I-954

213

ORNL/SUB-7537/15
GEAP-24965
CLASS I
JUNE 1980

MASTER

BWR URANIUM UTILIZATION IMPROVEMENT POTENTIAL

P. WEI
R. L. CROWTHER
L. E. FENNERN
P. J. SAVOIA
S. R. SPECKER
R. M. TILLEY
D. B. TOWNSEND
R. A. WOLTERS

GENERAL ELECTRIC
DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

ORNL/SUB--7537/15

DE82 007498

ORNL/SUB-7537/15

GEAP 24965

CLASS I

JUNE 1980

BOILING WATER REACTOR URANIUM UTILIZATION
IMPROVEMENT POTENTIAL

P WEI, RL CROWTHER, LE FENNERN, PJ SAVOIA, SR SPECKER,
RM TILLEY, DB TOWNSEND, RA WOLTERS

APPROVED:

PJ Savoia
PJ Savoia, Manager
Nuclear Design Technology

APPROVED:

RL Crowther
RL Crowther, Manager
Reactor Physics Technology

APPROVED:

JE Wood
JE Wood, Manager
Core Methods and
Analysis

Subcontract No. 7537

DISCLAIMER

This book was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

PREPARED BY
GENERAL ELECTRIC COMPANY
NUCLEAR ENERGY DIVISION
SAN JOSE, CALIFORNIA 95124
FOR
OAKRIDGE NATIONAL LABORATORY
OAKRIDGE, TENNESSEE 37830
DEPARTMENT OF ENERGY
CONTRACT NO. W-7405-eng-26

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

N O T I C E

This report was prepared by the General Electric Company under a subcontract with Union Carbide Corporation (No. 7537) for the account of DOE. No person acting on behalf of DOE, Union Carbide Corporation or General Electric Company will:

- (1) Make any warranty or representation, express or implied, with respect to the accuracy, completeness or usefulness of the information contained in this report, or that information, apparatus, method or process disclosed in this report may infringe privately owned rights, or;*
- (2) Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.*

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

*Printed in the United States of America. Available from
National Technical Information Service
US Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
NTIS Price codes - Printed Copy: A08; Microfiche A01*

BOILING WATER REACTOR URANIUM
UTILIZATION IMPROVEMENT POTENTIAL

A B S T R A C T

This report documents the results of design and operational simulation studies to assess the potential for reduction of BWR uranium requirements. The studies were the basis for BWR USA input to the International Fuel Cycle Evaluation (INFCE) and the Nonproliferation Alternative Systems Assessment Program (NASAP). The impact of the improvements on separative work requirements and other fuel cycle requirements also were evaluated. The emphasis was on analysis of the improvement potential for once-through cycles, although plutonium recycle also was evaluated.

The improvement potential was analyzed for several design alternatives including axial and radial natural uranium blankets, low-leakage refueling patterns, initial core enrichment distribution optimization, reinsert of initial core discharge fuel, preplanned end-of-cycle power coastdown and feedwater temperature reduction, increased discharge burnup, high enrichment discharge fuel rod reassembly and reinsert, lattice and fuel bundle design optimization, coolant density spectral shift with flow control, reduced burnable absorber residual, boric acid for cold shutdown, six-month subcycle refueling, and applications of a once-through thorium cycle design and plutonium recycle. Potential research, development and demonstration requirements were estimated for these design alternatives.

All of the improvement alternatives evaluated resulted in reduction of BWR uranium and separative work requirements, with the exception of reduction of fuel weight and an increase in lattice H/U ratio or a reduction in effective pellet density (through annular pellets or sphere pac fuel) which increased uranium and SWU resource requirements and depreciated performance when combined with extended burnup and other improvements. An optimal set of combined BWR improvements was selected and evaluated. Some technological and performance advantages were observed for the application of combinations of improvements as compared to application of individual improvements.

It was concluded that the potential for reduction of BWR equilibrium core uranium requirements in once-through fuel cycles was relatively large. Approximately 10 to 20 percent reductions in uranium requirements appear achievable in the short term, with potential for 20 to 30 percent reductions in the next decade. Three-dimensional coupled nuclear-thermal-hydraulic simulations of a combined set of design improvements indicated uranium and separative work savings of 28 and 27 percent, respectively, for improvements that appear to be retrofittable to both existing and future BWRs. It was observed that the incentive for application of the improvements was increased for 18 month as compared to 12 month refueling cycles. The appreciable SWU and U₃₀₈ savings from application of the identified improvement alternatives should result in major fuel cycle cost reductions for the BWR.

An improved Pu recycle design which utilized some of the same improved design features reduced uranium and separative work requirements by 45 and 44 percent, respectively. Thus, it was concluded that, although the BWR once-through cycle could be improved appreciably, significant additional uranium and SWU savings could be realized from implementation of Pu recycle.

T A B L E   O F   C O N T E N T S

	<u>PAGE</u>
ABSTRACT	iii
1. SUMMARY & CONCLUSIONS	1-1
1.1 Short Term Improvement Potential	1-1
1.2 Longer Range Improvement Potential	1-4
1.3 Plutonium Recycle Improvement Potential	1-4
1.4 Summary of Improvement Potential	1-4
1.5 RD&D Requirements	1-6
1.6 Conclusions	1-6
2. INTRODUCTION, PURPOSE & SCOPE	2-1
3. REFERENCE BWR CHARACTERISTICS	3-1
3.1 General Description	3-1
3.2 Core & Fuel Designs	3-1
4. CURRENT IMPROVED BWR FUEL & FUEL CYCLE DESIGNS	4-1
4.1 Natural Uranium Axial Blankets	4-1
4.2 Three Enrichment Initial Core & Radial Blankets	4-1
4.3 Improved Low Leakage Refueling Patterns	4-5
4.4 Lattice Geometry	4-10
5. ANALYSIS METHODS	5-1
5.1 Simplified Scoping Methods	5-1
5.2 Fuel Bundle Design & Local Nuclear Property Methods	5-2
5.3 Three Dimensional Simulation Methods	5-3
6. NUCLEAR, THERMAL, FUEL MECHANICAL & LICENSING CONSTRAINTS	6-1
6.1 MLHGR or Peak kW/ft	6-1
6.2 MCPR	6-2
6.3 Thermal/Mechanical Performance	6-3
6.4 Coupled Nuclear-Thermal-Hydraulic Stability	6-4
6.5 Loss-of-Coolant Accident (LOCA)	6-5
6.6 Cold Shutdown Margin	6-6

T A B L E   O F   C O N T E N T S
(continued)

	<u>PAGE</u>
7. SURVEY OF INDIVIDUAL ONCE-THROUGH IMPROVEMENT ALTERNATIVES	7-1
7-1 Alternatives Considered & Bases for Estimating Effects of Individual Improvements	7-1
7-2 Blankets, Improved Low-Leakage Refueling Patterns, Initial Core Optimization, Reinsert of Initial Core Discharge Fuel	7-2
7-3 Preplanned EOC Feedwater Temperature Reduction (FWTR) & Power Coastdown	7-7
7-4 Increased Burnup	7-15
7-5 High Enrichment Discharge Fuel Rod Reassembly/ Reinsert	7-21
7-6 Coolant Density Spectral Shift Control	7-27
7-7 Lattice & Fuel Bundle Design Optimization	7-32
7-8 Reduced Burnable Absorber Residual	7-38
7-9 Boric Acid Control	7-42
7-10 Six-Month Sub-Cycle Fast Refueling	7-46
7-11 Once-Through Thorium Cycle	7-50
8. COMBINED IMPROVEMENT ALTERNATIVE	8-1
8.1 Assumptions, Criteria, Methods of Approximation	8-1
8.2 Alternatives Considered & Bases for Selecting Combinations of Improvements	8-2
8.3 3-D Improvement Evaluations for High Enrichment Bundle	8-3
8.3.1 Extended Burnup & Reduced Gadolinia Residual (Case A)	8-5
8.3.2 Extended Burnup, Reduced Gadolinia Residual, Spectral Shift, FWTR, Coastdown (Case B)	8-5
8.3.3 Reconstitution & Reinsert of High Enrichment Discharge Fuel Rods Plus Other Improvements (Case C)	8-5
8.4 Reduced Pellet Density, Increased H/U Ratio	8-7

TABLE OF CONTENTS

(continued)

	<u>PAGE</u>
8.5 Optimal Combined Improvements	8-11
8.5.1 Assumptions, Criteria, Methods of Approximation	8-11
8.5.2 Resource Requirements for 12 Month Cycle	8-12
8.5.3 Resource Requirements for 18 Month Cycle	8-16
9. PLUTONIUM RECYCLE	9-1
9.1 BWR Pu Recycle Design Bases & Criteria	9-1
9.2 Mixed Oxide Fuel Experience Base for BWRs	9-5
9.3 Pu Recycle Improvement Impact Analysis	9-8
9.4 Once-Through Cycle Improvements Considered in Plutonium Recycle Evaluations	9-11
9.5 Current Improved Pu Recycle	9-15
9.6 Extended Burnup Pu Recycle	9-18
9.7 Plutonium Recycle Combined with Flow Control Spectral Shift	9-20
9.8 Pu Recycle Combined with EOC Feedwater Temperature Reduction & Power Coastdown	9-21
9.9 Best Case Pu Recycle Combined with Once-Through Cycle Improvements	9-22
9.10 Pu Recycle Summary & Conclusions	9-23
10. INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION (INFCE) AND NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM (NASAP) DATA	10-1
10.1 Assumptions, Criteria, Methods of Approximation	10-1
10.2 INFCE Fuel Cycle Data	10-1
10.3 NASAP Fuel Cycle Data	10-4
10.4 Decay Heat & Dose Rate Data as a Function of Burnup	10-4
11. REFERENCES	11-1

LIST OF ILLUSTRATIONS

FIGURE	TITLE	PAGE
3-1	Direct Cycle Reactor System	3-2
3-2	BWR Reactor Assembly	3-3
3-3	BWR/6 Bundle Cross Section, Typical Enrichment Distribution	3-4
4-1	Axial Blankets	4-2
4-2	Measured Cold Critical Axial Activation (Flux) Shapes at Quad Cities 1 Reactor	4-3
4-3	BWR/6 Initial Core Loading, Quarter Core Symmetric	4-4
4-4	Optimum Reactivity Profiles	4-6
4-5	Effect of Three-Enrichment Initial Core on Uranium Utilization & Refueling Batch Size	4-7
4-6	Multi-Enrichment Initial Core Loading - Hatch 2	4-8
4-7	Typical Low Leakage Refueling Pattern	4-9
6-1	Stable Steady Power Operation	6-4
6-2	Stable Power Increase	6-4
7-1	Fuel Utilization Improvements from Four-Enrichment Initial Core Design	7-4
7-2	Cycle Extension	7-8
7-3	Effect of Feedwater Temperature Reduction on Steam Void Axial Distribution	7-9
7-4	Cycle Extension	7-11
7-5	BWR Coastdown EOC Power Levels	7-13
7-6	BWR EOC Power Coastdown Experience	7-14
7-7	Reload Cycle Length Flexibility-Completed Cycles	7-16
7-8	Reload Cycle Length Flexibility-Current Cycles	7-17
7-9	Approximate BWR Uranium Ore Utilization for Equilibrium Cycle Operation (Increased Enrichment)	7-19
7-10	General Refueling Strategy	7-23
7-11	Typical BWR Power Flow Map	7-28
7-12	Reactivity vs. Burnup for Reduced Density Fuels	7-32a
7-13	Equal Energy Plot Comparing Reduced Density Fuels	7-34
7-14	Reactivity Effects of Lattice Geometry Changes	7-35

LIST OF ILLUSTRATIONS
(continued)

<u>FIGURE</u>	<u>TITLE</u>	<u>PAGE</u>
7-15	Typical Pre-Blanket Lattice	7-51
7-16	Blanket Assembly Breeding Ratio vs. Burnup	7-52
7-17	Infinite Lattice Neutron Multiplication Factors vs. burnup	7-53
7-18	K-Infinite Versus Exposure for Reference UO ₂ and Pre-Blanket Designs	7-59
7-19	Maximum Local (P/A) Power vs. Exposure for Reference UO ₂ and Pre-Blanket Designs	7-60
7-20	Dynamic Void Reactivity Coefficient for UO ₂ and Pre-Blanket Designs	7-62
8-1	Effects of Pellet Density, Burnup on Reactivity Slope	8-10
10-1	Decay Heat Generation vs. Cooling Time per Fuel Bundle	10-1
10-2	Dose Rate vs. Cooling Time (Receiver 1 meter Away in H ₂ O) Per Fuel Bundle	10-2
10-3	Dose Rate vs. Cooling Time (Receiver 1 meter away in air) Per Fuel Bundle	10-3

L I S T   O F   T A B L E S

<u>TABLE</u>	<u>TITLE</u>	<u>PAGE</u>
1-1	BWR Uranium Utilization Improvement Potential Relative to Current Operating Reactors	1-1
1-2	Current GE BWR Operating Experience with the Following Uranium Utilization Improvement Alternatives	1-3
1-3	Summary of Improvements from Best Combined 3-D Multi Cycle Evaluation	1-5
1-4	Optimal Combination Of BWR Improvements	1-7
1-5	Summary of Expected RD&D Requirements for Once-Through Fuel Cycle Improvements	1-8
1-6	Potential Development & Demonstration Requirements for BWR Fuel Cycle Improvement	1-9
3-1	Evolution of the General Electric BWR	3-5
3-2	Reference Reactor Core & Fuel Assembly Description	3-6
4-1	Current Improved Reactor Core & Fuel Assembly Descriptions	4-11
7-1	Preliminary Results on Equilibrium Cycle Bundle Reassembly/Reinsert Evaluation	7-24
7-2	Savings for BWR Spectral Shift vs. Cycle Applied	7-29
7-3	6 Month Sub-Cycle Fast Refueling 3-D Multicycle Evaluation	7-47
7-4	Material Flow in a 748 Bundle, 1100 MWe BWR/6 Once- Through Thorium Cycle Example	7-56
7-5	ThO ₂ Characteristics which may Improve Thermal Mechanical Performance/Lifetime	7-63
8-1	Summary of Combined Improvements Considered in 3-D Evaluation	8-4
8-2	Comparison of Combined 3-D Evaluations for 3.65 w/o U235 Bundle Enrichment	8-6
8-3	3-D Evaluations of Reduced Pellet Density	8-8
8-4	Summary of Twelve Month Cycle Optimum Combined Improvements	8-13
8-5	Summary of Within Cycle, Combined Improvement Incremental Effects on Equilibrium Cycle Energy	8-15
8-6	Summary of Optimal 18 Month Cycle Combined Improvements	8-17

LIST OF TABLES
(continued)

<u>TABLE</u>	<u>TITLE</u>	<u>PAGE</u>
9-1	GE BWR Mixed Oxide Fuel Experience Base	9-6
9-2	Preliminary Impact Analysis of Once-Through Fuel Cycle Options on BWR Pu Recycle Design	9-9
9-3	Summary of 3-D Multicycle Pu Recycle Analyses	9-13
9-4	Summary of Pu Recycle U_3O_8 & SWU Requirements	9-16
9-5	Summary of Discharge Fissile Material for Once-Through & Pu Recycle Alternatives	9-17
9-6	Equilibrium Cycle U & SWU Requirements for Pu+U Recycle Combined with Once-Through Cycle Improvements	9-19
10-1	General Reactor Performance Characteristics	10-1
10-2	INFCE BWR Fuel Cycle Information	10-3
10-3	INFCE, NASAP Fuel Cycle U_3O_8 & SWU SUMMARY	10-5

BOILING WATER REACTOR URANIUM UTILIZATION IMPROVEMENT POTENTIAL

1. SUMMARY & CONCLUSIONS

There is appreciable short and long range potential for reduction of Boiling Water Reactor (BWR) uranium ore* requirements in once-through fuel cycles by use of the design alternatives and improvements summarized in Table 1-1. These alternatives and others were considered in a DOE/ORNL funded program, reported herein, which had the primary objective of evaluating and identifying the long and short range BWR once-through fuel cycle improvement potential, the potential problem areas, and the general Research, Development and Demonstration required to bring the BWR fuel cycle improvements to fruition. BWR plutonium recycle also was evaluated to a limited extent, making use of available Pu recycle fuel bundle designs to provide bases for comparison with the once-through cycle alternatives. This report is a summary of the work conducted and the primary conclusions from the DOE/ORNL funded program.

Detailed 3-D coupled nuclear-thermal-hydraulic fuel cycle evaluations were used, where feasible, to evaluate the alternatives. The effects of refueling cycle length also were considered in the evaluations.

1.1 SHORT TERM IMPROVEMENT POTENTIAL

The short term BWR once-through cycle potential is in the range of 10 to 20 percent improvement in uranium utilization relative to designs with conventional "one-in-four distributed" or "scatter" refueling patterns without axial or radial blankets, at burnups of approximately 28 MWd/Kg. Designs now being introduced in the U.S. incorporate axial natural uranium blankets, improved low-leakage refueling patterns and initial core optimization, and limited axial gadolinia shaping to realize a significant short term reduction in both uranium and separative work requirements. Reinsert of initial core discharge bundles also has been applied in about one-half of GE-BWR plants to conserve uranium. End-of-cycle power coastdown is being applied to provide fuel cycle flexibility in a relatively large number of plants and is conserving uranium, although this alternative is not being applied to its full potential. Spectral shift and cycle extension with flow control, multi-enrichment initial cores, radial natural uranium blankets, and bundle repair have been applied to a limited extent and have considerable short range potential for further fuel cycle improvement. Table 1-2 is a summary of the current operating experience with uranium utilization improvement alternatives.

* Throughout the report "uranium ore" is used synonymously with U₃O₈ or yellow cake.

TABLE 1-1
BWR URANIUM UTILIZATION IMPROVEMENT POTENTIAL
RELATIVE TO CURRENT OPERATING REACTORS

	<u>URANIUM SAVINGS</u>
● BLANKETS, IMPROVED LOW-LEAKAGE OPTIMIZATION PATTERNS, INITIAL CORE OPTIMIZATION, REINSERT OF INITIAL CORE DISCHARGE FUEL	7 - 12%
● PREPLANNED EOC FEEDWATER TEMPERATURE REDUCTION & POWER COASTDOWN (TO 80%)	4 - 8%
● INCREASED BURNUP (WITH INCREASED ENRICHMENT)	5 - 15%
● HIGH ENRICHMENT DISCHARGE FUEL ROD REASSEMBLY/REINSERT	4 - 10%
● LATTICE & FUEL BUNDLE DESIGN OPTIMIZATION	0 - 5%
● COOLANT DENSITY SPECTRAL SHIFT CONTROL	4 - 8%
● ONCE-THROUGH THORIUM CYCLE	4 - 10%
● REDUCED GADOLINIA RESIDUAL	1 - 3%
● BORIC ACID CONTROL	2 - 4%
● SIX MONTH SUB-CYCLE FAST REFUELING	5 - 15%
● Pu RECYCLE	35 - 45%

BEST COMBINATION OF ONCE-THROUGH IMPROVEMENTS FOR BWR/6

URANIUM SAVINGS	28%
SEPARATIVE WORK SAVINGS	27%

BEST COMBINATION OF Pu RECYCLE IMPROVEMENTS

URANIUM SAVINGS	45%
SEPARATIVE WORK SAVINGS	44%

TABLE 1-2

CURRENT GE BWR OPERATING EXPERIENCE WITH THE
FOLLOWING URANIUM UTILIZATION IMPROVEMENT ALTERNATIVES *

● CYCLE EXTENSION WITH HIGH FLOW	1 PLANT
● EOC FEEDWATER TEMPERATURE REDUCTION	2 PLANTS
● EOC POWER COASTDOWN	18 PLANTS
● MULTI-ENRICHMENT INITIAL CORE	1 PLANT, FUTURE INITIAL CORES
● AXIAL NATURAL URANIUM BLANKETS	ALL RELOAD FUEL, NEW INITIAL CORES
● RADIAL NATURAL URANIUM BLANKETS	1 PLANT, FUTURE INITIAL CORES
● LOW-LEAKAGE REFUELING PATTERNS	ALL PLANTS
● AXIAL GADOLINIA SHAPING	ALL INITIAL CORES, SEVERAL RELOAD CORES
● RECONSTITUTION OR BUNDLE REPAIR	
7x7 DESIGN	~ 700 BUNDLES
8x8 DESIGN	46 BUNDLES
● REINSERT OF DISCHARGE BUNDLES (WITHOUT RECONSTITUTION)	ABOUT ½ OF PLANTS

---

*Experience as of December 1979.

1.2 LONGER RANGE IMPROVEMENT POTENTIAL

Longer range BWR potential for BWR once-through cycle improvement relative to the reference design is in the range of 20 to 30 percent reduction in uranium ore requirements. Extended burnup, high enrichment discharge fuel rod reconstitution and reinsert, expanded spectral shift with flow control, once-through thorium cycle applications, lattice and fuel bundle design optimization, and reduced gadolinia residual are the primary longer range alternatives that were considered. However, there is potential for considerable creativity and the worth of the longer range alternatives is much less certain than the shorter range improvements. Within the limited resources of this evaluation, it was not possible to consider other potential BWR once-through fuel cycle improvements.

Over the past 20 years, other design improvements have been identified for the BWR which would reduce uranium ore and separative work requirements by reducing neutron losses, increasing conversion ratios, optimizing H/U ratio, or reducing the fissile content of the discharge fuel. Some of these improvements were not evaluated in this study because of resource limits. For example, potential exists to improve BWR fuel cycles by shifting the neutron spectrum with axial power distribution control.²⁴ However, analysis of this alternative requires detailed simulation of control rod patterns within multiple refueling cycles. Many such improvements, innovations, and concepts have considerable potential for further BWR once-through cycle uranium utilization improvements beyond that identified in this report.

1.3 PLUTONIUM RECYCLE IMPROVEMENT POTENTIAL

The evaluations indicated that plutonium recycle also could be improved with a number of the retrofitable design alternatives such as axial blankets, spectral shift with flow control, low-leakage refueling patterns, and end-of-cycle power and feedwater temperature reduction. Relative to reference conditions, the improved Pu recycle alternative reduced uranium ore and separative work requirements by approximately 45 and 44 percent, respectively, and continued to indicate appreciable improvements over the best once-through cycle evaluated.

Existing Pu recycle designs which were developed in the mid-70's were used in the evaluation and these designs were not independently optimized. It is probable that optimization of the Pu recycle designs would result in greater uranium savings for the Pu recycle option.

1.4 SUMMARY OF IMPROVEMENT POTENTIAL

Table 1-3 is a summary of the reference, current improved, and best combined once-through cycles and the improved Pu recycle designs. The "current improved" design represents design features which are now being applied in new design or retrofit to existing reactors, but which will require a number of years to fully impact BWR uranium ore and separative work requirements.

TABLE 1-3
SUMMARY OF IMPROVEMENTS FROM BEST COMBINED 3-D MULTI CYCLE EVALUATION

	REFERENCE BWR	CURRENT IMPROVED BWR*	BEST COMBINED CASE STUDIECA	BEST COMBINED CASE WITH H Pu RECYCLE
AVERAGE CAPACITY FACTOR (%)	75	75	75	75
FRACTION OF BATCH REPLACED/REFUELING	0.24	0.24	0.23	0.20
NUMBER OF FRESH FUEL ASSEMBLIES/REFUELING	200	200	132	166
NUMBER OF REINSERTED FUEL ASSEMBLIES/REFUELING	---	---	60	---
FUEL PELLET DENSITY (% OF NOMINAL)	100	100	100	(a)
RELOAD ENRICHMENT (BUNDLE AVERAGE)	2.94	2.74	3.3	2.94
REFUELING INTERVAL (YEARS)	1.0	1.0	1.06	1.0
CONVERSION LOSS FRACTION	0.005	0.005	0.005	0.005
FABRICATION LOSS FRACTION	0.01	0.01	0.01	0.01
DISCHARGE FUEL BURNUP PRIOR TO REINSERTION (Mwd/MTU)	28439	28439	43000	36000
DISCHARGE BURNUP OF REINSERTED FUEL (Mwd/MTU)	---	---	49000	---
U ₃₀₈ REQUIREMENTS (ST/GWE)**				
INITIAL CORE	454	496	443	443
TRANSITION CYCLES	685	640	410	355
ANNUAL EQUILIBRIUM RELOAD	204	190	146	105
30 YEAR CUMULATIVE	6443	6051	4624	3528
SEPARATIVE WORK REQUIREMENTS (10 ³ SWU/GWE)				
INITIAL CORE	198	235	198	198
TRANSITION CYCLES	465	416	246	228
ANNUAL EQUILIBRIUM RELOAD	124	111	92	68
30 YEAR CUMULATIVE	3887	3490	2836	2194
PERCENT REDUCTION COMPARED TO REFERENCE CASE (%)				
U ₃₀₈ REQUIREMENTS**				
ANNUAL EQUILIBRIUM RELOAD	---	6.9	28.3	48.5
30 YEAR CUMULATIVE	---	6.1	28.2	45.2
SEPARATIVE WORK REQUIREMENTS				
ANNUAL EQUILIBRIUM RELOAD	---	10.5	25.6	45.2
30 YEAR CUMULATIVE	---	10.2	27.0	43.6

*Improved Initial Core, Axial and Radial blankets, Low-Leakage Fuel Management, and greater H/U (Design features being applied and offered in current designs).

**Tails Composition 0.2 W/O

Δ Current Improved Design plus EOC Temperature and Power Reduction, Spectral Shift with Flow Control, Extended Burnup and High Enrichment Discharge Fuel Rod Reassembly and Reinsert.

▽ All Features in Best Combined Once-Through except High Enrichment Discharge Fuel Rod Reassembly and Reinsert. Pu recycle Fuel is "Island" design. Pu recycle case was not optimized.

(a) MO₂ Fuel Rods utilize Annular Pellets.

GELP-1496

Table 1-4 is a summary of the optimal combinations of BWR improvements which was derived from the evaluations. All of these improvements appear to be retrofittable to current operating and future BWRs.

1.5 RD&D REQUIREMENTS

Research, Development and Demonstration (RD&D) requirements and associated approximate costs were estimated for the most promising improvement alternatives. In most cases the alternatives require additional analyses and design studies to define RD&D requirements. Data also are needed from measurements and demonstrations, reevaluation and modification of licensing criteria are needed, improved design and analysis methods must be developed and, in some cases, manufacturing improvements are needed.

Table 1-5 is a summary of the categories of RD&D requirements for the improvement alternatives. It is important to note that in most cases, additional more detailed analyses and design studies are needed to define the RD&D requirements. In all cases, data are needed from demonstrations and associated measurements and/or measurements on lead test assemblies to confirm predicted performance.

In a number of instances, improvements or modifications of licensing criteria, based on reevaluation of technical data, would increase uranium savings. The primary licensing criteria that have large impact on uranium ore requirements are summarized in Section 6.

Table 1-6 is a summary of the potential detailed RD&D requirements for the primary alternatives considered, including the estimated range of cost. The cost estimate range is relatively broad as a result of the current status of knowledge and the sensitivity of costs to plant specific details. Although the upper side of the RD&D cost estimate is large, this cost is small compared to potential utility fuel cycle cost savings.

1.6 CONCLUSIONS

Improvements available for introduction, or now being introduced in many BWRs reduce uranium ore requirements by 7 to 15 percent. Other improvements under consideration have potential of more than doubling these gains. Plutonium recycle, when combined with some of the improvement options, significantly increases uranium savings relative to the best once-through cycle. The current utility trend to increased cycle lengths between refueling tends to increase uranium ore requirements and will offset some of these gains. Several of the BWR design improvements introduced or being considered to improve uranium utilization also favorably affect thermal operating margins and reactor transient response. Many also are desirable independent of reprocessing. Such low-risk, "dual-incentive" improvements have highest potential for near term applications.

TABLE 1-4

OPTIMAL COMBINATION OF BWR IMPROVEMENTS

- OPTIMAL MULTI-BUNDLE ENRICHMENT INITIAL CORE WITH RADIAL NATURAL URANIUM BLANKETS AND WITH BUNDLE ENRICHMENTS OPTIMIZED TO MATCH EQUILIBRIUM CORE CONDITIONS;
- AXIAL NATURAL URANIUM BLANKETS;
- LOW LEAKAGE REFUELING PATTERNS WITH LAST CYCLE, LOWEST REACTIVITY FUEL ON PERIPHERY;
- SPECTRAL SHIFT USING FLOW CONTROL (90-105% FLOW);
- EOC FEEDWATER TEMPERATURE REDUCTION TO 250°F;
- EOC COASTDOWN TO 80% POWER;
- INCREASED RELOAD ENRICHMENT AND BURNUP;
- HIGH ENRICHMENT DISCHARGE FUEL ROD RECONSTITUTION AND REINSERT.

TABLE 1-5

SUMMARY OF EXPECTED RD&D REQUIREMENTS FOR ONCE-THROUGH FUEL CYCLE IMPROVEMENTS

	<u>Additional Analyses, Design Studies to Define RD&D Requirements</u>	<u>Data Needed From Measurements & Demonstrations</u>	<u>Licensing Criteria Improvements & Modifications</u>	<u>Design Analysis Methods Development</u>	<u>Fuel Manufacturing Improvements</u>
Increased Burnup	X	X	X	X	X
Improved Fuel, Control Man- agement	X	X		X	
Axial and Radial Blankets		X		X	
Coolant Density Control (Flow Control) Spectral Shift	X	X	X	X	
High Enrichment Discharge Fuel Rod Reassembly, Reinsert	X	X	X	X	X
Preplanned EOC Feedwater Temper- ature Reduction & EOC Power Coastdown		X		X	
Reduced Fuel Density & Lattice Optimi- zation	X	X	X	X	X
Once-Through Thorium Cycle	X	X	X	X	X
Boric Acid for Cold Shutdown	X	X	X	X	
Reduced Gadolinia Residual	X	X		X	
Six Month Fast Refueling Sub-cycles	X	X	X	X	

TABLE 1-6

POTENTIAL DEVELOPMENT & DEMONSTRATION REQUIREMENTS
FOR BWR FUEL CYCLE IMPROVEMENT

INCREASED BURNUP\$20-90M

PCI RESISTANT FUEL OR OTHER FUEL RELIABILITY IMPROVERS

FISSION GAS, CORROSION, MATERIALS MEASUREMENTS

IMPROVED FINE MOTION CONTROL ROD DRIVES

LONGER LIFE CONTROL BLADE DEMONSTRATION MEASUREMENTS

IMPROVED CHANNELS, SPACERS

NUCLEAR MEASUREMENTS ON LONG BURNUP CORE

INCREASED MANUFACTURING LIMITS ON PEAK PELLET ENRICHMENT AND BURNABLE ABSORBER CONTENT

SEGMENTED ROD IRRADIATIONS AND RAMP TESTS

LEAD TEST ASSEMBLIES (LTAs)

FUELS, ISOTOPIC, POWER AND BURNUP DISTRIBUTION MEASUREMENTS ON LTAs

OPTIMIZATION FOR SMALL REFUELING BATCH SIZE AND GREATER ENRICHMENTS TO MITIGATE ADVERSE POWER DISTRIBUTION AND COLD SHUTDOWN TRENDS AND TO MINIMIZE NEUTRON LEAKAGE AND FUEL INVENTORY

NUCLEAR AND FUELS DESIGN AND LICENSING METHODS IMPROVEMENTS

DEVELOPMENT OF IMPROVED FUEL DESIGNS

STABILITY AND TRANSIENT MEASUREMENTS

DEMONSTRATIONS IN POWER PLANTS

TABLE 1-6
(continued)

IMPROVED FUEL, CONTROL MANAGEMENT, AND BLANKETS

\$0.5-8M

IMPROVED NUCLEAR DESIGN OPTIMIZATION METHODS
(BUNDLE & CORE)

MANUFACTURING METHODS/STANDARDIZATION

NUCLEAR MEASUREMENTS (GAMMA SCANS, ISOTOPICS)
ON CORE WITH RADIAL AND AXIAL BLANKETS TO
CONFIRM PERFORMANCE

COOLANT DENSITY CONTROL (FLOW CONTROL) SPECTRAL SHIFT

\$0.5-20M

CLADDING TEMPERATURE LIMITS FOR BWR TRANSIENT CHF

LICENSING OF WIDER POWER/FLOW OPERATING MAP

FLOW INDUCED VIBRATION MEASUREMENTS

IMPROVED CONTROL SYSTEMS

STABILITY AND TRANSIENT MEASUREMENTS

CHANNEL, SPACER, INTERNALS, BYPASS FLOW
IMPROVEMENTS/MEASUREMENTS

RECIRCULATION AND JET PUMP AND LOOP IMPROVEMENTS

SOLID STATE OR MOTOR GENERATOR SET FLOW CONTROL
FOR PLANTS WITH VALVE FLOW CONTROL

STEAM SEPARATOR, DRYER MEASUREMENTS IMPROVEMENTS

DEMONSTRATION IN OPERATING PLANTS

MEASUREMENTS TO CONFIRM NUCLEAR PREDICTIONS
(GAMMA SCANS, CRITICALITY, ISOTOPICS, BURNUP)

HIGH ENRICHMENT DISCHARGE FUEL ROD REASSEMBLY/REINSERT

\$3.0-20M

INCREASED BURNUP CAPABILITY

LEAD TEST ASSEMBLIES (LTAs)

NUCLEAR, FUELS MEASUREMENTS TO CONFIRM PREDICTIONS

CHF AND LOCA MEASUREMENTS

IMPROVED NUCLEAR DESIGN METHODS

TABLE 1-5
(continued)

AUTOMATED QUALITY CONTROL, REASSEMBLY
LICENSING OF REASSEMBLY PROCESS
DEMONSTRATION IN OPERATING PLANTS

ONCE THROUGH THORIUM CYCLE

\$3.0-30M

LEAD TEST ASSEMBLIES (LTAs)
NUCLEAR, FUELS MEASUREMENTS ON LTAs
(GAMMA SCANS, ISOTOPICS)
MATERIAL PROPERTY MEASUREMENTS
PCI MEASUREMENTS
IMPROVED DESIGN METHODS
FULL SCALE REACTOR DEMONSTRATIONS
MANUFACTURING IMPROVEMENTS
NUCLEAR MEASUREMENTS TO CONFIRM PERFORMANCE

REDUCED BURNABLE ABSORBER RESIDUAL

\$1.0-5M

IMPROVED NUCLEAR DESIGN OPTIMIZATION METHODS
IMPROVED CONTROL BLADES
MANUFACTURING STANDARDIZATION
DEMONSTRATION AT OPERATING PLANTS
NUCLEAR MEASUREMENTS TO CONFIRM PERFORMANCE

BORIC ACID FOR CONTROL

\$10-90M

MATERIAL STUDIES, CLEAN-UP SYSTEMS DEVELOPMENT
COLD-TO-HOT STARTUP SYSTEMS DEVELOPMENT
NUCLEAR MEASUREMENTS
SAFETY AND INSTRUMENTATION SYSTEMS MODIFICATIONS
DEMONSTRATION AT OPERATING REACTOR

TABLE 1-6
(continued)

SIX MONTH SUB-CYCLE FAST REFUELING

\$2.0-20M

NRC PRELICENSING 18-24 MONTH CYCLE WITH 6
MONTH REFUELING

EQUIPMENT, PROCEDURES TO REDUCE REFUELING TIMES,
MINIMIZE STAFF REQUIREMENTS AND RADIATION
EXPOSURE

FUEL BUNDLE DESIGN REOPTIMIZATION

CORE AND FUEL MANAGEMENT OPTIMIZATION

MINIMIZATION OF INSPECTION AND MAINTENANCE
REQUIREMENTS DURING FAST REFUELING

DEMONSTRATION AT OPERATING REACTOR

NUCLEAR MEASUREMENTS TO CONFIRM PERFORMANCE

2. INTRODUCTION, PURPOSE & SCOPE

A primary purpose of this program was to complete a preliminary evaluation of the potential for improving BWR once-through fuel cycle uranium utilization. These evaluations were directed toward providing BWR inputs to the International Nuclear Fuel Cycle Evaluation (INFCE) Working Group 8 paper entitled, "Improved Once-Through Fuel Cycle for Light Water Reactors", and providing inputs to the report of the Non Proliferation Alternative Systems Assessment program (NASAP).

The scope of this Program was to:

1. Provide rough numerical estimates of the BWR potential for various uranium utilization improvement options.
2. Provide reference and approximate improved BWR fuel cycle design characteristics, to the extent feasible with available resources.
3. Evaluate combinations of the most promising BWR alternatives for once-through fuel cycle improvement to better estimate the net improvement potential. Develop a preliminary estimate of the fuel utilization savings from combinations of improvements which result in approximately 20 and 30 percent uranium utilization savings. Complete more detailed evaluations of combined applications of improvements, to the extent possible, and prepare the data required for the US INFCE paper for the combination of the most promising alternatives.
4. Estimate the practical difficulties in implementation and retrofit of the most promising alternatives based on available information and scoping studies. An accurate assessment of practical difficulties and mitigation measures will require detailed evaluations similar to those currently in progress in the DOE/TVA/GE program. Therefore, the scope of this program was to provide preliminary evaluations which to some extent were based on judgment and on simplified evaluations.
5. Estimate the Research, Development and Demonstration (RD&D) requirements to prove technical feasibility and gain NRC approval for the most promising uranium utilization alternatives and prepare preliminary scoping cost estimates and schedules. Estimate the implementation schedules with margin for uncertainties at the present stage of evaluation.
6. Evaluate the impact of once-through fuel cycle improvements on the plutonium recycle mode and complete evaluations that are feasible within available resources.
7. Update the preceding to respond to DOE and ORNL comments, to the extent feasible within available resources.

An overall purpose of the preliminary evaluations was to define incentives and potential for improving BWR once-through fuel cycles. The alternatives under consideration have different development requirements and implementation schedules. Detailed 3-D coupled nuclear-thermal-hydraulic fuel cycle evaluations, transient and safety evaluations and thermal-mechanical fuels evaluations will be required to assess the reliability of the preliminary estimates of development requirements for the alternatives. Although some such detailed evaluations and the consideration of development requirements were completed in the preliminary studies, considerably greater detail will be required to provide comprehensive cost estimates and schedules.

3. REFERENCE BWR CHARACTERISTICS

3.1 GENERAL DESCRIPTION

A 1270 MWe General Electric BWR/6 plant and core design was selected as the reference design for assessing the uranium utilization improvement potential of the various design and operational alternatives. The reference BWR/6 core design is an early design with fuel cycle characteristics typical of most of the BWRs now in operation. Although current BWR/6 designs, none of which have started up, incorporate fuel cycle improvements, the reference selected provides a basis for assessing the relative worth of improvements such as axial blankets that are not yet fully impacting uranium utilization savings in all BWRs.

The General Electric BWR is a direct cycle nuclear steam generating system (Figure-3-1), including auxiliary systems to accommodate the operational and safety requirements of the nuclear reactor, and necessary controls and instrumentation. The nuclear core and internal structures within the pressure vessel are shown in Figure 3-2. The fuel assembly design is shown in Figure 3-3. Water is circulated through the reactor core, producing saturated steam which is separated from recirculation water, dried in the top of the vessel, and directed to the steam turbine-generator. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization. Jet pumps are utilized to provide forced circulation flow through the core. A recirculation flow control system is utilized for load following and reactor power regulation. Automatic control systems maintain reactor vessel water level, regulate steam pressure, follow electrical demand and regulate overlapping safety and core cooling systems.

Table 3-1 is a brief summary of different GE-BWR product lines.

3.2 CORE AND FUEL DESIGNS

The reference reactor is rated at 3800 megawatts thermal (Mwt) with 848 fuel assemblies in the core. The core average power density at rated power is 51 kw/l, and the reactor operates with a core average steam void fraction of approximately 0.43. The fuel assembly is an 8X8 design with one internal water tube. The fuel rods are made up of sintered UO₂ pellets. Table 3-2 gives parameters on the reference reactor core and fuel assembly designs.

FIGURE 3-1

DIRECT CYCLE REACTOR SYSTEM

Figure 3-2 Reactor Assembly

Figure 3.3 BWR/6 Bundle Cross Section, Typical Enrichment Distribution

TABLE 3-1EVOLUTION OF THE GENERAL ELECTRIC BWR

<u>PRODUCT LINE NUMBER</u>	<u>YEAR OF INTRODUCTION</u>	<u>CHARACTERISTIC PLANTS</u>
BWR/1	1955	● DRESDEN 1, BIG ROCK POINT, HUMBOLDT BAY, KRB --INITIAL COMMERCIAL BWRS --FIRST INTERNAL STEAM SEPARATION
BWR/2	1963	● OYSTER CREEK --PLANTS PURCHASED SOLELY ON ECONOMICS --LARGE DIRECT CYCLE --FLOW CONTROL
BWR/3	1965	● DRESDEN 2 --FIRST JET PUMP APPLICATION --IMPROVED ECCS; SPRAY & FLOOD
BWR/4	1966	● BROWNS FERRY --INCREASED POWER DENSITY (21%)
BWR/5	1969	● ZIMMER --IMPROVED ECCS SYSTEMS --VALVE FLOW CONTROL --REDUCED FUEL DUTY (13.4 KW/FT)
BWR/5	1972	● BWR/6 --IMPROVED JET PUMPS & STEAM SEPARATORS --ADDED FUEL BUNDLES, INCREASED OUTPUT --IMPROVED ECCS PERFORMANCE --IMPROVED LICENSABILITY

TABLE 3-2REFERENCE REACTOR CORE & FUEL ASSEMBLY DESCRIPTIONPOWER PLANT PERFORMANCE

TOTAL THERMAL POWER	MWt	3800
ELECTRICAL POWER, NET	MWe	1270
THERMAL EFFICIENCY	%	33.4

REACTOR PARAMETERS

CORE VOLUME	ℓ	75045
CORE HEIGHT	ℓc	3.76
CORE EFFECTIVE DIAMETER	ℓc	5.01
CORE POWER DENSITY	kw/ℓ	51
SPECIFIC POWER	kw/kg U	23.77
COOLANT FLOW RATE	Mg/sec	14.8
COOLANT INLET TEMPERATURE	C	278.3
CORE AVERAGE VOID FRACTION	%	43.1
CORE PRESSURE	bar	73
NUMBER OF FUEL ASSEMBLIES		848
TOTAL URANIUM LOADING	MT	159.9

FUEL ASSEMBLY DATA

NUMBER OF RODS	m	63 (8X8 -1)
ACTIVE FUEL LENGTH	cm	3.76
FUEL ROD PITCH	cm	1.615
ROD OUTER DIAMETER	cm	1.252
CLADDING THICKNESS	cm	0.086
FUEL PELLET OUTSIDE DIAMETER	cm	1.057
TOTAL URANIUM LOADING	kg	188.6

4. CURRENT IMPROVED BWR FUEL & FUEL CYCLE DESIGNS

BWR design improvements, including axial and radial blankets, improved low-leakage refueling patterns that locate last cycle fuel on the reactor periphery, and improved initial core designs have been introduced recently! Development of these improvements was based on detailed 3-dimensional nuclear-thermal-hydraulic multicycle simulations, coupled with Monte Carlo and other benchmark analyses, where necessary. The Hatch 2 reactor which started up in 1979 is the first BWR to fully apply the improved design features. All GE reload fuel is being retrofit with axial blankets and low-leakage refueling patterns are being applied to the maximum extent feasible. The recent "current improved" design features resulted in reduction of BWR uranium ore and separative work requirements by approximately 7 and 10 percent, respectively, as summarized in Table 1-3.

4.1 NATURAL URANIUM AXIAL BLANKETS

Natural uranium approximately 6" long, has been incorporated in the top and bottom of each fuel assembly to reduce neutron leakage and improve void reactivity coefficients. Figure 4-1 is a plot of the average axial fuel enrichment distribution at approximate fuel discharge conditions with and without use of natural uranium axial blankets. In the cold condition, the core is most reactive in the top (Figure 4-2), and thus, the top of the reactor core strongly affects cold "one-stuck control blade" shutdown margin. The low reactivity of the natural uranium allows the active fuel length to be increased without any change in the length of the control blade. Axial blanket fuel assemblies are currently being used routinely in domestic BWR initial and reload cores.

4.2 THREE ENRICHMENT INITIAL CORE & RADIAL BLANKETS

Improved initial core designs utilizing three different bundle average enrichments are being applied to all new BWR cores to improve uranium utilization and to flatten radial power distributions. Figure 4-3 illustrates a typical core layout for BWR/6 class designs. The intermediate and high enriched fuel assemblies used in these evaluations had enrichments of 1.8 and 2.2 wt. percent, respectively. Natural uranium blanket assemblies are located on the core periphery to reduce neutron leakage and the negative void reactivity coefficient. One BWR fuel assembly width, which is about equal to one fast-neutron mean-free-path, was determined to be near optimum thickness (6") for the blanket.

Figure 4-1. Axial Blankets

Figure 4-2. Measured Cold Critical Axial Activation (Flux) Shapes at Quad Cities 1 Reactor

- ☐ HIGH ENRICHED BUNDLES
- ☒ INTERMEDIATE ENRICHED BUNDLES
- ☒ NATURAL ENRICHED BUNDLES

FIGURE 4-3. BWR/6 INITIAL CORE LOADING, QUARTER CORE SYMMETRIC

High-enrichment fuel assemblies are located just inside the peripheral assemblies to flatten radial power shape and raise radial blanket power, while maintaining reduced neutron leakage. The average enrichment of the core interior is selected to minimize the peak-to-average radial power of the three-zone core. This core design strategy is effective in maximizing the excess reactivity of the core while minimizing the radial power peaking. The optimization strategy developed is called the "maximum reactivity-minimum peaking" strategy and is characterized by the radial reactivity profiles shown in Figure 4-4.

The three-enrichment initial core also enables the fissile content of the first two discharge batches to be reduced relative to a single-enrichment initial core, thereby reducing the uranium requirements for the first two reload batches. This is illustrated by Figure 4-5 which summarizes the relative effects of three-enrichment and uniform initial cores on the refueling batch size and on the energy production per Kg of fuel loaded for each batch discharged.

Figure 4-6 is the core configuration of the Hatch 2 reactor, illustrating the layout of initial core fuel assemblies. This plant has operated for approximately two years.

4.3 IMPROVED LOW-LEAKAGE REFUELING PATTERNS

The maximum reactivity-minimum peaking strategy also is being applied to reload core designs. Highly exposed, low-reactivity, last cycle fuel is moved to the peripheral locations to form a low-leakage radial blanket. Relatively low-exposure, high-reactivity fuel is located just inside the peripheral assemblies to flatten the radial power distribution. The interior of the core is fueled in the conventional distributed load manner. This strategy has been adapted with success to a variety of reload core design situations.

Figure 4-7 is one equilibrium loading strategy that has been developed, using two types of fuel elements in the reload batch. The higher enrichment reload bundles are loaded inside the periphery and in the last cycle are located in central core regions. The lower enrichment reload bundles reside in the central core regions for four cycles and are moved to the periphery or blanket positions for the last cycle before discharge. The result is a general radial reactivity shape, similar to that shown in Figure 4-3, which both minimizes neutron leakage and flattens the radial power distribution.

Figure 4-4. Optimum Reactivity Profiles

Figure 4-5. Effect of Three Enrichment Initial Core on Uranium Utilization and Refueling Batch Size

Figure 4-6. Multienrichment Initial Core Loading

FIGURE 4-7

TYPICAL LOW-LEAKAGE REFUELING PATTERN

Each Cycle: Discharge Fuel: 4;D
 Fuel Moves: 3→4; C→D
 Change In Designation: 1→2, 2→3; A→B, B→C
 Fresh Fuel: 1 (Low Enrichment); A (High Enrichment)

Movement of fuel to low-power peripheral positions for the last cycle of irradiation and avoidance of the inward movement of fuel bundles, which have been irradiated in the steep power gradient peripheral positions, are further advantageous features that improve fuel reliability and simplify design analyses.

4.4 LATTICE GEOMETRY

The "current improved" design (Table 4-1) also has slightly smaller fuel rod diameter and a greater design H/U ratio than the reference design. Two large water rods are used in the center of the fuel assembly to reduce the peak-to-average fuel rod enrichments and reduce the magnitude of the negative steam void reactivity coefficients. The active fuel length is greater and the fission gas plenum volume is smaller, as was made possible by use of an upper natural uranium blanket. The water gap thickness between fuel assemblies is 13% greater and the control blade thickness is 24% greater than the reference design to further reduce steam void reactivity coefficients and increase control blade worth.

TABLE 4-1

CURRENT IMPROVED REACTOR CORE & FUEL ASSEMBLY DESCRIPTIONSPOWER PLANT PERFORMANCE

TOTAL THERMAL POWER	MWt	3800
ELECTRICAL POWER, NET	MWe	1270
THERMAL EFFICIENCY	%	33.4

REACTOR PARAMETERS

CORE VOLUME	ℓ	75045
CORE HEIGHT	m	3.81
CORE EFFECTIVE DIAMETER	m	5.01
CORE POWER DENSITY	kW/ℓ	51
SPECIFIC POWER	kW/kgU	24.58
COOLANT FLOW RATE	Mg/sec	14.8
COOLANT INLET TEMPERATURE	C	278.3
CORE AVERAGE VOID FRACTION	%	43.1
CORE PRESSURE	bar	73
NUMBER OF FUEL ASSEMBLIES		848
TOTAL URANIUM LOADING	MT	154.6

FUEL ASSEMBLY DATA

NUMBER OF RODS		62 (8X8 -2)
ACTIVE FUEL LENGTH	m	3.81
FUEL ROD PITCH	cm	1.615
ROD OUTER DIAMETER	cm	1.227
CLADDING THICKNESS	cm	0.081
FUEL PELLET OUTSIDE DIAMETER	cm	1.041
TOTAL URANIUM LOADING	kg	182.3

5. ANALYSIS METHODS

A combination of both sophisticated 3-D methods and simplified scoping methods were used in the analyses. In most cases comprehensive evaluation of licensing and design constraints was outside the scope of the evaluations.

5.1 SIMPLIFIED SCOPING METHODS

The fuel cycle requirements are normally specified in terms of an equilibrium cycle capacity factor and cycle length. The following energy balance equation relates the equilibrium fuel cycle management parameters:

$$fB_d = 365 \text{ LTS} \quad (5.1)$$

where f is the equilibrium fractional refueling batch size (fraction of fresh fuel in core) and the other parameters are defined as follows:

Parameters dependent on fuel and core design

B_d = Equilibrium average discharge burnup in MWD/MT

S = Core specific power, kwt/kgU

Parameters dependent on power system and refueling plan

L = Capacity factor

T = Equilibrium cycle length, years

For a specified L , T , B_d , and S , the equilibrium refueling batch fraction, f , is fixed.

If the rate of change of reactivity with fuel burnup is a constant (linear reactivity model), the equilibrium reload enrichment can be simply derived in terms of B_d , L , T , S , and f .

$$\begin{aligned} e &= \alpha + \beta (B_d + 365 \text{ LTS}) = \alpha + \beta B_d (1 + f) \\ &= \alpha + \beta (B_d + B) \end{aligned} \quad (5.2)$$

where

B = Cycle Average Burnup Accumulation

e = Average equilibrium reload enrichment, w/o U_{235}

In practice, the constants in Equation (5.2) are determined not from simple theory but from 3-D evaluations. β and α are fitted constants determined from 3-D multicycle evaluations. β is dependent on bundle design and α is dependent on core leakage. If the fuel cycle conditions are not too great, an extrapolation from previous 3-D analyses, Equation (5.2) provides a reasonably reliable scoping method for determining enrichment requirements. However, design variables that affect the conversion ratio and reactivity slope versus burnup or the neutron leakage, cause changes in α and β which must be reevaluated by detailed 3-D coupled nuclear-thermal-hydraulic fuel cycle analyses.

A one-dimensional fuel cycle model also was used in fuel cycle scoping analyses. The one dimensional model uses reactivity and isotopic data generated by the local fuel bundle nuclear property or lattice physics model and evaluates the effects of reloading patterns by use of a synthesis of one-dimensional neutron diffusion theory and superimposed interaction effects of two-dimensional fuel bundle loading patterns on reactivity and power distributions.

5.2 FUEL BUNDLE DESIGN & LOCAL NUCLEAR PROPERTY METHODS

BWR lattice physics methods^{2,3} consist of multigroup mixed one- and two-dimensional transport theory and transport corrected diffusion theory techniques. The epithermal spectrum is computed in 68 groups with determination of spatial weighting of the fuel, clad, and moderator and evaluation of the two dimensional Dancoff interaction between individual pins in the resolved resonance region. Resolved resonances are evaluated using the intermediate resonance (IR) approximation. The space-dependent thermal spectrum is computed by use of 30-group integral transport theory. Gadolinia rods are specially represented to include greater local spatial neutron spectrum detail and to permit space-dependent depletion calculations within the fuel pins.

Approximately 30 heavy nuclides and 30 individual fission products are evaluated as a function of burnup. The isotopic depletion equations are numerically integrated, including non-linear effects such as the change in neutron flux and spectrum with time and the change in resonance self-shielding with isotopic concentration.

BWR local nuclear property methods used in the evaluations are supported by continuing efforts to qualify the methods against benchmark analyses, such as Monte Carlo, and against benchmark critical experiments and detailed measurements in power reactors.⁴ In many of the evaluations, local nuclear property data from other design and operating data evaluations were used to minimize the computer resources required to accomplish the contract objectives.

5.3 THREE DIMENSIONAL SIMULATION METHODS

Three-dimensional coupled nuclear-thermal-hydraulic fuel cycle analyses, reactor core evaluations, and multicycle fuel cycle studies were performed using a three-dimensional coupled nuclear-thermal-hydraulic-fuel cycle simulator,^{5,6} which consists of coarse mesh, 1.5 group, steady state neutron diffusion theory coupled to steady state parallel channel thermal-hydraulics. The 3-D simulator was used to perform detailed three-dimensional fuel cycle evaluations as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables. A power-exposure iteration option was used to converge on consistent end-of-cycle target burnup and power distributions⁷ without following the detailed operating history within each cycle. This approach permits economic 3-D simulation of multiple operating cycles. The 3-D simulator is supported by continuing programs to qualify the model against detailed measurements and reactor data⁴ and to improve the accuracy of the nuclear determinations.

6. NUCLEAR, THERMAL, FUEL MECHANICAL AND LICENSING CONSTRAINTS

Nuclear, thermal, fuel mechanical and licensing constraints were lifted in some of the evaluations to make it possible to determine the long term potential for BWR fuel cycle improvements. Evaluations of the trends in performance relative to such constraints are important in determining RD&D requirements. There usually is a direct trade-off between fuel cycle improvements and thermal constraints. For example, if thermal margins are increased, the increased margin can be used to peak end-of-cycle neutron importance on the highest reactivity fuel, and to reduce end-of-cycle neutron leakage thereby increasing cycle energy and reducing uranium ore requirements. The full potential for fuel cycle improvements through thermal constraint relief was not evaluated fully in this study, but it is an important area for future increased uranium savings.

A number of the constraints used in current design have been conservatively derived and there is considerable potential for constraint revision through RD&D programs that quantify realistic influences of design variables on fuel reliability, transient performance, and reactor safety. Some of the more important constraints on current BWR design which have potential for are summarized in the following paragraphs.

6.1 MLHGR OR PEAK kw/ft

The Maximum Linear Heat Generation Rate, or the peak kw/ft, is determined by the overall reactor global peak-to-average power. Peak kw/ft determines peak central fuel temperature which affects the local release of fission products inside the cladding and the strain on the cladding from pellet-clad interaction. Local, axial and radial power distributions all contribute to the peak kw/ft. Current kw/ft limits have been conservatively selected to account for uncertainties and to provide for reliable fuel performance. Both greater knowledge of fuel performance fundamentals and trends versus design variables and improved design have potential to increase the operating peak kw/ft limit. Improved reactor measurement systems and operating reactor Process Computer nuclear methods which reduce uncertainty in the measured peak kw/ft and reduction in uncertainty in design nuclear methods have potential to permit decrease in the kw/ft design margin or increase in the peak kw/ft design basis. Such improvements would permit design and/or reactor operations improvements which would decrease uranium ore requirements through either extending fuel burnup or decreasing initial fuel enrichment requirements.

6.2 MCPR

MCPR, or Minimum Critical Power Ratio, is the power that a fuel assembly could be taken to without experiencing transition boiling divided by the peak power at which the fuel assembly is licensed to operate. Transition boiling in a BWR occurs when the liquid film flowing on the surface of the fuel rods dries out. Operation of a BWR fuel bundle in transition boiling causes rapid temperature oscillations on the surface of the fuel elements, which if continued over a long period of time, would increase the zircaloy corrosion rates. Because of detailed local flow and mixing characteristics within a fuel assembly, MCPR can be influenced by local power distribution. Some fuel rods have operated in film dryout transition boiling prototypical of BWR conditions for extended periods of time without failure.⁸

It is current practice to assume, for licensing purposes, that during a fast BWR pressure transient of a few seconds, BWR fuel rods will enter steady state transition boiling and that whenever transition boiling occurs, fuel rod failures will occur. This is a conservative basis for the following reasons:

1. In-pile data exist which show that BWR fuel rods can operate in transition boiling for minutes to hours without failures;⁸
2. The collapse of steam voids in a pressure transient is the primary cause of a neutron flux increase. The assumption usually is made that transient transition boiling can be approximated with steady state data. However, transient collapse of steam voids to cause a neutron flux increase tends to oppose dryout of the liquid film to cause a transition boiling cladding temperature increase. The dynamics of transient liquid film dryout during rapid steam void collapse are geometrically and physically complex. Steam void generation also feeds back to reduce local reactivity and power and in some cases, this interaction can be important;

3. Full-scale out-of-pile BWR fuel bundle MCPR measurements indicate that transient temperature increases during transition boiling in simulated pressure transients are small;
4. The worst pressure transients analyzed usually assume that all of the turbine bypass valves simultaneously fail to open. This is a low probability event which is more in the accident category than an operating transient, as currently assumed. Criteria for limiting frequent operating transients must be more restrictive than for low probability events.

A more realistic limiting fuel criterion would appear to be a limit on the peak zircaloy cladding temperature during pressure transients. Low probability pressure transients should be considered to be accidents, not operating transients. Adoption of such criteria for application to operating reactors would provide appreciable thermal margin that could be utilized to reduce uranium ore requirements by extending burnup or reducing initial enrichment requirements. For example, wider range flow control spectral shift operation may be feasible with adoption of more realistic transition boiling limits.

6.3 THERMAL/MECHANICAL PERFORMANCE

The model employed to determine the thermal/mechanical performance of the fuel designs incorporates the effects of fuel/cladding thermal expansion, fuel/cladding creep and fuel irradiation swelling, densification, re-location, and fission gas release as they affect pellet-cladding thermal conductance, fuel rod internal pressure, and pellet/cladding mechanical interaction. Included are the effects of fuel-cladding axial slip, cladding creepdown, clad irradiation growth, pellet and cladding plasticity, work hardening, creep and relaxation, and pellet hot pressing.

Improved knowledge of some of the parameters in the fuel model particularly at extended fuel burnups, may permit fuel cycle improvements. Better integration of the fuel model with the 3-D simulation model is likely to provide for improved refueling and control rod pattern optimization which may result in fuel cycle improvements by reducing the power distributions margins applied in fuel design.

6.4 COUPLED NUCLEAR THERMAL HYDRAULIC STABILITY

Reactor power level stability in a BWR is determined by the sensitivity of the neutron power to change in core voiding and the resulting perturbations in reactivity. A stable reactor's neutron power may fluctuate around:

1. A constant power level (as shown in Figure 6-1), or;
2. A steadily increasing or decreasing power level (as illustrated in Figure 6-2)

During stable operation, immediately following a small reactivity insertion, the second peak in power is smaller than the first as the power fluctuation damps out. The condition for stable operation is that the decay ratio (DR), defined as the second peak/first peak, must be less than 1.0.

Figure 6-1. Stable Steady Power Operation

Figure 6-2. Stable Power Increase

The most limiting stability conditions occur at natural circulation. Reactor stability also can limit the degree to which the flow can be reduced at rated power for purposes of shifting the neutron spectrum and increasing plutonium production.

Current practice is to design to a limiting decay ratio of 1.0 at natural circulation. Short term operation at a decay ratio of 1.0 may not be a problem provided control systems can keep the power changes within an acceptable limit cycle. No GE BWRs have yet been measured to be unstable at the most limiting natural circulation conditions.

Improvement of stability models through stability measurements and feedback to better physical representation and improved decay ratio limits may permit wider range spectral shift operation and associated improvements in BWR uranium ore requirements.

6.5 LOSS-OF-COOLANT ACCIDENT (LOCA)

The design basis loss-of-coolant accident for BWRs consists of assumed loss of the reactor core coolant through a double ended guillotine break in one of the recirculation loops after 2,000 days of continuous full power operation. A complete loss of normal power is assumed to occur simultaneously with the pipe break, which results in the longest delay time for the emergency core cooling systems to become operational. Licensing requirements are that the Peak Clad Temperature (PCT) should be limited to 2200°F for the duration of the LOCA.

Automatically initiated High Pressure Core Spray (HPCS) and low Pressure Coolant Injection (LPCI) systems provide core cooling. These systems are initiated by low water level in the reactor vessel and/or by high containment drywell pressure. Subsequent to depressurization of the reactor vessel, the temperatures in the clad rise, initially due to energy stored in the fuel pellets during power operation, and later due to decay heat from fission product and actinide decays. Some heat removal is realized when the hot channel wets due to the High Pressure Core Spray (HPCS) and the peak cladding temperature falls immediately when coolant from the HPCS or LPCI systems reaches the bottom of the active fuel length in the hottest channel (i.e., reflood).

Measurements indicate that actual peak cladding temperatures are in the 700°F to 1500°F range when currently approved reactor

licensing methods predict 2200 F. Modification of the LOCA analysis assumptions to include greater realism would permit appreciable improvement in BWR fuel cycle performance and reduction in BWR uranium ore requirements. For example, fuel could be operated with greater power peaking to reduce neutron leakage and increase end-of-cycle reactivity.

6.6 COLD SHUTDOWN MARGIN

BWRs are designed to be shutdown cold with any one control blade stuck out. Margins are included in design to assure that this condition can be met, including uncertainties in predictions. In the history of BWR operations, there have been almost no instances of a control rod not being capable of insertion cold and no instances of a BWR not meeting a one stuck control rod criterion. A soluble boron system is provided as a backup in case all other shutdown measures fail and to date, this system has never been used for shutdown at a BWR.

Cold shutdown criteria directly influence the allowable fuel loading patterns, and because loading patterns affect uranium and SWU requirements through their effect on neutron leakage and reactivity, the cold shutdown criterion impacts cycle energy, enrichment requirements and uranium and SWU requirements. The cold shutdown criteria also influence the amount of burnable absorber which is included in the fuel, which correspondingly affects the uranium and SWU requirements through the burnable absorber residual.

Cold shutdown is controlled by the top of the BWR because of higher conversion ratio, greater plutonium content, smaller U-235 depletion, and smaller burnup in the top. As fuel burnup is extended, or as steam voids during the cycle are increased by spectral shift options, the cold reactivity in the top of the core relative to the bottom is increased. Thus, many of the uranium utilization improvement options tend to adversely impact cold shutdown margins.

In-core instrumentation flux mapping systems are not operational in the cold shutdown state. Thus, although it is possible to measure criticality of simulated "one stuck control rod" configurations, only the reactivity is generally available. Consequently, in the past few years a number of special flux wire measurements have been made in cold, simulated one stuck control rod criticals.²⁴ Copper, gold and nickle wires are irradiated in TIP positions to measure the spatial distribution of the thermal, fast and resonance neutron flux. The

steep activation gradients measured in these criticals can be used to assess the accuracy of spatial cold neutron flux distribution predictions with 3-D simulation models. Measurements of detailed flux distributions in cold shutdown states and associated improvements in nuclear model predictions of cold shutdown as a function of uranium utilization improvement alternatives, should make it possible to reduce margins and to improve designs, with resultant reductions in uranium and SWU requirements through either reductions in burnable absorber residual or improvements in refueling patterns.

7. SURVEY OF INDIVIDUAL ONCE-THROUGH IMPROVEMENT ALTERNATIVES

7.1 ALTERNATIVES CONSIDERED AND BASES FOR ESTIMATING EFFECTS OF INDIVIDUAL IMPROVEMENTS

The following once-through fuel cycle design and operational alternatives with potential for BWR fuel cycle uranium utilization improvement were evaluated.

- Blankets, improved low-leakage refueling patterns, initial core optimization, reinsert of initial core discharge fuel;
- Preplanned EOC feedwater temperature reduction and power coastdown to (80%);
- Increased burnup;
- High enrichment discharge fuel rod re-assembly/reinsert;
- Coolant density spectral shift control;
- Lattice and fuel bundle design optimization;
- Reduced burnable absorber residual;
- Boric acid control;
- Six month subcycle fast refueling
- Once-through thorium cycle.

Many of these improvement alternatives are closely interrelated. For example, BWR spectral shift with flow control increases fuel burnup with constant reload enrichment and/or decreases the fuel enrichment required to achieve a given fuel burnup. A number of the alternatives reduce the fuel enrichments required to achieve greater burnups and thereby mitigate

the need for increased burnable absorber pins and concentrations, increased power peaking, more negative steam void reactivity coefficients and other related adverse trends associated with designing for greater burnup. Also, for example, when operating in the spectral shift mode with reduced coolant density throughout the cycle, neutron leakage is increased. Hence, application of optimal blankets and low neutron leakage refueling patterns enhances spectral shift operation. The complicated alternative interactions and the optimal coupled application of the alternatives was not fully evaluated or optimized in the evaluations. The savings summarized in Table 1-1 for the above alternatives were found to be non-additive in several cases.

It was not feasible within available resources to carry out comprehensive 3-D coupled nuclear-thermal-hydraulic fuel cycle analyses of all of the individual alternatives. Therefore, a mixture of approximate and more detailed analyses was used to assess the individual improvement potential. Some alternatives were studied more extensively than others, depending on improvement potential retrofitability, methods limitations and available resources. Combinations of alternatives, which are reviewed in following sections, were evaluated in greater detail than the individual alternatives.

The range of improvement potential associated with each alternative (Table 1-1) represents, for some alternatives, the degree to which these new features can be applied in operating reactors and for other alternatives, represents the uncertainty in the estimates of improvement potential.

7.2 BLANKETS, IMPROVED LOW-LEAKAGE REFUELING PATTERNS, INITIAL CORE OPTIMIZATION, REINSERT OF INITIAL CORE DISCHARGE FUEL

7.2.1 Assumptions, Criteria, Methods of Approximation

The "current improved BWR" incorporates most of these features and is discussed in Section 4.0. In the mid-seventies, a large multimillion dollar effort was applied to developing improved BWR fuel and core designs with blankets, low-leakage refueling patterns, and more optimal initial core features. Detailed 3-D coupled nuclear-thermal-hydraulic fuel cycle analyses, coupled with Monte Carlo and other benchmark analyses were used in this design activity. These design features have been extensively retrofit to most domestic BWRs, although few operating cores have yet to reach the full potential of the uranium savings. The uranium savings estimates in Table 1-1 for these design improvements are, therefore, based on extensive analyses, but still have a band of uncertainty due to the variability in BWR design features, and due to the long time

interval required to collect applicable operating plant data and required to verify design predictions against the data.

7.2.2 Uranium Requirements

As indicated by Table 1-3, the "current improved" design reduces uranium ore and separative work requirements by 7 and 10 percent, respectively, relative to the reference design. This design incorporates three bundle enrichments in the initial core (Figure 4-5), axial and radial natural uranium blankets, and low leakage refueling patterns with the lowest reactivity last cycle fuel loaded as a blanket at the core periphery.

For annual refueling cycles, a four enrichment initial core design is more optimum, as shown in Figure 7-1. This design alternative increases the energy from the initial core discharge batches by 6% relative to the "current improved design", with constant initial SWUs in both designs. Therefore, the uranium requirements in the first 5 years of operation also are reduced by 6%. On a 30 year basis, this savings averages 1% improvement in uranium utilization. However, reinsert of fuel discharged from a non-optimum initial core also can be utilized to recover some of the uranium losses from an inefficient initial core design.

The first few batches of fuel assemblies discharged from single-enrichment initial core designs (the reference design basis) have sufficient fissile content to justify reinsert in later cycles. Typically, if about one-half of the first discharge batch reinsert bundles were loaded in each of cycles 3 and 4, the number of new reload bundles for these cycles would be reduced by about 9%. Reinsert has been applied in the past for numerous BWRs (Table 1-2). Part of the uranium savings associated with initial core optimization can be obtained for all BWRs by reinsert of initial core fuel discharged from non-optimum designs.

There also is further potential for improvement of the "current improved" BWR design by refueling and control rod pattern optimization. The extent to which such optimizations can be effectively accomplished is dependent on available thermal and licensing margins. Additional uranium savings on the order of 4% have been estimated to be achievable from limited range optimization of the "current improved" BWR within current licensing criteria and constraints. If the improvements in criteria discussed in Section 6 were to be brought to fruition, the uranium savings would be appreciably greater as a result of reduced neutron leakage and reduced fuel enrichments.

Figure 7-1. Fuel Utilization Improvements From Four-Enrichment Initial Core Design

7.2.3 Economic Incentives & Considerations

The "current improved" BWR design reduced 10 year present worth BWR fuel cycle costs by 20 to 50 million dollars per 1000 MWe plant, depending on uranium ore cost, separative work unit (SWU) costs, and the magnitude of other fuel cycle cost components. Thus, although the uranium ore and separative work savings of about 7 and 10 percent, respectively, were modest compared to the ultimate BWR potential, the economic impact on utility fuel cycles was very large.

It has been the practice in BWR fuel manufacture to completely scan 100% of all fuel rods with a neutron scanner to assure reliability of fuel loading. Therefore, natural uranium axial blankets were easily retrofit with no major quality control or manufacturing difficulties. The presence of natural uranium pellets in each fuel rod provides a standard which can be used to determine the absolute enrichments of other fuel pellets in the fuel rod from the neutron scan.

7.2.4 Impact on Core Characteristics, Fuel Reliability, & Safety

Movement of low reactivity fuel to the periphery for its last cycle eliminates the need for movement of fuel from peripheral core positions to the interior and associated requirements for detailed consideration of historical effects of the steep power and local burnup tilts, caused by leakage and the reflector, on local power peaking and other performance parameters after peripheral fuel is located in central core regions. This refueling pattern also avoided subjecting the fuel to greater power after residence on the low power periphery, and reduced the power and stresses on the greatest burnup, longest residence time fuel for its last cycle of residence. The reduction of neutron leakage also reduces neutron irradiation of the reactor vessel internals and the reactor vessel. Consequently, the low-leakage refueling patterns improved fuel reliability and simplified design in addition to reducing uranium ore requirements. Power peaking also was reduced by optimal location of the highest reactivity fuel inside the periphery.

Use of 6" long natural uranium axial blankets increased axial power peaking by about 1%. This was more than compensated by design for reduced magnitude negative steam void reactivity coefficients and by a 1% increase in active fuel length. Axial neutron leakage was reduced significantly and the under utilization of enriched fuel at the ends of each assembly was eliminated. BWR control blade tips extend approximately 4" into the bottom of the active core when they are fully withdrawn. Therefore, there is little incentive to increase fuel in this region above natural U-235 concentrations. The large steam void fraction at the top of the

core reduces the power appreciably in the top 6" and therefore, there also is little incentive to enrich above natural levels in this region.

7.2.5 Technical Uncertainties, Problems & Mitigating Measures

There are uncertainties in the predictions of improved uranium utilization for cores with low reactivity blankets. Operating data is becoming available to test predictions. Special power distribution (gamma scan) and isotopic measurements are desirable to confirm details of blanket and low leakage refueling pattern performance.

7.2.6 Retrofitability

As indicated in Table 1-2, axial natural uranium blankets and low-leakage refueling patterns are being applied widely in all GE BWRs. However, these features are not widely applied by non-domestic BWR fuel and core suppliers. Improved 3- or 4- enrichment initial cores and radial natural uranium blankets are being applied to most domestic new initial cores. The Hatch 2 reactor, which started up in 1978, was the first BWR to employ all features of the "current improved" design. The Kuo Sheng reactor, which recently started up, is the first BWR/6 3 bundle enrichment initial core, including axial and radial natural uranium blankets.

Reinsert of initial core discharge fuel is being applied at about one-half of the GE BWR plants. Use of this alternative may increase further as uranium and SWU costs increase.

7.2.7 RD&D Requirements

For the retrofit "current improved" fuel now in operation, detailed fuel rod and fuel bundle gamma scan measurements are desirable to confirm power distribution predictions. Flux wire measurements in cold criticals that simulate "one-stuck control rod" conditions are desirable to provide detail on the reliability of the cold criticality predictions. Isotopic measurements on axial and radial natural uranium blankets also are desirable to confirm predictions of blanket conversion ratios and blanket local reactivity versus burnup.

These measurements will feed back to improved design methods which can be used to further optimize the uranium and fuel cycle savings from these alternatives.

7.3 PREPLANNED EOC FEEDWATER TEMPERATURE REDUCTION (FWTR) AND POWER COASTDOWN

7.3.1 Assumptions, Criteria & Methods of Approximation

BWR cycle extension by reduction of feedwater inlet temperatures reduces the steam void content of the core to extend the operating cycle beyond the end of full power life. This is accomplished by bypassing feedwater heaters, which for constant reactor power causes a reduction in electrical power, as a result of a poorer heat rate (Figure 7-2).

Lower than rated feedwater temperature at end-of-cycle conditions decreases coolant inlet enthalpy (increased core subcooling), and decreases core steam content to cause positive reactivity feedback (increased neutron moderation) which permits continued operation at core rated thermal output. For a typical case, the core average void fraction is reduced by approximately 20% when the feedwater temperature is reduced from 420°F to 250°F. Figure 7-3 is a typical comparison of the axial void distribution for nominal and reduced feedwater temperatures. The change in void distribution causes the power to peak toward the top of the core which further increases reactivity as a result of greater importance weighting of the highest fissile inventory, lowest burnup regions of the core. The extent to which refueling cycle length can be extended due to FWTR is proportional to the degree of feedwater temperature reduction. A feedwater temperature of 250°F was considered as the lower limit, based on systems and stress considerations.

Coastdown in power at end-of-cycle gains reactivity as a result of reduction in steam voids, reduction in fuel temperature (Doppler) and reduction in steady state xenon. The axial power distribution also peaks more toward the top of the reactor as power is decreased thereby peaking neutron importance on lower burnup, higher plutonium concentration regions, and reducing neutron leakage. Coastdown from rated power at end-of-cycle increases burnup of the fuel assemblies ready for discharge.

NOTE: CAN USE EITHER FEEDWATER TEMPERATURE REDUCTION OR THERMAL POWER COASTDOWN IN A PLANNED MANNER TO REDUCE URANIUM REQUIREMENTS

Figure 7-2. Cycle Extension

Figure 7-3. Effect of Feedwater Temperature Reduction on Steam Void Axial Distribution

7.3.2 Uranium Requirements

For feedwater temperature reduction from 420°F rated conditions to 250°F combined with power coastdown to 80% of rated power, the uranium savings are approximately 10%. A typical BWR achieves about 1 month cycle extension from a 10% power reduction. Considering that all BWRs will not apply coastdown, the uranium utilization improvement potential was estimated to range from 4 to 8 percent.

7.3.3 Economic Incentives, Considerations

The utilities are currently using FWTR and EOC power coastdown to provide flexibility in refueling schedules, coupled with uncertainty in the operating capacity factor. Both methods of cycle extension reduce EOC electrical power. Equipment is needed (valves) to bypass the feedwater heaters for FWTR and this is not available on all BWR plants. The uranium savings from use of power coastdown and/or feedwater temperature reduction must be economically balanced against the cost of the power needed to make up for any reduced electrical output during the cycle extension. The desirability and duration of cycle extension is thus sensitive to the type (coal, oil, gas, hydro) and cost of available replacement energy. The large and rapid escalation of oil and coal costs relative to nuclear fuel may tend to result in reduction of the optimum coastdown interval, with a resultant trend toward greater uranium use.

Figure 7-4 is a plot of the total power costs as a function of time for different levels of makeup cost. For makeup power costs that are twice nuclear (for example, coal) there is an economic incentive to coastdown for 12 weeks. This would reduce uranium ore requirements by approximately 10% and would require electrical power reduction to approximately 75% of rated. If uranium ore requirements were the only consideration, there would be incentive to coastdown to the break-even point where the overall power costs after coasting down are equal to the costs if no coastdown had been applied.

7.3.4 Impact on Core Characteristics, Fuel Reliability, Safety

Although end-of-cycle extension increases discharge burnup, the magnitude of increase is not expected to significantly impact core characteristics or fuel reliability, provided fuel enrichments are reduced to compensate.

Figure 7-4. Cycle Extension - Total Fuel Cost Change (Reactor Costs and Makeup Power Costs)

7.3.5 Technical Uncertainties, Problems and Mitigating Measures

The capabilities of 3-D simulation methods to accurately predict reactivity gains from FWTR and to optimize the coastdown burnup intervals is dependent on the specific core design features in use. Also, in some cases, the extent to which the FWTR option can be applied depends on the feedwater inlet sparger design to accommodate thermal transients and the ability to retrofit equipment that permits bypassing the feedwater heaters.

7.3.6 Retrofitability

There is no significant obstacle to retrofit of EOC power coastdown to existing plants. Cycle extension with coastdown is being applied to a large number of BWR plants down to about 50-60% EOC power. Figure 7-5 is a summary of recent data on plant coastdowns. The majority of coastdowns have been to powers that are greater than the economic optimum, illustrating that utilities are giving greater weight to use of this option for refueling scheduling flexibility. This may also be indicative that there are other factors which influence power costs that should be incorporated in the optimization depicted by Figure 7-4. Figure 7-6 illustrates that the number of BWRs applying end-of-cycle power coastdown is gradually increasing and in the 78-79 time frame climbed to 71% of the BWR operating plants.

7.3.7 RD&D Requirements

Of the tabulated alternatives, preplanned end-of-cycle FWTR and power coastdown have the least RD&D requirements. Development of improved methods for optimization of cycle extension, which includes consideration of the many factors that utilities use in making decisions, would potentially result in more effective application to conserve uranium.

The ability to accurately predict FWTR and power coastdown performance also may depend on the specifics of BWR core and fuel design and the coupled effects of other improvements that are being applied. For example, use of natural uranium blankets should reduce the reactivity gain from neutron leakage during FWTR or power coastdown and there currently is little data to determine the accuracy with which such coupled effects are being predicted. Also, other fuel cycle improvement options such as spectral shift with flow control tend to cause the EOC axial power

FIGURE 7.5
BWR COASTDOWN EOC POWER LEVELS

FIGURE 7.6
BWR EOC POWER COASTDOWN EXPERIENCE

shape to peak more to the top of the BWR and, if FWTR and/or power coastdown are coupled with spectral shift, this will make nuclear-thermal-hydraulic predictions and conversion ratio predictions at high steam voids more important.

Therefore, most of the technological RD&D requirements associated with FWTR and coastdown involve nuclear and other measurements to test the reliability of design and predictive methods when coastdown is being applied in conjunction with other fuel cycle improvements.

7.4 INCREASED BURNUP

7.4.1 Assumptions, Criteria, Methods of Approximation

Burnup can be increased by increasing the fissile content of fresh fuel or by applying other improvements that conserve neutrons, reduce neutron leakage, increase conversion ratio, or increase end-of-cycle reactivity. For purposes of separating individual effects the differential effects of increased burnup in this section are limited to the effects of increased reload enrichment.

As indicated in Section 5.1 initial enrichment requirements are approximately a linear function of the sum of core average and discharge burnup. The constants in the linear equation, Equation (5.2), are affected by other improvement alternatives such as spectral shift, natural uranium blankets, etc., but are not affected by limited range enrichment changes.

A major impact of increasing burnup is that the refueling batch size requirements or number of bundles per reload, are reduced (Equation (5.1)). Reduction of refueling batch size is a major contributor to reduction of uranium requirements with increased burnup.

The tendency among many utilities is to schedule reactor operation for 18 month or longer refueling cycles. Figures 7-7 and 7-8 illustrate the BWR trend to increased reload cycle length. Greater refueling cycle length increases the refueling batch size and increases the uranium requirements. Increased cycle lengths also generally require design with greater burnable absorber concentrations which increase the burnable absorber residual from the even atomic mass gadolinia isotopes. Because of the importance of this variable, refueling cycle length was included as a parameter in the extended burnup evaluations.

*ADJUSTED TO A FIXED 75% CAPACITY FACTOR

Figure 7-7. Reload Cycle Length Flexibility - Completed Cycles

*ADJUSTED TO A FIXED 75% CAPACITY FACTOR

Figure 7-8. Reload Cycle Length Flexibility - Current Cycles

7.4.2 U_3O_8 Requirements

Of the options, increased burnup has the greatest uranium utilization improvement potential. The range is estimated at 5-20 percent improvement corresponding approximately to increases in average batch discharge burnup from 25-75 percent. The optimum burnup from considerations of uranium utilization appears to be well beyond the economic optimum or the practical optimum (Figure 7-9).

Figure 7-9 summarizes uranium requirements versus burnup and refueling cycle length based on evaluations with the simple relationships summarized in Section 5. Fitted linear parameters (α and β) from 3-D analyses of 12 month cycles were used in the 6, 18, and 24 month analyses. Because the effects of cycle length on burnable absorber requirements are not included, this tends to result in optimistic predictions of uranium requirements for the 18 and 24 month cycles and pessimistic predictions for 6 month cycles. For large fuel enrichment increases, the number of burnable absorbers rods in the bundle increases, which will tend to result in poorer performance with burnup than indicated. Power peaking increases with extended burnup as a result of greater power mismatch between fresh and discharge fuel. Such power peaking increases generally require design revisions that feedback to increase uranium requirements. Preliminary 3-D simulations of the effects of increased burnup were completed in the combined case evaluations that are reported in a later section.

7.4.3 Economic Incentives, Considerations

Increased burnup has the largest uranium utilization improvement potential, but also the largest RD&D requirements. It is essential that fuel reliability not be depreciated by design for increased burnup. Large scale statistically significant irradiations are needed to assure the reliability of extended burnup fuel designs. Some hardware revisions such as improved control blades and/or fine motion control rod drives may be desirable to mitigate impacts of extended burnup on burnable absorber requirements, on power distributions, and on fuel reliability. Because of reduced factory thruput and fuel design revisions needed for increased burnup, fabrication costs increase as a function of burnup and these cost increases will feedback to decrease the optimum burnups. Hence, feedback from the RD&D programs will be needed to determine economically optimal burnup levels.

Figure 7-9. Approximate BWR Uranium Ore Utilization for Equilibrium Cycle Operation (Increased Enrichment)

7.4.4 Impact on Core Characteristics, Fuel Reliability, Safety

Increased burnup results in greater fuel residence time, greater zircaloy corrosion, greater fission gas pressure and the potential for greater fission product attack on the cladding interior. The smaller refueling batch size and greater mismatch in reactivity and in fission and absorption cross sections between fresh and last cycle fuel cause greater channel power peaking, or power peaking due to mismatch in the core.

However, axial power shapes may be flattened by the non-uniform axial burnup in last cycle fuel flattening the power in fresh fuel. Addition of burnable absorber to the lattice will tend to increase local power peaking and/or the peak-to-average fuel rod enrichment. Increased fuel enrichment tends to make steam void reactivity coefficients more negative. Because of these trends, extension of burnup with fuel enrichment increase results in more complicated and costly fuel bundle and core designs and in most cases, depreciation of thermal and transient margins. Consequently, improvements in licensing criteria, as discussed in Section 6, may be important in realizing the full potential of extended burnup design.

7.4.5 Technical Uncertainties, Problems & Mitigating Measures

The extent to which fuel design improvements such as pellet cladding interaction (PCI) resistant fuel must be utilized to protect the cladding from greater fission product attack at greater burnups cannot be easily determined without large scale irradiations which accumulate adequate statistics. Design basis fuel failure levels are so small as to require irradiation of appreciable quantities of fuel to achieve the necessary statistics.

There are uncertainties in fission gas release rates, corrosion rates, and PCI susceptibility as a function of design and operating variables. There also are uncertainties in predicted pin powers, peak burnups, reactivity coefficients, and other performance related parameters as a function of burnup.

Some of the uncertainties in increased burnup design are mitigated by use of other improvement alternatives to extend burnup without increasing fuel enrichment. For example, low-leakage refueling patterns which move fuel to the low power periphery during the last cycle of residence, produce increased burnup with no fuel enrichment or burnable absorber revisions, and also tend to mitigate the effects of increased burnup on fuel reliability by decreasing fuel power at greater burnups. Another similar design example

is the recently introduced Control Cell Core design which improves power distributions; reduces the amount of control rod movement; maintains low power, last cycle or next to last cycle fuel adjacent to inserted control blades; and avoids adjacent control blade motion throughout fuel life for about $\frac{1}{2}$ of the fuel.

Cold shutdown, one-stuck-control rod margin generally becomes more limiting as burnup is extended. Many of the options which improve burnup without increasing fuel enrichment, such as spectral shift with flow control, increase the conversion ratio. Increases in conversion ratio generally tend to make cold shutdown more limiting, thus increasing burnable absorber requirements and burnable absorber residual. As the burnable absorber content is increased, the residual reactivity penalty increases, which further increases fuel enrichment requirements, and it becomes more difficult to maintain adequate hot excess reactivity margins. Consequently, design for extended burnup creates many coupled reactivity, power distribution and thermal margin problems.

7.4.6 Retrofitability

Extended burnup is retrofitable to all BWRs, but the economic optimum may be different for different designs. Lower power density earlier class BWRs generally can be more easily designed for increased burnup, but because of smaller specific power (kW/ft) trend to have a reduced optimum burnup as a result of the important effects of refueling batch size on the optimum.

7.4.7 RD&D Requirements

Increase in discharge burnup requires extensive fuel and design development to assure reliability and to avoid erosion of thermal margins, fuel reliability and performance. BWR current design fuel rod reliability levels have been better than 99.9%, therefore requiring large volume irradiation statistics to reliably determine the effects of extended burnup. The RD&D requirements are summarized in Tables 1-5 and 1-6. Considerable work also remains to define fabrication costs versus burnup and the economically optimum burnup levels.

7.5 HIGH ENRICHMENT DISCHARGE FUEL ROD REASSEMBLY/REINSERT

7.5.1 Assumptions, Criteria, Methods of Approximation

In BWR fuel bundle design, the interior rods have higher initial enrichment. At discharge, the burnups on the fuel pins are fairly even. Therefore, the interior fuel rods of discharged fuel have higher residual fissile content and do not produce energy commensurate with their enrichment. On the average, the discharge fuel bundle no longer has enough reactivity to warrant further core residence. However, the interior rods have sufficiently high residual fissile content to justify reassembly without any reprocessing and reinsert into the reactor for further irradiation. This option has been under consideration since the late 60's but it has not been applied appreciably except in fuel assembly repair programs.

The current standard design BWR 8X8 (62 rods per fuel bundle) bundles have two water tubes located in the central region of the assembly. Recycle Rod Reinsert Bundle (RRRB) designs which incorporated increased numbers of water tubes in the central region of the fuel assembly were evaluated. The number of water tubes was increased to further increase the initial reactivity coefficients, and improve local power distributions. The optimum H/U ratio of any given fuel assembly increases with fuel assembly burnup due to plutonium buildup, thereby creating potential incentive for water rod addition late in fuel assembly life. However, there is a loss in weight and heat transfer surface, and the reactivity drops more rapidly with bundle energy production due to reduced conversion ratio when water rods are added.

7.5.2 U_3O_8 Requirements

A preliminary, approximate 3-D evaluation of the high enrichment discharge fuel rod reassembly and reinsert option was performed using the "current improved" BWR C-lattice design as the reference. Two discharge bundles were used to obtain rods for the RRRB bundles

Figure 7-10 is a summary of the refueling pattern that was utilized. Table 7-1 is a summary of the results for RRRB design with and without water rod additions. Uranium savings were 4 to 5 percent and there was little incentive indicated in this case for water rod addition.

Fuel management was carried out by holding the enrichment of the reload fuel constant and adding the RRRB bundles to the refueling batch, while decreasing the refueling batch size of the fresh fuel. It was assumed that all discharge fuel was desirable for reconstitution, independent of the reconstituted assembly reactivity relative to other fuel in the core. This is the most conservative basis for evaluation because of the drop in discharge burnup prior to high enrichment discharge fuel rod reconstitution and reinsert penalizes the fuel cycle and some of the discharge fuel is not desirable for reconstitution. Greater gains could have been realized by raising the reload fuel enrichment to maintain constant burnup prior to high enrichment rod recycle; however, this complicates the analysis because the higher enrichments feedback to affect RRRB fuel.

The detailed 3-D distribution of fuel properties inside the fuel bundle also was not considered in the analyses. RRRB properties were evaluated using approximate average isotopic compositions of the fuel rods. A more detailed 3-D fuel bundle synthesis may affect the conclusions and the estimated incentives.

The 3-D nuclear-thermal-hydraulic analyses indicate that power peaking was increased by about 5% by introduction of recycle rod reinsert fuel. However, no optimization of refueling patterns was carried out to mitigate this or to improve performance.

NO. ASSEMBLIES

42 = CYCLE 1 FUEL (FRESH)

42 = CYCLE 2 FUEL (ONCE BURNED)

42 = CYCLE 3 FUEL (TWICE BURNED)

41 = CYCLE 4 FUEL (THRICE BURNED)

20 = RECON BUNDLES

187

1/4 CORE BATCH SIZE

Figure 7-10. General Refueling Strategy

TABLE 7-1

PRELIMINARY RESULTS ON EQUILIBRIUM CYCLE
BUNDLE REASSEMBLY/REINSERT EVALUATION

ALL RESULTS ARE FROM 3-D MULTICYCLE HALING EVALUATIONS

PLANT SIZE 748	CASE 1 CURRENT IMPROVED BWR	CASE 2 $\frac{1}{2}$ BUNDLE 2 LARGE H ₂ O RODS	CASE 3 $\frac{1}{2}$ BUNDLE* 2 LARGE H ₂ O RODS & 4 REGULAR H ₂ O RODS
● AVERAGE CAPACITY FACTOR	0.75	0.75	0.75
● BATCH FRACTION	0.25	0.22	0.22
● # OF FRESH BUNDLES/REFUELING	188	168 + 80 REASSEMBLED BUNDLES	168 + 80 REASSEMBLED BUNDLES
● CYCLE LENGTH (YEAR)	0.997	0.930	0.934
● DISCHARGE FUEL BURNUP (MWd/MT) REASSEMBLED BUNDLES**	28,400	27,300	27,400
● EOC kw/ft***	10.55	11.04	11.00
● MCPR	1.337	1.273	1.277
● U ₃ O ₈ REQUIREMENTS (ST/GWE-YR)	189.3	181.4	180.6
● SWU (10 ³ SWU/GWE-YR)**	110.5	105.9	105.9
● IMPROVEMENT RELATIVE TO "CURRENT IMPROVED" DESIGN, %			
--URANIUM	---	4.2	4.6
--SWU	---	4.2	4.2

* " $\frac{1}{2}$ Bundle"- one reassembled bundle from two discharged bundles.

** Reassembled bundles had 27,600 MWd/MT burnup from previous core residence.

*** Improvement possible with more work on loading pattern.

Greater gains and improved performance are expected through further optimization. Application of RRRB designs for less efficient D-lattice designs, characteristic of most of the current operating BWRs, also is expected to result in greater uranium savings.

The range of uranium savings potential of this option, relative to the BWR reference case, is estimated to be 4 to 10 percent, including considerations of the inventory of discharge fuel now in storage.

7.5.3 Economic Incentives & Considerations

The economic incentives for high enrichment discharge fuel rod reassembly and reinsert are relatively large, since this option increases achievable burnup with no increase in average initial enrichment. However, reliable methods must be developed for carrying out and quality controlling the reconstitution at low cost, while also assuring licensability and fuel reliability. Keeping personnel exposures low also is an economic consideration.

Reconstitution and reinsert of discharge fuel reduces the fuel in storage, particularly during early years of application, and this is an additional factor that must be considered in economic analyses.

7.5.4 Impact on Core Characteristics, Fuel Reliability, Safety

Core design optimization studies will be needed to maximize uranium savings and minimize the radial peaking in the core. High enrichment discharge fuel rod reassembly and reinsert potentially can be considered for initial core fuel without exceeding current fuel burnup design bases. However, retrofit to reload fuel now in reactors generally will result in extension beyond the burnups for which these fuel rods were designed and licensed. On reinsert, the RRRB fuel rods also usually will experience appreciably greater power than occurred in the last cycle of residence prior to discharge. The power increase late in life coupled with extended burnup could impact reliability. Therefore, fuel reliability is a consideration in RRRB design and irradiation.

7.5.5 Technical Uncertainties, Problems & Mitigating Measures

Because of the individual axial burnup and isotopic distributions in the fuel rods, unique design and quality control procedures may be required to assure reliable application. Although BWR fuel assemblies easily can be manually disassembled and reassembled with different fuel rod configurations, as has been demonstrated many times in the past, large quantity processing will require different, more automated procedures. Design with additional water rods in the reassembled bundles may require critical heat flux and loss of coolant accident simulation measurements in out-of-pile test facilities. Because of geometric complexities in the spatial burnups and isotopics of the reconstituted rods, there are uncertainties in the accuracy of design method application to RRRB designs.

7.5.6 Retrofitability

High enrichment discharge fuel rod reassembly and reinsert appears to be retrofitable to all BWRs provided any adverse impacts on thermal and licensing margins can be mitigated through design.

7.5.7 RD&D Requirements

General application of this option to reload fuel will require many of the RD&D efforts of the increased burnup option discussed earlier.

Nuclear measurements, such as gamma scans, on LTA assemblies are needed to confirm predicted performance.

Improved design and licensing methods and automated quality control procedures will be required for large scale application. Automated disassembly and reassembly also may be desirable.

Critical heat flux and loss of coolant measurements may be needed for designs that use additional water tubes. Nuclear and fuel measurements will be needed to confirm the predicted performance characteristics of reassembly fuel.

7.6 COOLANT DENSITY SPECTRAL SHIFT CONTROL

7.6.1 Assumptions, Criteria, Methods of Approximation

Spectral shift by flow control takes advantage of the BWR's inherent capability to control the reactor with coolant, or moderator density.^{2,3} Provided there is sufficient thermal margin flow can be reduced at rated power throughout most of an operating cycle to reduce moderator density and maximize conversion ratio (hard spectrum). At end-of-cycle, flow can be increased to the maximum achievable to increase moderator density and maximize core reactivity (cold spectrum). The changes in flow at end-of-cycle were approximated by the step increases in flow necessary to keep the reactor critical over a burnup step. The accuracy is affected by the burnup step size, with the predicted spectral shift gain reduced by use of burnup steps which are too large.

Three-dimensional coupled nuclear-thermal-hydraulic fuel cycle methods are required and were used to evaluate the impacts of spectral shift operation. There is a transient in the energy gains after the first cycle of application, and therefore multicycle evaluations are required. Inlet subcooling varies with recirculation flow and therefore, it also is important to properly evaluate the heat balance to obtain core inlet temperatures. Axial power shape changes in the core, caused by spectral shift operation also contribute to the magnitude of the achievable gain and must be evaluated.

Figure 7-11 is a plot of a typical BWR power-flow map. The central region is the licensed region for normal operation. In order to apply the spectral shift with flow control option, this operating map must be expanded. The outer boundary is the boundary that is currently achievable for most operating BWRs.

7.6.2 U_3O_8 Requirements

Detailed 3-D evaluations were performed on a reference core containing radial (last cycle fuel) and axial blankets with spectral shift by flow control as an additional improvement feature. Table 7-2 is a summary of the incremental cycle burnup improvement as a function of flow range and cycle of application (the

FIGURE 7-11 BWR Operating Domain Showing Extensions for Spectral Shift via Flow Control Operation

TABLE 7-2

SAVINGS FOR BWR SPECTRAL SHIFT VS. CYCLE APPLIED

<u>WITHIN CYCLE % BELOW RATED FLOW</u>	<u>E-O-C % ABOVE RATED FLOW</u>	<u>PERCENT BURNUP INCREASE/U SAVINGS FOR SPECTRAL SHIFT CYCLE NUMBER</u>					
		<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>
90	105	3.9	1.4	2.2	2.2		
70	100	3.1	4.0	3.0	3.4		
70	105	5.7	3.6	3.9	4.1		
70	120	13.4	3.1	6.7	7.0	6.6	6.2
60	120	15.0	2.9	7.4	8.0	7.2	6.8

EQUILIBRIUM TRENDS:

FLOW BELOW RATED: ~ 0.13% URANIUM SAVINGS/% FLOW

FLOW ABOVE RATED: ~ 0.16% URANIUM SAVINGS/% FLOW

base case with no spectral shift was an equilibrium cycle). The greatest improvement occurs in the first cycle of application. Spectral shift converges to equilibrium in 3 to 4 cycles of application.

Uranium savings are approximately the same as the energy increase, assuming that the initial enrichment would be held constant and spectral shift would be used to increase burnup. Increases in flow above rated at end-of-cycle give greater gains per percent flow change than decreases in flow below rated during the cycle. For the reactor studied, there was an approximate 0.13% improvement in uranium utilization per percent flow below rated during the cycle and about 0.16% improvement per percent flow above rated flow at end-of-cycle.

The average, overall BWR spectral shift uranium utilization improvement potential is estimated to be in the range of 2 to 8 percent reduction in uranium ore requirements.

7.6.3 Economic Incentives and Considerations

Capability to operate at less than rated flow/rated power provides operational flexibility, aids load following, simplifies conformance to fuel PCI constraints, minimizes control rod movements in burnup control, speed up start ups and xenon control, and increases capacity factors. Therefore, there are other operational incentives to have capability for spectral shift operation. For these reasons, many utilities have re-licensed BWRs to permit operation in expanded flow domains.

There is appreciable incentive for limited range spectral shift (about 85% flow to 105% flow) that is reasonably easily achievable at current operating BWRs. Some operating BWRs can achieve flow ranges of 60 to 115 percent flow today.

Expansion of spectral shift capability beyond that achievable by current designs is dependent on equipment costs, retrofit cost, and licensing considerations.

Operation at less than rated flow with efficient pump speed control reduces pumping power requirements and improves plant heat rate. This would result in a further uranium ore savings per MWe generated which was not considered in this evaluation.

Retrofit of design, licensing and hardware to permit operation at further reduced flows during an operating cycle primarily would require control system and software revisions. Extended expansion of flow capacity above rated may require changes in pumping systems and in some cases vessel internal hardware to mitigate vibrations and/or stresses from increased flow. Therefore, it generally should be less costly to expand the rated power flow range in the downward direction rather than the upward direction.

Very wide spectral shift control may be considered in high conversion ratio, near breeder BWR applications. However, such BWR designs will require addition of fuel heat transfer surface.

7.6.4 Impact on Core Characteristics, Fuel Reliability, Safety

Reactor operation at rated power and less than rated flow may decrease the MCPR margins. A general rule of thumb for MCPR margin deterioration is approximately 1% MCPR margin loss for every 3% flow reduction. However, in some cases, the change in axial power shape and inlet subcooling, coupled with greater radial power flattening from steam voids can result in very small impacts on thermal margins.

Axial burnup shapes are affected by spectral shift operation and this can impact the steam void reactivity coefficients and reactor scram response. Such changes have to be evaluated with respect to effects on licensing analyses.

The greater operational flexibility from wider rated power flow control range and greater ease with which xenon transients can be controlled and load following actions implemented usually will result in improved fuel reliability and increased capacity factor being attributable to expanded spectral shift capability.

7.6.5 Technical Uncertainties, Problems & Mitigating Measures

There are uncertainties in the accuracy of 3-D nuclear simulation and prediction of spectral shift fuel cycle performance, particularly when this alternative is coupled with other improvement options such as use of low leakage refueling patterns and axial blankets. There also are uncertainties in the degree to which maximum potential spectral shift may be achieved. In some cases full potential could be more easily achieved by greater realism in licensing analyses. Spectral shift operation at less than rated flow/rated power is particularly sensitive to MCPR and stability licensing criteria (see Section 6)

7.6.6 Retrofitability

Coolant density control spectral shift can be applied to some extent in most BWRs with little or no modification. Most BWRs are able to operate over approximately a 90 to 105% flow range at rated power through completion of special licensing analyses. However, achievement of the full BWR spectral shift potential may require plant and core equipment revisions. The extent and potential for such modifications, as summarized in Table 1-5, would have to be considered on an individual plant basis.

7.6.7 RD&D Requirements

Table 1-6 summarizes the RD&D requirements for realization of maximum spectral shift potential. In the shorter term applications, nuclear measurements are needed to verify the predicted performance in full scale applications, including coupling with other uranium utilization improvement alternatives.

7.7 LATTICE & FUEL BUNDLE DESIGN OPTIMIZATION

7.7.1 Assumptions, Criteria, Methods of Approximation

Fuel density variations were evaluated to test the sensitivity of uranium ore requirements to H/U ratio. These evaluations, therefore, simulate the approximate impact of use of annular fuel pellets or lower density fuel. The H/U ratio also was varied by changing the fuel rod diameter. There are other fuel design and lattice design improvements that could be considered; however, these were outside the scope of these evaluations and the available resources.

The evaluation of reduced density fuel was performed by maintaining the fuel rod fissile inventory constant, independent of density. For example, an 80% density fuel rod requires 3.75 w/o U-235 enrichment to maintain the same fissile inventory as a 3.0 w/o U-235 fuel rod.

Fuel bundle lattice physics analyses were performed for different density fuel at an approximate average steam void content. Figure 7-12 illustrates the reactivity versus burnup curves for the different density fuels. In order to evaluate fuel cycle performance of these fuels on a common base, the reactivity curves were normalized to nominal design basis fuel density and plotted on an equal energy basis. The equal energy basis corrects the burnup for the weight and reactivity slope differences and provides a

Figure 7-12. Reactivity Versus Burnup for Reduced Density Fuels

realistic basis for intercomparing the reactivity lifetime of different weight fuels. Figure 7-13 is an equal energy plot of the different density fuels. The plots illustrate that there is a burnup "cross over" point below which the lower density design is advantageous, and above which the solid pellet fuel has greater reactivity. This is caused by the reduced conversion ratio of lower density fuel. The uranium utilization is sensitive to discharge burnup and end-of-cycle relative reactivity of the discharge fuel. If other options such as end-of-cycle coastdown are coupled with use of reduced density, the benefits of reduced density are decreased.

Figure 7-14 illustrates the trends in infinite multiplication when fuel rod diameter is used to vary the H/U ratio. The trends are similar to the trends with pellet density except that fuel rod diameter also affects pressure drop, MCPR and other thermal-hydraulic feedbacks. These effects are not reflected in Figure 7-14.

7.7.2 U_3O_8 Requirements

The uranium utilization improvement from reduced density fuel and increased H/U ratio is dependent on the relative end-of-cycle reactivity of the discharge fuel. The uranium utilization gains from reduced density tend to increase with larger refueling batch sizes and with greater relative reactivity of the discharge fuel. Design for extended burnup reduces the incentive to reduce pellet density or increase the H/U ratio. End-of-cycle coastdown, spectral shift with flow control and end-of-cycle feedwater temperature reduction increase the end-of-cycle H/U ratio and decrease any incentives to increase the lattice H/U ratio through changes in fuel design. When coupled with these improvement alternatives, increases in H/U ratio and/or reductions in fuel density can depreciate uranium utilization and increase fuel cycle costs.

7.7.3 Economic Incentives and Considerations

The uranium savings incentives for reduced fuel density and greater H/U ratio are small. However, for a limited range there is not a significant penalty if reduced density or greater H/U ratio were needed to improve performance. Thus, if reduced density fuel were desirable to provide fission gas plenum space (e.g., by using annular fuel pellets), to improve fuel reliability, to make steam void reactivity coefficients less negative, to reduce stored energy in the pellets, or to improve other performance related characteristics, there should be small penalties in uranium or SWU requirements, or fuel cycle costs.

Figure 7-13. Equal Energy Plot Comparing Reduced Density Fuels

Figure 7-14. Reactivity Effects of Lattice Geometry Changes

7.7.4 Impact on Core Characteristics, Fuel Reliability, Safety

Reduced effective fuel density by use of annular pellet fuel and/or increase of H/U ratio with fuel rod diameter tend to improve transient characteristics (smaller steam void reactivity coefficients), but increase power-mismatch power peaking due to greater reactivity change with burnup. Axial power peaking is reduced and radial power peaking is increased. The cold temperature reactivity coefficients become positive as H/U ratio is increased. A criterion that the cold channel interior temperature and void coefficients should be negative usually must be met. The fuel time constant decreases with use of annular fuel or reduction in fuel rod diameter and this tends to adversely affect stability.

Annular pellets reduce pellet stored energy which aids LOCA. Annular pellets increase fission gas plenum space and reduce fuel temperature which reduces fission gas pressures. However, reduced fuel rod diameter with the same overall length of rod and active fuel has an opposite trend. The plenum volume is reduced and there is no significant affect on fuel temperature. Therefore, increase of H/U ratio by reduction of fuel rod size tends to increase fission gas pressure.

Annular pellets reduce pellet-clad-interaction (PCI) during power ramps, contributing to improved fuel reliability. However, reduced diameter solid pellets, operating at the same kw/ft as larger pellets have no significant effect on PCI.

Increase in H/U ratio through reductions in fuel rod diameter causes performance trends similar to those associated with reduction in fuel density.

7.7.5 Technical Uncertainties, Problems & Mitigating Measures

Although there is appreciable data on annular pellet fuel, there are not large scale LWR applications with appreciable interior pellet I.D. There also are some uncertainties in application of smaller rod diameter, greater H/U designs. Several of the uncertainties are:

1. Diffusion of fission products (I, Xe, etc.) in central hole and associated impacts on core reactivity. Power reactor irradiations have indicated greater annular pellet axial fission product migration which was dependent on pellet I.D.⁹ However, the overall effects cannot be easily measured except in large scale, full core applications of annular fuel;
2. Gap conductance, fuel temperature, fission gas release and fission gas pressure for annular pellets. Although there is data for annular fuel, the applicability of existing fuel models to prediction of annular fuel is not well established;
3. Impact on coupled nuclear-thermal-hydraulic stability. Stability is very sensitive to coupled space dependent gap conductance, radial pellet power shape, fuel time constant and steam void reactivity coefficient. Final confirmation of the impact of both annular pellet and smaller rod diameter on stability will require large scale application to an operating BWR;
4. Impact of water logging on annular pellet fuel. Local fuel rod power peaking will be increased by the presence of water inside the pellet column. Hence, water logging of annular pellet fuel may cause greater rod damage and greater release of fission products and fuel to the reactor water. Defected fuel tests will be required to determine the impacts and/or mitigating measures; and
5. Coupling with other improvement options. The incentives for annular pellet or reduced rod diameter, greater H/U ratio designs are strongly dependent on whether other improvement options such as spectral shift with flow control, are being applied. However, the degree to which these options could cause uranium utilization and fuel cycle losses when coupled with other improvement options has been quantified only to a limited extent.

7.7.6 Retrofitability

Annular pellets and smaller rod diameter, greater H/U ratio designs appear to be retrofitable to existing BWRs, provided no unexpected unresolvable problems are identified in resolving existing uncertainties.

7.7.7 RD&D Requirements

Reliable application of annular pellet and/or smaller fuel rod diameter greater H/U ratio designs will require extensive testing including full scale demonstrations and coupled detailed nuclear-thermohydraulic fuel measurements in operating power reactors. Power distribution, pressure transients and stability tests will be required, coupled with a fuels measurement program.

7.8 REDUCED BURNABLE ABSORBER RESIDUAL

7.8.1 Assumptions, Criteria, Methods of Approximation

Burnable absorbers are used in BWR design for excess reactivity control. In current design, self shielded gadolinia absorbers are mixed with UO_2 in a few selected fuel rods in each fuel assembly.

Because of the large thermal neutron absorption, cross section of two isotopes, Gd-155 and Gd-157, the gadolinia is highly self-shielded and burns as a cylinder with decreasing radius, thereby causing a linear positive reactivity with burnup.¹⁰ The even isotopes of gadolinia, Gd-154, Gd-156 and Gd-158, have a large enough neutron absorption that the residual effect on reactivity has a non negligible fuel cycle effect. Optimization of the spatial distribution of the gadolinia in the fuel can reduce the end-of-cycle gadolinia residual by causing more complete burnup of the gadolinia. Axial shaping of the gadolinia within reload fuel also improves axial power shapes.

The first large scale application of gadolinia to an operating LWR (Dresden-1, 1966)¹¹ incorporated axial gadolinia shaping. However, it has been the practice in recent years to utilize uniform axial gadolinia for purposes of simplifying fuel design, licensing and manufacture. Hence, there is potential for BWR uranium utilization improvement through axial shaping of gadolinia within the reload fuel.

Optimization of the gadolinia shape to maximize thermal margins usually also produces fuel cycle improvement potential, since increased thermal margins usually can be converted to fuel cycle improvements and reduction of uranium requirements. However, this optimization is enhanced if the cold shutdown requirements are relieved by increases in the worth of control blades and/or use of soluble absorber in the water for cold shutdown.

The potential uranium savings from application of axial gadolinia shaping in reload fuel were estimated from work completed in other independent programs which intercompared design alternatives with uniform and axially shaped gadolinia. These studies were carried out using 3-D coupled nuclear-thermal-hydraulic fuel cycle analyses of reactors different than, but similar to, the reference design.

7.8.2 U_3O_8 Requirements

Evaluations for designs prototypical of the reference design, with discharge burnup levels of approximately 28,000 MWd/MT and annual refueling cycles, indicated uranium savings of approximately 1% from gadolinia shaping optimization. However, the prevalent current utility trend is to operate with 18 month or longer intervals between refueling. The longer 18 month cycles increase the required gadolinia concentration in the fuel, which increases the reactivity residual. Extended burnup design requires greater numbers of fuel pins containing gadolinia, which also increases the reactivity residual penalty from the even gadolinia isotopes. It was estimated that gadolinia shaping optimization for greater burnup, longer refueling cycle length designs could result in incremental 2% uranium utilization savings potential for these designs. BWR initial cores frequently are designed for an 18 to 24 month first cycle length which also increases required gadolinia concentrations in the fuel. The overall uranium utilization savings potential from gadolinia spatial shape optimization was estimated to range from 1 to 3%, depending on the number of gadolinia pins and the concentration of gadolinia in the design.

7.8.3 Economic Incentives, Considerations

The economic incentives for more optimal gadolinia spatial distribution in the fuel are reasonably large because quality control systems are now in place in fuel manufacturing to make this feasible in the short term with minimal quality assurance and manufacturing costs.

A superconducting magnetometer with a field strength of up to 75 kg is used to measure and quality assure gadolinia weight percent, gadolinia zone length and gadolinia zone position and to identify any deviant pellets.¹² This is fully automated and integrated with a gamma ray densitometer to measure fuel density and detect any gaps between pellets, an ultrasonic wall thickness gauge to measure cladding thickness and an automated fuel rod handling system. There also have been important developments in automation of fuel design and licensing which aid in gadolinia shaping.

The optimum axial distribution of gadolinia in the fuel depends on the detailed way in which the reactor is operated. For conventional core designs, this dependency is very large because the control rods directly interact with the gadolinia burnup and affect the gadolinia burnup shape. Consequently, changes in control rod patterns compared to design assumptions can lead to major differences in the gadolinia burnup shape which causes performance and economic differences. Development of the Control Cell Core (CCC) design,^{13,14} which avoids insertion of control blades adjacent to fresh fuel containing unburned gadolinia, has greatly reduced the dependency of optimal gadolinia shaping on the specific path followed by the reactor operator. Therefore, the CCC design has greatly increased the practicality of achieving economic and thermal margin improvements from axial gadolinia shaping.

The design, licensing and manufacturing automation improvements and the development of the Control Cell Core have made it possible to consider gadolinia spatial distributions in the fuel which previously would not have been practical. Spatial distribution of gadolinia in the fuel increases design, licensing and fabrication costs; however, the cost increases are not as great as would have occurred without the automation that has been developed and the reliable incentives are considerably greater with the Control Cell Core. The cost increase must be compared with the fuel cycle and performance improvement incentives in determining whether to utilize spatial gadolinia optimization in the fuel. The economic and thermal margin improvement incentives were estimated to be appreciably greater for Control Cell Core, extended burnup, and/or extended cycle length designs.

7.8.4 Impact on Core Characteristics, Fuel Reliability, Safety

Use of axially distributed gadolinia in the reload fuel improves axial power and burnup shapes and reduces the need for shallow control rod insertion. Reduction of shallow control rod insertion simplifies implementation of pellet-clad-interaction (PCI) constraints, reduces start-up time and simplifies power distribution control during

transient situations, particularly for non-Control Cell Core designs which are penalized much more than the CCC by shallow rod insertion. Axial shaping that is built into the fuel assemblies, can in some cases, depending on the hot excess reactivity, reduce flexibility for axial power shape control during operation. Design methods accuracy requirements also are increased in some cases, since the axial shaping is inflexibly built into the design and fuel assembly well before the shutdown for refueling. The CCC design, as previously summarized, greatly reduces the dependence on operating history and the performance uncertainty by decoupling the gadolinia shaping from control rod interactions and thereby providing flexibility for control rod pattern adjustments to correct for any differences in reactor operations relative to design assumptions.

Because of the adverse effects of gadolinia on thermal conductivity, the gadolinia pins in a fuel assembly sometimes are the limiting pins with respect to fission gas pressure and other thermal-mechanical design and licensing considerations. Optimal spatial distributions of the gadolinia can help to mitigate the adverse impacts of the reduced thermal conductivity of gadolinia-UO₂ pellets.

7.8.5 Technical Uncertainties, Problems & Mitigating Measures

The technical uncertainties in application of optimal gadolinia shaping are discussed to a large part, in the previous sections. One of the difficulties in applying gadolinia shaping in the fuel is that although the reactivity effects of gadolinia can be qualified reasonably easily through critical experiments and application of Monte Carlo methods, it is appreciably more difficult to qualify predictions of gadolinia burnup rate. This requires power reactor irradiations coupled with careful non-destructive and destructive measurements such as gamma scans to measure power shape and isotopic measurements of gadolinia, heavy elements and burnup. Unfortunately these measurements usually are not practical except at end-of-cycle when the gadolinia is depleted. Also, the low level gadolinia residual, while significant in the fuel cycle, has a small few tenths of a percent effect on core reactivity which is difficult to separate from other uncertainties in the neutron absorption cross sections of the even gadolinia isotopes. Consequently, significant uncertainty exists in the prediction of gadolinia fuel performance.

7.8.6 Retrofitability

Axial gadolinia shaping has been used in numerous BWR fuel designs and is retrofitable with minimal RD&D requirements. Short axial gadolinia segments which have significant pellet end burnup effects require unique design methods for reliable analysis. Design with optimal gadolinia requires reliable and accurate methods and, for non-CCC designs, depends on the utility following operating schedules and plans close to those used as inputs to fuel design. Retrofitability of such designs, therefore, can depend on both design method accuracy and utility operating plans and procedures. The Control Cell Core is retrofitable to most BWRs and this design improvement increases the feasibility of retrofit of improved, reduced gadolinia residual designs to reduce uranium and SWU requirements, increase thermal margins and reduce fuel cycle costs.

7.8.7 RD&D Requirements

Table 1-6 summarizes the RD&D requirements to achieve the ultimate potential. Up to approximately 1% uranium utilization improvement is achievable with minimal RD&D through optimal spatial distribution of gadolinia burnable absorbers. Measurements of spatial isotopic compositions and burnup are needed for gadolinia fuels which are extracted from the reactor prior to depletion of the gadolinia (such a measurement is planned for a fuel assembly which was accidentally damaged in handling during a mid-cycle maintenance outage). Spatial gadolinia shaping may aid in mitigating design for extended fuel burnup. Gadolinia fuel rods, because of their reduced thermal conductivity, are frequently limited with regard to fission gas pressure. Spatial shaping and extended burnup designs generally create a need for increased local gadolinia concentrations which may create greater fuel performance and nuclear design uncertainties. These considerations create a need for advanced lead test assembly irradiations and associated follow-on fuels and nuclear measurements.

7.9 BORIC ACID CONTROL

7.9.1 Assumptions, Criteria, Methods of Approximation

Boric acid has been used extensively for reactivity control in PWRs and has been used in some early BWRs, EBWR for example. Although boric acid was used at power in EBWR, there is a potential problem from such use, because of carry-over to the turbine and potential attack on turbine blade materials. However, PWRs have operated with

boric acid in the secondary water due to steam generator leaks. Control rod worths are appreciably greater hot than cold and therefore, there is incentive to consider boric acid for cold shutdown control only, with clean-up before there is significant power operation. This reduces the requirements for burnable absorber and improves the capability to improve thermal margins with optimum spatial distribution of absorber, thus improving uranium utilization.

Evaluations of the potential for BWR improvement were carried out by use of very approximate 3-D coupled nuclear-thermal-hydraulic fuel cycle analyses. These analyses were designed to estimate the upper limit potential for improvement by assuming no burnable absorber residual in the fuel. Thus, the estimated savings are the upper limit which would be achievable with no burnable absorber in the fuel.

7.9.2 U_3O_8 Requirements

A 3-D multicycle evaluation performed with the reference "current improved BWR" design as a base case indicated about 4% uranium savings when no burnable absorbers are used in fuel designed for 28,000 MWd/MT discharge burnup with a 12 month cycle length. No credit was taken for improvements in rod enrichment distribution within the fuel assembly which would increase the uranium savings. At greater burnups or greater cycle lengths, which require greater number of burnable absorber rods and/or greater burnable absorber concentrations, these savings will be increased; however, practical constraints also will limit the potential. The potential uranium savings were estimated to range from 2 to 4% above the savings achievable with optimal burnable absorber designs as discussed in the previous section.

7.9.3 Economic Incentives & Considerations

Use of boric acid for cold shutdown must be balanced with clean up costs, any effects on materials; any requirements for boration of safety system water supplies, boration of fuel storage pools, boric acid concentration monitoring systems, effects on startup systems, effects on liquid wastes, impact on startup time and capacity factor, etc. If there were any impacts on capacity factor, this could quickly offset fuel cycle incentives.

Potential economic effects of introduction of new possible accidents (for example the "fresh water" accident) also must be considered.

Use of boric acid for cold shutdown could result in hide-out in fuel corrosion films or surface deposits. If there were a cold shutdown too close to the end-of-cycle for the boron to be depleted by neutron captures, such hide-out could adversely impact capacity factors, replacement power costs, and fuel cycle economics.

Because of all of these considerations, the use of boric acid on a regular basis at refuelings or shutdowns is considered to be marginal. However, experience has indicated that the probability of a BWR control rod sticking in the fully withdrawn position cold is very small. Therefore, it may be practical to consider use of boric acid only of cold shutdown in the unlikely, low probability event of a stuck control rod. This would reduce burnable absorber residual by reducing the required reactivity compensation of the burnable absorber. Such application of boric acid would be infrequent and the required boric acid concentrations would be reduced compared to more extensive control with boric acid in the water.

Limited application of boric acid would reduce costs and increase retrofit potential. In some cases this may increase the economic incentive for this improvement option.

7.9.4 Impact on Core Characteristics, Fuel Reliability, Safety

Use of boric acid for cold shutdown reduces constraints on the maximum permissible reactivity of the core, to further improve capability to power flatten and to maximize the conversion ratio and minimize the fissile inventory requirements of BWR fuel assemblies. When boric acid is routinely used for control, the "fresh water" accident must be considered and protected against. Boric acid also is very toxic and must be kept out of the environment. Corrosion and embrittlement of plant materials also could impact safety and must be carefully considered.

Boron hideout in fuel surface films, if there are any feasible mechanisms for rapid release, could create a power excursion. Boric acid in the moderator makes temperature and steam void reactivity coefficients more positive. If the positive reactivity coefficients were too large, this could prevent heatup with nuclear heating and require addition of a supplementary heat source for cold startups.

7.9.5 Technical Uncertainties, Problems & Mitigating Measures

Use of boric acid for cold shutdown will require investigation of the effects on reactor materials under BWR water chemistry conditions. The required boric acid concentrations would be reduced and limited by use of enriched boron.

Cleanup systems are normally in operation during BWR shutdown. Boron cleanup would have to be controlled by revisions in the systems and/or the mode of operation. Boron hideout on fuel or core material surfaces also would have to be controlled. Dissolution of fuel surface deposits by the boric acid could clean the fuel, but also could cause water clarity problems that might interfere with re-fueling operations.

Safety systems which inject coolant at low pressure (LPCI) would have to be modified to inject borated water. It also might be necessary to inject boron in the feedwater flow, if there were any possibility of the feed pumps being turned on during cold shutdown conditions. Control rod drive flow and the scram accumulator tanks also would have to contain borated water, unless it could be shown that dilution from this source could not cause a significant reactivity excursion.

The feasibility of nuclear heat-up, as is current BWR practice, versus use of some other means of preheat system for cold-to-hot start-up would have to be investigated.

7.9.6 Retrofitability

Provided no significant materials and boron hideout problems are identified, the use of boric acid for control should be retrofitable. If necessary, enriched boron with B-10 concentrations above natural could be considered to reduce required boric acid concentrations, although this will increase costs.

7.9.7 RD&D Requirements

Table 1-5 summarizes the potential RD&D requirements for maximal applications of boric acid for control. These requirements can be reduced by limited, lower concentration applications.

7.10 SIX-MONTH SUB-CYCLE FAST REFUELING

7.10.1 Assumptions, Criteria, Methods of Approximation

Fast refueling at six month intervals reduces the enrichment required to achieve a desired burnup, and therefore, has the potential to reduce some of the technological requirements for extended burnup. The refueling batch size is reduced and burnable absorber concentrations are reduced. Analyses of this design alternative were carried out with 3-D simulation methods, using the same enrichments as for 12 month cycles. Therefore, the resulting evaluations included the effects of short refueling cycles on burnup. This may or may not be realistic, depending on the extent to which extended burnup can be practically achieved with smaller refueling batch sizes. Detailed optimal refueling and control rod patterns were not developed.

The utilities currently are trending toward longer refueling cycle lengths to mitigate observed costs and longer than desirable outages during each refueling. Licensing of refueling also is a factor in this trend. In order to make shorter refueling cycles practical, it would be necessary to mitigate the licensing requirements and to provide high assurance that short 7 to 15 day refueling times could be consistently achieved.

It was assumed that three to four 6-month refueling could be pre-licensed and that major inspections and maintenance could be carried out at 18 to 24 month intervals. It also was assumed that fast refueling equipment and procedures could be retrofit to achieve 7 to 15 day refueling outages. These assumptions are keys to the practicality of six month sub-cycle refueling.

7.10.2 U_3O_8 Requirements

A 3-D multi-cycle, equilibrium refueling evaluation was performed for a BWR 6-month cycle. This evaluation was carried out with axial blankets and low neutron leakage refueling patterns, using the same fuel enrichment and fuel bundles as for the "current improved" BWR 12-month cycle, but with reduced burnable absorber content and more frequent, fast refueling. This 3-D evaluation resulted, relative to the "current improved BWR" 12-month reference case, in a fuel discharge burnup increase of 15% and uranium and separative work requirement reductions of 13 and 14 percent, respectively. The results are summarized in Table 7-3. At greater burnups, it might have been necessary to reduce the reload fuel enrichment requirements, thereby reducing the uranium savings. The estimated range of potential uranium utilization improvement is 5-15%, depending on the extent to which this alternative is used to extend fuel burnups. The incentives for the 6 month sub-cycle refueling improvement alternative are one of the greatest of the improvements evaluated.

TABLE 7-3

6 MONTH SUB-CYCLE FAST REFUELING 3-D MULTICYCLE EVALUATION

	<u>6-MONTH SUB CYCLE REFUELING*</u>	<u>B W R CURRENT IMPROVED 12-MONTH CYCLE*</u>	<u>6-MONTH PERCENT SAVINGS</u>
● AVERAGE CAPACITY FACTOR (%)	75.00	75.00	
● FRACTION OF BATCH RELOAD/ REFUELING	0.11	0.24	
● NUMBER OF FRESH FUEL ASSEMBLIES/REFUELING	96.00	200.00	
● FUEL PELLET DENSITY (% OF NOMINAL)	100.00	100.00	
● RELOAD ENRICHMENT	2.75	2.74	
● REFUELING INTERVAL (YRS)	0.55	1.00	
● CONVERSION LOSS FRACTION	0.005	0.005	
● FABRICATION LOSS FRACTION	0.01	0.01	
● DISCHARGE FUEL BURNUP (MWD/MTU)	32,700.00	28,439.00	
● U_3O_8 REQUIREMENTS (ST/GWE-YR)** EQUILIBRIUM RELOAD	164.60	190.00	13%
● SEPARATIVE WORK REQUIREMENTS (10^3 SWU/GWE-YR) EQUILIBRIUM RELOAD	96.00	111.00	14%

---

* Axial Blankets & Low Leakage Fuel Management

** ^{235}U Enrichment 0.2 w/o

7.10.3 Economic Incentives, Considerations

In 1979, a BWR refueling of 15 days was achieved, coupled with considerable fuel sipping and some maintenance of control rod drives. Seven to fifteen day fast refueling at six month intervals within 18 or 24 month pre-licensed cycles may make consideration of this improvement economically desirable. Proliferation of inspection and maintenance during refueling and associated capacity factor losses, which increase dependence on expensive oil or other replacement power and increase refueling costs are serious difficulties which have made consideration of this alternative impractical in the past.

Fuel movement can be minimized in BWR design by use of refueling patterns in which the fresh fuel is inserted in vacant positions created by the discharge fuel, with no fuel shuffling. Such designs may or may not be desirable, depending on whether fuel movement is on the critical path during fast refueling outages. It is feasible to design so as to minimize fuel movement during the fast sub-cycle refuelings, coupled with more extensive fuel shuffling during the major inspection/maintenance outages.

The economic incentives for 6 month sub-cycle refueling are sensitive to many factors including outage time and capacity factor effects, personnel requirements, practicality of outage timing to schedule the outage during lows in power demand, personnel radiation exposures, practicality of pre-licensing several sub-cycle refuelings and/or carrying out refuelings under 10CFR50.59 criteria and practicality of avoiding outage time increases due to proliferation of inspection and maintenance. Six month sub-cycle refueling is probably impractical for utilities that have an annual major low in nuclear power demand, for example utilities with major spring hydro resources.

7.10.4 Impact on Core Characteristics, Fuel Reliability, Safety

The small refueling batch fraction (0.11 for the 6-month cycle design summarized in Table 7-3) may require that special measures be developed for excess reactivity and power distribution control. Unique axial gadolinia shaping may be optimal for the 6-month sub-cycle refueling and detailed strategies must be developed to maintain the greatest burnup fuel on the core periphery to minimize neutron leakage. There will be a larger mismatch between discharge and fresh fuel which must be carefully managed to avoid radial power peaking penalties.

Many of the potential problems associated with extended burnup designs should tend to be mitigated by 6 month sub-cycle refueling. Enrichments are reduced, gadolinia concentrations are reduced, reactivity coefficients are less negative, and excess reactivity requirements are reduced.

7.10.5 Technical Uncertainties, Problems & Mitigating Measures

Technical uncertainties are primarily related to the accuracy of prediction of power distributions and other performance related parameters for the smaller refueling batch sizes associated with 6-month sub-cycle refueling. There also are uncertainties relative to the degree to which refueling times can be reliably minimized. A practical demonstration at an operating plant is needed to provide answers to many of the open questions. In today's environment of licensing uncertainty, it is very unlikely that any utility would risk a multiple 6 month refueling cycle demonstration.

7.10.6 Retrofitability

Six month sub-cycle refueling, including any special equipment required to minimize refueling time, can be relatively easily retrofit to current operating BWRs. For example, automated fast refueling platforms and vessel head stud tensioners have been developed for retrofit. One of the major issues relative to retrofitability is whether the required pre-licensing and improved inspection/maintenance procedures can be retrofit.

7.10.7 RD&D Requirements

The RD&D requirements are summarized in Tables 1-5 and 1-6. Pre-licensed multiple refueling outages are being considered for other reasons under 10CFR50.59. Fuel bundle and fuel management design revisions will have to be developed and verified. Revised licensing and inspection/maintenance procedures will have to be developed. A first phase demonstration could be carried out with minimum risk by using existing standard reload fuel assemblies and thus, not obtaining the benefits of reduced gadolinia concentrations and residual. However, this would have to be carried out for several refueling intervals to obtain data beyond that now available.

7.11 ONCE-THROUGH THORIUM CYCLE

7.11.1 Assumptions, Criteria, Methods of Approximation

Problems in application of thorium cycles include greater fissile inventory requirements,¹⁵ the blending of highly enriched uranium or plutonium with thorium, and the necessity to recover and recycle the valuable U-233 produced in order to recover the costs of the initial inventory and enrichment. With these problems in mind, a preliminary "once-through" thorium cycle was evaluated for Boiling Water Reactors which minimizes the effect of thorium on the fissile inventory, is initiated with ThO₂ fuel containing no initial fissile material, and does not require U-233 recovery and recycle. The design makes advantageous use of the inherent heterogeneity and other characteristics of the BWR lattice to produce U-233 in ThO₂ without power distribution penalties and to improve reactor performance (thermal and transient margins). Standard BWR fuel assembly hardware was used to make the design backfitable with minimum manufacturing impact.

The design utilizes heterogeneous location of thorium in the highest thermal flux locations of the BWR lattice^{16,17} to maximize U-233 production rate during burnable absorber depletion.

Figure 7-15 is a typical BWR lattice arrangement showing the location of the pure ThO₂ rods. These maximum thermal neutron flux locations maximize U-233 production rate, maximize the reactivity value of the U-233 and maximize the "fertile-poison" effects of ThO₂, thereby maximizing the replacement of gadolinia burnable absorber with fertile absorption in thorium. In essence, in this design, some of the neutrons which would have been wasted in gadolinia absorptions are captured in Th-232 to produce the valuable fissile isotope, U-233.

After fuel assembly discharge, the U-233 bearing ThO₂ rods are re-assembled into breeder blanket bundles (BBBs) that are located at the core periphery. The breeder blanket assemblies have breeding ratios of 1.04 to 1.08 (Figure 7-16) and fissile inventory ratios of 1.05 to 1.7, depending on conditions in the pre-breeder irradiation. The reactivity is very flat with burnup (Figure 7-17) and; therefore, the residence time of the breeder blanket bundles is limited by mechanical integrity (corrosion, hydriding, etc.), not by reactivity lifetime.

It is current practice in BWR design to reduce neutron leakage from the core by locating either natural uranium (first core) or last cycle fuel on the core periphery.¹⁸ In the once-through thorium cycle, the BBBs replace the last cycle fuel on the core periphery. Thus, once the BBBs are in place, the number of bundles needed

Figure 7-15. Typical Preblanket Lattice

Figure 7-16. Blanket Assembly Breeding Ratio Versus Burnup

Figure 7-17. Infinite Lattice Neutron Multiplication Factors Versus Burnup

each cycle to sustain the reactor is reduced.

The concept may make it feasible to recycle U-233 without reprocessing, by recovering the ThO₂ fuel rods from BWR discharge fuel and by recycling these fuel rods into blanket bundles. If there is an excess of ThO₂ recycle rods beyond those needed to fill out the blanket, these rods can be recycled to make up the blankets of new reactors or, provided fuel reliability is adequate, the rods could be reinserted in lower neutron flux locations of fresh reload fuel assemblies.

Within the available resources, it was not feasible to carry out detailed 3-D coupled nuclear-thermal-hydraulic fuel cycle evaluations of the once-through thorium cycle concept. Studies were completed to compare the infinite fuel bundle lattice characteristics of the pre-breeder and breeder blanket bundles with similar infinite fuel bundle lattice analyses for reference fuel assemblies. The uranium utilization improvements were estimated by evaluating the impact of the breeder blanket bundles on the fresh fuel assemblies required in each reload batch.

Prior analyses¹⁹ have indicated improved performance from application of thorium fuel in BWRs. These studies have indicated that the prebreeder bundles have 5 to 15% less negative steam void reactivity coefficients, larger prompt negative Doppler reactivity coefficients, and flatter reactivity slope with burnup. The potential beneficial impacts of the performance improvements on uranium savings were not considered in the evaluation.

Analyses of the PBB and BBB bundle designs have been carried out with a 99 energy group spatial/spectral model. However, because of uncertainties in the analysis of mixed lattices of ThO₂/UO₂ rods, it is desirable that detailed full spectrum Monte Carlo analyses, additional experimental benchmark analyses and detailed 3-D coupled nuclear-thermal-hydraulic-fuel cycle analyses be completed. There may be important 3-D core interaction effects with the breeder blanket bundles which are expected to have a positive steam void reactivity coefficient. It may be possible for example, to optimally adjust the core inlet orificing to increase flow in the central regions of the core when the thorium assemblies are loaded in the blanket region.

7.11.2 U_3O_8 Requirements

Table 7-4 is a summary of material flow for a once-through BWR thorium application. The blanket is generated from the pre-breeder bundles (PBBs) in eight to ten cycles of operation. If PBBs were loaded throughout the lifetime of a BWR/6, three complete BBB blankets would be generated. Excess rods from the PBBs may be used in fresh fuel or may be used to fuel the blankets of other reactors. Earlier generation BWR/1-3s could incorporate greater number of ThO_2 rods in the PBBs and, therefore, could generate the blankets more rapidly as well as generating more excess U-233/ ThO_2 rods for other reactors.

As the number of blanket assemblies increase, the refueling batch size decreases as a result of the core becoming effectively smaller. If it is assumed that no excess ThO_2 rods are created or that the longest residence time blanket fuel rods are replaced with any excess ThO_2 rods, the uranium savings are estimated to be approximately 4%. If the low power blanket rods are able to remain in the reactor without replacement and the excess ThO_2 rods could be used to make up blankets of other reactors, the uranium savings were estimated to increase to approximately 10%.

These estimates did not include consideration of potential U-233 recovery and recycle. Because of the approximately 3 times smaller absorption cross section of U-233 as compared to Pu-239 and the larger absorption cross section of Th-232 as compared to U-238, the U-233 inventory in the thorium rods builds up to levels that typically are 3 or more times greater than the fissile Pu content of a uranium rod. U-233 also has better thermal and resonance spectrum nuclear properties than Pu-239 and studies have indicated that some of these properties may lead to performance advantages when U-233 is applied in BWRs.^{19,20} Consequently, there may be uranium savings incentive to recover the U-233 and to consider recycle in depleted uranium, diffusion plant tails, natural uranium, depleted uranium plus recycle plutonium, thorium diffusion plant tails blends, thorium or other optimal combinations of fertile-fissile material. If recovery and recycle were projected feasible and desirable in the future, this could increase the uranium savings incentives for this improvement alternative.

7.11.3 Economic Incentives & Considerations

A number of factors will affect the fabrication cost of the ThO_2 fuel rods. Thorium is approximately 3 times more abundant in the earth's crust than uranium. Therefore, the costs of thorium would be expected to remain below uranium. However, the biologically significant natural radioactivity from thorium is greater than for uranium, which could affect fabrication costs. ThO_2 containing no fissile material is not considered to be a special nuclear

TABLE 7-4

MATERIAL FLOW IN A 748 BUNDLE, 110 MWE
BWR/6 ONCE-THROUGH THORIUM CYCLE EXAMPLE

<u>END OF CYCLE#</u>	<u># OF PBBs DISCHARGED CYCLE</u>	<u># OF BBB RODS PRODUCED CYCLE</u>	<u>TOTAL ThO₂ RODS AVAILABLE</u>	<u>TOTAL BBBs</u>
1	---	---	---	---
2	192	768	768	9
3	151	608	1376	16
4	180	1360	2736	33
5	188	752	3488	43
6	176	704	4192	51
7	180	720	4912	60
8	180	720	5632	69
9	180	720	6352	78
10	180	720	7072	87*
11-30	180	720	21472	265**

---

* First blanket of (84 bundles) BBBs completed.

** Does not include ThO₂ rods from 1, 2, and 3 cycle fuel.

material. However, it probably will be necessary to fabricate ThO₂ pellets in a separate line and avoid any possible co-mingling of thorium and uranium. The fabrication costs of the ThO₂ rods are not well established.

The cost of reconstituting the ThO₂ rods into blanket fuel assemblies also is uncertain. If this is combined with reconstitution and reinsert of high enrichment discharge fuel rods, then incremental costs should be small. It probably will be most desirable to reconstitute into new fuel assembly hardware to assure long life-time of the blanket assemblies.

The economic incentives for the once-through thorium cycle designs are expected to be dependent on how optimally the core is designed and operated. If the reactor is being operated to extended burnup with maximal spectral shift control, feedwater temperature reduction and end-of-cycle coastdown, the economic incentive for the once-through thorium design should increase as a result of beneficial effects on gadolinia requirements and neutron leakage and as a result of the reactivity of the breeder blanket increasing relative to the reactivity of discharge fuel.

Design optimization to achieve economic credits for thermal or licensing margin improvements attributable to the thorium fuel should increase the economic incentives. Lifetime of the blanket assemblies, which will determine whether an excess of thorium/U-233 rods are being created, will have a significant economic impact. Longer term recovery of the U-233 also could be important.

Many of the economic parameters are uncertain for the once-through thorium design. Test and demonstration programs will be required to reduce these uncertainties.

7.11.4 Impact on Core Characteristics, Fuel Reliability, Safety

Although 3-D analyses were not completed for this design, some of the effects on core characteristics can be estimated from the infinite lattice fuel assembly characteristics. There also is considerable information available on the physical properties of ThO₂ which affect fuel performance and reliability.

Analyses of the pre-breeder bundle performance indicate that the ThO₂ rods make the BWR steam void reactivity coefficient 10-15% less negative due to combined effects of the ThO₂ rods on thermal utilization and resonance capture, coupled with the large U-233 resonance fission integral and relatively large epithermal fission neutron production per capture. The net effect is that axial power shapes and burnup dependent fuel assembly power mismatch should be improved as should performance in low probability pressure transients. The requirements for burnable absorbers also are decreased with a beneficial effect on gadolinia residual.

Figure 7-18 is a plot of the infinite fuel assembly neutron multiplication for a reference UO₂ fuel assembly and a similar "pre-blanket" fuel assembly containing four ThO₂ rods in the corners as shown in Figure 7-15. The number of Gd₂O₃ fuel pins were reduced by 25% in the pre-breeder design as compared to the reference design. The design containing ThO₂ rods has a flatter burnup dependent reactivity shape which will improve radial core power distributions and will contribute to improving the incentives for other improvement options which shift the neutron spectrum and extend the cycle length (flow control spectral shift, end of cycle power coastdown and feedwater temperature reduction, etc.).

The use of ThO₂ rods in the design increases local power peaking, as shown in Figure 7-19. Local power peaking is not as important when the reactivity of the fuel assembly is reduced either by burnable absorbers (0-5500 MWd/MT) or by fuel depletion and fission product buildup (>16,500 MWd/MT). Hence, the local peaking differences between 5,500 and 16,500 MWd/MT are the most important. In this range, the pre-blanket bundle has about 1-2% greater local power peaking. At larger burnups, the buildup of U-233 in the ThO₂ rods causes these rods to become the peak power rods in the fuel assembly. The rise in power of the ThO₂ rods at long burnup is the cause of the sharp increase in local peaking of the pre-blanket bundle beyond 33,000 MWd/MT (Figure 7-19). Because of the low reactivity of the fuel at the burnups at which this occurs ($k_{\infty} < 1.0$) peak power cannot occur in the ThO₂ rods and the local peaking increase is not important from consideration of thermal margins. However, the increase in power in the ThO₂ rods may be significant from considerations of pellet cladding interaction (PCI) and fuel reliability and this will have to be investigated, including consideration of the differences in properties between ThO₂ and UO₂, which also may affect fuel reliability (see following discussion).

Figure 7-18. K-Infinite Versus Exposure for Reference UO₂ and Preblanket Designs

Figure 7-19. Maximum Local (P/A) Power Versus Exposure for Reference UO₂ and Preblanket Designs

Fuel assemblies which are designed for greater fuel burnups and greater refueling cycle lengths must incorporate greater numbers of larger concentration Gd_2O_3 - UO_2 rods. Hence, as cycle length and burnup are increased, the impacts of the ThO_2 on local power peaking are likely to decrease and the beneficial impacts of the ThO_2 on cold shutdown margins and steam void reactivity coefficients are likely to become more important.

Figure 7-20 is a plot of the dynamic steam void reactivity coefficient of the reference UO_2 and pre-blanket designs. These analyses indicated that the steam void reactivity coefficient was approximately 10 percent less negative, while prior evaluations^{17,19} with a different design, had indicated that the steam void reactivity coefficient was 15 percent less negative. A less negative steam void reactivity coefficient generally improves reactor performance by increasing coupled nuclear-thermal-hydraulic stability and improving performance in low probability, licensing basis pressure transients. Because of the latter effect, the less negative steam void reactivity coefficients will increase margins to the onset of transition boiling (MCPR margins).

Table 7-5 is a summary comparison of UO_2 and ThO_2 thermal/mechanical properties. In general, ThO_2 is a better refractory material than UO_2 and appears to have a number of important properties, which will provide performance and reliability advantages over UO_2 , as summarized in Table 7-5. The reduced fission gas release and greater thermal conductivity of ThO_2 may be particularly important in making feasible long, reliable residence time of the breeder blanket bundles. The greater stability of ThO_2 in water, steam and in the products of radiolytic decomposition of water should result in significantly smaller release of fission products to the reactor water if any cladding defects occur.

7.11.5 Technical Uncertainties, Problems & Mitigating Measures

Thorium base fuels have not been used extensively in water reactors and therefore, the uncertainties are appreciable in some areas. Although, considerable RD&D has been completed as part of the LWBR program, there are appreciable practical design differences relative to the once-through concept being considered for the BWR and the applicability of the data and design methods developed under the LWBR program is uncertain. The uncertainties in basic neutron cross-section data for $Th-232$ and $Pa-233$ is appreciable. Generally, available thorium light water critical assemblies cannot be predicted

Figure 7-20. Dynamic Void Reactivity Coefficient for UO₂ and Preblanket Designs

TABLE 7-5

ThO₂ CHARACTERISTICS WHICH MAY IMPROVE THERMAL MECHANICAL PERFORMANCE/LIFETIME

<u>PROPERTY</u>	<u>ThO₂ VERSUS UO₂</u>	<u>AFFECTS</u>
MELTING POINT	ThO ₂ 700 F GREATER	IMPROVED TRANSIENT PERFORMANCE
THERMAL CONDUCTIVITY	ThO ₂ 20-50% GREATER	IMPROVES LOCA AND TRANSIENT PERFORMANCE; HURTS STABILITY; REDUCES PCI AND FISSION GAS RELEASE.
HEAT CAPACITY	ThO ₂ 5-40% SMALLER	IMPROVES LOCA AND TRANSIENT PERFORMANCE; HURTS STABILITY.
LINEAR EXPANSION COEFFICIENT	ThO ₂ 20% SMALLER	REDUCES PCI
FISSION GAS RELEASE	ThO ₂ 2-10 TIMES LESS	CLADDING DESIGN, PLENUM VOLUME AND DESIGN BURNUP
FISSION PRODUCT SWELLING	ThO ₂ 10% SMALLER	CLADDING DESIGN; PCI; FUEL DENSIFICATION
OXIDE STATES	ThO ₂ HAS SINGLE OXIDE STATE, MORE STABLE THAN UO ₂	CORROSION IN STEAM/WATER; FISSION PRODUCTS AND OFFGAS FROM LEAKERS

with the same degree of accuracy as for UO₂ criticals.¹⁹ Significant uncertainties in the basic nuclear data could affect reliability of design and performance predictions.

Use of mixed ThO₂ and UO₂ fuel rods in a BWR lattice introduces complexity in the reactor physics which may not be reliably modeled by current design methods. The Th-232 and U-238 resonances do not overlap appreciably¹⁶ and therefore, there is no Dancoff resonance shielding interaction between neighboring U-238 and Th-232 fuel rods. This effect was not accurately modeled in the evaluations.

The basic physical properties of ThO₂ indicate considerable potential for improved performance, however, there is considerable uncertainty in this area because of the lack of prototypical fuels irradiation data. There also is uncertainty on the performance of zircaloy cladding in the reduced power core peripheral regions for the very long core residence times desirable for the breeder blanket bundles. The uncertainty is compounded by the large time required to obtain the needed data.

7.11.6 Retrofitability

Provided no unexpected problems are identified from more detailed 3-D analyses and from test and demonstration programs, the design should be retrofitable to all BWRs. Thorium dioxide fuels have previously been applied in several BWRs (Elk River, Dresden-1, Lingen) with no major problems.

7.11.7 RD&D Requirements

Table 1-6 includes a summary of the RD&D requirements for the once-through thorium cycle in BWRs. Increases in availability of detailed information from the LWBR program could reduce many of the basic technology requirements.

8. COMBINED IMPROVEMENT EVALUATIONS

One of the primary objectives of the evaluations was to determine the overall range of BWR uranium utilization improvement potential. The individual alternatives discussed in the previous section are not necessarily additive and therefore, combinations of alternatives had to be considered in order to determine realistic ranges of BWR improvement. Table 8-1 is a summary of the combined improvements considered.

8.1 ASSUMPTIONS, CRITERIA, METHODS OF APPROXIMATION

Several of the alternatives cannot be evaluated realistically using simplified analysis methods and therefore, the combined evaluations, by necessity, had to be carried out with 3-D coupled nuclear-thermal-hydraulic-fuel cycle simulations. The simulations were simplified by considering the equilibrium core, only. Initial core and transition cycle effects were estimated based on known trends and the differentials from the equilibrium cycle evaluations. 3-D analysis complexities, coupled with available resources made it impossible to include all of the potential improvements in the combined evaluations. For example, it was not possible to combine the once-through thorium cycle with the other improvement alternatives. This should tend to make the evaluated combined improvement potential conservative.

In some cases, combinations of improvements enhance the overall worth. For example, BWR spectral shift can be used either to reduce fuel enrichment requirements for a fixed number of assemblies in the reload batch or to extend burnup with constant fuel enrichment and a reduced number of reload fuel assemblies. The latter alternative will result in greater uranium savings. Also, extending burnup with improvement alternatives such as spectral shift, which does not require a fuel enrichment increase, mitigates many of the power peaking and other problems associated with designing for extended burnup. Therefore, in some cases the larger uranium and SWU savings associated with combinations of several improvements may be easier to achieve than the smaller savings associated with individual improvements.

In some cases, combinations of improvements interact to affect the optimal combination. For example, it was found that reduced pellet density saved uranium for burnups near those of current design. However, at extended burnups, the opposite was true and reduced pellet density increased uranium requirements. This was due to the optimal H/U ratio being burnup dependent.

It was not possible to complete an overall optimization of the application of the improvement alternatives considered. Such a optimization would be expected to increase the uranium and SWU savings.

Although, 3-D fuel cycle evaluations were carried out, these were not iterated to maintain compliance with a fixed set of thermal and other constraining limits. Since some of the alternatives, such as extended burnup, tend to depreciate thermal and other margins this approximation should tend to make the results optimistic. However, there are potential mitigating design measures which can be applied to offset these tendencies.

An overall detailed design and simulation evaluation will be necessary to provide the information necessary to assess the importance of the approximations and to couple the pessimistic and optimistic assumptions. On the basis of experience in previous evaluations, the 3-D assessments are expected to be a reasonable estimate of the BWR improvement potential. This is to some extent borne out by the independent Norwegian assessment of BWR improvement potential included in the INFCE reports which indicated greater BWR savings potential than within the evaluations.²¹

In the 3-D evaluations, a sufficient number of repetitive refuelings were simulated (usually more than 8) until equilibrium conditions were reached. Refueling batch size was iterated to adjust the equilibrium refueling interval to the target cycle lengths.

8.2 ALTERNATIVES CONSIDERED & BASES FOR SELECTING COMBINATIONS OF IMPROVEMENTS

Of the alternatives listed in Table 1-1, the following alternatives were included in combined evaluations (Table 8-1):

- Axial Blankets
- Improved refueling patterns with last cycle, low reactivity fuel on periphery (low leakage refueling)
- Spectral Shift using flow (80-120% of rated)
- Feedwater Temperature Reduction (420-250°F)

- Thermal Coastdown to 80% Power
- Increased Enrichment and Burnup
- Reduced Gadolinia Residual
- High Enrichment Discharge Fuel Rod Reassembly and Reinsert
- Reduced Fuel Pellet Density (80% of Reference)

As indicated in Table 8-1, both 12 and 18 month cycles were evaluated for the "current improved" and one set of combined features in order to provide a realistic assessment of refueling cycle length effects on uranium and SWU resource requirements. A "current improved" and combined improvement plutonium recycle design also was evaluated to determine the resource requirements for Pu recycle and the effects of the improvements on Pu recycle. Pu recycle is discussed in the following Section.

Some improvement alternatives are not compatible, for example, increased burnup with increased enrichment and six-month sub-cycle fast refueling. Reduced gadolinia (burnable absorber) residual was limited to that estimated to be achievable from optimization of fuel bundle design coupled with limited use of boric acid for cold reactivity shutdown control in low probability events, such as inability to insert a control rod.

Table 8-1 is a summary of the combinations of improvements that were evaluated. The "current improved BWR" features (axial blankets plus low leakage refueling) were applied in all the combined evaluations. Available resources prevented inclusion of the once-through thorium cycle, six month sub-cycle refueling and lattice and fuel bundle design optimization.

8.3 3-D COMBINED IMPROVEMENT EVALUATIONS FOR HIGH ENRICHMENT BUNDLE

Three combined improvement 3-D evaluations were performed using the same increased enrichment 3.65 w/o U235 reload bundle. This fuel assembly design did not incorporate the increased gadolinia requirements that normally would be expected for this enrichment level. Therefore, the design is based on assumption of reduced gadolinia residual compared to current design practice.

TABLE 8-1

SUMMARY OF COMBINED IMPROVEMENTS CONSIDERED IN 3-D EVALUATION

IMPROVEMENT ALTERNATIVES	12 MONTH CYCLE								18 MONTH CYCLE		PU RECYCLE +	
	CIB*	A	B	C	D	E	F	G	H**	I	J	K
● AXIAL BLANKETS	X	X	X	X	X	X	X	X	X	X	X	X
● LOW LEAKAGE REFUELING	X	X	X	X	X	X	X	X	X	X	X	X
● EXTENDED BURNUP WITH INCREASED ENRICHMENT												
--3.3 W/O U-235								X				
--3.65 W/O U-235		X	X	X	X					X		
--4.55 W/O U-235						X						
● FLOW CONTROL SPECTRAL SHIFT												
--90-105% FLOW RANGE								X				
--80-120% FLOW RANGE			X	X			X			X		X
--70-120% FLOW RANGE					X	X						
● EOC FEEDWATER TEMPERATURE REDUCTION (TO 250°F)			X	X	X	X	X	X		X		X
● EOC POWER REDUCTION (TO 80% POWER)			X	X	X	X	X	X		X		X
● REDUCED GADOLINIA RESIDUAL		X	X	X	X	X	X	X		X		X
● HIGH ENRICHMENT DISCHARGE FUEL ROD RECONSTITUTION & REINSERT				X			X	X				
● REDUCED FUEL PELLET DENSITY (80% OF REFERENCE)					X	X						

* Designated the "Current Improved BWR" Case (2.74 w/o U-235 enrichment).

** Current Improved 18 Month Cycle (2.99 w/o U-235 enrichment)

+ 12 Month Cycle

GEAP-24965

The evaluations are summarized in Table 8-2 with the "reference BWR" and "current improved BWR" cases for comparison. Case A in Table 8-2 included blankets and low leakage fuel management plus extended burnup, so it is the "current improved BWR" with the addition of extended burnup. Case B includes most of the options in Table 8-1 except high enrichment discharge fuel rod reassembly/reinsert. Case C is the same as Case B with the addition of high enrichment discharge fuel rod reconstitution and reinsert.

8.3.1 Extended Burnup and Reduced Gadolinia Residual (Case A)

The cycle length for these evaluations were kept as close to 12 months as possible. A smaller reload batch fraction gives better uranium ore utilization, that is, the most efficient method of fuel management would be the extreme case of on-line refueling. Therefore, one of the major benefits of extended burnup is reduction of the reload batch fraction. With the cycle length and batch fraction as constraints, the simplified model discussed in Section 5.1 was used to estimate the bundle enrichment required.

The extended burnup case reduced uranium and SWU requirements by 23 and 17 percent, respectively, relative to the reference design and by 18 and 8 percent respectively, relative to the "current improved BWR" design. These estimates were discounted in the overall summary (Table 1-1) to reflect expected depreciation of power peaking and requirements for additional burnable absorber control.

8.3.2 Extended Burnup, Reduced Gadolinia Residual, Spectral Shift, FWTR, Coastdown (Case B)

Case B summarized in Table 8-2 is the same as Case A with the addition of spectral shift using flow (80%-120% of rated), EOC feedwater temperature reduction from 420°F to 250°F and EOC thermal power coastdown to 80% power. The capacity factor, reload batch fraction and reload enrichment are the same as in Case A. The additional improvement alternatives increased discharge burnups to 5,000 MWd/MT and reduced annual uranium and separative requirements by 28 and 23 percent, respectively, relative to the reference design. The incremental uranium and SWU savings relative to Case A were 5 and 6 percent, respectively.

8.3.3 Reconstitution and Reinsert of High Enrichment Discharge Fuel Rods Plus Other Improvements (Case C)

Case C has all the improvement alternatives contained in Case B, plus the high enrichment discharge fuel rod reassembly/reinsert improvement option. For the reference 75% capacity factor and 3.65 w/o bundle enrichment, the equilibrium reload batch fraction was 0.20, made up of 112 fresh fuel bundles and 56 reconstituted bundles. Each reconstituted reinsert bundle is reassembled from two 3.65 w/o

TABLE 8-2

COMPARISON OF COMBINED 3-D EVALUATIONS FOR 3.65 W/O U235 BUNDLE ENRICHMENT

	REFERENCE BWR	CURRENT IMPROVED BWR	CASE A**	CASE B***	CASE C****
● AVERAGE CAPACITY FACTOR (%)	75	75	75	75	75
● FRACTION OF BATCH REPLACED/REFUELING	0.24	0.24	0.14	0.14	0.20
● NUMBER OF FRESH FUEL ASSEMBLIES/REFUELING	200	200	120	120	112
● NUMBER OF REINSERT ASSEMBLIES/REFUELING	---	---	---	---	56
● FUEL PELLET DENSITY (% OF NOMINAL)	100	100	100	100	100
● RELOAD ENRICHMENT	2.94	2.74	3.65	3.65	3.65
● REFUELING INTERVAL (YEARS)	1.0	1.0	0.99	1.07	1.01
● CONVERSION LOSS FRACTION	0.005	0.005	0.005	0.005	0.005
● FABRICATION LOSS FRACTION	0.01	0.01	0.01	0.01	0.01
● DISCHARGE FUEL BURNUP PRIOR TO REINSERTION (MWD/MT)	28439	28439	46600	51000	49000
● DISCHARGE BURNUP OF REINSERTED FUEL (MWD/MT)	---	---	---	---	54500
● U ₃ O ₈ REQUIREMENT (ST/GWE)					
--ANNUAL EQUILIBRIUM RELOAD	204	190	156.4	146.1	144.8
● SEPARATIVE WORK REQUIREMENT (10 ³ SWU/GWE)					
--ANNUAL EQUILIBRIUM RELOAD	124	111	102.5	95.7	94.9
● PERCENT REDUCTION COMPARED TO REFERENCE CASE (%)					
--U ₃ O ₈ REQUIREMENTS					
-- ANNUAL EQUILIBRIUM RELOAD	---	6.9	23.3	28.4	29.0
--SEPARATIVE WORK REQUIREMENTS					
-- ANNUAL EQUILIBRIUM RELOAD	---	10.5	17.3	22.8	23.5

GEAD-14965

** Blankets & low leakage fuel management (current improved BWR) plus extended burnup.

*** See Table 8-1 for improvement alternatives included.

**** Same as B + reconstitution and reinsert of high enrichment discharge fuel rods.

discharge bundles. It was assumed that all discharge fuel assemblies would be reconstituted, independent of their discharge burnup. This assumption reduces the burnup of the 3.65 w/o fuel. The discharge burnup of the 3.65 w/o fuel was 49,000 MWd/MT. The high enrichment reconstituted discharge fuel rod reinsert fuel reaches a burnup of 54,500 MWd/MT. The reconstitution option results in small incremental uranium and SWU savings of 0.6 and 0.7 percent, respectively, or considerably less than that observed at lower burnups (Section 7.5.2). This may be in part due to the effects of extended burnup. However, it also appears that non-optimum reconstitution may be a contributor. Optimal selection of fuel assemblies to be reconstituted from the spectrum of burnups in the discharge batch and more optimum reconstituted fuel assembly reinsert patterns should increase the savings. Such detailed optimizations were outside the scope of this program.

8.4 REDUCED PELLETT DENSITY, INCREASED H/U RATIO

The pellet density was reduced to 80% of the reference design density on the basis of prior analyses which had indicated that this was near optimum. Two evaluations were carried out, one in which the enrichment of the pellet was held constant and another in which the pellet fissile content was held constant. An increased enrichment extended burnup design, Case B in Table 8-2, was used as the reference for the pellet density perturbation. The reduced effective pellet density could be achieved by use of annular fuel, sphere pac fuel, etc.

Table 8-3 is a summary of the results from the 3-D evaluations. As would be expected, there is significant increase in uranium and separative work requirements (Case D) when the pellet enrichment is held constant. The required number of reload assemblies increases proportional to fuel assembly weight, from 120 to 148. The reduced fissile inventory of the fuel assembly is the major contributor to the 5 to 6 percent increase in resource requirements.

Case E in Table 8-3 is a design in which the fuel enrichment was increased to hold the fissile content of the fuel bundles constant. The same number of reload assemblies were loaded as in Case B and the same improvement features were applied (axial blankets, low leakage refueling patterns, extended burnup, spectral shift with flow control, EOC feedwater temperature reduction, EOC power coast-down). The discharge burnup increases to 62,500 MWd/MT. In this case, unlike observations relative to the "current improved BWR" design, the uranium ore requirements increased by 1.4 percent and the SWU requirements increased by 7.5 percent.

These negative trends for reduced pellet density, are attributable to two factors. First, the optimum H/U ratio is burnup dependent and becomes lower as burnup is extended. Second, application of

TABLE 8-3

3-D EVALUATIONS OF REDUCED PELLET DENSITY

	REFERENCE BWR	CURRENT IMPROVED BWR	EXTENDED BURNUP DESIGN CASE B	REDUCED EFFECTIVE PELLET DENSITY ^a	
				CASE D	CASE E
● AVERAGE CAPACITY FACTOR (%)	75	75	75	75	75
● FRACTION OF BATCH REPLACED/REFUELING	0.24	0.24	0.14	0.18	0.14
● NUMBER OF FRESH FUEL ASSEMBLIES/REFUELING	200	200	120	148	120
● NUMBER OF REINSERT ASSEMBLIES/REFUELING	---	---	---	---	---
● FUEL PELLET DENSITY (% OF NOMINAL)	100	100	100	80	80
● RELOAD ENRICHMENT	2.94	2.74	3.65	3.63	4.55
● REFUELING INTERVAL (YEARS)	1.0	1.0	1.07	1.0	1.07
● CONVERSION LOSS FRACTION	0.005	0.005	0.005	0.005	0.005
● FABRICATION LOSS FRACTION	0.01	0.01	0.01	0.01	0.01
● DISCHARGE FUEL BURNUP PRIOR TO REINSERTION (MWd/MT)	28439	28439	51000	47500	62500
● DISCHARGE BURNUP OF REINSERTED FUEL (MWd/MT)	---	---	---	---	---
● U ₃ O ₈ REQUIREMENTS (ST/GWE)					
--ANNUAL EQUILIBRIUM RELOAD	204	190	146.1	156	149
● SEPARATIVE WORK REQUIREMENTS (10 ³ SWU/GWE)					
--ANNUAL EQUILIBRIUM RELOAD	124	111	95.7	103	105
● PERCENT REDUCTION COMPARED TO REFERENCE CASE (%)					
--U ₃ O ₈ REQUIREMENTS					
— ANNUAL EQUILIBRIUM RELOAD	---	6.9	28.4	23.5	27.0
--SEPARATIVE WORK REQUIREMENTS					
— ANNUAL EQUILIBRIUM RELOAD	---	10.5	22.8	16.9	15.3

^aPellet density reduced to 80% of reference design.

spectral shift with flow control, EOC power coastdown and EOC feedwater temperature reduction increases the end-of-cycle H/U ratio, which is the primary determinate in the H/U ratio optimization. Also, when spectral shift related improvements are being applied, it is desirable that the within cycle H/U be as small as possible so that the gains will be maximized when the H/U is shifted upward at EOC. Hence, reduction of pellet density shifted the H/U ratio in a non-optimum direction within cycle and shifted the H/U ratio above optimum at EOC for the extended burnup design basis.

Figure 8-1 illustrates the trends in reactivity slope versus burnup for solid and reduced effective density pellet fuel designed for burnups typical of the Reference Design (Curve "Set A") and for extended burnup (Curve "Set B"). These plots account for weight differences by shifting the burnup scale for the annular pellet fuel proportional to bundle weight (this is equivalent to plotting reactivity versus MWd energy generation). The "cross over point" is the point at which the infinite fuel bundle lattice neutron multiplication for the reduced density fuel becomes less than that of the reference fuel for further increases in energy generation. The discharge burnups are indicated on each curve. For the extended burnup designs, the burnup is well beyond the cross over point and this causes a greater penalty from the increased reactivity slope of the reduced density fuel. Decreased conversion ratio, due to the greater H/U ratio of the reduced density designs, is the primary cause of the steeper reactivity slope. Gains from spectral shift and EOC coastdown, which soften the EOC spectrum, are less for reduced density fuel because of both the steeper reactivity slope with burnup and the less negative steam void reactivity coefficients that are caused by the density reduction.

The larger reactivity swing with burnup for the reduced density fuel also will cause increased radial power peaking as a result of greater reactivity mismatch between fresh and discharge fuel. This may require use of less optimal, higher neutron leakage, refueling patterns to reduce radial peaking, which will feedback to further increase uranium and SWU requirements.

The SWU increase for the reduced density fuel is larger than the uranium requirement increase. The increase in SWU requirements will lead to economic penalties for reduced density design.

These trends are expected to be similar, independent of the method used to reduce the effective fuel pellet density and thus, are expected to indicate the trends which would result from use of annular pellets or sphere pac fuel. The observed trends also are expected to be indicative of the trends with designs which increase H/U ratio by other means such as reduction in fuel rod diameter, addition of water rods to the lattice, increase in fuel rod pitch, etc.

Figure 8-1. Effects of Pellet Density, Burnup on Reactivity Slope

On the basis of these evaluations, it was concluded that reduced pellet density, increased H/U ratio designs are undesirable when applied in conjunction with extended burnup and other design improvement options, such as spectral shift, which increase the end-of-cycle H/U ratio.

8.5 OPTIMAL COMBINED IMPROVEMENTS

The evaluations with the 3.65 w/o extended burnup fuel assembly indicated that there was potential for reduction of SWU requirements by reducing the reload fuel assembly enrichment and the associated burnable absorber content of the fuel. It further appeared that this would reduce the technological extrapolation by reducing discharge burnup and by increasing refueling batch sizes into a range in which there is applicable experience for current operating BWRs. With these trends and objectives in mind, a reduced enrichment fuel assembly was evaluated along with a coupled high enrichment discharge fuel rod reconstituted assembly which incorporated 6 water rods to shift the H/U ratio to more optimum conditions when there is significant plutonium in the fuel rods.

8.5.1 Assumptions, Criteria, Methods of Approximation

A 3-Dimensional nuclear-thermal-hydraulic-fuel multicycle evaluation was performed for the improvements summarized in Table 1-4. The lattice enrichment is about 3.53% U-235, and with the 6" long natural uranium axial blankets (on top and bottom of the fuel bundle) the bundle enrichment is about 3.3%. The discharge burnup for fuel before reinsertion is 43,000 MWd/MT. Each reinsert bundle was reassembled from two discharge 3.3 w/o bundles.

The reconstituted assemblies analyzed utilized four extra water tubes were added to the original two in the central region of the bundle. The reconstituted fuel assemblies were reinserted into the reactor for an additional cycle of irradiation and were discharged with an average burnup of 49,000 MWd/MT

The equilibrium cycle was simulated with the following sequences of reactor state variables:

THERMAL POWER	100%	100%	100%	100%	80%
FLOW	80%	95%	MAX	MAX	MAX
FEEDWATER TEMPERATURE	420°F (RATED)	RATED	RATED	250°F	250°F

Maximum flows of 120% of rated and 105% of rated were evaluated. Most operating BWRs have 5% or more excess flow capacity which can be applied to shift the end-of-cycle flow. Although pumping systems and other improvements might be retrofit to increase this capability, it is not likely that a utility would do this if any significant outage time were required.

Therefore, the 105% maximum flow case is the more likely case for existing reactors and the 120% max. flow cases are indicative of the potential for future reactors.

Both 12 and 18 month cycles were evaluated in order to provide a basis for assessing the effects of cycle length on resource requirements and on the savings from application of the improvement options.

8.5.2 Resource Requirements for 12 Month Cycle Lengths

The results of the 3-D evaluations are summarized in Table 8-4. The uranium requirements were increased by a small amount, 0.8 percent, relative to the higher enrichment design (Case C in Table 8-2). However, SWU requirements were decreased by 3.1 percent. Case F has about a 6% higher cycle length than Case C which was a partial contributor to the reduced uranium savings. It is likely that the greater burnups and greater in-core residence times of Case C will require mitigating changes in fuel assembly design to assure fuel reliability. Greater power peaking and more negative steam void reactivity coefficients associated with Case C also may require mitigation. In view of the greater conservatism, the smaller technological extrapolations, and the expected better power peaking of Case F and the significant reduced SWU requirements, it was concluded that Case F represented the optimal combined set of improvements for the combined improvements which were evaluated.

A flow range of 90 to 105% flow is achievable today in most operating BWRs. Therefore, an analysis was carried out with all of the improvement features of the best combined case, with the spectral shift flow range reduced to 90 to 105% flow. The results are summarized in Table 8-4, Case G. Refueling patterns were held constant and reconstitution was approximated by using the same reconstitution assemblies as in Case F. Case G is summarized in greater detail in Table 1-3.

TABLE 8-4
SUMMARY OF TWELVE MONTH CYCLE OPTIMUM COMBINED IMPROVEMENTS

	REFERENCE BWR	CURRENT IMPROVED BWR	BEST COMBINED IMPROVEMENT	
			CASE F 80-120% FLOW RANGE*	CASE G 90-105% FLOW RANGE*
● AVERAGE CAPACITY FACTOR (%)	75	75	75	75
● FRACTION OF BATCH REPLACED/REFUELING	0.24	0.24	0.23	0.23
● NUMBER OF FRESH FUEL ASSEMBLIES/REFUELING	200	200	132	132
● NUMBER OF REINSERT ASSEMBLIES/REFUELING	---	---	60	60
● FUEL PELLET DENSITY (% OF NOMINAL)	100	100	100	100
● RELOAD ENRICHMENT	2.94	2.74	3.3	3.3
● REFUELING INTERVAL (YEARS)	1.0	1.0	1.062	1.056
● CONVERSION LOSS FRACTION	0.005	0.005	0.005	0.005
8-13 ● FABRICATION LOSS FRACTION	0.01	0.01	0.01	0.01
● DISCHARGE FUEL BURNUP PRIOR TO REINSERTION (MWd/MT)	28439	28439	43450	42960
● DISCHARGE BURNUP OF REINSERTED FUEL (MWd/MT)	---	---	49520	48940
● U ₃ O ₈ REQUIREMENT (ST/GWE)				
--ANNUAL EQUILIBRIUM RELOAD	204	190	145.3	146.2
● SEPARATIVE WORK REQUIREMENT (10 ³ SWU/GWE)				
--ANNUAL EQUILIBRIUM RELOAD	124	111	91.7	92.3
● PERCENT REDUCTION COMPARED TO REFERENCE CASE (%)				
--U ₃ O ₈ REQUIREMENTS				
- ANNUAL EQUILIBRIUM RELOAD	---	6.9	28.8	28.3
--SEPARATIVE WORK REQUIREMENTS				
- ANNUAL EQUILIBRIUM RELOAD	---	10.5	26.0	25.6

*Improvements listed in Tables 1-4 and 8-1.

Changing the spectral shift flow range from 80-120% to 90 to 105% had very little effect on the cycle energy and the uranium and SWU requirements, assuming the same refueling patterns as Case F. These results were inconsistent with spectral shift evaluations at reduced burnup, without other combined improvements, which are reported in Section 7.6 and summarized in Table 7-2. Therefore, a careful check and comparison of Cases F and G was initiated in an attempt to assure validity of the results and to explain the observed differences.

Table 8-5 is a summary of the incremental cycle energy gains from each of the improvements applied to Cases F and G. At the 95/420/100, %/Flow/Feedwater Temperature/% Power state, the equilibrium 90 to 105% flow range case has accumulated more cycle energy than the 80 to 120% flow range case. However, on the first cycle of improvement application (non-equilibrium) this was not the situation, the 80 to 120% flow range case had accumulated more energy at the 95/420/100 state. These and other evaluations led to the conclusion that the observed trends between the two cases were attributed to 3 effects, all of which tended to penalize the 80-120% flow range case and make it conservative:

1. The 15 and 25% flow steps taken in 3-D analysis of the 80-120% range case were considerably greater than 5 and 10% steps used in the 90-105% case. Taking a larger flow step penalizes conversion ratio. For example, the assumption of 95% of rated flow over the 1762 to 3214 MWd/MT burnup interval reduced the steam void content and penalized the conversion ratio compared to more realistically subdividing this burnup interval into three substeps at 85%, 90% and 95% of rated flow. Hence, a reanalysis of the best combined design with smaller flow control spectral shift flow steps in the 3-D simulation would increase the cycle energy and uranium and SWU savings.
2. Operation at 80% flow at rated power greatly reduces the hot excess reactivity and reduces the required concentration of burnable absorbers in the fuel. By the time of all rods out, at 1762 MWd/MT cycle energy, little of the gadolinia has depleted. Consequently, for optimal conditions, a different fuel assembly design utilizing less gadolinia could have been used for design F as compared to design G and this would have further reduced uranium and SWU requirements.

TABLE 8-5
SUMMARY OF WITHIN CYCLE, COMBINED IMPROVEMENT INCREMENTAL
EFFECTS ON EQUILIBRIUM CYCLE ENERGY

BEST COMBINED DESIGN 80-120% FLOW RANGE, CASE F					BEST COMBINED DESIGN REDUCED 90-105% FLOW RANGE, CASE G				
REACTOR STATE			INCREMENTAL	CUMULATIVE	REACTOR STATE			INCREMENTAL	CUMULATIVE
% RATED	FEEDWATER	% RATED	CYCLE ENERGY	CYCLE ENERGY,	% RATED	FEEDWATER	% RATED	CYCLE ENERGY	CYCLE ENERGY,
FLOW	TEMP, °F	POWER	FOR STATE	MWd/MT	FLOW	TEMP, °F	POWER	FOR STATE	MWd/MT
			MWd/MT					MWd/MT	
80	420	100	1762	1762	90	420	100	3784	3784
95	420	100	1452	3214	95	420	100	286	4070
120	420	100	1540	4754	105	420	100	522	4592
120	250	100	1236	5990	105	250	100	1274	5866
120	250	80	1204	7194	105	250	80	1248	7114

3. It appears on the basis of preliminary evaluation that application of all of the improvements in tandem shifts the EOC H/U ratio above the optimum. Consequently, under these circumstances, there is reduced gain from increasing core flow from 105% to 120% of rated at EOC, since increases in H/U ratio above the optimum decrease core reactivity. It therefore appears, on the basis of preliminary evaluations, that flow increase above 105% could be improved by redesigning the fuel assembly or the lattice to reduce the hydrogen-to-uranium ratio at EOC. This conclusion should be confirmed by more detailed evaluations.

A very important conclusion from these evaluations is that the best combined improvement results could be achieved with a much less developmental 90 to 105% spectral shift flow range and that the results for the 80 to 120% range are conservative.

8.5.3 Resource Requirements for 18 Month Cycle Lengths

The prevalent utility trend is to operate for the longest possible cycle lengths between refuelings. Therefore, it appeared desirable to evaluate any impacts of this variable on the optimal selection of improvements. For purposes of comparison, it was necessary to develop a current improved 18 month cycle design. Greater enrichment fuel was used in the 18 month cycle evaluations to reduce the refueling batch size requirements. It was not feasible, for resource reasons, to include reconstitution/reinsert of high enrichment discharge rods in the 18 month cycle evaluations. The 18 month cycle gained more incremental savings from application of the improvement alternatives than did the 12 month cycle. For example, in comparison to the current improved design (Table 8-6) the incremental uranium and SWU improvements were approximately 4 percent greater for the 18 month cycle case.

On the basis of the evaluations, it has been concluded that the prevalent utility trend toward increased refueling cycle lengths will increase uranium and separative work requirements and will increase the incentive to apply the improvements which were evaluated in this program.

TABLE 8-6

SUMMARY OF OPTIMAL 18 MONTH CYCLE COMBINED IMPROVEMENTS

	REFERENCE BWR	CURRENT IMPROVED BWR*		BEST COMBINED IMPROVEMENT	
		12 MONTH CYCLE	18 MONTH CYCLE	12 MONTH CYCLE	18 MONTH CYCLE
● AVERAGE CAPACITY FACTOR (%)	75	75	75	75	75
● FRACTION OF BATCH REPLACED/REFUELING	0.24	0.24	0.39	0.23	0.25
● NUMBER OF FRESH FUEL ASSEMBLIES/REFUELING	200	200	292	132	188
● NUMBER OF REINSERT ASSEMBLIES/REFUELING	---	---	---	60	---
● FUEL PELLET DENSITY (% OF NOMINAL)	100	100	100	100.0	100
● RELOAD ENRICHMENT	2.94	2.74	2.99	3.3	3.65
● REFUELING INTERVAL (YEARS)	1.0	1.0	1.49	1.06	1.66
● CONVERSION LOSS FRACTION	0.005	0.005	0.005	0.005	0.005
● FABRICATION LOSS FRACTION	0.01	0.01	0.01	0.01	0.01
● DISCHARGE FUEL BURNUP PRIOR TO REINSERTION (MWd/MT)	28439	28439	27500	43000	47400
● DISCHARGE BURNUP OF REINSERTED FUEL (MWd/MT)	---	---	---	49000	---
● U ₃ O ₈ REQUIREMENT (ST/GWE)					
--ANNUAL EQUILIBRIUM RELOAD	204	190	216	146	156.2
● SEPARATIVE WORK REQUIREMENT (10 ³ SWU/GWE)					
--ANNUAL EQUILIBRIUM RELOAD	124	111	131	92	102.4
● PERCENT REDUCTION COMPARED TO REFERENCE CASE (%)					
--U ₃ O ₈ REQUIREMENTS					
— ANNUAL EQUILIBRIUM RELOAD	---	6.9	-5.9	28.3	17.4
--SEPARATIVE WORK REQUIREMENTS					
— ANNUAL EQUILIBRIUM RELOAD	---	10.5	-5.7	25.8	17.4
● PERCENT REDUCTION COMPARED TO CURRENT IMPROVED CORE (%)					
--U ₃ O ₈ REQUIREMENTS	---	---	---	23.5	27.6
--SEPARATIVE WORK REQUIREMENTS	---	---	---	17.1	21.7

*Axial blankets, low leakage refueling

9. PLUTONIUM RECYCLE

Although the primary emphasis of the program was to evaluate once-through fuel cycle improvements, it also was desirable to evaluate the effects of the improvements on the plutonium recycle option. Until the mid '70s decision to discontinue reprocessing and plutonium recycle, a large amount of effort had been devoted to development of optimal BWR Pu recycle designs. A full reload of BWR Pu recycle fuel was successfully fabricated, loaded in the Italian Garigliano reactor and used to produce power. Extensive RD&D programs were completed on Pu recycle, including manufacturing development, hot critical assembly nuclear measurements, and nuclear and fuel material measurements in test and power reactors. Reliable annular pellet designs were developed for Pu recycle fuel to adjust the lattice to the greater H/U ratio optimum of fuels containing significant plutonium. Extensive design optimization was carried out over a number of years, resulting in significant improvements in Pu recycle fuel performance and economics. This prior work and experience was applied to the Pu recycle evaluations under this program.

9.1 BWR Pu RECYCLE DESIGN BASES & CRITERIA

BWR plutonium recycle fuel is designed to be as interchangeable as possible with the uranium fuel it replaces. The key criteria in designing for optimal Pu recycle interchangeability with the uranium fuel it replaces are to:

- Minimize Differences in Reactivity vs. Burnup Characteristics
- Maintain same peak-to-average power
- Minimize differences in steam void reactivity coefficients
- Minimize effects on control blade strength

Experimental results from operating power stations and test facilities, and detailed evaluations with design models that have been compared with experiments have verified that the above objectives are achievable. General Electric plutonium recycle fuel assemblies have been designed to be totally interchangeable with uranium reload fuel. The plutonium recycle fuel is designed to the same thermal, mechanical and hydraulic conditions and limits as the uranium fuel. The

nuclear characteristics are affected by the plutonium recycle fuel. However, plutonium recycle designs have been developed with the following nuclear characteristics relative to uranium:

- Improved local and global power distributions
- Adequate control rod strength and cold shut-down margin;
- Reduced cold-to-hot operating reactivity swing
- Acceptable dynamic and safeguards characteristics

Interchangeability has been achieved in BWR design by use of an MO₂ island design in which the Pu bearing rods are located in central regions of the lattice away from the control blades and the water gaps between assemblies.²² Location of MO₂ in these lattice regions replaces the highest enrichment fuel rods in the BWR fuel assembly, thereby maximizing uranium ore and separative work savings. The low enrichment uranium rods, which are the highest conversion ratio rods in a BWR fuel assembly, remain in the highest neutron importance regions at the edge of the assembly and continue to have a high conversion ratio unimpeded by competition for neutrons with the plutonium recycle fuel. This design also maximizes the Pu concentration in each fuel rod to minimize the number of Pu recycle fuel rods and minimize fabrication costs. Competition of the large Pu absorption cross section with the control blades also is avoided, thereby preventing major adverse effects on reactor control. Self generated BWR Pu recycle designs require that less than 30% of the rods in the recycle fuel bundles be Pu recycle rods.

The basic BWR lattice and control rod geometry is one of the key factors which makes possible the practical and economical recycle of plutonium as a substitute for enriched U-235. One of the most important considerations in recycling plutonium in a light water reactor is maintenance of adequate control strength when plutonium is inserted into the lattice. The Pu isotopes have an absorption cross section approximately 3 times greater than U-235 and thus plutonium, if placed adjacent to a control blade, will compete with the control blade and reduce control system reactivity worth. Because of the mechanical design of the BWR cruciform control blades and the fact that they lie outside the individual fuel assemblies, plutonium may be physically isolated from the control blades by placing it within an "island" of mixed oxide (MO₂) rods in the interior of each fuel bundle. In this manner, loss of movable control worth due to the presence of

plutonium is minimized, even in the event large amounts of plutonium are recycled.

The flow channels surrounding BWR fuel assemblies and the water gaps between assemblies also aid in Pu recycle fuel design. The flow channels prevent cross flow and make it possible to mix fuel assemblies with different pressure drop and flow characteristics within the same reactor core. The water gaps between assemblies decouple the thermal neutron spectrum and minimize spectral interactions between fuel assemblies with large differences in fissile content or neutron absorption.

Burnable absorber (Gd₂O₃-UO₂) rods are located within or directly adjacent to the plutonium island to provide supplementary control. Gd-155 and 157 have absorption cross sections which vary much faster than inversely with thermal neutron velocity, while Pu-239 and Pu-241 have large fissile and absorption cross sections at 0.3 ev, near the top of the thermal neutron energy range. Thus, the combination of gadolinia and plutonium in an island arrangement produces a near ideal match from the standpoint of reducing cold-to-hot reactivity swing, improving reactivity coefficient interchangeability, and maintaining good local power distributions as a function of fuel assembly burnup.

The optimum H/U atom ratio is greater for a Pu recycle design. BWR reference designs utilize hollow or annular pellets in the MO₂ rods to improve economics and interchangeability with UO₂ fuel. The annular fuel reduces the steam void reactivity coefficient, thereby improving transient performance. Annular fuel in the central regions of the bundle also reduces stored energy in this region during a LOCA, thus reducing peak cladding temperatures. The annular fuel also reduces Pu fissile inventory requirements and improves economics.

The plutonium concentration is varied between fuel rods within the island geometry and the surrounding uranium fuel rod enrichments are varied to improve local power distributions. The island location coupled with combined use of burnable control absorbers, annular pellets and variable fissile concentration between fuel rods results in a fuel assembly with nuclear characteristics that make it interchangeable with uranium reload assemblies and that make it possible to design for a wide range of initial plutonium isotopic concentrations and quantities of recycle plutonium.

As a result of these design bases and considerations, BWR mixed oxide fuel bundles have been designed to be similar in mechanical, thermal, hydraulic, nuclear, dynamic and safety characteristics with uranium bundles. This interchangeability and similarity is significant because it preserves fuel management flexibility. For example, standard bundles may be applied to simplify the process of licensing and to facilitate obtaining emergency supply fuel bundles.

BWR Pu recycle design is sensitive to the location of MO2 and UO₂/Gd₂O₃ rods in the lattice and the effect of the MO2 rods on the surrounding UO₂ fuel rods. Key measures of the resource utilization and economic quality of Pu recycle design are:

- U235 Replaced/Fissile Pu Loaded
- SWU Replaced/Fissile Pu Loaded

Variations of these quantities for three alternate BWR Pu recycle bundle designs which all use the same fuel assembly dimensions and hardware is illustrated by the following comparison between several different BWR island Pu recycle designs.

	<u>U235 REPLACED/ FISSILE Pu LOADED</u>	<u>SWU REPLACED/ FISSILE Pu LOADED</u>
BWR BUNDLE DESIGN A	.73	117
BWR BUNDLE DESIGN B	.87	134
BWR BUNDLE DESIGN C	.97	142

This comparison illustrates that large differences in Pu recycle characteristics can be achieved by changes in location of the Pu rods in the lattice, Pu rod fissile contents relative to uranium rod fissile contents, etc. In BWR Pu recycle fuel design, such considerations can have large impacts on economics and uranium/SWU requirements.

Some of the MO₂ rods in the center of the BWR fuel assembly are very black to thermal neutrons, permitting storage of excess plutonium inventories in the reactor by increasing the concentration of these rods. Fissile Pu concentrations of 8 and 6 percent were used in Pu recycle fuel which has been designed and irradiated in the Big Rock and Garigliano power reactors.

The recycle Pu may be blended with diffusion plant tails uranium, natural uranium, or recycle uranium recovered from reprocessing. It also is feasible to consider non-self generated recycle in which an excess of plutonium from an outside source is recycled. All have been used in BWR designs and/or BWR power reactor applications.

9.2 MIXED OXIDE FUEL EXPERIENCE BASE FOR BWRs

Mixed-oxide (MO₂) fuel bundles have been designed utilizing the principles from the previous section and have undergone irradiation in a broad range of reactor environments. Some of the programs which have contributed, or are currently contributing, to the BWR experience base with mixed-oxide fuel are summarized in Table 9-1. These programs range from those involving a few fuel rods to a full reload batch of mixed-oxide bundles at the Garigliano reactor.

The results of these irradiation programs have confirmed the high degree of interchangeability between MO₂ and UO₂ bundles for operation in BWRs. Performance of MO₂ fuels has been found to be comparable to that of UO₂ fuel.

The on-going EPRI sponsored program at Quad Cities 1 has included a substantial number of fuel nuclear and mechanical performance measurements which provide benchmark data for verifying analytical models. Featured in the measurements are fuel rod isotopic distributions at low exposure (one cycle), nominal discharge burnup bundle rod-by-rod and axial gamma scans at each cycle refueling outage to assess power/exposure distributions, and a full range of fuel rod non-destructive tests and profilometry to verify mechanical integrity. The five test assemblies of this program have operated satisfactorily for 6 years of in-core residence.

TABLE 9-1

GE BWR MIXED OXIDE FUEL EXPERIENCE BASE

<u>PROGRAM</u>	<u>SCOPE</u>	<u>KEY RESULTS</u>
● EURATOM SPONSORED IRRADIATION OF PU FUEL IN VALLECITOS BWR (1960-1966).	PU FUEL IN $\frac{1}{4}$ OF A FUEL ASSEMBLY WAS IRRADIATED.	DETERMINATION OF ISOTOPIC CONTENT AS A FUNCTION OF FUEL BURNUP.
● PLUTONIUM RECYCLE TEST REACTOR (HANFORD)	80 RODS OF HOT PRESSED AND 3 RODS OF COLD-PRESSED PELLETS WERE IRRADIATED TO 11,000 MWd/MT	FUEL SUBJECTED TO SPECIFIC POWERS IN EXCESS OF 21 KW/FT TO PROVIDE OPERATING EXPERIENCE UNDER SEVERE ENVIRONMENTAL CONDITIONS.
● GARIGLIANO MO ₂ RELOAD FUEL (1975-PRESENT)	RELOAD FUEL BATCH OF 46 MO ₂ BUNDLES (32 MO ₂ RODS/ BUNDLE)	BUNDLES HAVE OPERATED SATISFACTORILY AND ARE EXPECTED TO REACH A NOMINAL DISCHARGE EXPOSURE OF >25,000 MWd/MT.
● BIG ROCK POINT-EEI PROGRAMS (1969-1974)	IRRADIATION OF 32 MO ₂ RODS IN 16 UO ₂ CARRIER BUNDLES AND IRRADIATION OF 3 MO ₂ RELOAD BUNDLES.	EVALUATION OF ANNULAR AND SOLID PELLET MO ₂ FUEL PERFORMANCE INCLUDING DESTRUCTIVE EXAMINATIONS. PEAK PELLET BURNUPS TO 36,000 MWd/MT.

TABLE 9-1
(continued)

PROGRAM	SCOPE	KEY RESULTS
● HALDEN PLUTONIUM PROGRAM	MO ₂ FUEL RODS WITH ANNULAR PELLET WERE IRRADIATED AT 17 TO 22 KW/FT.	MEASUREMENTS OF THE EFFECTS OF PELLET-CLAD INTERACTION SHOWED ACCEPTABLE BEHAVIOR OF THE MO ₂ RODS.
● KRITZ PROGRAM	A SERIES OF HOT CRITICAL EXPERIMENTS WAS COMPLETED INCLUDING POWER DISTRIBUTION AND OTHER MEASUREMENTS FOR GARIGLIANO MO ₂ RELOAD BUNDLES.	BENCHMARKING OF DATA TO ANALYTICAL PREDICTIONS PRODUCED GOOD AGREEMENT.
● QUAD CITIES 1 EPRI PROGRAM (1974-PRESENT)	4 MO ₂ BUNDLES AROUND THE CENTER CONTROL BLADE AND 1 MO ₂ BUNDLE ON THE CORE PERIPHERY ARE UNDERGOING IRRADIATION IN A HIGHLY CONTROLLED SITUATION WITH EXTENSIVE NON-DESTRUCTIVE MEASUREMENTS.	CENTER BUNDLES HAVE OPERATED SUCCESSFULLY FOR 6 YEARS OF CORE RESIDENCE TO BURNUPS ABOVE 27,000 MWD/MT DETAILED BENCHMARK DATA HAS BEEN OBTAINED ON LOCAL POWER DISTRIBUTIONS, FUEL ISOTOPICS AND ANNULAR MO ₂ PELLET MECHANICAL INTEGRITY.

Based upon the design and operating experience accumulated to date on mixed oxide fuel, it is expected that BWRs can be safely and reliably operated with MO2 fuel bundles replacing UO2 bundles. The MO2 fuel bundle designs used in the analyses detailed in following sections are representative of the best designs in the MO2 BWR data base. However, the designs were not re-optimized to reflect the impacts of the improvement alternatives. It is possible that such a re-optimization would result in greater uranium and SWU savings.

Minimization of uranium ore requirements has not been a major objective function in prior Pu recycle design. Therefore, it is probable that re-optimization of Pu recycle fuel bundle designs to minimize uranium ore requirements could result in further savings; however such an optimization was outside the scope of work under this program.

9.3 Pu RECYCLE IMPROVEMENT IMPACT ANALYSIS

A preliminary impact analysis was completed to estimate the effects of the once-through improvement options on Pu recycle. The once-through fuel cycle options which increase reactor conversion ratio or conserve neutrons also will have favorable impact on uranium requirements for plutonium recycle applications.

Table 9-2 is a summary of the preliminary evaluation of the effects of the BWR once-through improvement options on plutonium recycle. Options which reduce fissile inventory requirements for a given discharge burnup design improve the reactor conversion ratio by reducing competition of the fissile materials with fertile materials (conversion ratio is defined as fissile material produced by fertile capture per fissile atom destroyed). It was concluded that most of the once-through fuel cycle options have favorable impact on the Pu recycle mode.

Another consideration is the fact that fuel generally will be designed for minimum fuel cycle and power costs and/or maximum nuclear capacity factors, independent of uranium ore requirements. In the once-through cycle the minimum fuel cycle costs usually correlate with minimum uranium requirements, but this is not always the case for closed cycles. Once-through cycle improvement options which improve economics or reactor performance (thermal margins, safety margins, load following, etc) also are likely to be adopted in the plutonium recycle mode. Preliminary estimates of the effects of the once-through improvement options on the economics and performance of Pu recycle also are summarized in Table 9-2.

TABLE 9-2

PRELIMINARY IMPACT ANALYSIS OF ONCE-THROUGH FUEL CYCLE OPTIONS ON BWR PU RECYCLE DESIGN

<u>ONCE-THROUGH IMPROVEMENT OPTION</u>	<u>IMPROVEMENT FOR PU RECYCLE DESIGN</u>	<u>EXPECTED IMPACT ON URANIUM REQUIREMENTS</u>	<u>EXPECTED IMPACT ON ECONOMICS</u>	<u>EXPECTED IMPACT ON PERFORMANCE</u>
AXIAL BLANKETS	REDUCED NEUTRON LEAKAGE, REDUCED FISSILE INVENTORY	FAVORABLE	FAVORABLE	FAVORABLE
IMPROVED, LOW LEAKAGE REFUELING PATTERNS	REDUCED NEUTRON LEAKAGE, REDUCED FISSILE INVENTORY	FAVORABLE	FAVORABLE	FAVORABLE
INITIAL CORE OPTIMIZATION	REDUCED NEUTRON LEAKAGE, REDUCED FISSILE INVENTORY	FAVORABLE*	FAVORABLE	FAVORABLE
COOLANT DENSITY CONTROL SPECTRAL SHIFT	SPECTRAL SHIFT, REDUCED FISSILE INVENTORY	FAVORABLE	FAVORABLE	+ AND -
INCREASED BURNUP	REDUCES # OF ASSEMBLIES, INCREASES PU CONTENT OF RECYCLE FUEL RODS	UNFAVORABLE	FAVORABLE IF RE- PROCESSING AND PU FABRICATION COSTS ARE HIGH.	UNFAVORABLE
REDUCED FUEL DENSITY & LATTICE OPTIMIZATION	REDUCED FISSILE INVENTORY, HIGH CONVERSION RATIO RODS IN REGIONS OF GREATEST NEUTRON IMPORTANCE	FAVORABLE	FAVORABLE	FAVORABLE

*Uranium savings expected to be smaller than for the once-through option.

TABLE 9-2
(CONTINUED)

<u>ONCE-THROUGH IMPROVEMENT OPTION</u>	<u>IMPROVEMENT FOR PU RECYCLE DESIGN</u>	<u>EXPECTED IMPACT ON URANIUM REQUIREMENTS</u>	<u>EXPECTED IMPACT ON ECONOMICS</u>	<u>EXPECTED IMPACT ON PERFORMANCE</u>
ONCE-THROUGH THORIUM CYCLE	REDUCED NEUTRON LEAKAGE, REDUCED REFUELING BATCH SIZE, REDUCED FISSILE INVENTORY	FAVORABLE	FAVORABLE	FAVORABLE
PREPLANNED EOC FEEDWATER TEMP. REDUCTION & POWER COASTDOWN	SPECTRAL SHIFT, REDUCED EOC NEUTRON LEAKAGE AND XENON ABSORPTIONS, RE- DUCED FISSILE INVENTORY, INCREASED EOC H/U RATIO	FAVORABLE	FAVORABLE	+ AND -
HIGH ENRICHMENT DISCHARGE FUEL ROD RECONSTITUTION/RE- INSERT	REDUCED FISSILE INVENTORY, IN- CREASED EOC H/U RATIO	FAVORABLE *	FAVORABLE	UNCERTAIN
BORIC ACID FOR COLD SHUTDOWN AND/OR REDUCED GADOLINIA RESIDUAL	REDUCED FISSILE INVENTORY	FAVORABLE *	FAVORABLE	+ AND -

*Uranium savings expected to be smaller than for the once-through option.

On the basis of the preliminary impact analysis, it was decided to apply all of the improvements that were practical, within resource constraints, to the Pu recycle design. However, extended burnup was limited to lower levels estimated to be closer to the Pu recycle optimum, including considerations of fabrication and reprocessing costs. It was not practical within resource limits to include high enrichment discharge fuel rod reconstitution and reinsert or the once-through thorium cycle. All of the Pu recycle designs incorporated reduced effective fuel density in the MO₂ rods through use of annular pellets.

9.4 ONCE-THROUGH CYCLE IMPROVEMENTS CONSIDERED IN PLUTONIUM RECYCLE EVALUATIONS

9.4.1 Scope of Evaluations

The evaluations were limited to 3-D multicycle analyses of a Pu recycle reference design which included the "current improved BWR" improvement features plus annular pellets in the MO₂ rods, and a combined improvement case which included all of the features summarized in Table 1-4, except reconstitution and reinsert of high enrichment discharge fuel rods. Separate analyses were carried out to provide information on the independent effects on Pu recycle of extended burnup, spectral shift with flow control and end-of-cycle feedwater temperature reduction and power coastdown.

9.4.2 Assumptions, Criteria and Methods of Approximation

The analyses performed for BWR fuel cycles in the recycle mode were structured to parallel those performed for once-through operation. Identical analytical methods were employed, and a consistent basis for comparing fuel cycle analysis results was established. Thus, information previously presented on analytical methods, reactor design parameters, and reactor operating conditions is applicable for the recycle analyses.

The mixed-oxide (MO₂) reload bundle design used to establish a base recycle case was selected to produce a new self-generation fuel cycle in which the discharge fissile Pu equals the initial fissile Pu. The mechanical design features and performance characteristics of the fuel bundle were similar to the UO₂ bundles used in the once-through analyses. Use of the MO₂ reload bundle resulted in an annual equilibrium cycle at a 75% capacity factor with a 0.25 reload batch fraction. This MO₂ bundle design and others used in

the analyses were not necessarily optimized to reduce uranium requirements. The flexibility of the BWR "island" design may allow for additional improvements in fuel utilization beyond results obtained from this evaluation.

The fuel cycle analyses performed for the base recycle case included simplifying assumptions which are summarized as follows:

- The number of recycle bundles in the reload fuel batch was determined by the fissile Pu available from the previous cycle discharge batch, with a one year delay time for reprocessing.
- The nuclear data for the MO₂ bundle design was based upon a constant 0.68 fissile fraction; and,
- Credit was taken for the uranium in the discharge fuel without including a re-enrichment penalty for U-236 buildup.

The latter two assumptions resulted in slightly optimistic fuel cycle projections for a core loaded with recycle bundles. Using these assumptions, the base recycle case was established by building up the number of MO₂ recycle bundles in the reload batch until an all MO₂ equilibrium cycle was obtained. In other cases, UO₂ fuel assemblies were added to the reload batch as needed to provide the required cycle energy.

For purposes of the 30 year estimate of resource requirements, it was assumed that Pu recycle would start one year after the first discharge batch and would continue for 30 years, using make-up only from the BWR discharging the fuel. The initial core was assumed to be the same as the comparable once-through design with no MO₂ fuel in the initial core. Credit was not taken for the higher discharge plutonium fissile fractions which will occur in the initial core discharge fuel.

9.4.3 Once-Through Improvement Options Evaluated

Table 9-3 is a summary of the 3-D multicycle, equilibrium core evaluations that were completed to provide a incremental assessment of the effects of the improvements on Pu recycle.

TABLE 9-3

SUMMARY OF 3-D MULTICYCLE PU RECYCLE ANALYSES

<u>CASE ANALYZED</u>	<u>FEATURES</u>
1. CURRENT IMPROVED FUEL RECYCLE - BASE CASE	AXIAL BLANKETS, IMPROVED RE-FUELING PATTERN, ANNULAR MO ₂ PELLETS
2. EXTENDED BURNUP RECYCLE	CURRENT IMPROVED PLUS INCREASED FUEL ENRICHMENT TO INCREASE BURNUP BY 25% OVER BASE CASE.
3. FLOW CONTROL SPECTRAL SHIFT RECYCLE	CURRENT IMPROVED PLUS FLOW CONTROL SPECTRAL SHIFT OVER A FLOW RANGE FROM INITIAL CONDITIONS AT 80% FLOW TO 120% FLOW AT EOC.
4. EOC FEEDWATER TEMPERATURE REDUCTION AND POWER COASTDOWN RECYCLE	CURRENT IMPROVED PLUS TEMPERATURE REDUCTION FROM 420°F TO 250°F AT END OF NORMAL CYCLE AT 100% POWER FOLLOWED BY POWER COASTDOWN TO 80% POWER.
5. COMBINED RECYCLE IMPROVEMENTS	BASE CASE WITH SPECTRAL SHIFT OPERATION AS IN 3 AND FEED-WATER TEMPERATURE REDUCTION WITH POWER COASTDOWN AS IN 4

9.4.4 Economic Incentives and Considerations

Although MO2 recycle fuel has been successfully operating in BWR cores, there remain appreciable uncertainties with regard to the economics of recycle fuel. The most severe uncertainties focus on fuel reprocessing costs and on MO2 rod fabrication. Potential regulatory issues are key factors in the economic uncertainty. It is expected however, that the reduction in ore and SWU requirements would provide sufficient incentives for recycle in a non-restrictive regulatory climate.

9.4.5 Impact on Core Characteristics, Fuel Reliability and Safety

The utilization of the "island" BWR bundle design in which MO2 rods are selectively located in the bundle interior results in a recycle MO2 bundle which is interchangeable with standard UO2 reload bundle designs. Therefore, introduction of recycle fuel can be achieved in BWRs without major changes in core hardware. Similarly, fuel reliability and overall reactor safety response would be comparable to UO2 loaded BWR cores.

9.4.6 Technical Uncertainties or Problems and Mitigating Measures

Some technical uncertainties exist for the recycle alternative because of the effects of economic, regulatory, emotional and political uncertainties on design. Large scale demonstrations are needed for the designs found to be optimal for the environment which will exist at the time of re-introduction of the Pu recycle alternative.

It has been observed that Pu recycle design nuclear characteristics are more difficult to reliably predict than those of UO2 fuels and design methods improvements are needed to correct known deficiencies. This will result in more efficient design by reducing design margin requirements.

9.4.7 Retrofitability

MO2 recycle requires no major hardware changes for implementation in BWR cores.

9.4.8 Research and Developmental Requirements

Major efforts will be required to develop economic large scale fuel reprocessing and MO₂ fuel rod fabrication. The political and emotional factors which have influenced these areas in the past make it unlikely that such efforts would proceed without long term assurances of a stable environment.

9.5 CURRENT IMPROVED Pu RECYCLE

The multi-cycle results for the current improved recycle case are summarized in Table 9-4. Selected results for the reference and "current improved" UO₂ once-through designs are included for comparison. For the same refueling batch size of the UO₂ design, the Pu recycle fuel bundles must be designed for greater burnups as a result of the reduced weight of the annular pellet MO₂ fuel. The equilibrium cycle U₃O₈ and SWU requirements for the recycle case were reduced relative to the UO₂ once-through "current improved BWR" design by 41 and 36 percent, respectively. The 30 year cumulative requirements for U₃O₈ and SWU were reduced by 36 and 31 percent, respectively. Equilibrium core U and SWU savings, relative to the reference BWR design were 49 and 45%, respectively.

Relative to the best combined once through design, the current improved Pu recycle design reduced equilibrium uranium ore and separative work requirements by 22 and 23 percent, respectively. Therefore, there remained appreciable additional incentive to recycle uranium and plutonium, even in the event that fewer once-through improvements were applied to the Pu recycle design than had been applied to the once-through design.

Also included in Table 9-5 is a summary of the discharge fissile material inventories. Over 30 years, the "current improved" Pu recycle design reduced the Pu fissile inventory by a factor of 5 relative to the reference BWR and by a factor of 3 relative to the best combined once-through cycle. This reflected the increased plutonium burned in the reactor in the Pu recycle case.

TABLE 9-4

SUMMARY OF Pu RECYCLE U_3O_8 & SWU REQUIREMENTS

	UNIT	REFERENCE BWR	CURRENT IMPROVED ONCE-THRU	BEST COMBINED ONCE-THRU	CURRENT IMPROVED Pu RECYCLE	COMBINED IMPROVEMENT Pu RECYCLE
CAPACITY FACTOR	%	75	75	75	75	75
ENRICHMENT TAILS	%	0.20	0.20	0.20	0.20	0.20
RELOAD CORE FRACTION	%	0.24	0.24	0.23	0.25	0.20
REFUELING INTERVAL	-	1.0	1.0	1.06	1.0	1.0
EQ ^M RELOAD ENRICHMENT	%	2.94	2.74	3.30	2.96	2.94 ^a
FISSILE FABRICATION LOSS	%	1.5	1.5	1.5	1.5	1.5
AVG DISCHARGE BURNUP	MWD/MT	28439	28439	43000/49000	30250	36000
PEAK PELLETT DISCHARGE BURNUP	MWD/MT	<44000	<44000	<66000	<50000	<60000
U_3O_8 REQUIREMENTS						
INITIAL CORE	ST/GWE	454	496	443	496	443
TRANSITION CYCLES	ST/GWE	685	640	410	467	355
ANNUAL EQ ^M RELOAD	ST/GWE	204	190	145	113	105
30 YEAR CUMULATIVE	ST/GWE	6443	6051	4624	3901	3528
SWU REQUIREMENTS						
INITIAL CORE	10 ³ SWU/GWE	198	235	198	235	198
TRANSITION CYCLES	10 ³ SWU/GWE	465	416	246	316	228
ANNUAL EQ ^M RELOAD	10 ³ SWU/GWE	124	111	92	71	68
30 YEAR CUMULATIVE	10 ³ SWU/GWE	3887	3490	2836	2397	2194
PERCENT IMPROVEMENT RELATIVE TO:						
REFERENCE O-T DESIGN						
U3O8 - EQUILIBRIUM		---	6.9	28.3	44.6	48.5
- 30 YEAR CUMULATIVE		---	6.1	28.2	39.5	45.2
SWU - EQUILIBRIUM		---	10.5	25.8	42.7	45.2
- 30 YEAR CUMULATIVE		---	10.2	27.0	38.3	43.6
CURRENT IMPROVED O-T DESIGN						
U3O8 - EQUILIBRIUM		-7.3	---	23.7	40.5	44.7
- 30 YEAR CUMULATIVE		-6.5	---	23.8	35.5	41.7
SWU - EQUILIBRIUM		-11.7	---	17.1	36.0	38.7
- 30 YEAR CUMULATIVE		-11.4	---	18.7	31.3	37.1
BEST COMBINED O-T DESIGN						
U3O8 - EQUILIBRIUM		-40.7	-31.0	---	22.1	27.5
- 30 YEAR CUMULATIVE		-39.6	-31.1	---	15.5	23.5
SWU - EQUILIBRIUM		-34.8	-20.7	---	22.8	26.1
- 30 YEAR CUMULATIVE		-37.1	-23.1	---	15.5	22.6

^a reload batch contains 136 MO₂ plus 16 UO₂ bundles.

GEAP-2496

TABLE 9-5

SUMMARY OF DISCHARGE FISSILE MATERIAL FOR ONCE-THROUGH & Pu RECYCLE ALTERNATIVES

	<u>UNIT</u>	<u>REFERENCE BWR</u>	<u>BEST COMBINED ONCE-THRU</u>	<u>CURRENT IMPROVED Pu RECYCLE</u>	<u>COMBINED IMPROVEMENT Pu RECYCLE</u>
DISCHARGE FISSILE MATERIAL					
ENRICHMENT: U235	W/O H.E.	0.96	0.51	0.62	0.53
Pu-FISS	W/O H.E.	0.62	0.68	0.94	0.83
ANNUAL EQUILIBRIUM DISCHARGE					
U235	/GWE	281	96	172	114
Pu FISS	/GWE	181	130	260	197
END OF EQUILIBRIUM CYCLE CORE INVENTORY					
U235	kg/GWE	8988	3005	1050	870
Pu FISS	kg/GWE	5207	3780	1110	1050

9.6 EXTENDED BURNUP Pu RECYCLE

9.6.1 Assumptions, Criteria and Method of Approximation

The procedure utilized was similar to that used in the UO₂ once-through analyses. Specifically, an MO₂ bundle design was chosen with similar overall characteristics as the base case MO₂ bundle but with a higher initial fissile loading. Using the higher fissile content bundle design, multi-cycle analyses were carried out to an equilibrium annual cycle at a 75% capacity factor and a 0.20 reload batch fraction. This produced a 25% increase in the average burnup of the discharge fuel relative to the base case. The equilibrium fuel cycle for the extended burnup case required a reload batch with both MO₂ recycle and standard UO₂ reload bundles. This was necessary because the fissile Pu from the discharge bundles was insufficient for a full MO₂ reload batch. The UO₂ reload bundle design was selected to be similar in performance characteristics to the MO₂ bundle. The final reload batch ratio of MO₂ bundles to UO₂ bundles was 3.75. This produced the necessary material balance between discharged fissile material and reload fuel fissile requirements.

9.6.2 U₃O₈ and SWU Requirements

The equilibrium cycle U₃O₈ and SWU requirements are summarized in Table 9-6. Extended burnup in the recycle mode resulted in an increase in the U₃O₈ and SWU requirements relative to the base recycle case of 6% and 14%, respectively. Table 9-6 includes a summary of the differential percentage changes in U₃O₈ and SWU requirements relative to the reference, "current improved" and best combined once-through designs and relative to the "current improved" Pu recycle design. The uranium requirements were significantly lower than for the UO₂ once-through case. The increase relative to the base recycle case was caused by the reduced fissile Pu inventory in the extended burnup discharge fuel. The reduced inventory made it necessary to include UO₂ bundles in the reload batch. The UO₂ bundles increased the SWU requirement because they required substantially more SWUs than the Pu recycle fuel.

9.6.3 Economic Incentives and Considerations

Unlike the UO₂ once-through evaluations, the evaluations of recycle with extended burnup indicated a negative impact on uranium and SWU utilization. This penalizes extended burnup in recycle

TABLE 9-6

EQUILIBRIUM CYCLE U & SWU REQUIREMENTS FOR Pu+U RECYCLE COMBINED
WITH ONCE-THROUGH CYCLE IMPROVEMENTS

DESIGN BASIS	U ₃ O ₈ REQMTS. STU308/GWE	U ₃ O ₈ REQUIREMENT % IMPROVEMENT RELATIVE TO:				SWU REQMTS. 10 ³ SWU/GWe	SWU REQUIREMENT % IMPROVEMENT RELATIVE TO:			
		REF. O-T UO ₂	CURRENT IMPROVED O-T UO ₂	COMBINED IMPROVED O-T UO ₂	CURRENT IMPROVED Pu RECYC.		REF. O-T UO ₂	CURRENT IMPROVED O-T UO ₂	COMBINED IMPROVED O-T UO ₂	CURRENT IMPROVED Pu RECYC.
REFERENCE O-T UO ₂	204	---	-7.4	-40.7	-80.5	124	---	-11.7	-34.8	-74.7
CURRENT IMPROVED O-T UO ₂	190	6.9	---	-31.0	-68.1	111	10.5	---	-20.7	-56.3
COMBINED IMPROVE O-T UO ₂	145	28.9	23.7	---	-28.3	92	25.8	17.1	---	-29.6
CURRENT IMPROVED RECYCLE	113	44.6	40.5	22.1	---	71	42.7	36.0	22.8	---
EXTENDED BURNUP RECYCLE	120	41.2	36.8	17.2	-6.2	81	34.7	27.0	12.0	-14.1
FLOW CONTROL SPECTRAL SHIFT RECYCLE	109.5	46.3	42.4	24.5	3.1	69	44.3	37.8	25.0	2.8
EOC FWTR & POWER REDUCTION RECYCLE	107	47.6	43.7	26.2	5.3	69	44.3	37.8	25.0	2.8
COMBINED IMPROVEMENT RECYCLE	105	48.5	44.7	27.6	7.1	68	45.2	38.7	26.1	4.2

9-10

GEAP-24963

applications. It is likely, however, that the costs incurred for fuel reprocessing and MO₂ rod fabrication would result in greater optimum discharge burnup than would be desirable on the sole basis of uranium utilization.

9.7 PLUTONIUM RECYCLE COMBINED WITH FLOW CONTROL SPECTRAL SHIFT

9.7.1 Assumptions, Criteria and Methods of Approximation

The analysis of spectral shift operation with recycle reload fuel paralleled that completed for UO₂ once-through fuel. The analysis used the same recycle bundle design used to establish the base case for Pu recycle. Spectral shift conditions were simulated in the 3-D analyses by stepwise increase in core flow from 80% of full flow to 90% and from 90% to 120% flow at full power. Assumptions made for the Pu recycle evaluations were the same as for the once-through evaluations. The sequence of core flow steps was repeated in a multicycle analysis to obtain equilibrium cycle characteristics. Equilibrium cycle conditions were achieved with a full reload batch of Pu recycle bundles, since the discharged fissile Pu satisfied the material requirements of the reload batch. Other assumptions were the same as those used in the base case recycle analysis.

9.7.2 U₃O₈ and SWU Requirements

The equilibrium cycle U₃O₈ and SWU requirements are summarized in Table 9-6. Flow control spectral shift resulted in reductions of 3.1 and 2.8% over the base recycle case U₃O₈ and SWU requirements, respectively. These reductions were generally smaller than those obtained for UO₂ once-through operation. Part of the reason lay in the fact that the discharge burnup for the spectral shift case is approximately 3.4% greater than that for the base recycle case and the available Pu and U-235 content of the discharge fuel were reduced. Greater improvements in fuel utilization with spectral shift operation may be possible by selectively optimizing the MO₂ island design and/or reducing Pu fissile loading to maintain the burnup constant.

9.7.3 Economic Incentives and Considerations

Spectral shift operation in the recycle mode resulted in an improvement in fuel utilization and, therefore, would be desirable from an economics standpoint. Potential exists for optimizing the

recycle bundle design to produce greater uranium savings than obtained in this study.

9.8 Pu RECYCLE COMBINED WITH EOC FEEDWATER TEMPERATURE REDUCTION AND POWER COASTDOWN

9.8.1 Assumptions, Criteria and Methods of Approximation

The 3-D fuel cycle analysis for recycle combined with feedwater temperature reduction (FWTR) and power coastdown (PCD) incorporated the same criteria as used in the UO₂ once-through analyses. The feedwater temperature was reduced to 250°F in the FWTR mode and the reactor power was dropped to 80 % at full flow for the PCD mode.

The results of these operating conditions were to extend the cycle length and produce a greater fuel discharge burnup. An equilibrium cycle case could not be obtained with a full reload batch of recycle bundles due to insufficient fissile Pu in the discharge batch. The ratio of MO₂ bundles to UO₂ bundles in the equilibrium reload batch was 8.5 (11% of the reload bundles were UO₂). All other assumptions for reactor operation with FWTR and PCD were the same as used for the UO₂ once-through analyses.

9.8.2 U₃O₈ and SWU Requirements

Table 9-6 is a summary of the equilibrium cycle U₃O₈ and SWU requirements for recycle with EOC feedwater temperature reduction and power coastdown. The results indicated reductions relative to base recycle case U₃O₈ and SWU requirements of 5.3 and 2.8%, respectively. As in the case of spectral shift operation, the uranium utilization improvements in the recycle mode were generally less than those obtainable for UO₂ once-through cycles. This was due to the negative impact of greater fuel burnup on the recycle case and the need for UO₂ bundles in the reload batch.

9.8.3 Economic Incentives and Considerations

EOC feedwater temperature reduction and power coastdown operation in the recycle mode produced an improvement in uranium utilization

The flexibility available in BWR recycle bundle design may allow greater improvement in fuel utilization than was achieved in this evaluation. Cycle length and batch size optimization may also yield better results in terms of uranium utilization.

9.9 BEST CASE Pu RECYCLE COMBINED WITH ONCE-THROUGH CYCLE IMPROVEMENTS

9.9.1 Assumptions, Criteria and Methods of Approximation

The combination of Pu recycle with all the once-through improvements evaluated incrementally was evaluated for plutonium recycle on an equilibrium cycle basis. The combined improvements included axial blankets, low leakage refueling, limited extended burnup, spectral shift, EOC feedwater temperature reduction, and EOC power coastdown. The increased fissile inventory Pu recycle fuel assembly, evaluated for extended burnup in Section 9.6, was not utilized. However, the burnup was extended by application of once-through improvements without reducing the fuel assembly fissile enrichment.

The combined improvements case paralleled the equilibrium cycle analyses performed for single once-through cycle improvements. The fuel cycle was simulated in 3-D with reactor operation consisting of spectral shift, followed by feedwater temperature reduction and, finally, power coastdown. The reactor operating conditions were the same as those used in the individual improvement analyses.

As in the case with EOC FWTR and PCD, it was necessary to include UO₂ bundles in the reload batch in order to achieve the required Pu material balance. The ratio of MO₂ bundles to UO₂ bundles in the reload batch was determined to be 8.5 (11% of the fuel assemblies were UO₂). The recycle bundle design was not optimized to take advantage of the special operating conditions of the combined improvements.

9.9.2 Combined Improvements Case U₃O₈ and SWU Requirements

The U₃O₈ and SWU requirements for equilibrium cycle conditions are summarized in Tables 9-4 to 9.6. In comparison to the "current improved" recycle design, improvements of 7.1 and 4.2% in U₃O₈ and SWU requirements, respectively, were observed.

When compared to the UO₂ design, the combined recycle design results in equilibrium case reductions in U₃O₈ and SWU requirements of 48 and 45%, respectively. The 30 year U and SWU requirements were reduced 45 and 44 percent, respectively relative to the UO₂ reference design.

The net U and SWU savings from the combination of spectral shift and EOC FWTR and PCD was less than the sum of the incremental savings from applying these alternatives. This may reflect the disadvantageous effect of extended burnup on the resource requirements. Although no change was made in fuel assembly fissile content relative to the current improved Pu recycle design, the burnup was extended by 5,750 MWd/MT by application of the combined improvements to the "current improved" Pu recycle design. Greater resource savings could have been achieved by reducing the initial fissile inventory of the Pu recycle fuel assemblies to compensate for the effects of the combined improvements on burnup.

9.9.3 Economic Incentives and Considerations

Fuel recycle in conjunction with several once-through cycle improvements is viable from an economic perspective although the incremental gain is not as substantial as for UO₂ fuel. Economic optimization of the burnup of the recycle design, including consideration of reprocessing and MO₂ rod fabrication costs, should result in improved overall economics, but may not increase the incentive to apply the once-through cycle improvements to Pu recycle. A decrease in MO₂ rod fabrication and reprocessing costs would tend to reduce the optimum burnup and result in coupled savings in uranium and SWUs. If fabrication and reprocessing costs are high, it will pay to design for greater burnup even though this may waste uranium and SWUs. Thus, the extent to which uranium and SWUs are saved by Pu recycle, will depend on the extent to which reprocessing and MO₂ rod fabrication costs can be minimized.

9.10 Pu RECYCLE SUMMARY AND CONCLUSION

The key results from the Pu recycle 3-D analyses are summarized in Tables 9-4 and 9-6. The results of applying the once-through cycle improvements to Pu recycle were in agreement with conclusions

from the preliminary impact analysis summarized in Table 9-2.

The overall conclusions from the Pu recycle design evaluations are as follows:

- Pu recycle even when non-optimally applied with increased burnup, results in greater than 20 percent savings in uranium and separative work requirements compared to the best once-through cycle designs;
- Extension of the burnup of Pu recycle fuel may be desirable if MO₂ fabrication and reprocessing costs increase more rapidly than uranium and separative work costs. This will increase uranium and separative work requirements.
- Most of the fuel cycle improvements under consideration for once-through cycles have smaller, but significant, impact on resource requirements for the recycle alternative.

10. INTERNATIONAL NUCLEAR FUEL CYCLE EVALUATION (INFCE) NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM (NASAP) DATA

A primary objective of this program was to provide data and information which could be used to formulate the BWR related United States light water reactor inputs to the International Nuclear Fuel Cycle Evaluation and to prepare NASAP reports. This section is a summary of the approximations which were used in deriving the inputs which were submitted to the U.S. Government for use in INFCE and NASAP reports and a summary of the key data.

10.1 ASSUMPTIONS, CRITERIA, METHODS OF APPROXIMATION

Thirty year cumulative uranium and SWU requirements were estimated by using reference 30 year fuel cycles for the reference and current improved BWR/6 designs and perturbing these cycles using estimates derived from the equilibrium cycle 3-D coupled nuclear-thermal-hydraulic-fuel cycle evaluations. The initial core design for the reference BWR was a uniform bundle enrichment design. The initial core was a 3 bundle enrichment design with natural uranium radial and axial blankets. The initial core for the best combined improvement design was a four bundle enrichment design with axial natural uranium blankets and radial blanket bundles of 0.91 w/o enrichment. This design had been extensively optimized to match the initial core with a 12 month cycle length equilibrium core. All of the combined improved designs utilized the Control Cell Core^{13,14} to improve performance.

The INFCE and NASAP fuel cycle data tabulations were based on different units for uranium requirements and it is important to recognize this in making comparisons. The INFCE data was tabulated in metric tons of U per GWe, while the NASAP data was tabulated in short tons of U308 per GWe. The conversion is:

$$(tU/GWe) \times 1.2996 = STU_{308}/GWe$$

10.2 INFCE FUEL CYCLE DATA

Table 10-1 is a summary of the reference BWR characteristics and performance specifications. Table 10-2 is a summary of the fuel cycle data in INFCE format. Data was supplied for both 70 and 75 percent capacity factors for the reference and extended burnup designs because

TABLE 10-1
GENERAL REACTOR PERFORMANCE SPECIFICATIONS

	<u>UNIT</u>	<u>REFERENCE BWR</u>
A. POWER PLANT PERFORMANCE		
TOTAL THERMAL POWER	MW	3800
ELECTRICAL POWER, GROSS	MW	1344
NET	MW	1270
THERMAL EFFICIENCY	%	33.4
B. REACTOR PARAMETERS		
CORE EFFECTIVE DIAMETER	m	5.01
CORE LENGTH	m	3.81
CORE VOLUME	ℓ	75045
CORE POWER DENSITY	kW/ℓ	51
SPECIFIC POWER	kW/kgU	23.77
COOLANT FLOW RATE	T/Sec	14.8
COOLANT INLET TEMPERATURE	°C	278.3
COOLANT OUTLET TEMPERATURE	°C	
COOLANT PRESSURE	BAR	73
CORE AVERAGE VOID FRACTION	%	43.1
TOTAL URANIUM LOADING	T	154.6
NUMBER OF FUEL ASSEMBLIES		848
C. FUEL ASSEMBLY		
NUMBER OF RODS		8X8-2
ACTIVE LENGTH	m	3.76
FUEL ROD PITCH	cm	1.615
ROD OUTER DIAMETER	cm	1.230
CLADDING THICKNESS	mm	0.8
TOTAL URANIUM LOADING	kg	182.3

TABLE 10-2

INFCE BWR FUEL CYCLE INFORMATION

ONCE THROUGH FUEL CYCLES

	REFERENCE BWR		EXTENDED BURNUP BWR-a)		COMBINED IMPROVED BWR-b)	PU RECYCLE REFERENCE -d)	COMBINED IMPROVED PU RECYCLE-f)
AVERAGE CAPACITY FACTOR, %	75	70	75	70	75	75	75
ENRICHMENT TAILS, %	0.2	0.2	0.20	0.2	0.20	0.2	0.20
FRACTION OF CORE REPLACED/ REFUELED	0.24	0.22	0.14	0.13	0.16/0.07-c)	0.25	0.20
REFUELING INTERVAL, YEAR	1.0	1.0	1.0	1.0	1.06	1.0	1.0
EQ ^M RELOAD ENRICHMENT, %	2.94	2.91	3.81	3.79	3.30	2.96 -e)	2.94 -e)
FISSILE FABRICATION LOSS, %	1.5	1.5	1.5	1.5	1.5	1.5	1.5
AVERAGE DISCHARGE EXPOSURE, MWd/t	28440	28440	47000	47000	43000/49000-c)	30250	36000
PEAK PELLETT EXPOSURE, MWd/t	<44000	<44000	<66000	<66000	<66000	<50000	<60000
NATURAL URANIUM-REQUIREMENTS, tU/GWe							
INITIAL CORE	349	349	382	382	342	382	342
ANNUAL EQ ^M RELOAD	157	146	125	116	112	87	81
30-YEAR CUMULATIVE	4956	4610	3901	3825	3564	3000	2713
SWU REQUIREMENTS, tSWU/GWe							
INITIAL CORE	198	198	235	235	198	235	198
ANNUAL EQ ^M RELOAD	124	114	109	101	92	71	68
30 YEAR CUMULATIVE	3887	3594	3366	3130	2836	2397	2194
FISSILE MATERIAL IN SPENT FUEL							
ENRICHMENT, W/O U235	0.96	0.93	0.68	0.66	0.51	0.62	0.53
PU-FISS	0.62	0.64	0.74	0.77	0.68	0.94	0.83
ANNUAL EQ ^M DISCH. U235	281	257	110	110	96	172	114
Kg/GWe PU-FISS	181	169	110	110	130	260	197
30 YR CUMULATIVE U235	8988	8265	3725	3446	3005	1050	870
Kg/GWe PU-FISS	5207	4877	3549	3451	3780	1110	1050
NATURAL URANIUM SAVINGS, 30 YR CUMULATIVE	---	---	19.7	17.0	28.1	39.5	45.2

- a) Includes axial & radial blankets, improved 3 enrichment initial core design and low leakage fuel management.
- b) Includes axial & radial blankets, low leakage fuel management, preplanned end-of-cycle feedwater temperature reduction and power coastdown to 80% power, spectral shift with flow control, & high enrichment discharge fuel rod reconstitution and reinsert.
- c) Normal reload fuel/reinserted fuel made up of high enrichment discharge rods.
- d) Pu recycle fuel designed with "island" assemblies in which the Pu recycle rods replace the high enrichment fuel rods in the center of the BWR fuel assembly. Includes axial blankets and low leakage refueling. Initial core contains no Pu recycle fuel.
- e) Total initial fissile inventory (U-235 + Pu fissile) in total heavy metal.
- f) Reload batch contains 136 MO₂ and 16 UO₂ bundles, MO₂ bundles are "island" design with MO₂ in central regions. Combined improvements same as for once-through combined case -b) except reconstitution omitted.

the Norwegian BWR design contribution to INFCE was at 70 percent capacity factor and it was desirable to provide a U.S. BWR for the same condition.

10.3 NASAP FUEL CYCLE DATA

Table 10-3 is a summary of the fuel cycle data in NASAP units. The transition cycle data is included in this tabulation.

10.4 DECAY HEAT AND DOSE RATE DATA AS A FUNCTION OF BURNUP

Information was needed on the effects of fuel burnup on the decay heat and dose rates as a function of time after removal of a fuel assembly from the reactor. The decay heat in kilowatts and dose rates in R/hr were calculated for discharge burnups of 28,400 and 47,000 MWd/MT. The dose rates were estimated for both 1 meter in water and one meter in air.

The results are plotted in Figures 10-1 to 10-3 as a function of time after discharge of a fuel assembly from the reactor in years. The isotopic compositions for the 47,000 MWd/MT case were based on the 3.81 enriched fuel assembly design used in the "current improved, extended burnup, once-through" evaluation tabulated in Table 10-3. Extended burnup has a small effect on decay heat and dose rates.

TABLE 10-3

INFCE, NASAP FUEL CYCLE U_3O_8 & SWU SUMMARY

	UNIT	REFERENCE BWR	CURRENT IMPROVED [*] ONCE-THRU	CURRENT IMPROVED [*] EXT BURNUP ONCE-THRU	COMBINED ONCE THRU	BEST COMBINED ONCE THRU	CURRENT IMPROVED [*] PU RECYCLE	COMBINED IMPROVEMENT PU RECYCLE
CAPACITY FACTOR	%	75	75	75	75	75	75	75
ENRICHMENT TAILS	%	0.20	0.20	0.20	0.20	0.20	0.20	0.20
RELOAD CORE FRACTION	%	0.24	0.24	0.14	0.14	0.23	0.25	0.20
REFUELING INTERVAL	YEARS	1.0	1.0	1.0	1.07	1.06	1.0	1.0
EQ ^M RELOAD ENRICHMENT	%	2.94	2.74	3.81	3.65	3.30	2.96	2.94 ⁴
FISSILE FABRICATION LOSS	%	1.5	1.5	1.5	1.5	1.5	1.5	1.5
AVG. DISCHARGE BURNUP	MW /MT	28439	28439	47000	51000	43000/49000	30250	36000
PEAK PELLET DISCHARGE BURNUP	MW /MT	<44000	<44000	66000	66000	66000	50000	60000
U ₃ O ₈ REQUIREMENTS								
INITIAL CORE	ST/GWE	454	496	496	496	443	496	443
TRANSITION CYCLES	ST/GWE	685	640	541	468	410	467	355
ANNUAL EQ ^M RELOAD	ST/GWE	204	190	162	146	145	113	105
30 YEAR CUMULATIVE	ST/GWE	6447	6051	5174	4693	4624	3901	3528
SWU REQUIREMENTS								
INITIAL CORE	10 ³ SWU/GWE	198	235	235	253	198	235	198
TRANSITION CYCLES	10 ³ SWU/GWE	465	416	317	307	246	316	228
ANNUAL EQ ^M RELOAD	10 ³ SWU/GWE	124	111	109	96	92	71	68
30 YEAR CUMULATIVE	10 ³ SWU/GWE	3887	3490	3366	3012	2836	2397	2194
PERCENT IMPROVEMENT RELATIVE TO REFERENCE O-T DESIGN:								
U308-EQUILIBRIUM	---	---	6.9	20.6	28.4	28.9	44.6	48.5
-30 YR. CUMULATIVE	---	---	6.1	19.7	27.1	28.2	39.5	45.2
SWU -EQUILIBRIUM	---	---	10.5	12.1	22.5	25.8	42.7	45.2
-30 YR. CUMULATIVE	---	---	10.2	13.4	22.5	27.0	38.3	43.6
CURRENT IMPROVED O-T DESIGN								
U308-EQUILIBRIUM	---	-7.3	---	14.7	23.1	23.7	40.5	44.7
-30 YEAR CUMULATIVE	---	-6.5	---	14.3	22.4	23.8	35.5	41.7
SWU -EQUILIBRIUM	---	-11.7	---	1.8	13.5	17.1	36.0	38.7
-30 YEAR CUMULATIVE	---	-11.4	---	3.6	13.7	18.7	31.3	37.1
BEST COMBINED O-T DESIGN								
U308 -EQUILIBRIUM	---	-40.7	-31.0	-11.7	-0.7	---	22.1	27.5
-30 YEAR CUMULATIVE	---	-39.6	-31.1	-11.9	-1.5	---	15.5	23.5
SWU -EQUILIBRIUM	---	-34.8	-20.7	-18.5	-4.3	---	22.8	26.1
-30 YEAR CUMULATIVE	---	-37.1	-23.1	-18.7	-6.2	---	15.5	22.6

^{*}Includes radial and axial blankets, 3 enrichment, initial core, and low leakage refueling patterns.

⁴Reload batch contains 136 MO₂+16UO₂ bundles.

Figure 10-1. Decay Heat Generation Versus Cooling Time Per Fuel Bundle

Figure 10-2. Dose Rate Versus Cooling Time (Receiver 1 Meter Away in H₂O) Per Fuel Bundle

Figure 10-3. Dose Rate Versus Cooling Time (Receiver 1 Meter Away in Air) Per Fuel Bundle

11. REFERENCES

1. S. R. Specker, R. L. Crowther, J. M. Haun, "Optimization of the BWR for Once-Through Fuel Cycles," Trans. ANS 30, 277-79 (Nov. 1978).
2. R. L. Crowther, "The Physics of BWRs", Invited Paper, ANS National Topical Meeting on New Developments in Reactor Physics and Shielding, Kiamesha Lake, New York, CONF-720901 (September 1972).
3. C. L. Martin, "Lattice Physics Methods", NEDO-20913-A (July 1976). Also C. L. Martin "Lattice Physics Methods Verification", NEDO-20939 (June 1976).
4. R. L. Crowther, et al, "Feedback of Reactor Operating Data to Nuclear Methods Development," invited paper presented at ANS Topical Conference on Advances in Reactor Physics, Gatlinburg, Tennessee. CONF-780401 (1978).
5. R. L. Crowther, W. P. Petrick and A. Weitzberg, "Three-Dimensional BWR Simulation", CONF-690401 (April 1969). Also J. A. Woolley, "Three-Dimensional BWR Core Simulation", NEDO-20953 (May 1976).
6. R. L. Crowther, S. P. Congdon, B. W. Crawford, C. M. Kang, A. Leonard, C. L. Martin, C. T. McDaniel, L. A. Schwager, R. A. Wolters, "Three-Dimensional Simulation of Large Power Reactors", (Invited Paper), Presented at International Topical Meeting on "Advances in Mathematical Methods for the Solution of Nuclear Engineering Problems", Munich, Germany (April 1981).
7. R. L. Crowther, "Methods of Fuel Management Analysis--An Overview", Invited Paper, ANS Topical Conference on Mathematical Models and Computational Techniques for Analysis of Nuclear Systems, Ann Arbor, Michigan (April 1973).
8. R. Van Houten, "Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout", NUREG-0562 (June 1979).
9. D. M. Bishop and W. E. Baily, "Nondestructive Examination of Plutonium Recycle Fuel Rods Irradiated to 30,000 MWD/t", NEDO-12552 (1974).
10. R. L. Crowther, "Lectures on Control Power Reactors", IAEA Third International Advanced Summer School in Reactor Physics Herceg-Novi Yugoslavia (1970) IAEA Technical Report Series 143.
11. P. G. Aline, R. O. Brugge, W. R. Clancey, R. L. Crowther, D. L. Fischer, C. P. Ruiz, "Fuel Management and Isotopic Composition Prediction and Experiment in Light Water Power Reactors", Proceedings of International Conference on "The Physics Problems in Thermal Reactor Design", 297, London (1967).
12. F. C. Schoenig, et al, "Nondestructive Evaluation of Multizoned Gadolinia Bearing UO₂ Rods", Trans. ANS 33, 448-49 (1979).
13. S. R. Specker, L. E. Fennern, R. E. Brown, K. L. Stark, R. L. Crowther, "The BWR Control Cell Core Improved Design", Trans. ANS 30, 336-38 (Nov. 1978).

14. L. E. Fennern, R. L. Crowther, P. J. Savoia, S. R. Specker, K. L. Stark, "Improved BWR Control Cell Core Design Concept", Trans. ANS, 33, 642-44 (Nov. 1979)
15. N. L. Shapiro, et al, "Assessment of Thorium Fuel Cycles in Pressurized Water Reactors", EPRI-NP-359 (1977).
16. R. L. Crowther, "Use of Thorium in Boiling Water Reactors", Journal of Nuclear Energy Parts A/B, V. 16, 181 (1962).
17. D. B. Townsend, R. L. Crowther, R. A. Wolters, "A Once-Through Thorium Cycle for BWRs", Trans, ANS 32, 354-55 (June 1979).
18. R. L. Crowther, P. J. Savoia, P. Wei, R. A. Wolters, "BWR Fuel Management Improvements for Once-Through Fuel Cycles", (Invited Paper), Trans. ANS 33, 396-97 (Nov. 1979).
19. H. E. Williamson, et al., "Assessment of Utilization of Thorium In BWRs", NEDG-24071 (1978).
20. M. J. Colby, D. B. Townsend, C. L. Kunz, "Safety Analysis of Thorium-Based Fuels In the General Electric Standard BWR", NEDG-24817 (1980).
21. International Atomic Energy Agency, "Advanced Fuel Cycle and Reactor Concepts Report of Working Group 8", INFCE/PC/2/8. Vienna. (Jan. 1980).
22. R. L. Crowther, G. C. Hopkins, S. K. Lam, C. D. Sawyer, R. A. Wolters, "BWR Pu Recycle", Invited Paper, Trans. ANS Vol. 16 n2, (November 1973).
23. R. L. Crowther, and D. L. Fischer, "Nuclear Characteristics of Large Advanced Boiling Water Reactors", Proc. of ANS Topical Conference on Nuclear Performance of Power Reactor Cores, San Francisco (Sept. 1963).
24. RL Crowther and LK Holland, "Control Management and Automation of the Boiling Water Reactor", Nuclex 69, Basel, Switzerland (Oct. 1969).