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ONCE-THROUGH THORIUM FUEL CYCLE EVALUATION FOR TVA'S BROWNS FERRY-3 BOILING WATER REACTOR

COMPILED BY
G. C. HOPKINS

PREPARED FOR THE TENNESSEE VALLEY AUTHORITY
UNDER SUBCONTRACT 79P67-151502
U.S. DEPARTMENT OF ENERGY
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ONCE-THROUGH THORIUM-FUEL-CYCLE EVALUATION
FOR TVA's BROWNS FERRY-3 BWR

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Compiled by
G. C. Hopkins
Program Manager

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Approved: *J. E. Wood*
J. E. Wood, Manager
Core Methods and Analysis

Approved:

P. J. Savoia
P. J. Savoia, Manager
Nuclear Design Technology

Approved:

R. L. Crowther
R. L. Crowther, Manager
Reactor Physics Technology

NUCLEAR FUEL & SERVICES DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

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ABSTRACT

The once-through thorium fuel cycle involves producing U-233 in ThO_2 rods which are located in the high-importance, high-flux regions of a boiling water reactor (BWR) lattice, followed by removal of the Thorium-U-233 rods after fuel discharge and reconstitution of these fuel rods into special blanket fuel assemblies which are loaded at the core periphery for continued irradiation. The blanket assemblies have a breeding ratio above 1.0 and a relatively flat reactivity versus burnup which gives them potential to remain on the core periphery for long irradiation. Previous evaluations of the concept indicated potential to reduce the number of fuel assemblies required for refueling, to reduce uranium and separative work unit (SWU) requirements, and to improve core reactivity coefficients and dynamic performance. Analyses had indicated that design for the once-through thorium cycle would reduce BWR steam void reactivity coefficients by 10 to 15 percent, which would have a favorable effect on coupled nuclear-thermal-hydraulic stability and transient response. Large quantities of U-233 are built up in the ThO_2 rods which also could be considered for recovery and recycle.

This report documents benchmark evaluations to test thorium lattice predictive methods and neutron cross sections against available data and summarizes specific evaluations of the once-through thorium cycle when applied to the Browns Ferry-3 BWR. It was concluded that appreciable uncertainties in thorium cycle nuclear data cloud the ability to reliably predict the fuel cycle performance and that power reactor irradiations of ThO_2 rods in BWRs are desirable to resolve uncertainties. Benchmark evaluations indicated that the ENDF/B-IV data used in the evaluations should cause an underprediction of U-233/ ThO_2 fuel reactivity, and, therefore, the results of the preliminary evaluations completed under the program should be conservative.

The preliminary analyses which were completed with available cross-section data and methods indicated that the once-through cycle produced an increase in uranium and SWU requirements and depreciated fuel cycle economics. Alternative blanket fuel assembly designs and changes in flow pattern were identified which could improve blanket and coupled-core performance. It also was observed that performance of the once-through thorium cycle will improve as other BWR

uranium utilization improvement options are applied which extend burnup with no fuel enrichment increase, such as end-of-cycle coastdown, spectral shift with flow control, end-of-cycle feedwater temperature reduction, spectral shift with axial burnup shape optimization, etc. Additional improvement of both the prebreeder bundle design and breeder blanket design also may be possible through design optimization. However, these once-through thorium cycle improvements were not evaluated in depth in this program.

Removal of conservatism in the analyses and future design improvements have potential to improve the fuel cycle performance of the once-through thorium cycle such that economic and performance benefits can be realized.

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PRINCIPAL AUTHOR

M. J. Colby

CONTRIBUTING AUTHORS

D. C. Cambra

R. L. Crowther

D. W. Harris

P. C. Rose

P. J. Savoia

1. INTRODUCTION

Problems in application of thorium cycles include greater fissile inventory requirements, 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 once-through thorium cycle has been developed¹ to minimize the effect of thorium on the fissile inventory. The once-through cycle is initiated with ThO_2 fuel containing no initial fissile material, and U-233 recovery and recycle are not required. The design made advantageous use of the inherent heterogeneity and other characteristics of the boiling water reactor (BWR) lattice to produce U-233 in ThO_2 with minimal power distribution penalties and decreased thermal performance. Standard BWR fuel assembly hardware was used to make the design retrofitable with minimum manufacturing impact.

The design utilized heterogeneous location of thorium in the highest thermal flux locations of the BWR lattice to maximize U-233 production rate. After fuel assembly discharge of the prebreeder bundles (PBBs), the U-233-bearing ThO_2 rods were reassembled into breeder blanket bundles (BBBs) that were located at the core periphery. The BBB reactivity is relatively flat with burnup, 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. In the once-through thorium cycle, the BBBs replace the last cycle minimum reactivity fuel on the core periphery. Thus, once the BBBs are in place, the number of bundles needed in each cycle to sustain the reactor is reduced, thereby conserving uranium.

Analyses of the Browns Ferry-3 PBB and BBB bundle designs were carried out with a 100-energy group spatial/spectral model. However, because of uncertainties in the analysis of mixed lattices of ThO_2/UO_2 rods, full spectrum Monte Carlo analyses, experimental benchmark analyses, and detailed 3-D coupled nuclear-thermal-hydraulic-fuel cycle analyses were carried out to test the validity of the promising performance and trends that had been observed in analyses to date.²

2. SUMMARY AND CONCLUSIONS

The critical experiment benchmark analyses which were completed indicated relatively large uncertainties in thorium fuel cycle nuclear data and basic neutron cross sections. The evaluations indicated that the Evaluated Nuclear Data File/B, Version 4 (ENDF/B-IV) data set which was applied in the analyses should be conservative and should underpredict the worth of U-233/ThO₂ fuel rods when loaded in BWR lattices and cores. It was concluded that power reactor irradiations of ThO₂ rods are desirable to resolve the uncertainties and to provide more reliable bases for evaluation of this fuel cycle improvement option.

The preliminary evaluations of applications of the once-through thorium cycle to Browns Ferry-3 indicated that uranium and separative work unit (SWU) utilization would be reduced by application of this design alternative. In current GE-BWR design practice, low-leakage refueling patterns are used in which the lowest reactivity last-cycle fuel is loaded at the core periphery, where it acts as a blanket to reduce neutron leakage and improve fuel cycle performance. Thus, the viability of the breeder blankets is sensitive to the reactivity of the blanket bundles relative to the last-cycle fuel which they displace.

Figures 2-1 and 2-2 illustrate the reactivity trend. Except for the natural uranium assemblies used as blanket bundles for initial cores, the reactivity of the breeder blanket bundles is appreciably less than the reactivity of last-cycle peripheral fuel of current design. However, as burnup is extended (Figure 2-2) the reactivity of the breeder blanket bundles approaches or exceeds that of last-cycle fuel.

Several improvements were identified for increasing the reactivity of the breeder blanket bundles:

- (a) Reconstitute the 8x8 U-233/ThO₂ fuel rods into 9x9 or 10x10 blanket assemblies. This appears feasible because the power at the core periphery is small, and the peripheral bundle locations have greater inlet flow orificing in current design. The extra heat transfer surface of 9x9 or 10x10 assemblies would aid critical heat flux (CHF) performance. The reactivity trends between 8x8, 9x9, and 10x10 breeder blankets are shown in Figure 2-1.

Figure 2-1. K-Infinity vs. Energy Equivalent Burnup of Reference Design

Figure 2-2. K-Infinity vs. Energy Equivalent Burnup for Extended Burnup Design

- (b) Reduce the flow in the peripheral assemblies to take advantage of the positive steam void reactivity coefficient of the breeder blanket bundles. This could be accomplished either by revising the inlet orificing or by use of 9x9 or 10x10 breeder blanket bundles.
- (c) Increase flow in the center of the core to reduce steam void fraction and increase core reactivity. Either increased peripheral orificing or use of 9x9 or 10x10 breeder blanket bundles will tend to increase flow and reduce steam void fraction in the central regions of the core.

The fuel cycle performance of 8x8 breeder blanket bundles only was evaluated in this program.

The fuel cycle performance also is sensitive to the prebreeder design. The prebreeder design used in the evaluations required 2.4 percent more uranium and 3.0 percent more SWUs than the reference design, whereas previous estimates had indicated that the PBB/BBB designs would result in a savings of approximately 4 percent in uranium, ignoring any U-233 recycle. Hence, this had an important negative impact on the fuel cycle. Further optimization of prebreeder design has potential to reduce these penalties.

Analyses of the prebreeder bundle performance indicated that the ThO_2 rods made the BWR steam void reactivity coefficient 10-15 percent less negative because of combined effects of the ThO_2 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 was that axial power shapes and burnup dependent fuel assembly power mismatch were improved. The requirements for burnable absorbers also were decreased. These benefits were not fed back to improve fuel cycle economics.

2.1 BENCHMARK EVALUATIONS

Difficulties were encountered in benchmarking methods and neutron cross sections against available thorium benchmark experiments. Selected Babcock & Wilcox (B&W) thorium-uranium physics experiments (TUPE) critical (U/Th) O_2

configurations³ were analyzed to benchmark ENDF/B-IV cross-section data. It was concluded that the greater than 30 percent net neutron leakage from the small criticals and/or problems with the data caused inconsistencies in the predicted critical eigenvalues. Analyses with Monte Carlo methods did not resolve the observed discrepancies but did indicate that the measured critical buckling values are unreliable for predicting neutron leakage. Similar problems in analysis of the TUPE experiments have been reported by others,⁴ and large potential discrepancies in thorium cross-section data, including the most recent ENDF/B-V data, have been suggested by deSaussure and Perez.⁵ This led to the conclusion that there is appreciable unresolved uncertainty in the analysis of thorium lattices which could affect the viability of the once-through design option. The results also are sensitive to the Pa-233 cross-section data, which affects the conversion ratio, and the accuracy of Pa-233 cross sections could not be verified.

Comparisons of ENDF/B-IV and ENDF/B-V data⁴ indicate that the ENDF/B-IV data cause underpredictions of the eigenvalue of the BNL U-233/ThO₂/H₂O exponential experiments by 1.3 to 1.7 percent. Three-dimensional Monte Carlo analyses of the B&W TUPE 15B U-235/ThO₂/H₂O assembly with ENDF/B-IV data resulted in a 0.7 percent underprediction of reactivity. Hence, it was concluded that the use of ENDF/B-IV data in the evaluations probably has resulted in an underprediction of the worth of the once-through thorium cycle.

Fuel assemblies used in large power reactors operate in a low leakage, nearly infinite lattice environment. Therefore, power reactor predictions are not as sensitive to some nuclear parameters (e.g., fission spectrum needed for precise simulation of critical assemblies with large neutron leakage). Power reactor measurements of thorium rod performance, U-233 buildup, and conversion ratio are desirable to resolve uncertainties in the predictions.

2.2 PBB/BBB FUEL CYCLE PERFORMANCE

The preliminary analyses indicated that use of the PBB design concept in the Browns Ferry-3 reactor core would result in thermal operating margins comparable to those of the reference UO₂ core. However, cold shutdown margin was

reduced by 0.004 ΔK but still met the design limit. For the design evaluated, the basic PBB fuel cycle indicated reduced uranium utilization of approximately 2.4 percent and increased SWU cost of 3.0 percent.

The reconstituted 8x8 BBB peripheral fuel designs possessed less reactivity and operated at lower powers than the last cycle fuel they were intended to replace, thereby increasing the energy requirements of the remainder of the core. This would necessitate employment of a larger batch fraction or higher enrichment reload fuel to maintain the same cycle energy output, which would further decrease uranium utilization. The lower than expected BBB reactivity was a result of lower than expected concentrations of U-233 in the irradiated ThO₂ fuel pins that were removed from the PBBs after discharge.

2.3 EXTENDED BURNUP, ADVANCED BWR FUEL CYCLE PBB/BBB DESIGNS

As fuel burnup is extended, refueling batch sizes decrease, and the reactivity of the discharge fuel decreases. This is a favorable trend for the once-through thorium cycle because, if burnup is extended sufficiently, the reactivity of the last cycle fuel will be less than the reactivity of the breeder blanket bundles. Also, the PBB bundles have a flatter reactivity slope with burnup and reduce the need for burnable absorber pins, which aids design for extended burnup. If burnup is extended by BWR design alternatives such as flow control spectral shift, which require no fuel enrichment increase, the discharge fuel reactivity decreases more rapidly than if burnup is increased by increasing enrichments. The fissile inventory of the discharge thorium rods also increases with burnup.

The trend in advanced BWR fuel cycles is to maximize SWU savings by extending burnup with no initial enrichment increase.^{2,6} Hence, as BWRs are optimized to achieve maximum SWU requirements and fuel cycle costs, the economic viability of the once-through thorium cycle should improve.

3. REFERENCE REACTOR AND CORE DESIGN

3.1 REFERENCE REACTOR AND DESIGN REQUIREMENTS

The reference reactor selected for all the evaluations conducted under Phase I of the Improved Uranium Utilization Program was the Browns Ferry-3 BWR/4 reactor. This plant was selected by Tennessee Valley Authority (TVA) because it is representative of all three Browns Ferry units and, unlike the other units, has been initially loaded and refueled with 8x8 fuel assemblies. This makes the Browns Ferry-3 reactor a good candidate for any follow-on demonstration program using advanced 8x8 design concepts evaluated in the present studies.

The initial core was loaded with the original 8x8 fuel with a 144-in. active fuel length. The retrofit 8x8 fuel design with a 150-in. active fuel length and axial natural uranium blankets was introduced in Cycle 2 and has been used in all subsequent cycles. The equilibrium core fuel cycle, fueled exclusively with the retrofit design, was selected as the basis for comparison against the once-through thorium fuel cycle. The equilibrium fuel cycle requirements provide a consistent basis for the fuel cycle cost evaluations performed under the program.

Some of the key reactor design features and ratings of the Browns Ferry-3 reactor are listed in Table 3-1. The nuclear design evaluations have utilized these parameters along with the design basis requirements given in Table 3-2.

3.2 REFERENCE CORE DESIGN

The reference core design is based upon the control cell core (CCC) fuel loading and operating strategy. This design was developed by GE in 1975 and has completed successful demonstrations in both a BWR/3 and BWR/4 reactor.⁷⁻¹⁰ Based upon these successful demonstrations, as well as for the reasons discussed below, TVA has selected the CCC as the reference core design for Browns Ferry-3, commencing with Cycle 4. On this basis an equilibrium CCC cycle was used for comparison of fuel cycle performance of the once-through thorium fuel cycle concept.

The CCC design is based on an operating strategy in which control rod movement, used to offset reactivity changes during power operation, is limited to a fixed group of control rods that are surrounded by low power fuel (Figure 3-1). Each of these rods and its four surrounding fuel assemblies comprise a control cell. All other control rods are normally completely withdrawn from the core while operating at power. Low reactivity fuel assemblies are placed in the control cells so that control rod motion occurs adjacent only to low power fuel. Hence the control rod functions are separated, with some control rods used only for shutdown and other control rods in the designated control cells used for both power shaping and shutdown.

The CCC design increases the capacity factor by minimizing or eliminating control rod pattern swaps, by reducing the effects of pellet-clad-interaction constraints, and by decreasing startup time. Thermal margins are increased by elimination of partially inserted control rods adjacent to high-power fuel assemblies or fuel containing partially depleted burnable absorbers, which can increase the peak kW/ft above the tip of the control blade. For further details on the CCC design see References 6-10.

Using the CCC as a design basis, the reference equilibrium fuel cycle was developed. The reference evaluation was based on existing 8x8 fuel designs that were developed under separate GE programs and therefore provides a standard against which to compare the once-through thorium fuel cycle.

The evaluations were based upon detailed three-dimensional simulations of the Browns Ferry-3 reactor, using standard BWR design computer programs and procedures. The once-through thorium fuel assembly was evaluated using advanced assembly design methods. The design and performance of the reference equilibrium cycle is presented in the following subsection.

3.3 REFERENCE EQUILIBRIUM CYCLE

The reference loading pattern is presented in Figure 3-2. This loading is a one-stream refueling pattern* utilizing 264 assemblies with a bundle average

*A one-stream refueling pattern was applied, rather than the two-stream reference pattern used in other evaluations, to simplify the analyses.

enrichment of 2.84 weight percent U-235 (Bundle 2.84U2). The 2.84U2 bundle employs 6 inches of natural uranium on each end of the active fuel column with a uniform axial enrichment and gadolinia loading.

The objective of the reference loading was to maximize energy while maintaining adequate thermal margins. Therefore the fresh fuel in the center of the core was generally loaded into four-bundle cells with fuel in its second cycle of residence to help reduce power peaking in the fresh fuel.

Table 3-1
KEY REACTOR DESIGN FEATURES FOR BROWNS FERRY-3

Rated Thermal Power, MWt	3293
Net Electrical Output, MWe	1067
Rated Core Flow, lb/hr	102.5×10^6
Number of Fuel Assemblies	764
Number of Control Blades	185
Active Fuel Height,* inches	
original 8x8 fuel	146
retrofit 8x8 fuel	150
Core Average Power Density, kW/l	
with original 8x8 fuel	50.0
with retrofit 8x8 fuel	48.7

*The original 8x8 fuel contains one water rod and no natural uranium at either end of the bundle. The retrofit 8x8 fuel contains two water rods and 6 inches of natural uranium at both the top and bottom of the active fuel column. The fuel lattice geometry is such that two adjacent sides have wide water gaps to accommodate the control blade (D-lattice configuration).

Table 3-2
NUCLEAR DESIGN REQUIREMENTS

Thermal Limits

Maximum Linear Heat Generation Rate (MLHGR)	13.4 kW/ft
Minimum Critical Power Ratio (MCPR)	1.26*

Reactivity Limits

Minimum Cold Shutdown Margin (stuck rod)	1.0% $\Delta k/k$
Minimum Hot Excess Reactivity	1.0% $\Delta k/k$

Equilibrium Cycle

Cycle Length	18 months
Average Capacity Factor	74%

*Can vary depending upon results of transient analyses.

Figure 3-1. Typical Control Cell Locations for a 764-Bundle BWR Core

Figure 3-2. Browns Ferry-3 Reference Equilibrium Cycle CCC Loading Pattern for Once-Through Thorium Cycle Evaluations

4. LATTICE PHYSICS AND CORE SIMULATOR METHODS

4.1 LATTICE PHYSICS

BWR lattice physics methods 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, cladding, 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.

Improved methods which increase the accuracy of prediction of the two dimensional thermal spectrum variation within a fuel assembly and of Thorium-U-233 fuel were applied in the evaluations. It was necessary to analyze the reference design with the same methods to provide a reliable relative basis for comparison.

4.2 THREE-DIMENSIONAL BWR SIMULATOR

Core physics calculations were performed using a three-dimensional coupled nuclear-thermal-hydraulic simulator which represents the BWR core and coupled system hydraulics. The model consisted of coarse mesh, few group, static diffusion theory coupled to static parallel channel thermal-hydraulics. The simulator is used to predict detailed three-dimensional design thermal performance as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables.

5. THORIUM BENCHMARK COMPARISONS

The Browns Ferry-3 prebreeder bundle and breeder blanket bundle were designed using a 100-energy group spatial/spectral model which employs ENDF/B-IV cross-section data. However, because of uncertainties in the basic cross-section data used for the analysis of mixed lattices of (U/Th)O₂ rods, it was concluded that the fuel bundle assembly design method should be benchmarked by comparison with the Babcock & Wilcox thorium-uranium physics experiments³ and, by comparison, with detailed Monte Carlo analyses.

5.1 B&W TUPE ANALYSES

The main focus of the benchmarking effort was on the modeling of two selected B&W TUPE critical configurations to assess the suitability of the basic Th-232 ENDF/B cross-section data used for simulation of thorium-fueled BWR lattices. Infinite lattice analyses of the two critical lattices were performed using both the bundle design and full spectrum Monte Carlo simulation. Also, a three-dimensional simulation of one of the configurations was completed to investigate effects of the leakage of neutrons from the small criticals on the calculated critical eigenvalue.

5.1.1 B&W Cores 15A and 15B Configurations

The two B&W critical configurations (15A and 15B) that were selected for simulation are shown in Figures 5-1 and 5-2. Core parameters are given in Table 5-1. Table 5-2 lists the properties of the (U/Th)O₂ fuel pins. These two criticals were selected for analysis because their moderator/fuel (M/F) volume ratios of 1.642 and 1.385 simulate the hot operating H/U ratios of BWR lattices.

5.1.2 Infinite Lattice Analyses

Infinite lattice analyses of the B&W 15A and 15B critical configurations were performed using both the bundle design method and a full spectrum Monte Carlo simulation. The parameters of most interest in these simulations were the calculated K_{∞} and K_{eff} , which are indicators of total reaction rates.

Table 5-3 compares predicted to reported measured values. As shown, the calculated K_{eff} of both the bundle design and Monte Carlo analyses, using the measured bucklings, overpredicted the measured values by 2-3 percent. The design method predicted ~0.8 percent less reactivity than Monte Carlo analyses.

The B&W cores have diameters on the order of 30-40 cm and critical heights of 125-135 cm. These small core dimensions result in greater than 30 percent net leakage of neutrons from the cores. With core leakages on the order of 30 percent, an error of a few percent in buckling could give a significant error in the calculated critical eigenvalue. A three-dimensional Monte Carlo simulation of the B&W 15B critical configuration was performed to investigate the effects of leakage on the critical eigenvalue and to test the reliability of the measured bucklings.

5.1.3 Three-Dimensional TUPE Monte Carlo Simulation

The B&W thorium-uranium fueled critical configuration with a rod-to-rod pitch of 0.4027 inch, Core 15B, was modeled using full spectrum Monte Carlo methods to simulate a 1/8 sector of the core in three dimensions. A schematic of the simulation geometry is given in Figure 5-3. The entire core assembly configuration, including the base aluminum plate, top and bottom aluminum grid structures, air/water interface, and cylindrical water moderator, was accounted for in the simulation. Also, each fuel rod was explicitly represented.

The B&W configuration has a measured critical eigenvalue of $K_{eff} = 1.000 \pm 0.001$. The three-dimensional Monte Carlo simulation of the core resulted in a $K_{eff} = 0.9927 \pm 0.0036$. This result is in reasonable agreement with another Monte Carlo simulation¹¹ of the 15B core using ENDF/B-IV cross-section libraries, which yielded $K_{eff} = 0.9947 \pm 0.0013$. However, other Monte Carlo 3-D simulations have overpredicted the K_{eff} of B&W TUPE configurations with lower and higher moderator-to-fuel ratios than that of Core 15B (Table 5-4). Therefore, a consistent bias of K_{eff} between Monte Carlo and the B&W TUPE lattices cannot be determined using this analysis and the Reference 11 results.

5.1.4 Critical Eigenvalue Sensitivity to Small-Core Geometries

As noted earlier, the evaluation of ENDF/B Th-232 cross-section accuracy by simulation of the B&W TUPE critical configurations was complicated by the small-core dimensions and resultant greater-than-30-percent net leakage of neutrons from the fuel lattice assembly. The analyses indicated that infinite lattice evaluations combined with measured bucklings did not give accurate K_{eff} values. The K_{eff} for cores with high leakage is extremely sensitive to the buckling, a difficult parameter to measure with accuracy.

The 3-D Monte Carlo analyses¹¹ of the B&W TUPE configurations indicated that the ENDF/B-IV cross sections resulted in too small a predicted critical eigenvalue. This result is consistent with reported analyses of exponential experiments but is inconsistent with the Reference 11 reported analyses for the other TUPE experiments (Table 5-4). Hence, it is most likely that the ENDF/B-IV data are causing a reactivity prediction which is too low, but this conclusion cannot be firmed up without resolving the TUPE experiment inconsistencies.

Further comparisons with other thorium critical assembly benchmarks are desirable, applying 3-D Monte Carlo to reduce leakage uncertainties. Prototypical power reactor data are needed to resolve the major uncertainties.

5.2 MONTE CARLO/BUNDLE DESIGN COMPARISONS FOR TUPE 15B CORE

The Monte Carlo infinite lattice simulations of the B&W 15B core were used to benchmark the bundle design methods, which applied ENDF/B-IV cross-section libraries. The Th-232 neutron absorption rates of the B&W 15B core determined by the Monte Carlo and design method analyses (Subsection 4.1.2) are given in Table 5-5. The Monte Carlo and design method absorption rates agree within 0.5 percent in the fast and thermal neutron energy regions. However, the bundle design method predicts ~2 percent greater neutron absorption than Monte Carlo in the resonance energy region. Overall, the bundle design method overpredicts Th-232 neutron absorption by 2.4 percent, which accounts for its underprediction of the infinite eigenvalue relative to Monte Carlo, noted in the previous section.

5.3 MONTE CARLO DESIGN METHOD PBB BENCHMARK COMPARISONS

Based on earlier studies,² on the order of 1 wt percent U-233 should be present in the ThO₂ rods positioned in the lattice corners immediately prior to discharge of the PBBs. Therefore, as a benchmark, a simulation of a typical PBB combination with 5 ThO₂ rods containing 1 wt percent U-233 was modeled using both the Monte Carlo and bundle design methods at core average voids (40%). A comparison of the fission densities and lattice eigenvalue is given in Figure 5-4. Table 5-6 compares bundle absorption rates for U-233 and Th-232.

5.3.1 Local Fission Densities and Eigenvalue

As is shown by Figure 5-4, there is approximately 1 percent difference in relative fissions between Monte Carlo and the bundle design analysis observed in the U-233 bearing ThO₂ rods. Most of the fission rates differ by less than 2 percent, with the greatest disagreement along the bundle diagonal and along the bundle edge near the narrow-wide corner, where differences range from 2-4 percent. The bundle design method overestimated the Monte Carlo eigenvalue by only 0.29 percent, which is relatively good agreement.

5.3.2 Bundle Absorption Rates

The bundle design method overestimates the Monte Carlo Th-232 total absorption rate by 2.9 percent. This agrees well with the 2.4 percent greater Th-232 absorption observed in Subsection 5.2 for the infinite lattice evaluations and is attributed to resonance capture overprediction. The design method also overpredicted the thermal absorption in U-233, which probably is responsible for the design method eigenvalue being slightly greater than that of Monte Carlo, despite the predicted greater thorium absorptions.

These benchmark results were used to develop bases for nuclear uncertainty analyses to test the sensitivity of the design to the methods and nuclear data uncertainties.

Table 5-1
CRITICAL CONFIGURATION PARAMETERS

	Core	
	15A	15B
Fuel Rod Pitch (in.)	0.3850	0.4027
Moderator/Fuel Ratio	1.379	1.642
Critical Water Height (cm)	127.21	133.44
Volume Fractions		
Water	0.4942	0.5376
Pellets	0.3582	0.3274
Aluminum (Al)	0.0875	0.0800
Void	0.0602	0.0550

Table 5-2
PROPERTIES OF 15A AND 15B FUEL PINS

Outside Diameter (in.)	0.309
Clad Material	Al-2S
Clad Thickness (in.)	0.014
Clad Inside Diameter (in.)	0.281
Active Pin Length (in.)	60
Pellet Diameter (in.)	0.260
Effective Pellet Density (gm/cm ³)	8.35
U-235 Mass/Pin (gm)	24.04
Number of Th-232 Atoms/Number of U-235 Atoms	15.00
Th-232 Mass/Pin (gm)	405.0
Fuel Pellet Mass/Pin (gm)	434.6

Table 5-3
CALCULATED VERSUS MEASURED PARAMETERS
FOR B&W 15A AND 15B CORES

	Core	
	15A	15B
<u>MEASURED:</u>		
B^2 ($\text{cm}^{-2} \times 10^{-4}$)	85.31 ± 0.26	94.25 ± 0.29
K_{eff}	$1.000 \pm \text{assumed error of } 0.001$	$1.000 \pm \text{assumed error of } 0.001$
K_{∞}^* (inferred from meas.)	1.391 ± 0.002	1.407 ± 0.003
<u>PREDICTED:</u>		
Infinite Lattice Monte Carlo		
K_{eff}^*	1.0257 ± 0.0027	1.0284 ± 0.0028
K_{∞}	1.4269 ± 0.0015	1.4467 ± 0.0015
Bundle Design Method		
K_{eff}^*	1.018 ± 0.002	1.020 ± 0.002
K_{∞}	1.4158 ± 0.0002	1.4353 ± 0.0001
Calculated Migration Area M^2 (cm^2)	45.85	43.16

* K_{∞} inferred from the measurement using $K_{\infty} = K_{\text{eff}} * (1 + M^2 B^2)$; M^2 and B^2 values are listed above (B^2 is the measured geometric buckling), and M^2 is the predicted migration area.

Table 5-4
 CALCULATED CRITICALITY OF TUPE LATTICES
 (FULL CORE RCP01)¹¹

<u>Lattice Ident.²</u>	<u>Mod./Fuel Volume Ratio</u>	<u>Number of Rods</u>	<u>Moderator Height (cm)</u>	<u>Calculated Net Leakage</u>	<u>Calculated K_{eff} ± *</u>
15A	1.379	1108	127.21	0.313	1.0002 ± 0.0014
15B	1.642	880	133.44	0.325	0.9947 ± 0.0013
15D	2.945	514	135.05	0.317	1.0056 ± 0.0014
25D	3.636	1176	140.50	0.206	1.0075 ± 0.0013
25√2B	5.794	1146	135.27	0.165	1.0060 ± 0.0013

*Monte Carlo statistical error at one standard deviation.

Table 5-5

BUNDLE DESIGN VERSUS MONTE CARLO TH-232 REACTION
RATES FOR THE B&W TUPE 15B CORE

	<u>Neutron Absorption Fraction</u>		$\left(\frac{BD - MC}{0.2189} \times 100 \right)$
	<u>Monte Carlo (MC)</u>	<u>Bundle Design (BD)</u>	
TH-232:			
GP1	0.03142	0.03241	0.45%
GP2	0.09105	0.09596	2.24%
GP3 (thermal)	0.09639	0.09565	0.34%
TOTAL	0.2189	0.2240	2.35%

U-235:

TOTAL	0.7247	0.7196
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	<u>Neutron Production Fraction</u>		<u>(MC-BD) x 100</u>
	<u>Monte Carlo (MC)</u>	<u>Bundle Design (BD)</u>	
TH-232:	0.009	0.010	0.2%
U-235:	0.991	0.989	-0.2%

Table 5-6

BUNDLE DESIGN VERSUS MONTE CARLO ABSORPTION RATES
FOR PBB DESIGN WITH 1% U-233 IN ThO₂ CORNER RODS

Neutron Absorption Fraction			$\left(\frac{\text{BD-MC}}{0.02595} \times 100\right)$
Monte Carlo (MC)	Bundle Design (BD)		
U-233:			
GP1	0.00029	0.00029	0.00%
GP2	0.00485	0.00452	-1.27%
GP3	0.02081	0.02138	+2.19%
TOTAL	0.02595	0.02619	+0.92%
			$\left(\frac{\text{BD-MC}}{0.03857} \times 100\right)$
Th-232:			
GP1	0.00301	0.00301	0.00%
GP2	0.00969	0.01008	+1.01%
GP3	0.02587	0.02660	+1.89%
TOTAL	0.03857	0.03969	+2.90%

NUMBER PINS: 1108
 CORE RADIUS = 18.37 cm
 CRITICAL WATER HEIGHT = 127.21 cm

NOTE:

PINS ARE TO BE ASSUMED UNIFORMLY
 LOCATED AT ALL POSITIONS INSIDE
 RING SHOWN

Figure 5-1. Babcock & Wilcox Core 15A

NUMBER PINS: 880
CORE RADIUS = 17.12 cm
CRITICAL WATER HEIGHT = 133.44 cm

NOTE:

PINS ARE TO BE ASSUMED UNIFORMLY
LOCATED AT ALL POSITIONS INSIDE
RING SHOWN

Figure 5-2. Babcock & Wilcox Core 15B

Figure 5-3. Core 15B Critical Configuration

100,000 NEUTRON HISTORIES

KEY

MONTE CARLO BUNDLE DESIGN $\Delta\%$
--

W-W

ThO ₂ 0.780 0.783 +0.38	ThO ₂ 0.657 0.652	1.210 1.224	1.203 1.195	1.185 1.210	1.109 1.149	1.218 1.249	ThO ₂ 0.653 0.644
ThO ₂ 0.657 0.652 -0.76	1.210 1.231 +1.74	1.158 1.144	0.309 0.310	1.102 1.104	1.061 1.091	1.115 1.117	1.108 1.105
1.210 1.224 +1.16	1.158 1.144 -1.21	1.064 1.050 -1.32	1.036 1.006	1.069 1.072	1.025 1.021	0.310 0.307	1.103 1.096
1.203 1.195 -0.67	0.309 0.310 +0.32	1.036 1.006 -2.90	1.121 1.084 -3.30	0 0	1.024 1.027	1.013 1.006	1.075 1.076
1.185 1.210 +2.11	1.102 1.104 +0.18	1.069 1.072 +0.28	0 0	1.094 1.076 -1.65	0.985	1.035 1.018	1.135 1.152
1.109 1.149 +3.61	1.061 1.091 +2.83	1.025 1.021 -0.39	1.024 1.027 +0.29	0.985 0.987 +0.20	0.955 0.956 +0.10	0.970 0.977	1.172 1.153
1.218 1.249 +2.55	1.115 1.117 +0.18	0.310 0.307 -0.97	1.013 1.006 -0.69	1.035 1.018 -1.64	0.970 0.977 +0.72	0.309 0.308 -0.32	1.117 1.099
ThO ₂ 0.653 0.644 -1.38	1.108 1.105 -0.27	1.103 1.096 -0.63	1.075 1.076 +0.09	1.135 1.152 +1.50	1.172 1.153 -1.62	1.117 1.099 -1.61	1.150 1.126 -2.09

MONTE CARLO	$K_{\infty} = 1.0654 \pm 0.0014$
BUNDLE DESIGN	$K_{\infty} = 1.0685$
	$\Delta\% = +0.29$

Figure 5-4. Local Fission Densities 1% U-233 in ThO₂ Corner Rods

6. ANALYSIS OF ONCE-THROUGH THORIUM CYCLE PERFORMANCE

6.1 DESCRIPTION

The once-through thorium fuel cycle concept was evaluated for use in the Browns Ferry-3 reactor. The PBB design is based on placement of ThO_2 -bearing fuel pins in high thermal neutron flux regions of the BWR fuel bundle assembly to convert Th-232 to U-233. Because of the initial reactivity of the ThO_2 rods, the burnable absorber requirements of the fuel bundle may be reduced. Following discharge of the ThO_2 -bearing prebreeder bundle, the thorium fuel pins are removed and used to make up breeder blanket bundles that are positioned at the core periphery. The BBB reactivity is relatively flat with burnup, and therefore the BBB has a long nuclear lifetime. However, the BBB should be more reactive than last cycle fuel normally placed on the core periphery, to reduce the energy requirements of the remainder of the core, the equilibrium reload fuel batch size, and the feed uranium requirements.

The following subsections describe the PBB fuel design, resultant BBB fuel design, and the associated once-through thorium fuel cycle analyses. The PBB lattice design basis was to match approximately the energy of a standard reference fuel bundle design while meeting design and operating bases. Initially, in order to evaluate the once-through concept with the least amount of sophistication, the 2.84U2 axially uniform gadolinia bundle was selected for comparison, utilizing a single stream loading pattern.

6.2 PREBREEDER BUNDLE DESIGN AND CYCLE OPERATION

6.2.1 General Description

A fuel bundle assembly was analyzed which utilized pure ThO_2 rods placed in high thermal flux regions to convert fertile Th-232 to fissile U-233. The irradiated ThO_2 rods were reconstituted into assemblies intended to replace last-cycle peripheral fuel. The PBB utilized identical hardware and had an energy output similar to the 2.84U2 reference reload UO_2 fuel bundle design while using approximately the same U-235 bundle average enrichment. It had

greater local power peaking than the reference UO_2 bundle, but this was mitigated by the PBB's flatter slope of reactivity with burnup, and smaller steam void reactivity coefficient which yielded less bundle power mismatch and a flatter axial power shape.

6.2.2 Lattice Design

The once-through thorium PBB fuel bundle assembly shown in Figure 6-1 is identical in all respects to the reference UO_2 fuel assembly except for the composition of the fuel rods. The PBB employs two water rods, as does the reference UO_2 design, but has five pure ThO_2 rods and less burnable poison than the reference UO_2 bundle. The pure ThO_2 rods contain 9.6 g/cm^3 , 96 percent theoretical density thorium-dioxide. A comparison of the required pounds of feed U_3O_8 , separative work utilization (SWU), and average enrichment is given in Table 6-1.

6.2.3 BWR Operating Characteristics

Utilization of the PBB in BWRs is not expected to degrade thermal margins but may slightly reduce cold shutdown margin. These items are discussed more fully below.

6.2.3.1 Hot Excess Reactivity

An objective of the PBB design was to match as closely as possible the amount of excess reactivity throughout the equilibrium cycle with the reference UO_2 bundle while using about the same bundle average enrichment, thereby achieving approximate energy and uranium usage equivalence. For a PBB with less initial fissile inventory, Figure 6-2 gives the reactivity shape as a function of burnup compared to the reference UO_2 bundle. The hot excess reactivity of both designs, determined by identical equilibrium core loadings of the two designs, is given as a function of burnup in Figure 6-3. The hot excess reactivity was determined with all the control rods withdrawn fully from the core.

The two plots have similar shapes with the thorium-fueled core possessing slightly less excess reactivity than the reference UO_2 -fueled core. Therefore, the thorium-fueled core produced less energy during its equilibrium cycle operation. Uranium utilization of the PBB-fuel equilibrium cycle is compared to that of the UO_2 -fueled cycle in Subsection 6.2.4.

6.2.3.2 Cold Shutdown Margin

Cold shutdown margin, given for both the thorium-fueled and reference UO_2 -fueled cores as a function of burnup in Figure 6-4, must be 1.0 percent ΔK or greater to meet the design requirement. The decreased cold shutdown margin of the PBB cycle compared to the UO_2 cycle is a result of the smaller control rod worth associated with the PBB.

The control blade negative reactivity worth decreased by 1-2 percent at the beginning of the equilibrium cycle (BOEC) because five of the pins closest to the control rod are pure ThO_2 pins which operate at 10 percent of the bundle average power. Decreasing the power of these pins by control rod insertion does not decrease the total reactivity of the bundle as much as it decreases the reactivity of the reference UO_2 bundle. The UO_2 bundle pins that are closest to the control rod operate above the bundle average power and contribute a greater fraction of reactivity to the bundle than the ThO_2 pins in the PBB. As U-233 builds up in the U-233 pins, this effect should reverse, with the cold control blade worth being greater late in PBB life.

6.2.3.3 Local Thermal Margins

Local power peaking of the thorium PBB and reference UO_2 designs as a function of burnup is shown in Figure 6-5. Higher PBB local power peaking than the reference UO_2 bundle was observed at the beginning of the cycle, but comparable peaking was observed at peak bundle reactivity.

The local peaking of the PBB increases at high burnup, caused by U-233 buildup in the ThO_2 rods and the ThO_2 rods becoming the highest power rods in the fuel

assembly. PBB peaking at high burnup presented no problem because the bundle is operating at low power compared to the core remainder.

6.2.4 Cycle Energy and Core Thermal Margins

Equilibrium cycle energy and core radial and axial power shapes were determined for the reference loading pattern of the PBB-fueled Browns Ferry-3 reactor core using a three-dimensional coupled nuclear-thermal-hydraulic core simulator. Several important performance parameters for both the PBB and reference UO_2 equilibrium cycle are compared in Table 6-2. Evaluation of these parameters was based on end-of-equilibrium cycle (EOEC) consistent power-exposure (or Haling) shape, which is shown in Figure 6-6 for both fuel loadings.*

The PBB equilibrium fuel cycle reduced the cycle energy by 2.4 percent compared to the reference UO_2 cycle energy while using approximately the same amount of feed U_3O_8 and 3.0 percent more SWU per bundle. Therefore, use of the PBB in the Browns Ferry-3 reactor, without consideration of the BBB reconstitution and reinsertion on the core periphery, reduced the overall uranium utilization by 2.4 percent compared to the reference UO_2 cycle. To compete economically with the reference UO_2 -fueled reactor, use of the BBB along with the PBB must increase the equilibrium cycle energy by at least 5.4 percent.

The Haling EOEC axial and radial power shapes are comparable to that seen for the reference UO_2 reactor. This indicates that an operational strategy that meets BWR operating constraints could be developed because the UO_2 -fueled reactor already meets these operating limits.

It probably would have been desirable to either reoptimize the enrichment distribution or increase the fissile loading of the PBB design to match the energy capability of the UO_2 reference design. Increasing the fissile inventory would have penalized SWUs but would reduce uranium, fuel assembly, and spent fuel requirements.

*See Appendix for Haling discussion.

6.3 BREEDER BLANKET BUNDLE (BBB) DESIGN

6.3.1 General Description

Several breeder blanket reconstituted fuel bundles were evaluated which consisted of irradiated ThO_2 fuel pins which were removed from the PBBs after discharge (25-30 GWd/MT PBB bundle average exposure). The thorium pins were composed of Th-232, fissile U-233 produced from neutron capture in Th-232 during PBB irradiation, and fission products from burnup of Th-232, U-233, and fissionable capture products. If the BBB reactivity (and power) is higher than last-cycle fuel normally positioned on the core periphery, then replacement of the last-cycle fuel with BBBs would decrease the energy requirements of the remainder of the core, thereby reducing the reload fuel batch size and increasing uranium utilization. Because the BBBs stay at the core periphery, the core becomes effectively smaller, and less reload fuel is required.

6.3.2 Lattice Design

The $(\text{U/Th})\text{O}_2$ rods' composition, used to make up the BBBs, was specified by the discharge isotopics of the PBB. The designs evaluated contained 8x8, 9x9, and 10x10 fuel pin arrays with from two to six water rods. The rods were placed inside a standard size BWR fuel channel. Therefore, increasing the number of rods placed inside the channel decreased the M/F ratio, hardened its neutron spectrum, and increased the conversion ratio. The increased fissile content in the 9x9 and 10x10 arrays resulted in a flatter slope of reactivity with burnup. But the smaller M/F ratio decreased the bundles' slowing down power and gave bundle powers lower than that seen for 8x8 BBBs. Therefore, the 8x8 BBB design was used for the three-dimensional evaluation.

6.3.3 BWR Operating Characteristics and Cycle Energy

Peripheral bundles normally operate at about half the core average power. Therefore thermal performance does not normally limit operation. The only important parameter is bundle reactivity as a function of burnup, which

determines its power level relative to the remainder of the core. Higher BBB reactivity, compared to last-cycle PBBs, would reduce the energy requirements of the remainder of the core, thereby increasing uranium utilization. Conversely, lower BBB reactivity would decrease uranium utilization.

Reactivity is given as a function of burnup, for the core average void fraction (40% void), in Figure 2-1 for several BBB designs. All the BBB designs possess less reactivity than last-cycle PBB and reference UO_2 fuel throughout the range of the PBB/reference bundle last-cycle burnup (~ 22 - 28 GWd/MT). Therefore, the BBBs positioned on the core periphery operated at lower powers than the fuel they replaced. This required more energy production from the remainder of the core (the PBBs), which reduced calculated cycle energy by 2.5 percent compared to the all-PBB fuel loading and 4.9 percent compared to the reference UO_2 fuel loading. To achieve the same cycle energy as either the PBB only or reference UO_2 loading, the batch fraction of the reload fuel or enrichment would have to increase, which would require greater amounts of feed U_3O_8 , decrease uranium utilization, and increase SWU costs.

No rod patterns were developed because the 2.84U2 UO_2 fuel cycle energy was not matched by the PBB or PBB/BBB fuel cycle. It is possible that performance could have been improved by optimization of the refueling patterns. A comprehensive optimization was not completed during the evaluation.

6.4 HIGH BURNUP PBB/BBB FUEL CYCLES

Deployment of BBBs reconstituted from 2.84 PBBs decreased the utilization of uranium in Browns Ferry-3 because the BBBs replaced last-cycle PBBs which were of higher reactivity than the BBBs. If the last-cycle PBBs were operated to higher burnups, at some point the BBB reactivity and power would become greater than that of the PBBs they replace, thereby increasing cycle uranium utilization. Also, because the PBB reactivity decreases at a slower rate than UO_2 reactivity, once the first batch of fuel has passed its peak reactivity, higher burnups would imply greater uranium utilization.

In order to investigate the effect of high burnup on the PBB and PBB/BBB fuel cycle performance, a higher burnup PBB was analyzed to approximately match the 3.15A1 fuel bundle energy. The main thrust of these analyses was to increase fuel bundle discharge burnup. Thermal performance was not considered.

6.4.1 High Burnup PBB Design

A high burnup PBB was analyzed which approximately matched the 3.15A1 high burnup UO_2 bundle. The arrangement of fuel pins was identical to the basic 2.84 PBB lattice except that all the fuel pins were proportionately increased in enrichment until the PBB approximately matched the 3.15A1 bundle average enrichment. Figure 6-8 shows the high burnup PBB and 3.15A1 bundle reactivity as a function of burnup at core average voids (40% void).

6.4.2 PBB/BBB Fuel Cycle Performance

Table 6-3 compares PBB, PBB/BBB, and 3.15A1 cycle energies.

The equilibrium cycle energy was determined for both the PBB and 3.15A1 fuel bundles for the 3.15A1 loading pattern which utilized a 54-bundle reload batch. Then with all peripheral bundles replaced by BBBs, the analysis was repeated. Increasing the PBB enrichment and burnup reduced the PBB uranium utilization shortfall from -2.24 to -1.86 percent and reduced the PBB/BBB shortfall from -4.9 to -2.6 percent. Thus, increasing burnup further would continue to decrease the cycle energy shortfall.

6.4.3 Spectral Shift, Feedwater Temperature Reduction, and End-of-Cycle Coastdown for the PBB/BBB Fuel Cycle

Increased discharge burnup would occur for combined alternative analyses where spectral shift operation was followed by feedwater temperature reduction, followed by end-of-cycle coastdown. Such an analysis¹³ was made with the reload batch fraction being reduced to 46 bundles to maintain a consistent cycle energy.

Table 6-4 compares PBB, PBB/BBB, and 3.15A1 cycle energies. The PBB uranium utilization shortfall increased from -1.86 to -1.91 percent, while the PBB/BBB shortfall decreased from -2.6 to -2.05 percent. Further increases in burnup would turn the cycle energy shortfall into a gain.

Table 6-1
REFERENCE UO_2 AND PBB ASSEMBLY CHARACTERISTICS

	<u>Reference UO_2 Bundle</u>	<u>PBB</u>	<u>% Advantage</u>
Bundle Average Enrichment (wt % U-235) [over 62 rods]	2.84	2.84	----
Feed U_3O_8 (lbs)	2486	2485	+0.04%
SWU Requirements*	737.9	760.4	-3.0%

*SWU requirements are larger for PBB because of higher enrichment of UO_2 rods.

Table 6-2
REFERENCE UO_2 AND PBB EQUILIBRIUM CYCLE PARAMETERS

	<u>Reference UO_2</u>	<u>PBB</u>	<u>% Advantage</u>
Cycle Burnup, MWD/MT Heavy Metal	9573	9376	-----
Cycle Energy, GWd	1331	1298	-2.4
Axial Peak/Average Power*	1.178	1.187	-0.76
Radial Peak/Average Power*	1.317	1.310	
Hot Beginning of Cycle (BOC) Excess Reactivity, % ΔK	1.8	1.7	-----
Minimum Cold Shutdown Margin, % ΔK	2.5	2.0	

*Haling, consistent power-exposure, end-of-cycle (EOC)

Table 6-3

54-BUNDLE, HIGH ENRICHMENT LOADING CYCLE ENERGY

<u>Fuel Loading Case</u>	<u>Cycle Energy E (Gwd)</u>	<u>ΔE (%)</u>
Reference 3.15A1	1331	0.0
High Burnup PBB	1307	-1.8
High Burnup PBB/BBB	1296	-2.6

Table 6-4

CYCLE ENERGY FOR 46-BUNDLE, HIGH ENRICHMENT LOADING:
SPECTRAL SHIFT, FEEDWATER TEMPERATURE REDUCTION, AND
END-OF-CYCLE COASTDOWN

<u>Fuel Loading Case</u>	<u>Cycle Energy E (Gwd)</u>	<u>ΔE (%)</u>
Reference 3.15A1	1323.2	0.0
High Burnup PBB	1297.9	-1.91
High Burnup PBB/BBB	1296.1	-2.05

Figure 6-1. PBB Fuel Bundle Assembly Design

Figure 6-2. K-Infinity vs. Burnup 2.84U2 and PBB at Core Average Void Fraction

Figure 6-3. Hot Excess Reactivity vs. Burnup

Figure 6-4. Cold Shutdown Margin vs. Burnup

Figure 6-5. Local Power Peaking vs. Burnup

Figure 6-6. Power and Burnup Shape, EOEC (Haling)

8T-17/6-18

K-INFINITY UNCONTROLLED

GEAP-25506

Figure 6-7. K-Infinity vs. Burnup-High Enrichment PBB vs. 3.15A1

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Appendix A

DEFINITIONS

Throughout this report and other Improved Uranium Utilization Program (IUU) semiannual reports, as well as monthly progress and topical reports, several terms or phrases are used which may not be readily understood by a reader not closely associated with this IUU Program. This appendix, which appears in all IUU Program semiannual and topical reports, attempts to more fully explain these terms and their relationships to the analyses, evaluations, and studies reported herein.

A.1 THERMAL MARGINS

Practical application of uranium utilization alternatives requires equal, if not more, detailed consideration of thermal margins, licensing issues, and reliability.

Technical specifications limits (also referred to as "operating limits") define the licensed range of reactor operation. These technical specifications limits are Nuclear Regulatory Commission (NRC) requirements which must be observed or revised on the basis of justifiable safety considerations. Two important limits are the maximum linear heat generation rate (MLHGR) and minimum critical power ratio (MCPR) limits. The MLHGR limits the peak fuel temperature of a fuel pellet, while the MCPR effectively prevents operation in transition boiling.

To ensure that the reactor is not operated in violation of the technical specifications or operating limits, data from the in-core neutron detectors and other pertinent instrumentation are read into the on-site process computer, which in turn calculates the MLHGR and MCPR values for that operating state. These values are then compared to the respective operating limits to determine if the operator must make adjustments to the operating state. The difference between the process computer-derived MLHGR and MCPR and that specified by the corresponding operating limit is the "thermal margin" for that operating state.

During the nuclear design process, the term "thermal margin" has a slightly different connotation. Because projected operation (by the nuclear designer) and actual operation differ somewhat, it has been the practice to evaluate margin at the design stage. The nuclear design thermal margin is the difference between the predictions by the 3-D simulator and the MLHGR or MCPR technical specification limit.

A.2 HALING AND DETAILED CYCLE SIMULATION

The primary computation tool used in the analyses appearing in this report is a three-dimensional coupled nuclear thermal hydraulic BWR core simulator code used for calculation of core steady-state phenomena. Several different kinds of calculations are possible, all using the same basic calculational model.

Prior to the start of the 3-D core design calculations, the nuclear and thermal hydraulic characteristics of all the fuel bundles in the core are input to the simulator code. The first stage in the evolution of a core design, then, consists of arranging the fuel bundles in a particular pattern in the core. An approximate measure of the effectiveness of this pattern in yielding a desired cycle exposure and a desired degree of thermal margin throughout the cycle is provided by executing a Haling calculation for this particular fuel bundle pattern.

The Haling distribution is the end-of-cycle (EOC) unique spatial distribution wherein the power and incremental burnup shapes are identical. This distribution is computed by a set of power-exposure direct iterations. Operation of the reactor to the Haling distribution usually will maximize worst-case thermal margins within the cycle. The Haling calculation is typically used as the basis for approximating the cycle energy, and the Haling power distribution is used as the target power distribution for the later selection of detailed control rod patterns throughout the cycle. However, because particular features of the design may make it impossible to operate the reactor to the Haling EOC distribution, and because core reactivity can be sensitive to details of the local history as well as to the final EOC integral, the Haling

method only approximates cycle energy. Detailed cycle simulations including control rod effects are a more reliable determinant of cycle energy but are sensitive to the degree to which the control rod patterns have been optimized for cases being compared.

The Haling distribution is sensitive to the way in which the core is operated and therefore is primarily sensitive to refueling patterns. Using the same fuel bundles, different patterns of these bundles will result in different Haling cycle exposures and different Haling power distributions. Increasing the fraction of the core being reloaded with fresh high reactivity fuel bundles will usually result in longer Haling cycle lengths, depending on gadolinia considerations.

During iteration of refueling patterns, the core design is checked for compliance with limits set for the minimum cold shutdown margin and the minimum hot excess reactivity. If compliance is not met, then a new mixture of fuel bundle types, or a new fuel bundle pattern, or a combination of both is required, and the procedure of core design to this point must be repeated.

All the refueling pattern requirements having been satisfied, the next stage in the evaluation is the development of a set of critical control rod patterns throughout the entire cycle. This is referred to as a "detailed cycle simulation." Here, the entire operating cycle is subdivided into smaller burnup intervals, typically 1100 MWd/MT in duration. For each of these intervals, a critical control rod pattern is iteratively selected which results in a 3-D power distribution with adequate margin to thermal limits. At the same time an attempt is made to have the core average power distribution approach, as closely as possible, the previously calculated target Haling power distribution.

Once again, as in the design stage involving compliance with minimum cold shutdown margin and hot excess reactivity requirements, the failure of any of the many individual critical control rod patterns throughout the entire cycle to yield acceptable thermal margins results in a return to the design stage involving the selection of fuel bundles and the selection of a fuel bundle pattern in the core.