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FUEL UTILIZATION IMPROVEMENT IN PWRs
USING THE DENATURED ^{233}U -Th CYCLE

— Final Report —

Babcock & Wilcox

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Printed in the United States of America
Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

NTIS price codes
Printed copy: \$5.25
Microfiche copy: A04

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H. M. Jones
G. A. Schwenk
E. C. Toops
V. O. Uotinen

June 1980

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Prepared for

Oak Ridge National Laboratory
Operated by Union Carbide for
the U.S. Department of Energy

Contract No. W-7405-eng-26

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Report BAW-1608

March 1980

Fuel Utilization Improvement in PWRs Using the
Denatured ^{233}U -Th Cycle - Final Report

H. M. Jones, G. A. Schwenk, E. C. Toops, V. O. Uotinen

Key Words: Fuel Utilization, Thorium, Thoria, Pressurized
Water Reactors, Denatured Uranium-233, Thorium
Fuel Cycle, Thorium Blanket

ABSTRACT

A number of changes in PWR core design and/or operating strategy were evaluated to assess the fuel utilization improvement achievable by their implementation in a PWR using thorium-based fuel and operating in a recycle mode. The reference PWR for this study was identical to the B&W Standard Plant except that the fuel pellets were of denatured ($^{233}\text{U}/^{238}\text{U}$ -Th) O_2 .

An initial scoping study identified the three most promising improvement concepts as (1) a very tight lattice, (2) thorium blankets, and (3) ThO_2 rods placed in available guide tubes. A conceptual core design incorporating these changes was then developed, and the fuel utilization of this modified design was compared with that of the reference case. Calculations show that the maximum fissile mass savings achievable with the modified design amounts to 64% of the fissile mass utilized in the reference case. About 39% of the total savings results from the tight lattice, 41% results from the radial blanket, 8% results from the axial blankets, and 12% is contributed by the ThO_2 rods in the guide tubes.

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1. INTRODUCTION AND BACKGROUND

During the past few years there has been renewed interest in the thorium fuel cycle for pressurized water reactors (PWRs). The primary objective from the standpoint of the United States government has been to assess the potential for reducing the risks of nuclear weapons proliferation with the use of the "denatured" uranium/thorium fuel cycle.¹ In this concept the fissile isotope (^{233}U) is mixed with ^{238}U , which serves as a denaturant, rendering the uranium unusable directly for weapons fabrication.

As an important part of the current evaluation of alternative fuel cycles, the United States Department of Energy (DOE) has undertaken the Nonproliferation Alternative Systems Assessment Program (NASAP). Under this program, fuel cycle evaluations have been carried out to determine the potential benefits of using thorium-based fuels in PWRs.¹ These evaluations have indicated that the use of thorium could indeed result in considerable savings in U_3O_8 requirements if recycle of the bred ^{233}U were employed.

A concurrent major DOE study has involved the assessment of various design changes to improve fuel utilization in standard PWRs fueled with low-enriched uranium (LEU) and operated on the once-through fuel cycle.² These studies have indicated that certain changes in design or operating strategy can indeed result in sizable improvements in fuel utilization in the once-through cycle.

The study described in this report was undertaken by Babcock & Wilcox, under DOE sponsorship, to supplement the studies mentioned above, i.e., the studies of thorium utilization in existing PWRs and fuel utilization improvements in PWRs fueled with LEU. The specific mission of this study was to investigate how much the overall fuel utilization could be improved in a thorium-fueled PWR if appropriate changes were implemented in design and/or operating strategy. The reference design was a B&W standard PWR fueled with denatured ($^{233}\text{U}/^{238}\text{U}\text{-Th}$) O_2 , hereafter referred to as denatured ^{233}U , operating in a recycle mode.³

The study was laid out as a two-phased effort. The first phase was a scoping assessment of the fuel savings potential of several suggested concepts. The concepts were ranked and those that were most promising were identified. This in turn led into a second phase, which consisted of determination of the fissile fuel resource savings that could be achieved by a reactor design that incorporated the more promising improvements. Important ground rules to be observed in this assessment were that the design should retain a high degree of retrofitability, and core thermal and safety margins should not be adversely affected.

Thus, the overall goals of this study were (1) to provide first estimates of the fuel utilization improvement achievable by implementing each of the selected design changes and (2) to determine the potential fuel utilization improvement achievable through the use of a thorium-based reactor design incorporating a combination of the more promising design changes.

An overall summary of the study is presented in section 2, the Phase 1 effort is described in section 3, and the Phase 2 work is described in section 4.

2. SUMMARY AND CONCLUSIONS

A number of changes in reactor design and/or operating strategy have been identified under DOE's Nonproliferation Alternative Systems Assessment Program (NASAP) as having potential to improve fuel utilization in PWRs operating on the once-through, low-enriched UO_2 cycle. In the NASAP study reported herein, six retrofitable improvements have been evaluated for a Babcock & Wilcox standard PWR fueled with denatured ^{233}U with provision for ^{233}U recycle. An important aspect of the study was the use of a set of ground rules that would result in minimal plant design changes and the least possible sacrifices in plant safety margins. In general, the intents of all the ground rules were met, and those related to hardware and plant performance were addressed directly in the design development.

In the initial phase of this program a scoping study was conducted to estimate the potential for improved fuel utilization offered by the suggested concepts. It was concluded that those once-through improvements that do not increase the conversion ratio but obtain their benefit from reducing the residual fissile content of the fuel will not appreciably reduce fissile make-up requirements in the recycle mode. In keeping with this principle, certain concepts were eliminated from further study, and the results of the scoping study were used to identify the concepts showing the greatest potential. The concepts promising the largest net fuel utilization improvements for a PWR using denatured ^{233}U fuel were (1) a very tight lattice, (2) thorium blankets, and (3) ThO_2 rods in available guide tubes.

In a subsequent phase of the program, a conceptual reactor core design was developed, which incorporated all three of the more promising design changes. This core was designed to be retrofitable into the B&W Standard Plant. The core comprised 161 fuel assemblies surrounded by 44 radial blanket assemblies, and also included axial blankets at the top and bottom of the core. The outside envelope of both fuel and blanket assemblies was kept identical to that

of a standard B&W 17×17 fuel assembly. Thus, the standard control rod pattern could be used in the modified core.

Calculations indicate that this conceptual reactor design, incorporating the more promising fuel utilization improvement features, has the potential for reducing fissile fuel consumption significantly. The magnitude of the fissile mass savings, however, depends on the assumptions regarding the use of the fissile plutonium that is bred in the $\text{ThO}_2\text{-UO}_2$ core. In NASAP this plutonium is assumed to be used inside secure energy centers which contain plutonium-fueled reactors that produce ^{233}U for use in denatured ^{233}U -fueled reactors located at dispersed sites.

The largest savings attainable with the selected combination of design changes amounts to about 64% of the fissile mass destroyed in the reference case. (If no credit is taken for the plutonium, the net savings would be 23%.) The reference case is the B&W standard 205-fuel assembly PWR fueled with denatured ^{233}U .

For the optimum case the various design changes contribute to the total savings in the following proportions: about 39% of the total fissile mass savings results from the change to a very tight lattice, about 41% results from the radial blanket, 8% from axial blankets, and 12% from the ThO_2 rods in the guide tubes.

On a unit mass basis, the use of ThO_2 guide tube rods is the most efficient way to generate ^{233}U . The net production of ^{233}U in the guide tube rods per unit mass of ThO_2 is about two times that of the radial blanket and three times that of the axial blanket.

Overall, it is concluded that a substantial fuel utilization gain could be realized by employing the design changes examined in this study even if plutonium use is neglected. However, the principal contribution to the net fissile mass savings clearly comes from fissile plutonium buildup. Thus, for the full benefit to be realized from the design improvements reported herein, the overall reprocessing/recycle strategy should include utilization of the plutonium generated in the denatured $\text{UO}_2\text{-ThO}_2$ fuel. In light of current nuclear proliferation concerns, such plutonium utilization would have to take place within the confines of secure fuel cycle centers, which are needed in any event for the reprocessing and denaturing operations.

3. PHASE 1 — SCOPING STUDY TO IDENTIFY MOST PROMISING FUEL
UTILIZATION IMPROVEMENT CONCEPTS FOR DENATURED
 ^{233}U - ThO_2 -FUELED PWRs

3.1. Introduction

The initial phase of this program involved a quantitative assessment of six selected design improvement concepts with the potential for improving recycle fuel utilization and a ranking of their estimated potential for reducing fissile makeup requirements in a PWR fueled with denatured ^{233}U and operating in a recycle mode. The base case against which fuel utilization improvement options were compared was a B&W Standard Plant which utilized the B&W Mark C (17 × 17) fuel assembly design containing pellets of the denatured ^{233}U fuel. Safety and licensing aspects of this base case have been evaluated in a related study conducted by B&W and reported in reference 4. Key reactor design parameters for the B&W Standard Plant are listed in Table 3-1. The following specific options were selected for this initial scoping evaluation:

1. Lattice modifications.
2. Thorium blanket utilization.
3. Fertile (ThO_2) rods in place of conventional lumped burnable poison rods.
4. Removal of partial-length control rods at the end of life.
5. Moderator temperature control during the reactor operating cycle to achieve higher burnups.
6. Zone-loaded assemblies to allow "in-out" fuel management without the use of burnable poison rods.

3.2. Results of Scoping Assessment

3.2.1. Conclusions and Recommendations

The qualitative assessment of the six improvement options resulted in grouping them into two categories. The concepts that show the greatest potential improvements in fuel utilization for the recycle case under considerations are (1) lattice modification — specifically, use of a much tighter lattice than that of the reference case; (2) use of ThO_2 blankets around the core;

and (3) use of ThO_2 rods in available control rod guide tubes. These three design changes were recommended for further, more detailed studies in Phase 2.

The other three suggested modifications do not appreciably increase the conversion ratio, and their major benefit arises from reducing the residual fissile content of the fuel. However, simply reducing the residual fissile content will not appreciably reduce fissile makeup requirements in the recycle mode since the fissile content of spent fuel can be recovered through reprocessing. Thus, these suggested improvement concepts were not recommended for further study in the subsequent phase of the program. The preliminary screening assessment of these fuel utilization improvement concepts is discussed below.

3.2.2. Lattice Modifications

It is well known that larger conversion ratios can be achieved in lattices having larger metal-to-water volume ratios. The harder neutron spectrum that is characteristic of the tighter lattices enhances neutron absorption in fertile material relative to parasitic thermal absorptions. Thus, more fissile material is generated during reactor operation in a tight lattice than in a conventional PWR lattice. However, the tighter lattices also typically require larger initial enrichments and have larger discharge enrichments as well. Thus, the tighter lattices are not attractive for a once-through mode of fuel cycle but require reprocessing and recycle in order to recover the residual end-of-cycle fissile material.

The NULIF computer program, which is B&W's standard lattice cell code, was used in a preliminary assessment of the benefits of tighter lattices.⁵ To effect a metal-to-water increase in this evaluation, the fuel pin pitch was decreased. The range of interest extended from the reference pitch (which corresponds to a metal-to-water ratio of about 0.8) to much tighter pitches, approaching a M/W ratio of 2.0. The B&W Mark C fuel assembly lattice was the reference design.⁶ Design parameters that characterize the reference fuel cell are given in Table 3-1. Fuel cell depletions were performed to 33,000 MWD/THM, and soluble boron was cycled between 1000 and 17 ppm every 11,000 MWD/THM fuel exposure. The fuel cell neutron spectrum was modified with a NULIF option to account for the presence of water in the fuel assembly guide tubes, instrument tube, and interassembly water gap. Calendar time was incorporated into the fuel depletion time dictionary to accommodate decay of

protactinium, neptunium, and plutonium isotopes and the decay and buildup of xenon.

The results of the preliminary calculations indicated significant resource savings potential in the form of increased plutonium production as the lattice metal-to-water ratio is increased. At a metal-to-water ratio of 2.0, there is a reduction of about 25% in annual fissile makeup requirements relative to that of the reference lattice. The results of this assessment, namely, net fuel consumption and integrated conversion ratio as a function of fuel lattice metal-to-water ratio (Figure 3-1) were similar to results obtained by Argonne National Laboratory (ANL).⁷ Data points at two metal-to-water ratios from the ANL study are plotted on the same figure with the net fuel consumption data normalized to the B&W results at a M/W ratio of 1.20.

These results clearly indicated that resource savings due to lattice modifications are substantial. Thus, this design change was recommended for inclusion in the Phase 2 studies.

3.2.3. Thorium Blankets

External fertile material blankets are another promising design change for the recycle mode. Fertile blankets conserve fuel resources by reducing the neutron losses from leakage and by changing the average neutron energy to enhance absorption in fertile material relative to parasitic absorption in structural and coolant materials. The use of blankets also increases the overall conversion ratio.

A preliminary estimate of the benefits to be expected from the use of thorium blankets was performed on the basis of 1960 B&W studies⁸, which used highly-enriched ($^{235}\text{U}/\text{Th}$) O_2 fuel with recycle of ^{233}U . Extrapolation of the results of the early studies to the current B&W Standard Plant design led to an estimated 4% reduction in annual fissile material requirements by using a radial blanket around the core of a current B&W reactor. This significant resource savings potential, which results from enhancement of the conversion ratio, indicated that the blanket option should be included among the concepts to be evaluated more fully in Phase 2.

3.2.4. Thorium-Oxide Rods to Replace Burnable Poison Rods

In principle, the substitution of fertile material (thorium) for the parasitic absorber material (boron) will increase the overall conversion ratio and improve the overall utilization of fissile material.

It is standard practice in the industry to use lumped burnable poison (LBP) in PWR first cycles to achieve a more negative moderator temperature coefficient early in the cycle. The use of LBP reduces the soluble poison concentration requirements for lifetime reactivity control for a given core fuel loading. In addition, the manner of poison loading (inserting LBP in fuel assembly control rod guide tubes in those core locations that are not under control rods) provides a means for controlling both local and global power distribution. B&W has used this power-shaping capability of LBP in the development of a fuel management scheme (the LBP shuffle scheme) which uses LBP in all cycles to control power distribution in a manner that significantly reduces core leakage.⁹

To help in assessing the usefulness of substituting ThO_2 rods for boron-containing LBP rods, the design data for the B&W LBP shuffle scheme were reviewed to determine whether the core physics effects of the burnable poison could be approximated with thorium. Typically, about one-third of the LBP assemblies have a relatively low boron concentration. The LBP rods in these assemblies have a relatively low macroscopic thermal absorption cross section, which compares favorably with the corresponding cross section for a ThO_2 rod. On this basis, it was concluded that ThO_2 rods could serve the function of lightly loaded LBP assemblies in the core design and be compatible with the remaining two-thirds LBP complement containing boron.

In a core fueled with thorium-based fuel, thorium rods substituted for LBP will produce ^{233}U at about the same rate as the surrounding fuel rods. About 1% of the fertile material will be converted per cycle, so the fertile absorption rate will be nearly constant over the assembly lifetime. The added thorium, the number of cycles these rods remain in the core, and the absorption rate in thorium determine the mass of additional fissile material produced and the reduction in fissile makeup requirements. Based on macroscopic absorption cross section comparison between the fertile and fuel rods, it was estimated that the use of thorium rods for LBP duty in the B&W Standard Plant

LBP design could produce an annual reduction of about 1% in fissile material requirements. Thus, this resource savings option was recommended (with modifications) for inclusion in the subsequent Phase 2 studies.

It is recognized that special heat transfer problems are associated with the coolant flow characteristics of the guide tube unit cell when fertile rods that eventually become power-producing rods are inserted into guide tubes. The resolution of the heat transfer question was outside the scope of the present study and would have to be addressed to confirm the adequacy of any specific design before implementation.

3.2.5. Options Eliminated From Further Study

The three remaining options were judged to be unattractive or of marginal benefit in a recycle mode of operation. The concept of removing partial-length control rods at the end of life and the concept of temperature control during a reactor operating cycle to achieve higher burnups involve the principle of reducing the residual fissile content of the fuel. Thus, these concepts have potential benefits for a once-through mode of operation, but there is no incentive to employ them in a recycle mode since the fissile content of the spent fuel could be recovered through reprocessing. Thus, these modification concepts were not recommended for further detailed study in the subsequent phase of this program.

The last of the suggested improvement options was enrichment zone loading to eliminate the need for burnable poison rods. In principle, this would allow implementation of an "in-out" fuel management strategy to reduce core leakage without having to rely on parasitic absorber rods. Residual end-of-cycle negative reactivity effects of conventional LBP rods would be eliminated, and more complete utilization of the fissile loading in the core would be realized. Again, this concept is attractive within the context of a once-through cycle but it has minimal benefits in a recycle mode. It would also compete with the concept of ThO_2 rods replacing parasitic LBP rods. Thus, this concept was not pursued further.

Table 3-1. Reactor Design Data

	<u>Core Design Identification</u>		
	<u>Reference</u>	<u>Interm. lattice</u>	<u>Tight lattice</u>
<u>Reactor</u>			
Design heat output, MWt	3760	3760	3760
Vessel coolant inlet temperature, °F	569.0	569.0	569.0*
Vessel coolant outlet temperature, °F	626.0	626.0	626.0*
Core coolant outlet temperature, °F	631.6	631.6	631.6*
Operating pressure, psia	2250	2250	2250*
<u>Core</u>			
Total No. of fuel assemblies in core	205	177	161
Fuel assembly pitch spacing, in.	8.587	8.587	8.587
Avg thermal output, kw/ft	5.66	≤5.66	≤5.66
Avg core fuel temp, °F	1255	1255	1255
Total heavy metal (BOL, batch) Metric tons	29	29	29
Core dimensions, in.			
Equivalent diameter	138.7	128.9	122.9
Active height	143	132.6	130.6
<u>Fuel Assembly</u>			
Lattice M/W ratio	0.78	1.26	1.69*
No. of fuel rods per FA	264	328	404
No. of control rod guide tubes per assembly	24	32	36
No. of incore instr positions per FA	1	1	1
Fuel rod outside diameter, in.	0.379	0.379	0.362*
Fuel rod pitch, in.	0.502	0.446	0.402*
Fuel cladding ID, in.	0.332	0.332	0.316*
<u>Fuel</u>			
Material	(U ²³⁵ U-Th)O ₂		
Form	Dished-end cylindrical pellets*		
Pellet diameter, in.	0.324	0.324	0.308*
Density, % of theoretical	94.0	94.0	94.0*

* These lattice properties and a fuel temperature of 800F were used to describe blanket and guide tube rod regions.

Figure 3-1. Denatured $^{233}\text{U}/\text{ThO}_2$ Cycle Fuel Consumption and Conversion Ratio as a Function of Lattice Metal/Water

4. PHASE 2 — ASSESSMENT OF IMPROVEMENTS DUE TO A COMBINATION OF PROMISING DESIGN CHANGES

4.1. Introduction

The three design changes identified in Phase 1 as most promising were investigated in greater depth in Phase 2 by combining them into a conceptual modified reactor core design. Four variations of the core design were investigated to determine the fuel resource savings achievable by employing various combinations of the design changes.

The first variation involved two changes from the reference design: (1) an intermediate metal-to-water volume ratio (M/W), which was accomplished by changing from a 17×17 to 19×19 array within the reference assembly envelope and (2) the addition of a "thin blanket" consisting of 3-inch axial blankets and a 3-inch-thick radial blanket. (The radial blanket assemblies were assumed to have a 21×21 pin array.)

The second variation involved the same "thin blanket" plus a change within the core to a "high-M/W" lattice, corresponding to the 21×21 pin array used in the blanket assemblies.

A third variation maintained the same high-M/W lattice as above in the core, but had a considerably thicker blanket surrounding the core.

The final variation included the high-M/W core, the thick blankets, and ThO_2 rods in available control rod guide tube locations within the core.

Because of concerns related to economics, a key requirement of the Phase 2 work was that the design improvements be realistic or practical. The concept of practicality was delineated by the following guidelines observed in developing the design:

1. The final product should require no change in the size of the reactor vessel used in the B&W Standard Plant design.
2. Realistic and feasible modifications inside the reactor vessel can be tolerated.

3. No major changes should be required in the balance-of-plant equipment.
4. The capital cost of the conceptual plant should not be appreciably greater than that for a current light water reactor (LWR) plant.
5. Existing core thermal design limits — linear heat rate, core pressure drop, and DNBR — should not be affected adversely.
6. The design must not affect adversely any reactivity shutdown or accident safety margins.
7. The average batch discharge exposure should be kept at about 33,000 MWd/mtU. (It is recognized that reducing the burnup much below this value would increase the conversion ratio; however, this modification would not be commercially desirable nor economically attractive.)

The reference mechanical design was the B&W 205 Standard Plant as described in the FSAR and summarized in Table 3-1. The core barrel diameter set a limit on thickness of the radial blanket that could be accommodated. The reference fuel assembly length determined the maximum combined fuel and axial blanket length. To ensure mechanical design compatibility with the reference vessel head and reactor internals, neither the fuel assembly envelope nor the control rod guide tube array envelope were changed from the reference design. A square pin pitch was chosen to conform to the reference fuel assembly envelope and to be consistent in principle with familiar LWR fuel design and fabrication practice. Reactor power level and core heavy metal loading were held constant for all studies. These considerations minimize hardware changes for retrofitting a core design with a higher M/W ratio than the reference design and conform to the intent of guidelines 1 through 4.

The reference design reactor power level, coolant and fuel temperatures, and core heavy metal loading were used for all cases. The ratio of peak-to-average power and the linear heat rate were monitored to ensure that these reference design limits could be met in more definitive future analyses. The numbers of control rod assemblies and control rods and their configuration in a given assembly were selected to give reasonable assurance that reactivity shutdown and safety margins could be met. The conventional out-in fuel management scheme was used in all cases for a three-batch core with annual refueling and approximately 33,000 MWd/mtM average discharge batch exposure. These considerations minimize changes to the balance-of-plant design, thermal design, and reactivity safety margin and conform to the intent of guidelines 4 through 7.

4.2. Fuel Assembly Configurations

Because of the desire to retrofit the modified core to the reference B&W 205 Standard Plant, several restraints were imposed on the design of the fuel assembly, as discussed in the preceding paragraphs. These constraints guided the choice of fuel rod and unit cell dimensions, as well as the number and locations of control rod guide tubes.

4.2.1. Lattice Cell

Two lattices with M/W ratios higher than those in the reference assembly were investigated in this study. One of these modified lattices had an "intermediate M/W" of 1.26, which corresponds to a 19×19 fuel pin array in the assembly. The other modified lattice had a "high M/W" of 1.69, which corresponds to a 21×21 fuel pin array within the assembly. The M/W ratio of 1.69 was selected because it was judged to be the tightest lattice that could be accommodated with a square cell pitch and remain compatible with the thermal-hydraulic design of the B&W 205 Standard Plant primary loop.

This tightest lattice allowed for a gap between adjacent fuel pins (surface-to-surface) of 0.040 inch. This minimum gap thickness was based on thermal design analyses performed for other low-water-fraction core designs performed by B&W. Geometric parameters for the reference lattice cell and for the two modified cells are presented in Table 4-1.

Blanket assemblies were assumed in all cases to have a 21×21 pin array and thus corresponded to the high-M/W lattice as described in Table 4-1.

4.2.2. Control Rod Guide Tube Configurations

The number of control rod guide tubes in the modified fuel assemblies was chosen to provide reasonable assurance that reactivity shutdown could be achieved when using the maximum number of rodded assemblies (equal to the maximum number of control rod drives) available in the B&W 205 Standard Plant design. Control rod worth was not calculated; however, control requirements and availability were assessed as a basis for setting the guide tube density in the fuel lattice. The core average control pin density (ratio of control pins/fuel pins) that had been found to be sufficient for a plutonium-uranium-fueled tight lattice design was used for the low-enriched thorium core. This ratio was adjusted for fuel-related design differences in the two concepts. Finally, the average control pin density was weighted by the ratio of total fuel assemblies in the core to

the maximum number of control rod drives in the B&W 205 Standard Plant. It was assumed that any control rod worth deficiency could be compensated by using enriched boron as the control poison.

The configuration of the guide tubes in a fuel assembly was chosen so that the radial distance from the fuel assembly center to the outermost guide tube center would be no greater than the corresponding radial distance for the reference fuel assembly used in the B&W 205 Standard Plant. The guide tubes were spaced in an approximately uniform pattern in the circle defined by this radius. This ensures that the control rod assembly in the withdrawn position will minimize changes in the upper internals components.

The number of control pins and their configurations in the fuel assembly are illustrated in Figures 4-1, 4-2, and 4-3 for the reference, intermediate, and tight lattice designs, respectively.

4.3. Core-Blanket Configurations

The intermediate-M/W core contained 177 fuel assemblies and 28 radial blanket assemblies arranged as shown in Figure 4-4. The high-M/W core contained 161 fuel assemblies and 44 radial reflector assemblies arranged as in Figure 4-5.

4.4. Calculational Methods and Models

4.4.1. Calculational Methods

Two standard nuclear design computer codes were used to generate the required neutronic and fuel cycle parameters. One of these is NULIF, B&W's standard lattice cell code.⁵ The other nuclear code is B&W's version of PDQ-7, a multi-dimensional diffusion theory depletion code.¹⁰ These programs are described below along with other pertinent information concerning calculational models and methodology.

4.4.1.1. NULIF

NULIF is a fundamental-mode neutron spectrum generation and fuel depletion program, which is used to calculate reactivity, conversion ratio, and burnup-dependent, spectrum-weighted, few-group neutron cross sections for use in diffusion theory spatial calculations. The NULIF code was recently modified to include the capability to calculate thorium-based fuel systems.⁴ These modifications included the generation of multigroup neutron cross sections from ENDF/B-4 data files for the thorium chain isotopes, the incorporation of

extended depletion chains for the thorium chain isotopes, and the addition of resonance integral formulations for thorium cycle isotopes.

The thorium cycle depletion chain incorporated in the specially modified version of NULIF can be represented schematically by

The thorium cycle time-dependent depletion equations are solved using the same numerical methods as are applied for the uranium cycle chains. This method has checked very well against the exponential methods used in PDQ07 and CINDER¹¹ and is considerably faster with respect to computer time.

The resonance shielding in thorium cycle isotopes is based on the single level Breit-Wigner representation and the Adler-Nordheim model. This is the model used for the uranium cycle isotopes. It neglects resonance-resonance interference but accounts for resonance-potential scattering interference (approximately) using a smooth resonance contribution.

The modified NULIF code and newly generated multi-group cross sections were checked by calculating the effective multiplication factor of a number of (²³³U-Th)₂O₂ exponential experiments.¹² These results (presented in Table 4-2) are satisfactory and are generally consistent with benchmarking comparisons reported by others.

The NULIF code was applied in this study to calculate lattice parameters using four neutron energy groups defined as follows:

<u>Group No.</u>	<u>Energy range</u>
1	10 MeV - 9.12 keV
2	9.12 keV - 61.44 eV
3	61.44 eV - 1.85 eV
4	1.85 eV - 0

4.4.1.2. PDQ07

PDQ07 is a diffusion-depletion program (in one, two, or three dimensions) which is used to obtain the neutron flux and material distributions in a reactor at various times in life. In practice, the calculations are performed in

two steps. First, for an initial description of the reactor (geometry and material description), the neutron flux distribution for neutrons in several energy groups is obtained at discrete spatial mesh points in the reactor and the surrounding reflector regions. In the solution of the spatial problem an eigenvalue, which is a measure of the closeness of the described reactor to criticality, is found. The spatial flux is combined with the material inventory and nuclear cross sections to obtain the power distribution.

Once the spatial flux and power distribution have been obtained, the second step in PDQ07 (diffusion-depletion) is to simulate reactor operation during a specified time interval. Using the power-normalized flux from the spatial calculation, the differential equations describing the time behavior of the nuclide concentrations are solved for that time interval. The solution yields a new distribution of nuclide concentrations in the reactor, which is used in the generation of few-group macroscopic cross sections for the next spatial calculation.

4.4.1.3. PDQ07 Depletion Chains

A nuclide can be transformed by several modes of radioactive decay or by neutron absorption; likewise, it can be created when parent nuclei undergo such processes or by direct fission yield. Thus, the concentration of a particular nuclide may depend on more than one immediate parent as well as its own radioactive decay, neutron absorption rate, and yield from fission. A complete schematic of all coupling mechanisms between the precursors that feed and transform each nuclide may appear to be quite complicated. (The nuclear interactions between isotopes for the thorium depletion chain are given in Figure 4-6.)

The algorithm currently used in PDQ07 to solve the associated differential equations requires couplings in which nuclei are fed by a single parent (linear coupling). However, provided that a nuclide concentration is independent of the concentration of any of its progeny, it is always possible to resolve more complicated couplings into linearly coupled chains of nuclides because of the linearity of the equations.

The thorium depletion chain in Figure 4-6 is linearized into the two chains given in Figure 4-7 for this study. These linear chains branch at ^{233}Pa with the contribution to isotopes following ^{233}Pa being obtained as the sum of contributions from each chain.

4.4.1.4. Cross Section Preparation

A set of burnup-dependent microscopic cross sections was prepared for input into PDQ07 in order to perform the required diffusion-depletion analysis. A generic PDQ07 cross sectional tableset generating package was developed specifically to perform this analysis.

This package allows the user to create a four-energy group, burnup-dependent cross section tableset for any arbitrary NULIF case. The user sets up the NULIF portion of the generating deck to represent the initial fuel concentration and lattice geometry of interest. The generic fitting portion then will automatically process the NULIF code output into a punched-card PDQ07 tableset deck. The basic structure of the generic fitting scheme is as follows:

1. Group 1 and group 2 cross sections are considered constant throughout burnup with the reference microscopic values taken at 400 days.
2. The macroscopic transport cross section and the macroscopic removal cross section are fit versus burnup in all four energy groups.
3. The non-depletable materials are lumped together to form a macroscopic absorption cross section that is fit versus burnup in all four energy groups.
4. The boron concentration is held constant with burnup. The fission product concentrations are allowed to build up naturally in the NULIF depletion.
5. The microscopic cross sections that are fit versus burnup are given in Table 4-3.
6. The burnup points that are chosen as fit points are 0, 50, 200, 400, 600, 800, and 1000 days.

4.4.1.5. Fissile Enrichments

The source of ^{233}U for the denatured ^{233}U fuel was assumed to be a fast breeder reactor blanket, as suggested by ORNL. This feed material contained 97 wt % ^{233}U and 3 wt % ^{234}U . Denatured uranium was produced by diluting the feed material with depleted uranium (0.2 wt % ^{235}U) to reduce the fissile isotope content to a proliferation-resistant level specified by a weighted average enrichment limit according to the formula

$$\frac{0.12 M_{233} + 0.20 M_{235}}{M_{233} + M_{235}} \geq \frac{M_{233} + M_{235}}{M_T}$$

where

$$M_{233} = \text{mass of } ^{233}\text{U},$$

$$M_{235} = \text{mass of } ^{235}\text{U},$$

$$M_T = \text{total mass of uranium.}$$

The NULIF code was used to establish the enrichment-reactivity relationship, consistent with the methods used in other NASAP thorium studies.⁴ The feed enrichment was defined as

$$\frac{\text{wt of } ^{233}\text{U} + ^{235}\text{U}}{\text{wt of U} + \text{Th}} \times 100.$$

Feed enrichments are listed in Table 4-4 as a function of the M/W ratio of the lattice. These enrichments are for a three-batch core with a discharge batch average burnup of 33,000 MWd/mtM.

4.4.2. Core-Blanket (R,Z) Model and Calculational Procedures

The PDQ code was used with (R,Z) geometry to calculate the combined fuel utilization improvements of tight lattice cores and external fertile material blankets (both axial and radial). The core was modeled as a circular cylinder with three equal-volume, concentric, annular regions. These three homogenized regions corresponded to three fuel batches that were used in an out-in fuel management plan (see section 4.4.3) with the volume of each batch conserved in the cylindrical model. Two different core sizes, which corresponded to the two modified lattices mentioned above (M/W ratios of 1.26 and 1.69), were investigated.

In addition, two blanket thicknesses were investigated. One blanket consisted of a 3-inch-thick radial zone surrounding the core and 3-inch-thick axial zones at the top and bottom of the core. The other blanket configuration had a 9-inch-thick radial blanket and 6-inch-thick axial blankets at the top and bottom.

Figure 4-8 is a schematic drawing of the (R,Z) model. The outermost radial region (reflector) is a homogenized representation of the core barrel, support assembly, and downcomer water. The axial reflector was treated as an extension of the fuel assembly with the fuel replaced by void.

The M/W ratio of the basic unit cell underestimates the overall moderating characteristics of a reactor design because it neglects the presence of control rod guide tubes, instrumentation tubes, and clearance gaps between fuel assemblies. For this study the additional water of these regions was distributed

equally among all fuel cells prior to calculation of the four-group PDQ coefficients with the NULIF spectrum generation code. The volume fraction data for the lattices and cores are given in Tables 4-5 and 4-6, respectively. The equivalence between lattice and core M/W ratio and comparable fuel/water ratios is given in Table 4-7.

For each of the design variations, the initial core was fueled with fresh fuel in the outer region, and once- and twice-burned (11,000 and 22,000 MWd/mtM) fuel in the intermediate and inner regions, respectively. This configuration was depleted using the diffusion-depletion computer code PDQ07. After the cycle depletion, the central region fuel having approximately 33,000 MWd/t exposure was discharged, the isotopes for the first annulus (22,000 MWd/t exposure) were shuffled inward to the central region, and the outside annulus isotopes (11,000 MWd/t exposure) were shuffled inward to the first annulus. Fresh fuel was again loaded in the outside region. Isotopic decay for refueling (one half-life for ^{233}Pa) was accounted for. The blankets were loaded with thorium oxide pins for the initial cycle and irradiated for three cycles with no shuffling. For each design modification study, the core was depleted in this manner for three reactor operating cycles. In addition, Appendix A contains isotopic data for six cycles of blanket irradiation, computed by a less detailed technique than that used for the main report.

4.4.3. Modeling and Procedures for (X,Y) Calculation of ThO_2 Rods in Guide Tubes

Approximately half of the guide tubes in a PWR do not contain components in any given operating cycle. The configuration of this unused volume is fixed and determined by the design control rod pattern. This volume, which is about 4% of the core fuel volume, offers a relatively low cost method of introducing fertile material into the core. A study was performed to evaluate the resource savings potential for this concept when used in combination with the tight core lattice and the thick blanket design.

A radial cross section of the core and blanket was modeled in detail with a two-dimensional (X-Y) model with each unit cell represented by a mesh block as shown in Figure 4-9. The design was depleted for three cycles. The design parameters used in this study are identical to those given in Table 3-1. The core arrangement for cycle 3 is shown in Figure 4-10. Other key modeling and design details are as follows:

The active core comprised 161 assemblies surrounded by an outer row of 44 blanket assemblies. The reactor internals were modeled out to the pressure vessel. The ThO_2 rods replaced the normal LBP clusters in the 72 fuel assemblies that were not under control rods. The ThO_2 rod/fuel ratio was about 0.04. The core consisted of three fuel batches. Batches 1 and 2 had burnups of approximately 22,000 and 11,000 MWd/mtM, respectively, at the beginning of cycle 1. Fuel batches were shuffled in subsequent cycles with a constant feed batch of 4% $\text{D}^{233}\text{U}/\text{ThO}_2$ fuel. The discharged design burnup was approximately 33,000 MWd/mtM.

The guide tube and blanket pins, which were fresh natural thorium oxide at the beginning of cycle 1, remained in the reactor for all three cycles with each guide tube rod in the same core location for the three cycles. This assumes that these rods will be in different fuel assemblies since the fuel is shuffled. A 171×171 mesh array was used in quarter-core geometry with explicit representation of guide tube cell, water gap, instrument cell, fuel cell, radial blanket, baffle, and reflector regions. Each unit cell within the core and blanket was represented by a mesh block.

Three operating cycles were depleted in order to observe the buildup of fissile material in the guide tube ThO_2 rods and the change in blanket influence upon the core. Region isotopic masses by cycle were obtained. In addition, the reactivity worth of the guide tube ThO_2 was computed by replacing these rods with water at the beginning of each cycle and at the end of cycle 3.

4.5. Results and Conclusions Related to Fuel Resource Savings

The calculational methods and models described above were employed to determine the fissile fuel resource savings that could be achieved through the use of the most promising design changes, namely (1) a tighter pitch lattice (higher M/W ratio), (2) external fertile material (ThO_2) blankets, and (3) ThO_2 rods in available guide tube space.

Four variations of the modified reactor design were investigated and compared to the reference design to determine the fuel resource savings achievable by employing various combinations of design changes. The five designs (the reference design plus four modified designs) are summarized in Table 4-8. This table includes case identification that is used in the subsequent discussion of results.

The fuel utilization was evaluated in terms of the net amount of fissile mass destroyed to produce a given amount of energy. These results expressed as grams fissile mass destroyed per megawatt-day are provided in Table 4-9. Data are provided for each design improvement case studied. The results are expressed as combinations of an improved core plus radial blanket, axial blanket, or guide tube rod improvement. In addition, results of a design incorporating all improvements is provided.

The fissile mass savings for each of the design iterations are shown in Table 4-10. The fuel mass savings was defined as the difference in fissile mass destroyed between the improved core design and the reference design. This table shows the combined fissile mass savings for each case, as well as the contribution of each individual design change. The largest fissile mass savings is achieved for the case combining the high M/W lattice, the thick blanket, and ThO_2 rods in guide tubes. The total savings for this case amounts to 64% of the fissile mass destroyed in the reference case. Furthermore, in this optimum case about 39% of the total savings is due to the change to the very tight lattice. The calculations also indicate that the magnitude of this contribution is independent of the blanket thickness or the presence of ThO_2 in the guide tubes.

About 41% of the total savings in the optimum case (case 5) is due to the radial blanket. The resource savings in the blanket was found to be independent of the core lattice design, but it varies with the blanket thickness. The larger savings as a function of blanket thickness is related to the fact that as the blanket thickness is increased, the power fraction in the blanket increases, especially by the end of its third cycle of residence in the reactor.

The fissile mass savings provided by inserting ThO_2 rods in available guide tubes in the tight lattice core amounts to about 12% of the total savings for case 5. Furthermore, the fissile mass savings of the ThO_2 guide tube rods is about two times that of the thin axial blanket and about one half that of the thin radial blanket.

On a unit mass basis, however, the effectiveness of the ThO_2 guide tube rods is considerably greater. The net production of ^{233}U in the guide tube rods per unit mass of ThO_2 is about two times that of the radial blanket and three times that of the axial blanket.

Detailed fissile isotope masses at the beginning and end of life are given in Tables 4-11 through 4-17 for all five cases. These include three reference core cases differing only in batch burnup (for comparison purposes). The net fissile masses destroyed in each core batch are given by isotope in Tables 4-18 through 4-21. The mass values in Tables 4-18 through 4-21 include an approximate "fissile mass equivalence" of 1 gram mass per megawatt-day of energy for the energy produced in the blankets and guide tube rods. Because the power split between core and blankets (and guide tube rods) is different for each design iteration, the PDQ07 depletion calculations were taken to a constant reactor energy production (i.e., core plus fertile regions). The reported isotopic data were normalized to the same core energy production per cycle for each set of cases.

An important observation from these isotopic mass data is that the principal contribution to the reduction of net fissile mass consumption comes from the buildup of fissile plutonium. Thus, the design improvements reported herein will result in greatly limited fuel savings unless credit can be taken for plutonium. This would require that the overall reprocessing/recycle strategy include utilization of the plutonium generated in the denatured $\text{UO}_2\text{-ThO}_2$ fuel. In light of current nuclear proliferation concerns, such plutonium utilization would have to take place within the confines of secure fuel cycle centers, which would be needed in any event for the reprocessing and denaturing operations required for a thorium fuel cycle. It is assumed in NASAP that plutonium is utilized inside such secure fuel cycle centers in plutonium-fueled reactors that produce ^{233}U for use in denatured ^{233}U -fueled reactors located at dispersed sites.

4.6. Supplemental Results

In the course of performing the calculations described above, considerable information was generated which supplements the primary results reported in the preceding section. Supplemental information is presented below related to three aspects of the study. These are (1) the power split between core and blanket, (2) the reactivity worth of ThO_2 rods in guide tube locations, and (3) conversion ratios.

4.6.1. Power Split Between Core and Blanket

The power split between the core and surrounding blankets is shown in Figure 4-11 as a function of the blanket irradiation exposure, i.e., by reload cycle.

Over three cycles, the blanket power fraction builds up in a near-linear manner to about 4.5% for the thick blanket and 2.5% for the thin blanket with approximately 80% of this contributed by the radial blanket. Part of this large contribution is due to the larger radial blanket volume. On an equal volume basis, the radial blanket contribution drops to about two-thirds of the total blanket power. The highest blanket power (per unit volume) occurs at the radial blanket midplane adjacent to the core.

4.6.2. Reactivity Worth of Guide Tube ThO₂ Rods

The reactivity effects of thorium oxide rods in guide tubes were calculated with the PDQ code for a tight-lattice core blanket design modeled as a two-dimensional core radial cross section. The reactivity effect of the rods was calculated by replacing the rods with water. This was done at the beginning of each cycle to obtain burnup effects and at the end of cycle 3 to obtain the reactivity deficit. The results of these calculations are given in Table 4-22. The buildup of ²³³U with rod exposure provides a small positive reactivity change (0.20% Δρ) over three cycles. There is an end-of-cycle reactivity penalty of about -2.2% Δρ due to thorium absorptions and water displacement. This penalty will result in a higher fissile loading requirement, which is recovered upon reprocessing.

4.6.3. Conversion Ratios

Conversion ratios were calculated for each design change evaluated and are given in Table 4-23. The conversion ratio CR is defined as

$$CR = \frac{\text{atoms fissile generated}}{\text{atoms fissile destroyed}}$$

A denatured ²³³U system has four fissile nuclides: ²³³U, ²³⁵U, ²³⁹Pu, and ²⁴¹Pu. Destruction is accomplished by neutron absorption (capture plus fission) in these nuclides. Generation is accomplished by neutron capture in the fertile materials: ²³²Th, ²³⁴U, ²³⁸U, and ²⁴⁰Pu. Neutron capture by ²³⁴U leads to ²³⁵U, and capture by ²⁴⁰Pu leads to ²⁴¹Pu. Neutron capture in ²³²Th and ²³⁸U has a more complicated path. Capture in ²³²Th requires two successive beta decays prior to producing ²³³U. The half-life of ²³³Pa is 27.0 days, admitting neutron capture loss in the event that ²³³Pa captures a neutron prior to decay. In fact, the loss of ²³³U through ²³³Pa capture results in a decrease in conversion ratio of 1 to 2%. The half-life of ²³⁹Np is 2.35 days. Consequently, the probability

for capture in ^{239}Np prior to decay is reduced. The loss of ^{239}Pu through ^{239}Np capture represents a decrease in conversion ratio of only 0.3 to 0.5%.

The conversion ratio can be calculated from the decay rate of ^{233}Pa or the capture rate in ^{232}Th . The latter is always in equilibrium with the power and flux level, and hence always gives an "asymptotic" conversion ratio assuming instantaneous decay to ^{233}U without loss.

The decay from ^{233}Pa (in addition to neutron capture) is not constant because the long half-life of ^{233}Pa prevents equilibrium for other than long periods of operation at one power level. It is convenient to calculate the conversion ratio using ^{232}Th capture as is usually done and point out — but not necessarily apply — the ^{233}Pa correction. For this study it was decided to compute the conversion ratio as defined above and using the instantaneous capture rate in ^{232}Th without adjustment for ^{233}Pa capture. Volume average reaction rates at a given time in the core cycle were taken from three-dimensional PDQ calculations for the design improvement being evaluated. This method yields what is conventionally called the "instantaneous conversion ratio."

The incremental conversion ratio ΔCR given in Table 4-23 for the core lattices is the difference between the core conversion ratio for the improved lattices and that for the reference design core. The incremental conversion ratio for the blankets and guide tube rods is the difference between the conversion ratio for the tight lattice core plus blanket and/or guide tube rods and the conversion ratio for the tight lattice core only. The incremental conversion ratio was approximately the same for the thin blanket with either the tight-lattice or intermediate-lattice core. On this basis, the incremental conversion ratios are assumed additive as a basis for estimating the maximum achievable conversion ratio reported in Table 4-23 for a combination of improvements.

On a volume basis, the three types of fertile regions as measured by the incremental conversion ratio are ranked as follows: the guide tube rods, thin blankets, and thick blankets. This also corresponds to the relative ranking of the average epithermal neutron flux in the three regions.

Table 4-1. Description of Lattices

<u>Fuel assembly ident</u>	<u>Reference MKC</u>	<u>Interm. M/W</u>	<u>High (dry lattice) M/W</u>
M/W	0.78	1.26	1.69
Pin diameter, in.	0.379	0.379	0.362
Gap thickness, in.	0.123	0.067	0.040
Pin pitch, in.	0.502	0.446	0.402
Number of pin cells/FA	17 × 17	19 × 19	21 × 21

Table 4-2. $^{233}\text{U}/\text{ThO}_2$ Exponential Experiment⁵
Benchmark Comparison

<u>Moderator/fuel volume ratio</u>	<u>H_3BO_3 concentration, mg/cm^3</u>	<u>"Best" value of measured B_m^2, m^{-2}</u>	<u>Calculated k_{eff}</u>
0.9970	0	75.88	0.9923
1.3840	0	86.06	0.9941
1.7134	0	89.34	0.9982
2.1943	0	90.35	1.0005
3.0043	0	85.54	1.0015

Calculated critical $\bar{k}_{\text{eff}} = 0.9973 \pm 0.004$

Table 4-3. Microscopic Cross Sections for Input to PDQ07

<u>Isotope</u>	<u>Cross section type(a)</u>	<u>Energy group</u>	<u>Isotope</u>	<u>Cross section type(a)</u>	<u>Energy group</u>
^{232}Th	A	3	^{238}U	A	3
^{232}Th	A	4	^{238}U	A	4
^{233}U	A	3	^{239}Pu	A	3
^{233}U	A	4	^{239}Pu	A	4
^{233}U	F	3	^{239}Pu	F	3
^{233}U	F	4	^{239}Pu	F	4
^{234}U	A	3	^{240}Pu	A	3
^{234}U	A	4	^{240}Pu	A	4
^{235}U	A	3	^{241}Pu	A	3
^{235}U	A	4	^{241}Pu	A	4
^{235}U	F	3	^{241}Pu	F	3
^{235}U			^{241}Pu	F	4

(a) A: absorption, F: fission.

Table 4-4. Batch Feed Enrichments

	<u>Enrichment, (a) wt % $\text{D } ^{233}\text{U}$</u>
Reference lattice (Mark C)	3.1
Intermediate lattice	3.6
Tight lattice	4.0

(a) Uranium denaturant (0.20 wt % ^{235}U).

Table 4-5. Lattice Parameters

<u>Identification</u>	<u>Reference lattice (Mark C)</u>	<u>Interm. lattice</u>	<u>Tight lattice</u>
Pin OD, in.	0.379	0.379	0.362
Clad ID, in.	0.332	0.332	0.316
Pellet OD, in.	0.324	0.324	0.308
Pin pitch, in.	0.502	0.446	0.402
Pellet volume fraction	0.3272	0.4145	0.4610
Gap volume fraction	0.0164	0.0207	0.0243
Clad volume fraction	0.1041	0.1319	0.1516
Water volume fraction	<u>0.5523</u>	<u>0.4329</u>	<u>0.3631</u>
	1.0000	1.0000	1.0000
Metal/water ^(a)	0.7809	1.2624	1.6871
Fuel/water	0.5923	0.9576	1.2697
Water/fuel	1.6883	1.0443	0.7876

(a) Metal excludes fuel pin gap volume.

Table 4-6. Core Metal/Water Parameters

<u>Identification</u>	<u>Reference lattice (Mark C)</u>	<u>Interm. lattice</u>	<u>Tight lattice</u>
Pellet volume fraction	0.2952	0.3668	0.4082
Gap volume fraction	0.0148	0.0183	0.0215
Clad volume fraction	0.0939	0.1167	0.1342
Guide tube volume fraction	0.0087	0.0087	0.0079
H ₂ O volume fraction	<u>0.5874</u>	<u>0.4895</u>	<u>0.4282</u>
	1.0000	1.0000	1.0000
Metal/water ^(a)	0.6772	1.0056	1.2854
Fuel/water	0.5026	0.7493	0.9534
Water/fuel	1.9898	1.3346	1.0489

(a) Metal excludes fuel pin gap volume.

Table 4-7. Summary of Lattice and Core Metal/Water and Fuel/Water Ratios

Lattice ident	Lattice		Core	
	Metal/water ^(a)	Fuel/water	Metal/water ^(a)	Fuel/water
Reference (Mark C)	0.7809	0.5923	0.6772	0.5026
Intermediate	1.2624	0.9576	1.0056	0.5026
Tight	1.6871	1.2697	1.2854	0.9534

(a) Metal excludes fuel pin gap volume.

Table 4-8. Design Summary With Case Identification —
D ²³³U/ThO₂ Fuel, ThO₂ Blanket

Case ID	Core			Blankets			Guide tubes w/ThO2
	Enrichment, wt %	M/W	No. of FAs	M/W	Thickness, in.		
					Radial	Axial	
1	3.1	0.70	205	--	--	--	--
2	3.6	1.01	177	1.29	3	3	--
3	4.0	1.29	161	1.29	3	3	--
4	4.0	1.29	161	1.29	9	6	--
5	4.0	1.29	161	1.29	1FA ^(a)	--	45%

(a) Peripheral fuel assemblies (44).

Table 4-9. Comparative Fissile Utilization — Net Fissile Mass
Destroyed/Energy Produced, g/MWd

Design	Reference core ^(a)	Improved core	Improved core + blanket type		Improved core + guide tube rods	Combined improvements ^(b)
			Radial blanket	Axial blanket		
Interm. M/W, thin blankets	0.2718	0.2424	0.2091	0.2338	--	0.2006
High M/W, thin blankets	0.2738	0.2043	0.1716	0.1956	--	0.1630
High M/W, thick blankets	0.2754	0.2061	0.1331	0.1916	--	0.1192
High M/W, thick blankets, guide tube rods	0.2754	0.2061	0.1331	0.1916	0.1858	0.1002

(a) Each design was evaluated at slightly different discharge fuel exposure.

(b) Improved core + radial blanket + axial blanket + guide tube rods, as appropriate.

Table 4-10. Summary of Fissile Mass Savings (%)

Design	Individual design improvement				Combined savings
	Core	Blanket		Guide tube rods	
		Radial	Axial		
Interm. M/W, thin blankets	10.8	12.3	3.2	--	26.2
High M/W, thin blankets	25.4	12.0	3.2	--	40.5
High M/W thick blankets	25.2	26.5	5.3	--	56.7
High M/W, thick blankets, guide tube rods	25.2	26.5	5.3	7.4	63.6

Note: The percentage fissile mass savings was calculated from the comparative fissile utilization data of Table 4-9. The individual fissile mass savings for the blankets and guide tube rods was isolated by subtracting the improved core contribution.

Table 4-11. Core Batch Mass Inventory

Reference design, Case 1
M/W 0.70
E, wt % 3.1
Burnup, MWd/mtM 34,981

Isotope	Core batch isotopic mass, kg		
	<u>Loaded</u>	<u>Fissile discharged</u>	<u>Fissile destroyed</u>
²³³ U	884.47	510.41 ^(a)	374.06
²³⁵ U	13.02	24.77	-11.75
²³⁹ Pu	--	67.41	-67.41
²⁴¹ Pu	--	19.18	-19.18
²³² Th	21,577	--	--
²³⁴ U	27.40	--	--
²³⁸ U	<u>6,497</u>	<u>--</u>	<u>--</u>
Total	28,998.89	621.77	275.72

(a) Includes ²³³Pa.

Table 4-12. Core Batch Mass Inventory

Reference design, Case 1A
M/W 0.70
E, wt % 3.1
Burnup, MWd/mtM 33,716

Isotope	Core batch isotopic mass, kg		
	<u>Loaded</u>	<u>Fissile discharged</u>	<u>Fissile destroyed</u>
²³³ U	884.47	518.05 ^(a)	366.42
²³⁵ U	13.02	24.18	-11.16
²³⁹ Pu	--	67.40	-67.40
²⁴¹ Pu	--	18.57	-18.57
²³² Th	21,577	--	--
²³⁴ U	27.40	--	--
²³⁸ U	<u>6,497</u>	<u>--</u>	<u>--</u>
Total	28,998.89	628.20	269.29

(a) Includes ²³³Pa.

Table 4-13. Core Batch Mass Inventory

Reference design, Case 1B

M/W 0.70

E, wt % 3.1

Burnup, MWd/mtM 34,262

<u>Isotope</u>	<u>Core batch isotopic mass, kg</u>		
	<u>Loaded</u>	<u>Fissile discharged</u>	<u>Fissile destroyed</u>
^{233}U	884.47	514.75 ^(a)	369.72
^{235}U	13.02	24.44	-11.42
^{239}Pu	--	67.40	-67.40
^{241}Pu	--	18.84	-18.84
^{232}Th	21,577	--	--
^{234}U	27.40	--	--
^{238}U	<u>6,497</u>	<u>--</u>	<u>--</u>
Total	28,998.89	625.43	272.06

(a) Includes ^{233}Pa .

Table 4-14. Batch Mass Inventory of
Core and Blankets

Improved design, Case 2

M/W 1.01
E, wt % 3.6
Burnup, MWd/mtM (core batch) 34,981

Isotope	Core batch isotopic mass, kg		
	Loaded	Fissile discharged	Fissile destroyed
^{233}U	1,027.78	629.03 ^(a)	398.75
^{235}U	15.13	31.48	-16.35
^{239}Pu	--	111.19	-111.19
^{241}Pu	--	25.27	-25.27
^{232}Th	20,371	--	--
^{234}U	31.80	--	--
^{238}U	7,545	--	--
Total	28,990.71	796.97	245.94

Fertile Region Batch Data

	Blankets		Guide tubes
	Radial	Axial ^(c)	
Blanket thickness, in.	3	3	--
No. of core cycles irr'd	3	3	--
Energy produced, MWd	12,974	1,530	--
Burnup, MWd/mtM	3,659	1,062	--
M loaded (^{232}Th), kg	3,546	1,441	--
^{233}U discharged, kg ^(b)	31.12	8.40	--

(a) Includes ^{233}Pa .

(b) Includes ^{233}Pa and ^{235}U (<1% total).

(c) Top and bottom.

Table 4-15. Batch Mass Inventory of
Core and Blankets

Improved design, Case 3
M/W 1.29
E, wt % 4.0
Burnup, MWd/mtM (core batch) 34,262

Isotope	Core batch isotopic mass, kg		
	Loaded	Fissile discharged	Fissile destroyed
^{233}U	1,141.49	737.40 ^(a)	404.09
^{235}U	16.80	35.31	-18.51
^{239}Pu	--	155.56	-155.56
^{241}Pu	--	27.02	-27.02
^{232}Th	19,407	--	--
^{234}U	35.30	--	--
^{238}U	8,383	--	--
Total	28,983.59	955.29	203.00

Fertile Region Batch Data

	Blankets		Guide tubes
	Radial	Axial ^(c)	
Blanket thickness, in.	3	3	--
No. of core cycles irr'd	3	3	--
Energy produced, MWd	12,859	1,771	--
Burnup, MWd/mtU	3,867	1,214	--
M loaded (^{232}Th), kg	3,325	1,459	--
^{233}U discharged, kg ^(b)	30.31	8.30	--

(a) Includes ^{233}Pa .

(b) Includes ^{233}Pa and ^{235}U (<1% total).

(c) Top and bottom.

Table 4-16. Batch Mass Inventory of
Core and Blankets

Improved design, Case 4

M/W 1.29
E, wt % 4.0
Burnup, MWD/mtM (core batch) 33,716

Isotope	Core batch isotopic mass, kg		
	Loaded	Fissile discharged	Fissile destroyed
^{233}U	1,141.49	740.90 ^(a)	400.59
^{235}U	16.80	34.90	-18.10
^{239}Pu	--	154.60	-154.60
^{241}Pu	--	26.40	-26.40
^{232}Th	19,407	--	--
^{234}U	35.3	--	--
^{238}U	8,383	--	--
Total	28,983.59	956.80	201.49

Fertile Region Batch Data

	Blankets		Guide tubes
	Radial	Axial ^(c)	
Blankets thickness, in.	9	6	--
No. of core cycles irr'd	3	3	--
Energy produced, MWd	21,930	2,426	--
Burnup, MWd/mtM	2,010	832	--
M loaded (^{232}Th), kg	10,909	2,917	--
^{233}U discharged, kg ^(b)	68.43	13.64	--

(a) Includes ^{233}Pa .

(b) Includes ^{233}Pa and ^{235}U (<1% total).

(c) Top and bottom.

Table 4-17. Batch Mass Inventory of Core,
Blankets and Guide Tubes

Improved design, Case 5

M/W	1.29
E, wt %	4.0
Burnup, MWd/mtM (core batch)	33,716

Isotope	Core batch isotopic mass, kg		
	Loaded	Fissile discharged	Fissile destroyed
^{233}U	1,141.49	740.90 ^(a)	400.59
^{235}U	16.80	34.90	-18.10
^{239}Pu	--	154.60	-154.60
^{241}Pu	--	26.40	-26.40
^{232}Th	19,407	--	--
^{234}U	35.3	--	--
^{238}U	8,383	--	--
Total	28,983.59	956.80	201.49

Fertile Region Batch Data

	Blankets		Guide tubes
	Radial	Axial ^(c)	
Blanket thickness, in.	9	6	--
No. of core cycles irr'd	3	3	3
Energy produced, MWd	21,930	2,426	9,900
Burnup, MWd/mtM	2,010	832	8,824
M loaded (^{232}Th), kg	10,909	2,917	1,122
^{233}U discharged, kg ^(b)	68.43	13.64	18.03

(a) Includes ^{233}Pa .

(b) Includes ^{233}Pa and ^{235}U (<1% total).

(c) Top and bottom.

Table 4-18. Comparative Fissile Mass Savings

Improved design, Case 2			Blanket thickness, in.	
M/W		1.01	Radial	3
E, wt %		3.6	Axial	3
Burnup, MWd/mtM (core batch)		34,981		
Core batch mass destroyed, kg			Mass produced in blankets, kg	
Isotope	Reference design	Improved design	Net savings	Radial Axial ^(b)
²³³ U	374.06	398.75	-24.69	31.12 ^(a) 8.40 ^(a)
²³⁵ U	-11.75	-16.35	4.60	-- --
²³⁹ Pu	-67.41	-111.19	43.78	-- --
²⁴¹ Pu	-19.18	-25.27	6.09	-- --
Total	275.72	245.94	29.78	31.12 8.40
Mass equivalent of energy				12.97 1.53
Total				44.09 9.93

(a) Includes ²³³Pa and ²³⁵U (<1% total).

(b) Top and bottom.

Table 4-19. Comparative Fissile Mass Savings

Improved design, Case 3			Blanket thickness, in.		
M/W		1.29	Radial		3
E, wt %		4.0	Axial		3
Burnup, MWd/mtM (core batch)		34,262			
<u>Isotope</u>	<u>Core batch mass destroyed, kg</u>			<u>Mass produced in blankets, kg</u>	
	<u>Reference design</u>	<u>Improved design</u>	<u>Net savings</u>	<u>Radial</u>	<u>Axial^(b)</u>
²³³ U	369.72	404.09	-34.37	30.31 ^(a)	8.30 ^(a)
²³⁵ U	-11.42	-18.51	7.09	--	--
²³⁹ Pu	-67.40	-155.56	88.16	--	--
²⁴¹ Pu	-18.84	-27.02	8.18	--	--
Total	272.06	203.00	69.06	30.31	8.30
Mass equivalent of energy				<u>12.86</u>	<u>1.77</u>
Total				43.17	10.07

(a) Includes ²³³Pa and ²³⁵U (<1% total).

(b) Top and bottom.

Table 4-20. Comparative Fissile Mass Savings

Improved design, Case 4			Blanket thickness, in.		
M/W		1.29	Radial	9	
E, wt %		4.0	Axial	6	
Burnup, MWd/mtU (core batch)		33,716			
	<u>Core batch mass destroyed, kg</u>			<u>Mass produced in blankets, kg</u>	
<u>Isotope</u>	<u>Reference design</u>	<u>Improved design</u>	<u>Net savings</u>	<u>Radial</u>	<u>Axial</u> ^(b)
²³³ U	366.42	400.59	34.17	68.43 ^(a)	13.64 ^(a)
²³⁵ U	-11.16	-18.10	6.94	--	--
²³⁹ Pu	-67.40	-154.60	87.20	--	--
²⁴¹ Pu	<u>-18.57</u>	<u>-26.40</u>	<u>7.83</u>	<u>--</u>	<u>--</u>
Total	269.29	201.49	67.80	68.43	13.64
	Mass equivalent of energy			<u>21.93</u>	<u>2.43</u>
			Total	90.36	16.07

(a) Includes ²³³Pa and ²³⁵U (<1% total).

(b) Top and bottom.

Table 4-21. Comparative Fissile Mass Savings

Improved design, Case 5		Blanket thickness, in.
M/W	1.29	Radial 9
E, wt %	4.00	Axial 6
Burnup, MWd/mtM (core batch)	33,716	

Isotope	Core batch mass destroyed, kg			Mass produced in blankets, kg		GT rods, mass prod., kg
	Reference design	Improved design	Net savings	Radial	Axial ^(b)	
²³³ U	366.42	400.59	34.17	68.43 ^(a)	13.64 ^(a)	18.03 ^(a)
²³⁵ U	-11.16	-18.10	6.94	--	--	--
²³⁹ Pu	-67.40	-154.60	87.20	--	--	--
²⁴¹ Pu	-18.57	-26.40	7.83	--	--	--
Total	269.29	201.49	67.80	68.43	13.64	18.03
		Mass equivalent of energy		21.93	2.43	9.90
		Total		90.36	16.07	27.93

(a) Includes ²³³Pa and ²³⁵U (<1% total).

(b) Top and bottom.

Table 4-22. Reactivity Worth of Guide Tube ThO₂ Rods

Operating cycle time	% Δρ
BOC 1	-2.4
BOC 2	-2.3
BOC 3	-2.2
BOC 4	-2.2

Table 4-23. Conversion Ratio

	<u>Δ CR</u>	<u>CR</u>
Reference design	--	0.76
Interm. M/W lattice	0.06	0.82
High M/W lattice	0.10	0.86
Thin blankets	0.03	--
Thick blankets	0.07	--
ThO ₂ rods in guide tubes	0.02	--
Maximum achievable ^(a)	--	0.95

(a) The sum of high M/W lattice, thick blankets, and ThO₂ rods.

Figure 4-1. Fuel Assembly Fuel Pin and Control Pin Layout

Figure 4-2. Fuel Assembly Fuel Pin and Control Pin Layout

INTERMEDIATE LATTICE
19x19 CELLS
32 GUIDE TUBES

Figure 4-3. Fuel Assembly Fuel Pin and Control Pin Layout

Figure 4-4. Intermediate Lattice Core and Blanket

Figure 4-5. Tight Lattice Core and Blanket

$$\begin{array}{l}
 {}^{232}\text{Th} \xrightarrow{\sigma\gamma} {}^{233}\text{Th} \\
 \lambda |T_{1/2} = 2.22 \text{ m.} \\
 \downarrow \\
 {}^{233}\text{Pa} \xrightarrow{\sigma\gamma} {}^{234}\text{Pa} \\
 \lambda |T_{1/2} = 27.0 \text{ d} \quad \lambda |T_{1/2} = \begin{array}{l} 6.75 \text{ h stable state} \\ 1.17 \text{ m metastable state} \end{array} \\
 \downarrow \\
 {}^{233}\text{U} \xrightarrow{\sigma\gamma} {}^{234}\text{U} \xrightarrow{\sigma\gamma} {}^{235}\text{U} \xrightarrow{\sigma\gamma} {}^{236}\text{U} \xrightarrow{\sigma\gamma} {}^{237}\text{U} \\
 \lambda |T_{1/2} = 6.75 \text{ d} \\
 \downarrow \\
 {}^{237}\text{Np}
 \end{array}$$
$$\begin{array}{l} {}^{232}\text{Th} \xrightarrow{\sigma\gamma} {}^{233}\text{Pa} \xrightarrow{\lambda} {}^{233}\text{U} \xrightarrow{\sigma\gamma} {}^{234}\text{U} \xrightarrow{\sigma\gamma} {}^{235}\text{U} \xrightarrow{\sigma\gamma} {}^{236}\text{U} \xrightarrow{\sigma\gamma} {}^{237}\text{Np} \\ {}^{232}\text{Th} \xrightarrow{\sigma\gamma} {}^{233}\text{Pa} \xrightarrow{\sigma\gamma} {}^{234}\text{U} \xrightarrow{\sigma\gamma} {}^{235}\text{U} \xrightarrow{\sigma\gamma} {}^{236}\text{U} \xrightarrow{\sigma\gamma} {}^{237}\text{Np} \end{array}$$

Figure 4-8. R-Z Grid Model ($\frac{1}{2}$ Core)

Figure 4-9. Mesh and Depletion Blocks for PDQ Calculations

FIGURE A
QUARTER CORE ASSEMBLY
74 DEPLETION BLOCKS

Figure 4-10. 161 Core Arrangement, Cycle 3

	9	10	11	12	13	14	15	16	17
K	2	1	2	1	2	1	2	3	B
L	1	2	1	2	1	3	1	3	B
M	2	1	2	1	3	1	3	B	
N	1	2	1	2	1	2	3	B	
O	2	1	3	1	2	3	B		
P	1	3	1	2	3	3	B		
R	2	1	3	3	B	B			
S	3	3	B	B					
T	B	B							

<u>Batch</u>	<u>No. of assemblies</u>	<u>Initial enrichment</u>
1 (twice burned)	56	4.0
2 (once burned)	53	4.0
3 (fresh)	52	4.0
B	Blanket	--

Figure 4-11. Blanket Power Fraction Buildup

APPENDIX A
Fuel Cycle Isotopic Masses

1. Introduction

The overall goal of this study was to determine the potential fuel utilization improvement achievable with a thorium-based reactor design incorporating a combination of promising design changes. Design retrofitability was emphasized through the set of ground rules followed in the study. As a consequence, the two- and three-dimensional analytical methodology required to model the design details precluded the generation of isotopic mass data for a large number of cycles.

However, to provide a useful basis for the construction of mass flow data for potential economic comparisons, a supplementary study using a different analytical technique was performed for the radial blanket depletion over six cycles. The results of this study, grouped with appropriate isotopic mass data from the main study, form the body of this appendix. The isotopic change in the radial blanket as a function of irradiation exposure was calculated for a lattice cell with the B&W NULIF code.⁵ This code is a zero-dimensional neutron spectrum generation code. The power level was set to achieve a specific burnup (MWd/mtM) as a function of core cycle time. The burnup schedule was established using the three-cycle radial blanket average exposure history from the three-dimensional PDQ core design calculations¹⁰ as a base and extrapolating to the equivalent of six cycles of residence time. The isotopic data, representing an average lattice cell, were adjusted to the radial blanket volume.

2. Design Description

Detailed descriptions of the improved core designs studied are provided in section 4.2 of the main report. The key differentiating characteristics of the core design improvements are given below:

- Reference Core Design — The BAW 205 Standard Plant core design fueled with $(D^{233}\text{U}/^{238}\text{U-Th})\text{O}_2$.

- Intermediate-M/W Core Design — This design had the same fuel and fuel assembly pitch as the reference core design. The pin diameter and pitch were reduced to achieve a fuel assembly M/W ratio of 1.01. The core has fewer fuel assemblies but the same heavy metal loading as the reference core.

- High-M/W Core Design — This design also has the same fuel and fuel assembly pitch as the reference design. The pin diameter and pitch were reduced to achieve a fuel assembly M/W ratio of 1.29. The core has fewer fuel assemblies but the same heavy metal loading as the reference and intermediate-M/W designs.

- Thin Blankets — These are radial and axial blankets 3 inches thick. The blankets have material volume fractions identical to those of the fuel lattice design for the high-M/W fuel assembly (excluding the guide tubes and instrument tubes). The blanket pellets are thorium oxide.

- Guide Tube Rods — The guide tube rods contain ThO_2 pellets. The material volume fractions are identical to those of the fuel lattice design for the high-M/W ratio fuel assembly.

3. Isotopic Mass Flow Data

Isotopic mass data are presented on a cycle basis for all design improvements evaluated except the thick blanket. It was concluded that the latter underutilized reactor space relative to the thin blanket and would not have as good an economic potential as the thin blanket concept.

The isotopic mass data for all design improvements except the radial blanket, i.e., those components that would normally remain in the core for three cycles, were taken from the two- and three-dimensional core design analyses performed with the PDQ code.¹⁰ This information is reported on a cycle basis in Tables A-1 through A-4. The radial blanket incore residence time can be independent of the core fuel cycle and can be optimized relative to material and fabrication costs. Therefore, calculations were performed to generate radial blanket isotopic buildup data through six core cycles. Approximately 12,000 MWd/mtM average irradiation exposure was achieved compared to 4,000 MWd/mtM exposure for three cycles. The radial blanket isotopic inventory by cycle is also given in Table A-4.

Table A-5 is a listing of the feed enrichments used for the equilibrium cycle analyses. Estimates of the batch enrichments for annual cycle first cores are also provided.

Table A-1. Reference Design — Reactor Cycle Isotopic Inventory — Equilibrium Core Batch (0.33 Total Core)

	Isotopic mass, kg			
	BOC 1	EOC 1	EOC 2	EOC 3
²³² Th	21,577	21,395	21,154	20,989
²³⁴ U	27.4	62.4	93.3	108.8
²³⁸ U	6,497	6,408	6,291	6,208
²³³ U	884.5	712.3 ^(a)	573.8 ^(a)	514.7 ^(a)
²³⁵ U	13.0	14.5	20.2	24.4
²³⁹ Pu	--	47.9	65.5	67.4
²⁴¹ Pu	--	3.5	13.8	18.8
Burnup, MWd/mtM	0	11,477	25,325	34,262

(a) Includes ²³³Pa.

Table A-2. Intermediate M/W (1.01) Core Design — Reactor Cycle Isotopic Inventory — Equilibrium Core Batch (0.33 Total Core)

	Isotopic mass, kg			
	BOC 1	EOC 1	EOC 2	EOC 3
²³² Th	20,371	20,162	19,929	19,788
²³⁴ U	31.8	73.2	105.0	119.6
²³⁸ U	7,545	7,413	7,262	7,171
²³³ U	1,027.8	828.6 ^(a)	688.3 ^(a)	629.0 ^(a)
²³⁵ U	15.1	18.7	26.4	31.5
²³⁹ Pu	--	73.0	103.9	111.2
²⁴¹ Pu	--	5.7	18.3	25.3
Burnup, MWd/mtM	--	13,264	27,019	34,981

(a) Includes ²³³Pa.

Table A-3. High M/W (1.29) Core Design — Reactor Cycle
Isotopic Inventory — Equilibrium Core Batch
(0.33 Total Core)

	Isotopic mass, kg			
	BOC 1	EOC 1	EOC 2	EOC 3
^{232}Th	19,407	19,225	18,991	18,833
^{234}U	35.3	72.6	105.1	124.1
^{238}U	8,383	8,250	8,077	7,959
^{233}U	1,141.5	960.2 ^(a)	807.2 ^(a)	737.4 ^(a)
^{235}U	16.8	20.4	28.9	35.3
^{239}Pu	--	80.5	135.0	155.6
^{241}Pu	--	4.1	17.5	27.0
Burnup, MWd/mtM	0	11,477	25,325	34,262

(a) Includes ^{233}Pa .

Table A-4. Fertile Region Isotopic Inventories (Applicable to Intermediate and High M/W Cores)

	<u>BOC 1</u>	<u>EOC 1</u>	<u>EOC 2</u>	<u>EOC 3</u>	<u>EOC 4</u>	<u>EOC 5</u>	<u>EOC 6</u>
<u>Total Radial Blanket, 3 in. thick</u>							
^{232}Th , kg	9976	--	--	--	--	--	--
^{233}U , kg(a)	--	38.64	68.48	91.03	107.24	118.62	126.35
Burnup, MWd/mtM	--	457	1830	3867	6396	9275	12,395
Energy, kg (mass equiv)	--	4.56	18.26	38.58	63.81	92.53	123.65
<u>Total Axial Blankets, 3 in. thickness top and bottom</u>							
^{232}Th , kg	4376	--	--	--	--	--	--
^{233}U , kg(a)	--	9.66	17.86	24.89	--	--	--
Burnup, MWd/mtM	--	148	566	1214	--	--	--
Energy, kg (mass equiv)	--	0.65	2.48	5.31	--	--	--
<u>Total Guide Tube Rods, 72 fuel assemblies</u>							
^{232}Th , kg	3365	--	--	--	--	--	--
^{233}U (a)	--	25.88	42.75	54.09	--	--	--
Burnup, MWd/mtM	--	1258	4447	8820	--	--	--
Energy, kg (mass equiv)	--	4.23	14.96	29.68	--	--	--

(a) Includes ^{233}Pa and ^{235}U .

Table A-5. Feed Enrichments ^(a)

	<u>First core,</u> <u>%</u>	<u>Equilibrium</u> <u>enrichment, %</u>
I. Reference design		
Batch 1	2.0	--
Batch 2	2.5	--
Batch 3	3.0	--
Equilibrium batch	--	3.1
II. Intermediate M/W (1.01) design		
Batch 1	2.4	--
Batch 2	3.0	--
Batch 3	3.6	--
Equilibrium batch	--	3.6
III. High M/W (1.29) design		
Batch 1	2.7	--
Batch 2	3.3	--
Batch 3	4.0	--
Equilibrium batch	--	4.0
IV. Blankets and guide tube rods		
All regions	ThO ₂	

$$(a) \text{ Enrichment} = \frac{M_{233} + M_{235}}{MU + MTh} \times 100.$$

APPENDIX B

References

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- ² P. M. Lang, "Improved Uranium Utilization — DOE Objectives and Projects," Trans. Am. Nucl. Soc., 33 (1979), p. 389.
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