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URANIUM-232 PRODUCTION IN CURRENT DESIGN LWRs

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MASTER

UNION
CARBIDE

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ABSTRACT

The spectrum-dependent zero dimensional depletion code, LEOPARD, was modified to permit a study of Uranium-232 buildup in pressurized (PWR) and boiling water reactors (BWR). The results of the study showed good agreement with comparable calculations performed by Westinghouse for PWRs but poor agreement with General Electric calculations for BWRs. Differences noted were attributed to the nuclear cross section data and, in the case of the BWR results, the method of calculation. This study indicates that even at the high burnups projected for current light water reactor fuels, the discharged fuels do not reach concentrations of U-232 in excess of the current gaseous diffusion plant feed specification of 110ppb (U-232/U-235) provided the fuel charged to the reactor is free of thorium and U-236 and that the reactor achieves a reasonable load factor. However, the U-232 concentration in reactor fuel increases between the time it is discharged from the reactor and the time it is reprocessed. After about two years of storage the U-232 concentration reaches the 110ppb limit. The following conditions were shown to cause the U-232 concentration in reactor fuel at discharge to exceed the current GDP feed specification:

- (1) A fuel *loading* with in excess of 88ppb (U-232/U-235) in it.
- (2) A fuel *loading* with in excess of 0.5 w/o U-236 in it.
- (3) A fuel *loading* with in excess of 20ppm (Th-232/Uranium) in it.
- (4) A reactor load factor below 0.33.

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SUMMARY

The spectrum-dependent zero dimensional depletion code, LEOPARD,¹ was modified to permit a study of Uranium-232 buildup in pressurized (PWR) and boiling water reactors (BWR). The results of the study showed good agreement with comparable calculations performed by Westinghouse for PWRs but poor agreement with General Electric calculations for BWRs. Differences noted were attributed to the nuclear cross section data and, in the case of the BWR results, the method of calculation. This study indicates that even at the high burnups projected for current light water reactor fuels, the discharged fuels do not reach concentrations of U-232 in excess of the current gaseous diffusion plant feed specification of 110ppb (U-232/U-235) provided the fuel charged to the reactor is free of thorium and U-236 and that the reactor achieves a reasonable load factor. However, the U-232 concentration in reactor fuel increases between the time it is discharged from the reactor and the time it is reprocessed. After about two years of storage the U-232 concentration reaches the 110ppb limit. The following conditions were shown to cause the U-232 concentration in reactor fuel at discharge to exceed the current GDP feed specification:

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- (3) A fuel *loading* with in excess of 20 ppm (Th-232/Uranium) in it.
- (4) A reactor load factor below 0.33.

INTRODUCTION

In a closed uranium fuel cycle reactor fuel assemblies discharged from the reactor are reprocessed and returned to the gaseous diffusion plant as cascade feed material. One consequence of using spent reactor fuel to feed the diffusion plant is the introduction of U-236 and U-232 into the cascade as a standard part of the feed. Currently, ERDA requires that feed material contain no more than 110ppb U-232 relative to U-235.^{2,3} This limit was based on an assumption that any recycled fuel would be diluted by a factor of ten (10) with virgin feed (which contains no U-232). This specification was set because U-232 decay chain gamma radiation interferes with monitoring instrumentation used to detect U-235 deposits formed in the diffusion equipment.

Many studies have examined the expected U-232 buildup in a reactor,^{2,4,5,6,7,8} but none has heretofore considered all the factors which influence this buildup. Because little detail is provided with the earlier studies, it is difficult to piece together a complete understanding of the buildup and uncertainties which inhere in its calculation. This study was undertaken

to determine the sensitivity of U-232 buildup to such factors as cross section assumptions, reactor type, spent fuel cooling time, initial U-232 and U-236 concentration and final fuel burnup. The latest nuclear data available were used in the analysis and the LEOPARD computer code was modified to perform the calculations. This code was selected because: (1) its results are well validated for pressurized water designs, and it can be adapted to yield boiling water reactor isotopics; (2) its cross section library has been updated recently using the most widely accepted nuclear data files (ENDF/B-IV); (3) it is written for the IBM 360 system implemented in Oak Ridge; and (4) numerous calculations can be done at moderate computational cost. A description of the modified code will be available shortly.⁹

PRODUCTION ROUTES TO U-232

The principal production/depletion routes leading to U-232 in LWRs are shown in Figure 1. The amount of U-232 present in reactor fuel at discharge is primarily a function of the initial U-235 and U-236 concentrations, the degree of fuel burnup and quantity of thorium impurities. With no thorium impurities the U-232 buildup is controlled by the alpha decay of Pu-236 with its 2.85 year half life. Pu-236 is formed, as shown in Figure 1, as a result of the (n, 2n) reaction in Np-237 and subsequent rapid beta decay of Np-236. About 48% of the Np-236 goes to Pu-236, and the remainder decays to U-236. The Np-237 results from neutron captures by U-235 (n, γ), U-236 (n, γ) and U-238 (n, 2n) that yield U-237 which rapidly decays to Np-237. Significant quantities of U-232 are also produced when reactor fuel contains thorium impurities. Either an (n, 2n) reaction in Th-232 or an (n, γ) reaction in Th-230 produces Th-231 which rapidly beta decays to Pa-231. The Pa-231 can undergo an (n, γ) reaction to become Pa-232 which then beta decays to U-232. This thorium impurity chain is included because radiochemical analyses of PWR fuel have shown the presence of small quantities of thorium.

METHOD OF CALCULATION

All computer calculations performed in this study used a modified version of the LEOPARD code. Where possible ENDF/B-IV cross sections were used. Some nuclides are not included in ENDF/B-IV. The cross sections for U-237 and Pu-236 were processed at ORNL from preliminary ENDF/B-V data. Cross sections for Th-231, Pa-231, Pa-232, U-232 and Np-236 were developed from literature referenced in CINDA 75 using methods of Full Information Maximum Likelihood Parameter Estimation and Optimal Filtering¹⁰ to yield "best" cross section sets. (More details on data preparation are provided elsewhere.⁹)

LEOPARD has been extensively tested as a calculational tool for performing

Figure 1: IMPORTANT ISOTOPIC TRANSITIONS DURING (And After) REACTOR OPERATION

PWR isotopics studies. As a result, high confidence in its results for current design PWRs is attained. LEOPARD has not been well tested for BWRs. Because of the wide variations in core void fraction and the large control rod water slots in the BWR core design, serious questions about the validity of any zero dimensional code should be raised.

General Electric has provided the results of their own calculations of U-232 buildup in BWRs.¹¹ These results show U-232 concentrations in the fuel *at discharge* to be about twice the value deduced from LEOPARD and about 30% higher 25 years after discharge. Sufficient comparative information was not made available by General Electric to resolve the discrepancy.

RESULTS OF CALCULATIONS

Uranium-232 concentrations in reactor fuel depend on: (1) the initial isotopics of the fuel; (2) the type of reactor in which the fuel is irradiated; (3) the discharged fuel's burnup; (4) the reactor load factor and time from fuel discharge until reprocessing. (Further, any calculation of U-232 concentrations depends on the basic nuclear data and the appropriateness of the computer code used.) To serve as a basis for comparing the influence of these diverse factors specific PWR and BWR reactors were selected for analysis. The PWR selected was the WATTS BAR reactor. This reactor has a nominal 1100 MWe rating and uses the "15 x 15" Westinghouse fuel assembly.¹² The BWR selected was General Electric's generic BWR/6 design.¹³ This reactor also has a nominal 1100 MWe rating and uses the 8 x 8 GE assembly. Table I presents a summary of initial and discharge isotopics for the reference reactors.

It is frequently assumed by reprocessors that fuel must "cool" a minimum of 6 months before reprocessing in order to allow decay of the highly radioactive fission products in the fuel at discharge. With this limitation the minimum U-232 concentration achievable in reprocessed, unenriched fuel reaches 45ppb (U^{232}/U^{235}). Since the status of reprocessing is unsettled, selection of a specific reprocessing delay is almost impossible. Figure 2 shows the change in the U-232 concentration as Pu-236 decays to U-232 after discharge from the reactor, but before reprocessing. For long storage times the U-232 concentration approaches 220ppb.

Initial Fuel Isotopics

The amount of U-232 present at reactor discharge is a function of initial U-232, U-235, U-236 and thorium content. The effect of each of these nuclides was examined parametrically and quadratic fit equations were developed which express exit U-232 concentration as a function of initial concentrations. Figures 3-5 present U-232 concentrations as a function of burnup for different initial U-232, U-236 and Th-232 contents respectively for a PWR. Graphs 6-8 provide the same information for a BWR. For comparison, values calculated from a GE formula¹¹ are also presented. To

Table I: Typical Reactor Uranium Requirements and Returns*
(No fuel recycled)

Watts Bar PWR (w/o total U)		
<u>Isotope</u>	<u>Loading</u>	<u>Discharge**</u>
U-235	3.2 w/o	0.90 w/o
U-236	0.0 w/o	0.42 w/o
U-234	0.031 w/o	0.019 w/o
U-232	0	22 ppb (U ²³² /U ²³⁵)

BWR/6 (w/o total U)		
<u>Isotope</u>	<u>Loading</u>	<u>Discharge***</u>
U-235	2.715 w/o	0.90 w/o
U-236	0.0 w/o	0.35 w/o
U-234	0.026 w/o	0.016 w/o
U-232	0	22 ppb (U ²³² /U ²³⁵)

*Calculations performed by OA&P using a modified version of the LEOPARD computer code.

**Fuel burnup: 32,000 MWd/MTU

***Fuel burnup: 27,000 MWd/MTU

Figure 2:

EFFECT OF PU-236 ON U-232 BUILDUP
AFTER DISCHARGE FROM A PWR

Figure 3: $\frac{\text{U-232}}{\text{U-235}}$ VS. BURNUP

SQUARE 20 x 20 1 INCH
AS-0810

U-232/U-235 ppb

(1) Watts Bar PWR design

(2) U-232 concentrations at reactor discharge

100 ppb initial U-232

20 ppb initial U-232

10 ppb initial U-232

2 ppb initial U-232

Burnup (MWd/kgU)

Figure 4:

$\frac{U-232}{U-235}$ BUILDUP VS. BURNUP

- (1) Watts Bar PWR design
- (2) Discharge burnup 32 $\frac{MWd}{kg U}$
- (3) Initial U-235 assay adjusted to produce equal fuel life for all cases
- (4) U-232 concentration 6 months after discharge from the reactor

SQUARE 20 x 20 1 INCH
AS-0910 68

Fig. 4: U-232/U-235 vs. Burnup (MWd/kg U) for Watts Bar PWR design

Figure 5: $\frac{U-232}{U-235}$ VS. BURNUP

Figure 6: $\frac{U-232}{U-235}$ VS. BURNUP

(1) BWR/6 design

(2) U-232 concentrations at reactor discharge

U-232/U-235 ppb

— OA&P Calculation
 --- GE Calculation

Burnup (MWd/kgU)

Figure 7:

U-232
U-235 BUILDUP VS. BURNUP

- (1) BWR/6 design
- (2) Average discharge burnup $27 \frac{\text{MWd}}{\text{kg U}}$
- (3) Initial U-235 assay adjusted to produce equal fuel life for all cases
- (4) U-232 concentration 6 months after discharge from the reactor

SQUARE 20 x 20 1 INCH
AS-0319

FIG. 7: U-232 (U-232/U-235) vs. BURNUP (MWd/kg U)

Figure 8: $\frac{U-232}{U-235}$ VS. BURNUP

SQUARE 20 - 20 1/4 INCH
AS-0810-05

U-232/U-235 ppb

U-232/U-235 ppb

180

160

140

120

100

80

60

40

20

0

(1) BWR/6 design

(2) U-232 concentrations at reactor discharge

— OA&P Calculation

- - - GE Calculation

Current Cascade Feed Spec = 110 ppb

10 ppm Th-232 Impurity

No Thorium Impurity

Burnup (MWd/kgU)

10 12 14 16 18 20 22 24 26 28 30 32 34 36

illustrate the effects of initial enrichment, Figure 9 presents U-232 concentrations for selected initial cores as a function of burnup. Using a quadratic least squares fitting routine with the data we derive the following equations to express discharge U-232 (x_d) and Pu-236 (w_d) as a function of initial isotopics:¹⁴

$$x_d^{\text{PWR}} = (2.22 \times 10^{-8}) + 0.488x_i + (9.67 \times 10^{-8})y_i - (4.97 \times 10^{-9})y_i^2 \\ + (4.18 \times 10^{-5})z_i$$

$$w_d^{\text{PWR}} = (1.60 \times 10^{-7}) + (5.91 \times 10^{-7})y_i - (3.45 \times 10^{-8})y_i^2$$

$$x_d^{\text{BWR}} = (2.23 \times 10^{-8}) + 0.469x_i + (1.01 \times 10^{-7})y_i - (5.61 \times 10^{-9})y_i^2 \\ + (3.91 \times 10^{-5})z_i$$

$$w_d^{\text{BWR}} = (1.4 \times 10^{-7}) + 4.43 \times 10^{-7}y_i - (7.45 \times 10^{-9})y_i^2$$

where: w_d = discharged Pu²³⁶ concentration in w/o.

x_d = discharged U²³² concentration in w/o.

x_i = initial U²³² concentration in w/o.

y_i = initial U²³⁶ concentration in w/o.

z_i = initial Th²³² concentration in w/o.

Here we have assumed a burnup of 32MWd/kgU for PWRs and a 27MWd/kgU burnup for BWRs with reload cores installed. We make the generally accurate assumption that variations in initial Th-232, U-236 and U-232 produce independent effects on discharged U-232 concentrations. The comparable U-232 equation derived from GE data for a BWR is:

$$x_d^{\text{BWR}} = (5.5 \times 10^{-8}) + 0.704x_i + (2.04 \times 10^{-7})y_i + (5.74 \times 10^{-5})z_i$$

Reactor Capacity Factor

To this point U-232 calculations have all been calculated assuming a 100% load factor. Since the U-232 concentration in the fuel increases as a function of storage time before reprocessing (as a result of Pu-236 decay, see Figure 2) it will also be a function of the reactor capacity factor. As load factor is reduced and fuel residence time in the reactor increases,

Figure 9: $\frac{U-232}{U-235}$ Buildup vs. Burnup

- (1) ——— Watts Bar PWR
 - - - BWR/6

- (2) Discharge Burnup: 2.57 w/o 24,750
 2.14 w/o 14,203 $\frac{MWd}{MTU}$
 2.02 w/o 16,360
 1.87 w/o 11,503

- (3) U-232 concentrations 6 months after discharge from the reactor

SQUARE 20 x 20 1 INCH
 AS-0319

IN REACTOR OF THE U.S. NUCLEAR ENERGY

ppb U-232 (U-232/U-235)

280
240
200
160
120
80
40
0

Current Cascade Feed Spec = 110 ppb

1.87 wt % U-235

2.14 wt % U-235

2.02 wt % U-235

2.57 wt % U-235

Burnup (MWd/kg U)

10 12 14 16 18 20 22 24 26 28 30 32 34 36

the Pu-236 concentration is given more time to reach its equilibrium concentration in the reactor. This results in more U-232 at fuel discharge. However, the gross amount of Pu-236 produced in the reactor is not a function of residence time, but depends on burnup and nuclear cross sections. Simulations on the Connecticut Yankee reactor (575MWe) at power levels between 25-100% indicate the following relation can be used to relate U-232 concentration to fuel residence time at burnups of 32MWd/kgU:

$$N_x(22) = N_{100}(22) \left(\frac{T_x}{T_{100}} \right)^{0.8}$$

where: $N_x(22)$ = U-232 concentration at load factor, x.

T_x = fuel residence time at load factor, x.

This expression suggests that a reactor operating at a 50% load factor may produce 74% more U-232 than the same reactor operating at a 100% capacity factor. However, to determine U-232 concentrations 6 months after discharge, the proper exponent is about 0.5 and 2 years after discharge the value is reduced to about 0.25. Fourteen years after discharge the exponent is about 0.0 and the U-232 concentration would be independent of load factor. In short, the gross amount of U-232 produced does not depend on load factor if enough time is allowed for the Pu-236 to decay to U-232.

EVALUATION OF CALCULATED RESULTS

The extreme paucity of good experimental data and serious questions about the accuracy of basic nuclear data make the task of validating the computational results difficult. Only two major experimental programs sought to measure U-232 buildup in a power reactor. Westinghouse analyzed some fuel elements from Yankee Rowe (175MWe) cores I, II and IV and General Electric performed similar measurements on Dresden I (200MWe) fuel assemblies. (In the Westinghouse study, preselected fuel rods were removed intact from their assemblies and up to 6 fuel pellets were removed for analysis. In the GE program samples were drawn from the assemblies' dissolver solution.) By current standards these reactors are small and their design characteristics are quite different from those of current reactors. Nevertheless, computer simulations which give good agreement with experiment for these early reactors can increase our confidence that the computer calculations are accurate for larger reactors. Unfortunately, agreement between these experiments and our present calculations is poor for U-232. This fault is not unique to our calculations, but was discovered in the original Westinghouse and General Electric calculations which accompanied their experimental work.^{4,8} Figure 10 displays the calculated and measured U-232 concentrations as a function of burnup for the Westinghouse tests. Both our calculations and those of Westinghouse showed U-232 concentrations at fuel burnups about 30MWd/kgU which were only half the experimental value.

Figure 10. U-232 CONCENTRATION VS. BURNUP IN WESTINGHOUSE 'HIC' PROGRAM

The Westinghouse values were low despite the selection of questionable flux weighted cross sections intended to increase the calculated U-232 concentration.⁷ (At high energies they used 1/E weighted (n, 2n) cross sections for Np-237 and U-238 despite their admission that fission weighting was more appropriate.) However, they did achieve good agreement between experiment and calculations for Np-237 and Pu-236. Westinghouse concluded that the higher U-232 experimental findings resulted from thorium impurities in the fuel. A 10-20ppm (Th/U) Th-232 concentration was found sufficient to produce the noted discrepancy. Unfortunately, thorium contamination of this magnitude was not measured. In the Westinghouse study less than 1ppm (Th/U) was measured in the preirradiation analysis. Nonetheless, adducing experimental errors and the small preirradiation sample, Westinghouse attributed the difference between calculated and experimental U-232 to a Th-232 impurity with a concentration of about 10ppm (Th/U). The difference was not attributed to errors in the basic nuclear data because of the excellent agreement between calculated and experimental Pu-236 and Np-237 concentrations. The OA&P calculations are in close agreement with the original Westinghouse calculations for Pu-236 and Np-237, and in the case of Np-237 they are in better agreement with experimental results.

The data provided in General Electric reports on U-232 buildup in Dresden I assemblies was not detailed enough to perform the kind of comparisons made on Yankee Rowe. The General Electric experimental program as reported was significantly less extensive than the Westinghouse study. The calculated U-232 concentrations were a factor of 2.5 lower than experimental measurements. This discrepancy was attributed to a thorium impurity as was done in the Westinghouse study. Very little other experimental data were found which considered U-232 production. A series of measurements were performed on irradiated fuel delivered to Nuclear Fuel Services (NFS) in West Valley, New York from October 1969 to November 1971. These measurements were made on fuel from Dresden I (200MWe BWR), Yankee Rowe (175MWe PWR), Indian Point (265MWe PWR), Big Rock Point (72MWe BWR), and Humboldt Bay (65MWe BWR).¹⁵ Unfortunately, with data currently available in Oak Ridge it is not possible to determine the fuel burnup, the delay between reactor discharge and isotopic analysis, nor the experimental accuracy. These limitations make the data almost useless.

Uranium-232 and Pu-236 concentrations in three foreign BWRs were reported by Braun and Moy.¹⁶ They reached the same conclusions about thorium impurities in the fuel as Westinghouse and General Electric. However, no pre-irradiation measurements were performed to verify this conclusion. Although this conclusion appears reasonable, the failure to measure the requisite concentrations mitigates its veracity. No good explanation has been offered which explains how these thorium concentrations enter the fuel cycle. To date LWR fuel assemblies have used enriched uranium as fuel. Because thorium does not form a stable gaseous hexafluoride it is usually assumed that there is no thorium leaving the cascade. Once the enriched uranium is converted to UO₂ and fabricated into fuel we expect some of the U-234 to alpha decay to Th-230, and U-235 to alpha decay to Th-231. But the long half lives for U-235 and U-234 (7.13×10^8 years and 2.48×10^5 years) require

that the UO_2 "age" for many years before significant quantities of thorium can be generated. Fabricated fuels are not currently allowed to "age." This leaves contamination of fuel containers, housing, etc. as the only reasonable source of thorium impurities. General Electric feels that their fabricated fuel contains less than 5ppm thorium (Th/U) and less than 1ppb of this is thorium-230.¹⁷

Other possible sources of U-232 exist. One example of an alternative is the reaction:

The (n, 3n) reaction has a poorly known cross section and U-234 concentrations of about 0.03 w/o are known to exist in the fuel charged to the reactor. If this path is included as a source of U-232 and ENDF/B-IV data are used for the (n, 3n) cross section, LEOPARD calculations show a 20-25% increase in U-232 at discharge. However, values for the (n, 3n) cross section are reported as much as a factor of 2 different from the ENDF/B-IV value.^{18,19}

Current Westinghouse and General Electric calculations for the latest PWR and BWR designs treat thorium impurities parametrically and give no attention to (n, 3n) reactions in U-234. Westinghouse considers the thorium-232 in their fuel to range from 1-10ppm (Th/U). General Electric has considered thorium impurities from 0-20ppm with from 0-2ppb Th-230. It is difficult to make comparisons of calculated U-232 concentrations in spent fuels because of the plethora of different assumptions made in the analyses. However, Table II presents some reasonably comparable calculations. We note fair agreement among the calculated U-232 concentrations for PWRs, but poor agreement for BWRs. As indicated earlier, our results do not consider in detail the spatial aspects of fuel burnup--General Electric does. These and probable nuclear cross section differences are definitely a major cause of the discrepancies in the BWR calculations.

As indicated earlier, ENDF/B-IV data were used wherever they were available in this study. However, even these evaluated nuclear data are subject to great uncertainty for some nuclear reactions, especially the (n, 2n) and (n, 3n) reactions. Sensitivity analyses show that the variations in U-232 concentration at discharge burnups vary directly with the Np-237 (n, 2n) cross section, i.e. a 50% increase in cross section results in a 50% increase in U-232 concentrations. Recent measurements and analysis indicate the Np-237 (n, 2n) cross section in ENDF/B-IV may be as much as 60% too large.²² The U-232 concentration in reactor fuel is sensitive to other cross sections as well. A 50% change in the U-238 (n, 2n) or U-236 (n, γ) produces about a 25% change in U-232 concentrations.*

*Insufficient data is available to even add the U-232 cross section to the ENDF system. Different sources report widely different values for the (n, γ) reaction.²³

Table II: Summary of Calculated U^{232} Concentrations

Organization	U^{22}/U^{25} (in ppb)	Burnup (in MWd/kgU)	Thorium Impurities (ppm Th^{232}/U)	Cooling Time (in days)	Reactor
Westinghouse ²⁰	42	32	1	160	PWR (current design)
OA&P (4/77)	44	32	1	160	Watts Bar PWR
SRL ²¹	28	33	0	160	Westinghouse '17x17'
OA&P (4/73) ⁵	64	32	0	180	Diablo Canyon PWR
AGNES ⁴	34	33	0	160	PWR (current design)
ORNL ²	49	34.5	0	180	PWR (current design)
Westinghouse ²⁰	80	32	10	160	PWR (current design)
OA&P (4/77)	85	32	10	160	Watts Bar PWR
AGNES ⁴	79	33	10	160	PWR (current design)
ORNL ²	79	34.5	10	180	PWR (current design)
GE ¹⁰	80	27	0	180	BWR/6
OA&P (4/77)	46	27	0	180	BWR/6
OA&P (4/73) ⁵	60	27	0	180	BWR/5
GE ¹⁰	181	27	10 Th^{232} 1ppb Th^{230}	180	BWR/6
OA&P (4/77)	92	27	10	180	BWR/6

Considering the likelihood of compound errors, calculated U-232 values may be significantly more in error than is apparent from uncertainty in an individual cross section.

CONCLUSIONS

1. LEOPARD calculations of U-232 production in reactors are at best marginally adequate. This conclusion was reached because:
 - a. The great uncertainty in some relevant reaction cross sections can produce significant differences in expected U-232 concentrations.
 - b. Several issues revolving around the reprocessing and use of spent fuel are undecided. As a result spent fuel cooling times are unknown and U-232 levels in the fuel cycle are uncertain.
 - c. Trace amounts of thorium impurities result in significant concentrations of U-232. Because levels of thorium impurities are unknown, actual U-232 levels are uncertain.
 - d. OA&P methods for calculating discharge isotopics are not well suited to BWRs and General Electric is less than candid about their methods. (Since GE calculations are reputed to be based on proprietary GE experimental results, their U-232 concentrations are probably the more accurate.)
2. Better determination of U-232 production in reactors requires:
 - a. Better measurements of the U-236 (n, γ) and Np-237 (n, 2n) reaction cross sections or a resolution of current experimental differences.
 - b. Measurement of thorium impurities and identification of their sources in the fuel cycle.
 - c. Better methods for treating heterogeneities in BWR cores.

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