PHOENIX FUEL EVALUATION
IN A MARITIME REACTOR DESIGN

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PHOENIX FUEL EVALUATION
IN A MARITIME REACTOR DESIGN

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PHOENIX FUEL EVALUATION
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K. R. Wise and U. P. Jenquin

INTRODUCTION

Analytical studies have been made⁽¹⁾ that show possible advantages of using plutonium fuel rather than uranium fuel in certain reactor systems. The relatively high ^{240}Pu content of high exposure plutonium is sometimes considered a drawback of plutonium fuel since its large capture resonance at 1 eV makes it a poison to the system. However, neutron capture by ^{240}Pu results in the creation of fissile ^{241}Pu , thus diminishing the effect of the loss of ^{239}Pu from the system. This conversion of fertile ^{240}Pu to fissile ^{241}Pu has been named the "Phoenix Effect" and plutonium fuel with a high ^{240}Pu content has become known as Phoenix fuel.

Previous studies have identified general neutronic characteristics of Phoenix fuel relative to uranium fuel in water-moderated systems as:

1. Higher fuel loadings with a given control system can yield longer core lifetimes.
2. The use of burnable poisons can further improve reactivity-lifetime behavior. Large plutonium cross sections eliminate the need to lump burnable poisons.

3. The negative ^{240}Pu Doppler coefficient can be utilized to improve core safety characteristics.
4. On the negative side, reduced control rod worth and smaller delayed neutron fraction will offset some of the control requirements reduction; these considerations must be balanced with the neutronic advantages. Also, spatial power peaking might arise due to the short neutron diffusion length in plutonium cores.⁽²⁾

Some improvement in operating characteristics of compact reactors should be possible because of the favorable nuclear benefits. Some economic advantage should also be realized with the plutonium fuel.

This study was directed at the specific reactor application of ship propulsion. Choosing this application permitted comparison with an existing enriched uranium design. As a reference design, the CNSG-II (Consolidated Nuclear Steam Generator) reactor concept⁽³⁾ developed by Babcock and Wilcox was selected.

This Phoenix study provides an initial survey of this application and compares the fuel cycle costs for the plutonium fueled cores to the costs for the enriched uranium base case core.

The results of the study are presented graphically and in tabular form in the Results Section (page 6); discussion of these results is included. A discussion of the calculational methods is included in the Appendix.

SUMMARY

Phoenix fuel was found to be competitive with the slightly enriched uranium fuel generally suggested for merchant ship propulsion. The fuel cycle costs ranged upward from 2.54 mills/shp-hr (mills/shaft horsepower-hour) for the various Phoenix-fueled systems considered, while we calculated the enriched uranium base case to have a fuel cycle cost of 2.95 mills/shp-hr.

Four results which we consider significant are:

1. Some of the Phoenix-fueled cores had power costs lower than those calculated for the enriched uranium-fueled reference design.
2. The net change in fissile plutonium inventory, contrary to what had been expected, was only slightly dependent on moderator-to-fuel ratio for the heat transfer limited cores considered.
3. The plutonium with a high ^{240}Pu content exhibited a small economic advantage when compared to the system using plutonium with a lower ^{240}Pu fraction. This advantage would be increased with lower plutonium prices for higher burnup plutonium.

4. Enrichment and plutonium composition were identified as key parameters to the design of Phoenix fueled compact H_2O reactors.

DESCRIPTION OF CNSG-II

The Consolidated Nuclear Steam Generator (CNSG) concept as developed by Babcock and Wilcox⁽³⁾ is one approach to building a reactor for merchant ship propulsion. The reactor core is located in the center of the large pressure vessel and is surrounded by the steam generator, also within the pressure vessel. This geometry is claimed to have a number of advantages, particularly well suited to shipboard applications of reactor power. The primary coolant system is completely within the pressure vessel, thus the probability of loss of coolant due to a primary system rupture is greatly reduced compared to a system utilizing a separate steam generator and connecting piping. The steam generator location surrounds the core with a large amount of energy-absorbing material which would tend to isolate the reactor in the event of a collision. The system has been simplified to ease maintenance problems and permit refueling in a short time period. The system has some self-regulating characteristics as well as having a large thermal mass, permitting some transient power mismatch between reactor and load.

The CNSG-II design was used as the base case for this study. It produces 184 MW_t (megawatts thermal) to provide 66,000 shp (shaft horsepower) for propulsion. The fuel is UO_2 with an average enrichment of

5.3% ^{235}U . The fuel is clad with zirconium that has an OD of 0.43 inches and is 17 mils thick. The rods are located in a uniform square lattice at a pitch of 0.576 inches.

ANALYSIS APPROACH

This study evaluates one application, ship propulsion, where Phoenix fuel could exhibit improved operating characteristics and cost savings. A simple substitution of the Phoenix fuel for the enriched uranium fuel was considered and then design modifications were made to improve performance.

Three design variables were examined: (1) moderator-to-fuel ratio, (2) plutonium composition, and (3) fuel pin diameter. The first variable was investigated to determine how much gain in nuclear performance could be realized by lowering the moderator-to-fuel ratio in the core. Removing moderator from the core hardens the neutron spectrum, thus increasing the ^{241}Pu production rate. Two plutonium compositions were examined to observe the effect on power cost and determine if a more detailed evaluation of plutonium composition would be worthwhile. Fuel pin diameter was varied to observe the effect on core volume and cost.

The Phoenix systems considered were fueled with plutonium oxide and used ZrO_2 as the matrix material for the fuel. For these systems the clad material was 17-mil zirconium, the same as found in the base case.

The thermal-hydraulic, nuclear, and cost calculations were done separately. The nuclear analysis parametrically described the reactor systems of interest. The nuclear analysis was designed to yield the initial and final fuel inventories required for the economic analysis. The thermal-hydraulic analysis provided the nuclear study with a range of parameters to be covered in the analysis. It also provided the economic calculations with a number of realistic reactor geometries which, when coupled with the nuclear analysis, would indicate the effect of the variables of interest.

RESULTS

Because of the complex geometry of the CNSG-II design, the reactivity calculation for this base case system was done assuming no poison rods, control rods, nor structural material in the core. A burnup calculation consistent with this assumption, was also done, to determine the isotopic content of the fuel at the end of core life. A second burnup calculation was done in which it was assumed that the core consisted of a homogeneous mixture of poison rods, control rods, structural material, and fuel rods. The results of the two calculations indicated that the assumptions of a clean core would not introduce significant error.

Reactivity and burnup calculations for various plutonium fueled systems were done assuming that the core consisted of a clean lattice of only fuel, clad, and moderator. Results were obtained for various

moderator-to-fuel ratios (V_M/V_F), plutonium compositions, fuel pin diameters, and plutonium enrichments (volume fraction of plutonium in the fuel matrix). In a Phoenix system, reactivity should be relatively constant with decreasing plutonium content within the range of interest. This behavior was observed for the systems considered.

The infinite medium, neutron multiplication factor, k_∞ , is nearly independent of enrichment as shown in Figure 1. On the enrichment scale used in these figures, 1 vol% corresponds to about 1 1/2 wt%. For tight lattices (low V_M/V_F), the change in k_∞ is small (about 5%), for the range of enrichments considered. In going to looser lattices, the variation in k_∞ with enrichment tends to become still smaller. The results presented in Figure 1 also indicate that the moderator-to-fuel volume ratio has a small effect on k_∞ (difference between solid curves), but that plutonium isotopic composition affects k_∞ significantly. Utilizing plutonium with a high percentage of ^{240}Pu should simplify control problems because there is much less excess reactivity to control.

The effects of moderator-to-fuel volume ratio and fuel rod diameter on k_∞ are shown in Figure 2. These parameters have relatively small effects on reactivity, at most a few percent. Again, the importance of the plutonium composition appears as a large effect on k_∞ .

FIGURE 1
Variation of the Infinite Medium Multiplication, k_{∞} with
Enrichment for 0.4 in. Diameter Rods

FIGURE 2
Variation of Infinite Medium Multiplication, k_{∞} with
Moderator-to-Fuel Volume Ratio and Rod Diameter for 10 V/O
Enrichment

Two group parameters for these systems at room temperature are tabulated in Table I, so that effective multiplications for any size system can be calculated with the age-diffusion equation:

$$k_{\text{eff}} = \frac{k_{\infty} e^{-B^2 \tau}}{1+B^2 L^2}$$

or the two group point reactor equation:

$$k_{\text{eff}} = \frac{\bar{\eta} f_1 (1-p)}{1+B^2 \tau} + \frac{\bar{\eta} f_2 p}{(1+B^2 \tau) (1+B^2 L^2)}$$

The core lifetime as a function of enrichment is shown in Figure 3. On examination of these curves, we note that the lifetime is essentially proportional to the enrichment. The effect of increasing the ^{240}Pu content of the plutonium is to lower the initial neutron multiplication. The rate of loss of reactivity with irradiation is comparable to the system with the same plutonium enrichment but a lower percentage ^{240}Pu in the plutonium. Hence, core lifetime for a given amount of plutonium decreases as the percentage of ^{240}Pu increases. However, on a fissile atom basis there is very little change in lifetime due to using plutonium with a high ^{240}Pu content while control problems are reduced significantly.

The effect of moderator-to-fuel volume ratio on the core lifetime is shown in Figure 4. The increase in core lifetime with decreasing moderator-to-fuel volume ratio is proportional to the corresponding increase

TABLE I
PHYSICS PARAMETERS FOR PHOENIX FUELED SYSTEMS

Rod Diameter, (in.)	Enrichment, (vol%)	Moderator-to-Fuel Volume Ratio	$\bar{n}f_1$	τ	P_1	$\bar{n}f_2$	L^2	k_∞
0.3	10*	1.000	1.2219	41.32	0.5043	1.7453	0.42	1.4858
0.3	10*	1.515	1.1414	36.70	0.6064	1.7499	0.51	1.5104
0.3	10*	2.000	1.0915	34.41	0.6674	1.7473	0.59	1.5292
0.396	3*	1.000	0.8375	42.17	0.7181	1.7353	1.23	1.4822
0.396	3*	1.515	0.7806	37.17	0.7872	1.7107	1.35	1.5128
0.396	3*	2.000	0.7470	34.68	0.8261	1.6807	1.47	1.5184
0.396	6*	1.000	1.0586	41.29	0.6092	1.7483	0.66	1.4788
0.396	6*	1.515	0.9882	36.63	0.6978	1.7423	0.78	1.5144
0.396	6*	2.000	0.9458	34.30	0.7496	1.7287	0.89	1.5326
0.396	10*	1.000	1.2296	40.34	0.5104	1.7455	0.41	1.4929
0.396	10*	1.515	1.1514	36.04	0.6121	1.7475	0.52	1.5162
0.396	10*	2.000	1.1028	33.86	0.6741	1.7411	0.62	1.5330
0.396	15*	1.000	1.3645	39.38	0.4251	1.7400	0.28	1.5241
0.396	15*	1.515	1.2813	35.42	0.5342	1.7458	0.38	1.5294
0.396	15*	2.000	1.2283	33.40	0.6034	1.7432	0.47	1.5390
0.5	10*	1.000	1.2371	39.74	0.5174	1.7452	0.42	1.5026
0.5	10*	1.515	1.1573	35.54	0.6194	1.7428	0.55	1.5199
0.5	10*	2.000	1.1121	33.53	0.6796	1.7317	0.67	1.5331
0.396	3**	1.000	0.6585	42.17	0.7063	1.5920	1.45	1.3179
0.396	6**	1.000	0.8650	41.29	0.5981	1.6098	0.76	1.3104
0.396	6**	1.515	0.8069	36.65	0.6945	1.5994	0.89	1.3352
0.396	10**	0.667	1.1282	46.08	0.4042	1.5947	0.39	1.3168
0.396	10**	1.000	1.0402	40.37	0.5055	1.6048	0.46	1.3256
0.396	10**	1.515	0.9540	36.04	0.6033	1.6053	0.58	1.3469
0.396	10**	2.000	0.9006	33.85	0.6634	1.5969	0.69	1.3626
0.396	15**	1.000	1.1857	39.41	0.4228	1.5954	0.31	1.3589

* Plutonium composition 65.5/23.3/7.7/3.5 percents of 239/240/241/242

** Plutonium composition 43/40/10/7 percents of 239/240/241/242

FIGURE 3
Variation of Effective Multiplication with Irradiation for 0.4 in. Diameter Rods
at $v_M/v_F = 1.5$

FIGURE 4
Variation of Effective Multiplication with Irradiation for 0.4 in. Diameter Rods
with 10 V/O Enrichment

in total plutonium in the system. The rate of loss of reactivity is relatively independent of the fuel rod size as shown in Figure 5. Thus, the fuel rod size affects only k_{∞} (Figure 2) and the neutron multiplication (Figure 5) at any given exposure.

The results of the physics calculations indicate that enrichment and plutonium composition are key parameters to the design of plutonium Phoenix-fueled compact H_2O reactors.

The cores considered were all DNB (departure from nucleate boiling) limited. In the thermal hydraulic analysis they were assumed to have the same peaking factors, the same coolant conditions, the same thermal power, and the same core height-to-diameter ratio.

The physical characteristics and required fuel loading of the cores selected for the fuel cycle cost studies are shown in Table II. These enrichments were required for the various geometrical arrangements selected for thermal hydraulic and economic analysis. Since the size of these systems varied, it was necessary to account for the change in k_{eff} with varying core size. Thus, decreasing the core volume and the initial neutron multiplication necessitated an increase in the total amount of plutonium in the core in order to decrease the rate of loss of reactivity. The initial plutonium content is that necessary to produce 182,000 MWd of thermal energy for each system. Since no plutonium is made, the losses by ^{241}Pu decay are negligible, and about

FIGURE 5
Variation of Effective Multiplication with Irradiation for 10 V/O Enrichment
at $V_M/V_F = 1.5$

TABLE II
REACTOR CHARACTERISTICS

Case Number	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>
Moderator-to-Fuel Ratio	1.74	1.74	1.74	.997	.736	.938	.938	.678	.678
Core Power, MW	176	176	176	176	176	176	176	176	176
Pin Size, inch	.43	.43	.43	.43	.43	.35	.35	.35	.35
Number of Pins	7008	7105	7105	7530	7659	9868	9868	9953	9953
Coolant Flow Rate, lb/hrx10 ⁻⁷	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8
Coolant Inlet Temperature, °F	517	517	517	517	517	517	517	517	517
Coolant Outlet Temperature, °F	544	544	544	544	544	544	544	544	544
Core Inlet Pressure, psi	1300	1300	1300	1300	1300	1300	1300	1300	1300
Core Pressure Drop, psi	3	3	3	16	28	25	25	46	46
Maximum Heat Flux, Btu/hr-ft ² x10 ⁻⁵	4.60	4.59	4.59	4.82	4.99	4.96	4.96	5.20	5.20
Core Volume, l	2600	2615	2615	1820	1568	1380	1380	1175	1175
Fuel Loading, kgs (initial/final)	7388/7190	370/160	560/350	410/200	430/220	440/230	710/500	460/250	770/560
Fuel Composition, % (initial/Final)									
239Pu	/0.6	65.5/21	43/19	65.5/29	65.5/33	65.5/33	43/25	65.5/36	43/27
240Pu	/0.1	23.3/46	40/49	23.3/41	23.3/38	23.3/39	40/47	23.3/37	40/46
241Pu	/0.1	7.7/23	10/21	7.7/21	7.7/21	7.7/20	10/19	7.7/20	10/18
242Pu	/	3.5/10	7/11	3.5/9	3.5/8	3.5/8	7/9	3.5/7	7/9
235U	5.3/2.1								
236U	/0.4								
238U	94.7/96.7								

20 kg of ^{242}Pu are lost by absorption; the amount of plutonium in each system at the end of life had to be 210 kg (based on 211 MeV/fission) less than the beginning of life plutonium content. This was determined by assuming the loss of ^{241}Pu by decay was negligible and that the loss rate of ^{242}Pu was the same (20 kg) for all systems. The isotopic composition of the plutonium at the end of life was estimated from the burnup calculations. The isotopic percentages are most sensitive to the fractional depletion of the plutonium.

The figure of merit used for comparing the reactor systems examined in this study was the fuel cycle cost component of power cost. The assumption has been made that any changes in other components of the power cost among the various cases will be small compared to changes in fuel cycle cost.

Each case was examined to determine a fabrication cost. The other components of the fuel cycle cost for the various reactors (Table III) are either functions of the nuclear analysis inventories or the ground rules selected for the economic analysis.

Some explanation of what is included in each of the components of the fuel cycle cost in Table III may help in understanding the table. Depletion cost is the cost charged for the initial fissionable material. Bred fuel cost, a negative cost, is the credit for fissionable material at the end of core life. Fabrication cost, as explained in the Appendix,

TABLE III
FUEL CYCLE COSTS OF REACTORS EXAMINED

<u>Case Number</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>
<u>Fuel</u>	<u>UO₂</u>	<u>PuO₂*</u>	<u>PuO₂**</u>	<u>PuO₂*</u>	<u>PuO₂*</u>	<u>PuO₂*</u>	<u>PuO₂**</u>	<u>PuO₂*</u>	<u>PuO₂**</u>
Moderator-to-Fuel Ratio	1.74	1.74	1.74	.997	.736	.938	.938	.678	.678
Pin Size (inch)	.43	.43	.43	.43	.43	.35	.35	.35	.35
Depletion Cost	1.589	1.656	1.816	1.850	1.934	1.970	2.304	2.059	2.499
Bred Fuel Cost	-.286	-.430	-.856	-.616	-.726	-.745	-1.346	-.856	-1.542
Fabrication Cost	.450	.574	.574	.638	.688	.790	.789	.792	.792
Reprocessing Cost	.388	.215	.285	.285	.302	.305	.330	.311	.557
Shipping Cost	.025	.025	.025	.025	.025	.025	.025	.025	.025
Fuel Loss	.062	.049	.068	.058	.062	.063	.091	.068	.100
Fuel Inventory Capital Cost	.518	.270	.438	.326	.379	.389	.677	.442	.645
Fuel Cycle Capital Cost	.201	.240	.187	.257	.252	.275	.203	.267	.212
Total Fuel Cycle Cost (mills/shp-hr)	2.946	2.599	2.537	2.822	2.906	3.071	3.074	3.108	3.289

* Plutonium composition 65.5/23.3/7.7/3.5 percents of 239/240/241/242

** Plutonium composition 43/40/10/7 percents of 239/240/241/242

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is the cost of fabricating the fuel. Reprocessing cost is the cost of sending the fuel through a reprocessing plant. A small percentage fuel loss in fabrication and reprocessing is also assumed. Shipping costs are the accumulated cost of shipping the fuel assemblies from the fabrication to the reactor loading point and also shipping the spent core to the reprocessing plant. No interest charges are included in the above costs. All of the interest charges related to fuel cycle cost are included under fuel inventory capital cost and fuel cycle capital cost. Fuel inventory capital cost is the interest charge on the value of the fuel over the entire cycle. Fuel cycle capital cost is the interest charge on the money that is invested in fabrication, shipping, etc., which is the non-fuel material investment.

The minimum fuel cycle cost for any of the reactors considered was 2.54 mills/shp-hr. This fuel cycle cost was for the plutonium fueled system with the same geometry as the original CNSG core and utilizing the plutonium fuel with 40% ^{240}Pu . The enriched uranium core base case fuel cycle cost was calculated to be 2.95 mills/shp-hr, which is considerably higher than the 1.84 mills/shp-hr reported by Babcock and Wilcox.⁽³⁾ The difference between these figures lies in the economic assumptions made and is discussed in the next section.

Decreasing the moderator-to-fuel ratio increases the fuel cycle cost (Figure 6). This increase is due in part to the assumption of

FIGURE 6
Variation of Fuel Cycle Cost with Moderator to Fuel Ratio

constant core height-to-diameter ratio. Decreasing the moderator-to-fuel ratio causes a decrease in core size, which with constant core shape leads to decreasing fuel pin length and a corresponding increase in the number of pins required to supply adequate heat transfer area. This study does not evaluate the alternate approach of holding core length constant and changing only core radius with moderator-to-fuel ratio.

Fissionable material consumption provides one measure of the nuclear performance and is shown as a function of moderator-to-fuel ratio (Figure 7). The flatness of this curve, i.e., the small effect changing the moderator-to-fuel ratio has on fissionable material consumption, is one of the significant results from this study. Since the curves for both plutonium compositions are flat, the figure also shows this limited dependence of fissionable material consumption on moderator-to-fuel ratio to be independent of plutonium composition.

The lower moderator-to-fuel ratio also involves an increased pressure drop across the core (Figure 8). The effect of this pressure drop was not factored into the economics but the increased pressure drop would lower plant efficiency slightly as well as possibly cause mechanical design problems.

Different pin sizes were considered; 0.4 inch and 0.35 inch outside diameter. To provide adequate heat transfer area with the smaller pins the total number of pins had to be increased. The reason for considering the smaller pins was to reduce core volume and increase the range of

FIGURE 7
Net Fissile Material used to Obtain 182,000 MWd

FIGURE 8
Core Pressure Drop Variation with Moderator to Fuel Ratio

moderator-to-fuel ratios if the effect of decreasing moderator-to-fuel ratio had been beneficial. In some applications a decrease in core volume could be of significant value. For this application the value is negligible and, in fact, the power system volume would remain about the same since the smaller cores would require larger pumps to circulate the coolant.

An optimum plutonium composition cannot be determined from this study. The plutonium composition does appear to have a slight effect on fuel cycle cost, which could be magnified if plutonium utilization in thermal reactors brings about a plutonium price schedule dependent on isotopic composition.

DISCUSSION

We found the use of Phoenix fuel to be economically competitive with an enriched uranium fuel reactor design for merchant ship propulsion. The manner in which various assumptions affect the results and identification of some areas where future developments may be anticipated are points of interest. Core design improvements may be possible and are related to plutonium's inherent advantage of being chemically separable from the previously used uranium. The plutonium can thus be distributed in the core with more freedom in this application. Potential improvements lie in grading the enrichments axially as well as radially and thus controlling the power profile, matching local power generation rates to heat transfer limits. Pressurized water reactors are generally

burnout limited only in a fairly localized region of the core. By grading the enrichment axially and radially, it would be possible to transfer more heat from those regions of the core where the departure from nucleate boiling limits were not previously encountered. For a given power this will permit a reduction in core size and cost.

Lower density fuel may also show some advantages, again dependent on the fact that the required amount of plutonium for the reactor can be dispersed in any amount of matrix material desired. A low density fuel can be expected to have fabrication cost advantages and may also have improved heat transfer characteristics.

The difference between the Babcock and Wilcox cost of 1.84 mills/shp-hr and our calculated cost of 2.95 mills/shp-hr for the base case deserves some comment. The Babcock and Wilcox economic ground rules were not included in the CNSG-II document. It was therefore necessary to find typical ground rules used for such studies and compare them to those used in this study. An evaluation study⁽⁴⁾ was made in 1966 by NUS Corporation for the Maritime Administration. This evaluation suggests that CNSG first core fuel costs with a 7.5% interest rate would run about 2.3 mills/shp-hr. In addition this evaluation used an uranium cost of \$6/lb U_3O_8 compared to our value of \$8. The particular CNSG core design evaluated for the Maritime Administration had an initial enrichment of 4.27% compared to our base case average enrichment of 5.3%. The cost difference between our base case and the cost obtained from the CNSG-II document⁽³⁾ can be attributed to the economic ground rules assumed.

Plutonium with high Pu^{240} content will probably become available in the future, particularly at that time when plutonium is being utilized in thermal reactors. When plutonium is recycled in thermal reactors it is reasonable to assume that the plutonium cost, on a per gram fissile basis, will decrease as the Pu^{239} fraction decreases. If such a cost structure does develop in the future, Phoenix fueled reactors will appear even more economically justified than determined in this study.

A decreased fuel cycle cost dependence on moderator-to-fuel ratio might have been observed if the core height rather than the core shape (height-to-diameter ratio) had been constant. Holding core height constant would have decreased the total fabrication cost since this cost was essentially on a per pin basis. The fuel inventory cost would be expected to increase, however, since the required initial fuel loading would need to be higher as the core became elongated.

Zirconium oxide as a matrix material has been test irradiated in the past⁽⁵⁾ and there are development programs⁽⁶⁾ underway to further evaluate the irradiation behavior of mixed oxide ($\text{PuO}_2\text{-UO}_2$) with ZrO_2 as as diluent.

RECOMMENDATIONS

This fuel cycle cost evaluation of Phoenix fuel in maritime applications has indicated the fuel is competitive with other designs. Future design studies for ship propulsion should consider this fuel option.

There are transient characteristics demanded of a marine system which were not investigated and which will affect reactor design. Demand following behavior for maneuvering and control requirements should be examined in future marine application studies.

We did not evaluate the economic potential of graded axial enrichment to better match the power profile nor did we evaluate low density fuel. The economic incentive for both of these potential improvements, closely linked to plutonium fuel, should be examined.

Very small cores, requiring high fissile inventories, would use highly enriched uranium if uranium fuel were used. The greatest potential for Phoenix fuels appears to be in applications where it would compete with these highly enriched uranium fuels. A study similar to this one for a lower power reactor should be conducted to verify the economic attractiveness of Phoenix fuel in low power compact reactor applications.

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APPENDIX

METHODS OF ANALYSIS

METHODS OF ANALYSIS

The theoretical methods used for the reactivity calculations are similar to those used in the analysis of plutonium fueled H_2O critical experiments.⁽⁷⁾ They consist of using the codes HRG⁽⁸⁾, THERMOS⁽⁹⁾ and TEMPEST⁽¹⁰⁾ to obtain multi-group cell homogenized cross sections for use in a one-dimensional multiplication calculation with the HFN⁽¹¹⁾ code. The unit lattice cell with reflecting boundaries is assumed to consist of three cylindrical regions--fuel, clad, and moderator.

Nonthermal effects are computed using the HRG code which solves the time independent Boltzmann equation with isotropic sources of neutrons using the P-1 approximation. Resonance absorption in the fissile and fertile isotopes is treated according to the methods developed by Adler, Hinman, and Nordheim.⁽¹²⁾ The corrections for absorber lump interaction used in the resonance computation within the HRG code, were calculated by the Dancoff and Ginsberg method.⁽¹³⁾ In determining these lattice heterogeneity corrections, the moderator was assumed to consist entirely of H_2O .

Thermal effects are computed using THERMOS which solves the space-energy Boltzmann equation. The moderator scattering kernel was computed by GAKER which is a subsidiary code to THERMOS and is based on the Nelkin model for H_2O .⁽¹⁴⁾ All other thermal scatter-transfers were based on a gas model scattering kernel for the constituent materials.

Thermal diffusion coefficients are obtained from computations using the TEMPEST code with the Wigner-Wilkins spectrum. The transport cross section calculated by TEMPEST is divided by the thermal disadvantage factor from THERMOS to account for the spatial variation of the flux. The thermal diffusion coefficient is obtained from $1/3 \Sigma_{tr}$ where Σ_{tr} is the modified transport cross section.

The multi-group cross sections for the homogenized lattice cell are edited to four energy groups. The boundaries for these groups are selected to isolate the effects of the major resonances of ^{239}Pu , ^{240}Pu , and ^{238}U . The upper energies of the groups are 10 MeV, 11.7 KeV, 2.38 eV, and 0.683 eV.

The reactor systems are assumed to be one dimensional cylinders consisting of two regions, a homogeneous core and a reflector. The reflector was assumed to be at the same temperature and of the same composition as the moderator. The four group constants for the homogenized lattice obtained from the fine structure calculations, along with four group constants for the reflector (computed using the HRG and TEMPEST, codes) are utilized in computing lattice multiplication values, k_{eff} , assuming one-dimensional multi-group diffusion theory using the computer code HFN. A radial description of the reactor system is used in the HFN code. The code computes the radial neutron distribution in each energy group. The axial neutron distribution is approximated using energy-independent transverse bucklings.

One-dimensional burnup calculations were done with the ZODIAC (2+2)⁽¹⁵⁾ code. This code incorporates the TEMPEST, HRG, HFN, and ALCHEMY⁽¹⁶⁾ codes into an automated scheme for burnup calculations. The ALCHEMY code computes the isotopic transmutations.

The cross sections used with the HRG, THERMOS, and TEMPEST codes were obtained from the BNW Master Library.⁽¹⁷⁾ Values of the 2200 meters/sec constants for the fissile isotopes are those recommended by Westcott.⁽¹⁸⁾ The data for the fission products (excluding xenon and samarium) are those reported by Nephew.⁽¹⁹⁾

The thermal hydraulic analysis was performed using the computer code REPP.⁽²⁰⁾ REPP is a code designed to simulate the thermal hydraulic characteristics of either pressurized or boiling water reactors. The code can be run in a number of ways with various reactor characteristics held constant. For this study various pin diameters were selected (rather than letting the code adjust pin diameter to meet a fuel temperature constraint). The code adjusted the number of pins in the core and the core size to meet DNB criteria and keep the core volume fractions and shape (height-to-diameter ratio) at specified values. The coolant flow rate and clad thickness were held constant at the values used in the CNSG-II design. A number of different pin sizes and moderator-to-fuel ratios were examined to establish reasonable limits for the nuclear study and then to define the cases for the economic-study.

Assumptions made in the thermal-hydraulic analysis were:

1. The plutonium fueled reactors were assumed to be made up of hexagonal shaped subassemblies with pins in a triangular array.
2. The use of zoning of the core, in that it would lower the peak-to-average power ratio, was assumed to be the same for the reference case and the Phoenix fueled cores. The power profile was assumed to be the same as found for the reference design. The reference design profile was not given in the available document⁽³⁾ but was estimated from the thermal performance.
3. Zirconium oxide was used as a matrix material for the plutonium. The cost sensitivity to substituting another material was not evaluated.
4. The core height-to-diameter ratio was held constant.
5. The coolant inlet conditions and flow rate were held constant, at the CNSG reference design values.

Fuel cycle costs were determined for each case using the computer code QUICK.⁽²¹⁾ QUICK used the initial and final inventories from the nuclear analysis and fabrication costs estimated for each case to obtain the fuel cycle cost values. All of the reactors considered were fueled to operate 990 full power days at 184 MT_t .

The economic assumptions used were:

interest rate	7.5%
U_3O_8	\$8/lb
PuO_2	\$10/gm fissile
plant factor	.80

The fabrication costs were derived on a case-by-case basis. A basic assumption was made that a plutonium bearing rod would cost 25% more to fabricate than a corresponding uranium fueled rod. Using this premise, costs were estimated for each of the major fabrication components--oxide preparation, rod fabrication, rod assembly and hardware costs. Pin costs were calculated using these relationships. In order to make the results suitable for QUICK input all costs were reduced to units of \$/kg heavy metal.

Reprocessing costs were obtained utilizing the model in the QUICK code. This model charges the fuel a daily operation charge for the reprocessing plant and adjusts the plant throughput to meet constraints involving the fuel composition and criticality limits.

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