SSA-107

April, 1978

**STUDIES OF METALLIC URANIUM FUEL
AND MIXED MODERATOR SYSTEMS
IN PRESSURIZED WATER REACTORS**

Prepared For

U.S. ARMS CONTROL AND DISARMAMENT AGENCY

Prepared By

**Southern Science Applications, Inc.
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P.O. Box 10
Dunedin, Florida 33528**

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Contract No. AC7NC122

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This report is a summary of work performed for the U.S. Arms Control and Disarmament Agency under Contract AC7NC122. The work is directly related to the Agency effort to examine potential alternative fuel cycles that might enhance uranium resource utilization, minimize plutonium production, and reduce the need for reprocessing of spent light water reactor (LWR) fuel. Two alternative fuel cycles are examined in this report: (1) the possible use of metallic fuel in LWR fuel elements, and (2) the use of a mixture of light water and heavy water as coolant/moderator in a pressurized water reactor. In both cases, calculations of a survey nature were performed to determine the basic characteristics of the two cycles — reactivity, fuel depletion isotopics, uranium resource requirements and plutonium production. Results of these calculations are presented, together with a discussion of the state-of-the-art and design considerations relative to the utilization of metallic fuel.

2.0

CONCLUSIONS

Calculations were performed for metallic fuel in pressurized water reactors and for modern PWR systems with a mixed light- and heavy-water moderator. In terms of long-term resource requirements and plutonium production (non-proliferation concerns), no significant advantage from either mode of operation is evident. However, optimization studies of fuel lattice spacing were not made, and it is possible that such optimizations could lead to improvements in the cycles.

The basic analytical tool is a zero-dimensional, multi-group, point-depletion cell calculation, employing microscopic cross-sections compiled from the ENDF/B evaluated cross-section set. This program has previously shown excellent correlation with results of a large number of critical experiments and with experimental fuel burnup data in LWRs. Spatial calculations were not made; instead, a simple average of the k_{∞} at the end, at one-third, and at two-thirds of the fuel burnup was taken as the core-average k_{∞} . For the reference PWR, this yields an end-of-cycle k_{∞} of 1.064. For other calculations, the attainable burnup or the enrichment required to achieve a given burnup was determined to give the same end-of-cycle reactivity. The approximation used here neglects spatial effects, but should give reasonable values of the equilibrium fuel burnup or enrichment requirements. All calculations were based on the geometry and mechanical design of the Indian Point-2 reactor core.

The initial conversion ratio (ICR) is a simple indicator of the potential rate of production of fissile material in an operating reactor. Although the integrated value of conversion ratio over the fuel lifetime will differ somewhat from the initial value, the difference is not large. The conversion ratio, however, must be normalized to the critical reactor core to have any real significance. Reaction rates calculated by a point-depletion program can be used to estimate the normalized ICR, assuming that thermal neutron absorption is added in the amount needed to reduce the reactor to a just-critical state. In effect, this reduces the thermal neutron group flux to achieve criticality. The fast group contribution to reactivity, k_1 , is provided by the point-depletion program, and the thermal contribution is then $1-k_1$ for a critical system. Relative group fluxes are given by $k_1/\nu \sum_f^1$ in the fast group (ϕ_1) and by $(1-k_1)/\nu \sum_f^2$ in the thermal group (ϕ_2). With these relative flux values, the initial conversion ratio is the rate of neutron capture in fertile material (U-238 and Th-232) divided by the rate of neutron absorption in the fissile material (U-233 and U-235) initially present in the fuel, as follows:

$$ICR = \frac{\sum_{C,1}^{28} \phi_1 + \sum_{C,2}^{28} \phi_2 + \sum_{C,1}^{02} \phi_1 + \sum_{C,2}^{02} \phi_2}{\sum_{A,1}^{23} \phi_1 + \sum_{A,2}^{23} \phi_2 + \sum_{A,1}^{25} \phi_1 + \sum_{A,2}^{25} \phi_2}$$

where Σ is the macroscopic capture or absorption cross-section in the fast and thermal groups for U-238 (Σ_C^{28}), Thorium-232 (Σ_C^{02}), U-233 (Σ_A^{23}), and U-235 (Σ_A^{25}). Although the equation is an approximation, an independent check calculation for an exactly-critical reactor confirmed the validity of this normalization process. The integrated conversion ratio, defined as the time integral rate of fissile material production divided by the time integral rate of fissile material destruction, is computed by the point-depletion program and printed out at each time step.

In computing resource utilization, a 75% plant factor was assumed for all cases, and the annual requirements were normalized to a 1000 Mw(e) plant. The initial loading requirement was based upon a simple mixture of one-third of the core at the equilibrium loading, one-third at the U-235 loading corresponding to one-third of the final burnup, and one-third at the U-235 loading corresponding to two-thirds of the final burnup. This approximation, which is equal to twice the equilibrium annual loading plus the equilibrium discharge, neglects the perturbation due to the actual method used in the approach to equilibrium. The 30-year resource requirement is then 29 times the annual requirement plus the initial loading. Total plutonium production is 30 times the annual production rate. It should be noted again that the values obtained are those for a 1000 Mw(e) plant operating for 30 years at an average plant factor of 75%.

4.0 METALLIC FUEL

4.1 Summary

A review of the available information on metallic uranium fuel in LWR systems indicates a considerable level of confidence that such fuel could be satisfactorily utilized, although a significant R&D effort would be necessary to confirm the anticipated performance. A conceptual design of a metallic fuel element was developed for comparison with a conventional oxide fuel design. The metallic fuel element is dimensionally identical with the ceramic element and, consequently, could be a direct replacement. To accommodate fuel swelling during irradiation, a central axial void is provided, initially amounting to approximately 10% of the fuel volume, which is expected to be adequate to allow a maximum fuel exposure of about 40,000 Mwd/mtU. Simple calculations of the thermal performance of the metallic fuel design indicate acceptable temperatures: for example, the average fuel temperature of 675°F provides an adequate margin from the alpha-beta transition temperature in uranium.

Although there is a large body of available information on uranium and uranium with small amounts of alloying material, the data on uranium-thorium metallic fuel is limited. Generally, uranium-thorium should be superior to uranium in irradiation behavior, thermal conductivity, and compatibility with water. In the absence of more specific information on uranium-thorium alloys of higher thorium content (40-80%), the same conceptual design fuel element used for metallic uranium fuel was adopted for the metallic uranium-thorium fuel case.

Despite the encouraging information on the feasibility of using metallic fuel in PWR's, fuel utilization calculations did not reveal any significant advantage over the conventional oxide fuel cycles. Because uranium metal fuel is approximately twice as dense as oxide fuel (i.e., one-half the specific power), irradiating it for the same cycle lifetime as oxide fuel (in years) would result in the metal fuel attaining approximately half the fuel burnup of the oxide (17,000 Mwd/mtU in the metal). Conversely, if the metal fuel reached the same burnup as oxide fuel (~ 35,000 Mwd/mtU average), the cycle lifetime would be

approximately twice as long (~ 7.7 years at a 75% plant factor).

Results of the fuel utilization calculations are summarized in Table 1. Figure 1 is a plot of the k_{∞} variation and Fig. 2 shows graphically the initial and integrated conversion ratios as a function of thorium content. Comparison with the reference oxide fuel utilization (also shown in Table 1) shows that the 30-year U_3O_8 requirement is higher for the metallic fuel cycles, using either equal cycle time or equal burnup. Similarly, the plutonium production is also higher. With denatured fuel of various thorium contents (throwaway fuel cycle), the 30-year U_3O_8 requirements are higher in all cases than for oxide fuel, although the plutonium production is reduced with increasing thorium loading. Consequently, it is concluded that, from the standpoint of uranium resource utilization and plutonium production, the metallic fuel cycle does not have any advantage over the oxide fuel cycle currently being used. Table 2 shows the effect of a tandem fuel cycle (recovery of spent fuel and further irradiation in a CANDU type reactor) on metallic fuel utilization.* No advantage over the corresponding tandem oxide fuel cycle is noted. Figure 3 shows the k_{∞} variation with burnup for the metallic tandem cases. Details of the isotopic composition as a function of fuel burnup for the cases considered are shown in Figs. 4 through 13.

4.2 Physical Behavior of Metal Fuel

4.2.1 General

Early in the era of commercial nuclear power, a large fraction of the operating nuclear power plants used metallic uranium as reactor fuel. However, the large-scale use of metal fuel in those days (late 1950s - early 1960s) is misleading, for the metal fuel was used in a single class

*For further information on the tandem fuel cycle, see the companion report, Studies of Alternative Nuclear Technologies SSA-106, April, 1978.

Table 1 FUEL UTILIZATION IN PWR SYSTEMS WITH METALLIC FUEL

	Reference PWR Cycle	U-Cycle Throwaway Equal Time	U-Cycle Throwaway Equal Burnup	Denatured Fuel Throwaway 77% Th	Denatured U-Recycle 77% Th	Denatured Fuel Throwaway 40% Th	Denatured Fuel Throwaway 60% Th	Denatured Fuel Throwaway 80% Th
Discharge Burnup, Mwd/kg	33.0	17.0	34.6	35.3	35.3	35.3	35.3	35.3
Equilibrium Fissile Enrichment, % HM	3.2	2.15	3.525	4.6 (20% in U)	4.6 (20% in U)	4.7 (7.8% in U)	4.7 (11.8% in U)	4.4 (22.2% in U)
Initial Loading Requirements, kg/GWe-yr								
U-235	1892	2625	2013	2641	2641	2728	2739	2532
ST U ₃ O ₈	451	606	483	666	666	677	684	638
Equilibrium Loading, kg/GWe-yr								
U-235	839	1091	882	1127	802	1148	1157	1087
ST U ₃ O ₈	200	252	211	284	202	285	289	274
U-233					329			
Equilibrium Discharge, kg/GWe-yr, average								
U-235	214	443	249	387	301	432	425	358
U-233				245	335	161	208	253
Fissile Pu	164	302	221	75	75	163	117	66
U-235 discharge enrichment	.858	.897	1.05	7.86	6.08	3.15	4.74	8.46
Annual Net Requirements, kg/GWe-yr, average								
U-235	625	648	633	740	501	716	732	729
ST U ₃ O ₈		252	211	284	128	285	289	274
U-233				-245		-161	-208	-253
Fissile Pu		-302	-221	-75	-75	-163	-117	-66
Enrichment requirements, SWU	1.24x10 ⁵	1.27x10 ⁵	1.37x10 ⁵	2.58x10 ⁵	1.10x10 ⁵	2.3x10 ⁵	2.4x10 ⁵	2.51x10 ⁵
Enrichment of Makeup U, % (blending)		2.15	3.52	20	63.5	7.8	11.8	22.2
30-Year Requirements								
ST U ₃ O ₈	6251	7914	6602	8902	4378	8942	9065	8584
kg U-233				-7350		4830	6240	7590
kg Fissile Pu	4920	-9060	-6630	-2250	-2250	4890	3510	1980

57.5 wt

Table 2 FUEL UTILIZATION IN PWR/CANDU SYSTEMS WITH METALLIC FUEL IN THE TANDEM FUEL CYCLE (URANIUM CYCLE)

		U-Cycle Throwaway Equal Burnup	U-Cycle Throwaway Equal Time	
Discharge Burnup, Mwd/kg		(34.6/13.6) 48.2	(17/13.6) 30.6	
Equilibrium Fissile Enrichment, % HM		3.525	2.15	
Initial Loading Requirements, kg/GWe-yr				
U-235		1455	1458	
ST U ₃ O ₈		347	337	
Equilibrium Loading, kg/GWe-yr				
U-235		633	606	
ST U ₃ O ₈		151	140	
Equilibrium Discharge, kg/GWe-yr, average				
U-235		76	80	
Fissile Pu		109	155	
U-235 discharge enrichment		.44	.29	
Annual Net Requirements, kg/GWe-yr, average				
U-235		557	526	
Enrichment require- ments, SWU		9.8x10 ⁴	7.1x10 ⁴	
30-Year Requirements				
ST U ₃ O ₈		4726	4397	
kg Fissile Pu		3270	4650	

of reactor typical of those employed by the United Kingdom and France in their early nuclear programs. These are Magnox reactors, which operate to the modest fuel exposures obtainable through the use of natural uranium fuel in a graphite-moderated, CO₂-cooled system. In these systems, reactivity lifetime remains the limiting factor on fuel residence time, although work on metal fuels is required to obtain a metallurgical lifetime consistent with the achievable reactivity lifetime.

The following paragraphs contain discussions of the possible use of metallic fuel in commercial light water reactors. The economic operation of these reactor systems depends upon obtaining fuel exposures significantly greater than those obtained from metal fuel in other applications in thermal-spectrum reactors. Generally, for any given reactor system, an economically-optimum fuel exposure exists where the overall cost is a minimum: this optimum burnup is different for metal and ceramic fuel. Since fabrication costs for metal fuel are less than those for ceramic fuel, the range of economically-optimum burnups is lower for the metal fuel.

4.2.2 Uranium Metal Fuel

The state-of-the-art for metallic uranium fuel elements has improved markedly in the last ten or fifteen years. The basic fuel is uranium metal containing very small quantities of a few alloying elements (maximum concentration less than 1,000 ppm). Since the density of the fuel is very high, there is significantly less parasitic neutron capture than in less dense fuels. The material is compatible with zirconium alloys, and its thermal properties are very good.

For a time, the high corrosion rate of uranium metal in high temperature water was considered a potential problem of sufficient magnitude to cast doubt on metallic uranium utilization in LWRs. However, it presently appears that the material could be used without incurring unacceptable safety risks or operating inconvenience, although more corrosion data and systems analyses are necessary.

4.2.3

Design Considerations

The overriding consideration in the development of a metallic uranium fuel element is the attainment of a relatively-long fuel lifetime. In the case of metal fuel, lifetime is limited by distortion (swelling) of the fuel, leading to changes in thermal and neutronic performance of fuel assemblies through geometry shifts. Ultimately, fuel swelling results in diametrical growth that causes a circumferential strain in excess of the failure limit of the cladding. The problem, then, is to develop a fuel design capable of obtaining high fuel exposures with minimum risk of failure.

To a first approximation, a metal-fuel assembly contains twice the mass of uranium that the same size ceramic fuel assembly does. Consequently, in the same reactor, and with a specific power of approximately one-half that of the oxide fuel, a metal-fuel assembly operating to the same burnup (in Mwd/mtU) will remain in the core twice as long. Conversely, for operation for a given number of days, the metal fuel burnup in Mwd/mtU will be about one-half that of the ceramic fuel.

Early work on uranium growth had attributed the swelling to cavities formed by the agglomeration of fission gases at unexpectedly-low temperatures. Work at Savannah River showed, however, that fission-gas bubbles contributed only a small fraction of the volume increase. These investigations led, in 1961, to the recognition of cavitation swelling in uranium.¹ Subsequent investigation at Hanford confirmed the Savannah River observations and revealed another mode of swelling at somewhat higher temperatures — small pores aligned in planar arrays.

Both before and after the determination of the presently-accepted reasons for gross uranium swelling, large amounts of work had been done on alloying and other techniques to control the swelling. Some success in reducing the innate swelling of pure uranium in the temperature range necessary for light water reactor use had been achieved by alloying with small amounts of other metals, as mentioned above. In addition, high-strength cladding and high system pressure were used as external restraints on the volumetric swelling of fuel rods. However, the improvements obtainable through the use of the alloying and restraint techniques were not

sufficient to obtain the burnup necessary for economic use in light water reactors. Subsequently, work in the United States² and in the Soviet Union³ was directed toward determining the beneficial effects of a central void to accommodate fuel swelling internally, without increasing the external dimensions of the element. It is the combination of alloying constituents, external restraint, and a central, longitudinal hole in each fuel element that constitutes the basis for the fuel design evaluated here for light water reactors.

Before proceeding to describe the fuel element design and the calculations for the reference light water reactor, it should be noted that the alloying elements are assumed to have no effect on the neutronic behavior of the fuel (because of the properties of the alloying elements themselves and the very small quantities present).

4.3 PWR Fuel Design

4.3.3.1 Uranium Metal Fuel

As in past investigations, the light water reactor chosen as the basis for comparative calculations is the Indian Point-2 pressurized water reactor. The most logical metal fuel design was deemed to be one in which the ceramic fuel elements are replaced by dimensionally-identical metal fuel elements: that is, the cladding OD, rod spacing, etc. are the same as those used in the ceramic fuel design.

Although the contract specified that heat transfer calculations would not be performed, it was necessary to carry out a few very simple, brief, hand calculations to estimate the temperatures that would be encountered in a metal fuel design. In order to perform the physics calculations, one must know the average fuel temperature in the reactor, and it is also necessary to determine that a proposed design insures that the maximum temperature reached in the uranium metal gives a comfortable margin below the alpha-beta transition temperature in the uranium.

It appears that the ceramic fuel rods in Indian Point-2 could be replaced directly by metallic uranium rods of the design shown in Fig. 14. The Zircaloy-clad metal rod is

formed by the coextrusion process and has a 10% axial void. Reactor coolant temperatures are those that occur in Indian Point-2 with ceramic fuel. The average uranium metal temperature is 675°F, with an average cladding temperature of 625°F. The margin from the alpha-beta transition temperature in the uranium is substantially larger than in certain other reactor designs, so phase transformation does not appear to be a problem.

The 10% axial void was selected on the basis of the data given in Ref. 2. According to that reference, if 2% strain is allowed in Zircaloy cladding, a rod with a 10% axial void would be expected to survive exposures to a maximum of about 40,000 Mwd/mtU. The Russians claim³ that experiments show a fuel element with 5% void can attain a burnup in the range of 15,000 to 20,000 Mwd/mtU, and that the allowable burnup rises with an increase in the void volume. Although these statements are encouraging, irradiation of metallic uranium fuel elements to fuel exposures typical of those reached in present light water reactors would require additional experimental verification before the elements could be considered for commercial use.

It presently appears that the irradiation lifetime of metallic uranium fuel could be sufficient to maintain the refueling schedules used in commercial plants, which would result in a metallic fuel burnup of roughly one-half that reached in ceramic fuel. However, if a refueling interval twice that presently used in PWRs were anticipated (that is, the same overall fuel exposures as those presently obtained with ceramic fuel), additional tests would have to be carried out.

The physics calculations for Indian Point-2 were performed on the assumption that the fuel would reach an exposure of slightly over 34,000 Mwd/mtU, the value used for past calculations with ceramic fuel, or slightly over 17,000 Mwd/mtU, half the exposure of the ceramic fuel (same residence time). In order to attain these burnups, the uranium-metal fuel required initial enrichments of approximately 3.5% and 2.15% U-235, respectively.

4.3.3.2 Mixed Uranium-Thorium Metal Fuel

The work on thorium metal has not been extensive in comparison

to that on uranium metal. As far as irradiation behavior and compatibility with water are concerned, metallic thorium appears to be the superior material. It has excellent thermal conductivity (better than uranium), and its mechanical and physical properties are strongly influenced by small amounts of alloying elements. For example, the addition of carbon in amounts ranging from 0.03 to 0.2 w/o can increase markedly the yield stress, melting point, and corrosion resistance of pure thorium. Figure 15 gives the thermal conductivity for thorium and for uranium metal as a function of temperature.

The uranium-thorium alloys that have been the subject of the greatest investigation to date have actually been thorium alloys containing up to about 10 w/o uranium. For the situations considered in this study, the fuel material will be, say, 60 w/o uranium — 40 w/o thorium. The scarcity of data on alloys of interest here has resulted in the requirement that the mixed-metal fuel design be assumed to be the same as the uranium-metal design.

The simple heat-transfer calculations performed for the uranium metal fuel show that the fuel radial temperature gradient, used to determine the core-wise average temperature, is quite flat. Increases in thermal conductivity of the fuel will only flatten the gradient more, while reasonable decreases in conductivity will not have a great effect. Consequently, the average temperatures specified for the uranium metal fuel case were also chosen for the uranium-thorium fuel case.

The density of the alloy varies with composition, as shown in the following tabulation, and shown graphically in Fig. 16.

<u>w/o U in U-Th</u>	<u>Alloy density, grams/cm³</u>
20	12.6
30	13.1
40	13.7
50	14.4
60	15.1
70	15.9
80	16.9

The number densities used in the physics calculations were based on the appropriate alloy densities selected from the

values above. For the mixed U-Th metal alloys, the equilibrium U-235 enrichment was selected to give approximately the same burnup (Mwd/mtU) as the reference PWR with uranium oxide fuel.

The mixed moderator concept is a variant of the spectral shift reactor concept, in which a single constant H_2O - D_2O mixture is used in contrast to a continuously-variable heavy water content. D_2O in the moderator of a current PWR results in a large loss in reactivity, necessitating an increase in enrichment to accomplish a typical fuel burnup. A second fuel irradiation cycle in light water only is then required to salvage the potential reactivity remaining in the fuel and to consume a larger fraction of the U-235 initially loaded.

The first three cases considered, summarized in Table 3, show that, to achieve a design fuel burnup of 34,000 Mwd/mtU in the first cycle, the enrichment increases as the D_2O content increases and the duration (burnup) of the second cycle increases. Below about 40% D_2O , there is insufficient reactivity to support a meaningful second cycle. Resource requirements and plutonium production decrease for the higher- D_2O -content cases. The last two cases in Table 1 illustrate what could be accomplished if U-233 were available as feed material. It should be noted that the calculations reported here assume the reference PWR lattice spacing (i.e., same moderator-to-fuel volume ratio) and do not consider possible improvements that might be accomplished by optimizing the lattice spacing.

For a balanced cycle, the D_2O content and enrichment in the first irradiation cycle (mixed-moderator) at a burnup of ~ 34,000 Mwd/mtU should be adjusted so that the reactivity is sufficient to achieve the same burnup in the second cycle (light water only). On the basis of a number of calculations, searching for the appropriate D_2O /enrichment combination, it was found that 65% D_2O and an initial enrichment of 7.24 percent U-235 would give a burnup of 34,310 Mwd/mtU in the first cycle and ~ 45,000 Mwd/mtU in the second cycle. A slightly lower D_2O content and enrichment would tend to more nearly equalize the burnups, but further iterative calculations were not made. Calculations of the 30-year fuel utilization were made on the basis of 65% D_2O content and 7.24% U-235, and the results are summarized in Table 3.

Compared to the reference PWR cycle, the mixed moderator concept indicates approximately the same 30-year average

Table 3 FUEL UTILIZATION FOR THE FIXED-MIXED MODERATOR CONCEPT

	65% D ₂ O U-Cycle Throwaway	40% D ₂ O U-Cycle Throwaway	50% D ₂ O U-Cycle Throwaway	65% D ₂ O 60% Th U-233 Feed U-Recycle	65% D ₂ O 60% Th U-233 Feed U-Recycle			
Discharge Burnup, Mwd/kg	80 (34.3/45.7)	43.1 (34.3/8.8)	51.5 (34.3/17.2)	69.5 (34.8/34.7)	69.5 (34.7/34.8)			
Equilibrium Fissile Enrichment, % HM	7.24	5.0	5.69	N+12% U-233	20% U235 5.3% U233			
Cycle time, yrs	8.82	4.76	5.68	7.57	7.57			
Initial conv. ratio	.569							
Integrated conv. ratio	.586							
Initial Loading Requirements, kg/GWe-yr								
U-235	1713	2318	2185	85	2216			
ST U ₃ O ₈	424	566	535	14	558			
U-233				1450	757			
Equilibrium Loading, kg/GWe-yr								
U-235	783	1003	955	31	933			
ST U ₃ O ₈	194	245	234	6	235			
U-233				597	261			
Equilibrium Discharge, kg/GWe-yr, average								
U-235	147	312	275	23	350			
U-233				256	235			
Fissile Pu	133	200	186	84	98			
U-235 discharge enrichment	1.54	1.66	1.77	.53 (5.98% U233)	8.551 (5.75% U233)			
Annual Net Requirements, kg/GWe-yr, average								
U-235 consumed	636	691	680	8 (341 U-233)	583 (26 U-233)			
Enrichment require- ments, SWU	1.53x10 ⁵	1.78x10 ⁵	1.76x10 ⁵		1.46x10 ⁵			
30-Year Requirements								
ST U ₃ O ₈	6060	7671	7321	negligible	4845			
kg Fissile Pu	-3990	-6000	-5580	-2520	-2940			
kg U-233				11,339	1511			

uranium resource requirement, but produces somewhat less plutonium. Other combinations of D₂O and U-235 enrichments, for lower fuel burnups, tend to approach the performance of the spectral shift concept. Calculations at optimized lattice spacings were not made: conceivably, such calculations could show an improvement in the resource utilization for the mixed-moderator system. However, since no obvious advantage exists, and because it is unlikely that a total fuel burnup of around 80,000 Mwd/mtU will be technically feasible in the near future, the mixed moderator concept does not appear to warrant further consideration at this time.

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Fig. 1 k_{∞} with metallic fuel as a function of fuel burnup in the denatured fuel cycle.

Fig. 2 Conversion Ratio for metallic systems in the denatured fuel cycle.

Fig. 3 k_{∞} variation with metallic fuel in current PWRs, with tandem fuel cycle.

Fig. 4 Specific mass of U-235 (kg U-235/initial metric ton heavy metal) as a function of fuel burnup.

Fig. 5 U-233 production in denatured metallic fuel as a function of fuel burnup.

Fig. 6 Isotopic composition, uranium metallic fuel, tandem cycle 2.15% U-235 initial enrichment.

Fig. 7 Plutonium isotopic composition, uranium metallic fuel, tandem cycle 2.15% U-235 initial enrichment.

Fig. 8 Isotopic composition, uranium metallic fuel, 3.52% initial enrichment.

Fig. 9 Plutonium isotopic composition as a function of fuel burnup, 3.52% initial U-235 enrichment, uranium cycle

Fig. 10 Plutonium isotopic composition as a function of fuel burnup, 7.8% initial U-235 enrichment, denatured fuel cycle, 40% thorium.

Fig. 11 Plutonium isotopic composition as a function of fuel burnup, 11.8% U-235 initial enrichment, denatured fuel cycle, 60% thorium.

Fig. 12 Plutonium isotopic composition as a function of fuel burnup, 20% initial U-235 enrichment, denatured fuel cycle, 77% thorium.

Fig. 13 Plutonium isotopic composition as a function of fuel burnup, 22.2% initial U-235 enrichment, denatured fuel cycle, 80% thorium.

Fig. 14 Cross-section of uranium metal fuel rod for Indian Point-2 PWR.

Fig. 15 Thermal conductivity of thorium and uranium metal.

Fig. 16 U-Th metal density as a function of alloy composition.