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***POTENTIAL UTILIZATION OF SRS
URANIUM WITH HIGH U-236
CONTENT AS COMMERCIAL LWR
FUEL (Study of Neutronic and Safety
Characteristics) (U)***

D. Biswas

Publication Date: July, 1994

**Westinghouse Savannah River Company
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**POTENTIAL UTILIZATION OF SRS URANIUM WITH HIGH U-236
CONTENT AS COMMERCIAL LWR FUEL (U)
(Study of Neutronic and Safety Characteristics)**

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Abstract

The neutronics and safety characteristics of a PWR core loaded with fuel made from SRS recovered uranium have been analyzed. These results are compared to those of a standard commercial fuel loaded core. For both cases, equilibrium cycle three-dimensional core models have been developed to accurately quantify the fuel cycle penalty factor associated with high U-236 content in SRS derived fuel. A 0.28 wt% overenrichment is required to compensate for each 1.0 wt% of U-236 in SRS derived fuel in order to match the cycle length. Doppler and moderator coefficients have become slightly more negative with the increase in U-236 content along with a slight reduction of control rod worth, boron worth and shutdown margin. These changes will not impact the safety or operation of a commercial nuclear plant. Thus, the feasibility of using SRS recovered uranium in the form of commercial nuclear fuel has been established from the neutronic and safety point of view. The real issue is the radiological problem related to SRS derived fuel and its effect on worker safety.

1.0 Introduction

A large amount of highly enriched uranyl nitrate solution is stored in the H-Area. In addition, large quantities of highly enriched irradiated SRS (Savannah River Site) fuel and off-site fuels are also available for processing. A potential exists for enriched uranium recovered from processing of stored uranyl nitrate solution and SRS/off-site fuels to be blended down to about 5.0 wt% U-235 and utilized in fuel for commercial nuclear power plants (PWRs as well as BWRs). However, the recycle of uranium in the DOE reactor fuel cycle has resulted in the enrichments of U-234 and especially of U-236. The U-234 isotopic content in the blended uranium will be increased by a factor of two with respect to commercial fuel, and the U-236 isotopic fraction will be increased by more than 100 times over the content in commercial fuel.

The U-236 isotope acts as a neutron absorber and causes a fuel cycle penalty. Several earlier studies^{2,4,5} were done to determine the fuel cycle effect of U-236. They were essentially limited to calculating the fuel cycle penalty factor and associated cost without mentioning the effect of U-236 on reactor safety parameters. Different values of the penalty factor were also reported. The present study determines accurately the fuel cycle penalty using the latest Westinghouse design

codes and validated cross section sets. It concentrates on studying the neutronic and safety characteristics of a typical PWR using fuel derived from SRS recovered uranium with very high U-236 content. Important safety parameters such as Doppler and moderator coefficients as well as boron and control rod worths are calculated and compared with a reference core using commercial fuel of equivalent reactivity to see if there is any safety problem in using SRS uranium in a commercial nuclear plant.

2.0 Isotopic Composition of SRS Uranium

Table 1 gives the isotopic composition of fuels used in the subsequent neutronic calculations¹. SRS currently has stocks of uranium in the form of uranyl nitrate solution from the dissolution of highly enriched (> 60 wt% U-235) irradiated U/Al fuels, and irradiated unprocessed fuels. The U-236 content of these materials has become very high, due to repeated recycling of the fuel material. The U-236 isotope will increase in the discharged fuel due to neutron absorption in U-235 in a reactor. If these materials were to be dissolved and blended down isotopically to 5.0 wt% U-235, the result would be the isotopic composition shown in Table 1. This will be referred to henceforth as **SRS Derived Fuel**. Also shown in Table 1 for comparison are the uranium isotopic contents of natural uranium and commercial fuel enriched to 5.0 wt% U-235. Natural uranium contains a small amount of U-234. The content of U-234 in commercial fuel increases during the enrichment process. The isotopic fraction of U-234 quoted in Table 1 is an average value corresponding to 5 wt% enrichment. In reality, the U-234 content will fluctuate in the as-built fuel assemblies and is mainly due to contamination with reprocessed uranium in an enrichment plant. Similarly, even though there is no U-236 in natural uranium, the average value of U-236 is about 0.015 wt% in commercial fuel corresponding to an enrichment of 5 wt%. Contamination of natural UF₆ with enriched reprocessed UF₆ during the enrichment process has long been observed. When the recycled fuel is mixed with fresh fuel at the time of enrichment, U-236 is distributed throughout the enriched fuel, the effective concentration being a function of the ratio in which the recycled and fresh fuels are mixed. During the enrichment process, U-236 is also enriched² along with U-235, the relative increase in the enrichment of U-236 being ~80% of the increase in the enrichment of U-235.

Table - 1
Isotopic Compositions*

Isotope	Natural Uranium (wt%)	Commercial Fuel (wt%)	SRS Derived Fuel (wt%)
U-234	0.0055	0.0473	0.097
U-235	0.72	5.00	5.00
U-236	0.00	0.0146	2.00

*balance is U-238

Note that for SRS recovered uranium, the isotopic compositions of the uranyl nitrate solution stored presently in the H-Area of SRS, the irradiated fuel materials in the SRS Reactor Disassembly Basins and off-site fuels are somewhat different. For a conservative estimate of the fuel cycle penalty associated with high U-236 content in SRS uranium, the isotopics of the stored uranyl nitrate solution which has the highest value of U-236 content are chosen for this study and are shown in Table 1. The average isotopic composition of uranium in existing fuels, when blended down to 5.0 wt% enrichment, are 0.11 wt% and 1.7 wt% for U-234 and U-236 respectively. And the U-236 content of different off-site fuels vary to a large extent and are generally lower than that of existing fuel assemblies or the stored uranyl nitrate solution (Appendix 1).

The specifications for isotopic inventories in commercial fuel as well as in SRS derived fuel are given in Table 2. The isotopic contents of both U-234 and U-236 have exceeded the limiting values specified in the ASTM standard³ for fuel enriched to less than 5.0 wt% U-235.

Table - 2
Specifications for Enriched UF₆

Isotope	Commercial Fuel ($\mu\text{g/g U-235}$)	SRS Derived Fuel ($\mu\text{g/g U-235}$)	ASTM Limit ($\mu\text{g/g U-235}$)
U-232	<0.002	0.0016	0.002
U-234	9,460	19,400	10,000
U-236	2,920	400,000	5,000

It may be noted that the specifications used at nuclear fuel fabrication plants usually correspond to the ASTM standards⁷. The fraction of U-234 in commercial fuel usually approaches the ASTM limit which is mainly set for radiological protection. As seen from Table 2, the U-234 content in SRS derived fuel is twice the limiting value and may create a problem in a fuel fabrication plant unless there are some changes in equipment and procedures to minimize dose rate. The extremely high level of U-236 in SRS derived fuel will diminish the reactivity of SRS derived fuel for the same enrichment and is the major area of concern for potential customers in terms of fuel price.

3.0 Core Design Philosophy

A primary objective of this study is to determine the fuel cycle penalty in using PWR fuel fabricated from SRS uranium with very high U-236 content. In other words, the objective is to calculate the reduction in cycle length when the commercial fuel is replaced with SRS derived fuel having the same enrichment and to determine the equivalent enrichment using commercial fuel to have identical cycle length.

Moderator and Doppler coefficients will be compared between the two equivalent cases to see the impact on reactor safety. Control rod worths will also be calculated to make sure that adequate shutdown margin exists in a core using SRS derived fuel.

To accomplish the objective, a representative four-loop Westinghouse plant is chosen as the reference plant. A conceptual equilibrium cycle core loading pattern is then developed using SRS derived fuel as well as commercial fuel. An equilibrium cycle has more generic representation of core behavior and therefore the fuel cycle penalty calculations will be more accurate.

The reactor core consists of 193 fuel assemblies and employs a standard out-in loading pattern for the equilibrium cycle. The Westinghouse standard 17x17 Vantage 5-H fuel assembly is chosen for this study. The maximum permissible enrichment, 5.0 wt%, is used for all regions. The design model feeds and discharges 76 assemblies (one-third core) in the equilibrium cycle. Pyrex burnable absorbers are used to reduce power peaking and soluble boron in the core. There are 53 control rod cluster assemblies (RCCA) made of Silver-Indium-Cadmium in the reference design.

An acceptable loading pattern is obtained by shuffling the different feed, once burnt and twice burnt assemblies, and changing the burnable absorber configurations and locations till all peaking factors are within the design limits and the moderator temperature coefficients are negative at all operating conditions. All calculations are performed with the standard Westinghouse proprietary design codes with three dimensional core models. Three models are developed having the same loading pattern. They are as follows:

a. CASE 1 (reference case)-Equilibrium cycle model with SRS derived fuel (enrichment - 5.00 wt%).

b. CASE 2 (equivalent case)-Equilibrium cycle model with commercial fuel (enrichment - 4.44 wt%).

This enrichment is obtained after several iterations such that the cycle length for the case 2 is identical to that in the case 1. This model is equivalent to case 1.

c. CASE 3 -Equilibrium cycle model with commercial fuel (enrichment - 5.00 wt%).

This model is needed to find the decrease in cycle length compared to case 1.

4.0 Core Description

Table 3 gives important core design and operating characteristics for the typical 4-loop plant chosen for the analyses. Figure 1 shows the control rod locations, the burnable absorber locations, and three regions of the loading pattern for the reference case. The loading patterns are identical for all three cases. An "out-in" loading pattern is used, i.e., one that loads the fresh fuel on the core periphery and then shuffles the assemblies to interior locations in subsequent cycles. A small number of glass (pyrex) burnable absorbers are needed to reduce power peaking.

Table - 3
Reference Four-Loop Plant Design and Operating Parameters

Number of fuel Assemblies	193	Number of Control Rods	53
Core Power, (MWth)	3565	Number of Feed Assemblies	76
Lattice Configuration	17x17	Linear Power Density (kW/ft)	5.68
Fuel Rod OD, (inch)	0.374	Fuel Rods per Assembly	264
Cladding Material	Zircaloy-4	Hot Zero Power Temperature, (°F)	557.0
Fuel Pellet OD, (inch)	0.3225	Hot Full Power Temperature (°F)	589.6
Active Fuel Height, (inch)	144	System Pressure, (psia)	2250
Core Loading, MTM	89	Radial Peaking Factor Limit	1.53

5.0 Core Characteristics

The following sections describe important core characteristics of the equilibrium cycle for the three different cases. All calculations are documented in the calculation note⁸.

a) Fuel Cycle Penalty

Table 4 shows the results of the fuel cycle penalty calculations. A 0.56 wt% overenrichment of SRS derived fuel is required to match the cycle length of the equivalent core using commercial fuel mostly due to higher loading of U-236. In other words, a cycle length penalty of 2000 MWD/MTU or 50 equivalent full power days (EFPD) will be incurred by using SRS derived fuel.

Table -4
Fuel Cycle Penalty

PARAMETER	CASE 1 (reference case)	CASE 2 (equivalent case)	CASE 3
Fuel Type	SRS Derived Fuel	Commercial Fuel	Commercial Fuel
Enrichment (wt%)	5.00	4.44	5.00
PPM (BOL, HFP)	1990	1823	2083
PPM (EOL, HFP)	~10	~10	~10
Cycle Length (MWD/MTU)	16,200	16,200	18,200
Penalty in Cycle Length (MWD/MTU)	2000	-	-
Penalty in Enrichment (wt%)	0.56	-	-

A relationship between U-236 content and the corresponding increase in U-235 enrichment required to compensate for it can be expressed as

$$\Delta E = C * F(U-236)$$

where

ΔE is the overenrichment in wt%,

$F(U-236)$ is the U-236 fraction in wt% at the beginning of cycle and

C is the penalty factor.

In the present case, $F(U-236)$ is 2.0 wt%, and therefore the penalty factor (C) is $0.56/2.0 = 0.28$. The penalty factor will vary slightly depending on type of reactor, initial enrichment and U-236 fraction in the fuel (mainly depends on spectrum of the core and the self-shielding of U-236). This result is found to be consistent with an earlier study⁵.

The penalty in the form of a loss in reactivity due to additional capture in U-236 is also calculated as a function of U-236 loading. Table 5 shows the results of the parametric study. The base case contains no U-236 isotope, while the additional 4 cases contain increasing amounts of U-236 from 0.5 wt%, 1.0 wt%, 2.0 wt% and 3.0 wt% respectively. The loss in reactivity is calculated using the Westinghouse lattice (assembly) code. The results show that the reactivity penalty is not increasing uniformly with the increase in U-236 loading due to U-236 resonance self-shielding.

effect. The reactivity penalty also increases with burnup. The U-236 isotopes basically replaces the U-238 isotopes with the increase in U-236 loading. While U-236 and isotopes produced by neutron absorption remains in the core as neutron absorbers, some fraction of U-238 is converted to Pu-239 and increases the reactivity and therefore the penalty associated with high content of U-236.

Table - 5
Reactivity Penalty of U-236

Parameter	Case A (Base Case)	Case B	Case C	Case D	Case E
Enrichment (wt%)	5.00	5.00	5.00	5.00	5.00
U-234 (wt%)	0.097	0.097	0.097	0.097	0.097
U-236 (wt%)	0.0	0.5	1.0	2.0	3.0
Burnup (MWD/MTU)	Reactivity		Penalty		
	Base Case	$\Delta\rho$ (pcm)	$\Delta\rho$ (pcm)	$\Delta\rho$ (pcm)	$\Delta\rho$ (pcm)
0	-	720	1084	1687	2243
5,000	-	684	1075	1740	2360
10,000	-	664	1087	1824	2516
20,000	-	669	1169	2062	2908
30,000	-	713	1303	2379	3402
40,000	-	799	1487	2756	3972

Table 6 shows the variation of concentration of U-234 and U-236 with burnup. The reference case 1 or case D with 2.00 wt% of U-236 is chosen for illustration. It shows that while the concentration of U-234 decreases with depletion due to formation of U-235, the concentration of U-236 increases with burnup. The U-236 fraction increases by about 20% at a burnup of 40,000 MWD/MTU.

Table -6
Isotopic concentration with depletion

Burnup (MWD/MTU)	U-234 (wt%)	U-236 (wt%)
0	0.097	2.00
5,000	0.091	2.09
10,000	0.085	2.17
20,000	0.073	2.29
30,000	0.063	2.37
40,000	0.054	2.41

b) Critical Boron Concentration and Peaking Factor

The critical boron concentration variation with burnup is shown in Figure 2 for cases 1, 2 and 3. The loading patterns are identical in all three cases. The slope of the letdown curves for the case 2 and case 3 are practically the same as the two cases only differ in enrichment. The equivalent case 2 with 4.44 wt% enrichment has less reactivity compared to the reference case 1 with SRS derived fuel, but its slope is less steep, and it recovers some of its reactivity with the formation of more plutonium with burnup and thereby attains the same cycle length of 16,200 MWD/MTU. This trend is consistent with the reactivity penalty trend shown in Table 5.

Peaking factors for the three cases are shown in Table 7. The peaking factors for the reference case is practically the same as in the equivalent case. Therefore, there is no adverse effect in switching from the commercial fuel to SRS derived fuel as far as the peaking factor behavior is concerned.

Table - 7
Peaking factor

Parameter	Case 1 (reference case)	Case 2 (equivalent case)	Case 3
Fuel Type	SRS Derived Fuel	Commercial Fuel	Commercial Fuel
Enrichment (wt%)	5.00	4.44	5.00
$F_{\Delta H}$, BOL	1.41	1.42	1.44
$F_{\Delta H}$, EOL	1.32	1.32	1.33
F_Z , BOL	1.17	1.18	1.18
F_Z , EOL	1.12	1.12	1.12

where

$F_{\Delta H}$ is the nuclear enthalpy rise hot channel factor and is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power and F_Z is the axial peaking factor.

c) Cross Section

Some differences in the basic characteristics of the reference core using SRS derived fuel and the equivalent core using commercial fuel can be understood by comparing the cross sections of different isotopes of uranium. Table 8 gives the 2200 m/sec cross sections and resonance integrals of different uranium isotopes⁶. Table 9 gives the spectrum-averaged two-group cross sections for the reference fuel with 5.00 wt% enrichment and are calculated by the lattice code.

Table - 8
Thermal Cross Section and Resonance Integral*

Parameter	Absorption (capture + fission) 2200 m/sec cross section σ_a , (barn)	Resonance Integral (capture + fission), I, (barn)
U-234	103	666
U-235	682	421
U-236	5.2	355
U-238	2.7	282

* ENDF/B-V data ⁴

Table - 9
Two-Group Microscopic Cross Section

Parameter	σ_a , group-1 (barn)	σ_a , group-2 (barn)
U-234	17.3	41.3
U-235	11.5	268.4
U-236	4.3	2.3
U-238	0.9	1.2

The higher fraction of U-236 effectively replaces U-238 isotopes in the 5.00 wt% enriched SRS derived fuel. Therefore the absorption cross section of U-236 isotope needs to be compared with that of U-238 in order to understand the core characteristics. The thermal absorption cross section of U-236, as seen from Table 8 and 9 is almost twice that of U-238 and resonance integral of U-236 which determines the epithermal absorption is also higher by about 25% compared to U-238.

The higher thermal absorption makes the spectrum harder in a core using SRS derived fuel. The higher resonance absorption in U-236 also makes the Doppler coefficient more negative. It may be noted that any increase in U-234 content in the LWR fuel will add additional poison at beginning of cycle because of its high absorption cross section, but its effect will diminish gradually with burnup by the formation of U-235.

d) Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) is defined as the change in reactivity per degree change in the moderator temperature. It is an important parameter in reactor safety analysis and is usually kept negative at all operating conditions. The soluble boron concentration has an important effect on the MTC value. An increase in the soluble boron concentration introduces a positive component in the MTC. Therefore the maximum ppm level in a core should not exceed a certain value depending on other core parameters in order to make MTC negative at all operating conditions. Another important factor in determining the MTC is the neutron spectrum of the core. A hardened neutron spectrum results in an increase in resonance absorption in U-236 and other isotopes. The hardened spectrum also causes a decrease in the fission-to-capture ratio in U-235 and makes MTC more negative. With burnup, the moderator temperature coefficient becomes more negative primarily as a result of the reduced boron concentration in addition to the effects of the core average buildup of plutonium and fission products. The moderator temperature coefficients at BOL and EOL for three models are given in Table 10.

Table - 10
Moderator Temperature Coefficient *

Parameter	Case 1 (reference case)	Case 2 (equivalent case)	Case 3
Fuel Type	SRS Derived Fuel	Commercial Fuel	Commercial Fuel
Enrichment (wt%)	5.00	4.44	5.00
PPM, BOL, HFP	1990	1823	2083
MTC (pcm/ °F) BOL, HFP, No. Xenon	-12.1	-10.2	-9.7
MTC (pcm/ °F) BOL, HFP, Eq. Xenon	-16.3	-14.5	-13.6
MTC (pcm/ °F) EOL, HFP, Eq. Xenon	-41.2	-39.4	-40.4
MTC (pcm/ °F) BOL, HFP, No. Xenon	-2.0	-1.0	-1.0

* at all rod out, critical boron concentration.

Table 11 shows BOL and EOL fast (> 0.625 ev) and thermal flux ratios. The fast to thermal flux ratio in the reference core with SRS derived fuel increases significantly, primarily due to reduction in thermal flux. This phenomenon is due to the increase in enrichment and the presence U-236 and U-234 which are thermal absorbers.

Table - 11
Neutron Spectrum

Parameter	Case 1 (reference case)	Case 2 (equivalent case)	Case 3
Fuel Type	SRS Derived Fuel	Commercial Fuel	Commercial Fuel
Enrichment (wt%)	5.00	4.44	5.00
Fast to Thermal Flux Ratio, BOL	9.65	8.61	9.44
Fast to Thermal Flux Ratio, EOL	8.38	7.46	7.96

It is clear from Table 10 that the MTC is more negative for the reference case where SRS derived fuel is used. The reference core has the hardest spectrum and consequently more epithermal neutrons are absorbed in U-236. This is in spite of the fact that the reference core has more soluble boron which makes MTC more positive. To study the spectrum effect on MTC, boron concentrations are kept fixed at certain values for the reference case (Case 1) and the equivalent case (Case 2). The results are shown in Figure 3 and 4. The moderator temperature coefficients are always more negative by about 2 to 3 pcm/ °F if SRS derived fuel is used in designing the core compared to commercial fuel of equivalent reactivity.

e) Doppler Power Coefficient

The Doppler coefficient of reactivity is due to temperature changes in the fuel. The Doppler power coefficient (DPC) in pcm/% power, relates to the change in power which produces the fuel temperature change and is always negative. The Doppler coefficient is primarily a measure of the Doppler broadening of U-238 and U-236 resonance absorption peaks at BOL and U-238, U-236 and Pu-240 peaks at EOL. An increase in fuel temperature increases the effective resonance absorption cross section of the fuel and produces a corresponding reduction in reactivity. The Doppler coefficient changes as a function of core life representing the combined effects of the fuel temperature reduction with burnup and the core average buildup of Pu-240 and U-236. The Doppler coefficients calculated at different power levels also include the effect of thermal expansion of the fuel and clad. Expansion effects increase the magnitude of the coefficient by about 10%. The Doppler power coefficients at BOL and EOL are presented in Table 12. The moderator temperature is kept constant at the appropriate value for each power level in these calculations.

Table - 12
Doppler Power Coefficient & Defect

Parameter	Case 1 (reference case)	Case 2 (equivalent case)	Case 3
Fuel Type	SRS Derived Fuel	Commercial Fuel	Commercial Fuel
Enrichment (wt%)	5.00	4.44	5.00
Power (% of full power)	DPC (pcm/%power) BOL	DPC (pcm/%power) BOL	DPC (pcm/%power) BOL
0	-18.3	-16.1	-15.9
50	-13.0	-11.7	-11.7
100	-10.1	-9.2	-9.3
Power (% of full power)	DPC (pcm/%power) EOL	DPC (pcm/%power) EOL	DPC (pcm/%power) EOL
0	-15.0	-13.8	-13.5
50	-11.7	-10.6	-10.7
100	-9.2	-8.5	-8.5
Defect (HZP to HFP), BOL (pcm)	1337	1197	1194
Defect (HZP to HFP), EOL (pcm)	1180	1077	1087

It is seen that the Doppler coefficients are practically the same between the case 2 and 3, since commercial fuel with the same isotopic composition is used in both cases. But the reference core has the highest Doppler coefficients due to additional resonance contribution from U-236. It has already been noted in Table 8 that U-236 has a higher value of resonance integral than that of U-238. Therefore from the safety standpoint, SRS derived fuel will have a slight advantage compared to commercial fuel with equivalent reactivity.

f) Boron Worth

The boron worth defined as pcm per ppm of soluble boron is also an important safety parameter of a reactor core. The boron worths at BOL and EOL are calculated and are shown in Table 13.

Table - 13
Boron Worth

Parameter	Case 1 (reference case)	Case 2 (equivalent case)	Case 3
Fuel Type	SRS Derived Fuel	Commercial Fuel	Commercial Fuel
Enrichment (wt%)	5.00	4.44	5.00
U-236 (wt%)	2.00	0.0146	0.0146
Boron Worth, BOL, HFP (pcm/ppm)	-5.8	-6.4	-5.8
Boron Worth, EOL, HFP (pcm/ppm)	-6.8	-7.6	-7.1

The boron worth is primarily a function of boron absorption in the thermal energy range. The boron worth becomes less negative with an increase in boron concentration and with the hardening of spectrum. The boron worth for the reference core with SRS derived fuel has decreased by about 10%, but still is sufficiently negative to satisfy all safety criteria.

g) Control Rod Worth

The control rods compensate for reactivity changes in the operating power range and provide shutdown margin as per technical specification requirements of the core. The control rod worths of all banks are calculated for three different models to see the effect of U-236 on rod worth and the results are presented in Table 14. The D-bank worth for the reference case is decreased by about 20 to 30 pcm and the total control and shutdown bank worth is decreased by about 200 to 300 pcm due to hardening of spectrum. The decrease in bank worth by these small amounts will not impact the reactor operation or the shutdown margin limit of the core.

Table - 14
Control Rod Worth

Parameter	Case 1 (reference case)	Case 2 (equivalent case)	Case 3
Fuel Type	SRS Derived Fuel	Commercial Fuel	Commercial Fuel
Enrichment (wt%)	5.00	4.44	5.00
U-236 (w/o)	2.00	0.0146	0.0146
Control Bank Worth (pcm)	BOL, HFP, Eq. Xe	BOL, HFP, Eq. Xe	BOL, HFP, Eq. Xe
D	530	557	520
C	849	875	803
B	817	855	933
A	457	467	387
SD	3646	3840	3724
Total	6299	6594	6367
	EOL, HFP, Eq. Xe	EOL, HFP, Eq. Xe	EOL, HFP, Eq. Xe
D	542	561	542
C	1018	1068	1039
B	1127	1194	1169
A	527	549	541
SD	4008	4208	4054
Total	7222	7580	7345

h) Shutdown Margin

Shutdown margin is one of the key considerations in determining the safety of a reactor. Table 15 gives the end of life shutdown margin estimates for the reference core with SRS derived fuel and the equivalent core using commercial fuel. Shutdown margin is defined as the amount of negative reactivity by which a reactor is maintained in a subcritical state at hot-zero power condition after a control rod trip with no change in pre-trip boron or xenon concentration. Shutdown margin is usually more limiting at the end of cycle (EOC) mainly due to more negative MTC value at EOC. As given in Table 15, the control rod requirements consist of four components: the total power defect from hot full power (HFP) to hot zero power (HZP), the moderator temperature deadband allowance, the rod insertion allowance, and the allowance for void collapse. The moderator deadband allowance accounts for the fact that the moderator temperature swing from HFP to HZP could be higher than nominal because of moderator temperature control deadband, typically about 4°F. The rod insertion allowance accounts for allowable control rod insertion during normal operation. Control rods inserted during normal operation do not contribute to rod worth upon trip. The allowance for voids accounts for the small effect of void collapse in going from HFP to HZP. There is a small amount of void formation at HFP due to local or statistical boiling. In this study, generic values have been used for the rod insertion allowance and the allowance for voids.

In Table 15, the available shutdown margin for the reference core with high U-236 content decreases by about 0.45% $\Delta\rho$ compared to the equivalent core, but still is well above the limit. The control rod worth for the reference core is decreased by about 0.3 $\Delta\rho$ and the total control rod requirement is increased by about 0.2 $\Delta\rho$ due to more negative MTC and Doppler coefficients. Shutdown margin tends to decrease in a core because the harder neutron spectrum reduces control rod worth. Also, the power defect, the reactivity added to the core in going from HFP to HZP, increases due to harder spectrum.

Table - 15
Summary of Reactivity Requirements and Shutdown Margin

Parameter	Case 1 (reference case)	Case 2 (equivalent case)
Fuel Type	SRS Derived Fuel	Commercial Fuel
Enrichment (wt%)	5.00	4.44
U-236 (w/o)	2.00	0.0146
Reactivity Insertion (HFP to HZP),		
Power Defect (% $\Delta\rho$)	2.88	2.69
Rod Insertion Allowance (% $\Delta\rho$)	0.50	0.50
Voids (% $\Delta\rho$)	0.05	0.05
Moderator 4°F Deadband (% $\Delta\rho$)	0.16	0.16
(1) Total Control Requirements (% $\Delta\rho$)	3.59	3.40
Total Rod (N) Worth (% $\Delta\rho$)	6.61	6.95
Worst Stuck Rod Worth (% $\Delta\rho$)	0.69	0.74
All Rods inserted less Worst Stuck Rod (N-1)Worth (% $\Delta\rho$)	5.92	5.59
(2) (N-1) Worth less 10% (% $\Delta\rho$)	5.33	5.59
Available Shutdown Margin [(2)-(1)] (% $\Delta\rho$)	1.74	2.19
Required Shutdown Margin (% $\Delta\rho$)	1.30	1.30

6.0 Conclusion

Existing inventories of highly enriched uranium in the form of uranyl nitrate solution stored in the H canyon and irradiated fuel at the Savannah River Site contain a high percentage of U-236. It is possible to blend down enriched uranium with depleted uranium to about 5.0 wt% of U-235 and utilize the low enriched uranium as fuel for commercial power reactors. It is estimated that more than 100 metric tons of low enriched uranium may be produced from the SRS inventories. The U-236 content of this fuel will be more than 100 times the content in commercial fuel. U-236 is considered as a parasitic neutron absorber in reactor fuel. A penalty is associated with a high content of U-236 isotope, since U-236 will be displacing fertile U-238. For a PWR fuel cycle, the penalty associated with U-236 has been determined by developing a three dimensional equilibrium cycle model of a PWR core using an upper bound on the U-236 content in SRS recovered uranium and comparing with an equivalent core using standard commercial fuel. It is determined that a 0.28% overenrichment is required to compensate for each 1.0 wt% of U-236 in the fuel so that 'equal energy' is produced in a power reactor. Most fuel fabricators are currently licensed by the NRC to handle uranium with an enrichment of 5 wt% or less. Therefore the practical enrichment limit for the fuel using SRS uranium containing 2.0 wt% of U-236 will also be 5.00 wt%, which is equivalent to an enrichment of 4.44 wt% of commercial fuel. Many commercial reactor are presently operating with enrichment of 4.4 wt% or lower. It is, thus, feasible to develop commercial fuel cycle strategy using SRS uranium of 5.00 wt% enrichment.

The core physics calculations show that peaking factors remain practically unchanged when commercial fuel is replaced by fuel derived from SRS recovered uranium, slightly overenriched to compensate for high U-236 content in an equilibrium core. The critical boron concentration for the equilibrium core using SRS derived fuel will be a little higher, but can be kept within the design limit. The moderator and Doppler coefficients will be more negative due to harder neutron spectrum and higher resonance absorption in a core with high U-236 content. The U-236 absorption cross section is higher than that of U-238 by a factor of two. The rod worth and boron worth are slightly smaller in a core loaded with SRS derived fuel due to harder spectrum, but this will not impact the safety and operation of a nuclear plant. The available shutdown margin of the SRS derived fueled core is decreased due to smaller rod worth and higher power defect, but can easily be maintained much above the shutdown margin limit. This conceptual study shows the feasibility of using SRS uranium in a PWR from the neutronic point of view and quantifies the fuel cycle penalty associated with U-236.

In addition to high content of U-236, there is also a higher fraction of U-234 in fuel derived from SRS recovered uranium as well. At this concentration of about 0.1 wt%, U-234 does not significantly increase the neutronic penalty of an equilibrium core, even though its effect has been taken into account in calculating the penalty factor.

It should be noted that there will be some radiological and fabrication problems related to SRS derived fuel due to its higher content of U-236, U-234 and U-232, which have not been addressed in this study. For example, U-234 is a major factor of the exposure problem (high alpha activity) in a fuel fabrication plant and the higher value of U-234 exceeding the ASTM limit in SRS derived fuel might require some facility modification or some changes in procedures⁷. High U-236

concentration might contaminate the fuel fabrication line and extensive flushing and cleaning would be required following fabrication with SRS uranium. High U-236 concentration will also produce more Pu-238 in the spent fuel, and consequently more decay heat. U-232 is found to be lower than the ASTM limit and should not pose any additional exposure problem. If this option is pursued in the future, these concerns need to be addressed in further details.

7.0 References

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7. Personal Communication with Tom Freeman and Rich Montgomery, Westinghouse Nuclear Material Division at Columbia, South Carolina (May, 1994).
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NOTE:

1. TOTAL NO. OF CONTROL & SHUTDOWN RODS - 53
2. CYCLE LENGTH $\approx$ 16200 MWD/MTM
3. FEED ENRICHMENT: 5.0 wt%
4. NO. OF GLASS BURNABLE RODS: 672

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Figure 2. Critical Boron Concentration Versus Burnup (HFP, ARO, Eq. Xe)

Figure 3. Moderator Temperature Coefficient At Different PPM and Moderator Temperature (BOL)

Figure 4. Moderator Temperature Coefficient At Different PPM and Moderator Temperature (EOL)

Appendix 1

1.0 Stored Uranyl Nitrate Solution

The following table gives the isotopics of the enriched uranium solution stored in the H-Area¹ as well as the isotopics of the 5 wt% enriched uranium after blending with depleted uranium.

U-235 (wt%)	59.6	5.00
U232 (wt%)	2.00 E-6	1.62 E-7
U-233 (wt%)	1.00 E-2	8.08 E-4
U-234 (wt%)	1.20 E+0	9.70 E-2
U-236 (wt%)	2.44 E+1	1.97 E+0
U-238 (wt%)	1.479 E+1	9.293 E+1

2.0 Existing SRS Fuel Assemblies

The following table gives the average isotopics of different types of fuel assemblies stored at SRS¹ as well as the isotopics of the 5 wt% enriched uranium after blending with depleted uranium.

U-235(wt%)	62.9	5.00
U232 (wt%)	2.00 E-6	1.53 E-7
U-233 (wt%)	1.00 E-2	7.66 E-4
U-234 (wt%)	1.40 E+0	1.07 E-1
U-236 (wt%)	2.18 E+1	1.67 E+0
U-238 (wt%)	1.389 E+1	9.322 E+1

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