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Advanced Reactor Design Study: Assessing Nonbackfittable Concepts for Improving Uranium Utilization in Light Water Reactors

Final Report

September 1981

Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
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ADVANCED REACTOR DESIGN STUDY:

ASSESSING NONBACKFITTABLE CONCEPTS FOR IMPROVING URANIUM UTILIZATION IN LIGHT WATER REACTORS

FINAL REPORT

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1.0 INTRODUCTION

The U.S. Department of Energy (DOE) has undertaken a limited program to develop technology for improving light water reactor (LWR) performance. Aspects of performance being improved are plant productivity, dose reduction to plant personnel, and uranium utilization. The Advanced Reactor Design Study (ARDS) is part of the overall effort to improve uranium utilization in LWRs. This study involves a first effort to identify and evaluate nonbackfittable concepts for improving uranium utilization.

Once-through fuel utilization improvements are categorized into two groups: 1) those suitable for deployment in existing plants (backfittable), and 2) those that are nonbackfittable. In general, the backfittable concepts have been characterized as those that offer potential improvements in fuel utilization and economics, early implementation, and the likelihood that the concept will be suitable for licensing and deployment into currently operating reactors or commercial offerings without undue difficulty, cost, or impact on plant operational procedures.(1) On the other hand, nonbackfittable concepts are those that are too costly to incorporate in existing reactors, and thus, could only be economically incorporated in new reactor designs or reactor plants in very early stages of construction.

In some quarters the definition of a backfittable concept has been argued to include all potential improvements that are technically feasible in existing plants. However, in the extreme, this definition would include impractical concepts such as replacing all of an existing plant's LWR components with Liquid Metal Fast Breeder Reactor (LMFBR) components. Such extreme concepts would obviously require extensive downtime and power outage, and the potential benefits would not offset a utility's implementation costs and lost revenue. Therefore, there is a need to introduce the idea of economic "practicality" into the definition of backfittable.

At the other extreme, one could argue that no potential improvements be considered feasible in existing plants. In this case, all the concepts would be categorized as nonbackfittable. The rationale for this limitation of concept feasibility might, for example, be a utility's desire to avoid any changes in an operating plant for fear of increased regulatory exposure. This rationale would exclude the early implementation of such concepts as high burnup fuel. It is not valid, since selected utilities are proceeding to extended burnups.

Clearly, the practicality of implementing improvements in currently operating reactors must be considered in determining whether a concept is backfittable or nonbackfittable. Concepts that are too costly to incorporate in existing reactors are considered to be nonbackfittable. The implementation of nonbackfittable concepts is, thus, constrained to plants in early stages of construction or to future plants. However, nonbackfittable concepts can be demonstrated in an existing plant if sufficient government support is provided to offset retrofit costs and lost revenues.

Lead responsibility for managing the Advanced Reactor Design Study was assigned to the Pacific Northwest Laboratory (PNL) by DOE. As part of this reponsibility, PNL developed the program plan, schedules, and cost projections to meet the study's objectives. Since the expertise for performing the study's evaluations and assessments resided principally with LWR vendors/designers, their participation was actively solicited. The LWR vendors were also the most qualified to assess the impacts of nonbackfittable concepts on reactor design, operation, safety, licensability, and acceptability.

The Advanced Reactor Design Study is the first step toward implementing nonbackfittable concepts for improving uranium utilization in LWRs. The purposes of this study are to 1) identify nonbackfittable concepts having potential for implementation and 2) recommend research, development, and demonstration programs leading to implementation of those concepts. This report describes the work that was done to accomplish these purposes.

2.0 SUMMARY

The objective of the Advanced Reactor Design Study (ARDS) is to identify and evaluate nonbackfittable concepts for improving uranium utilization in light water reactors (LWRs). The results of this study provide a basis for selecting and demonstrating specific nonbackfittable concepts that have good potential for implementation. Lead responsibility for managing the study was assigned to the Pacific Northwest Laboratory (PNL).

As part of this responsibility, PNL consolidated the results obtained from the industrial evaluation and assessment studies, identified consensus positions among program contributors, established the technical bases for program planning, and developed recommendations for demonstrating specific design concepts.

Nonbackfittable concepts for improving uranium utilization in LWRs on the once-through fuel cycle were selected separately for PWRs and BWRs due to basic differences in the way specific concepts apply to those plants. Nonbackfittable concepts are those that are too costly to incorporate in existing plants, and thus, could only be economically incorporated in new reactor designs or plants in very early stages of construction. Essential results of the Advanced Reactor Design Study are summarized in the following sections.

2.1 RESULTS

For PWRs, nonbackfittable concepts that appear to have the greatest potential for implementation in the near/mid-term time frame are:

- rapid/frequent refueling
- low power density/radial blanket/Zircaloy shroud
- extended coastdown.

These concepts are listed in decreasing order of potential. The rapid/frequent refueling concept would be used to implement a 6-month cycle length. Low power density, radial blanket, and Zircaloy shroud were combined into a single concept because the three features should be considered together in optimizing the design. Coastdown would be extended beyond its normal length by reducing reactor moderator temperature and accommodating a larger volume of secondary system steam at reduced pressure. Extended coastdown would require major equipment modifications for accommodation of larger volumes of lower pressure steam at the turbine inlet.

Two nonbackfittable PWR concepts assessed by industry appear to have marginal benefits:

- smaller fuel assemblies
- higher system pressure and temperature.

Smaller fuel assemblies have a small potential for improving uranium utilization, but additional fuel handling and other potential drawbacks make implementation questionable. There appear to be substantial technical barriers to overcome before the PWR pressure and temperature could be increased significantly.

For BWRs, of the nonbackfittable concepts considered, those that appear to have potential for implementation in the near/mid-term time frame are:

- spectral shift by extended flow control
- rapid/frequent refueling
- higher system pressure and temperature
- low power density.

The potential benefit of increasing the system pressure by about 250 psi is larger than any of the other concepts, but is offset by a potentially adverse impact on reliability. The most promising BWR concept involves extending coolant flow control to a range of 40 to 150% of rated flow to vary the core steam volume fraction and thus, shift the thermal neutron spectrum during the cycle. The rapid/frequent refueling concept is similar to that for PWRs and would be used to implement a 6-month cycle length. The benefit of low power density in BWRs, while promising, is less substantial than for PWRs, since BWR power density is already lower than that of PWRs.

Two BWR concepts considered by industry have already been applied in existing reactors, and extensions beyond the limit of backfittability were assessed to have marginal benefits:

- extended coastdown
- radial blanket.

One other BWR concept was assessed by industry:

- soluble boron for cold shutdown.

This concept is promising, but it has also been applied in existing reactors and was determined to be backfittable.

2.2 CONCLUSIONS

When nonbackfittable concepts having the greatest potential for implementation are combined, future PWR uranium utilization can be improved by about 15% and future BWR uranium utilization can be improved by about 20%. These improvements are superimposed on potential uranium utilization improvements from backfittable concepts of 21% for PWRs and 28% for BWRs.[1] Thus, the total combined uranium utilization improvements from both backfittable and nonbackfittable concepts that have the greatest potential for implementation are about 33% for PWRs and 43% for BWRs. Since the reference BWR has about 7% lower uranium utilization than the current PWR,[1] uranium utilization for future LWRs incorporating the most promising design improvements is comparable.

Four basic nonbackfittable improvements appear to be most readily achievable in future LWRs:

- rapid/frequent refueling
- low power density/radial blanket/Zircaloy shroud
- spectral shift/end-of-cycle coastdown
- increased system pressure for BWR.

Each of these concepts has been assessed by industry and they all appear to be technically feasible on future LWRs. The earliest date for commercial operation of new plants employing these nonbackfittable improvements is around 1995. If these concepts are to be implemented on this schedule, development/demonstration programs must be established and government support must be provided. It is estimated that these development/demonstration programs would cost $10 to $15 million per year for 15 years to implement the most promising nonbackfittable concepts in both PWRs and BWRs.

Nineteen of the 28 nonbackfittable concepts reviewed by PNL were not selected for industrial assessment. The preliminary screening, made both by

PNL and industry, judged these concepts to have poor implementation potential. The PNL assessment and rationale for discarding each of these 19 concepts are contained in Appendix A.

2.3 RECOMMENDATIONS

Specific recommendations are made for each of the four basic nonbackfittable LWR improvements. These recommendations are based on industrial assessment of the research and development required for implementation and specific targets-of-opportunity identified to demonstrate crucial features of selected concepts.

Rapid/Frequent Refueling

- Initiate a program to determine actual rapid refueling outage times by demonstration of improved PWR refueling equipment being installed by Houston Lighting and Power Company at the South Texas Project. A practice demonstration of the refueling operation should be included before plant operation starts.
- Solicit proposals for innovative approaches to shorten the refueling outage time to approximately 7 days, when maintenance is deferred.
- Initiate a program to use the full-scale vessel refueling bridge and floor mockup at General Electric Company to determine actual operation times for comparison with theoretical minimums.
- Initiate a program to reduce the time used for low power and power ascension physics tests.
- Defer evolutionary development of refueling machines until more assurance is obtained that a sufficiently short refueling outage can be achieved.

Extended Coastdown/Spectral Shift

- Perform a study to determine the limits of present steam turbines, modifications required to relax limits, and costs of modifying existing turbines or incorporating capabilities to accept more low-pressure steam in future PWR turbines.

- Perform analyses of PWRs for conditions encountered during extended coastdown and end-of-cycle stretchout to establish the range of coastdown extension that can be accommodated.

- As Phase I of a demonstration program, perform a design study for equipment modifications to an existing PWR plant needed to extend its coastdown capability.

- Implement Phase II of the demonstration program during planned outages by making the equipment modifications identified in Phase I, and obtain operating data to verify plant performance characteristics.

- Demonstrate spectral shift in a BWR through extended flow control by making the necessary changes in an existing plant with government support and obtain operating data to verify plant performance characteristics.

Low Power Density/Radial Blanket/Zircaloy Shroud

- Perform a parametric study to determine the optimum configuration for using these concepts in PWRs using larger vessels, based on cost/benefit analysis decisions.

- Develop a preliminary conceptual PWR design using the results of the configuration optimization study.

- Identify targets-of-opportunity for demonstrating major features of the conceptual design that combines low power density, radial blanket, and Zircaloy shroud concepts in reactors under construction.

Increased System Pressure and Temperature

- Initiate a program to establish Zircaloy corrosion behavior at higher temperatures in water.

- For BWRs, use existing experience from Big Rock Point BWR to obtain information needed to increase pressure and temperature of future large BWRs.

3.0 TECHNICAL APPROACH

A work plan for executing the Advanced Reactor Design Study (ARDS) was prepared by PNL. This plan identified 1) specific tasks to be performed, 2) milestones and schedules, and 3) funding requirements for the study. Because the application of nonbackfittable concepts necessitates modifications to contemporary reactor designs, it was apparent that the most qualified organizations to assess implementation potential would be LWR designers/vendors. Accordingly, U.S. LWR designers/vendors were the principal participants in selecting, assessing, and evaluating the nonbackfittable concepts included in this study. Meetings were held with each of the four U.S. LWR vendors to describe the objectives of the ARDS and the approach planned to attain these objectives, and to obtain their suggestions as how to best achieve the objectives. Requests for proposals (RFPs) were sent to these vendors, seeking their participation in the study. The keystone of the study was the LWR vendors' evaluations and assessments, which provided the bases for the conclusions and recommendations included in this report.

Activities involved in the ARDS were organized into the following four tasks:

- Task I: Preliminary Review of Concepts
- Task II: Contracting Activities
- Task III: Industrial Evaluations and Assessments
- Task IV: Composite Assessments and Evaluations.

3.1 TASK I: PRELIMINARY REVIEW OF CONCEPTS

To provide information for RFP preparation, proposal evaluations, and assessment studies, PNL reviewed nonbackfittable concepts for improving uranium utilization in LWRs. The review included 28 concepts, identified by NASAP[1] and other fuel cycle studies,[2,3,4,5,6] which fell into three categories: 1) frequent refueling, 2) increased system efficiency, and 3) nonbackfittable nuclear fuel and core designs. A preliminary screening assessment of the concepts in each category provided the bases for grouping the concepts according to their implementation potential as shown in Table 3.1. Group I concepts were considered to have highest implementation potential.

TABLE 3.1. Nonbackfittable Design Modifications

Category	Frequent Refueling	Increased System Efficiency	Nonbackfittable Nuclear Fuel and Core Designs
I	PWR RAPID REFUELING SYSTEM	HIGHER TEMPERATURES AND PRESSURES	COASTDOWN[a]
	BWR RAPID REFUELING SYSTEM		BWR FLOW CONTROL[a]
	HOT STANDBY REFUELING		VARIABLE LATTICES
			LATTICE CHANGES[a]
			LOWER POWER DENSITY REACTORS
			FISSILE MATERIAL CONTROL
			VENTED FUEL
			BLANKETS[a]
			REFLECTORS
			SOLUBLE BORON FOR BWR COLD SHUTDOWN
II		INTEGRAL NUCLEAR SUPERHEAT	SEED BLANKETS
		ADD-ON NUCLEAR SUPERHEAT	FERTILE MATERIAL CONTROL
			SPECTRAL SHIFT (WITHOUT D_2O)
III	UNIT CORE REFUELING	ADD-ON FOSSIL FUELED SUPERHEAT	PWR FLOW CONTROL
	ON-LINE REFUELING		TUBULAR FUEL
		SUPERCRITICAL PRESSURES	HIGH PEAKING FACTOR REACTOR
		BOTTOMING CYCLES	ADVANCED CONTROL ROD SYSTEMS

(a) Beyond the limits of backfittability.

3.2 TASK II: CONTRACTING ACTIVITIES

Requests for proposals were prepared, proposals evaluated, and contracts were negotiated and administered as part of this task. RFPs were sent to the four U.S. LWR vendors seeking proposals for engineering studies for any of the concepts listed in Table 3.2 or any other concepts they wished to consider. Five proposals were received. A panel review board, using pre-established evaluation criteria, judged the proposal from Babcock and Wilcox on a radial blanket and Zircaloy shroud concept and the proposal from Combustion Engineering on a frequent refueling concept to be the only ones acceptable. Contracts were executed to perform engineering studies on these concepts.

Because these two engineering studies did not include all of the most promising concepts, an effort was undertaken to assess all concepts with good implementation potential. Contracts to participate in this assessment effort were executed with Babcock and Wilcox and with Combustion Engineering. General Electric participation was on a limited, voluntary basis and Westinghouse declined to participate.

3.3 TASK III: INDUSTRIAL EVALUATIONS AND ASSESSMENT

Two general areas of industrial (LWR vendor) participation in the ARDS were engineering studies of specific concepts and overall assessments of selected concepts.

Engineering Study of a Frequent Refueling Concept

Combustion Engineering performed an engineering study of a frequent refueling concept that used a dual refueling machine system. The study included the refueling system design, a review and evaluation of critical path operations for refueling/maintenance outage and refueling-only outage, identification of improvements in equipment, techniques, and procedures to minimize the duration of critical path operations, and the economic conditions (e.g., replacement power costs, capital costs) under which frequent refueling could be economically feasible.

Engineering Study of a Radial Blanket and Zircaloy Shroud for a PWR Core

Babcock and Wilcox performed an engineering study to evaluate the use of a radial blanket and a Zircaloy shroud for improving uranium utilization in a Babcock and Wilcox advanced reactor design. Both natural uranium and thorium were considered as blanket materials. Potential uranium utilization improvements from the improved neutron economy of a Zircaloy shroud were also evaluated. To evaluate a close to optimum blanket thickness, a modular fuel assembly design was used in this study. The modular assembly design allows for longitudinally segmenting the fuel into quarters so that two adjacent segments can be used as blankets and the other two adjacent segments as fuel. Individual segments could be used as blankets to provide the desired core peripheral configuration. The modular assembly design was also considered for use as core fuel assemblies to produce more flexibility in fuel management schemes. Such flexibility was applied to improving uranium utilization.

Assessment of Selected Concepts

Involvement of LWR vendors in the ARDS was also solicited for an assessment of the most promising concepts. Combustion Engineering and Babcock and Wilcox were full participants in this effort with General Electric participating on a more limited basis. The assessment effort consisted of 1) selecting the most promising of the 28 concepts identified in Task I, 2) developing criteria for assessment, 3) assessing concepts according to the criteria developed, 4) formulating conclusions regarding the relative merits of the concepts, and 5) making recommendations for further development of those concepts with greatest promise.

The assessment effort was initiated with a workshop to select the concepts to be assessed and to formulate assessment criteria. The selected concepts are shown in Table 3.2, and the criteria, in Table 3.3.

A period of six weeks was available for the participants to evaluate the concepts against the criteria and to develop conclusions and recommendations for each concept. Results of the participants' assessments were reviewed at a second workshop, conclusions and recommendations were developed, and the concepts were ranked by implementation potential. The industrial participants were allowed to revise their assessments following the workshop, and the revised results were compiled and summarized by PNL.

TABLE 3.2. Concepts Selected for Assessment by Industrial Participants

- Rapid/Frequent Refueling at Cold Shutdown
- Radial Blankets
- Low Power Density Core
- Spectral Shift/End-of-Cycle Coastdown
- Higher Temperature and Pressure
- Core Peripheral Modifications/Reflectors
- Small PWR Fuel Assemblies
- Soluble Boron for BWR Cold Shutdown

TABLE 3.3. Assessment Criteria Selected by Industrial Participants

Uranium Utilization

- U_3O_8 Savings
- SWU Savings

Operation

- Reliability
- Availability
- Operability

Economics

- Δ Capital Cost
- Δ Fuel Cycle Cost
- Δ Power Generation Cost
- Development Cost
- Δ Construction Time

Other Considerations

- Utility Acceptance
- Date of First Commercial Operation
- Compatibility with Other Concepts
- Potential for Retrofit
- Nonproliferation

Technology

- Technical Feasibility
- Safety

3.4 TASK IV: ANALYSIS OF COMPOSITE REACTOR SYSTEMS

The major efforts comprising the ARDS were 1) an engineering study on a frequent refueling concept by Combustion Engineering, 2) an engineering study of a radial blanket and Zircaloy shroud by Babcock and Wilcox, and 3) the assessment of selected concepts by Combustion Engineering, Babcock and Wilcox, and General Electric. In addition, PNL analyzed composite reactor systems that incorporate several potentially desirable nonbackfittable concepts to determine their combined benefits. These advanced LWR systems include both the backfittable concepts currently under development[1] and the most promising nonbackfittable concepts identified by the industrial assessments under this program. The competitive effects that multiple nonbackfittable concepts exhibit, and the extent to which the combined uranium utilization improvements recede, were analyzed for composite PWR and BWR systems. Based on the results of these analyses, and PNL's review of the engineering studies and assessments that were made under subcontracts to industry, overall conclusions and recommendations were developed for the ARDS. These results, conclusions, and recommendations are summarized in Chapter 2.0.

4.0 ENGINEERING DESCRIPTIONS AND FEASIBILITY EVALUATION OF SPECIFIC CONCEPTS

Two U.S. reactor vendors were awarded subcontracts by PNL to develop a functional description and perform a preliminary design and engineering study of specific nonbackfittable LWR concepts that can improve uranium utilization. One subcontract was awarded to Babcock & Wilcox Company (B&W) for "A Preliminary Feasibility Study of a Radial Blanket and Zircaloy Core Baffles and Formers in an Advanced PWR." The other subcontract was awarded to Combustion Engineering, Inc. (C-E) for "A Preliminary Engineering Study of a Frequent Refueling Design Concept for a PWR." Each of these subcontracts was scheduled over a 7-month time period. The industrial contractors had complete responsibility for the technical efforts expended. The bulk of the work involved the effort required to develop the functional description and preliminary design and engineering study. The remainder of the effort was to identify the major barriers that must be overcome before the concept could be implemented. A strategy for resolution of these problems was developed for each concept by the industrial contractors. Separate reports were issued by both B&W[7] and C-E[8] to document their engineering studies. Extracts from these two reports are presented in the following sections to summarize the efforts of these industrial contractors.

4.1 RADIAL BLANKET AND ZIRCALOY CORE SHROUD STUDY

This concept involves modifications to the core periphery region of an advanced PWR design. An advanced version of the Babcock-241 plant design served as the reference case for this study. The reference advanced design core contains several features that result in better uranium utilization than that achieved in PWRs currently operating or under construction.

The advanced Babcock-241 design used in the reference case features a low-power density core (Table 4.1), and a slightly higher average coolant temperature, which leads to an improvement of about 0.6% in thermal efficiency of the plant. This results in an increase in uranium utilization of approximately 2%.

TABLE 4.1. Advanced Babcock-241 Reactor Design Data

Reactor	
Design heat output (MWt)	3800
Design overpower (%)	112
Vessel coolant inlet temperature (F)	573.5
Vessel coolant outlet temperature (F)	629.7
Core coolant outlet temperature (F)	632.5
Operating pressure (nominal) (psig)	2250
Total reactor coolant flow (10^6 lb/h)	159
Core	
Total no. of fuel assemblies in core	241
Fuel assembly pitch spacing (in.)	8.587
Total core heat transfer surface area (ft^2)	75,229
Average heat flux (Btu/h-ft^2)	167,700
Maximum design heat flux (Btu/h-ft^2)	445,000
Average thermal output (kW/ft)	4.88
Maximum design thermal output, consistent with reference design peaking distribution (kW/ft)	12.95
Average core fuel temperature (F)	1230
Core average coolant velocity (fps)	14.6
Minimum DNBR at rated power	1.93 (BAW-2)
Minimum DNBR at design overpower	1.49 (BAW-2)
Core dimensions (in.)	
Equivalent diameter	150.4
Active height	131
Axial blanket height (in.)	
Top	6
Bottom	6

The advanced core design utilizes B&W's Mark-CZ fuel assembly (Table 4.2) that has been under development at B&W. This advanced 17x17 fuel assembly design contains two features that result in improved uranium utilization compared to assemblies of conventional design. One of these features uses Zircaloy-4 instead of inconel as the grid spacer material. The resultant inproved neutron economy improves uranium utilization by approximately 2%. The other feature in the Mark-CZ fuel design that provides improved uranium utilization is an improved water-to-uranium ratio. On an equal energy extraction basis, the Mark-CZ water-to-uranium ratio provides approximately 4.5% improvement in uranium utilization as compared to the Mark-C water-to-uranium ratio.

Other uranium saving features in the reference advanced core design include:

- increasing goal burnup from 30,000 or 34,000 MWd/MTU to 50,000 MWd/MTU to obtain about 10 to 15% improvement in uranium utilization
- providing axial blankets to obtain approximately 2 to 4% improvement in uranium utilization
- removing the axial power shaping rods (APSR's) near the end of each reactor cycle to extend cycle length and improve uranium utilization by about 1 to 2%.

The uranium savings resulting from the various reference case features are not strictly additive. Nevertheless, their combined uranium savings are expected to be significant (in the 20 to 25% range). All of these design changes except lower power density are considered to be backfittable.

The specific additional design features under study in this program (radial blanket and Zircaloy core baffles/formers) offer the potential to further improve uranium utilization over and above that achievable in the advanced reference core design. Using a zirconium base alloy instead of SS304 as the material for the core baffles and formers is estimated to result in U_3O_8 savings of 1 to 2% and use of a radial blanket is estimated to result in uranium savings of approximately 3 to 4%. Furthermore, extension of the modular fuel assembly concept to further optimize fuel management strategies offers the potential for additional improvement in fuel utilization, estimated to be

TABLE 4.2. Mark-CZ Fuel Assembly Design Parameters

Fuel Assembly	
Rod Array	17x17
Total Length (in.)	165-23/32
Envelope (in.)	8.536
No. of Fuel Rods	264
No. of Guide Tubes	24
No. of Instrument Tubes	1
No. of Spacer Grids	8
Fuel Rod Pitch (in.)	0.502
Fuel Rod	
Length	152-11/16
Clad O.D. (in.)	0.364
Clad I.D. (in.)	0.314
Clad Thickness	0.025
Pellet O.D. (in.)	0.307
Pellet clad gap (in.)	0.0070
Fuel density (% TD)	95
Fuel stack length (in.)	131
Bottom blanket stack length (in.)	6
Top blanket stack length (in.)	6
Pellet length (in.)	0.375
Pellet L/D (in.)	1.17
Guide Tube Assembly	
O.D. (in.)	0.465
I.D. (in.)	0.430
Spacer Sleeve	
O.D. (in.)	0.465
I.D. (in.)	0.428
Instrument Tube	
O.D. (in.)	0.420
I.D. (in.)	0.390
Spacer Grid	
Strip thickness, inner-outer (in.)	0.018/0.021
Height (inner strip) (in.)	2.0
Materials	
Fuel Clad	Zr-4
Guide Tube	Zr-4 (fully annealed)
Spacer Grids	6 intermediate Zr-4 2 end grids, inconel-718
End Fittings	Type CF3M SS
Instrument Tube and Spacer Sleeves	Zr-4 (fully annealed)

at least 2%. These savings are in addition to the benefits of the reference advanced design, which offers a 2% utilization improvement via increased plant efficiency because of the higher average coolant exit temperature. The design concept under study thus has a potential of a total 8 to 10% fuel utilization improvement beyond that achievable in current plants that incorporate the major backfittable design changes. However, when these improvements are combined, the composite uranium utilization improvement is reduced somewhat from the sum of the individual values because they compete to improve neutron economy. The composite uranium utilization improvement for this concept is analyzed in Appendix B.

Given the above potential for uranium utilization improvement, the objective of the program was to perform a preliminary engineering study to determine how to incorporate these design features into the reference design. The engineering study would lead to a preliminary determination of the feasibility, practicality, and operability of the advanced design modifications. A further aim of the study was to identify and assess remaining questions or barriers that could hinder the implementation of the proposed concept, and to identify further research, development, and demonstration efforts that would be required to answer those questions, eliminate those barriers, and facilitate implementation.

The program was divided into three main design efforts. The first effort focused on designing a modular fuel assembly. Within the context of this proposal, the modular assembly concept was perceived primarily as a means to accommodate a radial blanket of desired cross-sectional size. However, the engineering study of this area went beyond this application to investigate modular fuel assembly concepts that could also be used throughout the core. The second main design effort addressed the design of the radial blanket assembly, and the third main design effort addressed the design of Zircaloy core baffles and formers.

This summary is structured to reflect the three major design efforts mentioned above. The first section presents an overall summary of the results and conclusions. Then, the next three sections describe the modular fuel assembly design effort, the radial blanket assembly design, and the Zircaloy baffles/formers study. Finally, recommended future work aimed at resolving remaining questions is presented.

Summary and Conclusions

A preliminary engineering study was performed to determine the technical feasibility of incorporating certain uranium-conserving modifications to the core periphery region of an advanced version of the Babcock-241 pressurized water reactor design. The specific uranium-conserving features added to the low power density reference reactor design were a radial blanket and Zircaloy core baffles and formers.

A preliminary design was developed for a modularized fuel assembly concept in which the standard 17x17 fuel assembly was replaced by a group of four separable 8x8 pin modules. This modular concept can be employed on the core periphery, in which case an assembly would contain a combination of blanket modules and fuel pin containing modules. Additionally, the modular fuel assembly concept can be used throughout the core to enhance the potential for improving uranium utilization even further via added flexibility in fuel management. Fuel-containing modules retain the same water-to-fuel ratio as the reference 17x17 fuel assembly, whereas the blanket modules have a reduced water-to-fuel ratio to enhance neutron captures in the fertile blanket material. A preliminary design was also developed for the Zircaloy core baffles and formers.

The overall conclusion of the preliminary engineering study is that it appears technically feasible to incorporate the radial blanket, Zircaloy baffles/formers, and modular fuel assembly concepts into the B&W-241 fuel assembly advanced PWR design. Furthermore, it is concluded that these concepts are generically feasible as well. They could be incorporated even more effectively into advanced PWR designs that are not bound by the constraints of existing designs, such as the reference B&W-241 design.

No fundamental technological barriers have been identified that would rule out the implementation of these uranium-conserving features in future PWRs. Although additional development programs are needed to answer several key questions, it is judged that the resolution of all remaining technical questions is well within the bounds of current technology. However, before any detailed design work is undertaken on any of these concepts, it is recommended that a comprehensive and systematic study be performed of PWR designs combining low power density, radial blankets, and reflector region modifications. The objective of such a study would be to determine the optimum allocation of pressure vessel volume from the viewpoint of maximizing both uranium utilization and economics. Such a study should cover a range of pressure vessel sizes and power densities as well as a variety of blanket and reflector region designs.

Concurrently with such a comprehensive study it is recommended that several additional studies be performed to firm up the preliminary conclusions regarding the technical feasibility of these concepts. Further studies should address certain key areas that are judged to require more rigorous, in-depth study to confirm the preliminary conclusions reached in this study. The following studies are recommended by B&W:

- develop specifications for fabrication of optimized irradiation-resistant Zircaloy plate for core baffle/former components
- evaluate candidate blanket structural materials to maximize blanket life
- evaluate the dynamic response of modular fuel assemblies under seismic and LOCA conditions
- investigate minimum-shuffle fuel management schemes using small fuel assemblies.

Design of Modular Fuel Assembly

The modular fuel assembly concept was proposed to accommodate a radial blanket of half the thickness of a standard PWR fuel assembly on the periphery of the Babcock-241 core. The proposed modified core periphery (Figure 4.1) has blanket assemblies one quarter the cross-sectional size of standard Mark-CZ assemblies surrounding the core, and displacing parts of some of the peripheral fuel assemblies. The use of such small cross-sectional blanket assemblies allows an assembly arrangement that results in an average blanket thickness of approximately four inches. Earlier studies indicated that a

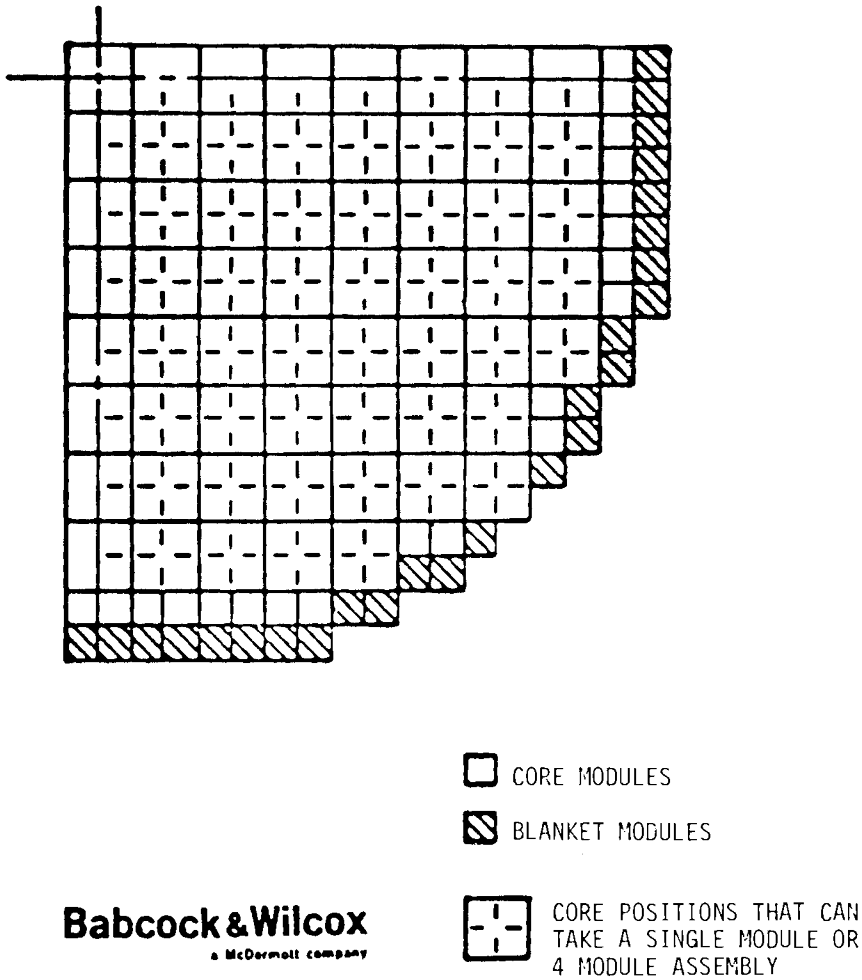


FIGURE 4.1. B&W Advanced Design Core and Radial Blanket Modular Assembly Arrangement

blanket of approximately this thickness provides more effective utilization of the fertile material than does a blanket with a thickness equivalent to the size of a standard PWR fuel assembly. A thicker radial blanket would sacrifice additional core volume without a corresponding gain in overall fuel utilization. A thicker blanket would also result in higher specific power in the core, due to a tradeoff for core volume, thus mitigating the fuel utilization improvement resulting from the inherently lower specific power of the reference design.

Although the modular concept was conceived primarily as a means of accommodating the small size blanket assemblies, it was also recognized that additional uranium utilization benefits could result if the small assembly concept could be used throughout the core as well. Thus, the modular fuel assembly design effort was directed toward developing a design that could serve both of these purposes.

The modular fuel assembly design was mainly a mechanical design effort. In addition, nuclear analyses were carried out in support of the modular assembly feasibility evaluation. These analyses were performed to assess the effectiveness of controlling relative power density between a group of adjacent modules by using burnable poison.

Design of Radial Blanket Module

The relationship of the core and blanket regions was illustrated in Figure 4.1. The reference 241 fuel assembly core volume was reduced by the equivalent of 18 fuel assemblies, these being replaced by blanket modules. In addition, other blanket modules were added to certain locations adjacent to the core periphery that could accommodate the quarter-size modules. This allowed 24 additional blanket modules to be added (a 33% increase in blanket volume) without changing the core support barrel design. The modified reactor design thus included the equivalent of 223 fuel assemblies and 24 blanket assemblies (each blanket "assembly" being equivalent to four blanket modules). With this core-blanket layout as the basis, the objective of this task was to formulate a preliminary design of a radial blanket module that would be compatible with the modular fuel assembly developed in the preceding task, and that would satisfy certain other design objectives.

The radial blanket was designed with the following functional requirements in mind:

1. reduce unproductive neutron leakage from the core by capturing neutrons; increase reflection of neutrons into the core; increase in situ power production from plutonium and fast fissions in uranium.
2. have coolant flow characteristics that provide for adequate heat removal from the blanket at the highest power density with minimum reduction in core flow availability.
3. efficient utilization of reactor volume.

4. efficient blanket feed utilization and reasonable fabrication.

The engineering study related to the radial blanket included a mechanical design effort plus supporting nuclear analyses.

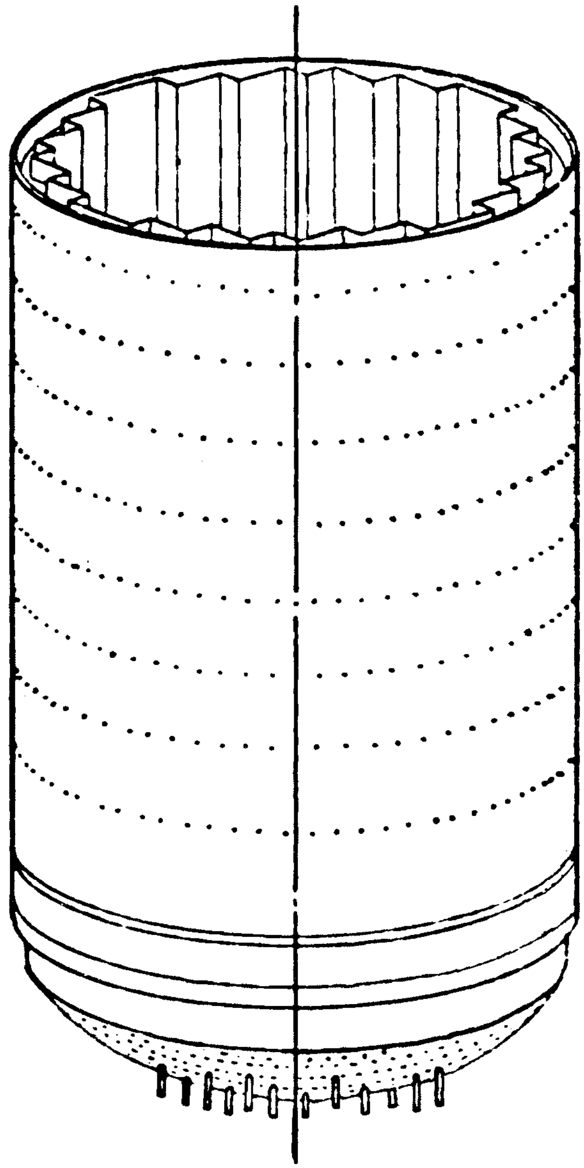
Design of Zircaloy Core Baffles/Formers

The concept of replacing the stainless steel core baffles and formers with Zircaloy was proposed as a uranium-conserving feature to be incorporated into the reference Babcock-241 reactor. The lower neutron absorption cross section of Zircaloy compared to steel reduces the parasitic capture of neutrons that leak from the core, and permits more of those neutrons to be reflected back into the core where they can be productively utilized.

The objective of this task was to perform a preliminary engineering study to formulate how to incorporate Zircaloy baffles and formers into the modified Babcock-241 reactor. The baffle would surround the core and radial blanket. The engineering study was to result in a description of the main features of such a Zircaloy design, identification of items that would require additional development, and a preliminary assessment of the technical feasibility of implementing such a concept.

The core baffles and formers are a part of the core basket assembly, which includes the baffle plates, the former plates, and the core barrel. The core basket assembly together with the lower grid and flow distributor assembly (Figure 4.2) provide flow distribution, neutron shielding, alignment of the lower end of the fuel assembly, and guidance for the in-core instruments. Specifically, the core basket guides the flow through the core and contributes to maintaining uniform axial flow throughout the core. During operation, reactor coolant from the steam generators enters one of the four inlet nozzles, flows down the annulus between the vessel and the internals, and is distributed by the lower internals. The flow continues through the core with some bypass flow around the core baffle plate to provide cooling. Exiting the core, the heated reactor coolant flows into the plenum region and is channeled to the outlet nozzles. Some flow goes up the control rod guide tubes to the upper head annulus and then down to the outlet nozzle.

The baffle plates are bolted to the former plates, which are then bolted to the inside of the barrel to provide the proper shape to surround the core.



Babcock & Wilcox
a McDermott company

FIGURE 4.2. B&W Core Basket, Lower Grid, and Flow Distributor Assembly

There are eight levels of former plates with two distinctive types of plates used at each elevation. A single flat former plate forms the core flat on the axis and located on each side of the axis is a stepped plate. Each of these plates is fastened to the core barrel with stainless steel bolts. Flow holes are provided in each former plate to allow a small percentage of reactor coolant flow to bypass the core for the purpose of cooling the core barrel.

The stacked vertical baffle plates span nearly the full length of the core barrel. These plates (often referred to as the core liner) include a tongue and groove joint that allows for differential thermal expansion between zirconium and stainless steel.

4.2 RAPID/FREQUENT REFUELING STUDY

This concept involves major modifications to plant design, fuel usage, and both maintenance methods and fuel handling operations for implementation. Refueling at semi-annual intervals while the reactor is at cold shutdown conditions was a basic guideline for the study. Uranium utilization improvement results from increasing the number of fuel batches resident in the core, thereby reducing core average exposure at end-of-cycle (EOC). This reduces the fraction of neutrons absorbed in both fission products and control poisons (such as boron) and decreases core excess reactivity requirement, thereby decreasing the ^{235}U enrichment required in fresh fuel for it to reach goal exposure at discharge.

Objective

The goal of this effort was to develop a viable procedure and provide a conceptual equipment design that would make frequent refueling an economically justifiable means for efficient uranium utilization in PWRs. Procedural changes were to be reviewed and justified and equipment modifications described in sufficient detail to demonstrate the capability of their commercial production. A secondary objective was the determination and listing of areas outside the program scope that would further justify the frequent refueling concept but which would require additional development.

Conclusions

It has been conservatively estimated that semi-annual refueling will be economically viable for approximately 25% of the total PWR installed capacity in 1995, based on the projected costs of replacement power and uranium oxide and the ability of these plants to accomplish a refueling-only outage in 11 to

13 days. Although the decision as to which plants can and should practice frequent refueling can only be made when the specific economic environment of that plant is known, it is concluded that the necessary equipment for frequent refueling should be installed since this capital investment can be justified based on the amount of time that such equipment can save over the lifetime of the plant.

Using advances in equipment design, modified procedures and special tooling, in conjunction with a fuel management program that minimizes the fuel handling requirements, it is concluded that a semi-annual refueling-only outage can be performed in 11 to 13 days. Although the prerequisite in-service inspections can be performed within this time frame, the multitude of maintenance and test items that must be performed on the balance of plant items can only be scheduled for the annual refueling outage, whose length is typically controlled by these items rather than those items on the refueling critical path.

It was determined that significant time savings in overall fuel handling operations could be affected by utilizing two independent fuel handling systems capable of simultaneous parallel fuel handling operations within the containment building. Various concepts were examined and it was concluded that two independent cantilever supported refueling machines offered the greatest potential for reducing fuel handling time. These refueling machines are illustrated in Figure 4.3. Results of the design review of this concept, which also employs two upenders and transfer systems, have shown that it is not only feasible but less complicated and more economic than the other systems investigated.

The in-depth review of both equipment and procedures has indicated that utilization of the proposed advanced techniques for reducing critical path activity times should be cost effective even for those plants that do not anticipate adoption of a frequent refueling fuel cycle. The cost of implementing the necessary modifications to allow the use of the semi-annual refueling concept is estimated at less than 4% of total physical plant cost, which would be quickly amortized by the resulting reduction in the cost of replacement power. Installation of this equipment during the construction

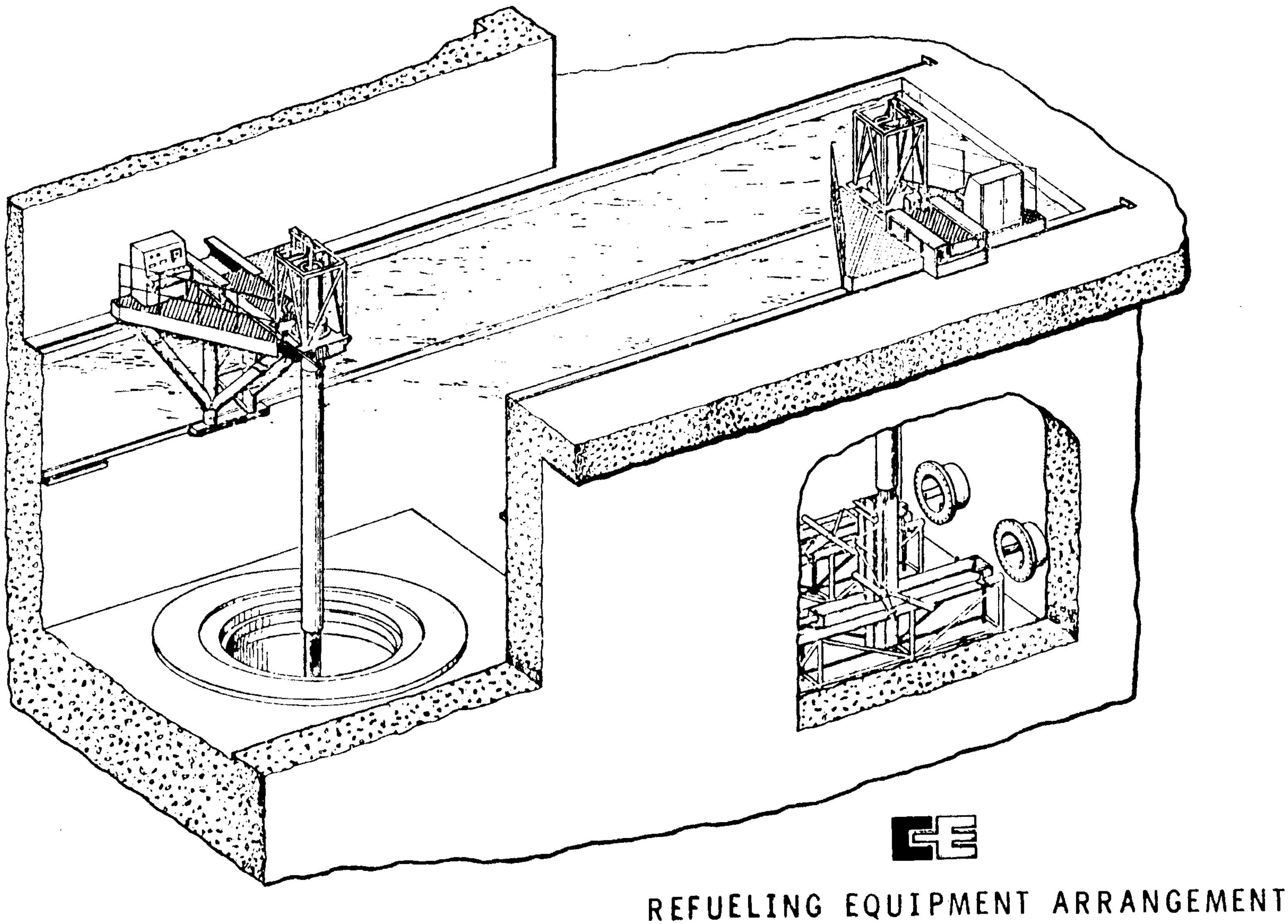


FIGURE 4.3. C-E Dual Independent Fuel Handling System

stage would allow the decision to utilize semi-annual refueling to be made whenever it becomes economically feasible for that particular plant.

Recommendations

This effort has produced results that not only support the concept of improving uranium utilization through frequent refueling but also provide a direction to be taken that will increase plant availability factors by decreasing the time required for a typical refueling and maintenance outage.[8] Some of the proposals can be utilized at plants presently in operation while others will require modification and additional equipment prior to their use. It is with this in mind that the following recommendations are made.

Cantilever Supported Refueling Machines

In order to expedite utilization of the cantilever supported refueling machine concept, it is recommended that DOE, in conjunction with a participating utility, sponsor a program to demonstrate the practicality of implementing the design. This program would include the completion of detailed design drawings, prototype fabrication, and equipment testing at a suitable facility. Although the overall design differs from equipment presently in use, the detail components are quite similar, which would allow this project to be completed in approximately two years.

Design and Procedural Improvements

With the design changes and procedural improvements developed in this study as a starting point, further plant modifications should be investigated that will eliminate or substantially reduce the time to perform the refueling critical path items to provide a positive economic incentive for adopting frequent refueling. This should include the optimization of low power and power ascension physics testing with initial programs developed to meet the requirements of a particular reactor plant, interactions with NRC for program approval, and start up design predictions to compare against the optimized measurements. The design changes can most expeditiously be accomplished through the participation of a reactor vendor and a utility that has a plant entering the initial phase of construction. The startup and power ascension physics testing could be implemented in a plant which is presently operating.

Balance of Plant Maintenance Items

A comparison of the refueling critical path schedule with the balance of plant schedule confirms the fact that the length of yearly refueling outages is presently controlled by the amount of time required to perform the many maintenance items. It is therefore proposed that an industry-wide program be initiated to upgrade this equipment such that maintenance procedures are simplified and testing of equipment prior to start-up is drastically reduced. This might best be accomplished by involving the manufacturers of turbines, pumps, electrical systems, and other balance of plant items in a DOE sponsored development program that would allow a detailed review of maintenance procedures for those items with excessive maintenance requirements.

Hot Standby Refueling

Results of the economic study indicate that the economy of frequent refueling improves as the length of the refueling outage decreases. Extrapolation of these results leads to the conclusion that a refueling procedure that could be performed without removing the reactor vessel head would be able to be accomplished in a minimum amount of downtime and therefore be most conducive to the frequent refueling concept. Although this is considered a radical departure from current technology, a feasibility study sponsored by DOE should be conducted to determine the practicality of an LWR design that utilizes hot standby refueling.

5.0 ASSESSMENT OF SELECTED NONBACKFITTABLE LWR CONCEPTS BY INDUSTRY

The industrial assessment study consisted of an initial workshop, a review and assessment period, and a final workshop. The initial workshop defined the framework for the assessment effort so that the work would be performed in a consistent, comparable manner by all participants. Nonbackfittable design modifications were selected and assessment criteria were established. Key questions relating to each modification were identified and a methodology for rating each modification and for ranking the concepts relative to each other was developed. During the two-month review and assessment period the selected nonbackfittable design modifications were evaluated based on the criteria established in the initial workshop. The industrial participants (Babcock & Wilcox, Combustion Engineering, and General Electric) answered the key questions for each design modification and made recommendations as to what, if any, additional effort should be expended on developing the selected design modifications. PNL reviewed the industrial assessment results and prepared the information for presentation at the final workshop. During the final workshop, each participant reviewed the results to determine areas of agreement and disagreement. A consensus was reached regarding the final results and conclusions of this study. Specific recommendations were made in terms of further design effort and/or hardware development and demonstration. Separate reports on the assessment of nonbackfittable PWR concepts were published by Babcock & Wilcox[9] and Combustion Engineering[10] to document their results.

5.1 DESCRIPTION OF NINE CONCEPTS SELECTED FOR ASSESSMENT BY INDUSTRY

Prior to the initial workshop, the assessments began with an industrial review of background material provided by a preliminary PNL review of nonbackfittable concepts for improving uranium utilization in LWRs. From the 28 concepts reviewed by PNL (listed in Table 3.1), the participating U.S. reactor vendors selected nine concepts as the most promising candidates for more detailed evaluation. The industrial participants selected the following concepts for assessment:

- PWR rapid/frequent refueling at cold shutdown
- BWR rapid/frequent refueling at cold shutdown
- low power density cores
- radial blankets
- core peripheral modifications
- spectral shift/end-of-cycle coastdown
- higher temperatures and pressures
- small PWR fuel assemblies
- soluble boron for BWR cold shutdown.

Each of these concepts is described briefly in the following sections.

Frequent Refueling at Cold Shutdown

Both the PWR and BWR concepts for rapid/frequent refueling at cold shutdown achieve improved uranium utilization by increasing the number of resident core fuel batches, thereby reducing core average exposure at end-of-cycle (EOC). This reduces the fraction of neutrons absorbed in both fission products and control poisons (such as boron) and decreases the ^{235}U enrichment required in the smaller fuel batches. A six-month cycle length was assumed for this concept, which doubles the number of core fuel batches from that for the reference annual cycle length. A short reactor outage of about 11 days for refueling would be alternated with a normal annual outage for both maintenance and refueling. Rapid refueling is a prerequisite for the economic feasibility of frequent refueling. In addition to reducing fuel handling time, equipment modifications would be required to reduce the time necessary for disassembly and reassembly operations.

Low Power Density Cores

Low power density cores provide reactivity gains from reduced neutron absorption in saturating fission products (such as xenon) due to lower neutron flux levels, reduced neutron absorption in ^{238}U due to less Doppler effect resulting from lower fuel temperatures, use of more in-core fuel batches as a result of a greater fuel loading for a fixed fuel discharge exposure, and reduced in-core neutron leakage due to a reduction of the surface-to-volume ratio of the core. These reactivity gains offer the potential for improved uranium utilization resulting from the reduced fresh fuel enrichment requirement. In addition to these direct savings in U_3O_8, the potential exists for further savings as a result of increased thermal margin. For example, increased core power peaking is usually accompanied by higher core reactivity due to increased flux weighting of the high reactivity fuel (which operates at high power density). Also, lower beginning-of-cycle (BOC) burnable poison concentrations can be used in low-leakage fuel management schemes if constraints on power peaking can be relaxed. Accompanying these lower BOC concentrations will be lower EOC concentrations (i.e., lower shim residuals), which will increase EOC reactivity and provide additional life for a given fresh fuel enrichment.

The low power density core was assumed to be implemented by increasing the diameter of the reactor vessel and internals, maintaining the same net plant electrical output, and adding at least one additional row of fuel assemblies at the periphery of current core configurations. Core volume increases of 14 to 45% were considered. Current spacing between the core, core support barrel, and reactor vessel would remain unchanged to maintain coolant flow velocity between the vessel and the barrel and to limit fast neutron fluence to the vessel.

Radial Blankets

By adding a radial blanket composed of fuel assemblies containing fertile material at the core periphery, it is possible to increase LWR uranium utilization. The increased utilization is due to three factors. First, the radial blanket shifts the neutron flux to the active core periphery, placing fissile material in relatively more important core locations. Second, the radial blanket acts as a reflector surrounding the active core, reducing core neutron leakage, and thus, increasing core reactivity. Finally, neutrons that leak from the active core are available in the radial blanket for conversion of fertile material to fissile fuel. This conversion leads to increased fissile material production, increased power production in the radial blanket, and substantial in situ consumption of the fissile material produced in the blanket during its service life. Four sources of fertile material were considered: natural uranium, depleted uranium (0.2 wt% ^{235}U), thorium, and spent fuel. Spent fuel has a shorter blanket life because of its prior irradiation, but it is readily available at no extra cost and has the highest fissile content, which will contribute to a greater reactivity gain. For BWRs, thorium corner rods were assumed to be placed in fuel assemblies and subsequently reconstituted into radial blanket assemblies for extended exposure following their discharge from the core.

For the purposes of this assessment, it was assumed that the radial blanket assemblies were added as an additional row on the periphery of current core configurations, while enlarging the pressure vessel and core support barrel, similar to that discussed earlier for low power density cores. Flow control orifices may be employed on the radial blanket assemblies to match coolant exit conditions from the fuel assemblies in the core. The service life of radial blanket assemblies containing fresh fertile material was assumed to range from 10 to 30 years. Following its irradiation in the core, spent fuel was assumed to reside in the radial blanket for only one year.

The potential improvement in LWR uranium utilization from the use of a radial blanket in a PWR, for extending its use in a BWR, competes with the potential improvement from use of a low power density core since both concepts require additional space inside the reactor vessel. Furthermore, a radial blanket containing spent fuel could be considered to be a low power density core with an extensive low-leakage fuel management scheme. These concepts should be considered together to optimize potential improvements. The radial blanket design optimization also depends on the blanket material, the water-to-fuel ratio in the blanket assembly (the optimum of which varies with blanket material), the blanket thickness, the reflector surrounding the blanket, the fuel management scheme employed in the reactor core, and the service life of the blanket.

Core Periphery Modifications

Uranium utilization improvements can be obtained by reducing neutron leakage from the radial boundary of the reactor core or blanket. Presently, neutrons leaving the outer boundary of a PWR core intersect a small water gap (~0.15 in. thick), a stainless steel core shroud (~1.0 in. thick), another water gap (~7 in. thick on the average), and then the core support barrel

(~2.6 in. thick). The majority of neutrons that leave the core are fast neutrons (>0.1 MeV). Some are reflected back toward the core as fast neutrons, while others are absorbed or are reflected back to the core as thermal neutrons. Improvements in uranium utilization can thus be made by increasing the number of fast neutrons reflected back to the core, and by reducing neutron absorption in these ex-core regions so that more neutrons can diffuse back into the core or radial blanket. The presence of a radial blanket reduces neutron leakage from the core's outer boundary and thus reduces the potential uranium utilization improvement from a reflector surrounding the blanket.

Several PWR modifications were considered to accomplish this increased neutron reflection. These modifications include increasing the existing water gap between peripheral assemblies and the core shroud, increasing the thickness of the stainless steel core shroud, replacing the stainless steel shroud with a Zircaloy shroud, and substituting alternate materials such as Be, BeO, or graphite for the existing reflector materials. Because BWRs have larger cores and individual BWR fuel assemblies have Zircaloy shrouds, adding another reflector material was not considered effective use of vessel space.

Spectral Shift/End-of-Cycle Coastdown

End-of-cycle coastdown can improve uranium utilization by providing the excess reactivity necessary to allow the reactor to operate at full thermal power beyond its nominal EOC life and thereby accumulate additional burnup on the fuel. This excess reactivity is obtained at EOC by several different methods, depending on the nuclear steam supply system, that accomplish an increase in moderator density.

In a PWR, average moderator density is increased by decreasing its average temperature to provide the excess reactivity needed to match reactor power with steam demand. To transfer the same amount of heat to the secondary system with a lower primary coolant 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 progressively opening the admission valve to its wide-open position. However, after reaching the valve-wide-open (VWO) position, the plant power level decreases because the turbine can no longer draw enough steam to produce rated power.(10)

Primary moderator temperature reduction for EOC coastdown in PWRs can be accomplished by decreasing the secondary system's feedwater temperature and reducing superheat to the minimum value required to prevent moisture carryover.

In this mode of operation, extracted steam flow to the feedwater heaters is sequentially decreased. With a lower feedwater temperature, the secondary fluid can experience a greater enthalpy rise for a given steam generator outlet pressure. In addition, the decrease in extraction steam flow for the feedwater heaters permits lower pressure operation before the volumetric flow limit on the turbine admission valve is reached. This mode of operation permits extended reactor operation at rated thermal power after the VWO position is reached but at a lower thermal efficiency.(10)

At that point, normal power coastdown is initiated. This 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 continue operating at declining power, accumulating additional fuel burnup. Normal power coastdown has been practiced by a number of utilities primarily to obtain cycle length flexibility when it was advantageous to delay a refueling outage.

In a BWR, moderator density is increased at EOC by increasing the reactor water recirculation rate, which increases the height in the core at which boiling occurs and reduces the fraction of core steam voids. In addition, moderator density can be decreased early in the cycle by decreasing the reactor water recirculation rate to lower the height at which boiling occurs in the core, reducing reactivity when large excesses must be controlled. Recirculation flow control could be extended to a range of 40 to 150% of rated flow to extend BWR spectral shift. This mode of operation reduces the parasitic neutron capture in boron control rods and increases the conversion of ^{238}U to plutonium during the early part of the cycle. This additional fissile material is substantially burned in situ later in the cycle as the water flow rate is increased and the moderator density is increased to improve neutron thermalization. Uranium utilization is improved by the increased conversion and in situ consumption of fissile material, by the additional excess reactivity provided by the increased moderator density at EOC, and by the improved heat rate and cycle efficiency which would occur due to reduced power input to the pumps during the cycle.

Higher Temperature and Pressures

All the other approaches to increasing uranium utilization in LWRs achieve the improvement through the extraction of additional thermal energy from a given amount of uranium fuel. The basic method of increasing uranium utilization through higher temperatures and pressures, however, is based on improving the conversion efficiency of thermal energy into electrical energy so that more electrical power can be generated from a given amount of consumed uranium. This increase in thermal conversion efficiency is accomplished by increasing turbine throttle pressure and inlet steam temperature conditions.

A significant improvement in turbine cycle efficiency requires a correspondingly significant increase in reactor coolant system outlet temperature. For PWRs, primary system coolant outlet temperature increases of up to 50^{o}F were considered. This would result in a maximum cladding surface temperature of about 700^{o}F, and primary system pressures of up to 3100 psia. A more conservative PWR approach was also considered. This approach would increase reactor coolant outlet temperature 15^{o}F by using flow control, reduced peak-to-average power distribution in the core, and improved thermal analysis methods, while the system pressure remains at 2250 psia. The maximum cladding surface temperature of about 665^{o}F for this approach is a reasonable bound for acceptable corrosion and creep behavior on Zircaloy materials currently used.

The increase in cycle efficiency, and hence the increase in electrical output, is not linear with respect to the increase in turbine throttle pressure. At elevated pressures, the increase in cycle efficiency achievable with increasing saturated steam pressure is increasingly less significant. To maximize cycle efficiency, some of the potential increase in steam pressure should, in practice, be converted into superheat by appropriate steam-generating equipment.[10]

For BWRs, there is a larger margin for increased reactor coolant pressure and temperature than for PWRs because the operating pressure of current BWRs (1050 psia) is less than half that of PWRs and cladding surface temperatures are about 22^{o}F lower. To make a significant improvement in steam turbine efficiency, an increase of about 250 psi (to a pressure of 1300 psia) in the reactor coolant system was considered. This would correspondingly raise the outlet saturated steam temperature by about 50^{o}F. Since the net thermal efficiency in this range increases 1% for every 100 psi increase in steam pressure, thermodynamic efficiency would be increased from 33.4 to 35.9%. It would be necessary to modify current BWR turbine design to accommodate the higher steam pressure. In addition, it would be necessary to increase the vessel and primary system piping thicknesses and to increase the internal jet pump capacity. The additional capital costs involved are believed to balance the higher efficiency so that the higher pressure design is near the economic optimum. In practice, the optimum BWR pressure is controlled by turbine pricing, but is also sensitive to jet pump capacity and heat flux maxima.

For PWR designs employing higher reactor coolant temperature and pressure, it would be necessary to reoptimize the fuel rod diameter and lattice pitch. At higher moderator temperatures, the neutron spectrum is hardened if the other lattice parameters are unchanged. This results in greater neutron resonance capture in ^{238}U. The greater neutron capture, in addition to producing more plutonium, also lowers the core reactivity during the operating cycle; this lower reactivity, in turn, is compensated by reduced absorptions in the control poisons. By modifying the lattice to restore the H/U atom ratio to a level approximating that for current (lower pressure and temperature) reactor coolant conditions, the reactivity necessary to reach current cycle burnups can be regained with the same EOC fissile inventory. It may also be necessary to use a more corrosion-resistant cladding material at high temperatures and a thicker cladding to prevent its collapse onto the fuel column early in life. These modifications to present fuel designs could adversely affect fuel cycle costs. Uranium utilization improvements potentially available from higher temperatures and pressures are essentially due only to thermodynamic efficiency gains.

In the BWR as pressure is increased, steam void fraction is decreased, which compensates for greater moderator temperature and makes the spectrum softer by increasing the H/U ratio. The optimum BWR lattice design trend with increased pressure is toward increases in fuel rod diameter. This adds heat transfer surface to help compensate for poorer critical heat flux performance with increasing pressure. The BWR can also be spectral shifted with pressure by operating at lower pressure, higher conversion ratio throughout the cycle followed by increasing pressure at the end of cycle to maximize reactivity. This alternative would reduce thermodynamic efficiency compared to operation continuously at higher pressures.

Small PWR Fuel Assemblies

Use of smaller PWR fuel assemblies can provide several small uranium utilization gains by permitting better mixing of fresh and highly exposed fuel, reducing the lumped burnable poison needed to control power peaking in fresh fuel, extending use of low-neutron-leakage fuel shuffle schemes, and increasing average fuel discharge exposure for the same maximum exposure. This concept would use a fuel assembly with perhaps one-fourth of the cross-sectional area of current PWR assemblies. These small assemblies would also mitigate the effects of fuel rod failures and provide greater flexibility in accommodating a more cylindrical peripheral core shape and a radial blanket.

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

With fresh fuel loaded in smaller assemblies, lumped burnable poison requirements to help control local power peaking are reduced. As a result, EOC core reactivity is increased due to reduced neutron absorption in residual poison shim. Particularly, in low-leakage fuel management schemes where fresh fuel is loaded in the core interior and highly exposed fuel is loaded on the core periphery, the burnable poison concentration can be lower, resulting in more complete poison material burnout by the end of this fuel's first operating cycle.

The large size of current PWR assemblies precludes dedication of peripheral core locations for use as a blanket region. Such dedication would necessitate an increase in the linear heat rate of the active core region in order to keep the same overall plant rating. By using small assemblies, a core blanket of one-half assembly thickness and composed of highly burned fuel could easily be constructed. The resulting increase in linear heat rate for the remainder of the core would be much less than when the entire peripheral row of assemblies is used as a blanket.

Soluble Boron for BWR Cold Shutdown

For BWR systems with higher initial ^{235}U enrichments or longer cycle lengths, such as 18-month refueling intervals instead of annual intervals, BOC reactivity at ambient water conditions may be too high to assure adequate shutdown reactivity margins entirely with the control blades. This requires the addition of gadolinium burnable poison to reduce BOC reactivity and provide power distribution control during burnup. The gadolinium required only for power shaping would be insufficient to provide the required negative reactivity for cold shutdown, so its concentration would have to be increased to provide an ample BOC shutdown margin. The residual reactivity poison from excess gadolinium loading would shorten the cycle length if additional initial fuel enrichment was not provided for compensation. In practice, increased natural uranium and separative work provide additional fuel enrichment to maintain the desired cycle lengths. The use of soluble boron to maintain cold shutdown margins would eliminate the need for excess gadolinium.

The soluble boron system includes an injection and removal system for introducing a concentrated boric acid solution into the water moderator and retrieving the boron from the diluted solution before returning to power. The soluble boron system would only have to be used when the moderator is below saturation temperatures so the boric acid would be confined to the reactor vessel. Potential uranium utilization improvements of 2 to 5% are expected from using the soluble boron system to obtain the required BWR cold shutdown reactivity margin.

5.2 CRITERIA SELECTED BY INDUSTRIAL PARTICIPANTS FOR ASSESSMENT OF SELECTED CONCEPTS

Criteria used to assess each concept's implementation potential were established by a consensus of industrial participants at the initial workshop. Five main categories were used to group the assessment criteria:

- uranium utilization
- economics
- technology
- operation
- other considerations.

These criteria are detailed in Table 5.1.

In the uranium utilization category, the measures used to rate the concepts included cumulative uranium requirements (short tons U_3O_8/GWe) for

TABLE 5.1. Assessment Criteria Selected by Industrial Participants

Uranium Utilization	Operation
U_3O_8 Savings	Reliability
SWU Savings	Availability
	Operability
Economics	**Other Considerations**
Δ Capital Cost	Utility Acceptance
Δ Fuel Cycle Cost	Date of First Commercial Operation
Δ Power Generation Cost	Compatibility with Other Concepts
Development Cost	Potential for Retrofit
Δ Construction Time	Nonproliferation
Technology	
Technical Feasibility	
Safety	

30 years of plant operation at an average capacity factor of 75%; percent improvement over the base case;[a] and cumulative separative work requirements for 30 years of operation.

In the economics category, the measures used to rate the concepts included relative power costs, concept development costs, and plant construction time impacts. The following factors were evaluated to determine relative power cost:

- changes in plant capital cost from the base case ($/kWe)
- fuel cycle costs (mills/kWh levelized over 30 years of operation for two uranium prices: $40/lb U_3O_8 and $100/lb U_3O_8)
- estimates of the percentage change in fuel cycle costs from the base case
- estimates of the average capacity factor, if changed from the 75% value used in the base case.

Development costs include all costs incurred by a reactor vendor up to the point when a firm bid proposal is offered for a system using the concept. The concept's impact on construction time was determined by the difference in time for plant design, licensing, and construction from the 10-year interval assumed in the base case.

In the technology category, both technical feasibility and safety aspects were evaluated. The following areas were considered:

- complexity
- degree of demonstration
- state-of-the-art
- manufacturability
- impact on the balance of the system.

The potential for concept retrofit was also considered from the standpoint of whether it would be possible to make minor design changes when the plant is built that would accommodate a later retrofit, if it becomes desirable.

(a) The base case is the composite improved PWR and BWR designs reported in Reference 1.

In the operation category, reliability, availability, and operability of the plant employing the concept were assessed. Plant reliability was evaluated considering maintainability, inspectability, and equipment lifetime. Estimates were made of the change in plant availability and capacity factor when possible. Plant operability was rated by considering changes in:

- number of staff
- quality of staff
- radiation exposure of staff
- plant maneuverability
- plant simplicity.

Other considerations were addressed in the industrial assessment criteria. A subjective judgment was provided concerning the acceptability of the concept to a utility ordering a new plant, in terms of the product's quality image, licensability, and adapatability for future needs such as load following. An estimate of the date for first commercial operation of a plant employing the concept was made, assuming the first step in the development process was begun in FY1981. The compatibility of the concept with nonproliferation objectives was assessed considering to what extent there is access to sensitive material, and whether safeguards changes would be required. The viability of the concept was considered in the context of whether the plant could be converted to recycle plutonium if national policy were to change during mid-life. Finally, a subjective analysis was provided to consider the interaction the particular concept under assessment has with the other selected nonbackfittable concepts and with backfittable concepts included in the base case.

Most of the key questions to be evaluated for each concept were considered in the process of developing the information necessary to systematically rate each concept according to the assessment criteria selected. These criteria proved to be relevant and discriminating, and as such were useful for accomplishing the assessment's objectives.

5.3 RESULTS OF CONCEPT ASSESSMENTS

Babcock & Wilcox and Combustion Engineering participants assessed the seven PWR concepts and General Electric assessed the seven BWR concepts.

Separate reports on the industrial assessments were published by Babcock & Wilcox[9] and Combustion Engineering.[10] General Electric's efforts were limited to the extent that they did not issue a comparable separate report. The most significant results of the concept assessments are listed in Table 5.2. The five most important criteria for determining a concept's relative implementation potential were uranium savings, decrease in fuel cycle cost, development cost, first commercial operation date, and decrease in power generation cost. Assessment criteria in the categories of technology, operation, and other considerations, described in Section 5.2, tended to be more subjective and as a result, less discriminating than the five quantitative criteria used to rate the concepts in Table 5.2. However, the judgments made for the more subjective categories are reflected in the quantitative criteria so the results are consistent.

After rating the concepts against the assessment criteria, the industrial participants ranked the concepts against each other in terms of relative implementation potential. The relative ranking of the individual PWR concepts is shown in Table 5.3. The PWR vendors also supplied a weighting factor to measure the relative worth they would assign to the ranked concepts. The agreement on concept ranking between the two PWR participants is apparent. The highest ranked design modification is rapid/frequent refueling, followed closely by the combined concept of low power density core, radial blanket, and core periphery modifications. These three concepts are viewed as a combined PWR package, since they all compete for reactor vessel space and capitalize on neutron core leakage. End-of-cycle coastdown and small PWR fuel assemblies were ranked considerably lower. The lowest ranked PWR concept is the use of higher temperatures and pressures.

The relative ranking of the individual BWR concepts is shown in Table 5.4. The General Electric participants ranked the concepts into two categories: those with greatest implementation potential, and those with marginal implementation potential. The highest ranked BWR design modification is spectral shift using extended flow control, followed closely by rapid/frequent refueling. Higher temperature and pressure in a BWR is judged to be the best design improvement opportunity, but was ranked in third position because of potentially adverse impacts on plant reliability. A somewhat lower ranking for low

TABLE 5.2. Results of Industrial Assessment

Concept	Uranium Savings (%)	Decrease in Fuel Cycle Cost (%)	Estimated Development Cost (Millions $)	First Commercial Operation Date	Decrease in Power Generation Cost (%)
Frequent Refueling	7-8 (PWR) 6 (BWR)	~8	6-18 (PWR) 25 (BWR)	1994-2002	~2[a]
Radial Blanket	2-6 (PWR) 3-4 (BWR)	2-6	5-10	1995-1999	~1
Low Power Density	3-5 (PWR) 2 (BWR	0-4[b]	8-15	1995-2000	0-1
Core Periphery Modifications	1-2	1-2	5-15	1995-2000	No change
Coastdown	1-1.5 (PWR) [c] (BWR)	1 (PWR) [c] (BWR)	2-4 15-20 (BWR)	1991-1994 (PWR) 1995 (BWR)	0-1
Small PWR Fuel Assembly	1-3	~2	61-12	1999	No change
Higher Temperature and Pressure	2-4 (PWR) 8 (BWR)	2-3 (PWR) 6 (BWR)	15-50	1996-2002	0-1
Soluble Boron in BWR	2-5	1-3	5	1990	0-1

(a) Based on zero replacement power cost.
(b) Dependent on fuel inventory charges.
(c) Estimates range from 6% to 11% due to spectral shift with flow control.

TABLE 5.3. Industrial Ranking of PWR Concepts (Ranking by Organization)

Concept	Combustion Engineering		Babcock & Wilcox	
	Ranking	(Weighting Factor)	Ranking	(Weighting Factor)
Rapid/Frequent Refueling	1	(1.0)	1	(1.0)
Radial Blanket/Low Power Density/Core Peripheral Modifications	2	(0.8)	2	(0.8)
Extended Coastdown	4	(0.4)	3	(0.5)
Small Fuel Assembly	3	(0.4)	4	(0.4)
Higher Temperature and Pressure	5	(0.0)	5	(0.1)

power density cores was due to smaller potential gains assessed for BWRs. Extended coastdown and radial blanket concepts were judged to have marginal implementation potential because these concepts are already deployed to a limited extent in current BWRs and further potential gains are small. The use of

TABLE 5.4. Industrial Ranking of BWR Concepts

Concept	General Electric Company Ranking	
Spectral Shift	1	Concepts with Potential for Implementation
Rapid/Frequent Refueling	2	
Higher Temperature and Pressure[(a)]	3	
Low Power Density	4	
Extended Coastdown		Concepts with Marginal Implementation Potential for Extension Beyond Limits of Backfittability
Radial Blanket		
Soluble Boron		Concepts Determined to be Backfittable

(a) Best opportunity offset by potential adverse impact on reliability.

soluble boron for cold shutdown reactivity control was considered to be backfittable in existing BWRs, so extending this concept to its nonbackfittable limit has marginal potential for improved uranium utilization.

5.4 CONCLUSIONS FROM INDUSTRIAL ASSESSMENTS

Industrial participants developed conclusions at the final workshop based on consensus of the concept assessments. These conclusions are listed by concept as follows:

Rapid/Frequent Refueling

Significant uranium savings are potentially available if rapid refueling equipment is used to implement frequent refueling. Furthermore, semi-annual refueling has potential economic viability for a substantial fraction of nuclear utilities that have replacement power costs below 20 mills/kWe hr.[8] Uranium savings potentially available from semi-annual refueling is about 8% for PWRs and 6% for BWRs employing extended burnup fuel. Savings are higher for plants using current fuel exposures. Development costs for rapid/frequent refueling appear to be reasonable in relation to potential benefits. Utility acceptance of rapid refueling equipment modifications and outage management changes should be high because of improved plant availability, reduced occupational dose exposures, and increased fuel management flexibility. However, utility decisions to use rapid refueling equipment for semi-annual refueling is questionable.

Low Power Density/Radial Blanket/Core Periphery Modifications

Three concepts: low-power-density cores, radial blankets, and core periphery modifications, should be considered together as a group because they compete for space inside the reactor vessel. Each of these concepts is feasible and improves uranium utilization. However, the PWR or BWR designs combining these concepts should be optimized to maximize potential benefits. No further consideration should be given toward the use of graphite, beryllium metal, or BeO reflector materials on future LWRs. On the other hand, the use of spent fuel in the radial blanket merits further consideration in the PWR. Current vessel sizes are not at the upper limit of existing technology, and further size increases should be considered. When the pressure vessel and core support cylinder of a BWR is increased in diameter to accommodate a lower power density core and a radial blanket, additional control blades may be required at the core periphery to handle the possibility of a misloaded high reactivity bundle in the radial blanket. For PWRs, the substitution of a Zircaloy core shroud for stainless steel in current designs should be considered as the best opportunity for improving the radial reflector. BWRs already use Zircaloy bundle shrouds and have small potential for implementation of an additional radial reflector.

End-of-Cycle Coastdown/Spectral Shift

The extension of normal coastdown by sequential reduction of bypass steam flow to feedwater heaters is a relatively simple and inexpensive method to improve uranium utilization if turbine modifications are not required. Modifications to steam turbine designs were beyond the expertise of the assessment group involved in this study. However, additional benefits may be available if turbine modifications are made to accept increased volumes of steam at reduced pressure and increased moisture content. Such modifications to extend coastdown are beyond retrofittable capabilities for PWRs. For BWRs, the spectral shift concept with expanded flow control capability provides the best opportunity for EOC coastdown. However, pump capacity and recirculation loop size may have to be increased and vessel internals strengthened to accommodate 150% EOC flow rates. An alternative is the use of internal pumps, which are being considered for advanced BWRs.

Higher Temperature and Pressure

For BWRs, small increases in temperature and pressure appear to be feasible. The economic optimum pressure for BWRs may be about 1300 psia, which is 250 psi above current operating conditions. In this range, thermal efficiency increases by 1% per 100 psi increase in pressure. The potential uranium savings from a 250 psi increase in pressure are about 8%.

For PWRs, higher pressures are not feasible or desirable. However, some small temperature increases at the same pressure may be feasible. The potential uranium saving from a 15^{o}F temperature increase in the primary coolant outlet conditions is only about 3%.

Small PWR Assemblies

Evaluated independently from other PWR concepts, small assemblies provide uranium utilization gains of only 1 to 2%. Smaller assemblies may be desirable for the implementation of other PWR improvements, however. For example, smaller assemblies may facilitate the use of a radial blanket and a more circular core periphery in PWRs. On the other hand, the development effort to permit handling substantially larger numbers of smaller fuel assemblies may be considerable in relation to potential benefits, so its implementation is questionable.

Soluble Boron for BWR Cold Shutdown

This concept has the potential for permitting gadolinium loadings to be reduced. It would be required to serve as a backup system when the core is in a high reactivity condition, such as at cold shutdown. A controlled amount of boron would be injected in case of a concurrent stuck control rod and the need for reactor shutdown to cold conditions. The frequency of the system's use is estimated to be one time per 5 reactor lifetimes. Potential uranium savings of more than 2% could result from the reduced gadolinium loadings that would be permitted. Development cost for demonstration of the concept is about $5 million.

5.5 RECOMMENDATIONS FROM INDUSTRIAL ASSESSMENTS

Specific recommendations were made by the industrial participants with regard to the development and demonstration efforts that they consider useful as next steps toward implementation of the nonbackfitable LWR concepts included in the assessment study. These recommendations are listed by concept as follows:

Rapid/Frequent Refueling

- Perform a limiting factor analysis on outage times to identify what maintenance operations can be reallocated between two outages to minimize total outage time.
- Determine what it would take to make a breakthrough in shortening refueling outage time to about 7 days.

- Determine actual times for outages limited only to refueling, by demonstration of PWR rapid refueling equipment at Houston Lighting and Power Company's South Texas Project. This demonstration should include a practice refueling outage before operation starts.
- Demonstrate the application of a shear pin type hydraulic closure device instead of headbolts on a BWR vessel for decreasing time necessary to gain access to the core and to reseal the primary system.
- Use the full-scale vessel refueling bridge and floor mockup at General Electric to determine actual refueling operation times to compare with theoretical minimums.

<u>Low Power Density/Radial Blanket/Core Periphery Modifications</u>

- Perform a parametric study to determine the optimum configuration for utilizing these concepts in a larger vessel, based on cost/benefit analysis.
- Develop a preliminary conceptual design for an LWR using the results of the configuration optimization study.
- Consider putting a Zircaloy shroud on a PWR presently under construction.
- Demonstration of reflector modifications could be preceded by use of triangular shaped reflector materials in unused space at core periphery and critical experiments to provide physics benchmarks.

<u>End-of-Cycle Coastdown/Spectral Shift</u>

- Perform a study to determine the limits of present turbines, modifications required to relax limits, and costs of modifying existing turbines or incorporating capabilities to accept more low-pressure steam in future turbines.
- Perform analyses of PWRs and BWRs for conditions encountered during EOC coastdown to establish a range of changes that can be made.

- Perform a design study for a specific plant, planning equipment modifications for extended coastdown capability, as Phase I of a demonstration.
- Implement Phase II of the demonstration during planned outages by making equipment changes identified by Phase I, and obtain operating data to verify performance of extended coastdown capability.
- Provide government support for the additional costs of making hardware changes in an existing plant to demonstrate extended coastdown capability.
- Demonstrate spectral shift in a BWR through extended flow control by making changes in an existing plant with government support.

Higher Temperature and Pressure

- Analyze the performance of the Big Rock Point BWR, which can operate at pressures 450 psi higher than current large BWRs, relative to increasing temperature and pressure of large BWRs in the future.
- Determine the relationship between increasing temperature and Zircaloy cladding corrosion rates.
- Perform an engineering study to optimize a design that increases PWR primary outlet temperature within existing pressure limits.

Small PWR Assemblies

- Analyze how small PWR fuel assemblies enhance the use of other concepts and improve uranium utilization under optimum conditions.
- Determine how frequent refueling would be impacted by smaller PWR fuel assemblies and how minimum shuffle fuel management schemes could mitigate fuel handling time during an outage.

Soluble Boron for BWR Cold Shutdown

- Since this concept is potentially backfittable, it should be considered for demonstration under government sponsored programs for retrofittable improvements.

6.0 ASSESSMENT OF CONCEPTS NOT SELECTED FOR STUDY

Following the industrial assessment study of nine concepts selected by the industrial participants, PNL undertook a similar assessment effort for the remaining 19 concepts that were not selected from the PNL preliminary review of 28 concepts, listed in Table 3.1. These remaining 19 concepts were judged by the industrial participants to have poor implementation potential in future LWRs during the next 10 to 20 years. Reasons for these judgments are primarily based on their perceived technical and/or economic viability. Although the 19 remaining concepts were not assessed in the same degree of detail as the 9 concepts evaluated by industry, the same evaluation criteria, described in Section 5.2, were used. In this sense, the assessment of all 28 concepts from the PNL preliminary review was made on a comparable basis. Results of the assessment of the 19 concepts not selected by industry are presented in Appendix A. Each concept is assessed separately using the same format. These concepts are presented in no particular order, which carries no implication as to preference. The only concept from this assessment that may have potential for implementation would appear to be hot standby refueling. An extensive development and demonstration program, supported by government funds, would have to be undertaken to achieve its implementation, however. None of these concepts offer potential benefits of sufficient magnitude to attract commercial development without government support.

7.0 EFFECTS OF COMBINING CONCEPTS ON URANIUM UTILIZATION IMPROVEMENTS

The concepts selected for study and reported favorably by industry have been examined for the most part individually. In this section, the general nature of the effects of combining concepts into a single advanced system are discussed. In some cases, the concepts may reinforce each other (synergism), while in other cases, they may oppose each other (antagonism).

7.1 PWR CONCEPTS

Concepts considered as potential PWR improvements are:

- Rapid/Frequent Refueling
- Extended Coastdown
- Low Power Density Cores
- Radial Blankets
- Core Periphery Modification
- Small PWR Fuel Assembly
- Higher Temperature and Pressure.

This discussion focuses on the impact that each major concept would have on other concepts with which it appears to interact. These impacts are not necessarily reciprocal. For example, adoption of "A" might not affect the incremental results of adopting "B", but adopting "B" might nevertheless affect the incremental results of adopting "A".

Whenever the impact reciprocity is asymmetric, the design utilizing both of the new features must be optimized and both concepts may not be implemented to the maximum extent possible. When the impacts are negative, this will certainly be the case. When the mutual impacts of two new features are synergistic, the optimum design will exploit both new features for maximum benefit.

Note that negative impacts of one new design feature on another do not necessarily mean that the impacted feature should not be used. It simply means that the improvements in system performance will not be as great as expected from summing the partial improvements for each of the features.

Rapid/Frequent Refueling

Rapid refueling is desirable even without the incentive of uranium economization. Decreased refueling times result in improved reactor availability and operability, and these result in economic benefits.

Frequent refueling results in fuel cycle savings, but also in reactor downtime. However, since reactor downtime is minimized by rapid refueling, frequent refueling is more favorable once rapid refueling has been achieved.

- extended coastdown - Frequent refueling will have little adverse affect on the practicality of utilizing extended coastdown routinely. Consider a rapid refueling scheme that permits an annual refueling-maintenance operation to be performed in 19 days and an additional refueling-only operation to be performed in 11 days.[8] Now add a 15-day normal coastdown (at an average 90% of maximum power) to the annual refueling scheme and two 8-day coastdowns to the semi-annual refueling schemes. For annual refueling, the maximum theoretical capacity factor over the year is:

$$\frac{(365 - 19 - 15) + 0.9(15)}{365} = 0.944$$

The semi-annual refueling scheme has a maximum theoretical availability of 0.913. If the coastdown periods are extended 3 days for the annual refueling scheme and 2 days each for the semi-annual refueling scheme, these numbers drop to 0.943 and 0.912, respectively. This is a capacity factor loss of about 0.1% for extending coastdown. The semi-annual (frequent) refueling thus imposes about the same capacity factor penalty on extended coastdown as that for annual refueling. When conditions are less than ideal (the capacity factor for normal operating periods is less than 100%), availability penalties for extended coastdown become smaller. Note that this consideration does not apply to uranium utilization improvements, where the effects of rapid refueling and extended coastdown are additive.

- <u>low power density</u> - Frequent refueling will improve the reactivity of larger (e.g., lower power density) cores and thus require less initial enrichment. This is because frequent refueling, which requires less initial excess reactivity, permits fresh fuel to be loaded in a slightly more reactive location without violating flux peaking requirements. Depending on the circumstances, the uranium utilization improvement could be about 1 to 5% larger the sum of the individual savings. Because the savings in uranium utilization from frequent refueling are about 7 to 8%, and about 3 to 5% from larger cores, a design combining these two features may yield a combined savings of up to 0.5% greater than the sum of the individual savings of 7.5% (frequent refueling) and 4% (larger cores). The design combination could improve uranium savings perhaps 12%.
- <u>higher temperature and pressure</u> - Frequent refueling would make higher temperature and pressure more difficult to engineer, primarily because of the increased frequency of thermally and mechanically cycling the primary system envelope (pressure vessel and piping). Existing practice is to design the primary pressure system so as to retain its physical properties of strength, creep resistance, and ductility over many more heatup and cooldown cycles than are expected from annual refueling requirements. Thus, forced cooldowns for regulatory and/or maintenance reasons have not been a problem so far. However, doubling the planned number of cooldowns, as for frequent refueling, would reduce the existing design margin. The degree of conservatism that it would be desirable to maintain under more severe temperature and pressure conditions would be harder to achieve.
- Frequent refueling would not affect the impact of the other concepts listed.

<u>Extended Coastdown</u>

Extended coastdown could affect the other concepts in two ways. Coastdown could either emphasize the impact of downtime introduced by other concepts, or it could change the neutronic balances that might be achieved.

- frequent refueling - Extended coastdown would have little adverse effect on the feasibility of frequent refueling, for reasons entirely reciprocal to those discussed under that heading.
- low power density - Extended coastdown designs would emphasize reactor lattices with increased negative temperature coefficients of reactivity around the operating point. These are drier, high-conversion lattices that show less reactivity as fresh fuel, but lose less reactivity on exposure. These lattices are closer to those now in use than to the wetter, reoptimized lattices that are better optimized for extended burnup. To the extent that low power density cores gain reactivity by loading fresh fuel in reactive positions, these advantages are reduced. Negative effects should be small and less than 1% of the uranium utilization savings that might be achieved (1% of 5 to 10%) would be lost.
- Extended coastdown does not affect the impact of the other concepts listed.

Lower Power Density Cores

Larger cores could affect the technical or operational feasibility of other concepts, or impact neutronic savings either favorably or unfavorably.

- frequent refueling - Since larger cores have more fuel assemblies, refueling downtime would increase slightly. Therefore, frequent refueling will be less attractive. The impact can be minimized because larger cores present more opportunity for fractional core fuel shuffling (salt and pepper loadings), but shuffling cannot be eliminated. For a 30% larger core, an extra day of refueling downtime might be realistic. If the minimum 11-day refueling cycle consists of 3 days each for cooldown and heatup, 1 day for vessel opening and closure, and 4 days for actual refueling operations, a 30% larger core would require 5.2 days (4 x 1.3) for fuel handling. Note that this has no effect on uranium utilization, but increases the reactor availability penalty by 0.1 to 0.2%.

- extended coastdown - Larger cores could make extended coastdown slightly more attractive. With less neutron leakage and more conversion (an effect that might be enhanced by reduced-leakage fuel management), their burnup reactivity changes would be reduced over the reference case. Then, a given change in reactor temperature (power) would lead to a longer coastdown period. This combination could improve uranium utilization 1 or 2%.
- radial blanket/core periphery modifications - Larger cores would slightly reduce the effect of both radial blankets and core periphery modifications because in both cases, with fewer neutrons leaking, improved utilization of leakage neutrons becomes less significant. Adverse effects, however, should be very small.
- smaller fuel elements - Larger cores would reduce the impact of smaller fuel elements insofar as they could be loaded into a more compact array, an effect of geometrical scale, but this effect would be very small.
- higher temperature and pressure - Larger cores could impose more engineering constraints on designs for higher temperature and pressure.

Radial Blankets

- frequent refueling - Radial blankets would have a very small negative impact on frequent refueling as they represent assemblies that require occasional replacement. This is extra work during refueling, but not very much on the average. For example, if radial blanket assemblies occupying 10% of core volume, in pieces one fourth the size of core assemblies -- and thus, amounting to 40% of the number of assemblies -- had to be changed every 10 years, the average number of pieces moved per refueling would be increased by only 2% for semi-annual and 4% for annual refueling. There are little if any combined or adverse effects on uranium utilization.

- extended coastdown - Radial blankets would have negligible impact on extended coastdown.
- low power density - Radial blankets would reduce the uranium utilization gains from larger cores, but again, the effects would be negligible.
- core periphery modifications - Radial blankets would reduce the neutron leakage from the core so that the improvement from the use of a Zircaloy core shroud would be reduced to about half of the uranium utilization improvement that could be obtained without a radial blanket, as shown in Appendix B. Furthermore, a radial blanket would compete for space with the addition of reflectors such as BeO or graphite at the core periphery. Radial blankets and reflectors both utilize the space between the reactor core and its support barrel.
- small fuel assemblies - Radial blankets have a favorable effect on the implementation potential of small fuel assemblies. Since radial blanket thickness constraints lead to use of small assemblies,[7] the potential use of small assemblies throughout the core would be enhanced because fuel handling capabilities for small assemblies would exist.
- higher temperature and pressure - Radial blankets might place some additional constraints on higher temperature and pressure design, but these would not be major.

Core Periphery Modifications

The discussion on radial blankets generally applies to other core periphery modifications, such as use of a Zircaloy core shroud, addition of solid radial reflection such as BeO or graphite, or both. The exceptions are:

- frequent refueling - Core periphery modifications would have no effect on frequent refueling.

- radial blanket/small fuel assembly - Use of a Zircaloy core shroud, without an extra reflector, would be compatible with a limited number of blanket assemblies or smaller assemblies.

Smaller Fuel Assemblies

The chief effects of smaller assemblies (quarter area) would arise from the increased number of assemblies to be handled, their capability for mitigating detailed power peaks, and their ability to fit into otherwise unused space.

- frequent refueling - Small assemblies would significantly impact the feasibility of frequent refueling. Using a whole core of small assemblies would multiply the number of fuel moving operations by 4, adding up to 10 days to the refueling operation. Only occasional use of small assemblies to avoid unusual power peaks could be tolerated. If the reactor would be improved by using a small number of quarter-area assemblies to make the desired core outline a more circular shape, this could be tolerated, but if more than about 5% of core area were taken up by such assemblies, the extra number of pieces to be moved during refueling would become significant.

- extended coastdown/higher temperature and pressure - Small assemblies would not affect gains from extended coastdown or higher temperature and pressure.

- low power density - Small assemblies would increase the potential of low power density cores to utilize high power-peaking, low-leakage fuel management. The increase comes from the capability of small assemblies to reduce power peaks that do not contribute to total reactivity.

- radial blankets - Small assemblies would compete with blankets for space at the core periphery. Blanket assemblies are often considered as ways of obtaining extra power and reactivity out of core peripheral space where full fuel assemblies cannot fit. Small fuel assemblies would provide more power and reactivity in these same locations.

- core periphery modifications - Cores with small assemblies would probably be built further out toward the core support barrel, slightly decreasing the effect of core periphery modifications.

Higher Temperature and Pressure

The impacts of higher temperature and pressure are primarily due to the engineering problems of implementing the other concepts under more severe environmental conditions.

- frequent refueling - Higher temperature and pressure have a slight negative impact on rapid refueling, under the assumption that more time for system cooldown and heatup would be required. This, in turn, would impact the desirability of frequent refueling.
- extended coastdown - Higher temperature and pressure could significantly improve coastdown capability since negative power coefficients will increase with temperature and more reactivity will be gained for a small temperature decrease. The PWR more closely approaches the BWR in this case. This concept, however, cannot be taken to extremes since higher temperature and pressure would demand lattices of greater moderator-to-fuel volume ratio to yield comparable moderator-to-fuel atomic densities with existing lattices at operating conditions. These lattices will exhibit less reactivity change than drier lattices as moderator density is increased by lowering its temperature.
- low power density - Higher temperature and pressure could impose significant engineering constraints on the space that could be made available for larger cores.
- core periphery modifications - Similarly, higher temperature and pressure could make it less feasible to install core periphery modifications.
- radial blankets/small fuel assemblies - Higher temperature and pressure would not affect the feasibilities or capabilities of blanket assemblies or small fuel assemblies.

Concept Interaction Matrix

Figures 7.1 and 7.2 are matrix summaries of positive and adverse effects among the concepts. The first column in each matrix lists the concept considered. The row corresponding to each concept is a rating of the effect of that concept on the other concepts, as labeled at the top.

Figure 7.1 is a total judgmental representation, including engineering and operational impacts. Figure 7.2 is a similar representation in which only impacts on uranium utilization improvements are considered.

Conclusions

To use Figures 7.1 and 7.2, the following process is used. First, the results are essentially added across a diagonal so that, for example, Figure 7.1 reduces to the matrix in Figure 7.3. Figure 7.3 represents the reciprocal impacts of two concepts without regard to possible transitive effects. Now, select a single concept, such as frequent refueling. Note that it is slightly synergistic with lower power density and neutral with regard to radial blankets and core peripheral modifications. But lower power density is neutral with extended coastdown, favorable with small fuel assemblies, and unfavorable with radial blankets or core periphery modifications, while radial blankets and core periphery modifications are strongly competitive with each other. Examining the relative magnitudes of these synergisms and competitions, note that only "frequent refueling, lower power density", yields net positive synergism.

Next, try extended coastdown as an initially assumed feature. It is synergistic with respect to higher temperature and pressure, negative with respect to rapid refueling, and neutral to the other features. Higher temperature and pressure is neutral with respect to small fuel assemblies and negative with respect with lower power density. This gives a choice of possibilities: "extended coastdown, high temperature and pressure," or "extended coastdown, low power density, small fuel assembly".

Also note that radial blankets and core periphery modifications are strongly competitive with each other and not synergistic with any of the other concepts.

Concept Considered \ Effect on Concept	Rapid/Frequent Refueling	Extended Coastdown	Lower Power Density Cores	Radial Blankets	Core Periphery Modifications	Small Fuel Assembly	Higher Temperature and Pressure
Rapid/Frequent Refueling		0	√√	0	0	0	x
Extended Coastdown	x		x	0	0	0	0
Lower Power Density Cores	x	√		x	x	0	xx
Radial Blankets	0	0	0		xx	√	x
Core Periphery Modifications	0	0	0	x		0	x
Small Fuel Assembly	xx	0	√	x	x		0
Higher Temperature and Pressure	x	√√	xx	0	xx	0	

FIGURE 7.1. Matrix of PWR Concept Interactions Considering All Effects[a]

(a) A value label is given for the impact of the concept listed to the left on the concept shown at the top. The value labels are:

√√, clearly favorable

√, slightly favorable

0, neutral

x, slightly unfavorable

xx, clearly unfavorable.

Concept Considered \ Effect on Concept	Rapid/Frequent Refueling	Extended Coastdown	Lower Power Density Cores	Radial Blankets	Core Periphery Modifications	Small Fuel Assembly	Higher Temperature and Pressure
Rapid/Frequent Refueling	X	0	+	0	0	0	0
Extended Coastdown	0	X	0	0	0	0	0
Lower Power Density Cores	0	0	X	0	0	0	0
Radial Blankets	0	0	0	X	-	0	0
Core Periphery Modifications	0	0	0	0	X	0	0
Small Fuel Assembly	0	0	+	0	0	X	0
Higher Temperature and Pressure	0	+	0	0	0	0	X

FIGURE 7.2. Matrix of PWR Concept Interactions Considering Uranium Utilization Only[a]

(a) Value labels are:

+, strong synergisms, adding >0.5% to uranium utilization over the effects separately

-, similarly strong antagonisms

0, weak or neutral effects.

Concept ↓ / Concept →	Rapid/Frequent Refueling	Extended Coastdown	Lower Power Density Cores	Radial Blankets	Core Periphery Modifications	Small Fuel Assembly	Higher Temperature and Pressure
Rapid/Frequent Refueling		x	✓	0	0	xx	xx
Extended Coastdown	x		0	0	0	0	✓✓
Lower Power Density Cores	✓	0		x	x	✓	xxxx
Radial Blankets	0	0	x		xxx	0	x
Core Periphery Modifications	0	0	x	xxx		x	xxx
Small Fuel Assembly	xx	0	✓	0	x		0
Higher Temperature and Pressure	xx	✓✓	xxxx	x	xxx	0	

FIGURE 7.3. Intransitive Matrix of PWR Concept Interactions for all Effects[a]

(a) The number of checks (synergism) or crosses (antagonism) in the box that connects two concepts indicates the strength of the mutual synergism/antagonism.

It must again be emphasized that "incompatibilities" among concepts do not necessarily mean that they cannot be used together. In many cases, it means that the gains from using them together will be less than expected from considering their individual gains relative to the base case. The only strong antagonism between concepts, considering only uranium utilization improvements, is the one noted between radial blanket core periphery modifications. The incompatibilities that have been noted between rapid refueling and extended coastdown, or higher temperature and pressure, or between lower power density and higher temperature and pressure, are more like warnings that designs using these features together will be subject to engineering tradeoffs and compromises.

7.2 BWR CONCEPTS

Nonbackfittable concepts considered for improved uranium utilization in BWRs were:

- spectral shift (by extending recirculation flow beyond currently available values)
- rapid/frequent refueling
- higher temperature and pressure
- low power density
- coastdown
- soluble boron (as a shutdown mechanism)
- radial blanket.

Similar to PWRs, the BWR concept interactions matrix is described as the impact that a decision to adopt any given option would have on the desirability of each of the other options.

Spectral Shift

For BWRs, spectral shift consists of the ability to control recirculation flow over a larger range of flow than is permitted by existing standard equipment. This would require more powerful recirculation pumps to permit higher recirculation rates and, perhaps, a dual pump system to permit good flow control at reduced flow. As so defined:

- rapid/frequent refueling - Spectral shift would affect rapid refueling capability.

- higher temperature and pressure - Spectral shift would make higher temperature and pressure operation less attractive, but only because of a feedback effect (see "Higher Temperature and Pressure").

- low power density - Spectral shift might require more downcomer space to achieve a higher recirculation rate, making larger, low power density cores less attractive.

- coastdown - Spectral shift could have a synergistic effect on coastdown. The assumption is that full control capabilities will be deployed at full power and the end of normal burnup life. This means that recirculation would then be at maximum. When reduced power is called for, there is no change from existing coastdown conditions, but the flatter long-term burnup characteristics (improved conversion) induced by spectral shift will slightly extend the time over which coastdown is feasible. Operating somewhat in the opposite direction is the fact that the high recirculation rate during a coastdown period would reduce the influence of subcooled feedwater on the nonboiling core length.

- soluble boron/radial blankets - Using spectral shift for reactivity control would not affect the feasibility of shutdown soluble boron control, nor of radial blankets.

Rapid/Frequent Refueling

Rapid refueling in BWRs requires rapid removal of steam separators and driers before refueling begins. Relative to PWRs, this is compensated by the fact that the bottom driven control rods remain in place.

- spectral shift - Rapid/frequent refueling would reduce the amount of reactivity to be controlled between reloads, and might make it possible to utilize spectral shift control over the entire range of burnup reactivity change. This is a strongly synergistic effect. In existing designs, considerable control of reactivity change over burnup is achieved by use of burnable gadolinium poison in the fuel pins. Averaged over total fuel residence time, 1 to 2% of all the neutrons absorbed by the fuel are absorbed by the gadolinium, the total elimination of which would add 7 to 15% to the energy extractable from fuel of a given initial enrichment. Going from annual (normal) to semi-annual refueling (frequent refueling) would cut down the burnable poison requirement by a factor of 2. The remaining burnup reactivity change might be within the capabilities of spectral shift to control completely. The quantity of burnable poison that has to be used decreases much faster than the amount of reactivity that has to be controlled, since there is less need for long-term reactivity control to make the poison last by incorporating it into strongly self-shielding lumps.

- higher temperature and pressure - Rapid/frequent refueling would add further constraints to the engineering problems of higher temperature and pressure design.
- low power density - As with PWRs, there is synergism with lower power density operation, but the gains compete with those from spectral shift control. This synergism occurs because, relative to the standard refueling case, frequent refueling adds smaller reactivity increments to the core, permitting the reactive fuel to be more advantageously located without violating power peaking constraints. For larger, lower power density cores, these constraints might be relaxed further. However, these constraints could become tighter if spectral shift is the chosen control mode.
- coastdown - Rapid/frequent refueling decreases the advantages of coastdown, but only if coastdown is used with each refueling cycle.
- soluble boron - Soluble boron for shutdown becomes less attractive under frequent refueling conditions. The reason for this is that frequent refueling sufficiently reduces the reactivity increment of refueling, and soluble boron could become superfluous; the mechanical control rods become more likely to provide assured shutdown without any supplement being needed.

- radial blankets - There is no impact, either way, on the attractiveness of radial blankets.

Higher Temperature and Pressure

Higher temperature and pressure impose more severe environmental conditions on all reactor components. The reactivity swing between zero and full power under operating conditions is reduced because of the reduced void fraction at full power (at constant recirculation rate). The reactivity loss at zero power, due to decreased moderator density, can however be compensated for by lattice redesign.

- spectral shift - Higher temperature and pressure make spectral shift flow control more difficult and less capable. The change in recirculation rate required to achieve a given amount of reactivity effect from steam voids is greater.

- rapid/frequent refueling - Higher temperature and pressure make rapid/frequent refueling less attractive from an engineering standpoint, without reference to uranium utilization. The number of heatups and cooldowns that the primary system can experience is a system design parameter that becomes more difficult to ensure the higher the working temperature and pressure are raised. The problem is not so much with high temperature, but that thicker vessels and pipes are required and that heatup and cooldown are subject to larger thermal differences across the vessel.

- low power density - Thermodynamic efficiency gains from higher temperature and pressure BWR designs are greater than in PWRs and thus higher pressure and temperature automatically decrease core size significantly (for constant net electric power). For BWRs a smaller core area increase over existing designs is needed to obtain usable decreases in power density and the balance swings from clearly unfavorable (incompatible) to slightly synergistic.

- coastdown - Higher temperature and pressure could impact coastdown capability. For example, if coastdown is accomplished by varying reactivity in voids, a given percentage power reduction would produce less reactivity effect at higher rather than lower pressures.

- soluble boron - If there are chemical problems in using soluble boron in a BWR environment (a matter of some uncertainty as to whether this is a major problem), higher temperature and pressure might aggravate them.
- radial blankets - Radial blankets are slightly impacted under higher temperature and pressure conditions, because the more severe thermodynamic environment makes long-term exposure more questionable.

Low Power Density

Neutronically, low power density (larger) cores have a very favorable impact on BWRs, because BWRs have high neutron leakage and thus more reactivity to be gained from increased core size. However, increasing radial dimension to decrease neutron leakage is not as effective as initially calculated because a large fraction of that leakage is in an axial direction from high reactivity zones that have large exit void fractions. Moreover, opportunities for improved uranium utilization by low-leakage radial fuel management are less for larger BWRs than for larger PWRs, simply because BWRs are already managed in a lower-leakage configuration: power shaping by steam voids supplements power shaping by fresh fuel location.

- spectral shift - Low power density cores interact negatively with spectral shift flow control since they compete for vessel space. This competition results from the desire to provide more core area to reduce power density, versus increasing downcomer area to provide more water recirculation capability to reduce voids near end of cycle.
- rapid/frequent refueling - As in PWRs, low power density cores would slightly lengthen refueling times, making frequent refueling less attractive.

- higher temperature and pressures - Low power density reacts slightly unfavorably on higher temperature and pressure because larger core sizes increase the engineering problems of such designs.
- coastdown - Low power density cores have less neutron leakage and higher conversion of fertile material to fissile plutonium. As a result, their burnup reactivity changes during a reactor cycle are reduced, leading to a longer coastdown period for a given change in water temperature. However, in a BWR, a large fraction of the neutron leakage is in the axial direction so that increasing core diameter has less effect on neutron leakage than in a PWR. As a result, low-power density is only slightly positive or neutral with regard to the desirability of coastdown.
- soluble boron - Low power density cores are intrinsically manageable with a larger number of refueling batches per core. This decreases core reactivity after reload, and consequently makes the need for soluble boron shutdown control less likely.
- radial blankets - Low power density through increased core space competes with radial blankets for vessel space.

Coastdown

Extended coastdown in BWRs is made feasible by the rather large reduction in core voids, at constant recirculation flow, that power reduction brings about. This reduction in voids has two origins. First, there is a direct effect that, at lower power, boiling begins further up the core and produces fewer voids per channel. Second, any increase in subcooling of the inlet water is reflected in an increased nonboiling length in the core. This can be designed to yield a large end-of-life reactivity/power coefficient and thus an extended coastdown.

- spectral shift - Coastdown capability does not affect spectral shift capability, but since lattice design to optimize either will improve the other, the impacts are correlated.

- rapid/frequent refueling - Coastdown decreases the desirability of frequent refueling, for operational reasons only. There is no antagonistic effect on uranium utilization improvements.
- higher temperature and pressure - Coastdown does not affect higher pressure operation, a case where impacts are not reciprocal.
- low power density/soluble boron/radial blankets - Coastdown capability does not seem to affect the desirability of lower power density, soluble boron, or radial blankets.

Soluble Boron Shutdown

This concept simplifies design by substituting for or augmenting the capability of control rods. Its benefits would come from eliminating some of the rods that might otherwise be needed for shutdown, thus eliminating rod channels in the core and somewhat simplifying mechanical design. Its adoption would neither enhance nor diminish the desirability or feasibility of other options, except in the case of higher temperature and pressure. In that case, a synergism might appear:

- higher temperature and pressure - Higher temperature and pressure designs might exhibit greater reactivity swings between hot operating and cold shutdown conditions, so that soluble poison shutdown could solve a problem.

However, note that the impacts are not reciprocal (see "Higher Temperature and Pressure").

Radial Blankets

The specification of radial blankets as a change from existing practice would not affect the arguments for or against spectral shift flow control, rapid/frequent refueling, coastdown or soluble boron shutdown. The effect of radial blankets on higher temperature and pressure design is also negligible, although the inverse may not be so. Only with regard to the use of larger, lower power density cores would the gains from radial blankets impact another concept. Here, there is definite competition between the two concepts for core space.

Concept Interaction Matrix

Figure 7.4 presents the concept interactions matrix for BWRs.

Conclusions

Although there are pairs of concepts that are antagonistic (e.g., low power density cores/radial blankets and low power density cores/spectral shift flow control), Figure 7.4 does not reduce to favored clusters of concepts. This is probably due to the intricacy of the thermohydraulic/neutronic/mechanical interactions of BWRs as compared with PWRs.

Concept Considered \ Effect on Concept	Spectral Shift Flow Control	Rapid/Frequent Refueling	Higher Temperature and Pressure	Lower Power Density Cores	Extended Coastdown	Soluble Boron Shutdown	Radial Blankets
Spectral Shift Flow Control		0	x	xx	✓	0	0
Rapid/Frequent Refueling	0		x	✓✓	x	x	0
Higher Temperature and Pressure	x	x		✓	x	x	x
Lower Power Density Cores	x	x	x		0	0	xx
Extended Coastdown	✓	x	0	0		0	0
Soluble Boron Shutdown	0	0	✓	0	0		0
Radial Blankets	0	0	0	xx	0	0	

FIGURE 7.4. Matrix of BWR Concept Interactions[a]

(a) A value label is given for the impact of the concept listed to the left on the concept shown at the top. The value labels are:

✓✓, clearly favorable

✓, slightly favorable

0, neutral

x, slightly unfavorable

xx, clearly unfavorable

8.0 ANALYSIS OF COMPOSITE REACTOR SYSTEMS INCORPORATING MULTIPLE DESIGN IMPROVEMENTS

In Chapter 5, nonbackfittable concepts for improving uranium utilization in LWRs were assessed by participating U.S. reactor vendors, and then ranked to identify the combination of concepts with the greatest implementation potential in either PWRs or BWRs. In Chapter 7, the basis for estimating effects on the improvement of uranium utilization of combining concepts into a single advanced reactor system was discussed. Those principles were then used to estimate composite uranium savings for both a PWR and a BWR system incorporating the combined concepts identified as having the greatest implementation potential for each system. These overall evaluations quantify the savings that could be realized from the combined concepts, taking into account the mutual competition or reinforcement of effects that result in uranium utilization improvements.

8.1 COMPOSITE ADVANCED PWR

Individual design enhancements to improve fuel utilization in PWRs, specifically nonbackfittable features (i.e., not feasible for retrofit into existing reactors), were examined. Five nonbackfittable PWR options have been rated. Listed in order of preference, these are:

- rapid/frequent refueling
- low power density core/Zr shroud/radial blanket
- extended coastdown
- small fuel assembly
- higher temperatures and pressures.

The first four of these items received generally favorable ratings. The last was rated as unfavorable, except possibly for incremental variations from existing conditions that represent normal commercial process improvements.

This chapter presents a design synthesis which combines favorably rated options into an advanced composite PWR to determine whether, if the favorable options are pursued parallel with each other, the resulting system would be compatible, and to evaluate the composite uranium utilization improvement. The following sections describe individual advantages and problems of the non-backfittable options included in the composite advanced PWR.

Rapid/Frequent Refueling

A reliable ability to rapidly refuel will react favorably on reactor economics by decreasing the length of scheduled shutdowns and improving reactor availability/capacity factors. The design would not permit compromises in technical safety or operational requirements, and is considered developable, although the minimum refueling time is not precisely predictable. Acceptability of this feature by utilities is considered to be very high.

By itself, rapid refueling does not improve uranium utilization. However, it decreases the economic penalty associated with refueling outages, and thereby makes frequent refueling more attractive. Frequent refueling has a very favorable impact on uranium utilization by reducing the reactivity change between fuelings, thus reducing the average loss of neutrons to absorbing controls. This increases conversion, which leads to an intrinsically slower reactivity loss with burnup, and also increases average core reactivity. These effects can be used to achieve lower initial enrichment, higher burnup, or an optimal combination of both. The result is lowering fuel cycle costs by outweighing the higher cost of purchasing energy (or supplying it from a reserve unit of high operating cost) during the refueling period.

More frequent refueling imposes extra stresses on the reactor system and greater demands on operating personnel. These qualitative disadvantages suggest that the economic advantages of frequent refueling must be firm and significant before it becomes accepted utility practice.

Extended Coastdown

A pressurized water reactor that has reached the end of its full-power reactivity margin can be made to operate somewhat longer at full thermal power by reducing the average temperature of the water coolant-moderator and increasing the volume and pressure of secondary steam entering the turbine.(10) However, the amount of electrical power that can be produced during this end-of-cycle extension becomes progressively reduced because the efficiency of thermal power conversion is reduced. Nevertheless, there are economies to be realized if this end-of-cycle stretchout by feedwater temperature reduction and secondary steam augmentation is carried out before normal coastdown is initiated.

The increased fuel burnup resulting from extended coastdown is "free" as far as fuel cycle costs are concerned. However, the fuel remaining in the core for the next cycle is not as reactive at full power, because it has been burned longer. Evaluating extended coastdown thus requires examination as a repeated phenomenon over complete fuel histories. When this is done, extended coastdown still results in some "free" burnup, an improvement in uranium utilization and fuel cycle cost beyond normal coastdown.

A design for utilization of extended coastdown would combine unfavorably with frequent refueling. As shown on page 7.2, there is a capacity factor loss of about 0.1% for extended coastdown. Since the maximum theoretical capacity factor is reduced by about 3.1% by semi-annual refueling, accommodation of this additional capacity factor loss makes the combination of extended coastdown and frequent refueling concepts slightly unfavorable. This concern is primarily from an economic standpoint, because the uranium utilization improvements combine without penalty. Thus, coastdown and frequent refueling must be planned together if they are to be used together.

Low Power Density Cores

Reactor pressure vessels can be increased in size, at increased capital cost, to permit more fuel to be loaded for a given power rating, reducing average power density. Such a core would have lower neutron leakage, lower flux, and thus less reactivity to xenon absorption, and consequently increased conversion, reactivity, and uranium utilization. The fuel cycle cost would be increased by the carrying charges on the larger fuel inventory in the core, but the economic advantages that accrue from improved uranium utilization outweigh the carrying charge disadvantage. Of course, not only the reactor vessel but many of the internal reactor structures and the containment vessel would increase in price and size.

All of these factors, taken together, project a small economic advantage to the lower power density system, which might not be enough in itself to outweigh certain other perceived disadvantages (in reactor construction and erection) of having larger and heavier reactor components. However, there are synergistic effects on the other options that are favorable. For the same refueling interval, for example, a larger number of fuel batches are used, which has the same favorable effect as going to more frequent refueling in a higher power density core. Lower power density cores can also accommodate higher power-peaking factors, which allows more flexibility in locating fresh fuel in the core, while still providing increased margin in several safety limits. This additional core volume and the capability for accommodating higher peaking factors also permit an additional cycle of exposure to be accumulated on high burnup fuel loaded at the core periphery that would otherwise be discharged from the core as spent fuel (see "Small Fuel Assembly" below.

<u>Small Fuel Assembly</u>

PWR fuel assemblies are large, containing nearly 300 fuel rods. Improvements in core reactivity, power-peaking factor, or both could be realized by designing each quarter of a conventional assembly as a small assembly that could be located independently. These improvements would result from the ability to reduce heterogeneities in the core and to take advantage of reduced burnup differences between fuel rods within a smaller assembly.

These possibilities, along with other reactor loading modifications such as using reflectors and fertile blankets at the core periphery, are evaluated as being marginally favorable for improved uranium utilization and fuel cycle economics, and thus low on a developmental priority list. An exception is made for the possibility of using quartered spent fuel assemblies to fill a half-assembly thickness at the periphery of the reactor core and at locations where the geometrical mismatch between the circular core support barrel and the conventional sized square fuel assemblies leaves some large water gaps. As indicated on page 7.6, the use of a Zircaloy core shroud, in place of the conventional stainless steel shroud, would be compatible with these blanket assemblies and small fuel assemblies. This combination of concepts is judged to be worth pursuing; however, the composite design should be optimized to maximize benefits from these concepts which compete for space in the reactor vessel.

Approach

After noting the effects in previous sections that should be considered in combining favorable nonbackfittable options, a composite PWR design was formulated by PNL based on judgments expressed by the industrial participants in the assessment of the nonbackfittable concepts individually.[9,10] This preliminary composite design was analyzed; some modifications were made to the preliminary composite design that would seem to improve it, and then the modified composite design was analyzed before drawing conclusions.

In developing this design, extended use of certain low-neutron-leakage fuel management was also considered. The lower power density core, which is a primary nonbackfittable feature, permits some increase in localized core neutron flux and power peaking. This allows fuel management schemes in which fresh fuel is loaded closer to the core center than is now the case. Such a fuel management scheme has increased reactivity and therefore increased potential for uranium savings.

Preliminary Composite Design Concept

As a basis for the composite advanced PWR the following design concept was adopted:

1. 3800 MW(t) reactor power output
2. 30% increase in cross-sectional area for reactor core and blanket
3. 6-month refueling intervals using an 11-batch core with minimum shuffling and low-neutron-leakage fuel management
4. "overburn" of spent fuel by placing it in positions at the core periphery available by using quarter-assemblies
5. extended coastdown at end of cycle using feedwater temperature reduction
6. substitution of Zircaloy for stainless steel in the core shroud.

The rationale for these choices is as follows. It is generally assumed that power increase would result in unit capital cost economies, but some studies indicate that there may be diseconomies in increasing scale.[11,12] Regardless, it is likely that the curve of the scaling law will be relatively flat in the vicinity of present practice.[12] Under these conditions, there are no good reasons for departing from present practice, since such departures have the potential for increasing first-of-a-kind design costs. Thermal power could always be reduced from the current limit of 3800 MW while using the largest vessels now being made, to lower core power density within the existing envelope. However, since large core volume has a favorable impact on uranium utilization because of reduced neutron leakage, it is preferrable to retain the 3800 MW thermal power limit and lower power density by increasing the core volume.

In determining the target figure for core power density reduction, the principal modification considered has been to increase core radius, while retaining current fuel assembly heights. This modification increases the number of assemblies in proportion to the inverse of power density, but retains most of the existing thermohydraulic limits on assembly power. It has been estimated[10] that reactor vessel internal diameter could be increased by 5.35% before current shop capabilities become limiting, but that the extra space provided by this change could accommodate 22% more complete fuel assemblies. This has been increased to 30% in consideration of the capability of filling the periphery with quartered assemblies and (if necessary) changing assembly dimensions to improve the "fit" of the square assemblies to the circular vessel outline.

The selection of 6-month refueling interval was principally dictated by the general pattern of U.S. utilities to experience minimum demands twice a year, in the spring and fall, regardless of whether they are winter-peaking or summer-peaking utilities. Thus, the refueling interval should be a multiple of 6 months to minimize the need for purchasing replacement-power. Since present practice is generally to refuel annually, any interval other than 6 months would not constitute feasible "frequent refueling." The fuel assembly shuffling approach is intended to minimize the amount of fuel handled at each refueling, limiting it to about one-third of the elements each time. An alternative not considered in the composite analysis might be to alternate 6-month and 12-month refuelings, if a fuel management scheme with small reactivity loss over the 12-month period can be found.

The "overburn" of fuel involves the irradiation of otherwise spent fuel for one additional reactor cycle at the periphery of the core. In addition, the core baffle plates (sometimes referred to as the core liner or shroud) and the core former plates (which connect the baffles to the core barrel) were designed of Zircaloy-4 instead of 304 stainless steel.[7]

Implementation of extended coastdown in the advanced composite PWR design was considered feasible because end-of-cycle stretchout by feedwater temperature reduction continues reactor operation at full thermal power, with only slight reduction of electrical power output, before normal power coastdown is initiated. The loss in capacity factor due to frequent refueling is essentially unchanged by extending coastdown, as indicated in Section 7.1. Furthermore, the capacity factor loss due to normal coastdown occurs at end-of-cycle in the spring and fall of the year when the need for purchasing replacement power is minimized.

Core Geometry

The composite advanced PWR design is based on the Combustion Engineering System 80™. This system is used for purely illustrative purposes; there is no reason why similar results cannot be obtained from other manufacturers' designs.

The present pressure vessel ID of 182.25 in. (4.629 m) is increased to a stated shop limit of 196 in. (4.978 m). The existing cross-sectional area of space between pressure vessel and core support barrel (2784 in.^{2} or 1.80 m^2) is retained, as is the thickness of the barrel (2.625 in. or 0.067 m). The result is a barrel of 177.5 in. (4.509 m) OD and 172.25 in. (4.375 m) ID. A further space of 6.12 cm is specified between the inside of the barrel and the closest part of the core, to accommodate the core shroud, barrel cooling, and necessary clearances. Thus, all fuel elements must be put in locations such that their furthest point from core center is less than 212.6 cm.

Using the existing square pitch of fuel assemblies, 20.78 cm, a lattice of 292 or 293 full assemblies can be loaded into the core, depending on whether the core center is an assembly corner or assembly center. The "center-a-corner" arrangement can accommodate 64 assemblies of quarter area and the central assembly arrangement, 60 of them. Thus, by the use of quarter-area assemblies on the periphery, a core equivalent to 308 full assemblies can be loaded. This is 27.8% more assemblies than the standard System 80™ core. To achieve a 30% increase in core volume, the core height is increased from 381 cm to 387.6 cm.

Such a system could be managed on a 6-month reloading cycle, charging 28 fresh fuel assemblies each cycle at steady state. (Of these, 16 would have to be capable of division into quarters.) This would amount to an 11 batch core.

A Reference Calculation

Before the details of the clearances were thoroughly worked out, a 13 batch, 312 assembly core was sketched for illustration, and critical calculations were performed. The core consisted of 300 full assemblies and 48 quarter assemblies, as diagramed in Figure 8.1.

There are 4 major zones, labeled A, B, C, and D. A blanket zone consisting of 3 full assemblies and 12 quarter assemblies per quadrant, is indicated by the label X. Within each major zone there are 3 sub-zones labeled 1, 2, and 3. At the first reload, only the 1's are moved, then the 2's at the next reload, then the 3's and so on. The successive movement of only the "1", "2", or "3" labeled fuel is what is meant by minimum shuffling reloading because two-thirds of the fuel in the core remains in place during each refueling. Thus, although there are $300 + \frac{48}{4} = 312$ assemblies in the core, at any given refueling only $\frac{288}{3} + 24 = 120$ assemblies have to be moved: the assembly of a single number plus the X's. Even noting that all 48 quarter assemblies must be moved, the number of pieces that must be replaced during each refueling is 156. The 1, 2, and 3 numbered assembly positions have been selected to be of approximately equal importance within each major zone. Table 8.1 gives the equivalent radial dimensions of the 312 assembly reactor. Table 8.2 gives the

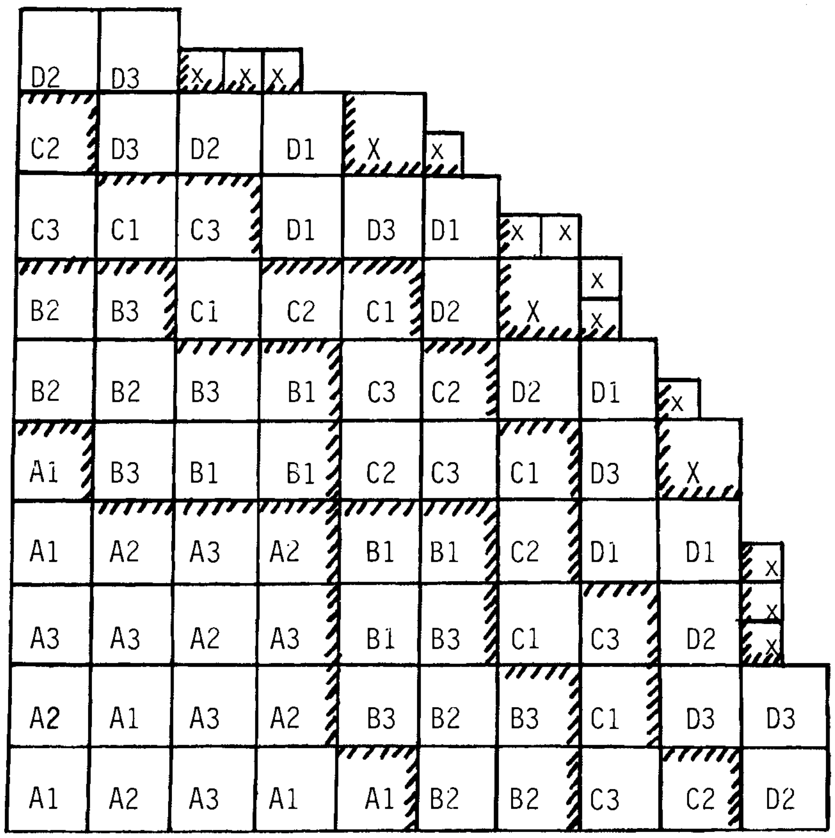


FIGURE 8.1. Zone Arrangement of a Core Quadrant[a]

(a) A,B,C,D, represent radial zones.

1,2,3 are parts of the various radial zones that are moved on refueling (i.e., all 1's, 2's or 3's), but only one of these part numbers is moved during a given refueling outage.

X are full peripheral "spent" assemblies; x are quarter peripheral "spent" assemblies

Each large square is 20.78 cm on a side and represents one full fuel assembly. Each small square is a quarter assembly.

TABLE 8.1. Radial Dimensions and Compositions of Equivalent Advanced Reactor

Zone	Inner Radius (cm)	Outer Radius (cm)	Composition
A	0	99.50	Core fuel
B	99.50	140.69	Core fuel
C	140.69	172.31	Core fuel
D	172.31	198.97	Core fuel
X	198.97	207.09	Core fuel
R	207.09	∞	Water

TABLE 8.2. Radial Dimensions and Compositions of Equivalent Comparison Reactor[a]

Zone	Inner Radius (cm)	Outer Radius (cm)	Composition
c	0	181.61	Core fuel
ws	181.61	206.06	30% steel, 70% H_2O[b]
w	206.06	231.46	Water
s	231.46	∞	Steel

(a) Source: CEND-380, page 5-4 (Reference 3).
(b) Core shroud, inner water gap and core support barrel.

radial equivalent dimensions of the comparison System 80™ C-E reactor. For both reactor designs, the core height is 381 cm of active fuel length.

Preliminary Reactivity Estimates

To get some idea of the purely configurational improvements in reactor neutron multiplication factor (k) some estimates of reflector savings and core migration area were made. From these estimates, fuel savings attributed to reactor core enlargment and use of a radial reflector can be approximated. Estimated parameters are given below:

Axial reflector savings (comparison and modified reactors): 10 cm at both top and bottom.

Radial reflector savings (comparison reactor): 8 cm

Radial reflector savings (modified reactor): 12 cm

Migration area (both reactors): 55 cm^2

These parameters lead to the following geometric bucklings.

$$\text{Comparison reactor:}\quad B_g^2 = \left(\frac{\pi}{401}\right)^2 + \left(\frac{j_o}{189.61}\right)^2 = .000222\ cm^{-2}$$

$$\text{Modified reactor:}\quad B_g^2 = \left(\frac{\pi}{401}\right)^2 + \left(\frac{j_o}{219.09}\right)^2 = .000182\ cm^{-2}$$

The difference in geometric buckling is $\delta B_g^2 = 40 \times 10^{-6}\ cm^{-2}$, leading to a reactivity difference in k_∞ (crit) of $55(40 \times 10^{-6})$ or 0.0022. Thus, a uranium utilization improvement of about 7 x 0.0022 or about 0.015 (1.5%) can be expected as the direct result of reactor enlargment. Any improvements above this must be attributed to improved fuel management that the use of the larger, modified reactor makes possible.

Fuel Management

For this preliminary work, fuel has been associated with radial zones. Within each zone (A, B, C, or D), the end of the (6-month) cycle fuel burnup is approximated by assuming that all the fuel in the zone has a burnup equal to an assumed initial burnup of fuel loaded into the zone plus two thirds (2/3) of the mean burnup accrued by the fuel while in the zone. The reasoning is that if the neutron flux in the zone is constant, at the end of any cycle 1/3 of the fuel will have 6 month's burnup in the zone (1/3 of its residence time), and the last third, which is the one to be shuffled, has had the complete (3/3) burnup of fuel in the zone. Zone X, however, is reloaded every six months and at the end of cycle has a corresponding burnup.

Criticality for this system was calculated by selecting two target burn-ups (50 and 65 MWd/kgU) and searching for initial enrichments that would permit these targets to be reached. First, results were obtained from LEOPARD[13] calculations reduced to one-group parameters from individual pin-cell burnup calculations for three different initial ^{235}U enrichments (2, 3, and 4%).

Then the calculated results for neutron multiplication factor, k_∞, and migration area, M^2 (equal to $\frac{k_\infty - 1}{B}$, where B^2 is the material buckling), were fitted to quartic curves as a function of burnup. The fitting was generally precise to about 0.1% in k_∞, and smoother than the original data. Because the data were conveniently available, a pin cell, representative of the Westinghouse 17 x 17 bundle design for the TROJAN reactor, was taken as the basis for these calculations. Geometric data are given in Table 8.3a.

A radial zone loading sequence (e.g., CBADX) was selected for low-neutron-leakage fuel management, and an initial estimate of the neutron flux distribution was made. The notation of the loading scheme selected means that fresh fuel is loaded into region C where it resides for 18 months, then moved in region B for 18 months, 18 months in A, 18 months in D, and 6 months in X. Each radial zone is characterized by its average, end-of-(6 month) cycle, burnup. This burnup is calculated from the initial estimated neutron flux distribution. Then, for a given estimated initial ^{235}U enrichment, k_∞ and M^2 were estimated for each radial zone by interpolating/extrapolating from the fitted curves. Using these data, a 1-dimensional (cylindrical), 1-group

TABLE 8.3. Lattice Dimensions

(a) Pin Cell Used in Reference Lattice

Cell Pitch: 0.4960 in. (1.260 cm)
Outer Fuel Radius: 0.16125 in. (0.410 cm)
Inner Clad Radius: 0.1645 in. (0.418 cm)
Outer Clad Radius: 0.187 in. (0.475 cm)
H_2O/Fuel volume ratio: 1.66

(b) Pin Cell of Improved Lattice

Cell Pitch: 0.4960 in. (1.260 cm)
Outer Fuel Radius: 0.15425 in. (0.392 cm)
Inner Clad Radius: 0.1575 in. (0.45440 cm)
Outer Clad Radius: 0.1800 in. (0.457 cm)
H_2O/Fuel volume ratio: 1.93

critical calculation was performed to obtain an improved neutron flux distribution and a value of the neutron multiplication factor for the reactor (k_{eff}). The k_{eff} value was used to improve the initial enrichment estimate and this, together with the improved neutron flux distribution, were used to obtain the final results by iteration of the process just described. Convergence to critical (k_{eff} = 1) determines the required ^{235}U enrichment value for fresh fuel and a set of radial zone neutron flux and burnup values.

The process just described produces results which only approximate more detailed calculations. A major defect of the simplified model is that it assumes 1-group neutron flux to be strictly proportional to power density, an approximation that can vary about (±10%) over the range of burnup considered. It also assumes that no neutron absorber material such as residual boron or gadolinium exists in the reactor at the end of cycle. For calibration, therefore, the high burnup, annual cycle, 5-batch OUT-IN equilibrium loading of the System 80™ C-E reactor[a] was calculated using the dimensions and compositions listed in Table 8.2. The calibration case, for which an initial enrichment of 4.3% was reported as necessary for steady-state operation, was computed to have k_{eff} of 1.026, and this value k_{eff} was used as the normalized "critical" value for the composite advanced PWR fuel management analysis previously described.

Four cases were analyzed: a CBADX radial zone (low-leakage) fuel management pattern, a DCBAX (high leakage) pattern, and the two cases from CEND-380.[3] Both the CBADX and DCBAX cases were calculated for 50 MWD/T and 65 MWD/T discharge burnups. The flux profiles all showed a peak in the freshly loaded region (respectively C and D). These cases, which were used for preliminary analysis, are summarized in Table 8.4.

In the course of performing these calculations, several values of k_{eff} and initial enrichment were calculated for many different fuel loadings. This enabled a validation of a semi-empirical formula, which held for all the cases under consideration:

(a) Described in Reference 3, pp 5-16 to 5-29.

$$\frac{E_1}{E_o} = \frac{1}{1.06\ E_o \left(\frac{50}{BU}\right)\left(\frac{k_o}{k_1}\right) + \frac{k_o}{k_1}} \qquad (1)$$

In Equation (1), E_o is an enrichment for which k_{eff} takes the value k_o, and E_1 is the enrichment for which k_{eff} will take the value k_1; BU is the burnup of the system, held constant for the two cases, in MWd/kg of heavy metal loaded. Equation (1) is a useful formula for interpolation purposes.

Burnup Adjustment

The results of Table 8.4 are for burnups that span the target value, which is 57.5 MWd/kg.[1] To adjust to that burnup, the uranium saving for each loading pattern was averaged, since this showed only small variation

TABLE 8.4. Initial Enrichments, Fuel Savings and Peaking Factors of Calculated Cases

Loading Pattern[a]	Final Burnup	Initial Enrichment	% U_3O_8 Saving[b]	Peaking Factor
DCBAX	50 MWd/kg	4.05%	5.05%	1.66
	65	5.04%	7.93%	1.91
CBADX	50	3.81%	11.04%	1.73
	65	4.80%	12.54%	1.81

(a) Indicates fuel progression through the reactor on reloads. CBADX indicates that fresh fuel is loaded into region C, moved to B after 18 months, to region A after 36 months, to region D after 54 months, to region X after 72 months and discharged after 78 months residence in the core.
(b) The comparison case is the CEND "improved" backfittable loading, calculated on an OUT-IN loading pattern. The uranium savings was calculated from the initial enrichment and assumed 0.25% enrichment tails assay from the formula:

$$\%\ \text{Fuel Savings} = 100\left(1 - \frac{50.6}{BU}\left[\frac{E-0.25}{4.05}\right]\right)$$

between 50 and 65 MWd/kg, and the ^{235}U enrichment resulted from the equation for uranium savings at a burnup of 57.5 MWd/kg. The results are:

- For the high neutron leakage fuel management loading pattern (DCBAX) the interpolated ^{235}U enrichment to provide equilibrium feed for 57.5 MWd/kg burnup is 4.55%.
- For the low neutron leakage fuel management loading pattern (CBADX) the interpolated feed enrichment is 4.31% for 57.5 MWd/kg.

A similar ^{235}U enrichment adjustment for achieving 57.5 MWd/kg fuel burnup was made to the cases reported in CEND-380[3] for high and low neutron leakage fuel management schemes. For the reference OUT-IN fuel loading scheme, the enrichment to achieve a 50.6 MWd/kg burnup loading was 4.30% ^{235}U, while for a lower neutron leakage loading scheme, it took an enrichment of 4.44% ^{235}U and a burnup of 53.7 MWd/kg to achieve a 2.57% uranium fuel saving over the OUT-IN case. For this adjustment, the result of the calculation reported in Table 8.4, that a 15 MWd/kg increase in burnup corresponded to a 0.99% increase in ^{235}U enrichment, was used for both loadings. This leads to the following adjusted enrichments for both CEND-380 cases:

- At 57.5 MWd/kg burnup, the equilibrium feed enrichment for the OUT-IN case is 4.76%.
- For the same fuel burnup the feed enrichment for the low neutron leakage case is 4.64%.

These results are listed in the top rows of Table 8.5, along with the uranium savings calculated relative to the 50.6 MWd/kg, OUT-IN loading that was used as the reference case.

<u>Adjustment to 308 Assembly Core with Longer Fuel</u>

The results reported in Table 8.4 and the top row of boxes in Table 8.5 are for a 312 assembly core with fuel assembly height of 381 cm. These results were adjusted to a 308 assembly core with fuel assembly height of 387.6 cm selected for the composite advanced PWR design. Taking 8-cm axial reflector savings top and bottom and 12-cm radial reflector savings, the geometric buckling of the calculated (312 assembly) core is 0.000182 cm^{-2}, whereas

TABLE 8.5. Comparative and Adjusted Data on Uranium Savings

	High Leakage Large Reactor[a]	Lower Leakage Large Reactor[a]	OUT-IN CEND-380	Low Leakage CEND-380
Enrichment for 57.5 MWd/kg	4.55%	4.31%	4.76%	4.64%
Uranium Savings[b]	6.57%	11.78%	2.00%	4.61%
Adjust Large Reactor to 308 Assembly Core with 6.6 cm Longer Fuel				
Enrichment	4.545%	4.305%	No Change	No Change
Uranium Savings	6.68%	11.89%		
Adjust to Comparable Effects of Low-Leakage Fuel Management				
Enrichment	4.50%	4.35%	No Change	No Change
Uranium Savings	7.55%	11.02%		
Adjust to Reduced Xe and ^{238}U Resonance Absorption in Large Core				
Enrichment	4.41%	4.26%	No Change	No Change
Uranium Savings	9.61%	12.87%		
Adjust to Lattice Improvements	(c)	(c)	(d)	(d)
Enrichment	4.05%	3.93%	4.64%	4.52%
Uranium Savings	17.4%	20.0%	4.6%	7.2%
Adjust to Zircaloy Core Shroud on Large Core				
Enrichment	4.00%	3.88%	4.64%	4.52%
Uranium Savings	18.4%	21.0%	4.6%	7.2%
Adjust to Coastdown in CEND-380 Cores and Extended Coastdown in Large Cores				
Enrichment	3.72%	3.60%	4.41%	4.29%
Uranium Savings	24.3%	26.9%	9.4%	12.0%

Enrichment of Composite Improved PWR with all Backfittable improvements (for comparison to adjusted low-leakage CEND-380 core) 4.31%[e]

(a) The large reactor results are consistent with those listed in Table 8.4.
(b) All uranium savings are in comparison with the 50.6 MWd/kg burnup, OUT-IN system described in CEND-380.
(c) Table 8.3b.
(d) CEND-380, p. 5-85, Table 5.4-1.
(e) CEND-380, p. 6-36, Table 6.3-3 (Reference 3).

the buckling for the 308 assembly core is 0.000178 cm^{-2}. The migration area of the lattice is about 55 cm^2, so that the 308 assembly core, with longer fuel, is 0.022% more reactive. This corresponds to a required enrichment that is about (Equation (1), E_0 = 4.3, BU = 57.5, k_0 = 1.00022, K_1 = 1) a factor 0.9989 times the previously calculated value. This adjustment is listed in the second row of Table 8.5.

Adjustment for Low-Leakage Loading

The CEND-380 comparison of OUT-IN and low-leakage loadings shows an effect almost half that of the two cases calculated for the large reactor as high leakage and low leakage. There is an inherent capability for a low-leakage loading to be more reactive than a uniform one by an amount on the order of the reactivity tied up in radial leakage. For the CEND-380 size core, this could make a difference of 0.88% in k. For the 308 assembly core, the difference would be less: 0.67%. Yet, the smaller core shows less difference than the larger one.

When examining the two calculations, note that the cases calculated here for the large core were characterized by large radial neutron flux peaks, which were ignored. Therefore, the large core cases contain a bias in favor of low-leakage fuel management that could be subject to substantial correction. To make this correction, the 308 assembly core was assumed to have a uranium savings change 0.9% larger than that of the CEND-380 core, in charging from high leakage to low-leakage fuel management. (This change in uranium savings, would be larger for the smaller CEND-380 cores, as just noted, except for the fact that the lower power density core tolerates greater neutron flux peaking). The 308 assembly, high-leakage uranium savings were adjusted upward; the 308 assembly, low-leakage uranium savings were adjusted downward. The uranium savings were adjusted by the same amount to give a 0.9% larger difference in uranium savings between OUT-IN (or high leakage) and low-leakage fuel management for the large core than the CEND-380 cores, because it could tolerate larger neutron flux peaks. The results are listed in the third row of Table 8.5.

Adjustment for Reduced Xe Absorption and Doppler Effect in Large Core

The effects of reduced power density on neutron absorption in Xe and capture of neutrons in ^{238}U resonances (the latter effect being the result of reduced Doppler effect) were calculated and reported in Section 4.2 of Reference 10. For a 30% increase in core volume, reduced neutron capture in xenon resulted in a 0.21% increase in k_∞. When power density, and as a result, fuel temperatures are reduced, a 0.20% increase in k_∞ was calculated as a result of reduced Doppler effect. These two effects combine to yield an increase of 0.41% in reactivity, decreasing ^{235}U enrichment and increasing uranium savings for the large core as shown in the fourth row of Table 8.5.

Adjustment for Improved Lattice

The reported calculations are for the Westinghouse 17 x 17 lattice defined in Table 8.3a, and those of C-E are for the standard System 80^{TM} lattice. The most important operative difference from a physics point of view is that the Westinghouse lattice has a (cold) water-to-fuel volume ratio of 1.66 and the C-E lattice has a ratio of 2.04.

An improved lattice on the Westinghouse pitch is described in Table 8.3b. It is "wetter" with a (cold) water-to-fuel volume ratio of 1.93, and should be more comparable to the C-E lattice.

However, it must be noted that C-E also explored a modified lattice (CEND-380, Table 5.4-1, page 5-85), with a water-to-fuel ratio of 2.25, for which a uranium utilization improvement of 2.6% was obtained.

The improved lattice (Table 8.3b) results in a calculated reactivity increase over that of the standard lattice (Table 8.3a) of 1.73%, that was nearly constant with burnup.

The adjustments for these lattice improvements are given in the fifth row of Table 8.5. In this row, uranium savings have been rounded off to more nearly reflect the reduced confidence in the calculation precisions.

Adjustment for Zircaloy Shroud

A calculational study was made to determine the uranium savings that could be achieved on a large PWR core by substitution of Zircaloy-4 for stainless steel in the core baffles and formers between the reactor core and barrel. The results of this study are summarized in Appendix B. Uranium savings attributable to this Zircaloy shroud are about 1.6% when no radial blanket is present and 0.7% when a natural UO_2 blanket of quarter assemblies surrounds the core. The uranium savings obtained from the Zircaloy shroud when using high burnup fuel loaded at the core periphery that would otherwise be discharged from the core as spent fuel, as is the case for the composite advanced PWR design selected, was estimated to be about 1%. This reduces the enrichment requirement for fresh fuel by about 0.05 wt% ^{235}U. These adjustments for the substitution of Zircaloy in the core shroud are given in the sixth row of Table 8.5.

Adjustment for Coastdown

The effects of end-of-cycle stretchout by either normal power coastdown or extended coastdown by feedwater temperature reduction (while maintaining full thermal power) followed by power coastdown were analyzed and reported in Section 3.3 of Reference 10. The uranium savings calculated for coastdown to 75% of rated net electrical output for the plant are 4.8% and 5.9% for normal and extended coastdown, respectively. This reduces the enrichment requirement for fresh fuel by about 0.23 wt% ^{235}U and 0.28 wt% ^{235}U, respectively. These adjustments for coastdown are given in the seventh row of Table 8.5. The adjustment made to the large cores was for extended coastdown; and the adjustment made to the CEND-380 cores was for normal coastdown.

Discussion

The data of Table 8.5 indicate that a larger core with extended fuel exposure at its periphery, 6-month refueling, Zircaloy core shroud, and feedwater temperature reduction at end-of-cycle can yield an improvement of about 15% in uranium utilization (over the composite, improved backfittable design described in CEND-380) when a low neutron leakage, fuel management scheme is used. The principal sources of this 15% improvement include about 2% due to decreased xenon and ^{238}U Doppler absorption, 1% due to reflector savings from the Zircaloy core shroud, 1% due to reactivity recovered at end-of-cycle by reduction of feedwater temperature, leaving 11% to be attributed to a combination of a 6-month refueling interval (frequent refueling); better ability to utilize excess reactivity, even in a high-leakage fuel management mode; and use of a "spent fuel" blanket.

The separate effect of implementing frequent refueling was estimated in the reference case. This is listed in Table 5.2 as a 7 to 8% improvement in uranium utilization. Thus, a 3 to 3.5% improvement can be attributed to the more flexible fuel management scheme made available by a larger core. Use of "spent fuel" as a blanket is considered to be an intrinsic part of such a fuel management scheme.

The calculations on which these comparisons have been based are simplified, as noted, and yield neutron flux distributions with rather high peaking factors. Before using them, a study was performed on the effect of such peaking factors on system reactivity in frequently refueled reactors. The conclusion of that study was that, for large reactors, peaking factor could be suppressed with only an extremely small reactivity loss. The apparent excess importance of neutron flux peaks in fresh fuel is almost exactly compensated by the greater rate of reactivity loss in that fuel at high neutron flux, and unless cores with quite high neutron leakage are involved, the penalty from neutron flux flattening is essentially zero.

Summary

The composite advanced PWR design was calculated to improve the uranium utilization by about 15% beyond the potential improvements from backfittable concepts, as shown in the final row of Table 8.5. The enrichment requirement for fresh fuel for the composite advanced PWR is reduced to 3.6 wt% ^{235}U from about 4.3% required for the composite improved PWR which includes all the backfittable concepts. The target burnup of the spent fuel was 57.5 MWd/kg, the same as that used for the composite improved PWR with backfittable improvements. The nonbackfittable features employed in the composite advanced PWR design include:

- 30% larger core volume with 23% lower power density than the composite improved PWR

- rapid refueling equipment and plant improvements to accommodate 6-month refueling intervals
- limited use of quartered assemblies at the core periphery to accommodate overburn of spent-fuel assemblies for one additional cycle
- substitution of Zircaloy-4 for stainless steel in the core baffles and formers surrounding the core
- extended coastdown, at the end of each cycle, using feedwater temperature reduction while maintaining full thermal power prior to normal power coastdown.

The sources of the 15% improvement in uranium utilization over the composite improved PWR's backfittable core that is annually refueled and optimized for the same burnup are approximately:

- 1% from reduced neutron absorption in xenon
- 1% from reduced Doppler effect in ^{238}U resonances
- 8% from reduced control absorption with semi-annual refueling
- 2.5 to 3.5% from improved fuel management flexibility
- 1% from reflector savings with Zircaloy core shroud
- 1% from reactivity recovered during feedwater temperature reduction at end-of-cycle.

These numbers are very little different from those obtainable by separate effect calculation, indicating that synergism of effects is not very important with regard to uranium utilization, for the nonbackfittable concepts included in the composite advanced PWR design.

8.2 COMPOSITE ADVANCED BWR

Individual nonbackfittable concepts for improving uranium utilization in BWRs were assessed in Chapter 5.0. Four of the seven BWR concepts assessed were ranked as having the greatest potential for implementation on a once-through fuel cycle. These concepts are:

- spectral shift with flow control
- frequent refueling (semi-annual)
- higher temperature and pressure
- larger core with lower power density.

Other concepts (extended coastdown by feedwater temperature reduction, and variation or change of existing radial blankets) were considered to have marginal potential for implementation in BWRs, as shown in Table 5.4. The use of soluble boron poison for shutdown control would probably not be necessary when 6-month cycles are used.

This section illustrates how these desirable nonbackfittable concepts might be combined in a single advanced BWR system and estimates expected uranium utilization improvements. However, precise design calculations, which require detailed consideration of neutronic-thermohydraulic coupling, are beyond the scope of this study. Therefore, a judgmental design concept is presented and analyzed by separate-effect estimates of the nonbackfittable concepts' impact on uranium utilization.

Spectral Shift Control

The steady-state steam void content of a BWR core can be changed by varying the mass flow rate of water through the core, or (describing the same conditions differently) by varying the recirculation rate. High recirculation rate is associated with less subcooling in the channel entrance, a longer boiling length, and a low exit steam quality. A low recirculation rate has high inlet subcooling, a shorter boiling length, and a higher exit steam quality. For constant channel power and exit steam mass flow, this variation in recirculation rate would be a small effect except for steam slip: the linear velocity of the steam exceeds that of the water, more so at higher flow rates, and the net result is that the core has fewer voids at high flow.

High steam voids at low flow permit more neutrons to be absorbed by ^{238}U resonances and give higher conversion of ^{238}U to plutonium. This additional plutonium is produced by neutrons that would otherwise have been lost to neutron absorbers, and makes additional reactivity available at the end of core life.

A core flow control system permitting 40 to 150% of rated recirculation flow to accomplish spectral shift could eliminate about half of the burnable gadolinia poison now used for burnup compensation. This could reduce initial reactivity requirements by an amount equivalent to about 2% in neutron multiplication factor, K_{eff}. For example, this would correspond to a 6.7% improvement in uranium utilization, when the initial enrichment is about 4%.

The flow control system specified would probably be appreciably more sophisticated than the existing downcomer jet pumps and, possibly, a duplex system of larger and smaller pumps would be needed.

Rapid/Frequent Refueling

Frequent (semi-annual) refueling reduces the required reactivity that has to be added at the beginning of cycle, and thus reduces the amount of compensating control absorption of neutrons. Initial estimates of uranium savings from frequent refueling are about 6%. The technical feasibility of rapid refueling, which is needed to make frequent refueling operationally attractive, is judged to be about as high for BWRs as for PWRs, although the detailed design problems are different.

Higher Temperature and Pressure

The existing BWR system pressure, 1050 psia, corresponds to a saturated steam temperature of 551^oF. It is considered feasible to raise the pressure to about 1300 psia, with a corresponding temperature of 577^oF. Steam cycle efficiency would increase from an existing 33.4% to about 35.9%, which is a relative improvement of about 8%. This improvement would be directly reflected in uranium savings. Use of Zr cladding under these conditions has been demonstrated in the Big Rock Point BWR for more than 15 years.

Lower Power Density

A larger core would require lower power density and, for the same specific flow rate as now exists, have fewer voids and more reactivity. A larger core would also have less neutron leakage. Even at higher temperatures and pressures, core cross-sectional area can be increased by about 14% to fit into a slightly larger pressure vessel. BWRs presently utilize natural uranium blankets in initial cores, which are replaced with high exposure fuel. To effectively utilize the larger core, additional high exposure fuel would be loaded at the periphery.

Reference BWR Reactor

Column A in Table 8.6 presents characteristics of a basic, large BWR system. The fresh fuel enrichment requirement has been extrapolated from current design to be consistent with the target burnup of 57.5 MWd/kg. This extrapolation is equivalent to increasing the fresh fuel enrichment, using essentially the same adjustment as for the PWR calculation in Section 8.1. The reference PWR was calculated to require 4.3% ^{235}U enrichment for a 50.6 MWd/kg burnup as compared to its current 3.0% ^{235}U enrichment for 30 MWd/kg. For the current BWR, the ratio of burnup (28.4 MWd/kg) to feed enrichment (2.8%) is 10.143 as compared to the PWR's ratio of 10. Then, for a 4.3% ^{235}U enrichment, the BWR is inferred to sustain a burnup of $1.0143 \times 50.6 = 51.3$ MWd/kg. This burnup is extrapolated to 57.5 MWd/kg using the PWR calculation shown in Table 8.4 that, in this enrichment range, an increase of 0.99% in enrichment is required to increase burnup 15 MWd/kg. The resulting fresh fuel enrichment for the reference extended burnup BWR is 4.71% ^{235}U.

Advanced BWR Reactor

Column B in Table 8.6 presents data on a composite advanced boiling water reactor that produces the same thermal power at 8% higher thermal efficiency using 300 psia higher pressure, and 14% greater core cross-sectional area. The required equilibrium fresh fuel enrichment, 4.08% ^{235}U, is obtained from the estimates for the nonbackfittable concepts included in the advanced design, shown below.

- correction for frequent refueling - An estimated 6% uranium savings can be expected from semi-annual refueling, lowering feed enrichment from 4.71 to 4.46%.

TABLE 8.6 Selected Properties of Reference and Advanced BWRs

Property	A. Reference, Extended Burnup	B. Advanced
Electrical Output (gross)	1220 MW	1220 MW
Thermal Power	3579 MW	3314 MW
Electrical Efficiency	34.09%	36.81%
Effective Core Diameter	4.90 m (193 in.)	5.23 m (206 in.)
Number of Fuel Assemblies	748	852
Mass of UO_2	155 Mg	177 Mg
Feed Water Temperature	376°F	367°F
Core Inlet Temperature	527°F	558°F
Core Outlet (Turbine Inlet) Temperature	551°F	581°F
System Pressure	1040 psia	1340 psia
Mean Exit Steam Quality	14.7 w/o	11.9 w/o
Mean Coolant Flow per Assembly	139,000 lb/hr	141,000 lb/hr
Mean Subcooled Fraction of Fuel Length	24%	30%
Mean Core Water Specific Gravity	0.47	0.48
Equilibrium Uranium Feed Enrichment	4.71% ^{235}U	4.08% ^{235}U
Burnup	57.5 MWd/kg U	57.5 MWd/kg U
Fueling	7 Annual Cycles	14 Semi-annual Cycles
Relative Uranium Requirement[a]	1.0 (reference)	0.787

(a) The normal, 4-cycle annual loading would require 1.158 times as much.

- correction for Xenon and Doppler effects - The BWR lattice at power has a lower power density than the PWR, even for the reference case, so that savings in k_∞ due to lower xenon absorption have been reduced from 0.20 to 0.15%. This is compensated by a higher change in resonance capture from Doppler effect, simply due to the drier (lower H/U atom ratio) BWR lattice at operating conditions. A k_∞ change of 0.4% was applied for these combined effects, which is estimated as being equivalent to reducing feed enrichment from 4.46 to 4.36%.

- correction for larger core size - Leakage is only slightly reduced, lowering feed enrichment from 4.36 to 4.35% ^{235}U.

- correction for spectral shift flow control - Estimates of uranium savings corresponding to flow control in the range of 40 to 150% of rated flow to accomplish spectral shift have been estimated to be in the range of 6% to 11% beyond the limit of backfittability. Taking into account the effects of six month cycles and extended burnup fuel, the achievable value may be near the lower estimate of 6%. A 6% uranium savings is equivalent to decreasing fresh fuel enrichment from 4.35 to 4.12% ^{235}U.

- correction for improved fuel loadings - A 1% uranium savings from these effects (which include some substitution of last half-cycle fuel for blanket) leads to a final estimated feed enrichment of 4.08 wt% ^{235}U.

Uranium Utilization Improvement

Estimates of uranium savings are based on the current BWR's design values of 2.8% feed enrichment to yield 28.4 MWd/kg and the reference BWR's extended burnup values of 4.71% feed enrichment to achieve 57.5% MWd/kg (both at 34.09% electrical efficiency).

Using the extended burnup BWR design as a reference, the composite advanced BWR design would require 20.3% less uranium.

Separative Work

The higher efficiency, composite advanced BWR design requires 22.3% less separative work than the reference, extended burnup system BWR design.

Summary

A composite advanced BWR design, featuring a core 14% larger than the reference design, 6-month refueling intervals, 300 psia higher pressure, and flow control in the range of 40 to 150% of rated flow to accomplish spectral shift, can achieve about 20% improvement in uranium utilization over that potentially achievable from the backfittable improvements included in the extended burnup reference design. The sources of improvement in uranium utilization for the composite advanced BWR design resulting from the various nonbackfittable features included in the design are approximately:

- 8% from higher temperature and pressure
- 6% from rapid/frequent refueling
- 6% from spectral shift with flow control
- 1% from large core, low power density.

The separate nonbackfittable improvements add to 21% uranium savings, whereas the estimated savings for the composite advanced design are 20.3%. This close correspondence supports the assumption that, with regard to uranium utilization, there are no strong synergisms or antagonisms among the separate effects. Any remaining synergisms/antagonisms would principally affect the details of the engineering designs of an advanced BWR system.

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APPENDIX A

ASSESSMENT OF 19 CONCEPTS NOT SELECTED BY INDUSTRY

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APPENDIX A

ASSESSMENT OF 19 CONCEPTS NOT SELECTED BY INDUSTRY

CONTENTS

ASSESSMENT OF 19 CONCEPTS NOT SELECTED BY INDUSTRY

INTRODUCTION

There are a number of other ideas that prior to evaluation were suggested as possibly having some potential for improving LWR utilization. They were not selected for further study and more complete evaluation because their potential improvements in uranium utilization were judged to be negated by perceived technical, economic, operational, and/or safety problems.

A very brief review of these concepts is presented below. For each concept the discussion follows the format used in the concept assessments by industry in this report. In each case the discussion concludes with a preliminary evaluation explaining why the concept was not considered further.

The order of presentation is for the convenience of the reader and carries no implications as to preference. First are presented various fuel lattice ideas, followed by fuel element ideas. Next come reactor core concepts. Then come new control schemes. The next group are ideas for rapid refueling. The final group are thermodynamic concepts, including three superheating methods.

FUEL ASSEMBLY LATTICE CHANGES

Most fuel lattice changes involve variations in pin size, density, cladding thickness or pitch to achieve specific objectives. A few of these changes are potentially backfittable within the envelope of existing fuel assembly design. Relaxing the fuel assembly envelope to accommodate different assembly sizes or shapes could lead to favorable nonbackfittable designs, as for example:

- Hexagonal shapes that could fit better into the circular core barrels and vessels.
- Larger assemblies that would require fewer lifting and lowering operations for fuel reloading, and permit fewer interruptions of the lattice between assemblies.

- Assemblies of special shape and function (viz. filler assemblies at core boundaries, special environments for control members).

Uranium Utilization

This category is too diverse for simple attribution. Uranium utilization could be improved by having larger cores within shroud envelopes, as some designs might permit, or by having assemblies that simplify spectral shift control, as others might. Fuel assembly envelope changes may improve uranium utilization by simultaneously providing desired values of H/U ratio, rod size, water channel spacing, and control rods that could not otherwise be fully optimized.

Economics

Fuel lattice assembly changes would not markedly affect the capital costs of an LWR and any uranium savings they permit would potentially improve economics.

Technical

Some types of change would require investigation of techniques for feasibility; for example, for forming a core shroud that has 120, rather than 90 bends.

Safety

No new problems are foreseen.

Potential for Retrofit

Most such schemes would require replacement of the core grid and shroud. Control rods would also have to be relocated and perhaps completely redesigned in order to accommodate different assembly designs.

Operational

No new problems are foreseen.

Preliminary Evaluation

As of this time, fabrication economies exist as the result of standardization of fuel assembly and control rod envelopes. These economies would be lost if different assembly types would proliferate. Most of the objectives of assembly lattice changes are achievable by other means. Industrial reviewers were of the opinion that the existing standard assembly lattices should be retained unless or until variants could be shown to have overriding advantages; this has not been shown to date.

VARIABLE LATTICE FUEL BUNDLES

A variable lattice fuel bundle is one in which the moderator to fuel ratio can be changed as a function of burnup. This is a method of improving the uranium utilization by controlling the moderator to fuel ratio, which affects reactivity and conversion through variations in resonance capture in ^{238}U and (to a lesser degree) thermal utilization of the fuel. The changes in the moderator to fuel ratio are achieved by mechanical variations in the fuel bundle and fuel pin configuration.

The moderator to fuel ratio affects both the reactivity of the system and the rate at which fissile plutonium is produced. A tight lattice spacing results in a high conversion rate of ^{238}U to ^{239}Pu but little excess reactivity. Conversely, a loose lattice spacing results in an increase in reactivity but reduces the conversion rate.

For a given enrichment, lattice spacing in PWRs are picked such that the longest fuel lifetime would result, always within the constraints imposed by safety. The choice becomes a tradeoff between maximizing the amount of plutonium produced and having enough reactivity available to use it.

Variable lattice designs can be used to change the relationship between initial k, conversion ratio, and end of life k. This can be done by using a tight lattice during earlier portions of a bundle's life to increase the conversion ratio, and compensating by using a looser lattice in later portions of the bundle's life to increase the excess reactivity.

Concepts involving H/U atom ratio changes during refueling by pulling rods from assemblies to produce "wetter" lattices and the formation of new assemblies from the rods that were removed are potentially backfittable. Conversely, those concepts involving lattice changes during each cycle to vary the lattice water content during normal operation of a reactor by mechanical means are clearly nonbackfittable.

Uranium Utilization

Preliminary evaluations of variable lattice design concepts[a,b] indicate that small improvements (<4%) in uranium utilization beyond optimized fixed-lattice values for fuel assembly exposures of 48 MWd/kg are potentially achievable by making lattice changes during refueling. The maximum potential improvement in uranium utilization for nonbackfittable concepts that continuously vary lattice water content during each cycle is about 10% for a five-batch core. These potential improvements decrease as the number of fuel batches in the core and the burnup of the fuel assemblies are increased.

Economics

The concept involving lattice changes during refueling would require an additional onsite facility for reconstituting irradiated fuel pins into assemblies, at a small incremental capital cost. It would also require either longer shutdown times or withholding fuel batches from irradiation while they are being reconstituted, the first of which leads to lower plant availability factors and the second to higher fuel inventory charges. The extra operations involved in reconstituting the fuel assemblies would lead to increased operating charges. These negative impacts must be balanced against cost savings in uranium purchases and/or fuel enrichment charges, which constitute the favorable components of the economic factors in improved uranium utilization.

A feasible scheme for continuously varying lattice water content during normal operation is assumed to be very costly. It may not be economically viable.

(a) Thomas R. Robbins. August 1979. "Preliminary Evaluation of a Variable Lattice Fuel Assembly and Reactor Design Concept." ORNL/Sub-79/13576/1, Pickard, Lowe & Garrick, Inc., Washington, D.C.

(b) Joseph A. Sefcik, Michael J. Driscoll, and David D. Lanning. January 1981. "Analysis of Strategies for Improving Uranium Utilization in Pressurized Water Reactors." DOE/ET/34022-1 (MITNE-234).

Technical

Equipment and systems for rebuilding the fuel assemblies remotely can be designed using standard techniques, but such systems would have to pass through specific pilot and proof-testing development stages.

The concept of having lattices of different channel spacing loaded into the reactor would probably require coolant flow orificing at each bundle in order to allocate the flow properly. For PWRs, this may involve putting each bundle in a "box," as is now done with BWRs. Experiments to validate codes used to design the orifices may be necessary for the licensing process.

Safety

Extra fuel handling, leading to the potential for mechanical damage and increasing employee exposure to radiation, is a negative feature. The rebuilding operation also increases the possibility of positioning fuel rods incorrectly. A positive feature arises from the opportunity to reinspect each pin while rebuilding the fuel. Since the reactor requires decreased control absorption, some large reactivity-excess accidents might be less severe.

Potential for Retrofit

No changes in the reactor primary system are necessary except as required for orificing. These may be built into the fuel assemblies. The "rebuilding" system could be in a new, separate operations building, attached or convenient to the fuel storage pool.

Operational

The contemplated operation would be significantly more complex than the present refueling operations, which many utilities find burdensome already--as evidenced by the growing popularity of 18-month refueling intervals.

Preliminary Evaluation

In view of the fact that this concept would require considerable developmental activity before it could be licensed, as well as requiring a considerable increase in the sophistication of utility operations staff, it was judged not to be attractive for development now.

HIGH PEAKING FACTOR WITHIN ASSEMBLIES

At present, one of the constraints on fuel management is that juxtaposition of fuel assemblies of different burnups often results in higher power in the rods of the more reactive (less burned) assembly that are adjacent to the "older" assembly. This can be compensated by variable enrichment loading in the initial fuel, preferential location of rods containing burnable poison, and other means, but it still contributes to hot spots in the reactor. If the higher peaking factors resulting from this phenomenon could be tolerated, more neutronically-efficient fuel management could be used.

Uranium Utilization

Higher peaking within assemblies does not in itself improve uranium utilization. Indeed, increased peaking could impose burnup constraints, decreasing uranium utilization.

Economics

No basic effects arise from higher peaking that affect costs directly. However, if higher peaking factors are compared with lower peaking factors for a given reactor, the lower peaking factor permits a higher average power density and thus a higher core power. It is not clear how unit costs scale with power for designs in the 1000 MWe range, and so the economics are unknown.

Technical

It is questionable whether assemblies with more heterogeneous burnups of fuel rods within them can be exposed to as high average burnups as assemblies with flatter power distributions.

Safety

There would be no major problems, assuming that, as a design requirement, existing fuel rod power limits are maintained.

Potential for Retrofit

The concept is not backfittable because higher power in fuel rods of current reactors cannot be tolerated; some derating of power would be necessary.

Operational

Reloading operations are simplified, since loading patterns are less constrained.

Preliminary Evaluation

Higher peaking factor is not in itself an economically attractive feature, but it can be considered as an adjunct to the potentially attractive option of designing for lower power densities. That is to say, if lower power density reactors show favorable economics and uranium utilization, part of the increased safety margin on heat transfer might be dedicated to the convenience of permitting higher peaking within fuel assemblies so as to achieve favorable loading patterns.

VENTABLE FUEL

The ventable fuel concept is based on the release and removal of those fission product isotopes with large neutron capture cross sections before they parasitically absorb neutrons. Because the fission yield of the mass-135 chain is high and because ^{135}I decays to the large capture cross section ^{135}Xe isotope, maximum uranium utilization depends on quick release and removal of either ^{135}Xe or ^{135}I. Fission product release might be enhanced by the use of annular UO_2 fuel pellets or through microstructures that enhance solid diffusion and interconnected pore migration. Removal from the active fuel zone, however, will very likely require purging to be effective to maximize uranium utilization.

In the present climate of nuclear regulation, venting of fission gases to the reactor coolant is unacceptable, so that the ventable fuel concept must be associated with a fuel plenum volume above the fueled zone in each rod or a complex, manifolded plenum.

Uranium Utilization

^{135}Xe captures enough neutrons to be responsible for a 1 to 2% loss in multiplication of an operating LWR. Using the rule of thumb that the percentage improvement in uranium utilization is seven times the percentage improvement in neutron multiplication factor (k), there is a theoretical upper limit of the order of 10% in uranium utilization improvement by this technique.

Economics

Some increases in capital costs would be expected for a reactor employing this concept. The plenum adds length to the fuel rods, and there might be some changes to accommodate this length. Longer (higher) reactor vessels might be needed. Minor increases in fuel fabrication costs can also be expected.

Technical

The concept of promoting the release of fission products from solid UO_2 is unprecedented and contrary to all current LWR fuel performance improvement efforts that seek methods to either minimize fission product release and/or minimize the thermo-mechanical/chemical pellet-cladding interaction (PCI) effects and the likelihood of fuel failure by iodine stress corrosion cracking (I_2SCC) and/or liquid/vapor metal embrittlement (L/VME) of the Zircaloy cladding. The development of ventable fuel must therefore include some accommodation to minimize or at least not significantly aggravate fuel failures due to PCI effects.

There is no experimental data to assure that the volatile ^{135}I and gaseous ^{135}Xe can be released from the solid UO_2 in sufficient quantities prior to neutron capture to be of practical value with respect to uranium utilization. With normal UO_2 fuel, less than 10% of all fission gases are relased from the UO_2 over the long term. The mechanisms for fission/decay product release to free surfaces of the UO_2 are not well understood.

Safety

Fission product venting could lead to increased fuel failure rates by the various mechanisms noted above. Much more importantly, all the worst fission products are concentrated and withdrawn through a fallible system rather than contained by the UO_2 crystal structure.

Potential for Retrofit

Assuming a viable system for fission product venting can be developed without increasing fuel assembly height or reducing reactor power, the concept is retrofittable.

Operational

There is a reduction in the likelihood of going into spatial xenon oscillations, and xenon removal--but more to the point, iodine removal--would very much simplify power recovery after hot shutdown.

Preliminary Evaluation

There is no technical basis for confidence that the concept can be successfully developed; this judgment relegates the ventable fuel concept to low priority in spite of its potentially desirable features.

TUBULAR FUEL

A tubular fuel element is a cylindrical fuel design that has a moderator in contact with the clad both inside and outside the fuel. Use of tubular fuel permits two moderator regions, one between the elements with water or steam and one inside the annulus with steam, graphite, or water.

The moderator to fuel ratio can be modified by varying both the degree of boiling in the exterior region and the degree of moderation in the interior region. This flexibility would improve reactivity control and ^{239}Pu buildup. Uranium utilization could be improved several ways with this concept, but only one of these would be applicable for a given reactor. By adding moderating capability to the system as the end-of-cycle approaches, the advantages of improved ^{239}Pu formation and reactivity increase with the amount of moderator that can be utilized. This would be accomplished by switching from steam to water inside the annulus as burnup progressed. If a solid moderator such as graphite were used, neutron economy would be improved and the required enrichment to obtain a given exposure would be reduced. By using coolant in the annulus, operating margins could be increased, which would relax the constraints on fuel management and permit improved uranium utilization.

Uranium Utilization

While tubular fuel is qualitatively conducive to improved utilization, the amount of improvement is strongly contingent on specific designs. No design that is compatible with LWR conditions has yet been brought forward for analysis.

Economics

Similarly, there is insufficient data for economic assessment. The individual tubular fuel elements would probably contain more fuel than existing fuel pins do. To the extent that fuel fabrication would therefore require fewer operations, economies might result. However, fabrication is one of the lesser expenses of the fuel cycle, and these economies could not be very great.

Technical

Tubular fuel would be clad by two cylindrical tubes, presumably made of Zircaloy: one on the inside and one on the outside. The manufacture of annular fuel pellets to the close dimensions required by maintaining well-controlled fuel-cladding gap tolerances could be a problem. Power gradients around the tubular fuel could cause unusual thermal stresses in it.

Safety

If the fuel is designed to be cooled inside and out, any internal flow blockages could starve the coolant channel everywhere above the flow restriction.

Potential for Retrofit

As a fuel change, this concept would not be difficult to retrofit, unless its effectiveness depends on moderator control in the inner channel. However, large tubular fuel elements are arranged more compactly in hexagonal assemblies, and such assemblies can not be retrofit.

Operational

No unusual problems are foreseen.

Preliminary Evaluation

There are no specific designs to evaluate and, in their absence, generic reasons for developing the concept are not overriding.

SEED AND BLANKET CORES

In seed and blanket reactors, high uranium utilization is achieved by using highly enriched driver zones to minimize parasitic neutron absorption, together with blankets of low-enriched uranium or thorium to catch the neutrons that leak from the seeds. Only the seed is changed at refueling, and the small number of seed assemblies permits (in principle) rapid and frequent refueling to minimize control requirements. Over reactor life, the blanket builds up fissile material that is burned in situ.

Uranium Utilization

The LWBR core that has been installed in the Shippingport reactor is designed to achieve approximately 1:1 conversion while operating on the thorium cycle. However, it is a consumer of enriched uranium that is burned in the early seeds, and only about a 10% improvement in uranium utilization over existing LWRs is expected if the blanket is not reprocessed to recover converted ^{233}U. This improvement only becomes evident late in reactor system life, after appreciable fissile material has accumulated in the blanket. In contrast, LWRs of conventional design are capable of more than 20% improvement in uranium utilization over present practice as a result of backfittable fuel design changes.[a]

Economics

Capital costs of seed-blanket reactors are expected to be significantly higher than those of standard LWRs. A major factor is the peaking of power generation in the seeds over most of the core life, which leads to a lower power density than conventional LWRs for a given sized pressure vessel.

(a) NASAP, DOE/NE-001/9, Vol. IX.

Separative work requirements for the early seeds are high. Therefore, neither the standard Shippingport reactor seed and blanket design, nor its LWBR modification, have generated commercial enthusiasm.

Technical

The long and continuing development activities of the Naval Reactor Program have resolved most of the technical problems that might arise. Indeed, oxide fuel, which is the backbone of commercial LWRs, is a spin-off from this program.

Safety

No unusual problems have been encountered in the Shippingport prototype. Application to large LWRs has not been evaluated.

Potential for Retrofit

Existing PWRs could be modified for seed-blanket operation by replacing essentially all core internals, but this would be both expensive and time consuming.

Operational

No problems have been uncovered at Shippingport and none are foreseen for this type.

Preliminary Evaluation

The concept is proven, but its commercial adoption would seem to entail economic penalties without compensating economic improvements from fuel cycle savings.

SPECTRAL SHIFT (Other than by D_2O-H_2O Mixing)

These concepts achieve control by varying the moderating power of the reactor. By displacing hydrogen (water molecules) with a less effective moderator, the neutron spectrum in the fuel is hardened. This spectrum change increases the formation rate of fissile ^{239}Pu, which flattens the reactivity burnup slope. Because the less effective moderators have smaller thermal cross sections than hydrogen, fewer thermal neutrons would be absorbed. Near the end of cycle the less effective moderator would be removed and water added to increase the reactivity of the core.

Uranium Utilization

These concepts reduce or eliminate the need for control poisons and have potentially large improvements in uranium utilization. To illustrate, the developers of the SSCR, a D_2O-H_2O version of a spectral shift reactor, had claimed savings of over 50% in uranium utilization for their concept over standard LWRs,[a] a number that also corresponds to about a 40% savings over extended burnup LWRs.

Economics

In spite of these very large potential fuel savings, the capital expenses and O&M costs of the spectral shift-controlled reactor (SSCR) have been judged to be too great for this to be a commercially viable reactor. The concepts discussed here are all attempts to avoid some of the D_2O-associated costs of SSCR, but none of them have been judged to be economically promising.

Technical

By varying the hydrogen density as a function of time, the neutron spectrum can be changed to obtain better reactivity and burnup characteristics. The moderator characteristics could be adjusted by the use of:

- carbon particles in the water
- gas bubbles in the water
- gas-filled tubes between fuel rods
- beryllium-filled tubes between fuel rods
- graphite-filled tubes between fuel rods.

All of these materials require a system for adding and removing the diluent as a function of time. Many of these ideas have been demonstrated experimentally, and seem more appropriate as special control devices than as whole-reactor modifications. The exception is variation of void content in BWRs by flow control, a concept that is in use and is considered separately in this report.

(a) ORNL/TM-5565, p. 158, Table 2-13.

Safety

As new control techniques, spectral shift concepts would require large-scale statistical testing under a variety of conditions before one could rely on their integrity against accidental reactivity insertions.

Potential for Retrofit

As whole-reactor concepts, potential for retrofit on existing LWRs is minimal.

Operational

If developed, none of the ideas would present operational difficulties. Essentially, control rod function is taken over by spectral shift parameter control.

Preliminary Evaluation

Spectral shift is more promising for special control devices, where it would serve as a form of fertile material control; these are separately covered in this appendix. No advantages, and considerable extra expense, are obtained by adopting spectral shift as a basis for whole-reactor design.

PWR FLOW CONTROL

In PWRs, the power varies from bundle to bundle. Coolant flow is, however, roughly constant. Thus, more coolant is pumped than is necessary: the flow rate is determined by the requirement for cooling the hottest bundle. As a result, mixed mean exit coolant temperatures are lower than in an ideal system and plant thermodynamic performance suffers.

Controlling the flow to individual bundles would not only permit better thermodynamic performance, but also makes it possible to improve the cooling of hotter elements, making operation at higher peaking factors possible. High peaking (low neutron leakage) fuel management, made possible by flow control, would save uranium.

Uranium Utilization

Improvements in plant thermal efficiency are unlikely to be more than 5%. The high peaking factor capability could add 3% to uranium utilization efficiency. However, the fuel assembly shrouds would increase the parasitic capture of neutrons in the reactor and detract from the potential savings. An upper limit to uranium utilization improvement would be about 8% for this concept.

Economics

No extraordinary capital costs are involved in this concept, and the potentially favorable fuel cycle improvements it could facilitate may make it economically attractive.

Technical

Since a PWR flow control system is one that regulates the flow in each bundle, the standard would be a system of orifices that vary as fuel is shuffled. Coupled with these would be shrouds around each PWR assembly to eliminate cross flow; the PWR fuel would then look, physically, more like BWR fuel. Perfect orificing is not possible, so this method would not achieve ideal performance. In addition, the fuel assembly shrouds would add both neutron parasitic absorption and flux peaking in the outside elements of the assemblies. Orificing the fuel assemblies increases core pressure drop and pumping power, cutting slightly into net plant efficiency.

Safety

No unusual problems are foreseen. Some checkout of orifice performance would be required. Shrouded fuel would permit, at some expense, more detailed measurement of radial thermodynamic performance, which could be helpful.

Potential for Retrofit

Retrofit potential is high. Indeed, clever mechanical design might make the orificed, shrouded assembly backfittable.

Operational

Extra operations (changing orifices) would be needed for fuel assemblies that are moved. No other operational problems are foreseen.

Preliminary Evaluation

The concept is attractive. It does not, however, in itself offer a basis for backfittable design. Therefore, it is considered as a technique possibly to be used in conjunction with other primary concepts, such as low neutron leakage fuel management and lower power density cores.

FERTILE MATERIAL CONTROL

During most of a cycle in present LWRs, excess reactivity is controlled by absorbing neutrons in some high cross-section material such as boron. This concept would utilize those neutrons instead to create fissile fuel that would eventually be used to drive the core. By using a fertile material such as ^{233}Th or ^{238}U to control the reactor, the excess neutrons generate fissile fuel. This fuel contributes to reactivity at the end of life, permitting a lower initial reactivity (enrichment) or longer burnup.

The concept could be implemented by using fertile material in place of control fingers in a PWR or by replacing the BWR control blades with fertile curtains. The major disadvantage to this method is that fertile materials do not produce a large reactivity swing and so large numbers of control elements would be required. Another difficulty of this otherwise obvious technique is that, if the rods remain in the reactor a long time, their control capability continually decreases. These disadvantages can be overcome with spectral shift control devices.

Uranium Utilization

Parasitic neutron absorbers control, on the average, 2 to 3% of the neutron multiplication factor (k) during a cycle. This corresponds to up to 20% potential improvement in uranium utilization.

Economics

The core space that must be allotted to fertile control elements, or to spectral shift elements, reduces the number of core fuel elements that can be put into a given vessel. This diseconomy negates potential fuel savings.

Technical

The size and weight of fertile control elements require more space and volume, as noted, and more powerful control drives than absorbing controls do. The variation with time of the fuel content of fertile assemblies requires overcooling at first so that later fissions in these assemblies can be adequately cooled. Fertile assemblies are more transparent to thermal neutrons than standard fuel is, and this creates hot spots in the adjacent fuel.

Safety

The technical problems just outlined have safety implications, none of which appear to be insurmountable by careful design and development, however.

Potential for Retrofit

Redesigned control motors could probably be retrofitted into reactors of present design. However, this would be a minor fraction of the changes that are necessary, and total backfittability is doubtful.

Operational

No unusual problems are foreseen; however, the operators must take into account the need for changing control assemblies as fissions accumulate in them.

Preliminary Evaluation

Fertile material controls have been considered for LWRs but have not been adopted. The primary reason is that the problems associated with their bulk, weight, and shifting cooling requirements have been judged to outweigh their uranium savings potential.

FISSILE CONTROL MEMBERS

Present LWRs use a combination of movable control materials, fixed control materials, and soluble control materials in order to maintain reactor criticality. Control materials function by absorbing neutrons. These neutrons are lost from the system without either causing a new fission event or generating new fissile material. In a typical PWR, 3% of the neutrons are lost in this way.

If the 3% of the neutrons that are lost in control materials could be eliminated, the uranium utilization could be increased by as much as 20%. This is of course an upper bound, but 10 to 15% might be achievable.

One way of eliminating control absorption is by varying the amount of fissile material in the core. Enriched fuel is substituted for control rods. If the reactor has too much reactivity, the enriched rods are withdrawn, to be inserted as the reactivity decreases.

Uranium Utilization

It is possible that on the order of 10% decrease in uranium requirements could be achieved. Separative work requirements are not expected to change much, as the extra separative work units (SWUs) needed for the enriched rods would be counterbalanced by the lesser SWU requirement for the main body of reactor fuel.

Economics

Although a fissile fuel motion control system would be quite different from a conventional rod and/or soluble poison system, it should not have a major effect on the capital cost of the plant. Avoidance of burnable poison should decrease fuel fabrication costs slightly. The uranium savings are reflected in lower fuel cycle costs.

Technical

Control by fissile fuel motion has not been used in large commercial LWRs. However, the techniques have been used in the Light Water Breeder Reactor (LWBR) program. Even though the objectives of the LWBR program are different from this program, some of the ideas are applicable. In particular, the use of a bypass inlet flow (BIF) system to scram upward-inserted rods by gravity, even in the presence of strong hydraulic lifting forces, has been demonstrated.

As in the LWBR method, the fuel would initially be below the reactor. Criticality would be obtained by pushing the fuel into the core. This is a major departure from existing PWR control design. An extended blanket could be included on the top of the fuel to reduce the water gap effect in the channel above the movable assembly.

The major technical problem with movable fissile fuel control is not in the design of the control system, but in the design of the fuel configuration such that the power peaking is maintained at an acceptable level. A suitable combination of fuel burnups in assemblies adjacent to the control members, or controlling the flow to each assembly, could resolve this problem. If movable fissile fuel control is used in a BWR, a flow orificing system exists to do this. In a PWR, the flow is presently not controlled to each assembly.

While specific problems in mechanical design and thermohydraulic design can be solved by straightforward bench and component testing, the concept is a sufficient departure from existing practice to require at least a demonstration loading or reworking of an existing reactor for demonstration purposes.

Safety

Assuring the safety of this concept would require extra attention above and beyond that already given to LWRs. The addition of mobility to a potential hot spot--the movable, enriched fuel--means that mechanical integrity problems are superimposed on the thermohydraulic considerations. Shutdown conditions would be different. There are likely to be more changes than usual in location of power peaks as "control" fuel is moved past "normal" fuel, and the resulting thermal cycling could cause increased fuel failures.

Potential for Retrofit

This concept entails a drastic design of core internals, which is not easily retrofittable.

Operational

The reactor is much more complicated both in flow characteristics and fuel heterogeneity than current designs. In early plants this may cause problems that reduce availability and consequently increase power costs. In mature designs it is not obvious that the increased complexity would result in a reduction in availability, but it is a possibility. The heterogeneity of the reactor power could result in an increase in fuel failure and potential unplanned outages or longer planned outages to locate and remove failed elements.

The refueling/maintenance outages would be complicated by requirements that the fissile control elements would need servicing and periodic replacement more frequently than conventional control elements.

Evaluation

Major developmental efforts (and costs) would be required to incorporate this concept into LWR designs. A major safety and licensing effort would also be required. The need for development and demonstration suggests that further first-of-a-kind design costs, costs associated with the learning curve on system engineering and operation and costs associated with licensing, will be considerable. It is judged a high-risk investment for the fuel cycle cost savings that would accrue.

FUEL MOTION CONTROL (Movable Assemblies)

One way of achieving fissile material control is with movable assemblies. In this system the reactivity would be controlled by the motion of selected fuel bundles in the core. These bundles would be movable from below. This concept combines the features of both "Fertile Material Control" and "Fissile Control Members" described in earlier sections of this appendix. Fuel Motion Control does not require as frequent replacement as fissile control members nor as many movable members as fertile material control. Movable fuel would be initially inserted into the core far enough to achieve criticality. As the reactivity decreased, the bundles would be inserted further into the core. Conceivably, they would be inverted between cycles to take full advantage of the uranium. A blanket or reflector could be underneath the active core to reduce the leakage. This system replaces the need for absorptions of excess neutrons in boron. The increase in neutron economy allocates more neutrons for conversion of fertile materials and decreases the initial enrichment.

Uranium Utilization

An estimate[a] of the magnitude of the improved uranium utilization indicates that a 10% to 20% improvement could be obtained as the result of complete elimination of neutron poison for regulation and shimming.

(a) Rampolla et al., WAPD-TM-1371.

Economics

Although provision for cooling the movable assemblies would add cost to the system, this added cost would not be likely to be enough to balance the savings in the fuel cycle cost.

Technical

A major difficulty with fueled control rods is simply the high rate of heat generation in the moving rods for which proper coolant flow must be designed. In PWR design, the problem is to make provision for flow near the inlet end; cross flow might be adequate to provide coolant further upstream. In BWR design (upward motion to decrease reactivity) the chief problem would be to assure that the moving inlet of the control assembly continues to receive its quota of flow. For both types of LWR, shifting axial power patterns must be dealt with.

Safety

Problems of providing emergency cooling to the control assemblies are compounded because of the vital role of these assemblies in maintaining safe shutdown.

Potential for Retrofit

Fuel motion control is not backfittable, and retrofit would be possible only to a limited extent (partial fueled control assemblies).

Operational

The need for monitoring the burnups of control assemblies and changing them periodically makes operation and fuel management more complex.

Preliminary Evaluation

The concept requires a great many changes in reactor design, complexity in operation, and poses new problems to the safety analyst. It is not clear that resolution of these problems would result in favorable economics.

HOT STANDBY REFUELING

The prime features of the hot standby concepts are an enlarged reactor vessel with space for storing used and unused fuel elements, and an internal transport mechanism for handling the stored fuel elements. The purpose of these features is to permit refueling to take place without the time-consuming phases of removing the reactor head in order to gain access to the core. With this capability, very frequent refueling, with attendant reduction of control poison requirements for burnup reactivity change, might be accommodated.

Uranium Utilization

If we assume that hot standby refueling would permit (on the time-scale governed by in-core fuel lifetime) virtually continuous refueling, one has gains of the order of 16% in uranium utilization over an equivalent five-batch fuel management scheme. Practical gains would, of course, be less, since pressure vessel cooling, opening, resealing, and reheating are not the only contributors to refueling shutdown time, which must be minimized to economic optimum. Six-month, rather than almost continuous refueling decreases the gains to about 8%.

Economics

Capital cost increments for this system are likely to be high. It must incorporate enlarged vessels and in-vessel mechanical equipment that must be rugged, sophisticated, and highly reliable. Considering that the fuel cycle gains would be less than 16% of that cost, it is questionable whether a hot standby refueling system would lead to economies in power cost.

Technical

In order to accommodate an array of both unused and used fuel elements within the reactor chamber, additional space must be incorporated. A choice of location arises for the location of the fuel element storage space. The space could be designed above or below the core assembly. While many of the problems are similar, above-core storage would be simpler for removal of spent fuel from storage during outages. This would require below-core control systems, already in use for BWRs, but a change for PWRs.

Additional space is required above the core to provide enough room to store the full length of the standby fuel elements and the remote fuel handling device to implement the transfer process. It is envisioned that the stored elements would be locked into positions around the perimeter of the storage space.

The remote fuel handling device for transferring fuel, to and from storage, would have to cover both the core and storage spaces by combined translation and rotation motions (as well as lifting and lowering ones). It would have to perform with accuracy and reliability in the hostile environment within the head of the reactor (2000 psi water and steam, 500°F, and radiation).

Safety

Storage of spent fuel within the reactor vessel adds to the fission product inventory that must be considered under accident conditions. Unless the refueling machine is very reliable, one can contemplate a variety of loading/unloading accidents which could damage fuel, expel control rods and so on. Personnel exposures to radiation during refueling would be reduced.

Potential for Retrofit

There is no potential for retrofit on existing designs. The changes required are fundamental and would call for extensive development and demonstration.

Operational

In many ways, hot shutdown refueling could simplify operations. It would mechanize what is now an operator-controlled activity. However, it is also possible that the refueling machine may require frequent maintenance, which could increase both downtime and personnel exposure to radiation.

Evaluation

Hot standby refueling would require a major development and demonstration program. If it worked well, it could improve uranium utilization and simplify operations. A development and demonstration program would have to assure that there are no safety impacts. Economically, uranium savings and increased vessel internals and mechanisms costs would somewhat cancel each other. The large expected development cost can not be expected to be financed by industry under these circumstances. The concept would require a detailed redesign of the entire LWR primary system, and might raise some separate new licensing issues.

ON-LINE REFUELING

The purpose of this concept is to permit refueling during reactor operation. This provides the most flexibility in adjusting fuel loadings and eliminates the downtime associated with refueling.

Uranium Utilization

Fuel savings of perhaps 16% could be achieved, since on-line refueling eliminates the need for using neutron poisons to provide control for burnup reactivity changes.

Economics

An LWR with on-line refueling would require a full development program, essentially as a new reactor type. The reactor would require additional equipment over and above that needed for hot standby refueling (discussed previously), and the equipment could be very costly.

Technical

On-line refueling requires all the development needed for hot standby refueling, and must in addition ensure that fission heat generated while the fuel is being moved can be safely removed. It is questionable whether a practical system can be developed.

Safety

The heat generated during fuel relocation considerably increased the possibility of fuel damage during that operation. Mechanical errors in moving fuel and inserting or discharging it from the core could lead to reactivity insertion accidents that damage the operating core, that would have serious consequences.

Potential for Retrofit

Essentially none.

Operational

By superimposing refueling operations on power generation operations, the complexity of the process is increased.

Preliminary Evaluation

The concept is probably not practical. Even if it could be developed, safety and operability would be compromised.

UNIT CORE REFUELING

Considerable time during the refueling period is spent removing and replacing individual fuel assemblies and relocating fuel assemblies within the core. The refueling period can be reduced significantly if these operations could be done either with complete cores or with portions of the core so that individual assembly movement can be done between refueling periods. A unit core refueling concept using multiple cores for the plant extends the "interchangability concept" to the core design and minimizes time spent during refueling manipulating core assemblies. This concept would best be implemented in a multiplant scenario where the impact of the increased fuel inventory is minimized. It is only effective when fuel shuffling becomes the critical path item on the shutdown.

Uranium Utilization

Unit core refueling is a means for expediting refueling, making frequent refueling more attractive. In itself it does not affect uranium utilization. Semiannual refueling would improve uranium utilization by about 8% in comparison to annual refueling.

Economics

Considerable expense could be incurred in designing the machinery to lift, cool, and properly emplace a whole reactor core at one time, or even a large fraction of it. The core itself might have to be redesigned and built along different principles, considering that about 200 fuel assemblies must be moved as a unit. These expenses are a drawback to the concept.

Technical

The economics problems reflect technical difficulties. Assuming that existing fuel assemblies are retained, the most likely design would comprise a secondary, movable grid structure that would support the fuel and must be accurately positioned with regard to the fixed core grid. Obtaining the required geometric precision using the necessarily heavy members would be difficult. The heavy PWR core shroud might also have to be movable under the same constraints. The design might be slightly easier for BWRs, but for both types of LWR the massive cooling required during transfer could cause problems.

Safety

The chief problems are making sure that control rods, required to be moved with the unit core, remain inserted during all operations, and removing the large amount of shutdown heat during transfer. The massive fuel movement magnifies the impact of what would normally be smaller accidents such as dropping a fuel assembly (e.g., dropping the core as a unit).

Potential for Retrofit

None

Operational

There are operational advantages to doing fuel shuffling away from the reactor.

Preliminary Evaluation

The technical and safety problems are formidable and forbidding.

INTEGRAL NUCLEAR SUPERHEAT

Superheating the steam that enters the turbine of a nuclear power plant would improve uranium utilization by increasing plant thermal efficiencies. In the integral nuclear superheat concept, boiling and superheating are accomplished in separate regions of the same core. The most popular concept[a]

(a) Mitchell, GEAP 20950.

boiled water in an outer annular region of the core and returned the steam through the center of the core for superheating. Beyond the reactor vessel, the equipment and controls are much the same as a modern fossil-fueled power plant.

Uranium Utilization

In contemplating superheat, one gains the advantages of improving plant thermal efficiency from 32 to 40%--a relative improvement of 25%, which corresponds directly as a uranium utilization improvement. However, superheating involves some sacrifices of neutronic performance, of which the most important, but not the only factor is due to the neutronic losses from substituting steel for Zircaloy in core cladding and structures. At superheat temperatures, Zircaloy cannot be used. Losses from these sources might be 10 to 15% in uranium utilization, leading to a net 10 to 15% improvement in uranium utilization as a more realistic target.

Economics

Construction of an integral superheat reactor is more costly than a saturated steam-producing BWR. Favored schemes have involved radial separation of boiling and superheat regions, requiring physical separation and manifolding. The experience with the two projects involving integral nuclear superheat, Pathfinder and Bonus, have been unfavorable. This indicates that integral nuclear superheat is uneconomic, even though some of the balance-of-plant system-capital costs/electrical kilowatt are lower (i.e., steam turbine and piping) and the thermal efficiency is higher.

Technical

Integral nuclear superheat reactors have two attractive features. First, with both the boiling and superheat sections in the same core neutronically, there is more potential for synergistic effects from spectrum control. Secondly, with all fuel in the same region of the pressure vessel, emergency cooling of the superheat fuel would be simplified.

The chief technical problem has to do with cladding and structural materials in the superheat zone. Zircaloy is not suitable and even stainless steels have not shown good endurance against corrosive attack. A substantial development task remains to find a material that will perform well in this demanding environment, and that is at the same time economically and neutronically acceptable.

Safety

The fuel in a superheat zone would be hotter and closer to melting, as well as having more stored energy than standard LWR fuel.

If a usable cladding material can be found for the superheat zone, the chief remaining safety problem is to cope with the neutronic and thermal coupling between boiling and superheat zones. Under transient conditions, instabilities might develop.

While boiling water reactors with saturated steam have shown good retention of solid fission products in the water phase (decontamination factor), there is some concern that superheating systems might carry over fission products, as aerosols into the turbine.

Potential for Retrofit

There is virtually no potential for retrofit of integral superheat modifications into an existing BWR.

Operational

Operation of an integral superheat reactor would require very careful balancing of reactivities, flows, and powers between boiling and superheating regions. Automatic alarms and controls might be required. Such reactors might be difficult to operate at power levels significantly different from the design-rated power.

The balance-of-plant power generation system would be very much like that of a standard modern fossil unit. Mechanical maintenance would be more familiar to fossil-trained operators, although fission-product carryover to the turbine would introduce additional operational problems during maintenance outages.

Preliminary Evaluation

The concept has the potential for substantial improvement in uranium utilization. It would also use turbine equipment of a more standard design, in comparison to fossil-fired units.

However, there is some risk that this reactor type might present problems of operating stability and maneuverability. It is speculative whether a cladding material can be developed for the superheat fuel. Significant material, operational, and safety problems must be resolved. Because integral nuclear superheat projects have been undertaken several times, and have failed each time, this design concept must be put into a low category for further examination.

ADD-ON NUCLEAR SUPERHEAT

Add-on nuclear superheat reactor concepts use a physically separate reactor to generate the superheat. A standard boiling water reactor can be used as the front end of the system to feed saturated steam into the superheater. The superheat core would be operated in tandem with a standard LWR in a separate vessel. Add-on superheat could also be used with the saturated steam from PWRs. The objective is to achieve the improved efficiency of superheated steam cycles.

Uranium Utilization

There are two ways of assessing add-on superheat. If we consider the incremental effect of extra energy added at superheat temperature to what would otherwise be a 32% efficient cycle, the extra electricity can be generated at well over 50% efficiency; or, the combined boiler-superheater combination could generate at over 40% efficiency. In either case, extra efficiency translates directly into uranium savings. Depending on the type of superheating reactor chosen, up to 30% uranium savings might be achieved in a LWR-superheater combination as compared to LWRs alone.

Economics

A superheating reactor is a gas (steam) cooled reactor. Since the rate of transfer of heat from fuel cladding to coolant is much lower for steam cooling than for water cooling, the power density of the superheating reactor is substantially less than that of current LWRs. In addition, the high coolant and fuel temperatures of the superheating reactor would rule out the use of Zircaloy cladding, increasing fuel fabrication costs, and decreasing uranium utilization. This superheating reactor system would have a capital cost that is considerably higher than LWRs. In the present economic climate, a reactor cooled by superheated steam could not be competitive.

Technical

Add-on nuclear superheat has the same problems as integral nuclear superheat of fission product carry over from the superheat core to the steam system as well as the problem of finding an adequate cladding for the fuel. One superheating reactor experiment has already been built, operated, and decommissioned.[a] Based on that experience, in which the cladding did not perform well, the development of cladding materials for superheat service was not considered to be attractive for private industry. In addition, emergency core cooling measures must be developed for the superheat reactor.

Safety

The problems listed as "technical" all have safety implications.

Potential for Retrofit

Add-on superheat intrinsically requires completely new reactor designs.

Operational

The chief operational problem is one of capacity factor. If the saturated-steam-producing reactor is down, the superheat reactor is useless unless an expensive Loeffler cycle capability is built in with it. Conversely, if the superheater is down, it and parts of the turbine system must be bypassed in order to operate the BWR or PWR. In short, the availability factors of the two units separately must be combined to find a basis for availability of the two together.

(a) HDR in Germany.

Preliminary Evaluation

Add-on nuclear superheat has little near-term or middle-term potential. It requires successful development of improved cladding and structural materials, achievement of high unit reactor availabilities, and resolution of economic problems concerning capital cost. The concept was tried at HDR in Germany, but was not pursued further due to technical failures experienced.

ADD-ON FOSSIL FUELED SUPERHEAT

In this design modification a standard LWR system is used to feed saturated steam into a fossil fired boiler. The boiler generates superheated steam. Fossil fueled superheaters have been added to nuclear power plants in the past[a] to increase the overall system efficiency to the 36 to 38% range. This concept does not introduce the problems of developing a nuclear superheat core, but it does not increase the uranium utilization as much either. The increased uranium utilization is accomplished in a different manner than with nuclear superheat. The fossil fuel is utilized at 40% efficiency. If it were burned in a fossil fueled plant without the preheat it would be used at 36% efficiency. The overall system, therefore, becomes 4 to 5% more efficient than the two individual systems.

Uranium Utilization

If the extra system efficiency were partitioned between the nuclear and fossil units, one could interpret this as a 2 to 2.5% improvement in uranium utilization.

Economics

The economics are not promising because of the high cost of such a hybrid system with its controls, as well as the lower total system availability. Add-on fossil fueled super heat nuclear plants would have lower-than-desirable capacity factors, as demonstrated by such projects in the past.[a].

(a) Indian Point Unit 1 and Elk River, in the United States, and Lingen in Germany.

Technical

Standard fossil-fueled superheaters use higher pressure steam than an LWR would provide, which is why this system would not be quite as effective as one might expect.

Safety

There are no new problems.

Potential for Retrofit

The superheated steam produced by the add-on fossil-fueled boiler would require replacement of the steam-turbine system due to the inability to control and match flow conditions between the existing (saturated steam) turbine and a separate turbine-generator system accepting superheated steam. The extensive modifications required to the plant to retrofit an add-on fossil-fueled boiler make it unfavorable.

Operational

Problems are foreseen in the ability to control the operation of two separate types of plants in matching conditions.

Preliminary Evaluation

Tandem operations of two plant types, and unfavorable economics make this concept unattractive.

SUPERCRITICAL PRESSURE LWR

Fossil-fueled supercritical pressure power plants went into commercial service in the late 1950s. Plant efficiencies of 40% were achieved. Soon after this, two supercritical pressure nuclear reactor concepts were studied.[a] Although both projected 40% efficiency and reasonable costs, the reactor designs were a considerable departure from PWR technology. The high efficiency would be achieved with a direct cycle, without phase change.

(a) Tower et al., WCAP 2042; Hanford report HW-59684.

Uranium Utilization

As with superheat reactors, the possible 25% improvement in uranium utilization from higher efficiency is somewhat compromised by the inability to use Zircaloy cladding. Savings of 10 to 15% are all that might be achievable. However, since supercritical reactors could operate with a wide range of coolant pressures available to a plant operator, this could result in the availability of a variable neutron spectrum for fuel management and extended life. A synergistic effect between increased efficiency and spectral shift could be achieved.

Economics

If the reactor could indeed be built to operate reliably, it might have very favorable economics.

Technical

Both Hanford and Westinghouse have studied this concept. The very high coolant pressures in the core (6000 psi) resulted in the coolant being contained in pressure tubes. The Hanford concept utilized an inverted fuel design (i.e., internally cooled matrix); the Westinghouse design used larger pressure tubing, and were direct cycle plants.

Probably the most important technical factor in the concept is that, after experience with supercritical pressures, fossil fuel designs reverted to subcritical pressures. There exists no really successful experience with the power conversion equipment therefore.

Safety

The direct cycle without a phase change would cause increased fission product carryover to the power conversion equipment. The very high plant pressures would be outside the range of current LWR emergency core cooling technology. All safety aspects would have to be addressed in evaluating this concept as well as the technology bases for a fuel design.

Potential for Retrofit

None

Operational

A successful supercritical pressure system would have operating characteristics rather like a BWR, which are quite acceptable.

Preliminary Evaluation

In the absence of acceptance of fossil-fired, supercritical pressure systems, the nuclear version faces an extremely difficult development program that includes a high risk of failure.

BOTTOMING CYCLES

The power conversion cycle can be improved in efficiency by bringing the heat rejection temperature of the working fluid down from about 100°F (present practice) to about 50°F, as might be available as cooling temperature at cold-water sites. Practically, this involves transferring heat from steam to a more volatile working fluid such as benzene, ammonia, or sulfur dioxide. This option is called a bottoming cycle. Bottoming cycles could in principle increase plant efficiencies by about 10%, with resulting savings in fuel, thermal power rating, and so on.

Uranium Utilization

A 10% improvement in uranium utilization would result from a 10% improvement in plant efficiency.

Economics

Bottoming cycles have not generally been adopted in power plants because the savings from them do not justify the extra capital costs of supplying a large heat exchange surface to exchange heat from water to bottoming fluid and a special bottoming turbine. Indeed, these same considerations are what make ocean thermal energy conversion (OTEC) uneconomic.

Technical

No technical problems exist.

Safety

No safety problems exist. Bottoming cycles are an add-on to existing power conversion cycles.

Potential for Retrofit

Bottoming cycles are potentially retrofittable from a technical standpoint, but not economically feasible. The only significant effect on the primary system would be that feedwater temperatures could be changed. This might affect the thermal power that could be achieved in the reactor.

Operational

No problems are foreseen.

Preliminary Evaluation

If bottoming cycles were economic there would already be considerable application of them, both in fossil-fired and nuclear units. The lack of such application attests to the economic disincentives.

APPENDIX B

ESTIMATE OF URANIUM SAVINGS ACHIEVABLE BY USE OF A RADIAL BLANKET AND ZIRCALOY CORE BAFFLE

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APPENDIX B

ESTIMATE OF URANIUM SAVINGS ACHIEVABLE BY USE OF A RADIAL BLANKET AND ZIRCALOY CORE BAFFLE

One task of the preliminary engineering study focused on identifying and assessing "barriers" that could hinder the implementation of the proposed concepts. One of these potential barriers relates to the question of uranium savings. Recent estimates of the uranium savings achievable via these design features have indicated 3 to 5% savings for a radial blanket and 1 to 2% for a Zircaloy baffle. These estimates, however, were generally not well documented regarding calculational methodology. Because of this, and because substantial uranium savings constitute a fundamental requirement for feasibility, a brief calculational study was performed to independently verify that the uranium savings expected from the use of a radial blanket and a Zircaloy core baffle are in the range indicated. B&W and PNL collaborated in this study, and the results are presented in this appendix.

One-dimensional (radial) calculations were performed at both B&W and PNL to determine the uranium savings over 30 years of reactor operation assuming annual refueling cycles and a 75% capacity factor. The core and baffle/reflector regions, as described in the preliminary engineering study, were modeled as concentric regions in these calculations. The two groups performed their respective analyses independently, and several differences exist in the methods and models used.

B&W used the PDQ07 code, and PNL used the ALTHAEA code to perform the reactor depletion calculations. Other differences included different homogenization techniques used by the two groups to define the regions surrounding the core, and different fuel management schemes assumed to define the fuel zones in the core. As expected, these differences led to somewhat different results. Nevertheless, the combination of results from the two groups provides a basis for making an independent assessment of the range of U_3O_8 savings achievable by implementing the radial blanket and Zircaloy baffle features.

B&W's calculations indicated that substitution of Zircaloy-4 for stainless steel in the core baffle improves uranium utilization by about 1.4% when no radial blanket is present, and by 0.7% when a natural UO_2 blanket surrounds the core. PNL's results for the corresponding cases were 1.8% and 0.7%.

B&W's calculations indicated that a four-inch-thick natural UO_2 radial blanket improves uranium utilization by about 5%. PNL's corresponding calculations indicated an improvement of about 2.5%. This rather large difference is due largely to different assumptions concerning the fuel management scheme in the core, and illustrates the point that the uranium savings resulting from use of a radial blanket are sensitive to the fuel management scheme assumed. Furthermore, this leads to the conclusion that an estimate of the maximum uranium utilization improvement achievable would require a careful and rigorous analytical investigation to develop fuel management strategies that are reoptimized to maximize the benefits of a radial blanket. Neither of the fuel management strategies used in this study was optimized, although that used by B&W did tend to enhance the benefits of a blanket more than that used by PNL.

B&W's calculated savings from a combination of Zircaloy baffle and natural uranium radial blanket were 5.7%, and PNL's were 3.2%. More rigorous calculations that represent the peripheral region with greater detail and use realistic and optimized fuel management strategies would be expected to change these numbers somewhat. An optimization of the blanket lattice would also be expected to affect the estimated uranium savings. Nevertheless, although not optimized and although based on a simplified one-dimensional model, these results provide an indication of the approximate range of U_3O_8 savings that could be achieved by using the proposed design features.

In addition to the natural uranium case, PNL's calculations also included depleted UO_2 (0.2% U-235) and ThO_2 blankets. The cumulative uranium savings from the three blanket materials are shown in Figure B.1. Natural uranium provides the greatest savings, followed in order by depleted uranium and thorium. B&W did not calculate core depletions using these alternate blankets. However, on the basis of B&W's calculated values of k-infinite shown

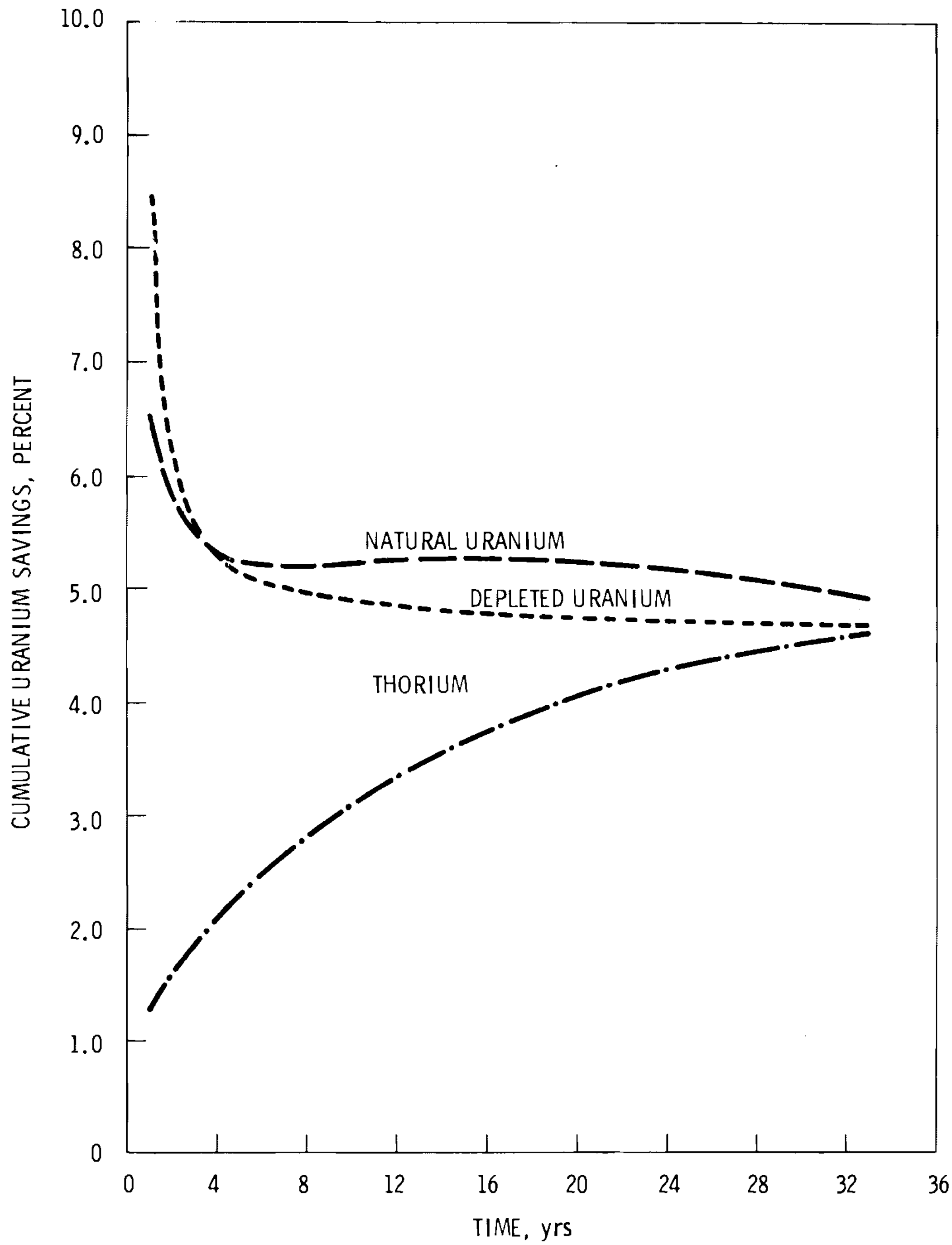

FIGURE B.1. Cumulative Uranium Savings Calculated for Various Blankets in Figures 4.2-9 and 4.2-12 of Reference 7, it can be concluded that B&W is in qualitative agreement with PNL concerning the relative uranium savings achievable from the three blanket materials.

Figure B.1 also illustrates the fact that it takes a much longer residence time for the thorium blanket to approach the cumulative uranium savings of the uranium blankets. In the preliminary engineering study the blanket residence time was identified as an item of uncertainty. In particular, there is concern that blanket assemblies may not be able to withstand residence times in the reactor as long as the 30 years assumed for this study. Should the residence time of blanket assemblies be limited, the thorium blanket would be even less attractive relative to uranium blankets. On the other hand, the resource savings resulting from the use of uranium blankets would not be adversely affected if the blanket residence time is limited.

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