116 1.193	0.972 1.123	1.057 1.106	0.994 1.020	
	18	19	G* 20	G 21	F 22	H* 23	F 24	G 25		
			1.070 1.126	1.156 1.193	1.038 1.148	1.241 1.301	0.986 1.072	1.138 1.195		
	26	27	28	G 29	G 30	F 31	G 32	F 33		
				1.251 1.298	1.202 1.260	1.000 1.086	1.217 1.277	1.070 1.096		
34	35	36	37	38	F* 39	G 40	F 41	G 42		
					0.919 1.013	1.098 1.138	1.076 1.142	1.240 1.270		
43	44	45	46	47	48	F 49	G 50	F 51		
						1.007 1.108	1.230 1.269	1.066 1.139		
52	53	54	55	56	57	58	F 59	G 60		
							1.038 1.120	1.143 1.153		
61	62	63	64	65	66	67	68	E 69		
								0.878 0.914		

Table 5.2-2

INVENTORIES FOR REFERENCE THREE-BATCH PWR

	<u>Initial Core^a</u>	<u>Equilibrium Batch</u>
Average Enrichment, wt% U-235	2.21	3.00
Full Power Days	328	274
<u>Fuel Loaded, kg</u>		
U-235	2195.51	1023.20
U-238	97123.00	33092.80
<u>Fuel Discharged, kg</u>		
U-235	233.60	272.80
U-236	83.20	127.20
U-238	31672.00	32312.00
Pu-239	150.40	180.00
Pu-240	57.60	70.40
Pu-241	30.40	44.00
Pu-242	8.80	15.20

a. Discharge quantities represent average of initial core batches

Table 5.2-3

FUEL MANAGEMENT INFORMATION
 UO_2 ONCE-THROUGH REFERENCE PWR CYCLE

Average capacity factor, %	75
Fraction of core replaced/refueling	0.33
Refueling interval, years	1
Fissile fabrication loss fraction	0.015
Equilibrium cycle enrichment, %	3.00
U_3O_8 requirements, tons/GWe ^a	
Initial Core	394 ^c
Annual equilibrium	194
30-year cumulative	6040
30-year net ^b	5840
Separative work requirements, 10 ³	
Separative work units (SWU)/GWe ^a	
Initial Core	200
Annual equilibrium reload	117
30-year cumulative	3600
30-year net ^b	3512
Plutonium in spent fuel	
Annual equilibrium discharge	
Plutonium fissile, kg/GWe	173
Plutonium total, kg/GWe	245
30-year cumulative discharge	
Plutonium fissile, kg/GWe	5169
Plutonium total, kg/GWe	7143
Residual U-235 in spent fuel	
U-235, wt%	0.85
Annual equilibrium discharge, kg/GWe	220
30-year cumulative discharge, kg/GWe	6498
Batch-average discharge burnup, MWd/kg	30.4
Peak discharge exposure, MWd/kg	55

a. Tails composition 0.2 wt%

b. The 30-year net cumulative requirement is the 30-year cumulative requirements less a credit for partially irradiated fuel discharged from the reactor after its final year of operation. This partially irradiated fuel could be used to reduce the fuel requirements for newer reactors still operating which are of the same design.

c. Long initial cycle and short second and third cycles required to make transition into equilibrium cycle with acceptable power distribution.

5.3 EXTENDED BURNUP WITH IMPROVED FUEL MANAGEMENT

One of the improvement options which was investigated in detail combined a low-leakage fuel management scheme with the five-batch, extended-burnup fuel cycle. During the initial investigation of potential improvement options, it was found that increasing the discharge burnup and reducing the fraction of the core which was replaced annually would significantly reduce uranium requirements. By increasing the discharge exposure of the fuel, better in situ utilization of bred fissile material is achieved. In addition, the quantity of (initially loaded) U-235 discharged in the spent fuel is reduced. This is particularly important for a once-through fuel cycle where unused fissile material is not recovered. Employing a larger number of in-core batches results in a lower core-average burnup at the end of cycle for a given discharge exposure, thereby reducing the fission product inventory (and therefore fission product poisoning), thus allowing the fuel to attain higher burnup values before refueling becomes necessary.

The analysis methods used in the investigation of extended-burnup fuel cycles are discussed in Section 5.3.1. Five-batch, OUT-IN fuel management and five-batch, low-leakage fuel management are described in Sections 5.3.2 and 5.3.3, respectively. The fuel performance, safety characteristics and the economic considerations of five-batch, extended-burnup cores are discussed in Section 5.3.4 through Section 5.3.6. The technical status, RD&D requirements and the likelihood of utility adoption are evaluated in the final two sections.

5.3.1 Analysis Methods

To analyze the five-batch, extended-burnup fuel cycle with low-leakage fuel management, two-dimensional core calculations were performed so that spatially-dependent information concerning fuel management schemes, power distributions and safety-related core physics parameters could be derived. To reduce computer requirements, coarse-mesh calculations were performed in which the System 80TM fuel assembly was homogenized with the water channels and surrounding gap, and then represented by a 4x4 or 2x2 nodal array. For the five-batch, OUT-IN fuel

management scheme, the multi-dimensional, few-group diffusion code PDQ (1) was used. The cross sections for PDQ were generated with the lattice cell code CEPAC (2) using ENDF/B-IV differential cross-section data. Pin peaking was estimated from the coarse mesh calculations by the CERISE (3) code. The five-batch core configuration with OUT-IN fuel management was also analyzed with ROCS (4), a two-group spatial diffusion code. The fuel assembly spectrum code DIT (5) was used to generate the few-group cross sections for the ROCS core model. Assembly pin peaks were determined from the ROCS assembly information with the FC pin peaking module (6). After comparing the assembly and pin power information from PDQ-CERISE and ROCS-FC, it was determined that ROCS-FC produced more accurate results due to the explicit modeling of the water hole in the FC pin peaking module. Whereas CERISE smears the water hole and surrounding fuel and consequently predicts lower flux levels around the water holes, FC reconstructs the flux for each pin and water hole to obtain more accurate pin peaking information. In addition, the coarse-mesh calculations are performed with ROCS in approximately one-third the time. Consequently, ROCS-FC was used to analyze the five-batch low leakage core. These codes and the models used are described in more detail in the following sections.

5.3.1.1 PDQ - Spatial Diffusion Calculations

PDQ solves the neutron diffusion-depletion problem in one, two or three dimensions through finite-difference techniques. A maximum of five lethargy groups can be employed, with the thermal neutrons being represented by one lethargy group or two overlapping groups. Adjoint or boundary value problems may be calculated in rectangular, cylindrical, spherical or hexagonal geometry. PDQ utilizes macroscopic data through the Harmony (7) code. Given the operating history (e.g., soluble boron rundown, power level, depletion schedule, etc.) PDQ calculates the core reactivity, the neutron flux, pointwise power level and nuclide number densities.

For the purposes of this study, a two-dimensional, four-group quarter-core PDQ model was constructed. Each assembly is individually modeled in the core along with the reflector and shroud. The System 80TM 16 x 16 fuel assembly and control rod channels are homogenized together with the surrounding water gap and are

represented by a 4 x 4 nodal array as shown in Figure 5.3-1. The array spacings of 2, 6, 6 and 2 cell pitch units separate adjacent mesh points. This coarse-mesh model accurately predicts core reactivity rundown, nuclide concentrations and assembly power while reducing computer time requirements as compared to fine mesh calculations in which each fuel pin and water region are more explicitly modeled.

5.3.1.2 CEPAC - Few-Group Cross Section Generation and Representation

The lattice cell code CEPAC was utilized to generate burnup-dependent, few-group constants for the PDQ calculation. CEPAC is a combination of several program modules including THERMOS (8), FORM (9) and CINDER (10). The input information required by CEPAC includes differential cross-section libraries, reactor core and lattice geometry data, and the reactor operating history. The neutron energy spectrum is represented by 83 groups spanning the energy range from 0 to 10 MeV. The 29 group thermal flux calculation is carried out in THERMOS for a one-dimensional model comprised of fuel, cladding and coolant. The 54 group epithermal calculation is performed in FORM for a homogenized cell. The B-1 approximation is used to account for leakage throughout the entire energy range.

Several criticality search options are available to maintain a critical spectrum. After the spectrum calculation has been carried out, a boron search or a group-independent buckling search is performed to determine the soluble boron level or buckling value which yields a critical system. Subsequently, the four-group fluxes are redetermined prior to the depletion calculation with the searched variable included. The burnup-dependent nuclide number densities, reactor fuel material inventories and reactivity are generated in the critical spectrum mode using CINDER. Spectra calculations are repeated after each burnup step to account for the effects of changing nuclide number densities.

Four-group microscopic cross sections were generated in the HARMONY format for the PDQ calculations. Individual CEPAC depletions with appropriate boron run-downs and burnup-dependent fuel temperatures were performed for the initial core and reload fuel enrichments.

Figure 5.3-1
PDQ COARSE MESH MODEL
SYSTEM 80™ 16 x 16 ASSEMBLY

5.3.1.3 CERISE - Generation of Pin Peaking Information

Since the coarse-mesh PDQ model utilized for this study does not provide pin peaking information, an auxiliary code, CERISE was used to generate the required pin powers. CERISE employs a spline fit technique to determine pin powers from the 16 coarse-mesh nodes for each assembly. The maximum peak pin predicted by the coarse-mesh model is typically 4% to 5% lower than the value obtained from a fine-mesh calculation. This difference is primarily due to the inability of CERISE to reconstruct the power peaking in the fuel rods located adjacent to water holes.

5.3.1.4 ROCS - Coarse-Mesh Diffusion Code

The second spatial diffusion code to be used for the fuel management calculations was ROCS, a multi-dimensional, coarse-mesh code. ROCS models each assembly as a 2×2 nodal array, with each node centrally located in the quarter assembly. ROCS employs two energy groups in the solution of the diffusion equation. The reflector and shroud regions are not explicitly represented, but rather are modeled through node dependent albedo boundary conditions. This reduces the number of nodes for solution and hence reduces the computer requirements. ROCS calculations can be performed with a variety of feedback options including moderator temperature, fuel temperature and xenon feedbacks which allow variation of these parameters between maneuvers (depletion steps). ROCS requires operating history in the form of soluble boron rundown and buckling values from which it calculates nuclide number densities, reactivities and assembly power levels.

The ROCS model employed for this study was a two-dimensional, quarter-core model utilizing DIT generated cross sections. Unlike the PDQ calculations where fuel temperature variation with burnup was included implicitly through the cross-section generation, a built-in fuel temperature correlation in terms of power level and burnup was used in ROCS through the fuel temperature feedback option. ROCS was chosen for the analysis of low-leakage fuel management because it performs the same spatial calculations as a coarse-mesh PDQ with comparable accuracy but requires one-third as much computer time. Furthermore, ROCS has more flexibility in performing end-of-cycle coastdown calculations than does PDQ because its structure was specifically designed with moderator and fuel temperature feedbacks and the ability to maneuver, e.g., change power level.

5.3.1.5 DIT - Few-Group Cross Section Generation

DIT is an assembly spectrum code based on integral transport theory which is used to generate two- or four-group cross-section data for spatial design codes such as ROCS and PDQ. The code consists of three major components: a spectrum calculation in approximate assembly geometry, a spatial calculation in the exact assembly geometry and a depletion calculation. Spectrum calculations are performed in an 85-group structure for selected spectrum geometries which are portions of an assembly. The 85-group cross sections are condensed to a few groups, and an assembly calculation is performed using integral transport methods. An 85-group B1 calculation with a buckling search is used to account for leakage and to maintain criticality. Depletion calculations are performed for each pin, and editing routines supply cross sections for spatial calculations.

DIT depletion calculations and additional spectrum calculations at selected time points were performed for each assembly type (i.e., unshimmed 16 shims, 12 shims, etc.) for different enrichments, soluble boron levels, fuel temperatures, moderator temperatures, etc. From these calculations, cross-section tables functional in the various core parameters were constructed. DIT was used to generate these cross-section tables instead of CEPAC since DIT explicitly models water holes and shims. As a result, the spectrum effect due to shims and water holes are incorporated directly into the cross sections rather than being included through correction factors as is the case for CEPAC. This is particularly important when analyzing low-leakage fuel management schemes because of the extensive use of burnable poison shims.

5.3.1.6 FC - Pin-Peaking Module

As was the case with PDQ, an auxiliary code was used in conjunction with ROCS to obtain pin peaking information. This code, the FC Pin Peaking Module, utilizes interassembly currents generated in the ROCS calculation to determine the power levels in individual fuel pins. A comparison of FC predicted pin peaks and those calculated through fine-mesh codes shows that FC over-predicts the pin powers by about 1%.

5.3.2 OUT-IN Fuel Management

To investigate the improvement in uranium utilization associated with an increase in fuel discharge burnup and a decrease in refueling batch size, two-dimensional coarse-mesh PDQ calculations were performed for the four initial cycles of the five-batch core. In addition, a near-equilibrium cycle was also analyzed.

The five-batch fuel management strategy is similar to that used in the reference three-batch core. Annually, one fifth of the fuel, that with the highest exposure, is discharged from the core. The fresh fuel is placed on the periphery of the core to attain a more uniform power distribution. The fuel which has been irradiated one cycle is moved into the core interior in a checkerboard pattern with more highly exposed fuel.

The fuel loading patterns and accompanying power distributions are given in Figures 5.3-2 through 5.3-5 for the Cycles 1 through 4. Figure 5.3-2 describes the enrichment and shim (B-10) loading in addition to the number of fuel assemblies of each batch type for the initial core. A near-equilibrium power distribution is given in Figure 5.3-6. Like the initial core of the three-batch fuel management scheme, the first cycle power distribution of a core employing five-batch, extended-burnup fuel management is centrally peaked. The maximum relative box power is 1.279 with a pin peak of 1.397. In order to provide a sufficient variation in reactivity to obtain an acceptable power distribution, two shim patterns were employed with two shim loadings in addition to the five-batch enrichments. During the transition cycles, the feed enrichment is increased gradually from 3.40 to 4.30 and the fresh fuel is placed on the edge of the core. Power peaking increases during the transition cycles as the higher enrichments are introduced. The near equilibrium cycle power distribution is fairly flat. The maximum relative box power is 1.335 in the once-burned fuel in the core interior. The maximum pin peak is 1.385.

As with the three-batch core, the first cycle is lengthened and the second cycle is shortened to obtain acceptable power distributions (i.e., relative pin peak power less than about 1.44). During the transition cycles, the large reactivity split between the initial core batches and the fresh reload fuel creates higher pin peaking than is seen in three-batch fuel management.

Figure 5.3-2
CORE POWER DISTRIBUTION EARLY-IN-CYCLE 1
FOR A FIVE-BATCH FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	SHIMS PINS/ASSY	ENRICHMENT WT %	SHIM LOADINGS GRAMS B-10 PER CM
A	48		1.70	
B	48		2.10	
C	28	16	2.60	0.0242
C*	24	16	2.60	0.0196
D	29	20	2.90	0.0242
D*	16	20	2.90	0.0196
E	40		3.40	
E*	8	16	3.40	0.0242

TQ
XX
YY

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

					E 1	E 2	E* 3	E 4	
					0.659 1.120	0.837 1.227	0.697 1.044	0.779 1.163	
				E 5	E 6	C* 7	B 8	D* 9	A 10
				0.673 1.174	0.954 1.282	0.949 1.138	0.988 1.062	0.989 1.187	0.831 0.902
			E 11	A 12	C* 13	B 14	D* 15	A 16	C* 17
			0.715 1.223	0.791 0.904	1.060 1.179	1.068 1.129	1.134 1.274	0.931 0.994	1.081 1.205
		18	19	C* 20	B 21	D 22	A 23	C 24	B 25
				1.055 1.173	1.075 1.135	1.141 1.283	0.970 1.029	1.134 1.235	1.123 1.187
		26	27	28	D 29	A 30	C 31	B 32	D 33
					1.141 1.285	0.973 1.034	1.148 1.241	1.138 1.200	1.204 1.325
	34	35	36	37	38	C 39	B 40	D 41	A 42
						1.150 1.241	1.140 1.201	1.192 1.329	0.996 1.058
	43	44	45	46	47	48	D 49	A 50	C 51
							1.195 1.333	1.007 1.066	1.161 1.274
	52	53	54	55	56	57	58	C 59	B 60
								1.191 1.299	1.186 1.253
	61	62	63	64	65	66	67	68	D 69
									1.279 1.397

Figure 5.3-3
CORE POWER DISTRIBUTION EARLY-IN-CYCLE 2
FOR A FIVE-BATCH FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	ENRICHMENT WT %
F	48	3.95

T	Q
XX	
YY	

T - BATCH
 Q - BOX NUMBER
 XX - BOX POWER
 YY - MAX 1 - PIN

Q

						F 1	F 2	F 3	F 4	
						0.768 1.171	0.998 1.282	1.050 1.301	1.060 1.303	
					F 5	F 6	E* 7	B 8	B 9	B 10
					0.816 1.271	1.089 1.352	0.990 1.088	0.775 0.811	0.780 0.818	0.776 0.810
				F 11	E 12	B 13	B 14	B 15	C* 16	C* 17
				0.856 1.230	1.001 1.124	0.853 0.883	0.848 0.891	0.941 0.998	0.946 0.996	0.896 0.943
	18		19	B 20	C* 21	E 22	D* 23	E 24	D 25	
				0.811 0.845	0.980 1.025	1.246 1.308	1.063 1.094	1.197 1.263	1.031 1.063	
	26		27	28	D 29	D* 30	B 31	D 32	C* 33	
					1.059 1.101	1.142 1.167	0.974 0.990	1.101 1.131	1.061 1.075	
34	35	36	37	38	E 39	C 40	E 41	C 42		
					1.267 1.386	1.048 1.073	1.308 1.414	1.041 1.061		
43	44	45	46	47	48	C 49	C 50	D 51		
						1.025 1.035	1.036 1.056	1.071 1.085		
52	53	54	55	56	57	58	E 59	C 60		
							1.267 1.389	1.012 1.032		
61	62	63	64	65	66	67	68	D 69		
								1.040 1.048		

Figure 5.3-4

CORE POWER DISTRIBUTION EARLY-IN-CYCLE 3
FOR A FIVE-BATCH FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	ENRICHMENT WT %
G	48	3.95

T	Q
XX	
YY	

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

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<u>BATCH</u>	<u>No. ASSY</u>	<u>ENRICHMENT WT %</u>
H	48	4.30

T	Q
XX	
YY	

[illegible]

Figure 5.3-6
CORE POWER DISTRIBUTION NEAR EQUILIBRIUM CYCLE
FOR A FIVE-BATCH FUEL MANAGEMENT SCHEME

<u>BATCH</u>	<u>No.</u> <u>ASSY</u>	<u>ENRICHMENT</u> <u>WT %</u>	<u>DESCRIPTION</u>
0B	48	4.30	ZERO-BURNED (FRESH) FUEL
1B	48	4.30	ONCE-BURNED FUEL
2B	48	4.30	TWICE-BURNED FUEL
3B	48	4.30	THRICE-BURNED FUEL
4B	48	4.30	FOUR-TIMES BURNED FUEL
5B	1	4.30	FIVE-TIMES BURNED FUEL

T	Q
XX	
YY	

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

Q — — —

						0B 1 0.758 1.125	0B 2 0.955 1.197	0B 3 1.025 1.242	0B 4 1.026 1.228	
					0B 5 0.789 1.177	0B 6 1.033 1.196	1B 7 1.058 1.101	4B 8 0.750 0.802	2B 9 0.996 1.081	4B 10 0.777 0.843
				0B 11 0.886 1.253	1B 12 1.114 1.177	4B 13 0.789 0.836	4B 14 0.766 0.807	2B 15 1.003 1.058	2B 16 1.055 1.115	3B 17 0.841 0.895
	18		19		4B 20 0.960 1.022	2B 21 1.105 1.152	1B 22 1.267 1.333	4B 23 0.830 0.860	1B 24 1.258 1.300	3B 25 0.982 1.012
	26		27			2B 29 1.139 1.189	3B 30 1.014 1.067	2B 31 1.130 1.168	3B 32 0.999 1.041	3B 33 0.987 1.020
34	35	36	37	38		1B 39 1.299 1.346	3B 40 1.014 1.053	1B 41 1.335 1.383	3B 42 0.959 0.982	
43	44	45	46	47	48		4B 49 0.855 0.892	3B 50 1.032 1.084	2B 51 1.141 1.190	
52	53	54	55	56	57	58		1B 59 1.282 1.339	4B 60 0.904 0.990	
61	62	63	64	65	66	67	68		5B 69 0.744 0.781	

The batchwise burnups for the first four cycles are presented in Table 5.3-1 along with estimates of the transition to the equilibrium cycle. These results show a transition to an equilibrium cycle length of 10,130 MWd/t for the five-batch OUT-IN fuel management scheme with an equilibrium feed enrichment of 4.3 wt% U-235. The batchwise discharge burnups approach their equilibrium cycle value of 50,650 MWd/t after 7 or 8 cycles.

The equilibrium enrichment predicted by PDQ, 4.3 wt%, is somewhat higher than that predicted by previous point reactor calculations, 4.15 wt% (see Figure 4.2-1). This can be explained by the following two effects:

- a. Increased Core Leakage - In order to increase the discharge exposure from 30,380 to 50,650 MWd/t, it is necessary to raise the feed enrichment by more than 1 wt%. Increasing the reactivity of the fresh fuel, which is located on the core periphery, appears to increase radial core leakage for the equilibrium cycle by $\sim 0.35\Delta\rho$ which represents an enrichment penalty of ~ 0.05 wt%.
- b. Higher Core-Average Exposure at End Of Cycle - The simple point reactor model assumes that each in-core batch receives exactly the same exposure each cycle. This is a reasonably accurate approximation for three-batch fuel management. However, the PDQ calculations performed for the five-batch scheme indicate that the fuel accumulates relatively more exposure in the earlier cycles which, for a given discharge exposure leads to a higher core-average exposure at the end of cycle (EOC).

Table 5.3-1
 BURNUPS ACCUMULATED BY EACH BATCH DURING EACH CYCLE FOR A FIVE-BATCH
 OUT-IN FUEL MANAGEMENT SCHEME
 (Cycles 1-4 are Calculated
 Transition Results for Cycles 5 through 11 are Estimated)

Batch	No. of Assem- blies Loaded	No. of Fuel Pins Loaded	Enrich- ment (wt%)	CYCLE												Batch Discharge Burnup (MWD/T)
				1	2	3	4	5	6	7	8	9	10	11	12	
A	48	11328	1.70	14150												14,150
B	48	11328	2.10	16007	5802											21,809
C	28	6160	2.60	18003	6939	6938										31,880
C*	20	4400	2.60	15457	6720	7246										29,423
C*	4	880	2.60	13827	6599	7878	7674									35,978
D	29	6264	2.90	18520	7093	7236	7674									40,523
D*	16	3456	2.90	16561	7517	7707	7674									39,459
E	40	9440	3.40	10501	8140	8417	9308	8426								44,792
E*	8	1760	3.40	10543	7033	8566	9308	8426								43,876
F	48	11328	3.95		6548	9854	10163	10070	8597							45,232
G	48	11328	3.95			8367	11398	10994	10274	7938						48,971
H	48	11328	4.30				8833	12330	11218	9487	8467					50,335
I	49	11564	4.30					9556	12581	10359	10119	8222				50,837
J	48	11328	4.30						9750	11617	11049	9826	8410			50,652
K	48	11328	4.30							9003	12391	10729	10051	8309		50,483
L	48	11328	4.30								9603	12032	10974	9930	8309	50,848
M	48	11328	4.30									9325	12307	10842	9930	
N	49	11564	4.30										9538	12160	10842	
O	48	11328	4.30											9424	12160	
BOC Exposure (MWD/T)				0	12057	14431	16538	18160	19581	21064	20995	21294	21065			
Cycle Burnup (MWD/T)				14971	6865	8530	9498	10276	10484	9682	10326	10027	10256			
EOC Exposure (MWD/T)				14971	18922	22961	26036	28436	30065	30746	31321	31321	31321			
Core Average Enrich- ment, wt%				2.53	2.99	3.37	3.71	3.98	4.16	4.23	4.30	4.30	4.30			

When the equilibrium cycle data of Table 5.3-1 is used to determine the ratio of the core-average end-of-cycle exposure (E_a) to the cycle length (E_c), the result is a value larger than that predicted by a point reactor model. For the same equilibrium cycle length, this results in a larger value of the core-average end-of-cycle exposure and a higher initial enrichment is required relative to a point model. Table 5.3-1 indicates incremental batchwise burnup ratios (batch burnup in cycle i /cycle length) of 0.93, 1.20, 1.07, 0.98 and 0.82 for each equilibrium residence cycle, respectively. A calculation of the ratio of the core-average exposure to the cycle length at EOC leads to a value of 3.09;

$$E_a/E_c = \frac{1}{5} (.93 + (.93 + 1.2) + (.93 + 1.2 + 1.07) + \dots) =$$

$$.93 + \frac{4}{5} (1.2) + \frac{3}{5} (1.07) + \frac{2}{5} (.98) + \frac{1}{5} (.82) = 3.09,$$

whereas the point model value is

$$E_a/E_c = \frac{Z+1}{2} = 3 \text{ (for } Z = 5\text{)}.$$

The higher EOC core-average exposure for the equilibrium cycle is equivalent to an enrichment penalty of 0.10 wt% relative to the enrichment predicted by the point model.

The inventories for the initial core fuel batches and the equilibrium core fuel batch are summarized in Table 5.3-2. The initial core fissile loading for the five-batch core is somewhat higher than that of the three-batch core (see Table 5.2-2) due to its extended cycle length. The first cycle of the five-batch core is 64 full power days longer than that of the three-batch core. The equilibrium batch loading, however, is 14% lower than that of the three-batch core. Approximately 17% of the initial fissile material is discharged in the five-batch core, whereas 27% of the U-235 loaded is discharged in the three-batch core, indicating the more complete consumption of fissile material in the five-batch core. The uranium and separative work requirements are summarized in Table 5.3-3 for the five-batch core along with other representative fuel management data. Over 30 years, the five-batch, extended-burnup cycle requires 11% less U_3O_8 than the reference three-batch cycle. The separative work requirements for the two cycles are almost identical.

Table 5.3-2

INVENTORIES FOR FIVE-BATCH, EXTENDED-BURNUP
PWR WITH OUT-IN FUEL MANAGEMENT

	<u>Initial Core</u>	<u>Equilibrium Batch</u>
Average Enrichment, wt% U-235	2.53	4.30
Full Power Days	392	274
Fuel Loaded, kg		
U-235	2514.6	880.8
U-238	96975.8	19603.1
Fuel Discharged, kg		
U-235	139.4	151.0
U-236	35.1	122.2
U-238	19579.5	18881.5
Pu-239	82.8	119.3
Pu-240	26.9	57.5
Pu-241	13.2	35.7
Pu-242	2.9	16.9

Table 5.3-3

FUEL MANAGEMENT INFORMATION
 UO₂ ONCE-THROUGH, OUT-IN FUEL MANAGEMENT

	<u>Three-Batch Reference</u>	<u>Five-Batch Extended Burnup</u>
Average capacity factor, %	75	75
Fraction of core replaced/refueling	0.33	0.2
Refueling interval, years	1	1
Fissile fabrication loss fraction	0.015	0.015
Equilibrium cycle enrichment, wt%	3.00	4.30
U ₃ O ₈ requirements, tons/GWe ^a		
Initial core	394 ^c	471
Annual equilibrium	194	171
30-year cumulative	6040	5395
30-year net ^b	5840	5124
Separative work requirements, 10 ³		
Separative work units (SWU)/GWe ^a		
Initial core	200	264
Annual equilibrium reload	117	118
30-year cumulative	3600	3670
30-year net	3512	3540
Plutonium in spent fuel		
Annual equilibrium discharge		
Plutonium fissile, kg/GWe	173	122
Plutonium total, kg/GWe	245	181
30-year cumulative discharge		
Plutonium fissile, kg/GWe	5169	3513
Plutonium total, kg/GWe	7143	5181
Residual U-235 in spent fuel		
U-235, wt%	0.85	0.78
Annual equilibrium discharge, kg/GWe	220	119
30-year cumulative discharge, kg/GWe	6498	3406
Batch-average discharge burnup, MWd/kg	30.4	50.65
Peak discharge exposure, MWd/kg	55	65

- a. Tails composition 0.2 wt%
- b. The 30-year net cumulative requirement is the 30-year cumulative requirements less a credit for partially irradiated fuel discharged from the reactor after its final year of operation. This partially irradiated fuel could be used to reduce the fuel requirements for newer reactors still operating.
- c. Long initial cycle and short second and third cycles required to make transition into equilibrium cycle with acceptable power distribution.

The QUT-IN fuel management calculations for Cycles 1 through 4 were reanalyzed with ROCS to provide a basis of comparison for later low-leakage analyses which would be performed with ROCS. DIT generated cross sections were used in the ROCS depletions. The agreement between the two calculations was fairly good overall. The ROCS and PDQ predicted values for core-average burnup and U-235 concentration differed by less than 0.3% and 0.9% at the end of Cycle 4, respectively. ROCS with DIT cross sections produced core reactivities from 0.6% to 1.7% higher than PDQ employing CEPAC cross sections. This reactivity difference is primarily due to a bias between the two cross-section sets, DIT versus CEPAC. The reactivity bias has been noted previously in other comparisons of DIT and CEPAC and follows the expected trend of increasing with burnup. Relative box power densities differed by a maximum of +3% to -3% at the end of Cycle 4. Typically, differences were $\pm 1.5\%$ for the majority of the assemblies.

Pin-peaking information was regenerated for those assemblies with the highest power densities using the FC pin peaking module. The maximum pin peaks obtained for five-batch fuel management are about 10% higher than those of the reference core. It appears that the CERISE-predicted pin powers shown in Figures 5.3-2 through 5.3-6 are somewhat optimistic. Two factors contribute to these underestimates: (1) pin peaking around water holes due to the increased thermal flux in these highly moderated water regions is not properly modeled in PDQ, and (2) the strong gradients between neighboring assemblies are not accommodated by the spline fit routines in CERISE which are used to predict pin peaks. Table 5.3-4 lists the pin/box ratio for some of the assemblies which were reanalyzed. The difference in the initial cycle pin-to-box ratios can be attributed to water-hole flux peaking. Typically, coarse-mesh calculations underpredict the pin peaks around the water holes by about 4% to 6%. The difference in the other pin/box ratios is due in part to water-hole peaking as well as the large gradients between assemblies as a result of the large difference in reactivity between fresh fuel and fuel that has burned for three or four cycles in the five batch core. The CERISE spline fit routines appear inadequate for the larger gradients

Table 5.3-4

PIN/BOX RATIOS FOR SELECTED ASSEMBLIES^a
 IN CYCLES 1 THROUGH 4 OF THE
 FIVE-BATCH EXTENDED-BURNUP CORE
 (OUT-IN FUEL MANAGEMENT)

<u>Cycle</u>	<u>Assembly</u>	<u>Box Power</u>	<u>CERISE Pin/Box Ratio</u>	<u>FC Pin Peaking Module Pin/Box Ratio</u>
1 ^b	69	1.302	1.083	1.123
2 ^c	6	1.221	1.225	1.314
3 ^d	22	1.385	1.037	1.154
4 ^c	3	1.197	1.199	1.328

-
- a. Assemblies with maximum pin peak values
 - b. 2000 MWd/t into the cycle
 - c. Beginning of cycle
 - d. 500 MWd/t into the cycle

which are present in the five-batch core. In the transition cycles, this situation is further aggravated by the enrichment difference between initial core fuel and the reload fuel. The resultant power distributions of the five-batch, extended-burnup core are not acceptable. The maximum pin peak values are too high by approximately 10% to 12% based on the reference value (a maximum pin peak of ~ 1.43). The basic problem with five-batch fuel management appears to be the power mismatch between fresh assemblies and assemblies which have been irradiated for three or four cycles. It may be possible to alleviate much of this pin peaking through a more optimal fuel-loading pattern. However, other techniques, such as the use of enrichment-zoned assemblies or the introduction of burnable poison, may also be necessary in order to reduce pin peaking to an acceptable level.

5.3.3 Low-Leakage Fuel Management

To assess the impact of reducing radial core leakage on uranium requirements, the five-batch, extended-burnup core employing a low-leakage fuel-management strategy was analyzed. The fuel-management parameters (e.g., cycle length, batch size, etc.) of the five-batch, OUT-IN cycle discussed in the previous section were duplicated in these analyses to facilitate assessing the benefit of employing low-leakage fuel management in the five-batch extended-burnup core. Only the fuel loading parameters (e.g., fuel enrichment, fuel placement, shim loading, etc.) were different from those of the OUT-IN fuel management analyses. The fuel loading strategy was deemed acceptable when the maximum pin peak was less than or equal to that obtained in the OUT-IN patterns. This practice ensured that the results were not biased by more stringent requirements than were followed in the OUT-IN scheme. In addition to repeating the first four cycles of the OUT-IN analysis, the calculations were extended for two more cycles to approach near equilibrium enrichments and cycle length.

The fuel management strategy for reducing radial core leakage involves loading the most reactive fuel in the core interior while the less reactive (once-burned or twice-burned fuel) is located on the core periphery. In order to situate the fresh fuel within the core while maintaining acceptable power distributions, burnable poison rods (shims) are employed in some of

the feed assemblies. In the C-E assembly design, these shim rods are placed in fuel rod locations and therefore displace fuel on a one-for-one basis. No burnable poison shim rods are placed in water hole locations. Although a penalty is paid in terms of higher enrichment to overcome the negative reactivity of any residual shim concentration remaining at the end of cycle, the savings of low-leakage fuel management more than compensates for this loss.

The effect on core leakage of placing burned fuel on the core periphery can be seen in Table 5.3-5. The end-of-cycle values differ by about 1% in leakage. Table 5.3-5 also shows that the shim residual reduces the potential savings in Cycles 2, 3 and 4 due to the improved fuel management since about 1% in reactivity is tied up in the shims at the end of cycle. In the equilibrium case, the shim residual should decrease to about 0.5% which would result in an additional savings of 8 kg to 10 kg U-235 in the feed batch. Since a major portion of the savings due to low leakage fuel management is offset by shim residual, significant incentive exists to employ a burnable poison design/material (such as gadolinium) which is more fully consumed by the end of cycle.

The fuel loading patterns and the corresponding power distributions for Cycles 1 through 4 are given in Figures 5.3-7 through 5.3-10. Values for pin powers are listed for those assemblies with the highest pin peaking values. The beginning-of-cycle power distribution for a near equilibrium cycle is shown in Figure 5.3-11. The shim patterns employed in the various cycles are shown in Figure 5.3-12. Power distributions in the low leakage core differ from those of the OUT-IN core in two aspects. The maximum box powers occur in the fresh fuel as opposed to once-burned fuel. Secondly, since once-burned and twice-burned fuel is placed on the core periphery instead of fresh fuel, the power distribution for the low-leakage core is more centrally peaked than the OUT-IN power distribution. The time point at which the pin power reaches the maximum value differs for the two fuel management schemes as seen in Figure 5.3-13. The maximum pin peak in low-leakage fuel management typically occurs toward the end of a cycle when the shims burn out, whereas the maximum pin peak for OUT-IN fuel management occurs in the beginning of the cycle when the fuel is most reactive.

Table 5.3-5

COMPARISON OF EOC LEAKAGE AND SHIM RESIDUAL FOR
THE OUT-IN AND LOW-LEAKAGE FUEL MANAGEMENT SCHEMES

Cycle	% Leakage		$\Delta\%$ Leakage (OUT-IN)-(Low-Leakage)	Shim Residual (% $\Delta\rho$) (OUT-IN)-(Low Leakage)
	OUT-IN	Low Leakage		
1	3.886	3.749	0.14	+0.12
2	5.219	4.252	0.97	-1.01
3	5.424	4.402	1.02	-0.97
4	5.383	4.250	1.13	-0.82
Near Eq.	5.237	4.179	1.06	-0.70

Figure 5.3-7

CORE POWER DISTRIBUTION BEGINNING OF CYCLE 1
FOR A FIVE-BATCH LOW-LEAKAGE FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	SHIMS PINS/ASSY	ENRICHMENT WT %	SHIM LOADING GRAMS B-10 PER INCH				
A	48	0	1.66					
B	48	0	2.05					
C	20	0	2.64					
C/	24	16	2.64	0.0242				
Cx	8	16	2.64	0.0115				
D	20	0	2.83					
D/	5	20	2.83	0.0242				
Dx	20	20	2.83	0.0168				
E	8	0	3.32					
E/	16	20	3.12	0.0242				
E*	8	20	3.32	0.0179				
Ex	16	24	3.32	0.0202				

						C 1	E 2	C 3	C 4
						0.588	0.950	0.893	0.909
			D 5	D 6	E* 7	B 8	Dx 9	B 10	
			0.580	0.816	1.004	1.040	1.113	1.085	
		D 11	B 12	Ex 13	A 14	Ex 15	A 16	E/ 17	
		0.660	0.870	1.064	0.914	1.132	0.941	1.157	
	18		Dx 20	B 21	Cx 22	A 23	Dx 24	B 25	
			1.076	1.090	1.167	0.939	1.108	1.071	
	26			E/ 29	A 30	C/ 31	A 32	D/ 33	
				1.189	0.978	1.103	0.951	1.098	
34	35	36	37	38	C/ 39	B 40	E/ 41	B 42	
					1.134	1.121	1.187	1.104	
							1.315		
43	44	45	46	47		48	C/ 49	A 50	C/ 51
							1.146	0.994	1.153
52	53	54	55	56	57	58	C/ 59	B 60	
							1.162	1.159	
61	62	63	64	65	66	67	68	D/ 69	
								1.203	

T Q
XX
YY

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

Figure 5.3-8
CORE POWER DISTRIBUTION BEGINNING OF CYCLE 2
FOR A FIVE-BATCH LOW-LEAKAGE FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	SHIMS PINS/ASSY	ENRICHMENT WT %	SHIM LOADINGS GRAMS B-10 PER INCH
F	8	0	4.08	
F/	8	8	4.08	0.0107
Fx	8	16	4.08	0.0107
F.	16	12	3.74	0.0242
F*	8	8	3.74	0.0242

T	Q
XX	
YY	

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

						D 1	C 2	F 3	Dx 4	
						0.525	0.691	1.102	0.802	
				C 5	E 6	Fx 7	Ex 8	Dx 9	D 10	
				0.501	0.787	1.223	1.169	1.175	1.256	
						1.540				
			C 11	F/ 12	Cx 13	Ex 14	D 15	E* 16	D/ 17	
			0.571	1.191	0.944	1.149	1.208	1.294	1.167	
				1.488						
	18		19	C/ 20	E/ 21	B 22	Dx 23	B 24	E/ 25	
				0.975	1.035	0.911	1.080	0.994	1.140	
	26		27		28	C/ 29	F. 30	B 31	F. 32	C/ 33
						1.005	1.300	0.872	1.299	1.013
34	35	36	37	38	B 39	C/ 40	B 41	C/ 42		
					0.863	0.917	0.845	0.947		
43	44	45	46	47	48	B 49	F* 50	B 51		
						0.845	1.300	0.851		
							1.494			
52	53	54	55	56	57	58	B 59	E/ 60		
							0.846	0.972		
61	62	63	64	65	66	67	68	D/ 69		
								0.899		

<u>BATCH</u>	<u>No. ASSY</u>	<u>SHIMS PINS/ASSY</u>	<u>ENRICHMENT WT %</u>	<u>SHIM LOADINGS GRAMS B-10 PER INCH</u>
G	16	0	4.20	
Gx	8	12	4.20	0.0210
G*	24	8	3.83	0.0242

T	Q
XX	
YY	

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

[illegible]

Figure 5.3-10
CORE POWER DISTRIBUTION CYCLE 4
FOR A FIVE-BATCH LOW LEAKAGE FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	SHIMS PINS/ASSY	ENRICHMENT WT %	SHIM LOADING GRAMS B-10 PER INCH
H	8	0	4.60	
Hx	8	12	4.60	0.0180
H*	16	12	4.18	0.0164
H/	16	12	4.18	0.0180

T	Q
XX	
YY	

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

						F* 1	G* 2	H 3	G* 4	
						0.369	0.732	1.100	0.925	
					Fx 5	G* 6	E/ 7	G 8	Ex 9	F/ 10
					0.467	0.656	0.632	1.260	1.030	1.228
				F. 11	G 12	D 13	Hx 14	C 15	H/ 16	Dx 17
				0.601	1.158	0.882	1.362	0.960	1.379 1.613	0.809
		18		19	F/ 20	F. 21	E* 22	Gx 23	C 24	E 25
					1.337	1.172	1.045	1.340	0.943	0.892
		26		27	28	E/ 29	H* 30	D 31	H* 32	E 33
						0.889	1.360	0.900	1.344	0.927
34	35	36	37	38		Dx 39	F 40	Ex 41	F. 42	
						0.821	1.143	0.874	1.072	
43	44	45	46	47	48		D 49	H/ 50	C 51	
							0.879	1.291 1.490	0.839	
52	53	54	55	56	57	58		E/ 59	G* 60	
								0.887	1.107	
61	62	63	64	65	66	67	68		Dx 69	
									0.774	

Figure 5.3-11
CORE POWER DISTRIBUTION NEAR-EQUILIBRIUM CYCLE
FOR A FIVE-BATCH LOW-LEAKAGE FUEL MANAGEMENT SCHEME

BATCH	No. ASSY	SHIMS PINS/ASSY	ENRICHMENT WT %
0B	48	0 OR 16	4.20
1B	48	0 OR 12	4.40
2B	48	0 OR 12	4.23
3B	48	0, 8 OR 12	4.11
4B	48	0, 8, 12 OR 16	3.86
5B	1	0	2.83

T	Q
XX	
YY	

T - BATCH
Q - BOX NUMBER
XX - BOX POWER
YY - MAX 1 - PIN

					1B 1	1B 2	0B 3	1B 4
					0.526	0.767	1.031	0.943
			1B 5	1B 6	2B 7	2B 8	3B 9	2B 10
			0.552	0.806	0.962	1.078	1.016	1.168
		2B 11	1B 12	3B 13	0B 14	4B 15	0B 16	3B 17
		0.612	1.111	1.021	1.276 1.507	0.834	1.262	1.012
	18		19	3B 20	2B 21	3B 22	3B 23	4B 24
				1.053	1.151	1.005	0.948	0.945
	26		27		28	4B 29	0B 30	4B 31
						0.894	1.279	0.900
							1.269	1.027
	34	35	36	37	38	3B 39	2B 40	4B 41
						1.038	1.156	0.920
	43	44	45	46	47		48	4B 49
								0.935
								1.272
	52	53	54	55	56	57		58
								4B 59
								0.949
	61	62	63	64	65	66	67	
								68
								5B 69
								0.749

Figure 5.3-12
SHIM LOADING PATTERNS

8 - SHIM PATTERN

12 - SHIM PATTERN

16 - SHIM PATTERN

20 - SHIM PATTERN

24 - SHIM PATTERN

Figure 5.3-13
EVOLUTION OF PLANAR PIN PEAK (Fxy)

As with the OUT-IN fuel management strategy, the power peaks are too high in five-batch, low-leakage fuel management due to inadequate power sharing between assemblies. As a result of the low leakage fuel management scheme, the maximum pin power peak is about 2% to 3% greater than that of the five-batch OUT-IN core. The pin powers are an additional 5%-6% higher relative to the OUT-IN core due to pin displacement by burnable poison shim rods. A reduction of these pin peaking values may be possible through zone-enriched assemblies and/or more optimal fuel management patterns.

The burnup values received for each cycle of exposure are summarized in Table 5.3-6. In the first cycle of exposure, the feed batch received relatively little exposure in the OUT-IN core since the fuel is on the periphery. It is not until the second, third and fourth cycles when the fuel is resident in the center of the core and is still fairly reactive that the majority of the burnup is acquired. This is in contrast to the low-leakage fuel management scheme where the fresh feed fuel receives its highest exposure in the first cycle when it is situated in the core interior. During the second cycle, the fuel is moved to the edge of the core where it receives only about 65% as high an exposure as it received during its first cycle of residency. The burnup accumulates more rapidly again in Cycles 3 and 4 when the fuel is returned to the center of the core. The total discharge burnup received by the equilibrium fuel batch in the low leakage scheme is higher than that of OUT-IN fuel management due to the displacement of fuel pins by burnable poison rods (about 5.6%).

The inventories for the initial core batches and the equilibrium feed batch are shown in Table 5.3-7. Due to the displacement of fuel by burnable poison rods, a higher equilibrium enrichment is required in the low-leakage scheme than in the conventional OUT-IN scheme. However, the total kg U-235 loaded is less than in the OUT-IN core due to the lower neutron leakage (cf. Table 5.3-2). The U-235 savings for Cycles 1 through 4 are summarized in Table 5.3-8. By Cycle 4, the cumulative U-235 savings amounts to 2.8%. The resource requirements are given in Table 5.3-9. Over thirty years, low-leakage fuel management results in a reduction of 2.5% in U_3O_8 requirements. The separative work requirements are almost the same due to the higher enrichment in the low-leakage case which compensates for the reduction in UO_2 loaded.

Table 5.3-6

EQUILIBRIUM BATCH BURNUPS BY CYCLE

Cycle of Exposure	Incremental Burnup, MWd/t	
	OUT-IN ^a	Low-Leakage ^b
1	9429	13492
2	12160	8648
3	10842	11068
4	9930	10574
5	<u>8309</u>	<u>9878</u>
TOTAL	50665	53660

a. 11328 pins in equilibrium batch

b. 10688 pins in equilibrium batch

Table 5.3-7

INVENTORIES FOR FIVE-BATCH, EXTENDED BURNUP
LOW-LEAKAGE FUEL MANAGEMENT

	<u>Initial Core</u>	<u>Equilibrium Batch</u>
Average Enrichment, wt%	2.47	4.44
Full Power Days	392	274
Fuel Loaded, kg		
U-235	2456.3	857.9
U-238	96998.1	18470.2
Fuel Discharged, kg		
U-235	129.8	128.0
U-236	34.3	147.0
U-238	19873.4	17747.0
Pu-239	82.7	114.1
Pu-240	28.2	54.2
Pu-241	14.4	33.6
Pu-242	3.6	15.8

Table 5.3-8

U-235 SAVINGS FOR CYCLES 1 THROUGH 4
FIVE-BATCH, EXTENDED-BURNUP, LOW-LEAKAGE FUEL MANAGEMENT

Cycle	U-235 Savings In Feed Batch ^a , kg	U-235 Savings, %	Cumulative U-235 Savings In-Feed ^a , kg	Cumulative U-235 Savings, %
1	58.3	2.3	58.3	2.3
2	40.0	4.9	98.3	3.0
3	7.6	0.9	105.9	2.6
4	32.7	3.7	138.6	2.8

a. OUT-IN Fuel Management Loading - Low Leakage Fuel Management Loading

Table 5.3-9

FUEL MANAGEMENT INFORMATION
 UO_2 ONCE-THROUGH EXTENDED EXPOSURE PWR

	OUT-IN	LOW LEAKAGE
Average capacity factor, %	75	75
Fraction of core replaced/refueling	0.2	0.2
Refueling interval, years	1	1
Fissile fabrication loss fraction	0.015	0.015
Equilibrium Cycle Enrichment, wt%	4.30	4.44
U_3O_8 requirements, tons/GWe ^a		
Initial core	471 ^c	459 ^c
Annual equilibrium	171	167
30-year cumulative	5395	5260
30-year cumulative, net ^b	5124	4998
Separative work requirements, 10 ³		
Separative work units (SWU)/GWe ^a		
Initial core	264	252
Annual equilibrium reload	118	116
30-year cumulative	3670	3588
30-year cumulative, net ^b	3540	3480
Plutonium in spent fuel		
Annual equilibrium discharge		
Plutonium fissile, kg heavy metal/GWe	122	116
Plutonium total, kg heavy metal/GWe	181	171
30-year cumulative discharge		
Plutonium fissile, kg heavy metal/GWe	3513	3317
Plutonium total, kg heavy metal/GWe	5181	4888
Residual U-235 in spent fuel		
U-235, wt%	0.78	0.72
Annual equilibrium discharge, kg heavy metal/GWe	119	101
30-year cumulative discharge, kg heavy metal/GWe	3406	2962
Batch-average discharge burnup, MWd/kg	50.65	53.67
Peak discharge exposure, MWd/kg	65	68

a. Tails composition 0.2 wt%

b. The 30-year net cumulative requirement is the 30-year cumulative requirements less a credit for partially irradiated fuel discharged from the reactor after its final year of operation. This partially irradiated fuel could be used to reduce the fuel requirements for newer reactors still operating.

c. Long initial cycle and short second and third cycles required to make transition into equilibrium cycle with acceptable power distribution

5.3.4 Fuel Performance at High Burnup

The analyses described above indicate that uranium utilization can be significantly improved by increasing both the discharge burnup and the number of refueling batches. However, before a commitment is made to extend burnup, the impact on fuel performance must be carefully assessed. It is necessary to estimate how much burnup limits can be extended for current fuel designs, and to evaluate design changes which would allow even higher exposures. To evaluate fuel performance at higher burnups, the following effects must be considered:

- . pellet-clad interaction (PCI)
- . fuel assembly dimensional changes
- . external clad corrosion
- . fuel rod internal pressure

5.3.4.1 Pellet Clad Interaction (PCI)

During irradiation, clad creepdown and fuel swelling and relocation eventually lead to pellet clad contact. Once contact has occurred, the clad is more susceptible to failure due to PCI, a mechanism that is possibly assisted by stress corrosion cracking (SCC). Fuel becomes more susceptible to PCI failures as burnup increases because of the following:

1. Higher clad stresses which occur due to the closure of the as-fabricated gap between fuel and cladding as a result of clad creepdown and pellet swelling and relocation, and
2. The increased availability of iodine and cesium (fission products), which are expected to be the primary corrosive species contributing to stress corrosion cracking.

The pellet-clad interaction mechanism has been identified as a cause of some fuel failures in irradiations to conventional burnup values (approximately 30,000 MWd/t). Thus, it is clear that pellet clad interaction could occur when extended burnups are employed. Guidelines are currently provided for the operation of reactors to avoid failures from PCI. In addition, DOE and industry programs are currently under way with the eventual intent of developing fuel immune to PCI-type failures. The major concern in this study is the question of whether the use of extended-burnup fuel will require additional R&D programs beyond that currently under way for conventional burnup fuel, and/or further restrictions on plant operation.

The probability that the fuel will fail due to the PCI mechanism is dependent both upon the probability that the fuel will be in a condition which is susceptible to PCI failures and to the probability that a transient of sufficient magnitude to initiate PCI failure will occur. As mentioned above, the conditions which make the fuel more susceptible to PCI failure are the higher cladding stresses which result from pellet-clad contact and stress corrosion cracking. Fresh fuel has little tendency for PCI failure because of the as-fabricated gap between fuel and cladding and the absence of fission products to contribute to stress corrosion cracking. The susceptibility of fuel to PCI-type failures (during a power change) increases with burnup as the gap between fuel and cladding closes due to clad creep-down and pellet swelling and relocation, and as corrosive fission products become available to contribute to stress corrosion cracking.

Although power ramp tests for high burnup fuel will have to be performed to definitively establish whether or not the conditions within the fuel rod which can lead to PCI failures during a power transient become more adverse for extended burnup fuels, there are a number of reasons for believing that the propensity for PCI-type failures (upon a given power rate change) will not be significantly higher at extended burnups than for fuel which has been irradiated to the burnup range of 20,000 to 30,000 MWd/t. The reasons for believing that propensity for PCI will not be significantly more adverse than for the standard burnup fuels are as follows:

1. The peak cladding stresses are not expected to be burnup dependent once clad creep-down and pellet-swelling and relocation have resulted in the onset of firm contact; the onset of such firm contact typically occurs in fuel after about two cycles of irradiation.
2. Corrosive fission products sufficient to result in SCC at a critical cladding stress level are likely to be available after two cycles of irradiation.
3. The oxide layer on the cladding increases in thickness with burnup and thus may provide an improved barrier to SCC at higher burnups.

Even if, as postulated above, the local conditions within a fuel rod which leave it susceptible to PCI-type failure upon a power ramp change are no more adverse for high-burnup fuel, the probability for PCI-type failure will be larger for extended-burnup fuel if such fuel undergoes a large number of power changes of sufficient magnitude to cause PCI-type failures during irradiation. For extended-burnup fuel, the length of time that the fuel pellet and the clad are in contact increases considerably (from about one year for current designs to about three years for high-burnup designs). This increase in the pellet and clad contact time, along with the increase in the number of power changes due to refueling and/or power maneuvers may subject the fuel to a larger number of transients of sufficient magnitude to cause PCI failure.

Figure 5.3-14 shows the expected limiting radial power peaking factors in fuel batches which have been in the reactor for a various number of cycles along with the maximum expected power change from one cycle to the next cycle. Pellet-clad interaction has not been observed in fuel with very low burnups. Data from recent ramp tests indicates that although the propensity of PCI failures is burnup dependent, it appears that the effects of fuel exposure tend to saturate before the burnups of the conventional fuel cycle are attained. For the five-batch, extended burnup fuel management scheme, the

Figure 5.3-14
CYCLE COMPARISON FOR A THREE-BATCH AND A FIVE-BATCH HIGH BURNUP
FUEL MANAGEMENT SCHEME FOR PCI COMPARISONS

maximum power change observed at the beginning of the second cycle is somewhat larger than that present in three-batch fuel management. This larger power change might be postulated to lead to more PCI failures. However, the increase in discharge burnups is not expected to further increase the susceptibility to PCI-type failures.

In summary, it appears that propensity of fuel to fail upon a power ramp from the PCI mechanism does not become more adverse with extended burnup, but that the probability of fuel failure due to PCI may be greater because of the longer period of time during which the fuel is susceptible to PCI-type failures. Power ramp tests for high burnup fuel will have to be performed in order to establish the acceptable power envelopes (i.e., the acceptable combinations of local burnup, power, power change, and rate of power change) for high-burnup fuel. Under the assumption that existing programs will develop fuel less susceptible to failure by the PCI mechanism, it is thought that PCI failure should not preclude operation of fuel to higher discharge burnups.

5.3.4.2 Fuel Assembly Dimensional Changes

Irradiation can cause both stress-free and stress-induced permanent dimensional changes in the components of the fuel assembly. These include: axial growth of the fuel rods, poison rods, and control-rod guide tubes; fuel and poison rod bowing; fuel assembly bowing; and retention grid relaxation. Although engineering evaluations and tests are needed to determine if current fuel assemblies have adequate margin at extended burnups and whether design changes are required, it appears likely that the effects of fuel assembly dimensional changes can be accommodated by providing adequate allowances for these dimensional changes in fuel rod and assembly design.

5.3.4.3 External Clad Corrosion

Clad corrosion occurs when the fuel clad chemically interacts with water to form Zirc oxide with hydrogen pickup (hydriding). Under normal steady-state operation, current PWRs have not experienced excessive corrosion on the outside surface of fuel rods. The heat fluxes and residence times have been accommodated without serious consequences. This phenomenon, however, is not well understood, and technical uncertainties exist when fuel exposure

is extended. If exposures and residence times are increased significantly, it is possible that a thicker layer of oxide and crud deposits may develop. The increased insulation may raise the oxide cladding interface temperature sufficiently to further accelerate the corrosion of the zircaloy cladding. Because of this potential for accelerated zircaloy corrosion, it will be necessary to develop further the data base for in-reactor corrosion of the zircaloy at typical power reactor conditions for in-reactor residence times anticipated for extended burnup fuel. Clad corrosion at high burnups is the subject of a joint program initiated in 1978 between EPRI, KWU, and C-E. Data will be available at the end of 1980.

5.3.4.4 Fission Product Release and Rod Internal Pressures

All modern PWR fuel rods are internally pressurized with helium to provide better gap conductance and resistance to cladding creep-down caused by the primary system pressure. The fission gas which is released as the fuel is irradiated increases the internal rod pressure. At high burnups, the end-of-life lead rod pressures may exceed design limits depending on the severity of the local power ramping during maneuvering cycles and the specific occurrences of Conditions I and II transients.* Significant steady-state irradiation data has been collected on standard burnup rods and shows that fission gas release is very low for normal fuel operation below approximately 12 kW/ft, even at 30 GWd/t. The effects of the larger fission gas release become more severe at high burnups because of the larger fission gas inventory within the fuel, an increase in the fission gas release rate which occurs at high burnups, and a decrease in the void space available for accommodating the mixture of helium and fission gas which results from clad creep-down and fission-product swelling. Further, the release of the gaseous fission

* Condition I Occurrences--normal operational events which are expected to occur during startup, shutdown, power operation and maneuvering.

Condition II Occurrences--faults of moderate frequency (i.e., assumed to occur one or more times during the lifetime of the plant) which can be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action.

products, xenon and krypton, can reduce the heat conductance through the gap between fuel and cladding, resulting in an increase in fuel temperatures and the subsequent release of additional quantities of fission gas.

Since it is currently believed that the release of fission gas at high burnup is the most significant factor which could potentially limit the discharge burnup of fuel in LWRs, a brief study has been undertaken to determine if fuel design changes will be necessary if the batch-average discharge exposure is increased from the current 30,000 MWd/t to 50,000 MWd/t. In this study, the fuel rod internal pressure was determined as a function of fission gas release rate, gas plenum size and configuration.

Ground Rules of the Rod Internal Pressure Study

Fission gas inventory calculations are based on the power history of the peak power rod during the fuel residence time. A single-node calculation is performed at a linear heat generation rate (LHGR) typical of the local (3-D) peak. Fuel temperature profiles used in the calculation of fission gas release are based on rod power history and consider normal operating transients which could be experienced by the lead power rod throughout its in-core residence time. Figure 5.3-15 shows the 3-D maximum pin-peaking factor as a function of maximum local (3-D) burnup.

This envelope was constructed based on the burnup and power distribution data of a five-batch, extended burnup equilibrium core. For the equilibrium cycle, an average discharge burnup of 50,650 MWd/t is achieved for a fuel enrichment of 4.3 wt%. The power history envelope used for the fission gas release calculation was obtained by raising the plotted envelope in Figure 5.3-15 by a factor of 1.34 to account for all the possible normal operating transients during the fuel residence time in the core.

Fuel temperature calculations at high burnups are based on a fixed gap conductance of $11.36 \text{ kW/m}^2 \text{ }^\circ\text{K}$ to reflect a stable fuel-cladding contact situation in an initially all-helium prepressurized PWR rod. The

Figure 5.3-15
POWER HISTORIES FOR FUEL PERFORMANCE IN FIVE-BATCH EXTENDED-BURNUP
EQUILIBRIUM CORE

effects of flux depression and a thermal conductivity change due to porosity added by swelling are accounted for in calculating the temperature profile through the fuel.

The rod internal pressure calculation accounts for the initial pre-pressurized helium fill gas, fission gas release, cladding creepdown and axial growth, fuel swelling induced by fission-product buildup, helium absorption by the fuel, and the differential thermal expansion of the fuel and cladding. Available void volume regions within the rod for the accommodation of the helium, xenon and krypton gas mixture include the plenum volume, pellet dish and chamfer volume, pellet open porosity and the annular gap volume that is transformed into pellet crack volume by relocated pellet fragments. The plenum is assumed to contain a standard System 80TM fuel column spring which occupies 0.40 cm³ per cm over the 24.59 cm working length.

Fission Gas Release Model

A model for the prediction of fission product release enhancement was first developed by the NRC based on high-burnup, mixed-oxide fuel. The model could not be verified against high burnup UO₂ data because of the lack of such data. C-E believes the NRC model to be conservative when applied to UO₂ fuel operating at typical PWR heat ratings, and to be excessively conservative for low-temperature operation of non-restructured fuel, for which little open-literature data exists at burnups greater than 20 GWd/t.

A version of C-E's gas release model has been modified for high-burnup UO₂ and applied in this study to the calculation of the EOL rod internal pressure. This developmental model has separate release rates and burnup dependencies for fuel operating in five temperature regions:

- . above columnar grain growth temperatures,
- . equiaxed to columnar grain growth temperatures, and
- . three temperature regions below the temperature for equiaxed grain growth.

Total gas release and release enhancement rates at high burnups are significantly lower below 1250°C than above this temperature.

The prediction of fission gas release is considered to be the factor which most strongly affects a high-burnup fuel rod design. The primary reason for this is the current uncertainties in the industry as to fission gas release enhancement at high burnups. To demonstrate this uncertainty, internal pressure calculations are included in this study for the use of two gas release models:

1. the more realistic C-E developmental high-burnup, gas-release model, and
2. the very conservative application of the NRC enhancement factor to an existing C-E low-burnup, gas-release model.

Rod Internal Pressure Results

The lead fuel rod internal pressure calculated for the standard System 80TM fuel design by the C-E developmental gas release model is 170 bar (2460 psia) which is slightly higher than the system pressure of 155 bar (2250 psia) but should be within a no-lift-off criterion. If the NRC model is used, the rod internal pressure is calculated to be 379 bar (5500 psia). This pressure is excessive and exceeds both the system pressure and no-lift-off criteria.

Although it is difficult to tell in the absence of better experimental data whether design changes will be necessary to reduce internal rod pressures, a number of design changes have been explored to ensure that there are design options available to reduce internal rod pressure should experiment confirm the gas releases of current predictive models. A number of these design changes are illustrated in Figure 5.3-16.

In the left-most graph, the internal rod pressure is plotted against an increase in rod length, i.e., an increase in the length of the fission gas plenum. For no increase in rod length, the plenum size would be 25.4 cm. By increasing the gas plenum length, the internal gas pressure can be substantially reduced. In the graphs on the right of this figure,

Figure 5.3-16
COMPARISON OF C-E AND NRC FUEL ROD INTERNAL GAS PRESSURE MODELS
FOR STANDARD LATTICE DESIGN

the active fuel length has been decreased by 12.7 cm and 25.4 cm, respectively, in order to allow more space to accommodate fission gas release. These curves are also plotted as a function of the increase in rod length. As anticipated, internal pressures decrease as additional fission gas plenum space is provided by increasing the length of the fuel rod and/or by decreasing the active length of the fuel.

As illustrated by the figure, when the NRC gas release enhancement factor is used, a relatively large increase in the plenum length or a combination of rod length increase and fuel column shortening will be required in order to meet an end-of-life pressure criterion which precludes cladding from lifting off the fuel during steady-state operation. With use of the C-E developmental model for fission gas release at high burnup, the standard fuel design should satisfy the no-lift-off criterion, while a small increase in the plenum length (about 3.3 cm) or a small decrease in the fuel column length (about 5 cm) can satisfy a more rigorous criterion that the EOL rod internal pressure will not exceed the system pressure (2250 psi). From the viewpoint of technical feasibility of the discussed design changes, there are, however, structural and operational considerations. The option of increasing the gas plenum length may only be applicable to new reactor designs, depending on the actual length increase, since the longer fuel rod might necessitate an increase in the length of the reactor vessel. The shortening of the fuel column will increase the peak linear heat rate, thus decreasing the margins for such events as the loss-of-coolant accident. Another method to provide more room for fission gases is the use of annular pellets. This option is discussed more fully in Section 5.4.

5.3.5 Impact on Core Physics Characteristics and Safety Evaluation

Selected core physics safety parameters were calculated for the equilibrium cycle to determine the effects of the five-batch, extended-burnup cycle on the reactor design. To separate the effects of the particular fuel management pattern used, i.e., five-batch, extended-burnup fuel management or low-leakage fuel management, several of the parameters were calculated for both OUT-IN and low-leakage loading strategies. These calculations were performed with two-dimensional PDQ and ROCS models for the OUT-IN and low-leakage cores, respectively. Included for

reference with these results are the safety-related, core physics parameters for the equilibrium three-batch PWR; these provide a basis for a qualitative safety evaluation.

5.3.5.1 Boron Worth

Table 5.3-10 gives the inverse boron worths at the beginning of equilibrium cycle and at the end of equilibrium cycle for the three-batch core and the two five-batch, extended-burnup cores. The inverse boron worth for the five-batch core with OUT-IN fuel management is larger than that of the three-batch core as a result of the harder neutron spectrum in the five-batch core. This harder spectrum is due to the higher feed enrichment and the higher fission product burden of the five-batch core which results from the higher core-average burnup. In the low-leakage, five-batch core, approximately 5.6% of the fuel pins are displaced by burnable poison shim rods. Consequently, neutrons have an increased chance of undergoing more moderation and the neutron spectrum is slightly more thermalized than that of the five-batch, OUT-IN core. The smaller inverse boron worth of the low-leakage, five-batch core reflects this more thermal spectrum. The inverse boron worth is important in the safety context for the overdilution accident. In the event that unborated water is mistakenly pumped into the reactor coolant system during cold shutdown, it is important to have sufficient time to detect and terminate the dilution operation before the shutdown reactivity margin is reduced to zero. Since the boron worths are lower for both five-batch cores than for the three-batch core, the boron dilution accident, for a given soluble boron concentration, is expected to be less severe than in the standard three-batch core.

5.3.5.2 Moderator Temperature Coefficient

The moderator temperature coefficients (MTCs) are summarized in Table 5.3-11 for the three-batch core, the five-batch, OUT-IN core and the five-batch, low-leakage core. The MTC for the five-batch core with OUT-IN fuel management is more negative than that of the three-batch core at the beginning and end of cycle due to the increased blackness of the core caused by the higher enrichments and larger fission product

Table 5.3-10

EQUILIBRIUM CYCLE INVERSE BORON WORTHS

<u>Core</u>	Inverse Boron Worth (ppm/% $\Delta\rho$)	
	<u>BOC</u> ^a	<u>EOC</u> ^b
Three-Batch	108.1	95.6
Five-Batch OUT-IN	130.3	113.9
Five-Batch Low Leakage	120.8	106.2

a. Evaluated at Boron Levels of 969, 940, and 648 ppm, respectively

b. Evaluated at Boron Level of 10 ppm

Table 5.3-11

EQUILIBRIUM CYCLE MODERATOR TEMPERATURE COEFFICIENTS (MTC)^a

Core	Moderator Temperature Coefficient	
	$\times 10^{-4} \Delta\rho/^{\circ}\text{F}$	$(\times 10^{-4} \Delta\rho/^{\circ}\text{C})$
	<u>BOC</u>	<u>EOC</u>
Three-Batch	-0.78 (-1.40)	-3.02 (-5.44)
Five-Batch OUT-IN	-1.77 (-3.19)	-3.54 (-6.37)
Five-Batch Low-Leakage	-2.03 (-3.65)	-3.32 (-5.98)

a. Cycle Length = 274 full-power days.

inventories. The beginning-of-cycle MTC for the five-batch core with low-leakage fuel management is more negative than the MTC of the five-batch OUT-IN core. This is primarily due to the lower beginning-of-cycle soluble boron concentration of the low-leakage fuel management, 648 ppm versus 940 ppm in the core with OUT-IN fuel management. At the end of cycle when the boron concentrations are identical, the MTC of the low-leakage core is slightly less negative due to the increased water-to-fuel ratio resulting from the displacement of fuel pins by burnable poison shim rods.

The more negative MTCs of the five-batch cores mitigate the consequences of warm-water accidents, e.g., loss of load or total loss of forced reactor coolant flow. As the reactor coolant temperature rises, more negative reactivity would be added to the system. For cold-water accidents, such as the steam-line-break accident or the idle-loop-startup accident, the more negative MTC increases the reactivity inserted by the cold water, and results in a more limiting accident. For the steam-line-break accident, the results are further aggravated by the reduced shutdown worth discussed below.

5.5.5.3 Fuel Temperature Coefficients

Table 5.3-12 gives the fuel temperature coefficients (FTCs) for the beginning and end of the equilibrium cycle. The end-of-cycle FTCs are more negative than the beginning-of-cycle values due to the increased Doppler absorption in Pu-240 which has a higher concentration at the end of cycle. The small differences between the end-of-cycle FTCs for the three cores is primarily due to differences in isotopic composition. The five-batch, extended-burnup FTCs are somewhat more negative than the three-batch values due to the harder spectrum and higher concentrations of Pu-240 in cores with greater core-average exposure. These differences in FTCs should not significantly alter the consequences of anticipated operational occurrences or other postulated accidents.

5.3.5.4 Control Rod Worths

Calculations of the shutdown worth at the end of the equilibrium cycle indicate that the total shutdown worth (all control rods inserted) for the five-batch, OUT-IN core is about 10% lower than that of a three-batch fuel management. The reason for this reduction in the worth of the

Table 5.3-12

EQUILIBRIUM CYCLE FUEL TEMPERATURE COEFFICIENTS (FTC)

	Fuel Temperature Coefficient	
	$\times 10^{-5} \Delta\rho/^{\circ}\text{F}$	$(\times 10^{-5} \Delta\rho/^{\circ}\text{C})$
<u>Core</u>	<u>BOC</u>	<u>EOC</u>
Three-Batch	-1.30 (-2.34)	-1.46 (-2.63)
Five-Batch OUT-IN	-1.35 (-2.43)	-1.47 (-2.65)
Five-Batch Low-Leakage	-1.39 (-2.50)	-1.49 (-2.68)

control rods appears to be the result of a neutron spectrum change due to the higher feed enrichments and the higher core-average exposures. The resulting higher fast-to-thermal flux ratio reduces the effectiveness of the control rods which are primarily thermal absorbers. The net rod worth, which considers the possibility of having the most reactive rod stuck out of the core, appears also to be about 10% smaller for a five-batch fuel management as seen in Table 5.3-13. This loss of net rod worth results in a loss in margin in various accident analyses including the steam-line break. Shutdown worth calculations were not performed for the low-leakage core. However, previous low-leakage design studies indicate that the rod worth is at least as large as that of the OUT-IN fuel management.

The rodded cooldown curves (most reactive rod stuck out) are shown in Figure 5.3-17 for the three-batch core and the five-batch, extended-burnup core (OUT-IN fuel management) at the end of the equilibrium cycle. The reactivity insertion between 311°C and 121°C is 9% higher in the five-batch core due to the more negative MTC. This more negative reactivity function for the five-batch core results in a loss in margin for the steam-line-break (SLB) accident. However, sufficient margin exists to prevent return to power with the SLB accident despite the increased reactivity insertion due to the MTC. So the consequences of SLB for the five-batch extended-burnup core are no more severe than those for the three-batch core.

Control element assembly (CEA) ejection calculations were performed for the three-batch core and the five-batch, extended-burnup core with OUT-IN fuel management. Table 5.3-14 summarizes the results for the two fuel-management schemes at hot full power (HFP) and hot zero power (HZP). The ejected rod worths and the pin peaks are similar for the three-batch and five-batch cores at both HZP and HFP. The consequences of a CEA ejection in the five-batch core would, therefore, be somewhat less severe than in a three-batch core due to the more negative MTC and FTC.

5.3.5.4 Kinetics Parameters

Variations in the kinetics parameters due to five-batch, extended burnup fuel management are small. Table 5.3-15 gives the delayed neutron fraction and the prompt neutron lifetime at the beginning and end of equilibrium cycle for the three-batch core and the five-batch, extended-burnup core

Table 5.3-13

END OF EQUILIBRIUM CYCLE SHUTDOWN WORTH AT HOT FULL POWER

Core	Total Rod Worth, % $\Delta\rho$	Net Rod Worth ^a , % $\Delta\rho$
Three-Batch	17.38	11.90
Five-Batch OUT-IN	15.55	10.80

*Most Reactive Rod Stuck Out

Figure 5.3-17

RODDED COOLDOWN CURVES FOR END OF EQUILIBRIUM CYCLE
THREE-BATCH, STANDARD-BURNUP AND FIVE-BATCH EXTENDED BURNUP CORES
(OUT-IN FUEL MANAGEMENT)

Table 5.3-14

HOT FULL POWER AND HOT ZERO POWER
CONTROL ELEMENT ASSEMBLY EJECTION CALCULATIONS
FOR THREE-BATCH, STANDARD-BURNUP AND
FIVE-BATCH, EXTENDED-BURNUP CORES
(OUT-IN FUEL MANAGEMENT)

	Three-Batch Core ^a		Five-Batch Core ^b	
	Ejected Rod Worth, % $\Delta\rho$	Fr	Ejected Rod Worth, % $\Delta\rho$	Fr
Hot Full Power (Bank 5 inserted)				
Rod 54 Ejected	0.09	1.614	0.07	1.599
Hot Zero Power (Banks 1 through 5 inserted)				
Rod 21 Ejected ^c	0.49	5.98	0.35	5.57
Rod 54 Ejected ^c	0.19	2.53	0.15	2.43
Rod 67 Ejected ^c	0.31	4.08	0.35	4.40

-
- a. Unrodded Fr = 1.223 (Fr = maximum value over all fuel pins of the ratio of the axially integrated pin power to the core-average pin power)
- b. Unrodded Fr = 1.299
- c. Figure 5.3-18 shows the rod locations in a typical System 80TM core.

Figure 5.3-18
SYSTEM 80™ CONTROL ROD PATTERN

S= SPARE CONTROL ROD LOCATION
PLR= PART LENGTH CONTROL ROD

Table 5.3-15

EQUILIBRIUM CYCLE KINETICS PARAMETERS FOR THE THREE-BATCH,
STANDARD-BURNUP AND THE FIVE-BATCH, EXTENDED-BURNUP CORES
(OUT-IN FUEL MANAGEMENT)

Delayed Neutron Fraction, β	Three-Batch Core	Five-Batch Core
BOC	0.00636	0.00616
EOC	0.00554	0.00562
Prompt Neutron Lifetime, λ^* ($\times 10^{-6}$ sec)		
BOC	21.3	19.6
EOC	24.8	22.4

(OUT-IN fuel management). The delayed neutron fraction of the five-batch core is slightly lower at the beginning of cycle due to the harder spectrum and the higher core-average burnup. At the end of cycle, the five-batch core has a larger delayed neutron fraction. The higher U-235 inventory coupled with a larger Pu-241 and Pu-242 inventories contribute to a higher β value.

The prompt neutron lifetime, ℓ^* , for the five-batch, extended-burnup core is smaller than that of a three-batch core at both beginning and end of cycle. These lower ℓ^* values are a result of the larger macroscopic absorption cross section due to the higher concentrations of fissile isotopes, particularly Pu-239 and Pu-241, and to the greater inventory of fission products.

5.3.5.5 Azimuthal Xenon Stability

Current PWR reactors are designed to be inherently stable with respect to azimuthal xenon oscillations. The tendency for such oscillations is generally more pronounced at beginning of a power cycle because of the less negative moderator temperature coefficient. Negative power feedback tends to dampen these oscillations.

The more negative moderator temperature coefficient of a five-batch fuel management design (OUT-IN or low-leakage) would indicate increased stability with respect to these oscillations. However, the fuel inventory distribution and resulting power shape also influences azimuthal xenon stability, and further analysis is required to determine whether or not the five-batch design has increased stability relative to a standard three-batch design.

5.3.5.6 Decay Heat

The larger inventory of fission products due to the higher burnups of a five-batch fuel management (compared to conventional three-batch fuel management) results in higher long-term decay heat rates as shown

in Figure 5.3-19. This larger inventory does not, however, significantly affect the decay heat during the first few days after shutdown. The relatively short-lived fission products, which contribute to most of the decay heat during this time interval, have reached a saturated concentration during the first year of irradiation. Increasing the irradiation time has, therefore, no significant effect on the decay heat during the first few days after shutdown. Safety analyses, such as LOCA (small break or large break) are, therefore, unaffected by decay heat considerations. Additionally, some safety calculations are required to use the ANS decay heat standard based on infinite operation plus a +20% uncertainty which consequently is conservative and already encompasses extended burnup fuels.

In addition, the heat removal requirements of the fuel storage pool are unaffected since the heating requirements are based on the early (larger) heating rates (subsequent to being discharged from the reactor) which are unaffected by increased fuel exposure.

The long-term decay heating rates (after several months), however, are dependent on the number of years that the fuel is irradiated as shown in Figure 5.3-19. This change in the heating rate may affect the cooling period for fuel before it may be shipped away from the reactor, or lead to potential increases in the fuel shipping cask heat removal requirements. This effect, however, leads only to a small perturbation. For example, if the heating rate is acceptable 90 days after fuel discharge for 3 years of irradiation, the same heating rate is obtained 105 days after shutdown for 5 years of irradiation.

The long-term decay heating rates (after several years) could have a significant impact on the design of a long-term waste storage facility. Such a facility should be designed to accommodate fuel with additional decay heat due to the potential use of high burnup fuel. If such a facility is designed to a lower decay heating rate, many additional years of decay may be needed before high-burnup fuel may be placed in the ultimate storage facility. This circumstance could cause a need for additional interim fuel storage.

Figure 5.3-19
DECAY HEAT GENERATION vs
COOLING TIME PER ASSEMBLY

5.3.5.7 Shroud Heating

Current PWR fuel assemblies are usually surrounded by a shroud which helps maintain lateral dimensional stability of the fuel assemblies. For C-E reactors this shroud consists of 2.5 cm thick steel plates which are welded together such that their faces are adjacent to peripheral fuel assemblies.

Results of power distribution calculations performed for both three-batch and five-batch, OUT-IN fuel management indicate that the peripheral fuel pins which are adjacent to the core shroud may have a linear heat rate which is 15-20% higher in the five-batch case. This increase in peripheral pin power will yield an increase in shroud heating and higher shroud temperatures and associated stresses. This increase in shroud heating and concomitant increase in shroud temperature and stresses may either be unacceptable, or may require a more detailed analyses to demonstrate acceptability. However, low-leakage fuel management in the five-batch core reduces the power in the periphery of the core by about 15-20%, reducing the shroud heating level in the five-batch, extended-burnup core to that of the three-batch core.

The high peripheral power in the five-batch, OUT-IN fuel management could also increase the vessel fluence accumulated over its lifetime. No calculations have been performed herein to investigate this phenomenon; however, enough margin probably exists so that this phenomenon is not likely to restrict PWR operation, and low-leakage fuel management would eliminate this problem.

5.3.5.8 Off-Site Dose

The off-site dose following a reactor accident may be larger due to the larger fission product inventory as a result of the five-year fuel assembly irradiation time. Although no quantitative analyses have been performed to date to determine the impact of this larger fission product

inventory on reactor operating limits, it is believed that the offsite dose following an accident will not be altered significantly when high-burnup fuel is employed. This conclusion is based on the fact that the majority of the off-site dose comes from Xe, Kr and I isotopes whose concentrations saturate rapidly, after a year or two of irradiation. A more detailed analysis is recommended for future follow-on work to confirm this tentative conclusion.

5.3.5.9 Summary

The safety concerns of the five-batch, extended-burnup PWRs are not substantially different from those of the three-batch reference PWR. The slightly larger inverse boron worth makes the boron dilution accident less severe than in the standard three-batch core. The effects of warm water accidents are reduced in the five-batch core due to the more negative MTC. This more negative MTC however adversely affects cold water accidents and results in a more limiting accident in the five-batch core. The net control rod worth of the five-batch core is approximately 10% lower than the three-batch and this reduces the margin for some accidents. However, the ejected rod worths are similar to those of the three-batch core. As a result of the ejected rod worth and the more negative MTC and FTC, the CEA ejection accident is somewhat less severe in the five-batch core. Decay heat requirements for the five-batch core are the same as those for the three-batch core in the short term (first few weeks after shutdown). The long term decay heating rates which depend on the irradiation period may impact the long term waste storage design. The effects of five-batch fuel management on shroud heating is an increase of 15%-20%. However, the shroud heating level can be reduced to that of the reference core by employing low leakage fuel management. Finally, it appears that the offsite dose following an accident will not be significantly altered when extended burnup fuel is employed.

5.3.6 Economic Assessment

The economic competitiveness of a fuel cycle or reactor type determines, to a large degree, whether it will be adopted by utilities. Hence, the three fuel management schemes discussed in this section were evaluated in terms of their fuel cycle costs. Two sets of cost assumptions (see Table 5.3-16), near term and long term, were employed to compute the

Table 5.3-16

FUEL CYCLE COST ASSUMPTIONS

	Near Term	Long Term
Uranium Price, \$/lb	40	100
Conversion, \$/kg	4	4
Separative Work, \$/kg/SWU	100	100
Fabrication Costs, \$/kg HM	140	120
Spent Fuel Disposal Cost (Shipping and Waste Storage), \$/kg HM	135	135
Enrichment Tails Assay, wt%	0.2	0.2
Reactor Plant Life	30	30
Discount Rate	4.525	4.525
Carrying Charge Rate	8	8
Capacity Factor, %	75	75

-
- a. Somewhat higher than current fabrication costs because they represent possibly more expensive processes for extended-burnup applications.

individual fuel cycle costs. The near-term cost assumptions represent the economic environment of the present nuclear industry. The U_3O_8 cost is relatively low, \$40/lb, while fabrication costs are \$140/kg HM. The relatively high fabrication cost reflects the potential technical difficulties and expense of manufacturing high burnup fuel. The long term cost set reflects an increased scarcity of U_3O_8 with a price of \$100/lb and higher throughput in fabrication facilities with a slightly lower fabrication cost of \$120/kg HM. These unit costs were used during the Nonproliferation Alternative Systems Assessment Program (NASAP) for various reactor characterization studies.

The fuel cycle costs presented here were analyzed using the C-E-developed fuel cycle cost code EVALU84. The characteristics of each fuel batch in terms of feed and discharge material inventories and batch exposure values were used in the calculations. The fuel cycle costs reported here are thirty-year levelized costs which incorporate the time value of money. These fuel cycle costs are given in Table 5.3-17 for the three-batch core and the five-batch, extended-burnup core with the OUT-IN and low-leakage fuel management schemes. The UF_6 fabrication and back-end direct costs of the five-batch core are lower than those of the three-batch core because of the reduced batch size and uranium requirements. The indirect charges for the five-batch core are higher as a result of the increased core-average enrichment accompanied by a larger in-core residence time.

Table 5.3-17

THIRTY-YEAR LEVELIZED COSTS FOR PWR FUEL CYCLES (mills/kWh)^a

FUEL MANAGEMENT	<u>UF₆</u>	<u>SWU</u>	<u>FAB</u>	<u>Back-End Direct</u> ^b	<u>Indirect</u>	<u>Total</u>
Three-Batch OUT-IN	2.47 (6.05)	1.76 (1.76)	0.60 (0.52)	0.58 (0.58)	0.82 (1.53)	6.23 (10.44)
Five-Batch OUT-IN	2.15 (5.26)	1.73 (1.73)	0.39 (0.33)	0.35 (0.35)	1.02 (1.87)	5.63 (9.53)
Five-Batch Low-Leakage ^c	2.11 (5.15)	1.70 (1.70)	0.41 (0.35)	0.33 (0.33)	.98 (1.80)	5.54 (9.38)

a. Values without parentheses are for near-term assumptions, while those with parentheses are for the long-term assumptions.

b. Disposal costs

c. Includes extra cost for fabrication of shimmed assemblies

In the near term, the fuel cycle costs for the five-batch, extended-burnup core with OUT-IN fuel management are approximately 9.6% less than those of the three-batch core. Including low-leakage fuel management saves an additional 1.4%. The fuel cycle cost savings are slightly larger with the long-term assumptions for the five-batch core. The higher cost of uranium increases the fuel cycle cost difference between the five-batch and three-batch cores. The fuel cycle costs were calculated with a constant fabrication cost and a fixed spent fuel disposal cost. It should be noted that even if fabrication and disposal costs for high-burnup fuel were to double, the five-batch, low leakage scheme would remain competitive.

5.3.7 Technical Status and RD&D Requirements

The five-batch, extended-burnup fuel cycle with improved fuel management would provide substantial uranium savings if incorporated in present-design light water reactors. Prior to adopting this improvement option, however, several questions must be resolved through further analysis and RD&D programs.

One of the principal areas of investigation for the five-batch, extended-burnup fuel management concept is fuel performance for exposures greater than the standard burnup values of about 30 MWd/kg. In order to assess fuel performance at higher burnup values, measurements of fission gas release, pellet-clad interaction, fuel dimension variation and clad corrosion must be made as functions of exposure. The initial effort to obtain this information is presently under way under the sponsorship of the Department of Energy. Several utility/vendor teams have initiated programs to irradiate current fuel designs to burnups in the range of 36 to 45 MWd/kg. The data gathered from these programs will be used in the design, safety analysis, and licensing of advanced fuel designs which will receive exposures in the range of 45 to 55 MWd/kg. With respect to fission gas release, C-E is developing a data base to benchmark their gas release model at high burnups in prepressurized PWR rods. Data is already available at burnups close to 30 GWd/t from the C-E/EPRI joint program at Calvert Cliffs-1. Gas release fractions were less than 1% in

all rods, and no burnup enhancement was evident at 29.1 GWd/t. Significant new data, at burnups up to 45 GWd/t, will be obtained in 1980. Furthermore, gas release data under transient conditions is being obtained from the ramp tests performed at the Studsvik and Petten reactors for fuels irradiated to 45 GWd/t. To demonstrate the extended exposure capability of the advanced fuel design, it will be necessary to irradiate an entire reload batch to the target exposure values.

A second issue which must be addressed for the five-batch, extended-burnup fuel cycle is the potential loss in thermal margin due to higher fuel pin powers. The primary source of the higher pin powers is intrinsic in five-batch fuel management. With the smaller batch size (48 assemblies versus 80 in a three-batch core), power sharing between assemblies becomes more difficult and higher power densities occur in the fewer fresh assemblies. In addition, the large reactivity difference between fresh fuel and fuel which has been burned for four cycles create larger flux gradients between neighboring assemblies. This results in higher pin-to-box peaking factors which further increase the pin powers. In the low-leakage scheme, these higher values are aggravated by the displacement of fuel pins by burnable poison shim rods. As seen in Section 5.3, in the equilibrium cycle, about 5.6% of the fuel pin locations in the five-batch, low-leakage core are loaded with poison shim rods. Analyses should be conducted to determine quantitatively the effects of the higher pin power values on thermal margin and the DNBR. If the margin loss is unacceptable, methods to reduce the pin power values such as zone-enriched fuel assemblies or more effective shim design (use of gadolinium) should be investigated.

Several other issues merit further analysis. More extensive fuel management development must be performed to optimize loading patterns in terms of pin power peaking. Safety issues such as the more negative MTC and the smaller control rod worth should be investigated in terms of their effect on margin in accident situations. Quantitative analyses of decay heat rate, criticality and shipping requirements for fuel storage should be performed. Finally, based on the aforementioned data, a licensing assessment should be conducted.

5.3.8 Likelihood of Utility Adoption

The primary objective of utility management is to supply the electrical power demanded by its customers in a reliable and economic manner. Based on this premise, a utility will alter reactor operation, i.e., fuel cycle length, batch size, fuel management, etc., if it results in more economic operation with essentially the same reliability. For example, several utilities have recently expressed an interest in 18-month cycles. Although the fuel cycle costs increase under this option, the savings in reactor downtime and the replacement power costs outweigh the added fuel cycle costs.

Since the cost savings for the five-batch, extended-burnup fuel cycle are substantial (about 10%), the motivation for utility adoption is high. The primary question for the utility would be licensing of high-burnup fuel. When the issue of fuel performance has been resolved and the licensability of extended-burnup fuel has been demonstrated, utilities will most likely elect to employ this improvement option.

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5.4 LATTICE MODIFICATIONS

5.4.1 Available Options

Uranium utilization in PWR cores operating on the once-through fuel cycle can be improved by lattice design changes that increase the water-to-fuel ratio and/or reduce parasitic absorption (e.g., in the cladding material). An increase in the water-to-fuel ratio results in a more reactive core which reduces the fissile inventory required for criticality. An increase in water-to-fuel ratio can be achieved in a number of fashions:

- by increasing the spacing between the fuel rods (increasing the lattice pitch),
- by reducing the pellet (and rod) diameter, or
- by reducing the effective fuel density through the introduction of low cross section diluent materials or through the use of annular fuel pellets.

Among these options, the increase in lattice pitch would require modifications of reactor design and hence violate the criteria of retrofittability, a primary ground rule of this study. Based on this consideration, the other two options for increasing the water-to-fuel ratio, i.e., the reduction of the rod diameter and the reduction of the effective fuel density were considered for incorporation into the composite improved PWR core design. Among these two options, however, a reduction in fuel rod diameter is preferred, both because it results in somewhat larger improvements in uranium utilization (cf., Section 5.4.3) and because less fuel rod research and development would be required to implement this option. The use of reduced density fuel (including annular pellet fuel) would require considerable fuel design development and proof testing to ensure satisfactory fuel performance.

There are, however, three major design constraints which could potentially limit the improvement in uranium utilization achievable by reducing fuel rod

diameter: the effect on the moderator temperature coefficient of reactivity, the effect on core thermal-hydraulic performance, and the effect of increased burnup on fission gas pressure. These effects are discussed in detail in Section 5.4.5.

An improvement in uranium utilization can be achieved by reducing the concentration of parasitic absorbers in PWR cores. Gains in this area are made difficult by the fact that the development of current core designs has included this improvement technique. With the important exception of soluble boron, most parasitic absorbers have already been eliminated or minimized. One item which has some potential for further optimization is captures in cladding material. As discussed in Section 4.2.14, an improvement of about 1.5% can be obtained by the use of zirconium enriched through advanced isotope separation techniques in the low cross section isotope Zr-90. It is doubtful, however, that the costs associated with the enrichment of zirconium will be low enough to justify the implementation of such a process. An alternative method of decreasing parasitic absorptions in the cladding is to reduce the cladding thickness. The potential adverse impacts of this action are increased fuel rod bowing and a higher probability of losing clad integrity due to external corrosion or PCI. These issues, along with the potential benefits of reducing clad thickness will be discussed in detail in the sections which follow.

5.4.2 Analysis Methods

The calculation of uranium utilization for different fuel designs was performed using the CEPAC point reactor model. A CEPAC calculation was initially performed for the standard fuel design. Using the generated reactivity depletion curve, the end-of-life core average reactivity was established for five-batch fuel management with a discharge exposure of 50,650 MWd/t. This core-average end-of-life reactivity value was then used to estimate the achievable discharge burnup for the modified lattices. Having obtained the discharge burnup estimates in this manner, the uranium utilization was calculated in terms of the energy production (MWd) per short ton of U_3O_8 required, i.e., MWd/ST U_3O_8 .

5.4.3 Impact on Uranium Requirements

The improvement in uranium utilization resulting from the use of lattice designs with higher water-to-fuel ratios is illustrated in Figure 5.4-1. The curves on this figure are based on the use of modified lattice designs in conjunction with a five-batch, extended-burnup fuel management scheme. Curve A shows the benefit due to the reduction in the UO_2 density while retaining the same U-235 loading. A maximum of 3% improvement in the uranium utilization results if the UO_2 density is reduced from 10.054 to 7.9 g/cm³. Curve B of Figure 5.4-1 shows the change in the uranium utilization obtained by increasing the lattice pitch. A maximum benefit of 6.5% can be obtained if the rod pitch is changed from 0.5063 in. (1.286 cm) to 0.570 in. (1.448 cm). Curve C of Figure 5.4-1 shows the improvement in uranium utilization obtained by reducing the fuel rod diameter while maintaining a constant clad and gap thickness. A maximum gain of about 5% is obtained by reducing the rod outside diameter from 0.382 in. (0.970 cm) to 0.346 in. (0.879 cm). The corresponding reduction in pellet diameter is from 0.325 in. (0.826 cm) to 0.293 in. (0.744 cm).

The improvement in uranium utilization achieved by a combination of pellet diameter reduction and clad thickness reduction is shown in Figure 5.4-2. Curve C in this figure represents the early ascending part of the similarly named curve in Figure 5.3-1 (i.e., the portion of the curve for water-to-fuel values of 2.0 to 2.4). Curve D illustrates the additional benefit obtainable from a clad thickness reduction (with gap thickness kept constant). This curve represents the uranium utilization of a fuel rod having a clad thickness of 22.5 mils (0.572 mm) (10% reduction). As indicated by Curve C, a reduction of the pellet diameter from 0.325 in. (0.826 cm) to 0.315 in. (0.800 cm) increases the water-to-fuel volume ratio from 2.04 to 2.25. The resulting improvement in uranium utilization is about 2.6%. If the clad thickness of the fuel rod containing 0.315 in. (0.800 cm) pellets is reduced from 25 mils (0.635 mm) to 22.5 mils (0.572 mm), with the gap thickness kept at a constant value of 3.5 mils (0.089 mm), the attainable benefit can be increased to about 4% as indicated by Point G. This 4% improvement in uranium utilization can be viewed as the combined result of two contributions: that due to the change in water-to-fuel ratio, approximately 3% from the reference

Figure 5.4-1

CHANGE IN URANIUM UTILIZATION FOR VARIOUS CHANGES IN LATTICE
PARAMETERS WHICH INCREASE THE WATER - TO - FUEL RATIO

Figure 5.4-2
EFFECT ON PELLETT DIAMETER AND CLAD THICKNESS REDUCTION
ON URANIUM UTILIZATION

point to Point F), and that due to the reduced neutron absorption in the clad material (1% from Point F to Point G).

5.4.4 Selection of the Modified Lattice

As a result of analyses performed during this study, a combination of reduced pellet diameter and thinner clad thickness (Point G in Figure 5.4-2) has been selected as the improved lattice design. These lattice modifications were adopted for the composite improvement design based on a balanced consideration of resource utilization benefit and fuel performance limitations.

The geometrical data of this modified lattice is compared with that of the standard lattice in Table 5.4-1. The impact on core characteristics and fuel performance of this new lattice design is discussed in more detail in the next section.

5.4.5 Impact on Core Characteristics and Fuel Performance

5.4.5.1 Moderator Temperature Coefficient

An increase in water-to-fuel ratio leads to a positive increase in the moderator temperature coefficient (MTC). Consequently, the allowable increase in water-to-fuel ratio is limited. The positive increase in the MTC arises primarily because of a more thermal spectrum which reduces changes in neutron thermalization due to changes in temperature. In addition, as the spectrum becomes more thermal, less neutrons leak from the core, thereby contributing to the positive increase in the MTC. Two-dimensional core calculations performed on different fuel designs indicate that a change of 0.6 - 0.8 in the water-to-fuel volume ratio would result in an increase in the MTC by $1.0 \times 10^{-4} \Delta\rho/^{\circ}\text{F}$ ($1.8 \times 10^{-4} \Delta\rho/^{\circ}\text{C}$). Since the MTC at BOC for the reference five-batch extended-burnup core was calculated to be $-1.67 \times 10^{-4} \Delta\rho/^{\circ}\text{F}$ ($-3.01 \times 10^{-4} \Delta\rho/^{\circ}\text{C}$), the estimated MTC at BOC for the modified lattice ($W/F = 2.287$) is therefore $-1.47 \times 10^{-4} \Delta\rho/^{\circ}\text{F}$ ($-2.65 \times 10^{-4} \Delta\rho/^{\circ}\text{C}$), which is considered to be acceptable.

Table 5.4-1

STANDARD AND MODIFIED LATTICE PHYSICAL DIMENSIONS

	<u>Standard Lattice</u>	<u>Modified Lattice</u>
Pellet O.D., in. (cm)	0.325 (0.8255)	0.315 (0.8001)
Gap Thickness, mils (mm)	3.5 (0.0889)	3.5 (0.0889)
Clad Thickness, mils (mm)	25 (0.6350)	22.5 (0.5715)
Rod O.D., in. (cm)	0.382 (0.9703)	0.367 (0.9322)
Rod Pitch, in. (cm)	0.5063(1.2860)	0.5063(1.2860)
Core Average Water-to-Fuel Ratio	2.041	2.287

5.4.5.2 Fuel Rod Internal Gas Pressure

Fuel rod internal gas pressure analyses have been performed for the modified lattice having a pellet diameter of 0.315 in. (0.800 cm) and a gap thickness of 0.0035 in. (0.089 mm). The same ground rules and power history as described in Section 5.3.4.4 were used for these analyses. However, the burnup values of the points on the power history curve (see Figure 5.3-15) were multiplied by a factor which accounts for the reduced fuel loading of the modified lattice.

The results of these analyses are shown in Figure 5.4-3 as a function of fuel design changes in gas plenum length and fuel column length in a similar way as in Figure 5.3-15. In Figure 5.4-3, the gas pressure curves of both the modified lattice and the standard lattice are plotted for comparison purposes. Compared to the standard fuel rod, the modified fuel rod with a pellet diameter of 0.315 in. (0.800 cm) has a 10% higher gas pressure at EOL when the C-E development model is used. This 10% higher gas pressure results from the higher burnup of the fuel (for the same energy extraction over the residence time) and the smaller gas plenum volume available in the modified fuel rod design.

As shown in Figure 5.4-3, the gas pressure of the modified fuel rod is about 500 psia (345 bar) higher than the system pressure; however, it can be reduced to system pressure by increasing the plenum length by about 2.5 in. (6.5 cm) or by shortening the fuel column by about 4 in. (10.2 cm).

If the no-lift-off criterion (~ 3000 psia--207 bar) is applied, the internal gas pressure of the modified fuel rod at EOL is acceptable. In this case, no design change is required. If, however, the NRC model is used to perform the gas pressure calculations, the gas pressures at EOL for the modified lattice are about 27% higher than that of the standard fuel design. Only drastic changes in fuel design (extending the gas plenum length by more than 15 in. (38.1 cm) or cutting back the fuel column by more than 30 in. (76.2 cm) could reduce the gas pressure below the no-lift-off criterion for the NRC model predictions.

Figure 5.4-3

FUEL ROD INTERNAL GAS PRESSURE FOR STANDARD AND MODIFIED LATTICE DESIGNS

Accurate prediction of fission gas release is extremely difficult at the present time because of uncertainties in fission gas release enhancement at high burnups. The lack of data inhibits the development of accurate gas-release models for high burnup and leads to significant differences in predicted internal gas pressures as indicated above when comparing predictions with the NRC model to those of the C-E developmental model. Fission gas release is currently the subject of a number of ongoing fuel-development programs; as data becomes available, it will be incorporated into the various fission-gas release models. At that time, a more definitive assessment of this potential problem area can be made.

In Section 5.5, the design option of natural uranium blankets at the top and bottom of the core is analyzed. As will be discussed in Section 5.5, an improvement in uranium utilization can be achieved at the expense of increased linear heat generation rate and burnup in the central portion of the core. In order to determine the impact on the fuel rod internal gas pressure of this higher linear heat rate and burnup in the central enriched fuel region, gas pressure results for the lead fuel rod with 6 in. (15.24 cm) natural uranium blankets at each end are also shown in Figure 5.4-3. The gas pressure of the axially blanketed lead fuel rod loaded with standard pellets is about 12% higher at EOL than that of the standard design if the C-E gas release model is used. As shown in Figure 5.4-3, the gas pressure curves calculated by the C-E developmental model for the fuel design with an axial blanket lie only slightly above those for the modified lattice.

Although the annular pellet option has not been chosen for incorporation into the improved composite design, it is still an attractive alternative since its thermal-hydraulic performance is expected to be essentially the same as that of standard design fuel and its fuel rod internal gas pressure at high burnups will be substantially less than that of standard design fuel. In order to assess the impact on fuel rod internal pressure of using annular fuel pellet designs, gas pressure in the lead fuel rod at EOL has been calculated for annular pellets with two different designs: one with a 0.09 in. (3.29 mm) diameter central hole and another with a 0.13 in. (3.30 mm) central hole. The results are shown in Table 5.4-2. Compared with the standard solid pellet design, the gas pressure in the hollow pellet fuel rod is significantly reduced, as shown by the results calculated

Table 5.4-2

LEAD ROD INTERNAL GAS PRESSURE AT EOL FOR
ANNULAR PELLET DESIGNS
(COMPARED WITH SOLID PELLET DESIGNS)

Hollow Pellet Designs

Central Hole Diameter, In.(mm)	Gas Release Model	EOL Internal Gas Pressure, psia(bar)
0.09 (2.29)	C-E	1810 (124.8)
0.09 (2.29)	NRC	4300 (296.5)
0.13 (3.30)	C-E	1530 (105.5)
0.13 (3.30)	NRC	3490 (240.6)

Solid Pellet Designs

Pellet Diameter in. (cm)	Gas Release Model	EOL Internal Gas Pressure, psia(bar)
0.325 (0.826)	C-E	2450 (168.9)
0.325 (0.826)	NRC	5550 (382.7)
0.315 (0.800)	C-E	2800 (193.1)
0.315 (0.800)	NRC	6900 (475.7)

by both C-E and NRC models. If the C-E developmental model is used, the internal gas pressure drops well below the system pressure, while the gas pressure is still above the no-lift-off criterion when the NRC model is applied. Annular pellets reduce fuel rod internal pressures for two reasons: first, the central region provides additional space to accommodate fission gas and, second, fuel temperatures are lower for annular pellets because the high temperature central region is removed. The reduction in fission gas release due to the lower temperatures together with the additional space to accommodate the fission gas which is released, proves to outweigh the higher burnup of annular fuel (the burnup of annular fuel is increased because the fuel loading is reduced; the burnup is increased in these calculations so as to maintain a constant number of megawatt-days of energy produced), resulting in significant reductions in internal rod pressures.

5.4.5.3 Thermohydraulic Performance

Based on the results of a thermohydraulic parameter study performed for another DOE-sponsored program (1), the thermohydraulic performance of the modified lattice has been evaluated and compared to that of the standard lattice design. The approach to the parametric study was to start with the System 80TM core configuration and determine the core flow rate for the various lattice configurations necessary for preserving the System 80TM overpower margin. That overpower margin was calculated to be 128.6% of 3800 MW(th) using the COSMO code(2), and the CE-1 critical heat flux correlation(3) to a DNBR of 1.195. Other assumptions used in this analysis are given in Table 5.4-3.

In Table 5.4-4, the thermohydraulic performance data of the System 80TM core containing the modified lattice is compared to that of the core containing the standard lattice. As indicated in Table 5.4-4, the use of the modified lattice with thinner fuel rods results in 5.4% larger core flow area, 3.6% smaller average flow velocity, 2.8% higher flow rate, while the total vessel pressure loss drops only slightly (1%). The required pumping power to maintain a DNBR of 1.195 at an overpower margin of 128.6% in the core containing the modified lattice is 3.6% greater than that for the standard design. Since the modified fuel design is

Table 5.4-3

ASSUMPTIONS USED IN THERMOHYDRAULIC PERFORMANCE ANALYSES

<u>Parameter</u>	<u>Value</u>
Core Power MW(th)	3800
Number of Fuel Assemblies	241
Number of Fuel Rods Per Fuel Assembly	236
Rod Array	16 x 16
Core Inlet Temperature, °F (°C)	565 (296)
Vessel Flow Rate for the System 80 Base Case (0.382 in. O.D., 0.506 in. Rod Pitch), lb/hr (0.970 cm O.D. 1.285 cm Rod Pitch), kg/s	164 x 10 ⁶ 207 x 10 ⁶
Core Bypass Leakage, %	4
Heated Length, ft (cm)	12.5 (381)
Rod Radial Peak P_1	1.55
Rod Peak to Bundle Average Peak	1.07
Axial Power Distribution Peak to Average Value at 85% from the Inlet	1.26
Standard Spacer Grid Loss Coefficient K	0.45
Fraction of Heat Generated in the Fuel Rod, %	97.5
Engineering Heat Flux Factor	1.03
Peak Rod to Four Pin Average in a Matrix Subchannel	1.03
Nuclear Enthalpy Rise Factor	1.05

Table 5.4-4

COMPARISON OF THERMOHYDRAULIC PERFORMANCE DATA
MODIFIED LATTICE vs. STANDARD LATTICE

	Standard Lattice Clad O.D. = 0.382 in. (0.9703 cm)	Modified Lattice Clad O.D. = 0.367 in. (0.9322 cm)	Change %
Core Flow Area, ft ² (m ²)	60.4 (5.61)	63.7 (5.92)	+5.4
Core Average Flow Velocity, ft/sec (m/s)	16.7 (5.09)	16.1 (4.91)	-3.6
Primary Flow Rate, lb/hr (kg/s)	164 x 10 ⁶ (207 x 10 ⁶)	168.6 x 10 ⁶ (213 x 10 ⁶)	+2.8
Pumping Power, MWe	6.34	6.57	+3.6
Vessel Pressure Loss, psi (bar)	55.56 (3.831)	55 (3.792)	-1
Quality at Minimum DNBR, %	10.5	8.72	-1.8

anticipated to be retrofitted into the existing standard plant system, the pumping power cannot be raised without replacing the coolant pumps. Under these circumstances, the overpower margin would suffer a 3 to 4% penalty. However, because of the conservative assumption of the overpower margin (128.6%) for the standard plant design, a reduction in that margin by 3 to 4% may not have a significant impact on the thermohydraulic performance of the core. Nevertheless, a detailed analysis is necessary to confirm the feasibility of the use of the modified lattice in the System 80TM core.

5.4.5.4 Other Fuel Performance Considerations

Reducing the clad thickness would influence clad creepdown. In the modified fuel rod design, the clad thickness is reduced from 0.025 in. (0.635 mm) to 0.0225 in. (0.572 mm) (10%), resulting in a clad-thickness-to-outside-diameter ratio of 0.061. This number is viewed as an index related to the mechanical resistance of the clad to creepdown. Since 0.061 still represents a conservative number for the clad thickness design, the modified fuel rod may provide an acceptable degree of resistance to clad creepdown.

The impact on bow-induced channel closure of the use of thinner pellet and clad in the modified fuel design irradiated to high burnup has been examined. Decreasing the pellet (and the clad) outside diameter by 10 mils (0.254 mm) while maintaining clad thickness constant increases channel closure by about 1% because of the less stiff clad cross section. A further reduction in clad stiffness brought about by reducing the clad thickness by 2.5 mils (0.0635 mm) increases the channel closure by an additional 10%. However, the total increase in channel closure of 11% can be offset by the 11% increase in the channel spacing achieved with the use of the smaller diameter rods.

The water-side clad corrosion of the modified fuel rod (pellet diameter 0.800 cm, gap thickness 0.089 mm, clad thickness 0.572 mm as specified in Table 5.4-1) may be slightly more severe than that of the standard fuel design because of higher clad temperature resulting from higher heat flux. To quantify the deterioration of the external clad corrosion, however, further operational data of extended burnup fuel has to be established.

5.4.6 Economic Assessment

Fuel cycle cost calculations have been performed for the five-batch, extended-burnup equilibrium core for three different fuel designs.

- standard System 80TM fuel design
- modified fuel design with thinner rods (cf. Table 5.4-1 for specifications)
- reduced effective density fuel (the fuel has the standard geometry but a 0.905 relative density to simulate the density effect of annular pellets with a central hole of 0.10 in (2.54 mm) diameter.

The relevant fuel cycle data used for the cost calculation for these three different fuel designs is listed in Table 5.4-5. The feed enrichments for the modified design (smaller pellet diameter and thinner clad) and the design with the reduced effective density are so determined that the same cycle length as that of the standard design can be achieved. The economic assumptions used for this analysis were the same as those shown in Table 5.3-16. However, the unit fabrication costs (\$/kg HM) for the modified thinner rod fuel design and for the reduced density fuel design were adjusted so that the costs per batch remain the same as for the standard fuel design. Since the reduced zircaloy demand for the fabrication of the modified fuel rods (i.e., thinner clad) was not accounted for in the cost calculation, this assumption of constant batch fabrication cost should be regarded as conservative.

The results of these fuel cycle cost calculations are listed in Table 5.4-6. By using near-term cost assumptions, the fuel cycle cost of modified fuel design is about 3% lower than that of the standard fuel design, while the cost benefit of the reduced density fuel design relative to the standard fuel design is 1.7%. If the long-term cost assumptions are used for the calculation, the relative cost benefits of the modified design and reduced density fuel are 3.4% and 2.3%, respectively.

Table 5.4-5

FUEL CYCLE DATA FOR COST CALCULATIONS

Fuel Design	Feed Enrichment (wt%)	Batch Heavy Metal Loading (kg)	Discharge Burnup (MWd/t)
Standard	4.30	20,570	50,608
Modified Lattice	4.40	19,322	53,890
Reduced Effective Density	4.60	18,614	55,928

Table 5.4-6

FUEL CYCLE COSTS FOR FIVE-BATCH, EXTENDED-BURNUP FUEL CYCLES
Equilibrium Core (mills/kWh)^a

	Fuel Design	UF ₆	Separative Work	Fabrication	Spent Fuel Disposal	Indirect Charges	Total Fuel Cycle Cost
96-5	Standard	2.17 (5.30)	1.80 (1.80)	0.34 (0.30)	0.31 (0.31)	1.15 (2.02)	5.77 (9.73)
	Modified Lattice	2.09 (5.10)	1.74 (1.74)	0.34 (0.30)	0.31 (0.31)	1.12 (1.95)	5.60 (9.40)
	Reduced Density	2.11 (5.14)	1.79 (1.79)	0.34 (0.30)	0.31 (0.31)	1.13 (1.97)	5.68 (9.51)

^aValues without parentheses are for near term cost assumptions, while those with parentheses are for the long term cost assumptions.

5.4.7 Technical Status and RD&D Requirements

Fuel rods with smaller pellet diameter and/or thinner clad than the C-E System 80TM design have been irradiated in operating reactors designed and built by other reactor vendors. The results show satisfactory fuel performance up to a batch burnup of 34 GWd/t achieved after a three-cycle residence time in the core. Irradiation tests have been planned to increase the batch burnup in excess of 40 GWd/t (4 cycles) and later to even higher burnups over 50 GWd/t (5 cycles). In addition to solid pellet fuel test programs, irradiation programs of modified fuel designs with annular pellets and reduced density pellets are being proposed and studies are being conducted to assess the irradiation behavior of these advanced fuel designs. In all these programs, the following four concerns are to be addressed at high burnup, (1) pellet-clad interaction/stress corrosion cracking in fuel rods, (2) possible enhanced fission gas release resulting in high end-of-life rod internal pressures, (3) fuel assembly dimensional stability, and (4) external corrosion of cladding.

To implement the use of a modified lattice in the present C-E standard System 80TM design, a detailed nuclear analysis and thermal-hydraulic study should be conducted for the specific design chosen. Hardware experiments should also be conducted which include hydraulic flow testing and critical heat flux testing and their supporting analyses. In addition, static and dynamic structural tests should be performed to determine the assembly responses under transient and accident conditions. The in-reactor demonstration program should include design, licensing, and fabrication of the test assemblies. After that, the assemblies would be irradiated for five cycles. Poolside examination would be performed after each operating cycle. The time necessary for the completion of the whole demonstration program might be 8 to 9 years, and the cost is estimated to be 2 to 3 million dollars.

Besides the modified lattice (a combination of reduced pellet diameter and thinner clad) suggested in this study, the use of annular pellets still represents an attractive alternative for applications with extended burnup because of lower rod internal pressure and practically unchanged thermohydraulic performance. Again, extensive R&D and demonstration programs are required in this case before the concept can be adopted for widespread use. It may be prudent to initiate the annular pellet fuel programs on time to make this option available if the thermohydraulic performance and fission gas pressure impose real limitations on the use of a modified lattice with thinner fuel rods.

5.4.8 Likelihood of Utility Adoption

Compared to the potential improvement in fuel utilization and fuel economics achieved by extended burnup with improved fuel management (12-15%, depending on the specific fuel management scheme used), the lattice modification offers only moderate benefit in uranium utilization ($\leq 4\%$) and minor reduction in the fuel cycle costs ($\sim 3\%$) which result in negligible savings ($< 1\%$) in the power generating costs. While the implementation of the extended burnup with improved fuel management does not require new reactor designs, the incorporation of the modified lattice in the existing reactor system may necessitate some hardware changes, therefore involving a certain amount of risk. It is anticipated that the modified lattice, which offers better uranium utilization, will eventually be adopted by the utilities, but only after the R&D efforts have been successfully completed and the in-reactor demonstration programs have been carried out.

To initiate these R&D efforts and demonstration programs, some governmental support is considered necessary, although pressure resulting from market competition may also prompt fuel manufacturers to consider shifting toward advanced fuel designs which show better fuel utilization and fuel economy. These efforts are relatively expensive and have only indirect benefits to the fuel manufacturers. Thus, there is a natural reluctance to finance and conduct these programs on their own. The DOE programs which have recently been initiated with a number of utility/fuel manufacturer teams may provide sufficient governmental support to satisfy these requirements.

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5.5 AXIAL BLANKETS

5.5.1 Introduction

Upon examining the axial profile of discharge exposure for a typical PWR, it is apparent that the fuel burnup falls off rapidly near the reflector boundaries. It can therefore be concluded that the fuel near the edge of the active core is under-utilized. This suggests that it may be possible to improve the overall resource utilization of a PWR by replacing the enriched fuel pellets at the ends (i.e., at the top and bottom) of the fuel rods with fertile blanket pellets composed of natural or depleted uranium. It is envisioned that this method can improve fuel resource utilization by decreasing the core leakage, increasing fertile conversion, and increasing the enriched fuel discharge exposure without significantly impairing fuel cycle performance or safety margins.

Any potential savings of resource requirements by the introduction of axial blankets depends not only on the properties of the axial blankets but also on the specific fuel cycle being analyzed. In the preliminary phase of this study, analyses of the axial blanket option were performed for a PWR employing conventional fuel management (i.e., three-batch, OUT-IN fuel management with feed enrichment of 3.3 wt% and a discharge exposure of 35 MWd/kg). In order that this evaluation of the axial blanket option be consistent with the broader examination of overall PWR once-through cycle resource utilization, analyses were also performed on the use of axial blankets for a PWR employing a five-batch, extended burnup fuel management scheme. For this scheme, the fuel enrichments and discharge exposure were 4.3 wt% and 50.6 MWd/kg, respectively. In all these analyses, a standard System 80TM core design, which has a 381 cm axial height and 241 assemblies was adopted.

5.5.2 Available Options

Two types (options) of axial geometries were examined as part of this study. The first option, which would be backfittable into existing PWRs, preserved the overall length of the fuel assembly (i.e., enriched fuel pellets at the top and bottom of the fuel rod were replaced with blanket pellets). The

second option, which would not be backfittable into current design PWRs, kept the enriched fuel length constant at 381 cm and concatenated blanket material at either end. Figure 5.5-1 shows the axial geometry of the reference case along with options one and two. Since the first option is considered a potential near-term improvement option while the second option is only applicable to new reactor designs, the first option was considered in greater detail than the second. In addition to the alternate geometries, there is a practical choice of two blanket materials (discounting more exotic materials such as Th-232 and hard to fabricate materials such as previously irradiated fuel): natural uranium and depleted uranium (with an assumed U-235 enrichment of 0.2 wt%). The benefits and drawbacks of each of these blanket materials will be discussed in detail in the sections which follow.

There is a final consideration in designing fuel with axial blankets--the length of the blanket to be used. After performing several survey calculations, it appeared that a blanket length of less than 10 cm (note that this study always assumed equal blanket lengths on the top and bottom of the core) resulted in negligible consequences. Likewise, a blanket length greater than 30 cm reduced the enriched fuel length excessively (for option one), resulting in generally unacceptable axial power peaking.

5.5.3 Method of Analysis

In order to determine the impact of the various axial blanket options on PWR fuel resource requirements and fuel cycle costs, a one-dimensional axial PDQ model was employed. This model had 75 axial mesh intervals with each mesh spanning about 5 cm. The model followed an average equilibrium batch from first cycle insertion through final cycle discharge. Previous two-dimensional (x-y) fuel management calculations provided batch buckling values to represent radial neutron leakage and soluble boron concentrations as a function of fuel exposure. Using consistent core operating environmental data, such as this, helped ensure that the physics model more accurately represented the behavior of the fuel in an actual reactor. Criticality was measured by

Figure 5.5-1
GEOMETRY OF AXIAL BLANKET STUDY

averaging the reactivity of the batch at each of the different end-of-cycle (EOC) exposures. This average EOC reactivity was then compared with that obtained from more detailed core analyses.

The actual calculative techniques employed were slightly different between the standard fuel management study and the extended-burnup fuel management study. The results, however, are completely consistent with each other when viewed in final form. The standard fuel-management study employed a reference enrichment of 3.3 wt% for the enriched portion of the blanketed cores. A depletion was then performed over the reference energy extraction. This energy extraction was then adjusted according to the end-of-cycle criticality, and a resource requirement (ST U_3O_8 /GWe) was thereby derived.

A consistent result for the extended-burnup fuel management was obtained in a somewhat different manner. Again, a reference case was analyzed in order to provide a standard from which criticality and resource savings could be measured. For the study of specific blanket options, the mesh spacing was altered to accommodate the exact blanket length, and appropriate blanket composition was assigned to different axial regions. Constant energy extraction from depletions was assured for each case by the use of equivalent power histories and operating times. However, instead of adjusting the energy extraction to a constant feed enrichment as was done for the standard fuel management cases, feed enrichment was adjusted to assure the meeting of the reference end-of-cycle criticality condition at constant energy extraction. Again, savings of resource requirements were then calculated.

5.5.4 Impact of Axial Blankets on Uranium Requirements

Results from this study can be quantified into three major areas:

1. U_3O_8 requirements as a function of blanket properties (i.e., composition and length),

2. Separative work requirement as a function of blanket properties, and
3. Axial power peak changes on a function of blanket properties.

Items 1 and 2 are the main topics of the present section; however, they are also related to economic incentives and therefore the commercialization potential for employing this improvement option. These latter aspects will be addressed in Sections 5.5.6 and 5.5.8. Item 3, the axial power peaks as a function of blanket properties, is a technical safety-related issue which will be discussed further in Section 5.5.5.

5.5.4.1 Results With Standard Fuel Management

The standard fuel management cases investigated as part of the axial blanket study are (1) a reference case with fuel rods 347 cm long and enriched to 3.3 wt% , (2) a "long fuel rod" case where 15 cm of natural uranium are added to the top and bottom, and (3) a "short fuel stack" case where 15 cm of enriched uranium at the top and bottom are replaced with natural uranium.

This portion of the study was performed primarily as a reference for the standard PWR fuel management and therefore was somewhat brief. The main emphasis of the axial blanket study was on the context of extended-burnup fuel management schemes. These will be discussed in detail in the next section.

Each case was depleted with a one-dimensional model of the core over a time interval which simulates three cycles. Results are shown in Table 5.5-1. Column 1 of Table 5.5-1 shows the burnup accumulated in the active core and in the blanket for each case for a fixed number of full-power hours. This column shows that after an equivalent number of full-power hours, the average burnup (MWd/t) in Case 2 is lower due to the addition of fuel, as expected.

The energy extracted from each fuel region is shown in Column 3; This assumes that each case is depleted for an equivalent number of full-power

Table 5.5-1

ENERGY EXTRACTION AND U_3O_8 REQUIREMENT COMPARISONS FOR
VARIOUS AXIAL BLANKET DESIGNS, STANDARD FUEL MANAGEMENT

Case	(1)		(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)
	Burnup at an Equivalent Number of Full Power Hours (MWd/t)		Mass of one Ass'y (tonnes)	Energy (MWd)	Energy ^a Adjustment (MWd)	Distrib- uted Ener- gy*Adjust- ment (MWd)	Adjust- ed Energy* (MWd)	Relative Change in Energy	U_3O_8 Requirement (tonnes of Natural U)	Relative Change in Uranium	Net Percent Utiliza- tion Increase
5-105 1. (Ref- erence)	35,120		.4268	14,990	---	---	14,990		2.5891	---	
2. (Long Fuel Rod)	Fuel	34,500	.4268	14,725	+299	+293	15,018	+0.2%	2.5891	---	
	Blanket	7,228	.0375	271		+ 6	277	+1.8%	.0375	+1.4%	
	Average	32,298	.4643	14,996			15,295	+2.0%	2.6266	+1.4%	+0.6
3. (Short Active Fuel Stack)	Fuel	37,690	.3893	14,673	-504	-493	14,180	-5.4%	2.3617	-8.8%	
	Blanket	8,403	.0375	315		- 11	304	+2.0%	.0375	+1.5%	
	Average	35,120	.4268	14,988			14,484	-3.4%	2.3992	-7.3%	+4.2%

a. Energy adjustment to an equivalent EOC reactivity as the reference case

hours. The addition of the blanket in Case 2 increases the reactivity throughout the depletion relative to the reference case. The energy adjustment for equal EOC reactivity is shown in Column 4 and Column 5. The total energy extractable from each region for each case is shown in Column 6. This data shows that the addition of an axial blanket (Case 2) allows a slightly increased energy extraction from the active fuel (to 0.2%). The increased energy extraction in the blanket itself is 1.8% so that the total increase is 2.0%. The addition of the axial blanket, however, requires a 1.4% increase in the uranium requirements. The net utilization improvement is, therefore, 0.6%.

Case 3 (short active fuel stack) shows a 5.4% decrease in the energy extraction from the active core. This is offset somewhat by a 2.0% energy production from the blanket, yielding a 3.4% net energy loss. However, the fuel rod contains 8.8% less enriched uranium due to the shorter active fuel length. This is mitigated by the 1.5% additional uranium for the blanket. This 7.3% net reduction in the uranium requirement is larger than the loss of the energy production, so that a 4.2% net uranium utilization benefit is achieved.

5.5.4.2 Results With Extended-Burnup Fuel Management

In order that this study be consistent with a broader examination of resource optimization, the use of the axial blanket was examined in the context of the previously described five-batch, extended-burnup fuel management scheme. The fuel design used for this phase of the study was that of the System 80TM PWR (i.e., 381 cm nominal length).

Table 5.5-2 summarizes the results of the extended-burnup blanket study for the constant fuel rod length option in terms of required enrichment of the active fuel as a function of blanket length and blanket material. The resource requirement ramifications of this data can be more easily seen on Figures 5.5-2 and 5.5-3. These figures show the U₃O₈ savings and separative work savings as a function of blanket length and blanket material. It can be seen that U₃O₈ savings achieve a maximum of about 3% at a blanket length of 25-30 cm regardless of the choice of blanket materials, while the separative work requirements are reduced by about 2% for a natural uranium blanket and about 0.5% for a depleted uranium blanket with a blanket length of 15 cm.

Table 5.5-2

ENRICHMENT REQUIREMENTS^a FOR
CONSTANT LENGTH FUEL ROD WITH AXIAL BLANKET

Blanket Length and Composition	Active Fuel Enrichment (wt%)	U ₃ O ₈ ^b Short Tons/GWe	Separative Work 10 ³ SWU/GWe
Base Case (no blanket)	4.30	170.78	118.38
10.2 cm (4 in.) natural ^c	4.44	168.37	117.10
15.2 cm (6 in.) natural	4.52	167.21	116.51
20.3 cm (8 in.) natural	4.61	166.44	116.28
10.2 cm (4 in.) depleted	4.67	168.21	117.99
15.2 cm (6 in.) depleted	4.56	166.94	117.79
20.3 cm (8 in.) depleted	4.66	166.07	118.00
30.5 cm (12 in.) depleted	4.93	165.45	119.43

a. Extended-burnup fuel management

b. Total Requirement: active fuel region plus blanket

c. 10.2 cm (4 in.) natural represents a 10.2 cm (4 in.) blanket composed of natural uranium at each end of the fuel rods.

Figure 5.5-2

U_3O_8 SAVINGS vs AXIAL BLANKET LENGTH
(FIVE-BATCH, EXTENDED BURNUP CORE)

Figure 5.5-3
SEPARATIVE WORK SAVINGS vs AXIAL BLANKET LENGTH
(FIVE-BATCH, EXTENDED BURNUP CORE)

The use of depleted uranium over natural uranium yields a slight savings in U_3O_8 requirements at the expense of reduced separative work savings. The reasons for this can be explained as follows. Unlike natural uranium, depleted uranium has no U_3O_8 requirement in and of itself. This more than compensates for any uranium savings in the enriched fuel due to the slight enrichment decrease attendant with the choice of a natural uranium blanket. However, neither blanket material involves any separative work. Therefore, the slight decrease in active fuel enrichment which accompanies the use of a natural uranium blanket (as compared to a depleted uranium blanket) results in an increased savings in separative work.

A savings of about 2% is obtained in both U_3O_8 and separative work requirements by the use of a 15 cm (6 in.) natural uranium blanket. This savings is smaller than that which was reported in Section 5.5.4.1 for a PWR employing a standard three-batch, 35 MWd/kg discharge exposure fuel management. This difference in uranium savings is easily explained by the fact that the axial blankets replace the under-utilized (low-burnup) fuel pellets at the ends of the rods. With extended-burnup fuel management schemes, these pellets achieve higher exposures and hence better utilization than similar pellets with standard-burnup fuel management schemes. Thus, there is less incentive to replace these end fuel pellets in extended-burnup schemes than in standard burnup schemes.

To quantify the fuel resource savings associated with the use of the second axial blanket option (i.e., adding blanket material at the ends of the nominal fuel rod), a single case was analyzed. This option was not studied in great detail because, as previously described, it requires a major core redesign. Also, the functional dependence of resource savings on blanket length was not expected to be as great as the first option since there is no attendant shortening of the enriched fuel region. In this case, the enriched fuel length was kept constant at 381 cm while concatenating unenriched blanket material on each end of the fuel rod.

The U_3O_8 and separative work savings due to a 15.2 cm (6 in.) depleted uranium blanket on either side of a 381 cm enriched length are 2% and 2.7%, respectively. Although savings of this size are not considered sufficient to justify the hardware modifications required to accommodate the increased length of the uranium rods, the reduced power peaking which results from the lengthening of fuel rods may make possible a slight increase in power rating, somewhat mitigating the expense of such hardware modifications. The longer length of these rods does, however, preclude general use of this option in existing PWRs.

5.5.5 Impact on Power Peaking and Peak Pellet Burnup

As previously stated, any potential resource benefits resulting from the use of axial blankets (with constant fuel rod length and power level) may be limited by core power peaking and peak burnup considerations. Table 5.5-3 presents estimates of the axial peaking factors and the change in maximum linear heat rate for the standard-burnup fuel management. For the long-fuel-rod case, the linear heat rate is expected to decrease by about 2% due to the addition on the blanket, whereas the short-fuel-stack case causes an increase in linear heat rate of about 5%. Thus, the 4.2% benefit obtained by the use of axial blankets with standard-burnup fuel management (short-fuel-stack case) is achieved at the expense of increased linear heat rate and burnup in the central portion of the core, which may lead to more limiting fuel performance conditions.

Table 5.5-4 gives the estimate of axial power peaks for beginning of equilibrium cycle of a five-batch, extended burnup core as a function of blanket length and material. Figures 5.5-4 and 5.5-5 show beginning of initial core and beginning of equilibrium cycle power distributions, respectively, for the reference case, a depleted uranium blanket of 15.2 cm (6 in.) and a natural uranium blanket of the same length. It can be seen that a 15.2 cm (6 in.) blanket may increase initial core axial power peaks by as much as 10% (i.e., an increase in the relative power peak of 0.10) for natural uranium blanket and 12% for depleted uranium blankets. The increase in axial power peaking as a consequence of employing axial blankets is only about half these initial core values in equilibrium. The lower increase in peak when using natural uranium is due to the higher U-235 content of the blanket and resulting higher power there and in the enriched fuel adjacent to the blanket.

Table 5.5-3

ESTIMATED CHANGES IN THE
AXIAL PEAKS AND THE
MAXIMUM LINEAR HEAT RATE
DUE TO THE ADDITION
OF AN AXIAL BLANKET

(Three-Batch, Standard-Burnup Fuel Management)

<u>Case</u>	<u>Axial Peak at BOC</u>	<u>Fractional Core Volume Change</u>	<u>Percent Linear Heat Rate Change</u>
1. Reference	1.104	-----	-----
2. Long Fuel Rod	1.170	.0878	-2.6%
3. Short Active Fuel Stack	1.163	0.0	+5.3%

Table 5.5-4

AXIAL PEAKING FACTORS^a FOR VARIOUS BLANKET LENGTHS
(Equilibrium Cycle)^b

Material	Blanket Length, cm (in.)			
	10.2 (4)	15.2 (6)	20.3 (8)	30.5 (12)
Natural Uranium	2.7%	4.8%	8.2%	---
Depleted Uranium	3.5%	5.1%	9.1%	15.9%

- a. Additional peaking beyond that of unblanketed core.
b. Initial core axial peaking about twice these values

Figure 5.5-4

NORMALIZED AXIAL POWER DISTRIBUTION vs RELATIVE AXIAL HEIGHT
FIVE-BATCH, EXTENDED-BURNUP FUEL MANAGEMENT
BEGINNING OF FIRST CORE

5-115

Figure 5.5-5
NORMALIZED AXIAL POWER DISTRIBUTION vs RELATIVE AXIAL HEIGHT
FIVE-BATCH, EXTENDED-BURNUP FUEL MANAGEMENT
BEGINNING OF EQUILIBRIUM CYCLE

An additional consideration in this study which would give preference to a natural (as opposed to depleted) uranium blanket is that of fission gas internal rod pressure. The lower power peaks associated with the use of a natural uranium blanket result in a reduced peak and average discharge burnup in the active fuel region. This reduced discharge burnup gives rise to a smaller fission gas release. Discharge burnup distributions for the core with no blanket, a natural uranium blanket and a depleted uranium blanket are shown in Figure 5.5-6. The cores with blankets have higher active core burnups than the unblanketed core. The higher active core power and discharge burnup will aggravate the already potentially troublesome problem of fission gas internal rod pressure. The effect on fission gas release of employing axial blankets is discussed briefly in Section 5.4.4.

5.5.6 Economic Assessment

The primary incentive for the use of axial blankets is fuel resource conservation. The degree to which resource conservation can be attained, however, is typically constrained by operational considerations (in this case by thermal margins and technical specifications).

From a commercialization standpoint, the prime motivator of implementing a resource efficient improvement option is usually the overall economics of the fuel cycle when the option is included. As described in the previous section, any significant increase in the total fuel rod length will be associated with the extreme expense of a major core redesign. A study of capital cost changes to implement this blanket option is beyond the scope of this report and is not undertaken. However, for the option in which fuel rods of constant length were employed, fuel cycle cost calculations were performed to assess the economic incentive to deploy this backfittable blanket option. The economic assumptions used for these fuel cycle cost calculations are those adopted by NASAP and are felt to be representative of industry values. These assumptions are shown on Table 5.4-16. The results of these analyses are in graphic form in Figure 5.5-7, where the batch fuel costs are shown as a function of blanket length and material. It can be seen from this figure that the fuel cycle costs decrease to a minimum (corresponding to approximately a 2% savings) at about a 20-25 cm blanket length with a slight economic advantage in favor of natural (as opposed to depleted) uranium. This slight advantage is due to the fact that the separative work savings economically outweigh any U_3O_8 savings which may be obtained through the use of depleted uranium. It

Figure 5.5-6
AXIAL BURNUP DISTRIBUTION
FIVE-BATCH, EXTENDED BURNUP FUEL MANAGEMENT

Figure 5.5-7
FUEL CYCLE COST vs BLANKET LENGTH
(\$40/LB U_3O_8)

is of interest to note that the safety considerations of peak linear heat rate and pellet burnup also favored the use of natural uranium blankets over depleted uranium blankets. If thermal margins constrained the natural uranium blanket length to about 15 cm, a fuel cycle cost savings of 1.5% would be obtained.

5.5.7 Technical Status and RD&D Requirements

As mentioned in the previous sections, the use of axial blankets in standard length fuel increases the axial power peaking factor and the peak fuel pellet burnup compared to that of unblanketed fuel. The major issues that need to be investigated are therefore associated with these two effects. In particular, the impact on core thermal margins, the susceptibility of the fuel to pellet-clad interaction (PCI), and the buildup of fuel rod internal pressure as a result of fission gas release will need to be assessed for the higher power densities and peak burnups which accompany the use of axial blankets. These issues are quite similar to those associated with the use of extended-burnup fuel, although they may be influenced somewhat differently by the more severe axial gradients that occur with blanketed fuel.

An RD&D program has already been initiated by the U.S. Department of Energy (DOE) in cooperation with The Sacramento Municipal Utility District (SMUD). Four lead test assemblies will be fabricated with axial blankets and loaded into the Rancho Seco reactor for irradiation in Cycle No. 5. The performance of these assemblies will be tracked through the use of in-core detectors and the reactor's on-line computer. The data provided from this irradiation test will then be used to confirm the analytical models and methods used in the design and licensing of the blanketed assemblies. As a further demonstration of the use of axial blanketed fuel, it is presently planned to implement this concept for three full reload batches, to begin with Rancho Seco Cycle No. 6. The design of the assemblies for this phase of the program will be optimized within the present hardware constraints and reactor operating limits. Complete mechanical, thermal hydraulic, and nuclear analysis assessments will be used in the development of the fuel management and in support of licensing these reload batches.

Although the above demonstrations will provide valuable data to confirm the licensability of fuel with axial blankets, additional demonstrations will probably be necessary if this concept is to be implemented in the context of extended-burnup fuel cycles. A program similar to that mentioned above appears adequate and possibly can be combined with one of the extended-burnup demonstrations which include an advanced fuel design.

5.5.8 Likelihood of Utility Adoption

Utilities will adopt the axial blanket option if the impact on core thermal margins and fuel performance is not significantly adverse. Techniques, as yet undecided, will have to be employed to mitigate the higher local power densities that occur when blankets are added to the ends of the fuel rod. The small savings in uranium and fuel costs can not be justified if the performance of the fuel or the safety of the core is significantly compromised. The willingness of a utility to participate in the demonstration of this concept is indicative of the realization, on their part, that methods such as this will be important in the long term to maximize the energy extractable from a fixed natural resource. However, reliability of energy supply, the safe operation of their plants, and costs of generation are their primary concerns.

5.6 FULL USE OF EARLY BATCHES OF INITIAL CORE

The initial core of a reactor employing a multi-batch fuel management scheme contains several batches which are discharged after fewer irradiation cycles than a typical reload batch experiences. Although the enrichments employed are lower than that of the equilibrium batch, the reactivity and fissile content of the initial core fuel at discharge are higher than that of reload fuel irradiated to equilibrium burnups. Consequently, a potential exists to irradiate these initial core fuel assemblies for an additional cycle.

There are several ways that the initial core fuel could be used. The two methods investigated in this study are reinsertion during later cycles in the same reactor and insertion in another reactor for its first cycle of operation (fuel sharing). In either case, a portion of the fresh fuel batch is replaced with the previously irradiated initial core fuel to reduce uranium requirements.

The following sections describe this improvement option in greater detail. Section 5.6.1 describes the analysis methods employed, Sections 5.6.2, 5.6.3, and 5.6.4 describe the impact of full use of early batches on the uranium requirements, on the core power distribution and on the fuel cycle economics. The RD&D requirements and likelihood of utility adoption for this improvement option are discussed in the final two sections.

5.6.1 Analysis Methods

Several evaluations of the full use of initial core batches have been conducted in recent years in support of C-E's commercial position as a fuel manufacturer. These evaluations have normally been carried out using two-dimensional core models and standard design methodology; these have provided the reference analyses from which more extensive surveys of fuel management schemes with fuel reinsertion have been prepared. In addition, these analyses have provided general guidelines with respect to loading patterns, power sharing and safety-related parameters for cores employing this fuel management option. The present study of the full use of initial core batches relies heavily on these earlier evaluations to establish the general feasibility of this

option. Although these earlier studies were conducted for fuel management schemes which employed three or fewer in-core batches and fuel discharge exposures in the 30 MWd/kg range, it is felt that their overall trends and conclusions still apply. To particularize the full use of early batches option to the improved five-batch, extended-burnup cycle, the methodology used to perform reload fuel surveys was employed. This methodology utilizes various derivative rules to relate the particular case being investigated back to the detailed reference cases, so that it becomes practical to evaluate the effect of fuel reinsertion over a large number of operating cycles.

Calculations for reinsertion in the five-batch, extended-burnup core were performed using a C-E developed survey code. Based on enrichment and burnup history, core batch loading, cycle lengths and operating reactor biases, the code determines the necessary core-average enrichment to achieve the proposed cycle lengths. Two cases were analyzed: five-batch, extended-burnup fuel management and five-batch, extended-burnup fuel management with reinsertion of the first four batches of the initial core fuel. An OUT-IN fuel shuffling scheme was assumed for both of these fuel management strategies. From these calculations, the uranium requirements and fuel cycle costs were determined to assess the impact of reinsertion on the five-batch, extended-burnup fuel cycle.

The second method by which initial core fuel can be employed, fuel sharing, was also analyzed. Calculations were performed for the reuse of the low enrichment batch of a three-batch PWR initial core. Cases were run for both donor and recipient reactors over the initial 10 years of operation.

5.6.2 Impact on Uranium Resource Requirements

The effect of reinserting the early batches of the initial core fuel on uranium requirements was investigated for a five-batch, extended-burnup PWR. To maximize the savings, the analysis was conducted so that all of the assemblies of the initial four batches were reinserted. After the initial cycle of operation, forty-eight assemblies are loaded, eight of which originated from the initial discharge batch. In the subsequent refuelings, these eight assemblies are replaced with eight of the remaining assemblies from the initial core and the discharged fuel is replaced with the appropriate number of fresh

assemblies (40 or 48). Table 5.6-1 lists the fuel cycle data for the five-batch, extended-burnup PWR used as the basis for the reinsertion study. The transition cycle lengths in this five-batch core were chosen different from those described earlier in Section 5.3.2 to conform more closely to those used in conventional PWR practices. Instead of an extremely long first cycle and a very short second cycle, cycle lengths closer to the equilibrium cycle length were selected. Because of the survey nature of these calculations, the enrichment predicated in Table 5.6-1 differs slightly from those previously reported for the five-batch core; however, this is of little significance since the parameter of interest is the relative uranium savings which can be achieved with fuel reinsertion.

Table 5.6-2 gives the corresponding data for the same core when reinsertion is implemented. Over thirty years of reactor operation, approximately 2.4% fewer assemblies are loaded with this reinsertion scheme. However, since reinserting initial core assemblies results in a higher core-average exposure (about 350 MWd/t), slightly higher feed enrichments are required as compared to the five-batch core without reinsertion. The thirty-year resource savings due to fuel reinsertion in the five-batch, extended-burnup cycle are approximately 1% and 0.4% for uranium and separative work, respectively.

The second method for implementing this improvement option, i.e., fuel sharing, shows resource savings of similar magnitude. The fuel-sharing analysis performed for this study consists of a donor reactor with a normal first cycle enrichment which is operated to near depletion of reactivity. After completion of the first cycle, Batch A is transferred to the recipient reactor unit. Tables 5.6-3 and 5.6-4 show the reactor material inventory and cycle burnup information for the donor unit. Tables 5.6-5 and 5.6-6 show similar information for the recipient unit, assuming that the startup schedule for the recipient allows the donor unit to be burned to the end of reactivity.

The uranium utilization benefit of this fuel-sharing concept is obtained as follows:

Let L_i^D , L_i^R be the donor and the recipient fuel loadings of a batch to be discharged in Cycle i respectively (kgU).

Table 5.6-1

FUEL CYCLE DATA FOR A FIVE-BATCH CORE
WITHOUT REINSERTION

Batch	Number of Assemblies	Cycle Number		Average Discharge Burnup (MWd/t)	Fuel Loaded Uranium (kg)	Enrichment (wt%)
		In	Out			
1	48	1	1	12130	20471	1.37
2	48	1	2	20940	20471	1.77
3	48	1	3	31396	19084	2.22
4	48	1	4	41709	18765	2.56
5	49	1	5	51850	20777	3.07
6	48	2	6	49850	20471	4.65
7	48	3	7	51170	20471	4.51
8	48	4	8	50844	20471	4.20
9	48	5	9	50661	20471	4.20
10	49	6	10	50650	20898	4.20
11	48	7	11	50650	20471	4.20
12	48	8	12	50650	20471	4.20
13	48	9	13	50650	20471	4.20
14	48	10	14	50650	20471	4.20
15	49	11	15	50650	20898	4.20
16	48	12	16	50650	20471	4.20
17	48	13	17	50650	20471	4.20
18	48	14	18	50650	20471	4.20
19	48	15	19	50650	20471	4.20
20	49	16	20	50650	20898	4.20
21	48	17	21	50650	20471	4.20
22	48	18	22	50650	20471	4.20
23	48	19	23	50650	20471	4.20
24	48	20	24	50650	20471	4.20
25	49	21	25	50650	20898	4.20
26	48	22	26	50650	20471	4.20
27	48	23	27	50650	20471	4.20
28	48	24	28	50650	20471	4.20
29	48	25	29	50650	20471	4.20
30	49	26	30	50650	20898	4.20
31	48	27	31	50650	20471	4.20
32	48	28	32	50650	20471	4.20
33	48	29	33	50650	20471	4.20
34	48	30	34	50650	20471	4.20

Table 5.6-2

FUEL CYCLE DATA FOR A FIVE-BATCH CORE
WITH REINSERTION

Batch	Number of Assemblies	Cycle Number		Originally Discharged	Batch Average Discharge Burnup (MWd/t)	Fuel Loaded Uranium (kg)	Enrichment (wt%)
		In	Out				
1	48	1	1		12130	20471	1.37
2	48	1	2		20940	20471	1.77
3	48	1	3		31395	19084	2.22
4	48	1	4		41709	18765	2.56
8	49	1	5		51839	20777	3.07
6	8	2	2	1	20940	3412	1.37
7	40	2	6		49839	17059	5.52
8	8	3	3	1	20940	3412	1.37
9	48	3	7		51159	20471	4.73
10	8	4	4	1	20940	3412	1.37
11	48	4	8		50833	20471	4.06
12	8	5	5	1	20940	3412	1.37
13	48	5	9		50650	20471	3.72
14	8	6	6	1	20940	3412	1.37
15	49	6	10		50650	20898	4.00
16	8	7	7	1	20940	3412	1.37
17	40	7	11		50650	17059	4.00
18	8	8	8	2	31070	3412	1.77
19	48	8	12		50650	20471	4.82
20	8	9	9	2	31070	3412	1.77
21	48	9	13		50650	20471	4.68
22	8	10	10	2	31070	3412	1.77
23	48	10	14		50673	20471	3.62
24	8	11	11	2	31070	3412	1.77
25	49	11	15		50696	20898	3.72
26	8	12	12	2	31070	3412	1.77
27	40	12	16		50719	17059	4.97
28	8	13	13	2	31070	3412	1.77
29	48	13	17		50742	20471	4.60
30	8	14	14	3	41548	3180	2.22
31	48	14	18		50765	20471	4.68
32	8	15	15	3	41548	3180	2.22
33	48	15	19		50765	20471	3.82
34	8	16	16	3	41548	3180	2.22
35	49	16	20		50770	20898	4.00
36	8	17	17	3	41548	3180	2.22
37	40	17	21		50775	17059	3.91
38	8	18	18	3	41548	3180	2.22
39	48	18	22		50780	20471	4.95
40	8	19	19	3	41548	3180	2.22
41	48	19	23		50785	20471	4.70
42	8	20	20	4	41548	3128	2.56
43	48	20	24		50785	20471	3.83
44	8	21	21	4	41548	3128	2.56
45	49	21	25		50785	20898	3.77
46	8	22	22	4	41548	3128	2.56
47	40	22	26		50757	17059	4.81
48	8	23	23	4	41548	3128	2.56
49	48	23	27		50729	20471	4.78
50	8	24	24	4	41548	3128	2.56
51	48	24	28		50701	20471	4.53
52	8	25	25	4	41548	3128	2.56
53	48	25	29		50673	20471	3.65
54	56	26	30		50650	23879	3.58
55	40	27	31		50650	17509	4.81
56	48	28	32		50650	20471	4.78
57	48	29	33		50650	20471	4.53
58	48	30	34		50650	20471	3.63

Table 5.6-3

FUEL SHARING STUDY
FUEL CYCLE DATA TO DONOR UNIT

BATCH NUMBER	NO. OF ASSYS.	CYCLE NUMBER		AVERAGE DISCHARGE BURNUP MWd/t	FUEL LOADED	
		IN	OUT		URANIUM kg	ENRICH. Wt% U-235
1	80	1	1	14448	34119	1.83
2	1	1	2	23341	426	1.83
3	75	1	2	24582	29817	2.49
4	13	1	3	31507	5168	2.49
5	32	1	3	31507	12722	2.95
6	28	1	3	31507	11942	2.95
7	12	1	4	35055	5118	2.95
8	56	2	4	30203	23893	3.45
9	24	2	5	35631	10236	3.45
10	44	3	5	31244	18765	3.00
11	32	3	6	36527	13648	3.00
12	32	4	6	29558	13648	3.00
13	41	4	7	35700	17486	3.00
14	24	5	7	29526	10236	3.25
15	44	5	8	35729	18765	3.25
16	24	6	8	29593	10236	3.27
17	44	6	9	36484	18765	3.27
18	20	7	9	31954	8530	3.27
19	44	7	10	36484	18765	3.27
20	21	8	10	31954	8956	3.27
21	44	8	11	36484	18765	3.27
22	24	9	11	29433	10236	3.27
23	44	9	12	36484	18765	3.27
24	20	10	12	31954	8530	3.27
25	44	10	13	36484	18765	3.27

Table 5.6-4

FUEL SHARING STUDY
BATCHWISE BURNUPS FOR THE DONOR UNIT

BATCH NUMBER	CYCLE										AVERAGE DISCHARGE BURNUP Mwd/t
	1	2	3	4	5	6	7	8	9	10	
1	14448										14448
2	14448	8293									23341
3	15689	8393									24582
4	12004	10734	8765								31507
5	12004	10734	8765								31507
6	12004	10734	8765								31507
7	7622	10702	9098	7633							35055
8		10420	10828	9555							30803
9		7650	11075	8909	8047						35681
10			10070	10764	10410						31244
11			7476	11074	9640	8337					36527
12				9150	10631	9777					29558
13				7711	10004	9754	8227				35700
14					9821	10456	9249				29526
15					7538	10566	9352	8273			35729
16						9941	10865	8787			29593
17						7806	10868	9586	8124		36484
18							11216	10959	9779		31954
19							7806	10868	9686	8124	36484
20								11216	10959	9779	31954
21								7806	10868	9686	36434
22									10268	10104	29433
23									7806	10868	36484
24										11216	31954
25										7806	36484
<hr/>											
CYCLE BURNUP Mwd/t	13740	9780	9568	9450	9450	9450	9450	9450	9450	9450	
CYCLE TIME YEARS	1.51	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	

Table 5.6-5

FUEL SHARING STUDY

FUEL CYCLE DATA FOR RECIPIENT UNIT

BATCH NUMBER	NO. OF ASSYS.	CYCLE NUMBER		AVERAGE DISCHARGE BURNUP MWd/t	FUEL LOADED	
		IN	OUT		URANIUM kg	ENRICH. Wt%
1	80	1*	1	7752	33368	.77
2	1	1	2	19760	426	1.83
3	75	1	2	20577	29817	2.49
4	5	1	3	30366	1988	2.49
5	40	1	3	30366	15903	3.05
6	28	1	3	30366	11942	3.05
7	12	1	4	35229	5118	3.05
8	56	2	4	31317	23683	3.00
9	24	2	5	35122	10236	3.00
10	44	3	5	30792	18765	3.22
11	32	3	6	36109	13648	3.22
12	32	4	6	29756	13643	3.18
13	41	4	7	35909	17486	3.18
14	24	5	7	29932	10236	3.00
15	44	5	8	36040	18765	3.00
16	24	6	8	29593	10236	3.27
17	44	6	9	36484	18765	3.27
18	20	7	9	31954	8530	3.55
19	44	7	10	36184	18765	3.55
20	21	8	10	31954	8956	3.27
21	44	8	11	36484	18765	3.27
22	24	9	11	29433	10236	3.01
23	44	9	12	36484	18765	3.01
24	20	10	12	31954	8530	3.35
25	44	10	13	36484	18765	3.35

*Batch 1 initially utilized in donor unit where it received burnup of 14448 MWd/t

Table 5.6-6
FUEL SHARING STUDY
BATCHWISE BURNUPS FOR THE RECIPIENT UNIT

BATCH NUMBER	CYCLE										AVERAGE DISCHARGE BURNUP MWd/t
	1	2	3	4	5	6	7	8	9	10	
1	7752										7752
2	9703	10057									19760
3	10520	10057									20577
4	9977	12144	8245								30365
5	9977	12144	8245								30366
6	9977	12144	8245								30366
7	7137	12103	8553	7431							35229
8		11783	10232	9302							31317
9		8651	10418	8674	8379						36122
10			9473	10479	10840						30792
11			7033	10781	10038	8337					36189
12				8908	11070	9777					29756
13				7507	10421	9754	8227				35909
14					10227	10456	9249				29932
15					7849	10566	9352	8273			36040
16						9941	10855	8787			29593
17						7806	10868	9686	8124		36484
18							11216	10959	9779		31954
19							7806	10868	9686	8124	36484
20								11216	10959	9779	31954
21								7806	10868	9686	36484
22									10268	10104	29433
23									7806	10868	36484
24										11216	31954
25										7806	36484
CYCLE BURNUP MWd/t	9228	11060	9012	9200	9840	9450	9450	9450	9450	9450	
CYCLE TIME YEARS	1.02	1.17	.94	.97	1.04	1.00	1.00	1.00	1.00	1.00	

B_i^D, B_i^R be the respective average discharge burnup (MWd/t).

E_i^D, E_i^R be the respective feed enrichment. (wt% U-235).

For the donor unit, the cumulative uranium utilization after n fuel batches are discharged is

$$U_n^D = \frac{\sum_{i=1}^n L_i^D B_i^D}{\sum_{i=1}^n L_i^D \left(\frac{E_i^D - 0.2}{0.511} \right)} \left(\frac{\text{MWd}}{\text{TeM}} \right) \quad (5-1)$$

The above equation assumes a tails assay of 0.2 wt% U-235.

For the recipient unit, no uranium needs to be mined for the first batch. The uranium utilization is therefore

$$U_n^R = \frac{L_1^D \Delta B_1^R + \sum_{i=2}^n L_i^R B_i^R}{\sum_{i=2}^n L_i^R \left(\frac{E_i^R - 0.2}{0.511} \right)} \left(\frac{\text{MWd}}{\text{TeM}} \right) \quad (5-2)$$

where ΔB_1^R is the additional burnup of the shared batch which is accumulated in the recipient unit.

The relative uranium utilization benefit can then be written as

$$R_n = \frac{U_n^D + U_n^R}{2 U_n^D} = 1/2 \left(1 + \frac{U_n^R}{U_n^D} \right) \quad (5-3)$$

This ratio is presented in Figure 5.6-1 using the data in Tables 5.6-3 through 5.6-6. The relative uranium savings is about 50% for the first discharged batch, but it decreases to 1.5% after twelve cycles.

Figure 5.6-1

FUEL SHARING STUDY - URANIUM UTILIZATION OF FUEL DISCHARGED AT THE END OF A GIVEN CYCLE (MWd/STU₃O₈) FOR A DONOR AND A RECIPIENT REACTOR, WHERE THE RECIPIENT REACTOR USES BATCH A OF THE DONOR REACTOR, COMPARED TO THE URANIUM UTILIZATION WITHOUT FUEL SHARING

5.6.3 Impact on Core Power Distribution

The spatial calculations for a core with fractional batch refueling ($n = 2.47$) indicate that acceptable power distributions can be obtained with fuel insertion. The maximum planar pin-peaking value is usually no larger than that obtained without reinsertion. For the five-batch, extended-burnup core, it is expected that the maximum pin peaking value will remain approximately the same with and without reinsertion if the number of fresh assemblies loaded remains a constant (48). Every fifth year, however, a smaller number of fresh assemblies (40) is loaded into the core. At these times, the maximum pin-peaking factor will probably increase slightly since there will be fewer fresh assemblies over which to distribute the power.

5.6.4 Economic Assessment

The economic evaluation of the full use of early batches of the initial core was performed with the fuel cycle cost code EVALU84 described in Section 5.3. Thirty-year levelized fuel cycle costs were calculated for the five-batch, extended-burnup core. Each of the fuel cycle cost components is slightly lower in the core with reinsertion. Due to the fewer number of fresh assemblies loaded and discharged, the charges for U_3O_8 , separative work, fabrication, and fuel disposal decrease along with the indirect charges. The total savings for fuel reinsertion amounts to approximately 0.03 mills/kWh, which is about 0.5% of the fuel cycle costs.

5.6.5 Technical Status and RD&D Requirements

The RD&D requirements for the full use of early batches improvement option are minimal. Reinsertion of initial core fuel has been analyzed extensively by many reactor vendors and is presently being employed for three-batch cores by some utilities. The remaining work in the area of core physics for this improvement option lies in developing the explicit fuel loading pattern for specific reactors in conjunction with fuel reinsertion for five-batch, extended-burnup, low leakage fuel management. In particular, the effect of fuel reinsertion on five-batch power distributions must be determined. Since core power distributions are sensitive to the specific loading pattern and prior exposure of the fuel, this option must be analyzed on a case-by-case basis using the exact core conditions and prior fueling history of the host reactor.

Another practical consideration in pursuing this option is the effect on fuel performance of many years of storage in the spent fuel storage pool. Reinserting eight assemblies at a time means that some assemblies would be awaiting reinsertion for over 20 years. The major questions that need to be resolved are the effect of this long waiting time on clad and overall assembly corrosion and on the susceptibility of the fuel to PCI failure due to the unusual fuel duty cycle.

5.6.6 Likelihood of Utility Adoption

Although the savings due to the full use of early batches are relatively small, it requires very little effort in terms of RD&D and therefore is attractive from the standpoint of the utility. For almost no additional RD&D cost, a reduction of approximately 0.5% in fuel cycle costs can be obtained. The supplementary benefit of reducing spent fuel storage requirements (by about 2.4% over the life of the plant), although not large, is another incentive for pursuing this option. Indeed, a number of utilities have already employed fuel reinsertion. Perhaps, as an integrated program along with several other improvement options, this option might receive more widespread usage.

5.7 END-OF-CYCLE STRETCHOUT

5.7.1 Introduction

End-of-Cycle (EOC) stretchout provides a means of operating the reactor beyond its nominal depletion of reactivity by modifying the primary and secondary plant operating conditions. It is an attractive operating strategy, often adopted by utilities, to accommodate unexpected grid requirements and to increase flexibility in their refueling schedules. Some utilities, for example Yankee Rowe (1), have employed planned EOC stretchout from cycle to cycle to obtain significant savings in uranium requirements and power generation costs.

Recent concern for nuclear proliferation and the desire to improve the attractiveness of the once-through LWR fuel cycle by improving its fueling efficiency and economics has emphasized this type of operation. Planned EOC stretchout has the potential to reduce uranium requirements by as much as 3-4%, and energy costs by 1-2%. The extent of benefits that can be realized from this operation will, however, depend on the primary and secondary plant characteristics and the operating limitations of the plant turbine generator unit. This study introduces various stretchout methods for a standard three-batch PWR fuel-management scheme and for an improved five-batch, extended-burnup fuel-management scheme. The constraints imposed by the secondary plant characteristics and turbine operation are stressed to assure that the potential of this strategy is not overestimated. The impact of planned EOC stretchout on uranium requirements and cycle energy costs is assessed.

5.7.2 Available Options

Of the several methods available to accomplish planned cycle stretchout, two have been evaluated in this study because they show the greatest potential for reducing uranium demand. These two methods are power coastdown and feedwater-pressure augmentation followed by 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 steam 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 turbine 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 operation 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. 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 beyond VWO position 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.

5.7.3 Analysis Methods

5.7.3.1 Nuclear Characteristics

Nuclear Steam Supply Systems (NSSS) are inherently capable of operating for a substantial length of time beyond the nominal end of the fuel cycle at a variety of power levels. This inherent capability is the result of a large negative moderator temperature coefficient (MTC) at the end of cycle, the negative fuel temperature coefficient (Doppler effect), and the xenon worth. The reactivity contribution due to the MTC is by far the most significant factor. The large negativity of the MTC at end-of-cycle (EOC) lies primarily in the fact that the soluble boron concentration is essentially zero and in its dependence on core average burnup which for the five-batch, extended-burnup core is about 50% greater than that of the conventional three-batch, standard-burnup core.

A more negative MTC implies a larger reactivity gain as a result of lowering the core average moderator temperature. This larger surplus in reactivity renders possible greater cycle burnup extension beyond the nominal end of cycle. Figure 5.7-1 shows for the five-batch, extended-burnup core and the three-batch, standard burnup core the reactivity gain as the core average moderator temperature is lowered. The significantly more negative MTC of the five-batch core is clearly evident in this figure. With a lower core average moderator temperature and power level, the fuel temperature drops and the Doppler effect comes into play but with much less significance than the moderator reactivity effect. The reactivity gain as a result of a drop in fuel temperature is given in Figure 5.7-2 for various power levels. Over the range of temperatures under consideration, the fuel temperature (Doppler) coefficient can be assumed constant at a value of $-1.44 \times 10^{-5} \Delta p / ^\circ F$. As Figure 5.7-2 shows, the fuel temperature defect is a function of both power level and moderator temperature. At 100% core thermal power and normal coolant operating temperature (moderator temperature equal to $592^\circ F$), the fuel temperature defect is zero. As the power level decreases, the fuel temperature

Figure 5.7-1

REACTIVITY CHANGE DUE TO VARIATION IN MODERATOR TEMPERATURE

Figure 5.7-2
FUEL TEMPERATURE DEFECT

drops inserting positive reactivity through a reduction in resonance capture. Since the fuel temperature is directly coupled to the moderator temperature through the heat transfer relationship, the fuel temperature defect will increase with a decrease in moderator temperature. Figure 5.7-3 shows the effect of xenon on core reactivity as the power level is reduced during coastdown. The worth of xenon is nearly a linearly decreasing function of core power level.

5.7.3.2 Methodology

To quantify the reactivity gain as a result of a decrease in core average moderator temperature, two-dimensional (2-D) coarse-mesh PDQ calculations were performed at the end of cycle for a five-batch, extended-burnup, near-equilibrium core. A 2-D coarse-mesh calculation was judged to be necessary for the cooldown calculations since the radial leakage and the assembly power redistributions are vital in the determination of the reactivity effects. In contrast, point PDQ model calculations were employed to determine the excess reactivity resulting from lower xenon levels at lower power levels. This was felt to be adequate since radial leakage has a negligible effect on the reactivity gain from changes in xenon level. Finally, the reactivity gain as a result of a drop in fuel temperature during coastdown was determined using a constant Doppler coefficient of $-1.44 \times 10^{-5} \Delta\rho/^\circ\text{F}$ and assuming linearity of reactivity gain versus fuel temperature at the power levels under consideration. The fuel temperature is dependent both on the core average moderator temperature and the power level (2).

The total reactivity gain during stretchout is the sum of the components due to moderator and fuel temperature decrease, and lower equilibrium xenon level at lower thermal powers. A derivative value of 0.8% $\Delta\rho$ per 1000 MWd/t was used to translate this excess reactivity into a burnup extension; this derivative rule was obtained from analyses of the reactivity loss with burnup at EOC for the five-batch, extended-burnup equilibrium cycle. This derivative depends primarily on the core average burnup and secondarily on the fuel-management scheme employed. Figure 5.6-4 shows the resulting first-time burnup extension in MWd/t for various power levels as a function of the core inlet temperature. The core inlet temperature is used here since secondary plant heat transfer analyses are parameterized on core inlet temperature rather than core average temperature. These two primary temperatures are coupled through an enthalpy balance across the core at various power levels.

Figure 5.7-3
EQUILIBRIUM XENON WORTH

Figure 5.7-4

REACTOR CORE OPERATING CHARACTERISTICS
(FIVE-BATCH, EXTENDED-BURNUP CYCLE)

The first-time extension refers to the initial time EOC stretchout is employed. This can be contrasted to the equilibrium extension which refers to repeated use of EOC stretchout from cycle to cycle.

A comparison of the first-time EOC stretchout curves for a three-batch, standard-burnup core, as given in Figure 5.7-5 with a five-batch, extended-burnup core shows that a five-batch core has a greater first-time benefit than a three-batch core. Primarily, this is a consequence of the more negative MTC at EOC for the five-batch, extended-burnup core as a result of its higher core average burnup.

5.7.3.3 Secondary Plant Characteristics

The secondary plant characteristics control the extent to which the coast-down operation and the feedwater-pressure augmentation operation can be employed. Feedwater-pressure augmentation is limited by the sizing of the steam-piping system, e.g., the turbine admission valve, and the margins allowed in the turbine warranties which determine how far the secondary pressure can be lowered before power must be reduced. In addition, the amount of secondary pressure reduction during power coastdown also can be limited by the low pressure trip setpoint (60 bar for the System 80TM PWR). This low pressure trip setpoint, however, can be manually adjusted by the operator to 13.8 bar below existing system pressure during the stretchout operation and still provide protection against an excessive rate of heat extraction such as occurs during a postulated steam-line-break accident. The most severe restriction appears to be on warranted turbine operation. The discussion which follows strictly applies to a specific turbine and turbine manufacturer. However, it is felt to be representative of the turbine operating limitations which currently exist for large-capacity 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

Figure 5.7-5

REACTOR CORE OPERATING CHARACTERISTICS
(THREE-BATCH, REFERENCE CYCLE)

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, throttle valve position must be reduced until the loadings are equal to or less than their design values (3).

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 would be possible, of course, to design turbines with more flexibility to operate at reduced feedwater temperature (i.e., in the FWPA mode). Such changes would not, however, be backfittable into existing plants because of the excessive cost. Present turbines may have the capability to exceed the 25°F and 50°F limitations discussed above, but this capability has not, in general, been fully assessed by the turbine manufacturers.

Figure 5.7-6 provides a plot of the secondary plant allowable operating space for the System 80TM PWR. The upper boundary of this space defines the best route for a plant to operate during power coastdown; it yields the

Figure 5.7-6
SECONDARY PLANT ALLOWABLE OPERATING SPACE
(SYSTEM 80™ PWR)

maximum thermal power output subject to limitations in the secondary plant characteristics. This plant operating route is shown in Figures 5.7-4 and 5.7-5 as a dashed line. The calculational process for generating this relationship can be broken down into two general steps: the determination of the minimum steam generator pressure versus power curve, and the determination of core inlet temperature versus steam generator pressure. This information can then be correlated to produce a curve of allowable power versus core inlet temperature.

In the analysis for FWPA followed by power coastdown, a typical System 80TM NSSS plant was chosen for study. FWPA was accomplished by reducing feedwater temperature through isolation of certain high-pressure feedwater heaters. The FWPA analysis was performed with two temperature reductions (25⁰F and 50⁰F) followed by standard power coastdown. Figure 5.7-7 shows the net electrical output versus the core inlet temperature for the coastdown operation and the FWPA operation followed by coastdown.

5.7.4 Impact on Uranium Requirements

The prime motivation for the use of planned cycle stretchout is its potential for significant uranium savings without the licensing complications entailed in major design changes. Analyses were performed for the first-time extension and the final results were cast in terms of an equilibrium extension. The benefits are smaller for the equilibrium extension because a large fraction of the burnup increase as a result of coastdown is carried over to the next cycle, thereby reducing the capability of the next cycle to perform coastdown. It can be shown that the cycle burnup extension (δE_n) of an equilibrium cycle is a fraction of the first-time extension (δE) given by the relationship

$$\delta E_n = \frac{2}{n+1} \delta E ,$$

where n is the number of batches employed. This relationship shows that a smaller fraction of the first-time extension carries over to the equilibrium cycle for a five-batch core than for a three-batch core. However, because of the more negative MTC of the five-batch, extended-burnup core, a greater

Figure 5.7-7
SECONDARY PLANT OPERATING CHARACTERISTICS
SYSTEM 80™ PWR

first-time extension is possible than for the three-batch core, and the resulting equilibrium stretchout benefit is very similar to that of the three-batch core. Figures 5.7-8 and 5.7-9 give the net electrical power versus the cumulative days for the first-time extension of the five-batch, extended-burnup, and three-batch, standard-burnup cores, respectively. These plots show how the electrical power decreases as a function of time. Once power coastdown is initiated, the plant electrical output drops about 7 MWe (about 0.55%) per day of additional operation for the five-batch core and 10 MWe (about 0.80%) per day for the three-batch core.

Table 5.7-1 shows the impact of cycle stretchout on uranium savings for the equilibrium extension of a five-batch, extended-burnup core. When the electrical output is allowed to decrease to 85% of rated power, a uranium savings of 3.1% can be achieved in the equilibrium with conventional power coastdown. For FWPA followed by coastdown, a savings in uranium of 3.3% with a feedwater temperature reduction of 25°F and 3.6% with a reduction of 50°F can be achieved. At the economic optimum cycle stretchout time (which will be discussed in the next section) of 14 days for FWPA, this mode of operation saves 4.8% U_3O_8 compared to the U_3O_8 savings which accrue for the normal coastdown mode of operation.

Another interesting point, from the aspect of maximum uranium resource utilization, is the cycle stretchout time at which the energy cost is equal to that without stretchout (i.e., the break-even cycle stretchout). As seen in Figure 5.7-10, for a replacement power cost of 20 mills/kWh, the break-even cycle stretchout for FWPA is about 30 days. This corresponds to a total uranium savings of 8.4% or an additional 3.6% savings in U_3O_8 compared to that achieved when stretchout is stopped at the economic optimum.

The 25°F and 50°F feedwater temperature reductions used in the present study to assess the benefit of FWPA are achieved by taking out of service one to two feedwater preheater strings. This study also showed that substantially higher uranium savings can be obtained by taking more preheaters out of service. However, turbine operation limits the removal of more feedwater preheaters at the present time.

Figure 5.7-8
POWER HISTORY DURING END-OF-CYCLE STRETCHOUT
FIVE-BATCH, EXTENDED-BURNUP CYCLE
(FIRST TIME EXTENSION)

Figure 5.7-9
POWER HISTORY DURING END-OF-CYCLE STRETCHOUT
THREE-BATCH, REFERENCE CYCLE
(FIRST-TIME EXTENSION)

Table 5.7-1

THE IMPACT OF EQUILIBRIUM CYCLE STRETCHOUT ON U_3O_8 REQUIREMENTS
FOR FIVE-BATCH, EXTENDED-BURNUP CORE

Minimum Plant (Net) Electrical Output (% of rated)	Cycle Stretchout Time (Days)		U_3O_8 Savings (%)	
	Normal Coastdown	FWPA ^a	Normal Coastdown	FWPA ^a
95	3.3	5.1	1.2	1.8
85	9.2	10.4	3.1	3.6
75	14.8	16.0	4.8	5.2
65	20.4	21.7	6.2	6.6
55	25.4	27.0	7.3	7.8

Table 5.7-2

THE IMPACT OF EQUILIBRIUM CYCLE STRETCHOUT ON U_3O_8 REQUIREMENTS
FOR THREE-BATCH, STANDARD-BURNUP CORE

Net Electrical Output (% of Rated)	Cycle Stretchout Time (Days)		U_3O_8 Savings (%)	
	Normal Coastdown	FWPA ^a	Normal Coastdown	FWPA ^a
95	5.1	6.8	1.6	2.1
85	11.8	13.4	3.5	4.1
75	17.8	19.2	5.1	5.6
65	24.2	25.2	6.6	6.9
55	31.0	31.0	7.9	8.0

- a. FWPA represents Feedwater Pressure Augmentation (with a maximum of 50°F feedwater temperature reduction) followed by power coastdown.

A comparison of Tables 5.7-1 and 5.7-2 shows that a standard three-batch core benefits slightly more than a five-batch, extended-burnup core from equilibrium cycle stretchout. When the electrical output is allowed to decrease to 85% of rated power, a uranium savings of 3.5% can be achieved compared to 3.1% for the five-batch, extended-burnup core. For FWPA followed by coastdown, a uranium savings of 4.1% can be achieved for the standard three-batch core compared to 3.3% for the five-batch core.

5.7.5 Economic Assessment

Economic evaluations were performed for an equilibrium cycle using the fuel cycle cost code EVALU8. The scope of the economic calculations includes direct and indirect nuclear fuel cycle costs as well as replacement energy costs. Figure 5.7-10 shows the relative energy cost of the five-batch, extended-burnup cycle as a function of the number of days of cycle stretchout for the two different modes of operation, i.e., normal coastdown and FWPA followed by coastdown. 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 Figure 5.7-10 indicates, the FWPA mode of operation is more economic than normal coastdown. Assuming \$40/lb U_3O_8 and a replacement energy cost of 20 mills/kWh, the FWPA mode of operation saves approximately \$1.12 million per cycle at the optimum extension period of 14 days compared to the case of no stretchout, or \$0.16 million per cycle compared to the normal power coastdown mode of operation. FWPA is more economic than normal coastdown because less replacement energy needs to be purchased during the stretchout. The effect of higher replacement energy costs is to shift the optimum stretchout period to a fewer number of days and to increase the relative savings of FWPA compared to normal coastdown.

The sensitivity of the above results to the assumed price of uranium has also been investigated; this is shown in Figure 5.7-11 for \$100/lb U_3O_8 . The results of this sensitivity analysis show that for a uranium price of \$100/lb U_3O_8 and an assumed replacement energy cost of 20 mills/kWh, the economic optimum stretchout period is 22 days. The relative savings of the

Figure 5.7-10

EQUILIBRIUM CYCLE STRETCHOUT: RELATIVE ENERGY COST vs STRETCHOUT TIME
(FIVE-BATCH, EXTENDED-BURNUP CYCLE)

Figure 5.7-11

EQUILIBRIUM CYCLE STRETCHOUT: RELATIVE ENERGY COST vs STRETCHOUT TIME
(FIVE-BATCH, EXTENDED-BURNUP CYCLE)

FWPA mode of operation compared to no stretchout for this case is \$2.84 million per cycle. Compared to normal power coastdown, FWPA saves \$0.24 million per cycle.

Results similar to those reported above have been obtained for the reference three-batch cycle. The main differences between this cycle are the slightly longer optimum stretchout period and the somewhat higher economic savings. Figures 5.7-12 and 5.7-13 summarize the results of the economic analyses for the three-batch cycle. The economic optimum extension period for FWPA operation is 19 days compared to 14 days for the five-batch, extended-burnup cycle when \$40/lb U_3O_8 and 20 mills/kWh are assumed. When the price of uranium reaches \$100/lb U_3O_8 , the economic optimum extension period increases to 27 days compared to 22 days (cf. Figure 5.7-13) for the five-batch, extended-burnup cycle. In terms of dollar savings per cycle, the FWPA mode of operation saves approximately \$1.75 million at the optimum extension compared to the no-stretchout case, or \$0.23 million per cycle compared to the normal power coastdown mode of operation.

In summary, because of the higher incremental core average exposure carried over from one cycle to the next in the five-batch, extended-burnup cycle compared to the reference three-batch cycle, slightly shorter optimum stretchout periods and somewhat smaller fuel resource and economic savings are obtained for planned end-of-cycle stretchout. However, because of the more negative moderator temperature coefficient of reactivity, the first-time extension capability of the five-batch, extended-burnup cycle is greater than that of the three-batch cycle. Consequently, the five-batch, extended-burnup cycle would be more responsive to the use of cycle stretchout to gain cycle length flexibility in the event of an unplanned change in the refueling interval.

5.7.6 Effect on Core Characteristics

The use of planned EOC stretchout impacts core characteristics primarily with respect to the moderator temperature coefficient (MTC). If the same cycle energy is desired as for the no-stretchout case, then a lower feed

Figure 5.7-12

EQUILIBRIUM CYCLE STRETCHOUT: RELATIVE ENERGY COST vs STRETCHOUT TIME
(THREE-BATCH, REFERENCE CYCLE)

Figure 5.7-13

EQUILIBRIUM CYCLE STRETCHOUT: RELATIVE ENERGY COST vs STRETCHOUT TIME
(THREE-BATCH, REFERENCE CYCLE)

URANIUM \$100/LB U_3O_8

○ ECONOMIC OPTIMUM

△ BREAKEVEN STRETCHOUT TIME

enrichment setting can be used and a lower soluble boron concentration will be sufficient to control the excess reactivity at the beginning of cycle. The former factor introduces a more positive MTC while the latter makes the MTC more negative. These two compensating effects tend to cancel, and there is no foreseeable problem with safety calculations on BOC MTC for planned EOC stretchout when a lower enrichment setting is used.

When the same enrichment setting as the no-stretchout case is used, more cycle energy can be extracted, resulting in higher core average exposures both at beginning and end of cycle. As a result of the higher core average burnup, the MTC becomes more negative and this results in a loss in shutdown margin for the steam-line break (SLB) accident. However, based on present safety analyses there should be sufficient margin in the rod worth to cover the extra reactivity incurred in the SLB accident as a result of a more negative MTC.

The impact on other core characteristics, such as the fuel temperature coefficient, delayed neutron fraction, etc., should be minimal. Hence, there is no foreseeable concern in the licensing of planned EOC stretchout operations.

5.7.7 Technical Status and RD&D Requirements

End-of-cycle stretchout has been practiced by various utilities for a number of years. In the majority of cases, utilities have reserved this practice to gain flexibility in their refueling schedule (e.g., moving their refueling outages to periods of low load demand). At least one utility (1), has practiced planned cycle stretchout and has for some time operated each cycle beyond its normal end of full power life. In almost every case, the feedwater heaters were not taken out of service but, rather, standard power coastdown was employed.

Approximately one-half of Combustion Engineering's operating plants have used cycle stretchout for extended periods. The Maine-Yankee plant, operated by the Yankee Atomic Co., is a good example of such operation. This plant operated for approximately 40 days during the winter and spring of 1977 beyond its normal end-of-cycle. During this time, the turbine admission valve was

maintained in the fully open position. Core thermal power declined from 2450 MW(th) to about 1870 MW(th) while the electrical output declined from 770 MWe to 575 MWe. During this power coastdown, the plant operated very well with no observable problems in licensing, fuel integrity or fuel management in subsequent cycles.

Other Combustion Engineering plants which have operated beyond their normal depletion of reactivity are Calvert Cliffs Unit 1, which extended its first cycle approximately 1500 MWd/t, Fort Calhoun which extended its second cycle 485 MWd/t and Millstone Unit 1 which extended its second cycle about 600 MWd/t.

Based on the already wide usage of cycle stretchout, it is felt that little, if any, research and development is required, particularly for standard power coastdown. It may be beneficial, however, to perform a systematic study of the capabilities and limitations of the feedwater pressure augmentation mode of operation. In particular, restrictions on turbine operation should be established with a view toward implementing FWPA. It may be possible to remove overly conservative restrictions which were evoked with little knowledge of the potential benefits of reducing feedwater temperature during cycle stretchout.

5.7.8 Likelihood of Utility Adoption

Clearly, utilities have already adopted cycle stretchout as a means of gaining cycle length flexibility; in at least one instance, planned cycle stretchout has also been practiced. As to whether the electric utility industry will adopt planned cycle stretchout as a widespread practice is difficult to assess. It has been shown that planned cycle stretchout is economical and therefore commercially attractive. Furthermore, even if planned cycle stretchout is practiced, considerable flexibility still exists to allow a utility to delay its refueling outage because of some unforeseen occurrence. For example, if a utility planned to operate a reactor to its economic optimum stretchout extension of each cycle, about two weeks of operation can still be accomplished with no economic penalty (relative to the no-stretchout case) at the end of such a stretchout period.

It appears that the likelihood of a utility adopting planned cycle stretch-out will be tied closely to their management's perception of the benefits versus the risks. The benefits are reduced energy costs and lower U_3O_8 and separative work requirements. The risks lie mainly in somewhat reduced flexibility. Better communication of the benefits and emphasis of the plant operating flexibility that still remains when planned cycle stretchout is practiced may be sufficient to convince utility management to adopt this uranium utilization improvement option.

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Section 6

DISCUSSION OF COMPOSITE IMPROVEMENT CASE

6.1 IMPROVEMENT OPTIONS CONSIDERED IN THE COMPOSITE ANALYSIS

The composite case combines the individual improvement options described in the previous section in order to assess the potential for overall improvement in uranium utilization. To determine the uranium utilization and to identify any safety or performance penalties incurred due to collectively implementing these improvement options, spatial diffusion calculations were performed. The improvement options which were incorporated in the composite case were extended burnup with modified fuel management, low-leakage fuel management, lattice modifications, planned end-of-cycle stretchout, axial blankets, and reinsertion of early batches.

The improvement options which were explicitly analyzed through two-dimensional fuel management calculations were:

- . extended burnup with modified fuel management,
- . low-leakage management,
- . modified lattice design (in the form of reduced pellet diameter), and
- . end-of-cycle stretchout.

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% operational capacity factor. Low-leakage fuel management entails loading fresh makeup fuel in the core interior with burnable poison used to suppress the power in these assemblies during its 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 "wetter" lattice was adopted for the composite improved core. This modified design consists of reducing the fuel pellet diameter from 0.825 cm to 0.800 cm, and reducing

the clad thickness from 0.635 mm to 0.572 mm. The fuel rod pitch was kept constant at 1.286 cm. This corresponds to an increase in the water-to-fuel ratio from 2.041 for the standard design to 2.287 for the modified design. End-of-cycle stretchout, utilizing feedwater-pressure augmentation 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.

These options were combined in the composite case since they constituted the majority of the potential uranium savings and were most easily incorporated into the two-dimensional core model. These improvement options and their contributions to the composite case are discussed in greater detail in Section 6.3.

The improvement options which were not included in the two-dimensional fuel-management calculations were:

- . full use of early batches of the initial core, and
- . axial blankets.

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 (15.24 cm) of the 150 in. (381 cm) fuel rods are loaded with natural uranium pellets. For a more detailed discussion of each of the improvement options included in the composite design, the reader is referred to Section 5 of this report.

These improvement options were not included since the savings in uranium attributable to them were small and well established in the individual assessments, and because the composite analysis would have been complicated considerably by their inclusion. Incorporation of the full use of early batches option in the composite calculations would require a longer approach to equilibrium cycle conditions, since discharged batches from previous cycles are reinserted

later in core life (i.e., over the next 24 years). In addition, the fuel management of each individual cycle would have been complicated by the need to find suitable locations to load the low-enrichment, low-burnup fuel assemblies of these initial core batches. The inclusion of axial blankets in the fuel management calculations would have required that these calculations be performed in three dimensions. Since many depletion calculations were performed to assess the feasibility of the loading pattern for each cycle and to obtain equilibrium conditions, use of a three-dimensional model would have been prohibitive both in terms of computer time and job turnaround. Instead, the initial cycle of the composite improvement case was reanalyzed with a three-dimensional model. Two cases were depleted: one with a uniform axial fuel loading and one with a six-inch (15.2 cm) natural uranium blanket. These calculations were then used to verify the previously performed axial blanket assessment.

6.2 ANALYSIS METHODS

The analysis of the composite uranium utilization improvement case was performed in two and three dimensions using ROCS, the multi-dimensional, coarse-mesh diffusion code previously described in Section 5.3.1. Individual pin peaks were calculated by the FC code using nodal currents generated in the ROCS calculation. The three major changes in the ROCS model which were required for the composite case were: (1) expansion to three dimensions to properly treat the axial blankets, (2) modifications to implement end-of-cycle (EOC) stretchout, and (3) preparation of a consistent set of neutron cross sections for the modified lattice design. These modifications in the ROCS model are discussed in the following sections in greater detail.

6.2.1 Three-Dimensional ROCS Model

The three-dimensional ROCS model consists of 14 planes, one each for the top and bottom 15.24 cm natural uranium blanket, and the remaining 12 planes to model the 350.52 cm active core region. Top and bottom reflector regions were treated by boundary conditions obtained from a previous one-dimensional axial calculation. ROCS utilizes a closed-channel thermal-hydraulic model to calculate the effect on the power distribution of changing moderator and fuel temperatures and moderator density as coolant flows up the channel. Neutron

cross sections are parameterized as functions of fuel and moderator temperatures, soluble boron concentration, U-235 enrichment, and fuel exposure. Thermal-hydraulic feedbacks, as well as fuel temperature and xenon feedbacks, are performed during the flux calculation until neutronic/thermal-hydraulic convergence is achieved.

6.2.2 End-of-Cycle Stretchout

In the composite case, end-of-cycle stretchout is accomplished by feedwater-pressure augmentation followed by power coastdown. The effect of the resulting modified operating conditions on the reactor core is a reduction in the moderator temperature and core power level (with a corresponding drop in fuel temperature and xenon level). The secondary plant characteristics dictate the manner and extent to which end-of-cycle stretchout can be employed (i.e., a reduction in core power level must be coupled with the corresponding change in core inlet temperature determined from the secondary plant characteristics).

During the power coastdown portion of the stretchout, the core power level and moderator temperature decrease continuously with time. Since only discrete changes in core power level and moderator temperature are possible in the spatial diffusion code ROCS, a series of small coastdown steps were employed, typically with a 10% reduction in power level for each step. The corresponding reduction in core inlet temperature is first obtained from the secondary plant operating characteristics along with an estimate of the coastdown time (burnup) from the fuel temperature coefficient, moderator temperature coefficient, and xenon worth. As Figure 6.2-1 illustrates, full thermal power operation is continued past the nominal EOC (feedwater-pressure augmentation phase) until the onset of power coastdown. This corresponds to Point 1 on the curve. At this point, a flux calculation is performed at the corresponding (reduced) moderator temperature in order to determine the core reactivity at this condition. The core power level and moderator temperature are then reduced and a new flux distribution is calculated. (Decreasing the core power level before the depletion step rather than after insures the proper xenon level at the end of the depletion step). A depletion calculation is then made using this flux distribution to arrive at Point 2 on the coastdown curve. The process is then repeated until the desired reduction in core power level is achieved.

Figure 6.2-1
MODELING OF EOC STRETCHOUT IN THE COARSE-MESH
SPATIAL DIFFUSION CODE, ROCS

The coastdown curves change during the transition cycles, primarily due to the MTC becoming more negative with increased core average exposure. This effect is illustrated in Figure 6.2-2, where a substantial shift in the core operating characteristics curves between the initial and equilibrium cycles is apparent. This necessitates a separate calculation of the characteristic coastdown curve for each cycle until equilibrium cycle conditions are reached.

6.2.3 Few-Group Cross Section Generation

The few-group neutron cross sections used in the ROCS calculations were generated for the modified lattice design (cf. Section 5.3.3 for detailed geometry data) in a similar fashion as described in Section 5.3.1.5. The only difference was the use of a 41-group condensed cross section library in the DIT calculations instead of the normal 85-group library, which resulted in a 35% reduction in computer time. This condensed library was carefully verified and proven to give satisfactory results. The maximum error in reactivity for a burnup of 50,000 MWd/t was about 0.08%, while average nuclide concentrations agreed to within 0.2%. The spectral impact of these small errors on the cross section calculation is negligible and results in no noticeable error in the determination of fuel resource requirements.

6.3 FUEL MANAGEMENT AND RESOURCE REQUIREMENTS

The following sections describe the fuel management strategy and the resulting uranium requirements for the composite improved PWR. Section 6.3.1 describes the low-leakage fuel management which was developed for this core design and the effects of the various improvement options on the fuel management. In Section 6.3.2, the uranium requirements for the composite case are presented, and a comparison is made of these requirements to those of the reference three-batch PWR.

6.3.1 Fuel Management

The fuel management strategy for the composite case is similar to the low-leakage fuel management discussed in Section 5.3. Highly reactive fresh fuel is loaded in the interior locations of the core. Burnable poison rods (shims) displace some fuel rods in most of the feed assemblies in order to control their reactivity early in cycle. These poison rods consist of fine B_4C particles dispersed

Figure 6.2-2

COMPARISON OF COASTDOWN CURVES FOR THE INITIAL AND EQUILIBRIUM CYCLES

in an Al_2O_3 matrix. Typical B_4C concentrations are 0.7 to 3.5 wt% (B_4C in Al_2O_3 - B_4C). For shimmed assemblies, a maximum of 24 shim rods was used for the first cycle, and 16 burnable poison rods per shimmed assembly were utilized in subsequent cycles. A greater number of shim rods per assembly is required in the first cycle in order to provide sufficient reactivity control for the long initial cycle. Also, using more shim rods with a lower B_4C concentration reduces self-shielding effects and results in more complete consumption of the poison material. However, this is at the expense of somewhat higher rod linear heat rates and higher enrichments due to the smaller number of fuel pins in the core. Shim placement patterns for 16 and 24 shim assemblies are included in Figure 5.3-12. The peripheral locations of the core were loaded with a combination of once-burned and twice-burned fuel in order to reduce the radial leakage. To maintain uniform discharge burnup values for each batch, no assemblies were loaded on the core periphery for more than once cycle with the exception of Cycle 2. The initial cycle length of the composite case was selected as 13,800 MWd/t while that of Cycle 2 was set at 7,183 MWd/t, as indicated in Table 6.3-1. This distribution of cycle energies is different from those used previously for the low-leakage fuel management (cf. Section 5.3-3). In order to accommodate the more positive moderator temperature coefficient of the modified (wetter) lattice design, a shorter initial cycle was chosen. The cycle lengths of subsequent transition cycles was dictated by a limitation on how quickly the fresh fuel enrichment could be increased from that of the initial core to that of the equilibrium cycle and still maintain realistic core power peaking factors. The core-average enrichment for the initial core is substantially lower than the equilibrium enrichment (2.23 wt% versus 4.40 wt%), and consequently the reactivity between the remaining batches of the initial core and the fresh feed fuel in Cycles 2 through 5 is quite large. Nevertheless, a longer than equilibrium initial cycle is advantageous since this allows somewhat higher enrichments to be loaded in this cycle and thereby aids in this transition.

In Chapter 5, it was noted that power sharing between assemblies becomes more non-uniform as the number of batches in the core increases; i.e., the power density in the fresh and low-burnup fuel is high relative to the power production in more highly irradiated fuel batches. This is further aggravated by the fact that the power is distributed over a fewer number of fresh assemblies in a core with a relatively large number of batches. For five-batch fuel management, this non-uniformity in power sharing between assemblies results in an

Table 6.3-1

COASTDOWN PARAMETERS

Cycle	Nominal ^a Cycle Length, MWd/t	Stretchout Cycle Length, MWd/t	End-of-Cycle Net Electrical Power, %
1	12788	13800	71.2
2	6225	7183	79.5
3	6265	7402	79.5
4	7150	8242	79.5
Equilibrium	9785	11425	72.5

-
- a. Cycle length when excess reactivity is depleted at normal operating conditions. All previous cycles to the one considered have employed stretchout as per the next column.

increase of 10-12% in power peaking relative to the reference three-batch core. This problem is particularly troublesome in the early transition cycles where power-sharing difficulties often occur even in conventional three-batch fuel management schemes. In order to alleviate these difficulties during early transition cycles, 60 fresh fuel assemblies were loaded in the second cycle of the composite improvement case, rather than the 48 assemblies normally replaced each year with five-batch fuel management. With a larger number of assemblies, feed enrichments are reduced, thus improving power sharing. This improved power sharing between the feed assemblies decreased pin peaking by about 8%. In addition, the cycle length (which tends to be rather short in the first transition cycle) increased due to the loading of these 12 extra fuel assemblies. For subsequent cycles, the normal 48 feed assemblies were loaded.

Additional modifications to the fuel management strategy were also necessary in order to incorporate the modified lattice and end-of-cycle stretchout options into a single spatial diffusion calculation for the composite core. These improvement options affected the low-leakage fuel management in two ways. First, the modified lattice design increased the power density in the peripheral assemblies by approximately 10% relative to the standard lattice design with the same fuel management scheme as a result of the differential increase in batch reactivities caused by its softer neutron spectrum. The power roll was compensated for by adjusting the batch enrichments and shim loadings. Since the modified lattice was more reactive than the standard lattice design, the core-average enrichment was also reduced to maintain the same cycle length. The second effect on the fuel management was the reduction in power sharing between burned and fresh fuel assemblies during transition cycles as a result of employing end-of-cycle stretchout. This effect was most pronounced in the four-times-burned fuel, which was typically situated adjacent to fresh fuel. This highly-burned fuel had accumulated additional exposure above that normally achieved as a result of end-of-cycle stretchout during its previous four irradiation cycles. This additional exposure increased the burnup of the four-times-burned batch by approximately 4%. As a result, the burned fuel is less reactive than it would normally have been if EOC stretchout had not been implemented in the previous cycles. This increase in burnup, coupled with the increasing fresh fuel enrichment during the transition to equilibrium, decreases power sharing between the fresh and burned fuel assemblies, resulting in a higher pin peak

in the fresh fuel. However, since limiting pin peaks typically occur near or at EOC in low-leakage fuel management schemes employing burnable poison rods, the increase in pin peaking is partially offset by the reduction in core power level during EOC stretchout. Figure 6.3-1 shows the maximum planar pin peak as a function of burnup for the four transition and equilibrium cycles. In the first cycle, the peak planar pin power is approximately 1.46 and occurs at 7000 MWd/t. The limiting pin peak is observed to increase with cycle, primarily due to increasing non-uniformities in power sharing between fresh and highly irradiated fuel assemblies. These difficulties can be partially alleviated by assembly enrichment zoning, as discussed in Section 6.3.2.

6.3.1.1 3-D Analysis - Cycle 1

The fuel loading pattern and corresponding beginning-of-cycle planar power distribution for Cycle 1 are given in Figure 6.3-2; the maximum planar pin peak is also shown in this figure. At beginning of cycle, the peak power is located in a fresh fuel assembly located on the core periphery. The peak pin power for Cycle 1 decreases with burnup until about 1000 MWd/t, as indicated in Figure 6.3-1. At this point, a fresh fuel assembly in the core interior becomes limiting as its shim rods burn out. This effect is seen in the ascending part of the curve in Figure 6.3-3. The peak pin power in the cycle occurs at approximately 7000 MWd/t.

The evolution of the Cycle 1 core axial power shape with and without the 15.2 cm (6 in.) natural uranium axial blankets is illustrated in Figure 6.3-3 and Figure 6.3-4, respectively. The typical trend of decreasing power density with burnup in the core central region is observed. The peak axial power for the blanketed core is greater than the axial peak for the composite core without blankets. Axial peaking at BOC1 near the core midplane is about 20% greater when axial blankets are used, and only about 10% greater at EOC1.

Figure 6.3-5 is a comparison of the two-dimensional (2-D) planar power distribution with the axially integrated three-dimensional (3-D) power distribution at BOC1. The maximum difference in assembly powers is 2.76% in Box 2. This bias is primarily due to slight differences in the average moderator temperature between the 2-D and 3-D models. In the 3-D ROCS model, a core inlet temperature is specified and the enthalpy rise is calculated as coolant flows up the channel. Thus, each axial plane has a different moderator temperature. In the 2-D ROCS model, an average moderator temperature is specified. These modeling differences resulted in approximately a 2°F difference in the average moderator temperature, which

Figure 6.3-1
EVOLUTION OF PLANAR PIN POWER PEAK

Figure 6.3-2
CORE POWER DISTRIBUTION
BEGINNING OF CYCLE 1 FOR THE COMPOSITE FIVE-BATCH,
EXTENDED BURNUP FUEL MANAGEMENT

BATCH	SHIMS	SHIM LOADING, gm B-10/IN.	ENRICHMENT WT %	No. ASSEMBLIES
A	0	—	1.519	52
B	0	—	1.752	44
C	0	—	2.352	24
C/	16	0.0154	2.352	17
Cx	16	0.0096	2.352	8
D	0	—	2.584	16
D/	20	0.0163	2.584	8
Dx	20	0.0154	2.584	16
E	0	—	2.855	8
E/	20	0.0224	3.049	8
E*	20	0.0125	3.049	8
Ex	24	0.0148	3.049	16
E·	20	0.0224	2.855	16

BATCH IDENTIFIER
 N A
 X.XXX ASSEMBLY NUMBER
 Y.YYY ASSEMBLY RELATIVE POWER
 MAX PIN RELATIVE POWER

						C 1	E 2	C 3	C 4
						0.646	1.026	0.947	0.958
							1.445		
			D 5	D 6	E* 7	B 8	Dx 9	B 10	
			0.619	0.916	1.159	1.081	1.173	1.088	
		C 11	B 12	Ex 13	A 14	Ex 15	A 16	E/ 17	
		0.622	0.854	1.173	0.975	1.242	0.968	1.246	
	18		19	Dx 20	B 21	Cx 22	A 23	C/ 24	B 25
				1.087	1.071	1.189	0.945	1.086	1.025
	26		27		28	E/ 29	A 30	E/ 31	A 32
						1.255	0.956	1.125	0.887
									1.040
34	35	36	37	38	Dx 39	B 40	E/ 41	A 42	
					1.113	0.992	1.063	0.839	
43	44	45	46	47		48	C/ 49	A 50	D/ 51
							1.041	0.865	1.017
52	53	54	55	56	57	58	C/ 59	B 60	
							1.035	0.989	
61	62	63	64	65	66	67		68	C/ 69
									1.083

Figure 6.3-3

CYCLE 1 COMPOSITE FIVE-BATCH FUEL MANAGEMENT WITH A BLANKET EVOLUTION
OF AXIAL POWER SHAPE

Figure 6.3-4

CYCLE 1 COMPOSITE FIVE-BATCH FUEL MANAGEMENT WITHOUT AXIAL BLANKET
EVOLUTION OF AXIAL POWER SHAPE

accounts for the small differences in the planar power distributions. At EOC1, the maximum difference in box powers is 2.21% and occurs in assembly 69, as indicated in 6.3-6. A comparison of the integrated 3-D power distribution for the composite core and the composite core with 15.2 cm axial blankets is given in Figure 6.3-7 for BOC1. A maximum difference of 0.54% occurs at BOC1, and the maximum difference at EOC1 is 0.97%, as indicated in Figure 6.3-8. These small differences in Cycle 1 power distributions indicate that 2-D calculations are sufficiently accurate, provided the axial leakage is represented by axial bucklings which are consistent with the blanketed core. Therefore, the 2-D ROCS model was used in all subsequent composite cycles. The use of 2-D rather than 3-D calculations for subsequent cycles saved a considerable amount of computer time and turn-around time on completing the composite fuel management calculations.

The critical soluble boron concentrations for the blanketed and unblanketed cores are given in Figure 6.3-9. The enrichment of the active core region was adjusted such that the end-of-cycle eigenvalues were the same. This resulted in about a 2% savings in U-235 loading for the composite blanketed core as compared to the composite core without blankets. As Figure 6.3-9 indicates, a higher boron concentration is required at BOC for the blanketed core due to importance weighting of the higher enriched active core region and the reduced axial leakage. As the core-average burnup increases and the axial power at the core mid-plane decreases (cf. Figure 6.3-3), the importance weighting of the active core region decreases in the blanketed core causing the critical boron curves to merge. The fact that the active core of the blanketed case achieves a higher burnup than the equivalent region of the unblanketed case also contributes to this merging of core reactivities. The final 1000 MWd/t of the boron curve represents the coastdown phase plant operation, when the soluble boron concentration is a constant 10 ppm.

6.3.1.2 2-D Analysis - Transition and Equilibrium Cycles

The beginning-of-cycle power distributions and fuel loading patterns for Cycles 2 through 4 are presented in Figures 6.3-10 through 6.3-12, respectively. Figures 6.3-13 through 6.3-16 show the planar power distributions for the equilibrium cycle case at BOC, 4000 MWd/t, 8000 MWd/t, and at EOC, respectively. Two effects are apparent upon examining these planar power distributions. First, the power

Figure 6.3-7
COMPARISON OF BLANKETED AND UNBLANKETED
INTEGRATED 3-D RELATIVE POWER DISTRIBUTIONS
FOR THE COMPOSITE FIVE-BATCH CORE
(BEGINNING OF CYCLE 1)

BATCH IDENTIFIER	
N A	ASSEMBLY NUMBER
X.XXX	3-D ASSEMBLY RELATIVE POWER, WITH 6" (15.2 cm) BLANKET
Y.YYY	3-D ASSEMBLY RELATIVE POWER, NO BLANKET
Z.ZZ	% DIFFERENCE

				C 1	E 2	C 3	C 4		
				0.643 0.642 0.17	0.996 0.997 -0.17	0.924 0.925 -0.06	0.933 0.934 -0.12		
		D 5	D 6	E* 7	B 8	Dx 9	B 10		
		0.6234 0.6211 0.37	0.910 0.908 0.22	1.143 1.146 -0.24	1.060 1.058 0.17	1.154 1.158 -0.33	1.067 1.066 0.08		
		C 11	B 12	Ex 13	A 14	Ex 15	A 16	E/ 17	
		0.627 0.624 0.37	0.857 0.854 0.40	1.168 1.172 -0.30	0.973 0.969 0.35	1.232 1.238 -0.48	0.966 0.964 0.23	1.234 1.240 -0.54	
18		19	Dx 20	B 21	Cx 22	A 23	C/ 24	B 25	
			1.090 1.092 -0.15	1.068 1.065 0.28	1.186 1.187 -0.09	0.950 0.948 0.30	1.092 1.094 -0.22	1.027 1.025 0.18	
26		27	28	E/ 29	A 30	E/ 31	A 32	D/ 33	
				1.252 1.257 -0.41	0.964 0.961 0.32	1.135 1.140 -0.44	0.898 0.895 0.29	1.056 1.059 -0.33	
34		35	36	37	38	Dx 39	B 40	E/ 41	A 42
						1.126 1.129 -0.27	1.004 1.002 0.25	1.081 1.086 -0.42	0.859 0.856 0.33
43		44	45	46	47	48	C/ 49	A 50	D/ 51
							1.060 1.061 -0.09	0.885 0.881 0.41	1.040 1.042 -0.20
52		53	54	55	56	57	58	C/ 59	B 60
								1.056 1.056 0.01	1.006 1.002 0.41
61		62	63	64	65	66	67	68	C/ 69
									1.102 1.102 0.08

Figure 6.3-8
COMPARISON OF BLANKETED AND UNBLANKETED
INTEGRATED 3-D RELATIVE POWER DISTRIBUTIONS
FOR THE COMPOSITE FIVE-BATCH CORE
(END OF CYCLE 1)

BATCH IDENTIFIER	
N	A
ASSEMBLY NUMBER	
X.XXX	3-D ASSEMBLY RELATIVE POWER, WITH 6" (15.2 cm) BLANKET
Y.YYY	3-D ASSEMBLY RELATIVE POWER, NO BLANKET
Z.ZZ	% DIFFERENCE

				C 1		E 2		C 3		C 4	
				0.550 0.545 0.90		0.749 0.744 0.67		0.717 0.711 0.76		0.704 0.699 0.82	
				D 5		D 6		E* 7		B 8	
				0.549 0.544 0.86		0.820 0.817 0.39		1.121 1.123 -0.12		0.946 0.940 0.57	
				Dx 9		B 10		1.137 1.139 -0.14		0.949 0.944 0.51	
				C 11		B 12		Ex 13		A 14	
				0.561 0.555 0.97		0.819 0.813 0.71		1.240 1.245 -0.39		0.974 0.969 0.53	
				Ex 15		A 16		E/ 17		1.300 1.309 -0.63	
				1.311 1.318 -0.58		0.990 0.987 0.33					
18				19		Dx 20		B 21		Cx 22	
						1.166 1.169 -0.27		1.032 1.029 0.28		1.169 1.169 -0.01	
						A 23		C/ 24		B 25	
						0.989 0.986 0.29		1.180 1.185 -0.38		1.040 1.039 0.05	
26				27		28		E/ 29		A 30	
								1.315 1.325 -0.75		0.997 0.995 0.19	
								E/ 31		A 32	
								1.274 1.285 -0.89		0.993 0.993 0.01	
								C/ 33		1.2275 1.2369 -0.76	
34				35		36		37		38	
										Dx 39	
										1.2347 1.2430 -0.67	
										B 40	
										1.0327 1.0336 -0.09	
										E/ 41	
										1.2570 1.2693 -0.97	
43				44		45		46		47	
										48	
										C/ 49	
										1.1429 1.1471 -0.37	
										A 50	
										0.9450 0.9424 0.28	
52				53		54		55		56	
										57	
										58	
										C/ 59	
										1.0907 1.0911 -0.04	
										B 60	
										0.9507 0.9459 0.51	
61				62		63		64		65	
										66	
										67	
										68	
										C/ 69	
										1.0727 1.0707 0.19	

Figure 6.3-9

CYCLE 1 SOLUBLE BORON RUNDOWN FOR THE FIVE-BATCH COMPOSITE CORE
(WITH AND WITHOUT 6 INCH AXIAL BLANKET)

Figure 6.3-10
CORE POWER DISTRIBUTION
BEGINNING OF CYCLE 2
FOR THE COMPOSITE FIVE-BATCH,
EXTENDED BURNUP FUEL MANAGEMENT SCHEME

BATCH	SHIMS	SHIM LOADING gm B-10/IN	ENRICHMENT WT %	No. ASSEMBLIES
F	0	—	3.50	24
F/	16	.0173	3.10	20
Fx	16	.0155	3.50	8
F*	16	.0063	3.50	8

N A		BATCH IDENTIFIER
X.XXX		ASSEMBLY NUMBER
Y.YYY		ASSEMBLY RELATIVE POWER
		MAX PIN RELATIVE POWER

						F 1	E 2	F 3	E/ 4
						0.694	0.652	0.944	0.705
			E. 5	F 6	Ex 7	Fx 8	Dx 9	Ex 10	
			0.444	1.007	1.030	1.242 1.460	1.035	1.062	
		E/ 11	F* 12	B 13	D 14	B 15	E 16	D/ 17	
		0.494	1.082	0.823	1.165	0.878	1.187	1.081	
	18		Ex 20	E* 21	Cx 22	F/ 23	B 24	F/ 25	
			1.089	1.176	1.019	1.260	0.927	1.257	
	26			D/ 29	D 30	C/ 31	C 32	C/ 33	
				1.106	1.175	1.036	1.052	0.979	
	34	35	36	37	38	Dx 39	C 40	B 41	C 42
						1.177	1.156	0.886	1.082
	43	44	45	46	47		B 49	F/ 50	Dx 51
							0.894	1.201	1.009
	52	53	54	55	56	57		C/ 59	C 60
								0.945	1.007
	61	62	63	64	65	66	67		C/ 69
									0.946

Figure 6.3-11
CORE POWER DISTRIBUTION
BEGINNING OF CYCLE 3
FOR THE COMPOSITE FIVE BATCH, EXTENDED BURNUP
FUEL MANAGEMENT SCHEME

BATCH	SHIMS	SHIM LOADING, gm B-10/IN.	ENRICHMENT WT %	No. ASSEMBLIES
G	0	—	3.45	8
G/	16	.0173	3.45	24
Gx	16	.0136	3.45	8
G*	16	.0058	3.45	8

N	A	BATCH IDENTIFIER
X.XXX		ASSEMBLY NUMBER
Y.YYY		ASSEMBLY RELATIVE POWER
		MAX. PIN RELATIVE POWER

ACTIVE POWER					Ex 1	F 2	G 3	Ex 4	
					0.466	0.931	1.084	0.704	
				F 5	F 6	F* 7	C 8	E. 9	Gx 10
				0.712	1.025	1.216	0.970	1.085	1.342
			Ex 11	G* 12	E* 13	D 14	E. 15	Gx 16	D/ 17
			0.508	1.247	1.067	1.045	1.052	1.357 1.564	0.935
		18	19	C 20	F/ 21	D 22	F/ 23	C 24	E/ 25
				0.949	1.210	0.976	1.190	0.906	0.967
		26	27	28	Dx 29	G/ 30	Dx 31	G/ 32	D/ 33
					0.906	1.246	0.889	1.176	0.822
34	35	36	37	38	Dx 39	E 40	C/ 41	G/ 42	
					0.845	0.875	0.789	1.174	
43	44	45	46	47	48	E/ 49	Fx 50	C 51	
						0.949	1.155	0.899	
52	53	54	55	56	57	58	C/ 59	F/ 60	
							0.789	1.032	
61	62	63	64	65	66	67	68	C/ 69	
								0.734	

Figure 6.3-12
CORE POWER DISTRIBUTION
BEGINNING OF CYCLE 4
FOR THE COMPOSITE FIVE-BATCH, EXTENDED BURNUP
FUEL MANAGEMENT SCHEME

BATCH	SHIMS	SHIM LOADING, gm B-10/IN.	ENRICHMENT WT %	No. ASSEMBLIES
H	0	—	3.80	8
H/	16	.0155	3.80	16
Hx	16	.0127	3.80	12
H*	16	.0033	3.80	8
H.	16	.0182	3.80	4

N A BATCH IDENTIFIER
 X.XXX ASSEMBLY NUMBER
 Y.YYY ASSEMBLY RELATIVE POWER
 MAX PIN RELATIVE POWER

IDENTIFIER NUMBER RELATIVE POWER RELATIVE POWER						F/ 1 0.395	G/ 2 0.704	H 3 1.011	G/ 4 0.868			
						G/ 5 0.497	Gx 6 0.753	Fx 7 0.929	Hx 8 1.267	E* 9 0.906	Hx 10 1.356 1.568	
						F/ 11 0.519	H* 12 1.171	F/ 13 1.035	G* 14 1.317	E 15 0.960	G 16 1.355	Ex 17 0.923
						18	19	E/ 20 0.877	F* 21 1.156	E. 22 0.919	F 23 1.299	Ex 24 0.988
					26	27	28	E/ 29 0.899	H/ 30 1.358	E. 31 0.951	H/ 32 1.325	D 33 0.844
34	35	36	37	38	Ex 39 0.963	F 40 1.249	C 41 0.861	Gx 42 1.195				
43	44	45	46	47	48	D 49 0.851	F 50 1.126	C 51 0.809				
52	53	54	55	56	57	58	D 59 0.792	H. 60 1.211				
61	62	63	64	65	66	67	68	C 69 0.754				

Figure 6.3-13
CORE POWER DISTRIBUTION
BEGINNING OF CYCLE EQUILIBRIUM CYCLE
FOR THE COMPOSITE FIVE-BATCH, EXTENDED BURNUP
FUEL MANAGEMENT SCHEME

BATCH	SHIMS	SHIM LOADING, gm B-10/IN.	ENRICHMENT WT %	No. ASSEMBLIES
F	0	—	4.40	8
F/	16	.0195	4.40	12
Fx	16	.0086	4.40	8
F*	16	.0107	4.40	8
F.	16	.0135	4.40	12

N	A	BATCH IDENTIFIER
X.XXX		ASSEMBLY NUMBER
Y.YYY		ASSEMBLY RELATIVE POWER
		MAX PIN RELATIVE POWER

						D/ 1	E* 2	F 3	E/ 4
						0.427	0.810	1.091	0.945
				Dx 5	Ex 6	C 7	F* 8	B 9	F/ 10
				0.429	0.762	0.862	1.329 1.623	0.866	1.339
			Dx 11	Fx 12	C/ 13	F/ 14	B/ 15	E 16	B/ 17
			0.537	1.134	0.957	1.328	0.920	1.273	0.893
	18		19	C/ 20	E/ 21	C/ 22	E/ 23	B/ 24	E. 25
				1.131	1.266	1.030	1.210	0.860	1.161
	26		27	28	B/ 29	D* 30	B/ 31	F. 32	B/ 33
					0.888	1.149	0.859	1.311	0.849
	34	35	36	37	38	C/ 39	Dx 40	C/ 41	D* 42
						1.122	1.168	0.992	1.085
	43	44	45	46	47	48	C/ 49	D 50	C/ 51
							1.077	1.024	0.943
	52	53	54	55	56	57	58	B/ 59	F. 60
								0.786	1.205
	61	62	63	64	65	66	67	68	A 69
									0.653

Figure 6.3-14
EQUILIBRIUM CYCLE CORE POWER DISTRIBUTION
4000 MWd/t
FOR THE COMPOSITE FIVE-BATCH, EXTENDED BURNUP
FUEL MANAGEMENT SCHEME

BATCH IDENTIFIER
 ASSEMBLY NUMBER
 ASSEMBLY RELATIVE POWER DENSITY
 MAXIMUM RELATIVE PIN POWER DENSITY

						D/ 1	E* 2	F 3	E/ 4
						0.456	0.823	1.047	0.933
				Dx 5	Ex 6	C 7	F* 8	B 9	F/ 10
				0.475	0.812	0.887	1.352 1.646	0.8560	1.318
		Dx 11	Fx 12	C/ 13	F/ 14	B/ 15	E 16	B/ 17	
		0.589	1.211	0.992	1.359	0.930	1.233	0.888	
	18		19	C/ 20	E/ 21	C/ 22	E/ 23	B/ 24	E. 25
				1.155	1.294	1.032	1.228	0.872	1.192
	26		27		28	B/ 29	D* 30	B/ 31	F. 32
						0.887	1.106	0.849	1.316
									0.857
	34	35	36	37	38	C/ 39	Dx 40	C/ 41	D* 42
						1.044	1.090	0.950	1.044
	43	44	45	46	47	48	C/ 49	D 50	C/ 51
							0.996	0.970	0.911
	52	53	54	55	56	57	58	B/ 59	F. 60
								0.780	1.215
	61	62	63	64	65	66	67	68	A 69
									0.678

Figure 6.3-15
EQUILIBRIUM CYCLE CORE POWER DISTRIBUTION
8000 MWd/t
FOR THE COMPOSITE FIVE-BATCH, EXTENDED BURNUP
FUEL MANAGEMENT SCHEME

N	A
X.XXX	ASSEMBLY RELATIVE POWER DENSITY
Y.YYY	MAXIMUM PIN RELATIVE POWER DENSITY

PIN RELATIVE POWER DENSITY					D/ 1	E* 2	F 3	E/ 4
					0.475	0.843	1.042	0.945
			Dx 5	Ex 6	C 7	F* 8	B 9	F/ 10
			0.499	0.835	0.909	1.402	0.863	1.355
						1.653		
		Dx 11	Fx 12	C/ 13	F/ 14	B/ 15	E 16	B/ 17
		0.615	1.254	0.995	1.398	0.934	1.210	0.887
18	19	C/ 20	E/ 21	C/ 22	C/ 23	B/ 24	E. 25	
		1.132	1.259	1.007	1.214	0.872	1.203	
26	27	28	B/ 29	D* 30	B/ 31	F. 32	B/ 33	
			0.852	1.049	0.831	1.340	0.862	
34	35	36	37	38	C/ 39	Dx 40	C/ 41	D* 42
					0.980	1.035	0.927	1.021
43	44	45	46	47	48	C/ 49	D 50	C/ 51
						0.949	0.947	0.904
52	53	54	55	56	57	58	B/ 59	F. 60
							0.797	1.285
61	62	63	64	65	66	67	68	A 69
								0.727

Figure 6.3-16
EQUILIBRIUM CYCLE CORE POWER DISTRIBUTION
9785 MWd/t (NOMINAL EOC)
FOR THE COMPOSITE FIVE-BATCH, EXTENDED BURNUP
FUEL MANAGEMENT SCHEME

N	A
X.XXX	
Y.YYY	

BATCH IDENTIFIER
 ASSEMBLY NUMBER
 ASSEMBLY RELATIVE POWER DENSITY
 MAXIMUM PIN RELATIVE POWER DENSITY

						D/ 1	E* 2	F 3	E/ 4	
						0.487	0.847	1.036	0.947	
					Dx 5	Ex 6	C 7	F* 8	B 9	F/ 10
					0.507	0.841	0.916	1.412 1.665	0.864	1.369
				Dx 11	Fx 12	C/ 13	F/ 14	B/ 15	E 16	B/ 17
				0.621	1.258	0.991	1.413	0.933	1.198	0.885
	18		19	C/ 20	E/ 21	C/ 22	E/ 23	B/ 24	E. 25	
				1.116	1.238	0.997	1.205	0.871	1.202	
	26		27	28	B/ 29	D* 30	B/ 31	F. 32	B/ 33	
					0.839	1.031	0.827	1.353	0.867	
34	35	36	37	38	C/ 39	Dx 40	C/ 41	D* 42		
					0.963	1.023	0.925	1.020		
43	44	45	46	47	48	C/ 49	D 50	C/ 51		
						0.941	0.949	0.912		
52	53	54	55	56	57	58	B/ 59	F. 60		
							0.814	1.330		
61	62	63	64	65	66	67	68	A 69		
								0.756		

density in the peripheral assemblies increases with burnup, and the power density of assemblies in the central locations of the core decreases with burnup. Secondly, superimposed on this overall power shift is an increase in power density with burnup of the shimmed fresh fuel assemblies due to the burnout of poison material.

The planar power distributions for the composite case are similar to those obtained for the low-leakage fuel management discussed in Section 5.3. A slight increase in pin peaking results during the transition cycles and is due to the implementation of end-of-cycle stretchout during all previous cycles. In order to minimize the propensity of EOC stretchout to increase pin peaking during the transition cycles, a slower approach to equilibrium cycle conditions must be made. As shown by the thirty-year mass flows in Appendix A, the composite improvement fuel management requires approximately eight cycles before the equilibrium enrichment can be introduced into the core as compared to three cycles for the reference three-batch core. A more rapid approach to equilibrium cycle conditions could be made by not implementing EOC stretchout during the transition cycles; however, this would require the loading of approximately 4% more fissile material during these cycles.

End-of-cycle stretchout was implemented in each cycle of the composite improvement case. Table 6.3-1 summarizes the stretchout operation for the four transition cycles and equilibrium cycle. In the first cycle the plant net electrical output was reduced to 71.2%, which is approximately the economic optimum value (cf. Section 5.7). The net electrical power in Transition Cycles 2 through 4 was only reduced to 79.5% in order to reduce the power peaking problems during these cycles. The power level in the equilibrium cycle was again reduced to a near-optimum value of 72.5%.

Equilibrium burnup values by batch are summarized in Table 6.3-2 for each cycle of exposure. The total exposure for the composite case is about 6.5% greater than the exposure of the low-leakage fuel management (see Table 5.3-6) due to the reduced pellet diameter of the modified lattice. In the first cycle of residency, fresh fuel receives its highest exposure since it is loaded in the core interior. During the second and third cycles, portions of each batch are

Table 6.3-2

EQUILIBRIUM BATCH BURNUPS BY CYCLE

COMPOSITE FIVE-BATCH, EXTENDED-BURNUP FUEL MANAGEMENT

CYCLE OF EXPOSURE	INCREMENTAL BURNUP, Mwd/t
1	14493
2	12229
3	9371
4	11200
5	9868
TOTAL	57161

moved to the core periphery. Thus, the batch-average exposure for the second and third cycles is only about 84% and 65%, respectively, of the first cycle exposure. In the fourth cycle of residency, the exposure increases when the fuel is returned to the center of the core. These batch exposures are similar to those for the standard lattice design low-leakage fuel management (see Table 5.3-6).

6.3.2 Assembly Enrichment Zoning

It was noted previously that five-batch fuel management results in approximately 10-12% higher power peaking than the reference three-batch PWR. One technique to reduce this potential loss in core thermal margin is by enrichment zoning of fuel rods within an assembly.

In order to assess the effect of enrichment zoning as it applies to the composite case, depletion calculations were performed using the coarse-mesh diffusion code, ROCS. Nodal interassembly currents from the ROCS calculations were then used in the auxiliary code, FC, to determine individual pin power levels. The fuel assembly geometry used in these calculations is shown in Figure 6.3-17. Sixteen burnable poison rods were employed in an assembly with 100 fuel pins at 3.8 wt% enrichment and 120 fuel pins at 4.9 wt% enrichment. This is approximately the optimum enrichment split for this lattice configuration as indicated in Figure 6.3-18. This figure also shows the benefit of optimizing the number (and location) of the low-enrichment fuel pins. The enrichment of the fuel pins in the central region of the assembly around the instrumentation location and adjacent to the four control rod locations is 3.8 wt%. The use of lower-enrichment fuel in these locations reduces the steep flux gradient across the fuel assembly which is characteristic of five-batch fuel management, and also reduces water hole peaking. In addition, the single pin in the upper left-hand corner of Figure 6.3-17 is at the lower enrichment to reduce power peaking adjacent to the interassembly water channel.

The relative pin power distribution is illustrated in Figure 6.3-17 for a fresh fuel assembly at the beginning of the equilibrium cycle. The peak-pin-to-assembly power for this assembly with uniform pin enrichments is 1.154, and

Figure 6.3-17

EFFECT ON LOCAL POWER DISTRIBUTION OF ENRICHMENT ZONING ASSEMBLIES

Figure 6.3-18
EFFECT OF ENRICHMENT SPLIT ON PIN/BOX POWER

occurs adjacent to the diagonal water hole. Zoned enrichment reduces the pin-to-assembly power to 1.085. Note that the peak pin power is shifted away from the water hole by enrichment zoning.

This zoned enrichment pattern was depleted to the equilibrium cycle length and compared with an identical calculation for an assembly with a uniform enrichment. The maximum planar pin peak in the core as a function of burnup for uniform and zoned enriched assemblies is compared in Figure 6.3-19. The maximum pin power peak (which occurred at 9800 MWd/t) is about 5% lower for the zoned enrichment case. Thus, zonal enrichment can reduce the potential loss of thermal margin due to five-batch fuel management by approximately 40%.

6.3.3 Resource Requirements

The material inventories of the initial core batches and the equilibrium feed batch for the composite improvement case are given in Table 6.3-3. The initial core fissile loading for the composite five-batch core is approximately 8% lower than the three-batch core (see Table 5.2-2). The equilibrium batch fissile loading, however, is 23.5% lower than the three-batch core. The smaller initial core savings is partially due to the longer first cycle of the composite five-batch core compared to the three-batch core. In addition, the benefit of the high discharge burnup values is not realized for several cycles.

The uranium and separative work requirements are summarized in Table 6.3-4 for the composite five-batch and reference three-batch cores. Over 30 years, the composite five-batch PWR requires 21% less U_3O_8 than the reference three-batch PWR. Separative work requirements for the improved PWR are 9% less than the reference three-batch PWR. The peak pellet discharge exposure increases from about 55 MWd/kg for the reference PWR to about 69.5 MWd/kg for the improved PWR. Spent fuel storage and disposal requirements are about 40% less for the composite PWR since the number of discharged fuel assemblies is reduced from 80 for the reference PWR to 48. The amount of plutonium discharged is also reduced by about 40% for the composite improved PWR; annual fissile plutonium discharge

Figure 6.3-19
EFFECT OF ENRICHMENT ZONING ON THE EVOLUTION OF
PLANAR PIN POWER PEAK
(EQUILIBRIUM CYCLE)

Table 6.3-3

INVENTORIES FOR COMPOSITE IMPROVEMENT CASE^a

	Initial Core	Equilibrium Core
Enrichment	2.18	4.31
Full Power Days	339	274
Fuel Loaded (kg)		
U-235	2037.7	783.0
U-238	91363.2	17376.7
Fuel Discharged (kg)		
U-235	136.4	81.9
U-236	39.5	134.2
U-238	23396.4	16741.8
Pu-239	88.2	83.6
Pu-240	32.0	48.7
Pu-241	14.8	27.4
Pu-242	3.8	18.9

-
- a. Includes all improvement options except reinsertion of early core batches.

Table 6.3-4

FUEL MANAGEMENT INFORMATION - PWR ONCE-THROUGH CYCLE

	Reference PWR	Composite Improved PWR
Average capacity factor, %	75	75
Fraction of core replaced/refueling	0.33	0.20 ^d
Refueling Interval, years	1	1
Fissile fabrication loss fraction	0.015	0.015
Equilibrium cycle enrichment, %	3.00	4.70 ^c
U ₃ O ₈ requirements, short tons/GWe ^a		
Initial core	394	376 ^b
Annual equilibrium	194	150
30-year cumulative	6040	4789
Separative work requirements, 10 ³		
Separative work units (SWU)/GWe ^a		
Initial core	200	196
Annual equilibrium reload	117	106
30-year cumulative	3600	3267
Plutonium in spent fuel		
Annual equilibrium discharge		
Plutonium fissile, kg/GWe	173	95
Plutonium total, kg/GWe	245	140
30-year cumulative discharge		
Plutonium fissile, kg/GWe	5169	2575
Plutonium total, kg/GWe	7143	3807
Residual U-235 in spent fuel		
U-235, wt%	0.85	0.48
Annual equilibrium discharge, kg/GWe	220	69
30-year cumulative discharge, kg/GWe	6498	1854
Batch-average discharge burnup, MWd/kg	30.4	57.13
Peak discharge exposure, MWd/kg	55	69.45

a. Tails composition 0.2 wt%

b. Long initial cycle and short transition cycles required to attain the equilibrium cycle with acceptable power distributions.

c. Active fuel region; blanket enrichment = 0.711 wt%

d. This is the nominal value; the use of fuel reinsertion lowers this value by about 2.4% on the average.

is 173 kg/GWe for the reference core and only 95 kg/GWe for the composite PWR.

6.4 CORE CHARACTERISTICS AND IMPACT ON SAFETY

Selected core physics parameters were calculated for the equilibrium cycle to determine the effects of the combined improvement options on reactor safety. The physics parameters which were analyzed were those which had the greatest potential impact on plant safety and which were expected to deviate the most from those of the reference PWR. These calculations were performed with two-dimensional PDQ and ROCS models for the reference three-batch and composite cores, respectively. The three safety parameters which were evaluated are boron worth, the moderator temperature coefficient, and the fuel temperature coefficient. These core physics parameters are discussed below in greater detail.

6.4.1 Boron Worth

Table 6.4-1 gives the inverse boron worths at the beginning and end of equilibrium cycle for the reference three-batch and composite five-batch cores. At beginning of cycle, the inverse boron worth for the two cores is nearly the same, while at nominal end of cycle the inverse boron worth for the composite five-batch core is about 6% less than the reference three-batch core. At the beginning of cycle, the more thermal spectrum due to the increased water-to-fuel ratio of the modified lattice design is apparently offset by the higher enrichment of the composite core, which tends to harden the neutron spectrum. Thus, the inverse boron worths at the beginning of cycle for the two cores are nearly identical. At the end of cycle (EOC), the inverse boron worth is less for the composite core because of the softer neutron spectrum. This softer spectrum is due to the depletion of the burnable poison rods, which tend to harden the neutron spectrum at BOC. Hence, at EOC the spectral "softening" effects of the modified lattice design are dominant, resulting in a smaller inverse boron worth.

The inverse boron worth is important in the context of safety for the boron dilution accident. In the event that unborated water is mistakenly pumped into the reactor coolant system during cold shutdown, it is important to have

Table 6.4-1

EQUILIBRIUM CYCLE SAFETY PARAMETERS^a

Safety Parameters	Reference BOC	Three-Batch EOC ^b	Composite Five-Batch		
			BOC	EOC ^c	Nominal EOC ^b
Inverse Boron Worth (ppm/% $\Delta\rho$)	108.2	95.6	106.3	82.0	90.1
Moderator Temperature Coefficient (MTC) ($\Delta\rho \times 10^{-4}/^{\circ}\text{F}$)	-0.78 ^d	-3.02	-1.01 ^e	-1.76	-2.61
Fuel Temperature Coefficient (FTC) ($\Delta\rho \times 10^{-5}/^{\circ}\text{F}$)	-1.30	-1.46	-1.36	-1.38	-1.44

-
- a. Cycle length = 274 full power days.
- b. 100% core thermal power, inlet temperature = 564^oF.
- c. 80% core thermal power, inlet temperature = 528^oF.
- d. Boron level = 969 PPM
- e. Boron level = 958 PPM

sufficient time to detect and terminate the dilution operation before the shutdown reactivity margin is reduced to zero. Since the inverse boron worths for the composite core are virtually identical to those of the reference three-batch core, the boron dilution accident is expected to be no more severe than in the three-batch core.

6.4.2 Moderator Temperature Coefficient

The moderator temperature coefficients (MTCs) are summarized in Table 6.4-1 for the reference three-batch and composite five-batch cores. The MTC for the composite five-batch core is more negative than the three-batch core at the beginning of cycle (BOC) due to the increased blackness of the core caused by higher enrichments and larger fission product inventories. The propensity of increased core blackness to decrease the MTC is partially offset by the wetter lattice design of the composite core, which tends to increase the moderator temperature coefficient. Therefore, the MTC at BOC for the composite five-batch core is more positive than the five-batch, low-leakage core described in Chapter 5 (see Table 5.3-1). At the nominal end of cycle, the MTC for the composite core is somewhat less negative than the three-batch core.

A less negative MTC increases the consequences of warm water accidents (e.g., loss of load or total loss of forced reactor coolant flow), since less negative reactivity is inserted as the reactor coolant temperature increases. Conversely, cold water accidents (such as the steam-line break accident) are aggravated by a more negative MTC. Since the MTC becomes more negative with burnup, warm water accidents are limiting at BOC and cold water accidents are limiting at EOC. Thus, warm water accidents at BOC are more severe for the reference core than for the composite PWR due to the more positive MTC; cold water accidents at EOC are also more limiting for the reference PWR. Since the reference PWR design is within acceptable safety standards, the composite improved PWR is not expected to be limited by MTC constraints.

6.4.3 Fuel Temperature Coefficients

Table 6.4-1 gives the fuel temperature coefficients (FTCs) at the beginning and the end of cycle for the reference three-batch core and the composite five-batch core. The end-of-cycle FTCs are more negative than the beginning-of-cycle values primarily due to increased Doppler absorption in Pu-240, which has a higher concentration at the end of cycle. At the beginning and nominal end of cycle, the FTCs for the two cores are nearly identical. The end-of-cycle FTC for the composite core is smaller than the three-batch core due to the reduced power level during EOC stretchout. Since the FTCs are virtually identical for the reference and composite cores, no significant FTC-related safety impacts are anticipated.

6.5 ECONOMIC ASSESSMENT

An important aspect in determining whether a particular fuel cycle or reactor type will be accepted by a utility is its economic benefit. With this motivation, the composite improvement case was evaluated in terms of its fuel cycle costs and compared with those of the reference three-batch PWR. The previously described set of cost assumptions (see Table 5.3-16) were used for the near and long-term economic assessment. The fuel cycle costs were analyzed using EVALU84, a C-E developed fuel cycle cost computer code.

The thirty-year levelized fuel cycle costs for the composite five-batch PWR and the reference three-batch PWR are given in Table 6.5-1. Near-term fuel cycle costs for the improved five-batch PWR (excluding benefits of reinsertion) are 16.4% less than the reference three-batch core. Incorporating fuel reinsertion in the composite improved PWR would further reduce these costs by approximately 0.5%. The UF_6 , fabrication, and back-end direct costs are less for the composite PWR due to the reduced uranium requirements and batch size. The separative work costs (SWU) are less for the five-batch PWR since the reduced uranium batch size offsets the higher enrichment of the fuel in this core. The indirect charges for the composite five-batch PWR are higher as a result of the higher enrichments and longer in-core residency time. In the long term, fuel cycle costs for the composite five-batch PWR are 16.3% less than the three-batch PWR. Thus, there is a fairly strong economic incentive for a utility to adopt the improved PWR design.

Table 6.5-1

THIRTY-YEAR LEVELIZED COSTS FOR PWR FUEL CYCLES (mills/kWh)^a

Fuel Management	UF ₆	SWU	FAB	Back-End Direct ^b	Indirect	Total
Three-Batch OUT-IN	2.47 (6.05)	1.76 (1.76)	0.60 (0.52)	0.58 (0.58)	0.82 (1.53)	6.23 (10.44)
Five-Batch Composite ^c	1.98 (4.84)	1.59 (1.59)	0.41 (0.35)	0.36 (0.36)	0.87 (1.60)	5.21 (8.74)

a. Values without parentheses are for near-term assumptions, while those with parentheses are for the long-term assumptions.

b. Disposal costs.

c. Includes extra cost for fabrication of shimmed assemblies; excludes benefits from reinsertion.

One expense which is not included in the fuel cycle costs shown in Table 6.5-1 but which should be considered is the cost of replacement power during end-of-cycle stretchout. The fuel cycle costs were recalculated for three replacement power cost values and are shown in Table 6.5-2. The fuel cycle cost savings of the composite improved PWR design decreases from 16.4% to 14.9% as compared to the reference three-batch PWR when a replacement power cost of 40 mills/kwh is assumed. Although this expense is fairly small, it could be reduced through judicious schedule planning so that power coastdown and refueling outages coincide with the low power demand periods. The economic penalty of coastdown and the importance of schedule planning will increase as replacement power costs rise.

6.6 ENVIRONMENTAL ASSESSMENT

To assess the environmental impact of employing the composite improved fuel cycle in pressurized water reactors (PWRs), two main areas must be examined: the nuclear fuel support industry and the operation of the reactor itself. The nuclear fuel support industry affects the environment in terms of land, water and energy requirements as well as liquid and solid effluent and radiological releases. The operation of a light water reactor affects the environment through radiological releases during normal operation and accident situations. The changes in the nuclear fuel support industry and the reactor operation which result from implementing the composite fuel cycle are addressed in the subsequent sections in terms of their environmental consequences.

6.6.1 The Nuclear Fuel Support Industry

Deployment of the composite improved fuel cycle in PWRs would alter the operation of the fuel cycle support industries, particularly the fuel supply industries. Due to the reduced U_3O_8 requirements of the composite improved PWR, the demand for uranium mining and milling and UF_6 production would decrease by about 21% averaged over a 30-year reactor lifetime. Similarly, separative work requirements would decrease by approximately 9%. The majority of the environmental impact of the nuclear fuel cycle is associated with these processes. Since the reload batch size

Table 6.5-2

FUEL CYCLE COSTS FOR THE COMPOSITE IMPROVED PWR
INCLUDING REPLACEMENT POWER COSTS^a

Replacement Power Costs (mills/kWhe)	Fuel Cycle Costs (mills/kWhe)
0	5.21
20	5.26
40	5.30
60	5.35

^a
Near Term Cost Assumptions

is 40% smaller than that used in the conventional three-batch PWR fuel cycle, fabrication, fuel transportation, and spent fuel storage requirements would decrease by this amount. Implementation of the composite improved fuel cycle would reduce land use, liquid and solid effluents, as well as radiological effluents associated with these processes and, therefore, would result in a reduced environmental impact compared to the traditional three-batch, once-through fuel cycle.

In light of the higher discharge burnup of the composite improved fuel cycle, two aspects of spent fuel storage must be examined further: the decay heat and criticality of stored fuel. In the short term (i.e., the first few weeks after shutdown), the decay heat production for extended-burnup fuel is essentially identical to that of fuel receiving standard exposures (see Figure 6.6-1), since the great majority of the decay heat produced shortly after shutdown is generated by short-lived fission products which reach saturation levels during the first year of irradiation. Extended burnup fuel does not generate higher decay heat until longer periods after shutdown when the contribution of the long-lived (non-saturating) fission products become more important. However, the decay heat remains far below the short-term value. Since the heat removal requirements of the spent fuel storage pool are based on the short-term decay heat level, extended-burnup fuel does not affect this aspect of spent fuel storage. From the standpoint of transportation of spent fuel, an additional cooling period (on the order of 30 days) will be required prior to shipment to allow the gamma energy and decay heat release to drop to that of the conventional fuel cycle. The second area of concern with the storage of extended burnup spent fuel is the consideration of criticality. Since the reactivity of extended-burnup is generally less than or equal to that of spent fuel discharged at standard burnups, criticality is not expected to be a problem. However, the spent fuel storage racks are designed for the possibility of intentional or inadvertent storage of low-burnup or fresh fuel. With the higher feed enrichment of the five-batch composite core compared to conventional core designs (4.44 wt% vs. 3.00 wt%), some fuel rack redesign may be necessary. Other than changes in fuel storage rack design, the higher discharge burnup of the composite improved fuel cycle would not adversely affect spent fuel storage.

Figure 6.6-1
DECAY HEAT GENERATION vs COOLING TIME PER ASSEMBLY

6.6.2 Reactor Plant Operation

The radiological impact of plant operation with the composite improved fuel cycle will be determined, to a large degree, by the fuel performance and the activity in the reactor coolant system. The environmental impact assessment made here is based on the following two assumptions (1) the composite fuel cycle employs fuel with the equivalent performance of conventional fuel (i.e., no significant increase in the propensity for fuel failures), and (2) the primary coolant activity limits presently used are maintained. To ensure that these assumptions are valid, it will be necessary to develop fuel designs to prevent degradation of fuel performance at extended-burnup values. Issues such as pellet clad interaction, fuel assembly dimensional changes, external clad corrosion and fission gas release are presently being examined in various DOE, EPRI and industry-sponsored research and development programs for extended-burnup fuel. Data from these programs will be available in the early to mid 1980s. The impact of low-leakage fuel management, axial blankets; modified lattice dimensions and end-of-cycle stretchout on the fuel performance of extended-burnup fuel appears to be minimal. No significant new problems or considerations are introduced which would result in basically different performance than the high burnup option. However, a demonstration program of the composite design will be necessary to quantitatively determine the effects of these other improvement options on the performance of extended-burnup fuel.

If these assumptions are valid, the radiological impact due to reactor operation of the composite improved fuel cycle would be negligible. The radiological activity in normal gaseous and liquid releases is determined by the quantity of fuel failures and reactor coolant system activity. Although the extended-burnup of the composite improved fuel cycle results in more long-lived isotopes, it is primarily the short-lived isotopes which contribute to the radiological activity. For the case when accidents occur involving fuel-rod gas gap activity release, the short-lived isotopes with high specific activities are the primary components in site boundary dose. Since these activities are already saturated at standard burnups, no significant increase in radiological impact is expected for the composite improved fuel cycle. This is also true for accidents involving release of reactor coolant system activity, where clad integrity is maintained, since the primary coolant activity is maintained at the same level as that of conventional fuel cycles by proper coolant chemistry and adherence to technical specification limits.

6.7 RD&D REQUIREMENTS FOR THE COMPOSITE IMPROVED FUEL CYCLE

In each of the sections in Section 5, the RD&D requirements to implement the individual improvement options were described. The RD&D requirements presented earlier are summarized here because they form the basis of an integrated development program for the composite improvement design. The extended-burnup, low-leakage fuel management option requires substantial effort in the area of RD&D, particularly with respect to establishing acceptable performance for fuel with exposures significantly greater than conventional burnup values of 30 MWd/kg and for possible advanced burnable poison designs (e.g., those using admixed gadolinia or gadolinia wafers). Furthermore, the potential loss in thermal margin due to higher fuel pin power peaking factors must be investigated and techniques instituted to diminish or to recover these potential losses to prevent core derating. Several additional evaluations should be performed for this improvement option including more extensive fuel management development, further safety studies to determine the effects of core parameter changes (e.g., more negative MTC and smaller control rod worth) on thermal margin, and a quantitative analysis of the radiological concerns for extended-burnup fuel (i.e., decay heat rate, criticality and shipping requirements for spent fuel storage). In addition, a licensing assessment should be made for the extended-burnup, low-leakage fuel management option.

To implement a modified lattice design, detailed thermal-hydraulic and neutronics calculations should be conducted. Thermal hydraulic experiments should be performed on test assemblies for both normal and accident conditions. An irradiation program would be necessary to verify the in-core performance of the modified lattice design; this program should include the design, licensing and fabrication of assemblies which incorporate various design modifications that both improve uranium utilization and fuel performance. For the axial blanket option, the RD&D requirements involve further demonstration of axial blankets (in addition to the Rancho Seco irradiations) in the context of extended-burnup fuel cycles. Assessments of the impact of axial blankets on core thermal margin should be made in light of the higher axial peaking factors. The reinsertion of early batches of the initial core requires the development of explicit loading patterns and the determination of the effect of these loading patterns on core power peaking factors.

Finally, end-of-cycle stretchout requires little research and development. However, to further improve the benefits of stretchout a systematic study of the capabilities and limitations of feedwater pressure augmentation should be investigated. To deploy the composite improved fuel cycle, these individual RD&D efforts would be necessary along with an integrated demonstration of feasibility. Figure 6.7-1 outlines the required RD&D effort and a proposed schedule for completion. The program would take approximately 15-18 years to demonstrate the composite fuel cycle on a full-core basis.

6.8 LIKELIHOOD OF UTILITY ADOPTION

The impetus for utility adoption of the composite improved fuel cycle is fairly high, a 16% reduction in fuel cycle costs coupled with a 21% reduction in uranium requirements and a 40% reduction in spent fuel storage requirements. However, several potential problems exist which could deter utilities from implementing all of these improvements or of implementing them to the degree implicit in the composite core. In particular, the reliability and licensability of the extended-burnup fuel option would have to be demonstrated prior to utility adoption.

Secondly, the potential losses in thermal margin due to high pin power peaking in the composite improved core would have to be quantified and reduced. If substantive R&D programs were initiated and these problems were resolved, a utility would be more inclined to employ the composite improved fuel cycle. In all probability, however, not all improvement options in the composite case will be adopted by all utilities. Rather, utility management will selectively choose improvement options as they become available. It should also be pointed out that some of these improvement options may prove unacceptable after more detailed evaluations and demonstrations due to their adverse effect on thermal margin.

Based on the fuel cycle cost savings and uranium utilization improvement, the five-batch, extended-burnup option is the most likely option to be adopted compared to the other potential improvement options. A utility would probably also include fuel reinsertion and end-of-cycle stretchout in this improved fuel cycle since these options are presently employed in operating reactors (although to a much more limited extent than considered here) and would require minimal effort to license. Substantial uranium utilization improvement results from combining these three options (approximately 15%).

Figure 6.7-1
INTEGRATED RD&D REQUIREMENTS FOR COMPOSITE IMPROVED PWR

Section 7

EFFECT OF ONCE-THROUGH IMPROVEMENT OPTIONS ON THE RECYCLE MODE

7.1 INTRODUCTION

A number of promising options to improve the uranium utilization of the once-through cycle in pressurized water reactors (PWRs) have been identified through the efforts of the Nonproliferation Alternative Systems Assessment Program (NASAP). These options, which were discussed in detail in Section 5 of this report, were selected because they are well defined and appear to be economic, technically sound, and backfittable into existing PWRs. Furthermore, it is felt that these options could be demonstrated during the next decade and could contribute to uranium resource conservation in the near term if sufficient incentive exists to promote them.

The question has arisen as to the impact of these once-through improvement options on the performance of the recycle mode of operation, that is, whether the improvement in uranium utilization from each of these options also carries over to recycle, whether economical fuel cycle costs obtainable from these options carry over to recycle, and whether the incentive for recycle is diminished as a result of the implementation of these once-through improvement options. Consequently, each of the once-through uranium improvement options has been assessed in the recycle mode and the results of this assessment are reported in this section.

The analyses reported here were performed on the C-E System 80TM reactor (1) which is typical of current design PWRs. For the reference case (i.e., the standard lattice and conventional fuel management), cycle-by-cycle calculations were performed to establish the U_3O_8 requirements for this reactor operating on the self-generated recycle mode. For other cases, usually only the equilibrium cycle was investigated and with physics models of varying complexity. The resource requirements and fuel cycle costs were determined under the assumption that recovered uranium would be returned to the diffusion plant for reenrichment. In all calculations, a 75% capacity factor and a 0.2 wt% diffusion plant tails assay were used, and the worth of

recovered uranium was assumed to be reduced by about 10% due to the buildup of U-236 (2).

For the reference case, self-generated recycle of plutonium and residual uranium reduces the equilibrium U_3O_8 requirement by about 38%*, while cumulative uranium requirements averaged over the 30-year life of the reactor are reduced by about 32%. These percentage savings have been adopted as the base line benefit of recycle in the remainder of this report.

7.2 WHEN IMPROVEMENT OPTIONS BENEFIT THE RECYCLE MODE

The uranium utilization of the once-through mode is related directly to the fissile material (i.e., U-235) charged per unit energy production, while for the recycle mode it is related almost exclusively to the conversion ratio. Figure 7.2-1 illustrates a number of considerations which are important in establishing the fissile requirements and hence the resource utilization of a PWR in both modes of operation. This figure shows how the required fissile loading of a UO_2 core operating on the once-through cycle varies as a function of the core average exposure (burnup). The total fissile requirement, which includes both loaded U-235 and the plutonium produced, is represented by the solid line in Figure 7.2-1, and can be viewed as consisting of four components:

1. A fissile inventory component which is required to obtain criticality in an infinite media--this component compensates for parasitic captures in the moderator, clad, and other structurals;
2. A fissile inventory component which is required to account for the loss of neutrons resulting from core leakage;
3. A fissile inventory component which is required to overcome the buildup of fission products; and
4. A fissile depletion component to provide for the consumption of fissile material which occurs in the production of energy.

*This value also represents the net savings of U_3O_8 . The net requirement can be determined by taking the 30-year cumulative requirements less a credit for the ex-core fuel batches and fuel remaining in core after a reactor's final year of operation. It is essentially equal to the equilibrium requirement.

Figure 7.2-1

CORE FISSILE REQUIREMENTS vs END OF CYCLE CORE AVERAGE EXPOSURE
(1270 MWe PWR UO_2 ONCE-THROUGH CYCLE, 75% CAPACITY FACTOR, ANNUAL REFUELING)

Figure 7,2-1 should be interpreted in the following manner. The total fissile loading shown for a particular end-of-cycle (EOC) core average exposure is that fissile loading which is required to obtain an annual operating cycle at a 75% capacity factor. At the end of such an operating cycle, the core has the indicated average exposure.

The fuel inventory required to maintain criticality in an infinite media (Curve A) is determined by the balance between neutron production in fissile material and neutron captures in all core materials. It is a relatively strong function of the lattice design and the composition of materials in the core. In the once-through mode, this fissile inventory is discharged with the spent fuel while in the recycle mode the majority of this material is recovered and used to fuel later cycles.

The fuel inventory required to maintain criticality in an actual reactor (Curve B) includes a component for neutron leakage from the core. The magnitude of this component is dependent on the size of the reactor core, on the fuel management employed (e.g., whether a low leakage fuel management scheme is used), and on the composition and thickness of the core reflector/blanket (if incorporated). As with the previously discussed core inventory, this material is discarded with the spent fuel in the once-through mode, but recovered when the fuel is reprocessed in the recycle mode.

The fuel inventory required to maintain criticality at the end of an operating cycle includes sufficient fissile material to balance the parasitic captures in fission products. This contribution to the required fissile inventory is primarily dependent on the core average exposure, but can be influenced somewhat by the fuel design and resulting neutron spectrum. Again, this inventory is discarded in the once-through mode, but recovered in the recycle mode.

Finally, the total fissile requirement includes a depletion component which depends essentially only on the conversion ratio of the system. If only U-235 were present in the fuel, then about 1 gram of U-235 would be consumed

per megawatt-day energy produced. In such a fuel, the conversion ratio is zero since no fissile material is being produced through conversion of fertile material. For low-enriched uranium fuels typically employed in PWRs, the production and subsequent in-situ utilization of plutonium reduces U-235 consumption by roughly a factor of two. The depletion component can be reduced by increasing the conversion of U-238 into fissile plutonium. This can be achieved by design changes which increase the probability of neutron absorption in U-238. Since U-238 is primarily a resonance absorber, design changes which harden the neutron spectrum (e.g., decreasing the amount of water present in the lattice) are effective. Since the depletion component is consumed during the operating cycle, it must be made up with additional U-235 during refueling.

In summary, when recycle is employed, the end-of-cycle inventory is of no consequence with respect to resource utilization; resource utilization is determined solely by the conversion ratio. The situation is considerably different in the once-through mode. The only parameter affecting resource utilization is the initial enrichment required to achieve a given burnup, i.e., Curve D of Figure 7.2-1. The end-of-cycle inventory and depletion components must be minimized. As with recycle, the resource utilization can be improved by reducing the depletion component through enhanced conversion. However, the conversion ratio is, in general, a poor overall index of the performance in the once-through mode since design modifications which tend to improve the conversion ratio (e.g., reducing the W/F ratio) also increase the required EOC fissile inventory. Most often, optimal performance of the once-through fuel cycle tends to be achieved at the expense of conversion ratio by minimizing end-of-cycle inventory requirements. Thus, in the once-through mode, the most productive area for improving fuel utilization lies in exploring options which reduce the EOC inventory requirement.

It should be noted that the considerations involved in the improvement of fuel cycle costs are somewhat different from those with respect to resource utilization. This is particularly true in the recycle mode. Fuel cycle costs are incurred both due to the depletion of fuel and from carrying charges on the

fuel. Thus, while only the depletion component is important to the resource utilization in the recycle mode, an economic assessment involves a balance between depletion charges and carrying charges on fuel inventories. This competition between depletion charges and fuel inventory carrying charges tends to result in designs which minimize the fissile inventory component at the expense of conversion ratio and resource utilization (i.e., designs which, from a resource utilization standpoint, are more nearly optimal for once-through fuel cycles).

From the above discussion, it is apparent that the once-through mode improvements will have differing effects on the recycle mode depending upon how the conversion ratio is altered. Those once-through improvements which reduce uranium demand by increasing the conversion ratio will result in a similar improvement in the recycle case. Those once-through improvements which do not alter the conversion ratio but obtain their benefit from more completely utilizing the fissile content of the fuel will not alter the uranium requirements in the recycle mode. In this case, the percentage benefits of recycle are reduced since the uranium requirements with recycle do not change, while the uranium requirements of the once-through mode are reduced so that they more closely approach those of recycle. Lastly, if once-through improvements reduce the conversion ratio, uranium requirements in the recycle mode are increased. Benefits in the once-through mode in this case are derived from the more complete utilization of fissile material which proves to outweigh the adverse effect of reduced conversion ratio. If such once-through improvements were employed in the recycle mode, resource utilization would be degraded.

Since various once-through improvement options will have differing effects on the conversion ratio, it is necessary to address these improvement options explicitly so that the effects on conversion ratio and subsequent resource utilization in the recycle mode can be established. It is useful to distinguish between those improvement options which are likely to be demonstrated in the near term and which are viewed to have acceptable safety and economic characteristics from the less developed, longer term options so that improvements in the uranium utilization of the recycle mode can be similarly classified.

The once-through improvement options whose impact on the recycle mode will be discussed in detail in this section are listed in Table 7.2-1. The benefits attributed to each option are those for the equilibrium cycle. It is felt that these are the most appropriate in a discussion involving recycle. Each option on this list is considered to be technically and economically sound and demonstrable within the next decade. All items on this list have, at least to some degree, already been demonstrated in operating PWRs, or R&D programs for their demonstration have been initiated.

7.3 EFFECT OF INDIVIDUAL IMPROVEMENT OPTIONS

7.3.1 Increased Burnup

As indicated in Table 7.2-1, increasing the discharge burnup to about 50,000 MWd/t and employing a five-batch, OUT-IN, fuel management scheme is expected to reduce uranium requirements in the once-through mode by about 12%. The evaluations performed as part of this study indicate that the use of such extended burnup schemes in the recycle mode leads to a lower conversion ratio and thus to a decrease in uranium utilization. This can be seen in Figure 7.3-1 which shows the instantaneous conversion ratio as a function of cycle exposure for the standard cycle and the five-batch extended burnup cycle. Because of the higher fission product absorption in the extended burnup cycle, the conversion ratio of this cycle (averaged over the entire cycle) is about 2.5% lower than that of the standard cycle. The higher fission product absorption of the extended burnup cycle is quantified in the neutron balance given in Table 7.3-1. Averaged over the cycle, 2% more of the total neutrons are lost to parasitic captures in fission products in the extended burnup cycle than in the standard cycle. Captures in other non-fuel core materials are reduced somewhat in the extended burnup cycle because of the harder neutron spectrum which exists; however, the dominant effect is the increased fission product absorption.

Based on the conversion ratios shown in Figure 7.3-1, a 5-6% higher uranium requirement would be expected in the recycle mode for the extended burnup fuel cycle compared to the standard SGR cycle. This higher requirement would be mitigated slightly because less uranium would be recycled to the enrichment plant in the extended burnup cycle and therefore less U-235 would be

Table 7.2-1

NEAR TERM IMPROVEMENTS IN THE PWR ONCE-THROUGH CYCLE

	Reduction in Equilibrium U_3O_8 Requirements, %
1. Extended Burnup, Modified Fuel Management	12.0
2. Lattice Optimization	3.0
3. Axial Blankets	2.5
4. Full Use of Early Batches of Initial Core	0.9
5. Low-Leakage Fuel Management	2.6
6. End-of-Cycle Stretchout	3.0
Total*	22.0

* Percentage reductions in U_3O_8 requirements are not directly additive. The reduction, R_T , can be approximated by

$$R_T = \left[1 - \left(1 - \frac{R_1}{100} \right) \left(1 - \frac{R_2}{100} \right) \dots \left(1 - \frac{R_i}{100} \right) \right] \times 100$$

where R_i is the improvement from each option. Explicit calculations of the total improvement which might be realized by implementing these options in a single reactor are in progress.

Figure 7.3-1
CONVERSION RATIO vs CYCLE EXPOSURE
(SYSTEM 80TM PWR)

Table 7.3-1

NEUTRON BALANCE FOR THE UO_2 ONCE-THROUGH PWR^a

ISOTOPE	BEGINNING OF CYCLE ^b		MIDDLE OF CYCLE		END OF CYCLE	
	3-Batch 30 Gwd/t	5-Batch 50 Gwd/t	3-Batch 30 Gwd/t	5-Batch 50 Gwd/t	3-Batch 30 Gwd/t	5-Batch 50 Gwd/t
U-235	0.3242	0.2988	0.2638	0.2568	0.2159	0.2251
U-236	0.0038	0.0064	0.0052	0.0077	0.0064	0.0086
U-238	0.2752	0.2550	0.2748	0.2566	0.2767	0.2589
Pu-239	0.1359	0.1546	0.1877	0.1875	0.2233	0.2118
Pu-240	0.0294	0.0390	0.0406	0.0467	0.0529	0.0544
Pu-241	0.0157	0.0251	0.0250	0.0328	0.0372	0.0411
Pu-242	0.0006	0.0016	0.0011	0.0022	0.0020	0.0029
Hydrogen	0.0377	0.0302	0.0383	0.0313	0.0401	0.0330
Oxygen	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032
Soluble Boron	0.0700	0.0544	0.0373	0.0293	0.0008	0.0006
Zirconium	0.0120	0.0112	0.0120	0.0113	0.0121	0.0115
Iron	0.0006	0.0005	0.0006	0.0005	0.0007	0.0005
Samarium	0.0042	0.0048	0.0046	0.0048	0.0046	0.0048
Xenon	0.0193	0.0175	0.0202	0.0186	0.0203	0.0189
Fission Products	0.0317	0.0525	0.0466	0.0660	0.0635	0.0795
Axial Leakage	0.0114	0.0125	0.0120	0.0122	0.0124	0.0127
Radial Leakage	0.0251	0.0326	0.0271	0.0324	0.0281	0.0324
Core Average Exposure, MWd/t	9,933	20,442	14,868	25,372	20,200	30,146
Boron Concen- tration, ppm	969	940	509	490	10	10

a. Equilibrium Cycle: 75% capacity factor with annual refueling

b. With equilibrium xenon

lost in the tails stream during the re-enrichment process. Evaluations performed on the recycle mode confirm this higher uranium requirement in the recycle mode for the extended burnup cycle. A summary of these evaluations is presented in Table 7.3-2. As this table illustrates, the 30-year cumulative uranium requirements in the recycle mode are about 4% higher for the 50,000 MWd/t extended burnup case compared to the standard 30,000 MWd/t case. Similarly, the net requirements taking credit for ex-core batches and final core inventories are about 5.5% higher for the extended burnup cycle. The benefits of recycle are thus reduced from the 32% cumulative (38% net) reduction of the reference SGR case to 19% cumulative (26% net) for the 50,000 MWd/t extended burnup case primarily because of the improved performance of extended burnup in the once-through mode of operation. A similar reduction of the benefit of recycle is seen in terms of separative work requirements when going from the reference case to the 50,000 MWd/t extended burnup case.

Uranium utilization in the closed cycle could be improved if the discharge burnup were decreased below current values, but economic considerations would limit the practicality of this approach. Reducing the discharge exposure results in the more frequent need for fuel reprocessing and mixed-oxide fabrication. Since both reprocessing and mixed-oxide fabrication are expensive, economic considerations generally preclude such reduced discharge exposures. In fact, economic evaluations, which are summarized in Table 7.3-3, indicate that 50,000 MWd/t burnup is more economic than current burnup values in the recycle mode for uranium prices up to about \$100/lb U_3O_8 . Savings from the reduced requirements for reprocessing and mixed-oxide fabrication are larger than the cost of the additional U_3O_8 required at higher burnup. Thus, reprocessing and mixed-oxide fabrication costs would have to drop dramatically before reducing the discharge burnup below current values would prove economical. From the results emerging from NASAP, such dramatic decreases in reprocessing and mixed-oxide fabrication costs should not be expected.

Consequently, it can be concluded that the once-through fueling mode improvement in uranium utilization derived from extended burnup does not result in

Table 7.3-2

EFFECT OF EXTENDED BURNUP/MODIFIED FUEL MANAGEMENT ON URANIUM UTILIZATION

	Reference Case (3-Batch, 30 Mwd/kg)		Extended Burnup (4-Batch, 40 Mwd/kg)		Extended Burnup (5-Batch, 50 Mwd/kg)	
Uranium Requirements	ST/ GWe	% Change From O-T	ST/ GWe	% Change From O-T	ST/ GWe	% Change From O-T
Once-Through Mode						
Initial Core	394	-	402	-	408	-
Equilibrium Cycle	194	-	178	-	171	-
30-Year Cumulative	6040	-	5600	-	5340	-
30-Year Net	5820		5345		4147	
Recycle Mode						
Initial Core	394	0	402	0	408	0
Equilibrium Cycle	120	-38	121	-32	127	-26
30-Year Cumulative	4130	-32	4200	-25	4300	-19
30-Year Net	3600	-38	3640	-32	3800	-26
Separative Work	MT SWU/ GWe	% Change From O-T	MT SWU/ GWe	% Change From O-T	MT SWU/ GWe	% Change From O-T
Once-Through Mode						
Initial Core	200	-	206	-	216	-
Equilibrium Cycle	117	-	116	-	118	-
30-Year Cumulative	3600	-	3590	-	3606	-
30-Year Net	3510		3480		3540	
Recycle Mode						
Initial Core	200	0	206	0	216	0
Equilibrium Cycle	81	-31	86	-26	95	-19
30-Year Cumulative	2740	-24	2910	-19	3093	-14
30-Year Net	2430	-31	2580	-26	2850	-19

Table 7.3-3

PWR FUEL CYCLE COST^a
(mills/kWh)

Cycle	Burnup (MWd/t)	30-Yr. Levelized Fuel Cycle Cost (mills/kWh)			
		\$40/lb U ₃ O ₈		\$100/lb U ₃ O ₈	
		Reference	Range	Reference	Range
PWR Once-Through, Three Batch	30,400	6.22	(5.97-6.45)	10.58	(10.33-10.81)
PWR Once-Through, Five-Batch	50,650	5.66	(5.53-5.80)	9.66	(9.53- 9.79)
PWR Pu/U Self-Generated Recycle, Three-Batch	30,400	5.43	(5.01-6.20)	8.19	(7.76- 8.96)
PWR Pu/U Self-Generated Recycle, Five-Batch	50,650	5.14	(4.92-5.53)	8.25	(8.03- 8.64)

a. Unit cost assumptions are given in Table 7.3-4

Table 7.3-4

UNIT COST ASSUMPTIONS

	Reference Value	(Range)
U_3O_8 Price, \$/lb . U_3O_8	40/100	
Conversion Cost, \$/kg . U	4	
Enrichment Cost, \$/kg . SWU	100	
Fabrication Cost, \$/kg . HM		
LWR LEU Fuel	110	
LWR Recycle Pu/U Fuel	370	(280-540)
Throwaway Cost, ^a \$/kg . HM	135	(70-200)
Recovery Cost, ^b \$/kg . HM	270	(180-430)

a. The throwaway cost is the sum of spent fuel shipping charges and the permanent spent fuel storage charge.

b. The recovery cost includes reprocessing, shipping, and waste storage costs.

a similar improvement in the recycle mode; the uranium requirements in the recycle mode increase by about 5%, while the percentage reduction in cumulative (net) uranium requirements realizable with recycle as compared to once-through fueling is reduced to 19% (26%). Furthermore, independent optimization of uranium requirements by changing the burnup in the recycle mode does not appear attractive since this is achieved by decreasing the discharge exposure which would result in economic penalties due to increased reprocessing and mixed-oxide fabrication requirements.

7.3.2 Low-Leakage Fuel Management

Evaluations of low-leakage fuel-management schemes indicate that the once-through uranium utilization benefit is derived almost solely from a reduction in core leakage, which impacts the EOC fissile inventory required to sustain criticality. Detailed two-dimensional calculations indicate that the conversion ratio of the low-leakage fuel-management configuration is only slightly larger (0.005) than the conversion ratio of the equivalent standard fuel management scheme. Based upon this change in conversion ratio, it is estimated that low-leakage configurations will reduce uranium requirements by about 1.2% in the recycle mode as compared to about 2.6% in the once-through mode.

7.3.3 Lattice Optimization

The improvement in the uranium utilization of the once-through cycle attributable to lattice optimization stems from design changes which increase the water-to-fuel volume ratio. As discussed in Section 5.4, the reduction in uranium requirements shown in Table 7.2-1 for lattice optimization is not the largest which can, in principle, be obtained with lattice changes; rather, it is the gain anticipated from design changes which are retrofittable into existing reactors with the further constraint that moderator temperature coefficients of reactivity must remain negative. Such increases in the water-to-fuel ratio can be obtained by either decreasing fuel rod diameter slightly or by employing annular fuel pellets (annular fuel pellets have a void central region, thereby reducing the fuel volume). Since either option does not change the overall dimensions of the fuel assembly, they can be deployed in existing reactors. The small reduction in heat transfer area resulting from reducing rod diameter is not expected to necessitate a reduction in reactor power output; the heat transfer

area is not altered when annular fuel is used. Reductions in fuel rod diameter are currently being implemented into a number of PWRs while annular fuel has been irradiated in BWRs and further irradiations in PWRs are planned.

Evaluations of modified lattice geometries indicate that such increases in water-to-fuel ratio would decrease the conversion ratio of the recycle core by about 1.5 - 2% as indicated in Figure 7.3-2. The reduced conversion would increase uranium requirements in the recycle mode of operation by about 3-4%. Since increasing the water-to-fuel ratio reduces the uranium requirements of the once-through fueling mode by 3% while increasing the uranium requirements of the recycle mode by 3%, this design change if incorporated into the recycle mode would decrease the resource incentive for recycle by about 6%.

In principle, uranium requirements in the recycle mode can be reduced by design changes which reduce the water-to-fuel ratio. Although it is clear that such reductions in water-to-fuel ratio are technically feasible in the context of new reactor designs, it is doubtful whether these design changes can be back-fitted into existing or committed reactors without reducing power output and thus incurring considerable cost penalties. In particular, thermal margins are expected to decrease due to the increased pressure drop across the core and the resulting decrease in coolant flow. This would necessitate either a power derating or a modification of the primary coolant pumps to increase coolant flow and of the reactor internals to accommodate the higher fuel assembly uplift forces resulting from that increased flow. The LOCA analysis will also be adversely affected since it will be more difficult to cool and reflood the fuel because of the closer packing of fuel rods. Lastly, although control rod worths will decrease as a result of reducing the water-to-fuel ratio, the more negative moderator temperature coefficient of reactivity will require additional control rod worth. The reactor would consequently have to be modified to increase the number of control rod poison fingers; some modification to reactor internals would also be required so that control rods would insert into the new guide tube locations of the modified fuel assembly design.

Figure 7.3-2
CONVERSION RATIO vs WATER-TO-FUEL RATIO
(PWR RECYCLE MODE)

An additional deterrent to modifying the fuel lattice in the direction of lower water-to-fuel ratio is poorer economics. As a result of the lower water-to-fuel ratio, core fissile material inventories will increase, as indicated in Figure 7.3-3, compromising fuel cycle economics through increased carrying charges. The economics of plutonium recycle presently favor lattices with higher water-to-fuel ratios than are used for current design PWRs. This can be seen in Figure 7.3-4 which shows the change in power costs as a function of water-to-fuel ratio. At current U_3O_8 prices (approximately \$40/lb), the economic optimum for a PWR employing recycle is at a water-to-fuel ratio of approximately 2.3. It should be noted that the design of most modern LWRs was developed with the intent of eventually utilizing the recycle fueling mode, so it should not be surprising that lattice changes which improve uranium utilization in the recycle mode are at the expense of economic performance.

Consequently, it appears doubtful that such reductions in water-to-fuel ratio could be retrofitted into existing or committed reactors without reduced power output and considerable cost penalties, and further that fuel cycle economics will not favor this fueling mode for some time until uranium prices have risen significantly. It is therefore concluded that this design change is unlikely to prove attractive, but at any rate cannot be employed in the near term and hence can be considered only as a long-term option.

7.3.4 Axial Blankets

As discussed in Section 5.5, the use of axial blankets improves the uranium utilization of the once-through cycle by replacing the under-utilized enriched fuel at the ends of the fuel rod with fertile material. By employing this strategy, neutron leakage from the core is decreased, fertile conversion through captures in the blanket material is increased, and the exposure on the enriched active core fuel is increased. In the recycle mode, only the second of these effects is important from a fuel resource standpoint. The first effect allows the end-of-cycle fissile inventory to be lower, which is important from an economic point of view, but not from the standpoint of resource utilization. The third effect merely represents a relocation of the fission products in the core. Again, this can have a beneficial effect on

Figure 7.3-3
FISSILE REQUIREMENTS vs WATER-TO-FUEL RATIO
(MIXED OXIDE FUEL)

Figure 7.3-4
CHANGE IN POWER COST vs WATER-TO-FUEL RATIO
(PWR RECYCLE MODE)

the required EOC fissile inventory, but since the total inventory remains the same (for equal energy extraction), it has no effect on the neutron economy of the core. The conversion ratio of a core with 15 cm axial blankets is compared to that of a core without a blanket in Figure 7.3-5. It can be seen that only a small increase in conversion ratio is achieved if the total fuel rod length is limited to that of the reference unblanketed rod (i.e., 381 cm). In this case, the conversion ratio is increased only 0.005, which represents an improvement of about 1.2% in uranium utilization for the recycle mode. This benefit is only about half that predicted for the once-through fueling mode. The difference in predicted benefits is due to the fact that, in the once-through mode, reductions in required fissile inventory impact directly the resource requirements, while in the recycle mode they have essentially no impact.

The conversion ratio of a core which has blanket regions added onto the standard fuel rod is also shown in Figure 7.3-5. In this case there is a greater benefit for the recycle mode. However, since this blanket option could not, in general, be incorporated into existing reactor designs because of the increased fuel length, it is considered to be a long-term non-backfittable option. This particular option is discussed further in Section 7.4 along with other selected non-backfittable options.

7.3.5 Full Use of Early Batches

The full use of early batches of the startup core will not result in a reduction in U_3O_8 requirements in the recycle mode. Two factors contribute to this consequence. First, reinserting this partially burned fuel results in the accumulation of additional exposure; therefore, this once-through improvement option suffers the effect in the recycle mode of increased burnup (cf. Section 7.3.1). Secondly, inserting these early core batches in later cycles delays the recycle of plutonium. This delay in reaping the resource benefits of recycle increases slightly the 30-year cumulative uranium requirements; it has little, if any, impact on the net requirements.

Figure 7.3-5
CONVERSION RATIO vs CYCLE EXPOSURE
(AXIAL BLANKET STUDY FIVE-BATCH, EXTENDED BURNUP CORE)

7.3.6 End-of-Cycle Stretchout

End-of-Cycle coastdown has been used in the past primarily as a means of obtaining added cycle length flexibility when unforeseen factors make it desirable to do so (e.g., because of the forced outage of another unit). In the once-through mode, however, there is both a fuel resource conservation incentive and an economic incentive to practice planned cycle stretchout. As shown in Table 7.2-1, a fuel resource savings of about 3% can easily be achieved through stretchout; a similar percentage savings in cycle energy costs can be obtained by employing this operating strategy in an economic optimum fashion.

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% 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.

7.4 SELECTED NON-BACKFITTABLE IMPROVEMENT OPTIONS

Selected non-backfittable uranium improvement options that are relatively well defined and appear technologically feasible have been included in this discussion. In general, implementation of these options would be relatively straightforward, although substantial changes from present reactor design may be required. These options would, most likely, be economic in the long term if the price of U_3O_8 increases beyond \$100/lb. A number of other non-backfittable uranium improvement options have been left out of the present discussion, e.g., use of a low-power-density core. These omitted options

either have not been analyzed in sufficient detail to assess their commercial potential or are thought to represent more significant departures from present PWR design.

7.4.1 Increasing Fuel Rod Length With Axial Blankets

The axial blanket option discussed in Section 7.3.4 was considered backfittable in existing PWR designs since the blanket materials displaced enriched fuel at the ends of the rod while the overall length of the fuel remained unchanged from the reference design. An alternative application of axial blankets which is often envisioned but which is not backfittable consists of adding blanket material at either end of the enriched region of the fuel rod so that the rod becomes longer and neutron leakage is reduced. In this case, blankets would lead to higher uranium savings with recycle than in the once-through cycle. In the once-through cycle, this application of axial blankets would result in about a 2% reduction in uranium requirements. Because of the larger integrated conversion ratio (0.59, as shown in Figure 7.3-5) which results when longer fuel rods are employed, the U_3O_8 savings in the recycle mode is estimated to be about 3.5%. Since this axial blanket option has slightly less potential in the once-through mode than the previously discussed backfittable option, it would most likely not be employed in the once-through mode. Use of this axial blanket option would, however, benefit the recycle mode and therefore has a higher probability of adoption in this mode. It is felt that this axial blanket option would be applicable only in the long term when new reactor designs could be developed to accommodate the longer length fuel.

7.4.2 Increased Frequency of Refueling

Increased frequency of refueling will result in an improvement in the recycle mode about half as large as that realized when once-through fueling is employed. Analyses for the extended-burnup, self-generated recycle mode have shown that decreasing the refueling interval from 12 months to 10 months reduces the

30-year cumulative U_3O_8 requirement by about 1.8%. From these results, it is estimated that reducing the refueling interval to 6 months would result in about a 4.5 - 5.0% reduction in U_3O_8 requirements. Since increasing the frequency of refueling is considered to be non-economic in the context of existing designs which have not been designed explicitly for rapid refueling, the potential for reducing uranium requirements of the once-through fuel cycle by more frequent refuelings has not been included in the near-term improvement options. As a longer term non-backfittable option, more frequent refuelings could contribute to improved uranium utilization for both once-through fueling and for the recycle mode. Economic evaluations have indicated that refueling times on the order of one week would be required to make six-month refuelings attractive commercially.

7.4.3 Other Non-Backfittable Improvement Options

There are a number of other non-backfittable improvement options which have been identified by various investigators within NASAP. These options have not been investigated as part of the present study, but are mentioned for completeness. The details of these evaluations are reported elsewhere as indicated.

7.4.3.1 Variable Lattice Design

The variable lattice design concept involves altering the water-to-fuel ratio present in the fuel assembly at each refueling outage(3). The assembly is designed with a relatively low water-to-fuel (W/F) ratio which is increased by removing a fraction of the fuel rods after an initial period of operation. Anticipation of this change in W/F ratio allows the initial ratio to be set at a lower value than would be desirable or optimum for a fixed lattice design. This lower W/F ratio increases the conversion ratio and reduces the initial excess reactivity of the assembly. As a result of the improved conversion ratio during the initial period of operation, the subsequent increase in W/F ratio can result in larger reactivity increases than could otherwise be obtained by the same change. Thus, the variable lattice design offers the potential of improved fuel utilization by increased production of fissile material from fertile material and subsequent improved utilization of that fissile material in an optimized lattice. The potential reduction in uranium

requirements is estimated to be on the order of 4% in the once-through mode and 8% in the recycle mode. Certain applications of this technique may be considered backfittable, while others would not be. In either event, it is relatively long term in nature because of the requirement to demonstrate and/or verify various safety, licensing, environmental, and material considerations related to its use.

7.4.3.2 Movable Fuel and/or Fertile Control

Use of movable fuel and/or fertile control has the potential for eliminating the neutron losses associated with poison control (i.e., soluble boron control in a PWR.) This mechanism can significantly enhance the conversion ratio of a PWR, resulting in large fuel resource benefits. This technique embodies the basic design features of the Light Water Breeder Reactor (LWBR) being demonstrated in the Shippingport Reactor(4). Additional features of the LWBR are its use of different lattice designs in the movable seed and stationary blanket regions and the use of U-233/thorium fuel. If a uniform lattice design were employed with the uranium fuel cycle, then U_3O_8 savings of about 12% in the once-through mode and 20% in the recycle mode could be obtained with this concept. More substantial design changes, such as those included in the LWBR, would result in significantly greater gains; in general, the potential for these more substantial gains are limited to the recycle mode.

7.5 EFFECT OF COMPOSITE IMPROVEMENTS ON THE RECYCLE MODE

The effect of various improvement options on uranium utilization in the PWR is summarized in Table 7.5-1. Percentage improvements, potentially realizable in the near and long term, are tabulated based upon the information provided in the previous sections. The benefits attributed to each option are those for the equilibrium cycle. It is felt that these are the most appropriate in a discussion involving recycle. In developing this table for the near term, three cases are shown for the recycle mode: one case assumes that all once-through improvement options are included in the recycle mode; the second case assumes that those once-through options which degrade the resource utilization performance in the recycle mode are not included; finally, the third case assumes that only those options which are economic and therefore commercially desirable are included. For example, it has been assumed that extended discharge

Table 7.5-1

EFFECT OF IMPROVEMENT OPTIONS IN THE PWR ON URANIUM UTILIZATION

Percent Improvement in Equilibrium Cycle

	Near Term ^c				Long Term	
	Once-Through ^e	Recycle ^f			Once ^e Through	Recycle ^f
		With All O-T Options ^b	With Only Resource Eff. Options	With Only Economic Options		
1. Increased Burnup/Modified Fuel Management	12	-4	0	-4	12	0
2. Lattice Changes	3	-2.5	0	-2.5	3	5 ^d (0)
3. Enrichment Zoning/Blankets	2.5	1.2	1.2	1.2	2.5	3.5 ^d (1.2)
4. Full Use of Early Batches of Startup Core	0.5	-0.2	0	0	0.5	0
5. Improved Fuel Management	2.6	1.2	1.2	1.2	2.6	1.2
6. End of Cycle Coastdown	3	-0.5	0	0	3	0
7. Increased Refueling Frequency	0	0	0	0	9 ^d (0)	4.5 ^d (0)
8. Variable Lattice Design	0	0	0	0	4	8
9. Movable Fuel and/or Fertile Control	0	0	0	0	12 ^d (0)	20 ^d (0)
TOTAL ^a	22	-4.5	2	-4.0	40 ^d (25)	36 ^d (10)

a. Percentage reductions in U_3O_8 requirements are not directly additive. The reduction, R_T , can be approximated

$$\text{by } R_T = \left[1 - \left(1 - \frac{R_1}{100} \right) \left(1 - \frac{R_2}{100} \right) \dots \left(1 - \frac{R_i}{100} \right) \right] \times 100, \text{ where } R_i \text{ is the improvement from each option.}$$

b. This column includes the effect on the recycle mode of incorporating all once-through (O-T) improvement options even though they may degrade the resource utilization performance in the recycle mode.

c. All near-term options are backfittable.

d. Includes selected non-backfittable options; values in parenthesis represent backfittable improvement options.

e. As compared with reference three-batch core

f. As compared with reference recycle core.

exposure was employed in the first case even though a resource penalty occurs with recycle, but, in the second case, the standard discharge exposure was employed. Since fuel cycle economics favor higher burnups even in the recycle mode, the extended discharge exposure option was included in the third near-term case. In the long term, more substantial modifications, some of which are non-backfittable into existing reactors, are considered for incorporation into the improved design. In each case, the non-backfittable options are clearly identified as such and separate composite improvements are specified for the case where only backfittable options are incorporated and for the case where non-backfittables are also included. It should be noted that for some of the long-term options, analyses have not progressed to the same detail as for the near-term options; however, the values shown for these improvement options are considered to be reasonable estimates of their potential. Furthermore, since only selected non-backfittable options are included in the long-term cases shown in Table 7.5-1, the total composite improvements are not the maximum possible. If additional options are incorporated, it may be possible to realize larger total savings.

In the near term, it appears that the maximum composite improvement in uranium utilization for the recycle mode is about 2%. This is obtained when only the resource efficient options for the recycle mode are included in the composite design. If economic considerations are included in the selection process (as would be the case for commercial deployment), then the recycle mode would suffer about a 4% reduction in uranium utilization compared to the reference recycle case as a result of incorporating the economically attractive improvement options.

In the long term, the range of potential composite improvement values for the recycle mode is larger than it is in the near term, reflecting the impact of including non-backfittable options. The major non-backfittables considered for the long term are reduced water-to-fuel ratio, longer fuel to accommodate axial blankets without reducing the active fuel region, increased

refueling frequency through the use of rapid refueling designs, and the use of movable fuel/fertile control. Inclusion of some or all of these non-backfittables (depending on whether they benefit the particular fueling mode being considered) increases the potential uranium composite improvement to 40% for the once-through mode and 36% for the recycle mode. Although detailed economics evaluations have not been performed for the long-term uranium utilization improvement options, evaluations performed thus far indicate that, in general, the price of U_3O_8 would have to increase substantially (above about \$100/lb) before there would be sufficient economic incentive to implement most of these options.

Table 7.5-2 summarizes the 30-year U_3O_8 requirements for the reference case, near-term improvements, and long-term improvements for both the once-through and recycle modes. The 32% cumulative improvement (38% net improvement) due to recycle in the reference case is seen to be reduced to a 14% (22% net) advantage when near-term improvements are implemented in both the once-through and recycle refueling modes. If all once-through improvements are included in the recycle mode, this value drops to 8% (17% net) because of the degradation in resource utilization efficiency of some of these options when recycle is employed. When only those options which are economic in the recycle mode are included, the advantage in uranium utilization of recycle over the once-through cycle is only 9% (17% net). In the longer term, U_3O_8 requirements for recycle might be about 27% (34% net) lower than those of once-through fueling.

Consequently, it is concluded that the improvements identified for near-term demonstration--those improvements which are well defined and economic, and which are retrofittable into existing PWRs--can obtain a substantial fraction of the uranium utilization improvement possible with recycle. These near-term improvements will reduce the difference in uranium utilization between the open and closed cycles from about 32% (38% net) to about 8-14% (17-22% net). From a resource improvement standpoint, little of the R&D required to implement these near-term improvements will be directly applicable to the recycle mode. However, from an economics standpoint, much of the once-through

Table 7.5-2

30-YEAR CUMULATIVE U₃O₈ REQUIREMENTS^a

	Once Through ST/GWe	Recycle ST/GWe	Percentage Gain With Recycle
Reference Case	6040/5840	4130/3603	-32/-38
Near-Term Improvements ^b	4789/4500	4047/3530 ^c (4315/3768) ^d [4295/3748] ^e	-15/-22 ^c (-10/-16) ^d [-10/-17] ^e
Long-Term Improvements	4494/4345 ^f (3599/3480) ^g	3708/3246 ^f (2643/2305) ^g	-17/-25 ^f (-27/-34) ^g

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- a. The first number shown for each table entry is the 30-year cumulative requirement while the second number is the 30-year net requirement.
- b. All near-term options are backfittable.
- c. These table entries include only once-through improvement options which are resource efficient in the recycle mode.
- d. These table entries include the effect on U₃O₈ requirements of incorporating all once-through improvement options in the recycle mode.
- e. These table entries include the effect on U₃O₈ requirements of incorporating all economic once-through improvement options in the recycle mode.
- f. These table entries include only backfittable improvement options.
- g. These table entries incorporate selected non-backfittable improvement options.

R&D effort will benefit the recycle mode. In the longer term, it may be possible to develop additional improvements (new reactor designs) to further reduce uranium requirements. These long-term improvements will, in general, benefit both the once-through and recycle refueling modes.

REFERENCES

1. Combustion Engineering Standard Final Safety Analysis Report (CESSAR) Docket Number STN-50-470.
2. NUREG-2002, Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed-Oxide Fuel in Light Water Cooled Reactors, Volume 4, August 1978.
3. T. R. Robbins, "Preliminary Evaluation of a Variable Lattice Fuel Assembly and Reactor Design Concept," ORNL/SUB-79/13571/1, Pickard, Lowe & Garrick, Inc., February 1979.
4. Shippingport Atomic Power Station - Safety Analysis Report for the Light-Water Breeder Reactor.

Appendix A

REACTOR FUEL CYCLE CHARGE AND DISCHARGE INVENTORY DATA

In this appendix, the material inventory data for the PWR fuel cycles examined by C-E as part of the uranium utilization improvement program are presented. This data is given as year-by-year charge and discharge data for the following six fuel cycles:

- a. UO_2 once-through, three-batch (30,400 MWd/t), OUT-IN fuel management cycle;
- b. UO_2 once-through, five-batch (50,650 MWd/t), OUT-IN fuel-management cycle;
- c. UO_2 once-through, five-batch (53,660 MWd/t), low-leakage fuel-management cycle;
- d. UO_2 once-through composite cycle;
- e. UO_2 self-generated recycle, three-batch (30,400 MWd/t), OUT-IN fuel management cycle; and
- f. UO_2 self-generated recycle, five-batch (50,650 MWd/t), OUT-IN fuel management cycle.

Data have been presented in this format to be consistent with the systematics developed during the Nonproliferation Alternative Systems Assessment Program (NASAP) and to facilitate use in future nuclear energy system economic and resource utilization analyses. All the reactor charge and discharge data are given in kilograms for a 1270 MWe PWR. Empty spaces on data sheets assume the values last stated in years one through thirty.

Fuel cycle f. listed above strictly refers to a five-batch, extended-burnup, OUT-IN fuel-management cycle. The inclusion of the other once-through improvement options into the recycle mode has only a minor effect on overall resource requirements and material inventories. Therefore, detailed year-by-year data were not generated for other self-generated recycle (SGR) cases which include various combinations of the once-through options (e.g., the SGR case where only economics options are included). The results determined for these other cases were obtained from equilibrium analyses only and consequently year-by-year data are not available for them.

Table A-1a
 UO_2 ONCE-THROUGH, THREE-BATCH FUEL MANAGEMENT SCHEME
 CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Total Heavy Metal</u>
1	2195.5	97123.0	99318.5
2	1081.4	33034.6	34116.0
3	1036.0	33506.5	34542.5
4-30	1023.2	33092.8	34116.0

Table A-1b
 UO_2 ONCE-THROUGH, THREE-BATCH FUEL MANAGEMENT SCHEME
 DISCHARGE DATA BY YEAR
 (kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1	264.0	55.4	32887.2	136.2	41.5	20.9	4.0	33409.2
2	217.3	81.2	30684.5	146.1	57.8	31.4	9.6	31227.9
3	219.5	110.6	31444.3	167.1	71.2	40.7	15.1	32038.5
4	291.8	127.0	32288.3	174.9	71.0	42.1	15.0	33649.1
5-30	272.8	127.2	32312.0	180.0	70.4	44.0	15.2	33021.6

Table A-2a
 UO_2 ONCE-THROUGH, FIVE BATCH OUT-IN FUEL MANAGEMENT SCHEME
 CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Total Heavy Metal</u>
1	2514.6	96975.8	99490.4
2	809.1	19675.0	20483.9
3	809.1	19675.0	20483.9
4-30	880.8	19603.1	20483.9

Table A-2b
 UO_2 ONCE-THROUGH, FIVE-BATCH OUT-IN FUEL MANAGEMENT SCHEME
 DISCHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1	139.4	35.1	19879.5	82.8	26.9	13.2	2.9	20179.8
2	127.2	50.6	19681.2	93.2	37.5	19.9	6.3	20015.9
3	117.8	70.2	19847.2	103.1	47.5	26.6	11.3	20223.7
4	77.0	69.3	16434.6	89.1	45.9	26.2	13.7	16755.8
5	104.60	94.0	18701.5	107.1	54.4	32.1	16.5	19110.2
6	154.6	110.0	19021.3	115.9	54.2	33.3	14.7	19504.0
7	129.2	112.3	18955.3	115.2	57.20	34.8	17.3	19421.3
8-30	151.0	122.2	18881.5	119.31	57.5	35.7	16.9	19384.1

Table A-3a
 UO_2 ONCE-THROUGH, FIVE BATCH LOW-LEAKAGE FUEL MANAGEMENT SCHEME
 CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Total Heavy Metal</u>
1	2456.3	97005.2	99461.5
2	769.2	18905.9	19675.1
3	801.5	19162.9	19964.4
4	848.1	18768.3	19616.4
5	851.5	18765.9	19617.4
6	854.1	18474.0	19328.1
7-30	857.9	18470.2	19328.1

Table A-3b

UO₂ ONCE-THROUGH, FIVE-BATCH LOW-LEAKAGE FUEL MANAGEMENT SCHEME
DISCHARGE DATA BY YEAR
(kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1	129.8	34.3	19873.4	82.7	28.2	14.4	3.6	20166.4
2	124.5	49.4	19697.5	90.0	36.3	19.5	6.5	20023.7
3	105.5	73.2	17996.7	86.9	42.2	23.5	10.8	18338.8
4	111.8	78.2	18145.3	87.3	42.0	23.5	10.6	18498.7
5	75.0	93.6	17651.4	88.3	49.3	28.0	18.0	18003.6
6	133.9	142.3	17905.6	99.2	49.2	29.4	15.7	18375.3
7	71.6	137.3	18412.6	118.4	56.2	34.9	15.7	18346.7
8	118.2	145.3	18033.4	115.9	55.1	34.1	15.8	18517.8
9	121.6	145.9	18031.1	115.7	54.9	33.8	15.8	18513.8
10	124.2	146.3	17750.7	114.6	54.5	33.7	15.9	18239.9
11-30	128.0	147.0	17747.0	114.1	54.2	33.6	15.9	18239.8

Table A-4a
 UO_2 ONCE-THROUGH, COMPOSITE FUEL MANAGEMENT SCHEME.
 CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Total Heavy Metal</u>
1	2037.7	91363.2	93400.9
2	762.3	22317.8	23080.1
3	614.0	17545.7	18159.7
4	676.2	17483.5	18159.7
5	711.8	17447.9	18159.7
6	747.4	17412.3	18159.7
7-30	783.0	17376.7	18159.7

Table A-4b

UO₂ ONCE-THROUGH, COMPOSITE FUEL MANAGEMENT SCHEME

DISCHARGE DATA BY YEAR

(Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1	136.4	39.5	23396.4	88.2	32.0	14.8	3.8	23711.1
2	91.1	45.8	18247.9	74.4	33.1	16.3	6.0	18514.6
3	63.5	57.0	17251.6	74.5	37.5	18.9	9.0	17512.0
4	75.3	62.7	17652.6	78.0	41.2	21.3	11.0	17942.1
5	72.3	76.5	16805.9	76.7	42.5	22.5	13.7	17110.1
6	84.1	94.9	17747.9	81.8	45.5	24.0	14.8	18093.0
7	74.3	93.9	16970.2	79.1	44.2	24.2	15.4	17301.3
8	77.8	109.5	16871.6	80.6	46.1	25.7	17.1	17228.4
9	74.5	117.4	16810.4	82.0	47.1	26.2	18.3	17175.9
10	78.2	128.1	16776.1	83.1	48.5	27.0	18.7	17159.7
11-30	81.9	134.2	16741.8	83.6	48.7	27.4	18.9	17136.5

Table A-5a
 PU/U SGR THREE-BATCH FUEL MANAGEMENT SCHEME
 MAKEUP CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Total Heavy Metal</u>
1	2195.5	97123.0	99318.5
2	1081.4	33034.6	34116.0
3	1036.0	33506.5	34542.5
4	828.0	26467.0	27295.0
5	886.0	26409.0	27295.0
6	800.0	25216.0	28016.0
7	791.0	24798.0	25589.0
8	811.0	24778.0	25589.0
9	763.0	23547.0	24310.0
10	746.0	23137.0	23883.0
11	755.0	23128.0	23883.0
12	754.0	23556.0	24310.0
13	742.0	23141.0	23883.0
14	726.0	21451.0	22177.0
15	721.0	21883.0	22604.0
16	709.0	21468.0	22177.0
17	714.0	21463.0	22177.0
18	718.0	21886.0	22604.0
19	708.0	21469.0	22177.0
20	714.0	21463.0	22177.0

Table A-5a (Continued)
 PU/U SGR THREE-BATCH FUEL MANAGEMENT SCHEME
 MAKEUP CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Total Heavy Metal</u>
21	692.0	20206.0	20898.0
22	707.0	21470.0	22177.0
23	710.0	21467.0	22177.0
24	692.0	20206.0	20898.0
25	682.0	19789.0	20471.0
26	684.0	19787.0	20471.0
27	690.0	20208.0	20898.0
28	680.0	19791.0	20471.0
29	686.0	19785.0	20471.0
30	692.0	20206.0	20898.0

Table A-5b
 PU/U SGR THREE-BATCH FUEL MANAGEMENT SCHEME
 RECYCLE CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1							
2							
3							
4	47.0	6583.0	132.0	40.0	18.0	4.0	6824.0
5	47.0	6545.0	140.0	56.0	27.0	9.0	6824.0
6	59.0	8196.0	160.0	66.0	35.0	14.0	8538.0
7	59.0	8174.0	176.0	68.0	38.0	14.0	8529.0
8	59.0	8174.0	176.0	69.0	38.0	14.0	8530.0
9	70.0	9786.0	202.0	100.0	54.0	23.0	10235.0
10	70.0	9761.0	212.0	107.0	58.0	27.0	10235.0
11	70.0	9742.0	215.0	114.0	62.0	33.0	10236.0
12	70.0	9732.0	220.0	117.0	64.0	33.0	10236.0
13	70.0	9731.0	221.0	117.0	64.0	33.0	10236.0
14	82.0	11376.0	233.0	135.0	74.0	42.0	11942.0
15	81.0	11363.0	237.0	140.0	76.0	45.0	11942.0
16	81.0	11349.0	241.0	144.0	78.0	49.0	11942.0
17	81.0	11337.0	246.0	147.0	80.0	50.0	11941.0
18	81.0	11347.0	244.0	147.0	80.0	49.0	11942.0
19	81.0	11320.0	246.0	153.0	83.0	58.0	11941.0
20	81.0	11306.0	250.0	157.0	86.0	61.0	11941.0

Table A-5b (Continued)
 PU/U SGR THREE-BATCH FUEL MANAGEMENT SCHEME
 RECYCLE CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
21	93.0	12994.0	249.0	160.0	87.0	65.0	13648.0
22	81.0	11303.0	247.0	158.0	86.0	66.0	11942.0
23	81.0	11301.0	248.0	158.0	86.0	66.0	11941.0
24	93.0	12973.0	254.0	166.0	90.0	72.0	13648.0
25	93.0	12962.0	257.0	169.0	92.0	74.0	13647.0
26	93.0	12971.0	251.0	165.0	90.0	77.0	13648.0
27	93.0	12961.0	256.0	169.0	92.0	77.0	13648.0
28	93.0	12960.0	257.0	169.0	92.0	77.0	13648.0
29	93.0	12956.0	254.0	169.0	92.0	83.0	13648.0
30	93.0	12952.0	254.0	171.0	93.0	84.0	13648.0

Table A-5c
 PU/U SGR THREE-BATCH FUEL MANAGEMENT SCHEME
 MAKEUP DISCHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1	264.0	55.4	32887.2	136.2	41.5	20.9	4.0	33409.2
2	217.3	81.2	30684.5	146.1	57.8	31.4	9.6	31227.9
3	219.5	110.6	31444.3	167.1	41.2	40.7	15.1	32038.5
4	291.8	127.0	32288.3	174.9	71.0	42.1	15.0	33649.1
5	272.8	127.2	32312.0	180.0	70.4	44.0	15.2	33021.6
6	226.0	99.0	25838.0	145.0	57.0	35.0	12.0	26412.0
7	260.0	104.0	25800.0	147.0	55.0	35.0	11.0	26412.0
8	222.0	95.0	24621.0	139.0	54.0	33.0	11.0	25175.0
9	220.0	94.0	24214.0	137.0	53.0	33.0	11.0	24762.0
10	232.0	96.0	24200.0	137.0	52.0	33.0	11.0	24761.0
11	216.0	90.0	22996.0	130.0	50.0	31.0	10.0	23523.0
12	210.0	88.0	22594.0	128.0	49.0	31.0	10.0	23110.0
13	216.0	89.0	22588.0	128.0	49.0	31.0	10.0	23111.0
14	211.0	90.0	23002.0	130.0	50.0	31.0	10.0	23524.0
15	208.0	88.0	22597.0	128.0	49.0	31.0	10.0	23111.0
16	215.0	85.0	20958.0	120.0	45.0	28.0	9.0	21460.0
17	208.0	85.0	21374.0	122.0	46.0	29.0	9.0	21873.0
18	205.0	84.0	20969.0	119.0	45.0	28.0	9.0	21459.0
19	203.0	84.0	20966.0	119.0	45.0	28.0	9.0	21459.0
20	206.0	85.0	21376.0	121.0	46.0	29.0	9.0	21872.0

Table A-5c (Continued)
 PU/U SGR THREE-BATCH FUEL MANAGEMENT SCHEME
 MAKEUP DISCHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
21	206.0	77.0	19748.0	109.0	41.0	26.0	8.0	20221.0
22	202.0	79.0	20976.0	115.0	44.0	28.0	9.0	21459.0
23	204.0	79.0	20974.0	115.0	44.0	28.0	9.0	21459.0
24	204.0	77.0	19750.0	109.0	41.0	26.0	8.0	20221.0
25	204.0	75.0	19343.0	107.0	40.0	25.0	8.0	19808.0
26	206.0	76.0	19340.0	107.0	40.0	25.0	8.0	19808.0
27	204.0	76.0	19751.0	109.0	41.0	26.0	8.0	20221.0
28	202.0	75.0	19345.0	107.0	40.0	25.0	8.0	19808.0
29	206.0	76.0	19340.0	107.0	40.0	25.0	8.0	19808.0
30	206.0	77.0	19748.0	109.0	41.0	26.0	8.0	20221.0
30	206.0	76.0	19340.0	107.0	40.0	25.0	8.0	19808.0
30	208.0	76.0	19338.0	107.0	40.0	25.0	8.0	19808.0

Table A-5d
 PU/U SGR THREE-BATCH FUEL MANAGEMENT SCHEME
 RECYCLE DISCHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1								
2								
3								
4								
5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
6	19.0	5.0	6427.0	64.0	47.0	27.0	12.0	6601.0
7	21.0	5.0	6395.0	72.0	56.0	32.0	17.0	6598.0
8	26.0	6.0	8006.0	84.0	64.0	30.0	23.0	8247.0
9	27.0	6.0	7990.0	92.0	69.0	41.0	23.0	8248.0
10	27.0	6.0	7989.0	92.0	69.0	41.0	23.0	8247.0
11	33.0	7.0	9564.0	112.0	90.0	54.0	33.0	9893.0
12	34.0	7.0	9543.0	118.0	96.0	57.0	36.0	9891.0
13	35.0	7.0	9526.0	121.0	100.0	60.0	41.0	9890.0
14	36.0	7.0	9518.0	125.0	103.0	61.0	41.0	9891.0
15	36.0	7.0	9517.0	125.0	103.0	61.0	41.0	9890.0
16	40.0	8.0	11121.0	134.0	114.0	68.0	51.0	11536.0
17	41.0	8.0	11109.0	138.0	117.0	70.0	54.0	11537.0
18	41.0	8.0	11097.0	140.0	120.0	72.0	57.0	11535.0
19	42.0	8.0	11086.0	144.0	123.0	74.0	57.0	11534.0
20	42.0	8.0	11088.0	143.0	123.0	73.0	57.0	11534.0

Table A-5d (Continued)
 PU/U SGR THREE-BATCH FUEL MANAGEMENT SCHEME
 RECYCLE DISCHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
21	44.0	9.0	12263.0	142.0	124.0	74.0	69.0	13179.0
22	41.0	8.0	10672.0	142.0	125.0	74.0	68.0	11529.0
23	42.0	8.0	10671.0	142.0	125.0	74.0	68.0	11529.0
24	45.0	9.0	12243.0	146.0	128.0	77.0	75.0	13177.0
25	45.0	9.0	12454.0	148.0	131.0	78.0	76.0	13176.0
26	45.0	9.0	12239.0	144.0	127.0	76.0	79.0	13175.0
27	45.0	9.0	12231.0	147.0	130.0	78.0	79.0	13175.0
28	46.0	9.0	12230.0	147.0	130.0	78.0	79.0	13175.0
29	46.0	9.0	12226.0	146.0	130.0	77.0	83.0	13174.0
30	46.0	9.0	12221.0	147.0	131.0	78.0	84.0	13173.0
30	46.0	9.0	12227.0	144.0	127.0	76.0	86.0	13173.0
30	46.0	9.0	12218.0	147.0	131.0	78.0	87.0	13173.0

Table A-6a

PU/U SGR FIVE-BATCH FUEL MANAGEMENT SCHEME
 MAKEUP CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Total Heavy Metal</u>
1	2184.69	97383.31	99568.0
2	951.90	19519.10	20471.0
3	923.24	19547.76	20471.0
4	791.54	16267.46	17059.0
5	779.59	16279.40	17059.0
6	774.48	16284.52	17059.0
7	765.95	16293.05	17059.0
8	754.28	15025.72	15780.0
9	724.66	14628.34	15353.0
10	712.37	14640.62	15353.0
11	718.52	14634.48	15353.0
12	713.91	14639.09	15780.0
13	732.19	15047.81	15353.0
14	704.70	14648.29	15353.0
15	707.77	14645.23	15353.0
16	709.31	14643.69	15353.0
17	710.84	14642.15	15353.0
18	717.99	15062.01	15780.0
19	704.70	14648.30	15353.0
20	701.63	14651.37	15353.0

Table A-6a (Continued)

PU/U SGR FIVE-BATCH FUEL MANAGEMENT SCHEME
MAKEUP CHARGE DATA BY YEAR
(Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Total Heavy Metal</u>
21	701.63	14651.37	15353.0
22	707.77	14645.23	15353.0
23	722.72	15057.27	15780.0
24	704.70	14648.29	15353.0
25	690.25	13810.75	14501.0
26	694.59	13806.40	14501.0
27	693.15	13807.85	14501.0
28	709.03	14217.97	14927.0
29	690.25	13810.75	14501.0
30	687.35	13813.65	14501.0

Table A-6b

PU/U SGR FIVE-BATCH FUEL MANAGEMENT SCHEME

MAKEUP DISCHARGE DATA BY YEAR

(Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1	111.46	27.90	19945.69	80.55	26.37	12.71	2.82	20207.5
2	93.63	42.84	19724.61	88.88	39.04	20.20	7.31	20016.1
3	73.07	53.79	18141.75	92.86	45.61	25.57	12.38	18445.03
4	55.92	61.94	17608.06	99.11	50.98	30.38	17.99	17924.38
5	55.62	61.91	19237.26	114.87	61.35	37.29	24.95	19593.25
6	193.65	121.92	18818.48	130.15	54.88	37.22	15.52	19371.82
7	168.99	119.32	18814.04	128.56	56.40	37.78	16.91	19342.0
8	142.3	99.2	15687.0	107.2	46.6	31.4	13.8	16127.5
9	136.7	97.8	15694.0	106.4	46.8	31.3	14.0	16127.0
10	134.0	97.1	15698.0	106.1	46.9	31.3	14.2	16127.6
11	130.2	96.1	15703.0	105.6	47.0	31.3	14.3	16127.5
12	143.5	94.6	14495.0	100.7	42.8	29.2	12.3	14918.1
13	134.7	90.9	14110.0	97.3	41.8	28.3	12.1	14515.1
14	136.1	91.2	14108.0	97.5	41.7	28.3	12.1	14514.9
15	131.2	90.1	14114.0	96.9	41.9	28.3	12.3	14514.7
16	129.2	89.6	14117.0	96.6	41.9	28.3	12.4	14515.0
17	131.9	91.9	14510.2	99.2	43.2	29.1	12.7	14918.2
18	124.6	88.4	14123.0	96.0	42.1	28.2	12.6	14514.9
19	129.7	91.3	14513.0	98.9	43.2	29.0	12.8	14917.9
20	126.6	88.9	14120.0	96.3	42.0	28.2	12.5	14514.5

Table A-6b (Continued)
 PU/U SGR FIVE-BATCH FUEL MANAGEMENT SCHEME
 MAKEUP DISCHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
21	127.5	89.1	14119.0	96.4	42.0	28.3	12.4	14514.7
22	125.3	90.2	14519.0	98.3	43.3	29.0	13.0	14918.1
23	124.3	88.3	14123.0	95.9	42.1	28.2	12.6	14514.4
24	123.2	88.1	14125.0	95.8	42.1	28.2	12.6	14515.0
25	123.2	88.1	14125.0	95.8	42.1	28.2	12.6	14515.0
26	126.2	88.8	14121.0	96.2	42.0	28.2	12.5	14514.9
27	127.5	90.7	14516.0	98.6	43.3	29.0	12.9	14918.0
28	124.5	88.4	14123.0	96.0	42.1	28.2	12.6	14514.8
29	130.2	86.6	13322.0	92.3	39.4	26.8	11.3	13708.6
30	132.3	87.1	133190.0	92.6	39.3	26.8	11.3	13708.4

Table A-6c
 PU/U SGR FIVE-BATCH FUEL MANAGEMENT SCHEME
 RECYCLE CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1							
2							
3							
4	23.4	3271.0	77.7	25.4	11.0	2.7	3411.2
5	23.2	3241.0	85.8	37.7	17.5	7.1	3412.3
6	23.1	3221.0	89.6	44.0	22.2	12.0	3411.9
7	22.9	3200.0	95.6	49.2	26.4	17.4	3411.5
8	34.8	4856.0	110.9	59.2	32.4	24.1	5117.4
9	34.8	4857.0	125.6	53.0	32.3	15.0	5117.7
10	34.8	4855.0	124.1	54.4	32.8	16.3	5117.4
11	34.4	4796.0	131.4	93.6	39.9	22.7	5118.0
12	34.4	4802.0	136.4	93.7	42.8	28.2	5137.5
13	34.3	4793.0	143.1	79.2	45.8	29.9	5125.3
14	34.3	4780.0	142.6	81.1	46.6	32.8	5117.4
15	34.2	4773.0	140.0	78.6	45.9	37.8	5109.5
16	34.2	4777.0	143.8	81.8	47.7	32.9	5117.4
17	34.2	4776.0	143.9	81.8	47.8	33.8	5117.5
18	33.9	4736.0	155.3	98.7	54.2	39.3	5117.4
19	34.0	4742.0	152.7	93.2	54.0	42.3	5118.2
20	33.9	4735.0	154.7	95.8	55.4	43.0	5117.8

Table A-6c (Continued)
 PU/U SGR FIVE-BATCH FUEL MANAGEMENT SCHEME
 RECYCLE CHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
21	33.9	4726.0	156.6	98.6	56.7	45.7	5117.5
22	33.8	4718.0	159.4	99.6	57.5	49.9	5118.2
23	33.8	4723.0	157.8	99.6	57.5	45.7	5117.4
24	33.8	4723.0	157.9	99.6	57.6	46.2	5118.1
25	39.7	5535.0	169.4	112.4	63.7	51.0	5971.2
26	39.7	5543.0	164.5	108.4	62.2	52.7	5970.5
27	39.7	5538.0	166.3	110.5	63.4	53.4	5971.3
28	39.6	5531.0	167.8	112.6	64.4	55.4	5970.8
29	39.6	5525.0	169.4	113.8	65.1	58.2	5971.1
30	39.6	5523.0	171.6	114.8	65.7	55.7	5970.4

Table A-6d
 PU/U SGR FIVE-BATCH FUEL MANAGEMENT SCHEME
 RECYCLE DISCHARGE DATA BY YEAR
 (Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
1								
2								
3								
4								
5								
6								
7								
8	5.5	2.9	3138.0	29.1	23.7	14.5	9.6	3223.3
9	6.8	2.9	3116.0	29.2	23.8	14.5	9.6	3202.8
10	7.5	2.9	3100.0	29.2	23.8	14.5	9.6	3187.5
11	8.2	2.8	3083.0	42.2	37.0	22.4	19.7	3215.3
12	9.8	4.4	4667.0	48.4	40.9	25.1	28.0	4823.6
13	10.2	4.3	4671.0	51.7	43.0	26.6	22.0	4828.8
14	10.2	4.3	4669.0	51.6	43.1	26.7	22.9	4827.8
15	11.7	4.4	4619.0	64.0	60.0	34.1	28.4	4821.6
16	12.2	4.1	4624.0	61.6	54.7	33.9	31.4	4821.9
17	12.5	4.1	4616.0	63.8	57.3	35.5	32.2	4821.4
18	13.0	4.1	4605.0	66.3	60.1	37.1	34.8	4820.4
19	13.3	4.0	4598.0	64.5	58.4	36.2	37.8	4812.2
20	13.1	4.0	4602.0	67.2	61.2	38.0	34.9	4820.4

Table A-6d (Continued)

PU/U SGR FIVE-BATCH FUEL MANAGEMENT SCHEME :
RECYCLE DISCHARGE DATA BY YEAR
(Kg)

<u>Year</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>	<u>Total Heavy Metal</u>
21	13.1	4.0	4601.0	67.2	61.2	38.0	35.5	4820.0
22	14.3	3.9	4564.0	77.2	73.2	44.3	39.9	4816.8
23	14.3	3.8	4568.0	74.6	70.2	43.4	42.0	4816.3
24	14.5	3.8	4562.0	76.5	72.4	44.8	42.7	4816.7
25	14.8	3.7	4553.0	78.1	74.6	46.0	44.8	4815.0
26	15.2	3.7	4545.0	79.4	75.9	46.8	47.8	4813.8
27	14.9	3.7	4550.0	79.2	75.7	46.7	44.8	4815.0
28	15.0	3.7	4550.0	79.2	75.8	46.8	45.2	4815.7
29	16.4	4.5	5331.0	84.7	81.3	49.9	50.7	5618.5
30	16.1	4.6	5339.0	81.3	77.6	48.0	51.8	5618.4

Appendix B

FUEL RESOURCE REQUIREMENTS AND MATERIAL INVENTORY DATA FOR 18-MONTH REFUELING CYCLES

Because of their potential for increased availability, 18-month cycles are seriously being considered by a large number of utilities. Such long refueling intervals are resource efficient only in the context of extended burnup. For a discharge exposure of 50,000 MWd/t, the uranium requirements of an 18-month cycle are about 5% lower than those of the reference 12-month, 30,000 MWd/t cycle. On the other hand, a 12-month cycle with extended burnup has significantly better uranium utilization than an 18-month cycle. The impact of cycle length, discharge exposure, and the number of in-core fuel batches on uranium utilization is discussed in detail in Section 4.2.1.

In this appendix, the fuel resource requirements and material inventory data of two 18-month fuel cycles are presented. These two cycles have fuel discharge exposures that are similar to the reference (three-batch) and the extended-burnup (five-batch) cycles discussed in Sections 5.2 and 5.3, respectively. In all cases, the System 80TM PWR was used as the reference design.

Table B-1

FUEL RESOURCE REQUIREMENTS FOR 18-MONTH REFUELINGS¹

	Fuel Management Scheme	
	30,000 MWd/t Burnup ² (Z=2)	50,000 MWd/t Burnup ² (Z=3.35)
U ₃ O ₈ Requirements (ST/GWe)		
Initial Core	479	479
Equilibrium Cycle	318	280
30-Year Cumulative	6483	5773
Separative Work Requirements (10 ³ SWU/GWe)		
Initial Core	269	268
Equilibrium Cycle	199	199
30-Year Cumulative	4020	4040

Notes: (1) 75% Capacity factor and 0.2 wt % tails assay.

- (2) Because of the burnable poison shim rods used each cycle, the total heavy metal in the core is less than for the annual refueling scheme. Consequently, the discharge burnup increases to 30,868 MWd/t and 52,244 MWd/t for Z=2 and Z=3.35 cases, respectively. Here, Z represents the average number of cycles that a fuel batch remains in core.

Table B-2a

FUEL CYCLE INFORMATION FOR 18-MONTH REFUELING

30,000 MWd/t BURNUP¹

1270 MW(e) PWR

Charge Data

Batch	Cycle		Heavy Metal Loaded (kg)	Enrich- ment (wt%)	Discharge Exposure (MWd/t)
	In	Out			
A	1	1	34,119	1.97	15,722
B	1	1	16,329	2.63	15,722
C	1	2	15,903	2.63	26,722
D	1	2	32,962	3.14	26,722
E	2	3	50,795	2.81	26,434
F	3	4	50,376	3.25	30,868
G	4	5	50,795	3.41	30,868
H	5	6	50,376	3.15	30,868
I ⁽²⁾	6	7	50,795	3.40	30,868
J ⁽²⁾	7	8	50,376	3.19	30,868

Table B-2b

DISCHARGE DATA
(kg)

<u>Batch</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-240</u>	<u>Pu-241</u>	<u>Pu-242</u>
A	270.2	67.5	32,989.1	146.8	47.2	24.8	5.6
B	207.2	38.3	15,713.5	68.3	19.2	9.7	1.7
C	113.8	49.5	15,155.9	75.8	31.9	17.9	6.2
D	342.9	116.3	31,290.4	167.5	61.0	36.3	10.3
E	432.9	164.6	48,341.7	256.1	97.2	57.8	17.9
F	467.4	193.6	47,600.7	268.8	104.0	64.2	21.0
G	521.4	202.2	47,940.4	274.2	102.7	64.1	19.9
H	437.4	189.1	47,634.5	266.7	105.3	64.5	21.8
I ⁽²⁾	518.2	201.8	47,944.0	274.0	102.8	64.1	20.0
J ⁽²⁾	449.4	190.9	47,621.1	267.5	104.7	64.4	21.5

Notes: (1) Burnable poison shims used with each reload, thereby reducing total heavy metal in the core. This makes discharge burnup 30,868 MWd/t for 75% capacity factor.

(2) Repeat batches I and J cyclic for 30-year data.

Table B-3a.

FUEL CYCLE INFORMATION FOR 18-MONTH REFUELING

50,000 MWd/t BURNUP¹

1270 MW(e) PWR

Charge Data

Batch	Cycle		Heavy Metal Loaded (kg)	Enrich- ment (wt%)	Discharge Exposure (MWd/t)
	In	Out			
A	1	1	30,707	1.97	15,722
B	1	2	28,885	2.55	26,722
C	1	3	29,420	3.02	42,416
D	1	4	10,300	3.14	58,036
E	2	4	19,510	4.38	42,314
F	2	5	10,378	4.38	57,922
G	3	5	19,510	5.24	46,922
H	3	6	10,378	5.24	62,530
I	4	6	19,510	4.69	46,836
J	4	7	10,378	4.69	62,444
K	5	7	19,510	4.45	46,824
L	5	8	10,378	4.45	62,432
M	6	8	19,510	4.94	46,824
N	6	9	10,378	4.94	62,432
O	7	9	19,510	4.94	46,824
P	7	10	10,378	4.94	62,432
Q(2)	8	10	19,510	4.79	46,824
R(2)	8	11	10,378	4.79	62,432

Table B-3b

DISCHARGE DATA
(kg)

Batch	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242
A	298.4	60.7	29,634.8	132.2	42.5	22.4	5.0
B	246.2	87.8	27,491.7	136.9	58.5	32.6	11.7
C	177.6	115.0	27,482.3	155.1	79.2	46.6	25.5
D	35.0	41.2	9455.7	54.9	32.5	18.8	14.7
E	227.7	102.4	18,090.1	114.8	47.9	30.9	11.8
F	65.1	57.8	9483.1	61.0	32.0	19.9	11.6
G	283.9	122.9	17,903.3	119.1	49.3	32.9	11.7
H	84.9	70.5	9419.3	65.1	32.9	21.6	10.5
I	224.8	111.7	17,976.1	117.1	50.8	33.1	13.5
J	60.8	63.2	9431.8	62.5	33.5	21.2	12.2
K	200.4	106.6	18,006.3	115.3	51.6	33.1	14.4
L	50.6	59.6	9440.1	61.2	33.7	20.9	12.9
M	251.5	116.8	17,944.2	118.5	50.0	33.0	12.6
N	71.1	66.6	9424.9	63.8	33.2	21.4	11.4
O	251.5	116.8	17,944.2	118.5	50.0	33.0	12.6
P	71.1	66.6	9424.9	63.8	33.2	21.4	11.4
Q ⁽²⁾	235.4	113.8	17,963.5	117.7	50.5	33.1	13.1
R ⁽²⁾	64.8	64.6	9428.9	63.0	33.4	21.2	11.8

Notes: (1) Burnable poison shims used with each reload, thereby reducing total heavy metal in the core. This makes discharge burnup 52,244 MWd/t for 75% capacity factor.

(2) Repeat batches Q and R cyclic for 30-year data.