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Assessment of the neutron and gamma sources of the spent BWR fuel

Interim report on Task FIN JNT A 1071 of the Finnish support programme to IAEA Safeguards

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ABSTRACT

The neutron and gamma sources of the spent nuclear fuel were assessed by calculating the nuclide inventory of the spent fuel using SAS2H/ORIGEN-S code. The calculations were performed by VTT Energy under Task JNT A 1071 FIN Partial Defect Test on Spent Fuel LWRs. According to the calculation results the neutron emission of the spent LWR fuel is dominantly from spontaneous fission of Cm-244 at cooling times longer than two years. The parameters that affect the Cm-244 concentration at discharge are fuel bundle type, burnup, and in BWR also void fraction. The power history has only a slight influence on buildup of Cm-244. Also the axial distribution of the neutron source can be determined. Spent LWR fuel contains a variety of gamma emitting nuclides. However, only Cs-137 and Cs-134 gamma peaks stand out from the gamma spectrum. Cs-137 is a better indicator of the burnup than Cs-134, but the ratio between the amount of Cs-134 and Cs-137 can be used to estimate the cooling time.

CONTENTS

1 INTRODUCTION

LWR spent fuel assemblies are being placed in storage facilities where they are considered difficult to access. According to the safeguards concept, IAEA verifies the assemblies before they become difficult to access. Therefore, a standard verification procedure needs to be developed. The aim of this work was to calculationally assess the neutron and gamma source of the spent LWR fuel. The calculations were performed by VTT Energy under Task JNT A 1071 FIN Partial Defect Test on Spent Fuel LWRs.

In this work the nuclide inventory of the spent BWR and VVER-440 type fuels were calculated applying SAS2H/ORIGEN-S code that is part of the modular SCALE code system [1]. CASMO-4 was applied to the BWR case as an additional reference code. In parallel with this work, the neutron emission rate and gamma spectrum of some BWR bundles of the Olkiluoto reactors were measured with IAEA BWR Fork detector coupled with compact underwater gamma spectrometer [2].

2 CALCULATIONAL TOOLS

2.1 SAS2H/ORIGEN-S in SCALE

The SCALE (Standardized Computer Analysis for Licencing Evaluation) system was developed by Oak Ridge National Laboratory for the Nuclear Regulatory Commission to satisfy a need for a standardized method of analysis for the evaluation of nuclear fuel facility and package designs. In its present form, the system has the capability to perform criticality, shielding, and heat transfer analyses using well established functional modules tailored to the SCALE system. In this work SCALE-4.4 acquired through NEA Data Bank was applied for assessment of the neutron source of spent LWR fuel.

ORIGEN-S is a functional module of the SCALE system that performs a depletion and decay calculation for spent fuel characterization and source term generation. A matrix exponential expansion model supplemented with Batemann chain equations is used to solve the nuclide generation and depletion equations. Over 1600 nuclides are followed in the depletion analysis [1, S2.2.10]. The SAS2H module uses ORIGEN-S to perform a fuel depletion analysis (to characterize spent fuel and/or generate source terms) and an optional one-dimensional radial shielding analysis of a cask-type geometry. SAS2H execution includes: repeated passes through BONAMI, NITAWL-II, XSDRNPM, COUPLE, and ORIGEN-S. This simple procedure has been shown to produce conservative actinide inventories for both PWR and BWR spent fuel. The neutron flux used to update the cross sections during the depletion was based on a one-dimensional model of the fuel assembly. Even though the flux in the axial direction of the assembly is assumed to be constant, the procedure characterizes the major 2-D effects.

Several multigroup cross section data libraries are available for SAS2H/ORIGEN-S depletion analysis. In this work 44-group 44GROUPNDF5 cross section library was used. It is based on the ENDF/B-V evaluated nuclear data. SAS2H/ORIGEN-S has been validated against several western type PWRs and BWRs [1, S2.3.1]. Validation studies for BWR have been conducted but they are still to be published.

2.2 CASMO-4

CASMO-4 is a multigroup, two-dimensional transport theory code for burnup calculations on BWR and PWR fuel assemblies. The code can handle geometries consisting of cylindrical fuel rods of varying composition in a square or hexagonal array. Two multigroup cross section data libraries are available for CASMO-4 calculations: 70-group E4LBL70 and a condensed 40-group E4LBL40. Both libraries are based mainly on the ENDF/B-IV evaluated nuclear data. In this work CASMO-4 was applied as an additional reference code.

3 SPENT LWR FUEL NEUTRON AND GAMMA SOURCES

3.1 Nuclide inventory calculations

In this work SAS2H/ORIGEN-S and CASMO-4 were applied to assess the nuclide inventory of spent BWR fuel. The BWR fuel bundle was a common 8x8 BWR fuel bundle with four gadolinium burnable absorber rods. The amount of Gd_2O_3 in the burnable absorber rods was 2 w-%. The fuel average enrichment was 2.75 w-%, its average discharge burnup was 37.8 MWd/kgU, and the average coolant density was 0.426 g/cm^3 which corresponds to average void fraction 0.45. The results of the SAS2H/ORIGEN-S calculations in Appendix A give the nuclide inventory right after discharge (column: initial) and at several cooling times. SAS2H/ORIGEN-S was also applied to assess the nuclide inventory of spent VVER-440 fuel. The VVER-440 assembly studied was a typical uniformly enriched (3,6 w-%) hexagonal bundle with 126 fuel rods and a central water rod. Its average discharge burnup was assumed to be 36 MWd/kgU. The results of the SAS2H/ORIGEN-S calculation are in Appendix B.

3.1 The major components of the spent LWR fuel neutron and gamma sources

In appendices C and D are the major components of the primary neutron and gamma sources of the above described spent BWR fuel bundles calculated by SAS2H/ORIGEN-S. The major components of the neutron and gamma sources of the spent VVER-440 fuel were basically the same. All the sources were normalized to an assembly. ORIGEN-S gives the gamma source spectrum in eighteen gamma energy groups so that the source terms are normalized to the average energy of each group in order to preserve the total gamma energy production.

Table I. *The major components of the neutron source of BWR spent fuel at discharge (n/s/assembly), burnup was 37.8 MWd/kgU.*

Components	SAS2H/ORIGEN-S	CASMO-4
S.F. of ^{244}Cm	$1.42 \cdot 10^{8}$	$9.97 \cdot 10^{7}$
S.F. of ^{242}Cm	$6.98 \cdot 10^{7}$	$6.64 \cdot 10^{7}$
Spontaneous fission total	$2.14 \cdot 10^{8}$	$1.67 \cdot 10^{8}$
$(\alpha$,n)-reaction	$1.58 \cdot 10^{7}$	$8.20 \cdot 10^{7}$
Total	$2.30 \cdot 10^{8}$	$2.49 \cdot 10^{8}$

Table II. *The major components of the neutron source of VVER-440 spent fuel at discharge (n/s/assembly), burnup was 36 MWd/kgU.*

Components	SAS2H/ORIGEN-S
S.F. of ^{244}Cm	$4.87 \cdot 10^{7}$
S.F. of ^{242}Cm	$3.55 \cdot 10^{7}$
Spontaneous fission total	$8.50 \cdot 10^{7}$
$(\alpha$,n)-reaction	$7.89 \cdot 10^{6}$
Total	$9.29 \cdot 10^{7}$

The major components of the calculated neutron source at discharge are shown in Table I and Table II. According to the calculational results the major part of the neutron source is produced from spontaneous fission of heavy nuclides, namely ^{244}Cm and ^{242}Cm. A significant neutron source is also produced from $^{17}O(\alpha,n)$ and $^{18}O(\alpha,n)$ reactions in the UO_2 and other oxygen compounds of the spent fuel.

Spent LWR fuel contains a variety of gamma emitting nuclides. After ten years cooling time ^{137}Cs or actually its daughter nuclide ^{137m}Ba is the dominant gamma emitter. It has been shown that the amount of ^{137}Cs depends linearly on burnup [3], and with gamma spectroscopy the ^{137}Cs contents of spent fuel can be determined based on the measured 662 keV gamma line emission. Other major gamma emitters are ^{154}Eu, ^{134}Cs, ^{106}Ru and

^{144}Ce [4]. However, last two have a relatively short half-life.

3.3 The accuracy of the calculational results

The average difference in the SAS2H/ORIGEN-S computed and measured PWR isotopic concentration for ^{244}Cm has been found to be -9% when applying 44GROUPNDF5 nuclear data library [1, S2.3.1]. The combined uncertainty in the computed neutron source from spontaneous fission is in the range 4 to 8%, with a decrease as a function of time resulting from the decay of ^{242}Cm. In ORIGEN-S thin target cross sections for (α,n)-reactions and alpha stopping powers data are applied to compute neutron yields of the fuel material. The data and method originally used in ORIGEN overpredicted the (α,n) yields by a factor of approximately 2. This is in accordance with the calculations of Anttila using ORIGEN-2 [5]. The uncertainty of the present data and method is estimated to be of the order of 20%. Thus, overall uncertainty of the total neutron source after 5 years cooling time is estimated to be of the order of 10%.

According to the calculational results CASMO-4 seems to slightly underpredict the neutron source from spontaneous fission, and drastically overpredict neutrons from (α,n)-reactions. Nevertheless, the total neutron source given by CASMO-4 agrees relatively well with the SAS2H/ORIGEN-S result. The cross section data used in the CASMO-4 calculations was based on ENDF/B-IV evaluated nuclear data library, and the depletion calculation of CASMO-4 follows only some 100 most important nuclides. The data and method used to estimate (α,n) yields is basically similar to that of ORIGEN-2.

The accuracy of gamma source spectra is dependent upon the energy. Photon rates computed for fission products tend to be more accurate than those for actinides, because the fission product inventory computed for a given burnup depends directly upon calculation of removable energy per fission. Thus, the uncertainty of the ^{137}Cs and ^{134}Cs gamma sources should be of the order of 5% [1, S2.3.3].

4 NEUTRON SOURCE ASSESSMENT

4.1 Cooling time decay

The decay of the major components of the neutron source during cooling is shown in Figure 1. At discharge (α,n)-reactions contribute 7% of the neutron source, and their contribution decreases in the course of time. ^{242}Cm contributes about 30% of the neutron source at discharge, but due to itse short half-life (163 d) it decays leaving ^{244}Cm the dominant neutron source at longer cooling times.

After discharge ^{244}Cm decays with a half-life 18.1 years, and a straightforward correction can be introduced to account for it

$$S_p(T_c) = S_p^0 e^{-\lambda T_c} \qquad (1),$$

where S_p^0 is the primary ^{244}Cm neutron source at discharge,
λ is the decay constant of the ^{244}Cm (λ = 1.2135·10^{-9} 1/s), and
T_c is the cooling time after discharge.

The cooling time correction based on the decay of ^{244}Cm was verified by SAS2H/ORIGEN-S calculation, where the neutron source from spontaneous fission of ^{244}Cm was calculated as a function of time. The cooling lasted for 50 years, and the results given by a simple decay model agreed well with the SAS2H/ORIGEN-S results.

4.2 Simple ^{244}Cm neutron source model

According to Würz [6] The assembly-averaged primary ^{244}Cm neutron source at discharge of the spent LWR fuel with burnup higher than 18 MWd/kgU can be determined using a simple formula

$$S_p^0 = a(BU)^b \qquad (2),$$

where S_p^0 is the primary ^{244}Cm neutron source at discharge,
BU is the discharge burnup,
a and *b* are fuel bundle type dependent constants.

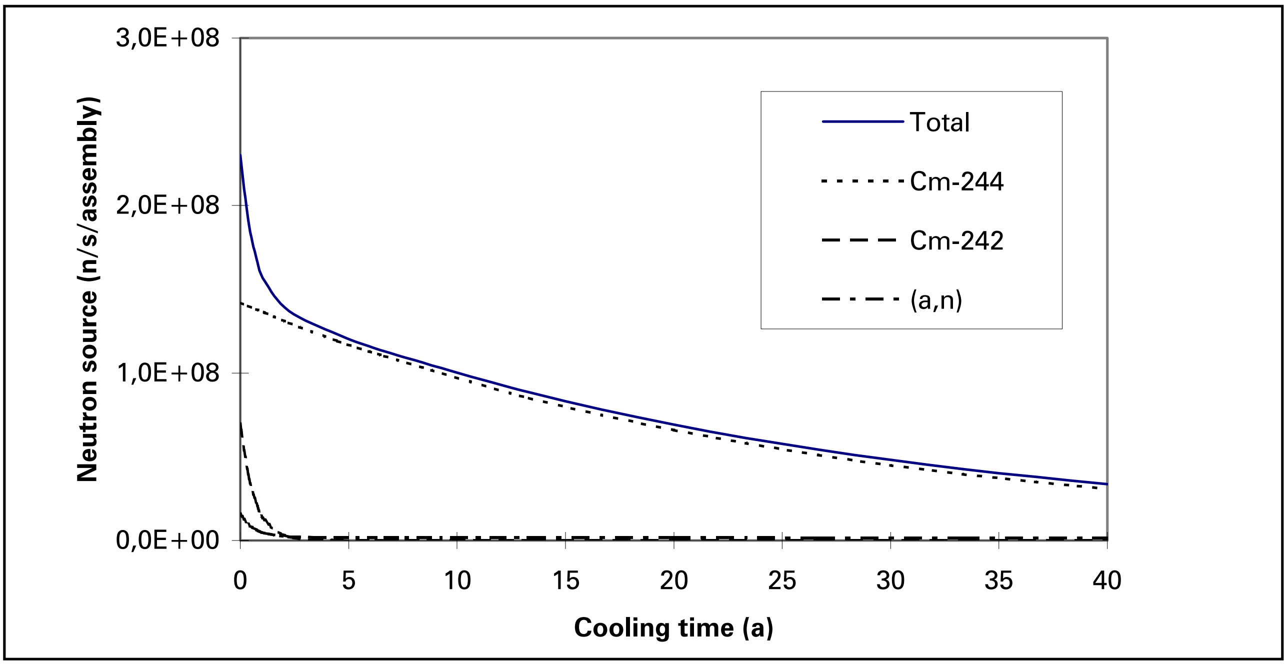

Figure 1. *Total neutron source and its major components as a funtion of cooling time (enrichment 2.746 w-%, burnup 37.8 MWd/kgU, and coolant density 0.426 g/cm*^{3}*).*

At Kernforschungszentrum Karlsruhe burnup-dependent neutron emission profiles of spent BWR and PWR fuel were measured by a combination of active and passive neutron measurement, and the mean neutron emission of the fuel bundles were determined. The neutron emission values were corrected for the decay of ^{244}Cm, and the ^{242}Cm contribution was eliminated. According to Würz *b* is 3.97 for BWR, and 3.94 for PWR spent fuel; and the initial enrichment does not affect *b* [6]. At VTT Energy Anttila has studied the burnup-dependence [5]. His ORIGEN-2 calculations implied that *b* varies from 4.5 to 4.8.

The burnup-dependence of the primary ^{244}Cm neutron source at discharge was studied by a serie of SAS2H/ORIGEN-S calculations. Discharge burnup varied from 15 to 40 MWd/kgU. The fuel bundle was a common 8x8 BWR fuel bundle without burnable absorber rods. A uniform enrichment of 3.0 w-% was applied, and the moderator density was set 0.5. Average power of the bundle was set 4.5 MW, and the desired discharge burnup was achieved by adjusting the time that the bundle was irradiated in the reactor. The calculated burnup-dependence of total neutron source, and ^{242}Cm and ^{244}Cm spontaneous fission neutron sources are shown in Figure 2. The value of *b* was determined using least-square-method, and it was found to be 4.67. The value of *a* was found to be 5. The least-square fit was good (R^2=0.9992), and the model was capable of predicting the neutron source within 7% when burnup varied from 15 to 40 MWd/kgU. The model tended to overpredict the neutron source at low and high burnups, and give too low neutron source strength at intermediate values of burnup. The determination of the *b* value was repeated at moderator density 0.752 g/cm^3 (zero void fraction), and it *b* was found to be 5.03. The fact that *b* does not stay constant when void fraction varies implies that the simple ^{244}Cm neutron source model is not adequate for accurate assessment of the neutron source.

4.3 Enrichment dependence

The enrichment-dependence of the primary ^{244}Cm neutron source at discharge was studied by a serie of SAS2H/ORIGEN-S calculations. The uniform enrichment was varied from 1.6 up to 4.4 w-%. The fuel bundle was a common 8×8 BWR fuel without burnable absorber rods. The void fraction was set 0. The parameter study was performed at three different discharge burnups: 20, 30 and 40 MWd/kgU. The average power of the bundle was set 4.5 MW, and the desired discharge burnup was achieved by adjusting the time that the bun-

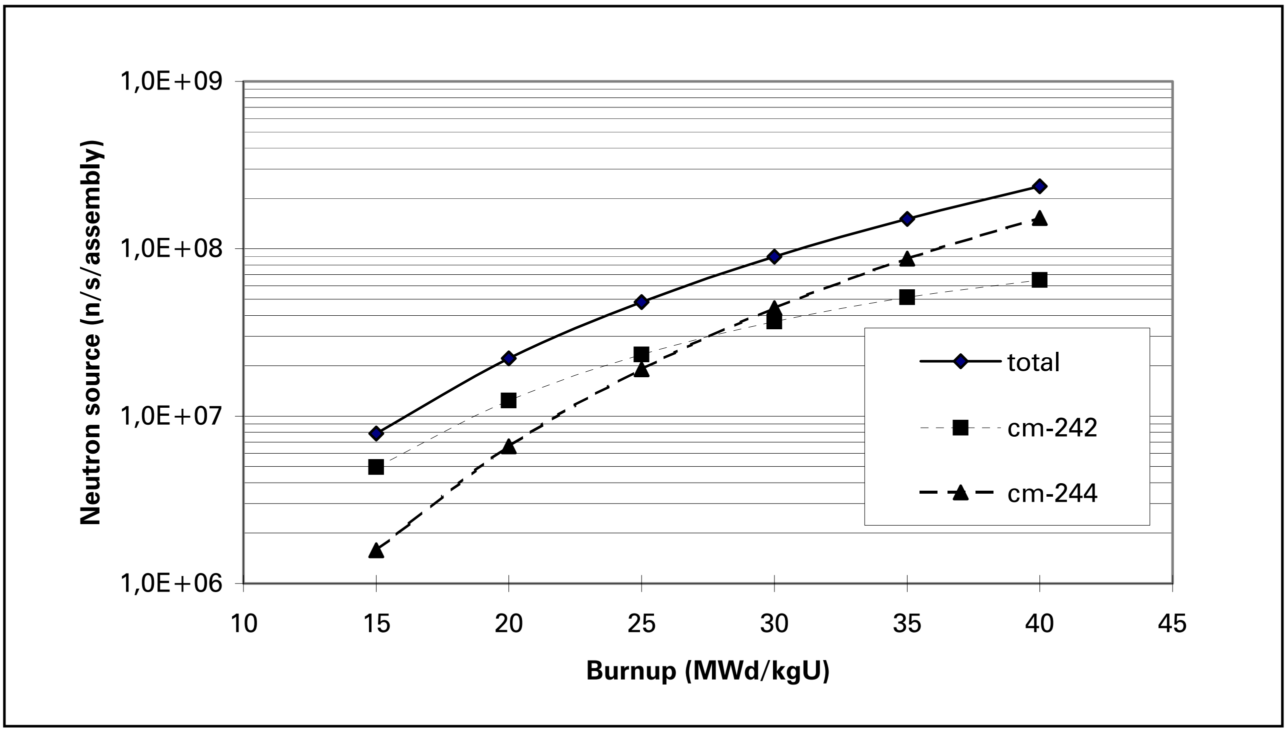

Figure 2. *Burnup-dependence of total neutron source, and ^{242}Cm and ^{244}Cm spontaneous fission neutron sources according to SAS2H/ORIGEN-S calculations.*

dle was irradiated in the reactor. The calculated primary ^{244}Cm neutron sources at discharge are shown in Table III.

The parameter *a* in the simple neutron source model (Equation 2) was calculated by dividing the computed primary ^{244}Cm neutron source at discharge by $(BU)^b$, where *b* was 5.03. The enrichment dependence of *a* is shown in Figure 3. For the same burnup, lower enrichment gave higher neutron source. The value of the *b* parameter was determined at 3 w-% enrichment, and the simple ^{244}Cm neutron source model was able to give good results around and above that enrichment. The curves diverge at lower enrichments, which reveals the inadequacy of the model.

Table III. *The calculated enrichment dependent ^{244}Cm neutron source at discharge (n/s/assembly).*

Enrichment	Burnup (MWd/kgU)		
(w-%)	20	30	40
1.6	$1.60 \cdot 10^7$	$9.83 \cdot 10^7$	$3.00 \cdot 10^8$
2.0	$9.78 \cdot 10^6$	$6.77 \cdot 10^7$	$2.30 \cdot 10^8$
2.4	$6.23 \cdot 10^6$	$4.66 \cdot 10^7$	$1.74 \cdot 10^8$
2.8	$4.16 \cdot 10^6$	$3.27 \cdot 10^7$	$1.30 \cdot 10^8$
3.2	$2.89 \cdot 10^6$	$2.34 \cdot 10^7$	$9.72 \cdot 10^7$
3.6	$2.09 \cdot 10^6$	$1.72 \cdot 10^7$	$7.35 \cdot 10^7$
4.0	$1.56 \cdot 10^6$	$1.30 \cdot 10^7$	$5.63 \cdot 10^7$
4.4	$1.19 \cdot 10^6$	$1.00 \cdot 10^7$	$4.39 \cdot 10^7$

4.4 Void fraction dependence

The void fraction dependence of the primary ^{244}Cm neutron source at discharge was studied by a serie of SAS2H/ORIGEN-S calculations. The coolant void fraction inside the assembly varied from 0 to 0.8, while the channel void fraction was set 0. The fuel bundle was a common 8x8 BWR fuel without burnable absorber rods. A uniform enrichment of 3.0 w-% was applied. The parameter study was carried out at three different values of discharge burnup: 20, 30 and 40 MWd/kgU. Average power of the bundle was set 4.5 MW, and the desired discharge burnup was achieved by adjusting the time that the bundle spent in the reactor. The calculated primary ^{244}Cm neutron sources at discharge are shown in Table IV.

Table IV. *The calculated void fraction dependent ^{244}Cm neutron source at discharge (n/s/assembly).*

	Burnup (MWd/kgU)		
Void fraction	20	30	40
0.0	$3.45 \cdot 10^6$	$2.76 \cdot 10^7$	$1.12 \cdot 10^8$
0.2	$4.52 \cdot 10^6$	$3.35 \cdot 10^7$	$1.28 \cdot 10^8$
0.4	$6.19 \cdot 10^6$	$4.22 \cdot 10^7$	$1.49 \cdot 10^8$
0.6	$8.87 \cdot 10^6$	$5.46 \cdot 10^7$	$1.77 \cdot 10^8$
0.8	$1.32 \cdot 10^7$	$7.19 \cdot 10^7$	$2.10 \cdot 10^8$

The parameter *a* in the simple ^{244}Cm neutron source model (Equation 2) was calculated by di-

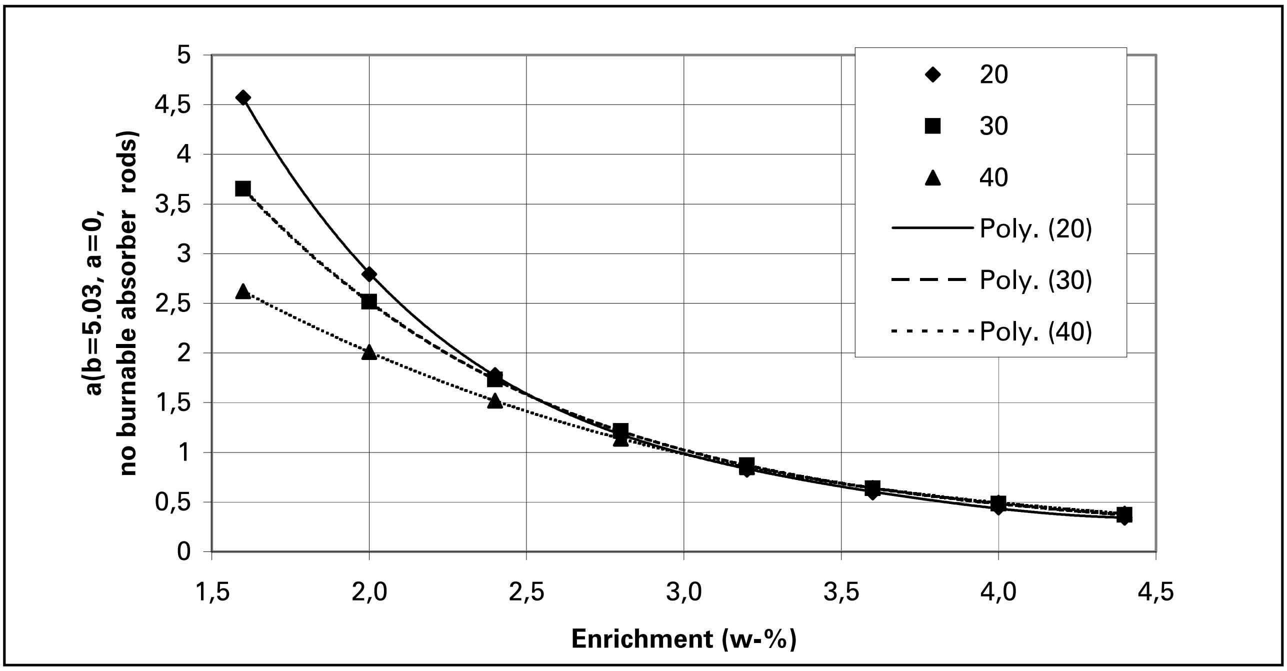

Figure 3. *Enrichment dependence of the parameter a in simple ^{244}Cm neutron source model.*

viding the computed primary ^{244}Cm neutron source at discharge by $(BU)^b$, where b was 5.03. The void fraction dependence of a is shown in Figure 4. Void fraction had a strong influence on buildup of the ^{244}Cm in BWR: for the same burnup higher void fraction gave a higher neutron source. The value of b was determined at zero void fraction, where the value of the parameter a was the same for all three burnups. The curves diverge at higher void fractions, which reveals the inadequacy of the simple ^{244}Cm neutron source model.

4.5 Effect of the amount of Gd_2O_3 in burnable absorber rods

The effect of the amount of Gd_2O_3 in the burnable absorber rods on the primary ^{244}Cm neutron source at discharge was studied by a serie of SAS2H/ORIGEN-S calculations. The fuel bundle was a common 8x8 BWR fuel bundle with four burnable absorber rods. The amount of Gd_2O_3 in burnable absorber rods was varied from 2.0 up to 4.0 w-%. The void fraction was set 0. The parameter study was performed at three different discharge burnups: 20, 30 and 40 MWd/kgU. Average power of the bundle was set 4.5 MW, and the desired discharge burnup was achieved by adjusting the time that the bundle was irradiated in the reactor. The calculated primary ^{244}Cm neutron sources at discharge are shown in Table V. According to the results the primary ^{244}Cm neutron source at discharge is only weakly dependent on the amount of Gd_2O_3 in the burnable absorber rods.

Table V. *The calculated ^{244}Cm neutron source at discharge (n/s/assembly).*

Gd_2O_3 in BA rods	Burnup (MWd/kgU)		
(w-%)	20	30	40
2.0	$3.58 \cdot 10^6$	$2.80 \cdot 10^7$	$1.13 \cdot 10^8$
2.5	$3.62 \cdot 10^6$	$2.81 \cdot 10^7$	$1.13 \cdot 10^7$
3.0	$3.64 \cdot 10^6$	$2.82 \cdot 10^7$	$1.13 \cdot 10^7$
3.5	$3.66 \cdot 10^6$	$2.83 \cdot 10^7$	$1.13 \cdot 10^7$
4.0	$3.68 \cdot 10^6$	$2.84 \cdot 10^7$	$1.13 \cdot 10^7$

4.6 Axial neutron source distribution

The fuel bundle neutron source has an axial distribution because burnup and void fraction vary axially. The axial distribution of the ^{244}Cm in ID6124 BWR fuel assembly was calculated with SAS2H/ORIGEN-S using the node-wise (25 nodes axially, 0=bottom) burnup and void fraction data acquired from SIMULATE-3 runs. The ID6124 bundle is a 8×8 BWR fuel bundle with four burnable absorber rods. Its average enrichment is 2.75 w-%, and its average burnup at discharge is

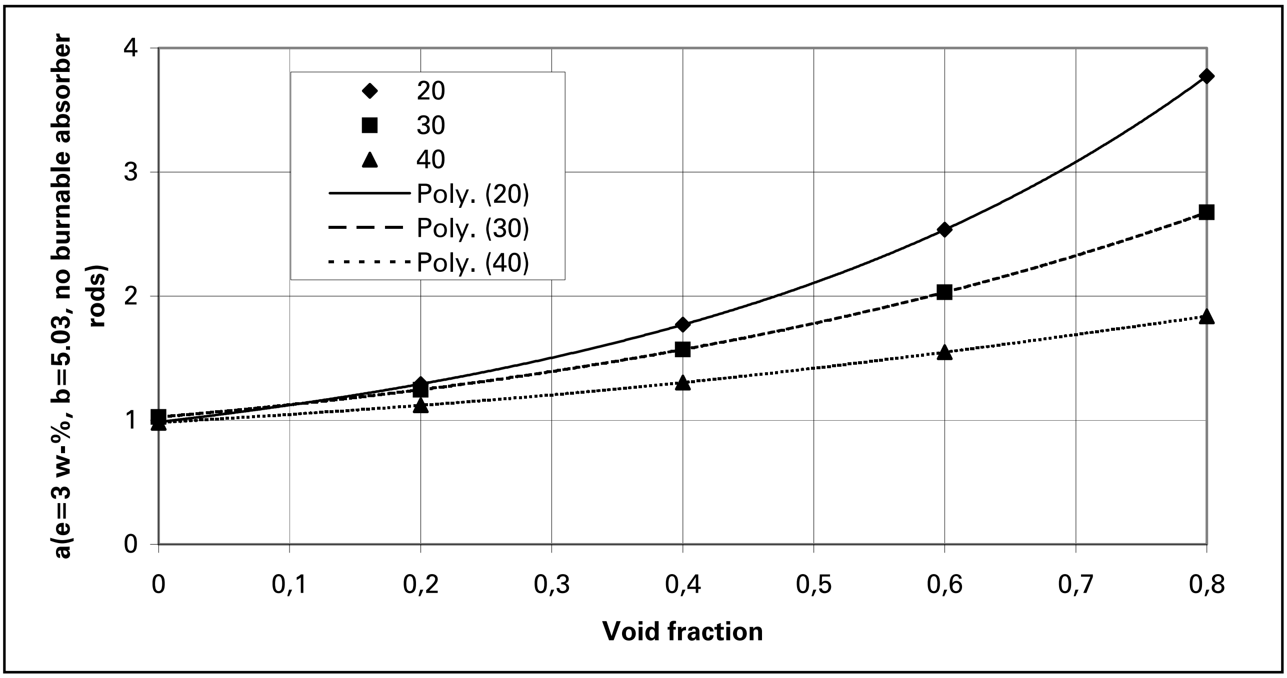

Figure 4. *Void fraction dependence of the parameter in a simple ^{244}Cm neutron source model.*

37.8 MWd/kgU. The axial variation in gadolinium content was not taken into account in the calculations since the amount of the burnable absorber does not significantly influence the magnitude of the neutron source. The normalized calculated axial ^{244}Cm distribution and the assumed discharge burnup of the ID6124 BWR bundle are shown in Figure 5. The figure implies that the higher void fraction in the upper part of the fuel bundle moves the maximum axial neutron source towards the top of the assembly.

4.7 Pinwise neutron source distribution

There is also a radial or more likely pinwise neutron source distribution in the fuel bundle. This is due to the variations in the pin enrichment and the irradiation conditions (neutron flux) of each pin. In this work the pinwise distribution of the ^{244}Cm concentration was studied by CASMO-4. However, it was realised that the calculation method was not valid for studying the pinwise distribution: CASMO-4 is a fuel assembly burnup code that assumes reflecting boundary conditions at the bundle boundaries. The surrounding fuel bundles or control rods affect the buildup of neutron source. In order to soundly assess the pinwise neutron source distribution one should use a core simulator coupled with a pin power reconstruction model.

4.8 Assessment of the neutron source of the Olkiluoto BWR bundles

The neutron source of nine Olkiluoto BWR fuel bundles were calculated with SAS2H/ORIGEN-S using as exact models and input data as possible. The input data included data about the fuel bundle type, average burnup and average void fraction from SIMULATE-3 runs and cooling time. The neutron emission of the nine fuel bundles were measured in November 1998 under Task JNT 1077 in the KPA Store in Olkiluoto with IAEA BWR Fork detector coupled with a compact underwater gamma spectrometer. The measured neutron emissions are plotted against calculationally assessed neutron sources in Figure 6. There is a clear correlation between the measured neutron emission and the calculated neutron source. However, to be precise the multiplication of the neutrons in the fuel matrix should also be taken into account. The isotopic contents of the spent fuel affect the neutron emission and the effective source in a subcritical source is

$$S_{eff} = \frac{S_p}{1-k} \quad (3),$$

where S_p is the primary ^{244}Cm neutron source,
k is the multiplication factor of the system.

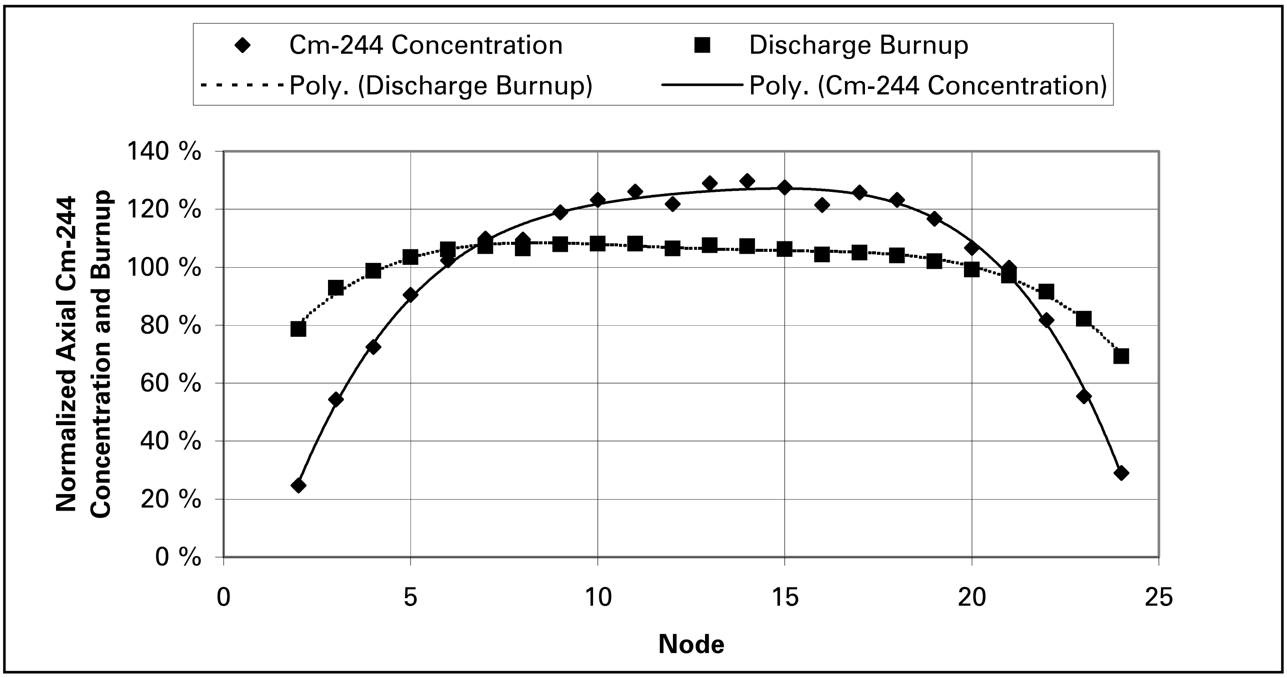

Figure 5. *The calculated axial distribution of the Cm-244 concentration and assumed discharge burnup.*

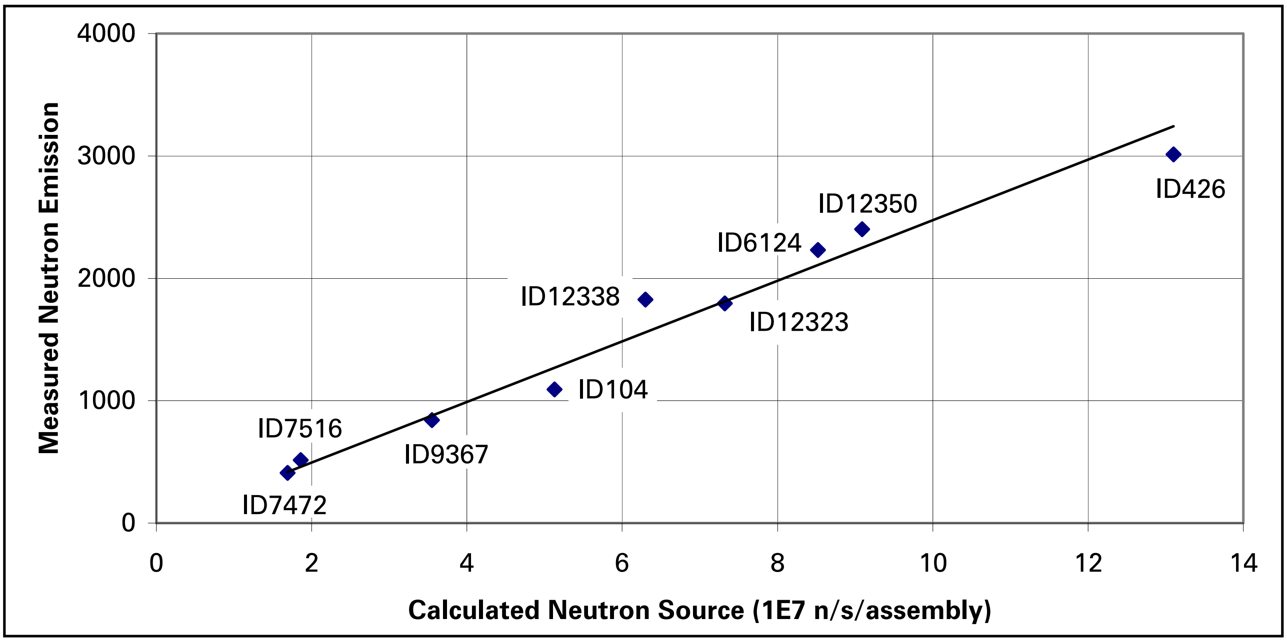

Figure 6. *Measured neutron emission vs. calculationally assessed neutron source.*

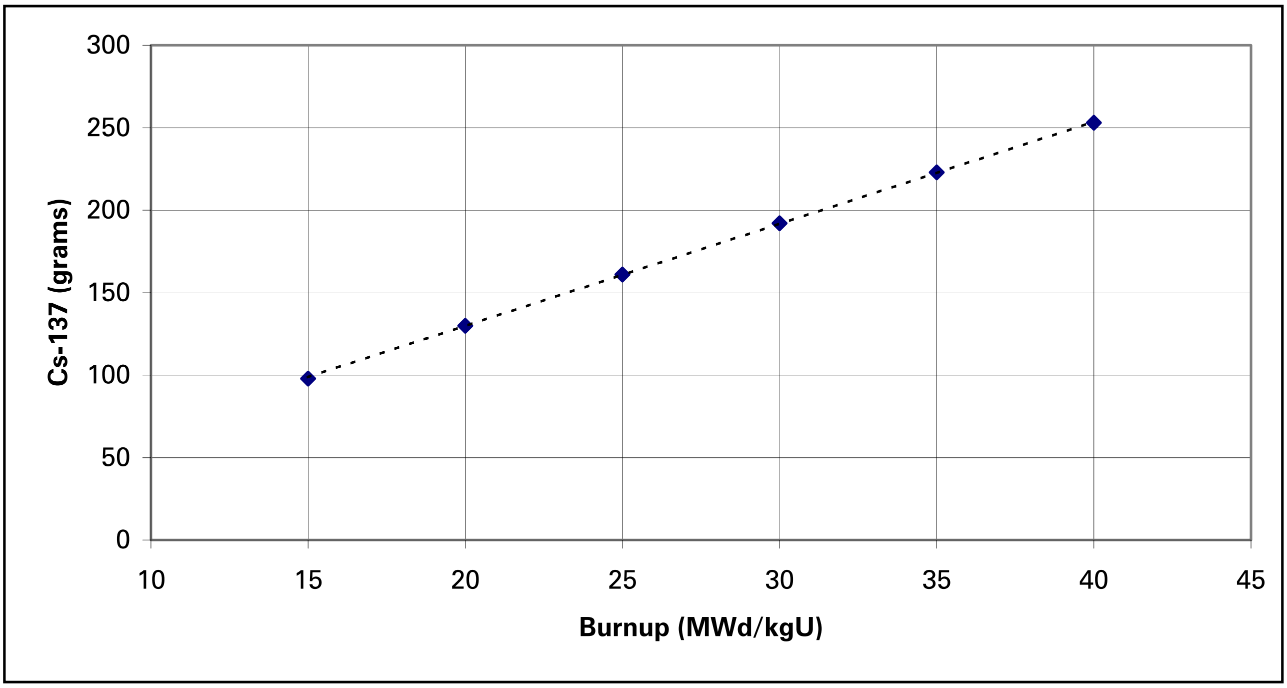

Figure 7. *Amount of ^{137}Cs as a function of burnup according to SAS2H/ORIGEN-S.*

5 ASSESSMENT OF THE ^{137}Cs AND ^{134}Cs GAMMA SOURCES

Spent LWR fuel contains a variety of gamma emitting nuclides (Appendix D). However, only ^{137}Cs and ^{134}Cs gamma peaks stood out from the gamma spectrum measured during the underwater gamma spectrometry measurement campaign [2]. The fission product contents of the spent fuel is closely correlated to its burnup. ^{137}Cs is mainly produced directly from fission but ^{134}Cs is mainly produced from neutron reaction with ^{133}Cs.

The SAS2H/ORIGEN-S runs performed to study the neutron source of spent BWR fuel also produced information about the ^{137}Cs and ^{134}Cs contents of the fuel. As expected, the only significant relations between the amount of the cesium isotopes and the parameters studied were those of burnup and cooling time decay (no relation between the amount of the cesium isotopes and enrichment, void fraction or the amount of burnable absorber). The calculated ^{137}Cs and ^{134}Cs contents of the spent BWR fuel are shown in Figures 7 and 8. The results imply that ^{137}Cs is a very good indicator of the fuel burnup; and that ^{134}Cs is not as good because its production increases at higher burnups. Anyhow, the ratio between the amount of ^{134}Cs and ^{137}Cs can be used to estimate the cooling time T_c according to formula

$$T_c = 3.198 \ln\left(\frac{r_0}{r_c}\right) \quad (4),$$

where r_0 is the ratio between the amount of ^{134}Cs and ^{137}Cs at discharge,
r_c is the ratio after T_c years of cooling.

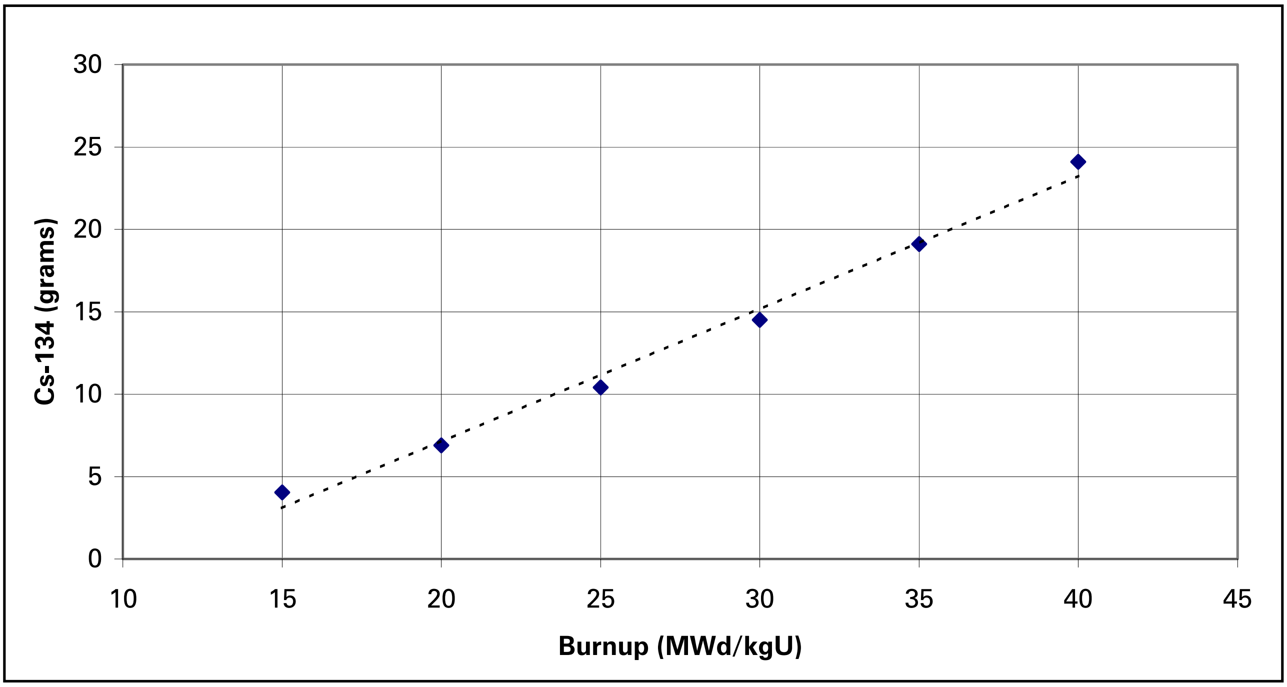

Figure 8. *Amount of ^{134}Cs as a function of burnup according to SAS2H/ORIGEN-S.*

6 CONCLUSIONS

SAS2H/ORIGEN-S was succesfully applied to assess the neutron and the gamma sources of the spent BWR and VVER-440 fuel. CASMO-4 was applied as a reference code to estimate the neutron source of the BWR fuel. The parameter studies performed gave valuable information about what has to be taken into account in order to accurately assess the neutron and gamma sources of the spent LWR fuel.

The neutron emission of the spent LWR fuel is dominantly from spontaneous fission of ^{244}Cm at cooling times longer than two years. Because of the long half-life of ^{244}Cm, neutron emission remains a characteristic feature of the spent LWR fuel even after relatively long cooling times. The parameters that affect the ^{244}Cm concentration at discharge are fuel bundle type, burnup, and in case of BWR also void fraction. The power history has only a slight influence on buildup of ^{244}Cm. The axial distribution of the neutron source can be determined if all the required input data is available. The accurate determination of the pinwise neutron source distribution is a demanding task.

Spent LWR fuel contains a variety of gamma emitting nuclides. However, only ^{137}Cs and ^{134}Cs gamma peaks stand out from the gamma spectrum. According to the SAS2H/ORIGEN-S results ^{137}Cs is a better indicator of the burnup than ^{134}Cs. Anyhow, the ratio between the amount of ^{134}Cs and ^{137}Cs can be used to estimate the cooling time.

Modern BWR fuel assemblies are highly heterogeneous: there are pins with different enrichments, burnable absorbers are used, and also axial variations (for example part length fuel rods, axial variation in the amount of burnable absorber, natural uranium end parts). Moreover, control rods affect the way the fuel is burnt in the reactor. Thus, assessment of the radiation source distribution of BWR fuel bundle is a relatively complicated task and some uncertainty is to be accepted.

When neutron or gamma radiation is being detected one should take into account the fact that in addition to the primary source particles there are secondary particles induced by the primary ones (for example, the fuel matrix of the spent fuel is acting as a neutron multiplier). Thus, for sound interpretation of the detector signals the production of the secondary particles should also be considered. This requires knowledge of the spatial source distribution, and rigorous modelling of both the spent fuel matrix and the detection geometry.

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APPENDIX A

NUCLIDE INVENTORY OF SPENT BWR FUEL

bu= 37816, bwr, 5 cyc, e=2.746, vf=0.426, 8x8-e2-275-1-200-10
actinides
decay, following reactor irradiation identified by:
power=4.31mw, burnup=6299.mwd, flux=3.43E+13n/cm**2-sec
nuclide concentrations, grams
basis =single reactor assembly

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
he 4	1.45E-01	2.32E-01	2.66E-01	2.99E-01	3.31E-01	3.62E-01	3.94E-01
pb208	1.23E-07	6.79E-07	1.97E-06	3.93E-06	6.35E-06	9.07E-06	1.20E-05
th228	3.36E-07	1.24E-06	2.23E-06	2.99E-06	3.50E-06	3.80E-06	3.96E-06
th230	1.65E-06	6.42E-06	1.54E-05	2.86E-05	4.59E-05	6.71E-05	9.24E-05
th232	5.50E-05	9.51E-05	1.35E-04	1.75E-04	2.15E-04	2.56E-04	2.96E-04
th234	2.28E-06	2.28E-06	2.28E-06	2.28E-06	2.28E-06	2.28E-06	2.28E-06
pa231	3.98E-06	5.64E-06	7.29E-06	8.94E-06	1.06E-05	1.22E-05	1.39E-05
pa233	2.73E-06	2.74E-06	2.75E-06	2.76E-06	2.77E-06	2.78E-06	2.79E-06
u232	3.97E-05	9.38E-05	1.24E-04	1.40E-04	1.48E-04	1.51E-04	1.52E-04
u233	7.17E-05	1.32E-04	1.92E-04	2.53E-04	3.14E-04	3.75E-04	4.36E-04
u234	4.35E-01	1.11E+00	1.79E+00	2.45E+00	3.11E+00	3.75E+00	4.38E+00
u235	7.60E+02	7.60E+02	7.60E+02	7.60E+02	7.60E+02	7.60E+02	7.60E+02
u236	6.15E+02	6.15E+02	6.15E+02	6.15E+02	6.15E+02	6.15E+02	6.16E+02
u237	1.19E+00	6.49E-06	5.82E-06	5.22E-06	4.69E-06	4.21E-06	3.78E-06
u238	1.57E+05	1.57E+05	1.57E+05	1.57E+05	1.57E+05	1.57E+05	1.57E+05
np236	1.38E-04	1.38E-04	1.38E-04	1.38E-04	1.38E-04	1.38E-04	1.38E-04
np237	7.95E+01	8.08E+01	8.10E+01	8.12E+01	8.15E+01	8.19E+01	8.23E+01
np239	1.15E+01	2.74E-05	2.73E-05	2.73E-05	2.73E-05	2.73E-05	2.73E-05
pu236	1.36E-04	7.99E-05	4.68E-05	2.74E-05	1.60E-05	9.38E-06	5.49E-06
pu238	3.64E+01	3.92E+01	3.86E+01	3.79E+01	3.72E+01	3.66E+01	3.60E+01
pu239	8.84E+02	8.95E+02	8.95E+02	8.95E+02	8.95E+02	8.95E+02	8.95E+02
pu240	4.56E+02	4.56E+02	4.57E+02	4.58E+02	4.59E+02	4.59E+02	4.60E+02
pu241	2.38E+02	2.14E+02	1.92E+02	1.72E+02	1.55E+02	1.39E+02	1.25E+02
pu242	1.27E+02	1.27E+02	1.27E+02	1.27E+02	1.27E+02	1.27E+02	1.27E+02
am241	1.05E+01	3.49E+01	5.66E+01	7.61E+01	9.35E+01	1.09E+02	1.23E+02
am242m	2.13E-01	2.11E-01	2.08E-01	2.06E-01	2.04E-01	2.02E-01	1.99E-01
am242	2.14E-02	2.72E-06	2.69E-06	2.66E-06	2.63E-06	2.60E-06	2.57E-06
am243	3.18E+01	3.18E+01	3.18E+01	3.18E+01	3.18E+01	3.18E+01	3.18E+01
cm242	3.37E+00	1.05E-01	3.75E-03	6.37E-04	5.35E-04	5.26E-04	5.20E-04
cm243	1.09E-01	1.03E-01	9.75E-02	9.23E-02	8.74E-02	8.28E-02	7.84E-02
cm244	1.23E+01	1.13E+01	1.03E+01	9.49E+00	8.71E+00	7.99E+00	7.33E+00
cm245	4.57E-01	4.57E-01	4.57E-01	4.57E-01	4.56E-01	4.56E-01	4.56E-01
cm246	8.85E-02	8.85E-02	8.85E-02	8.84E-02	8.84E-02	8.84E-02	8.83E-02
cm247	1.85E-03	1.85E-03	1.85E-03	1.85E-03	1.85E-03	1.85E-03	1.85E-03
cm248	2.18E-04	2.18E-04	2.18E-04	2.18E-04	2.18E-04	2.18E-04	2.18E-04
cf249	6.02E-07	3.57E-06	4.06E-06	4.13E-06	4.13E-06	4.11E-06	4.09E-06
total	1.61E+05	1.61E+05	1.61E+05	1.61E+05	1.61E+05	1.61E+05	1.61E+05

bu=37816, bwr, 5 cyc, e=2.746, vf=0.426, 8x8-e2-275-1-200-10
fission products
decay, following reactor irradiation identified by:
power=4.31mw, burnup=6299.mwd, flux=3.43E+13n/cm**2-sec
nuclide concentrations, grams basis = single reactor assembly

h 3	9.17E-03	8.09E-03	7.13E-03	6.29E-03	5.54E-03	4.89E-03	4.31E-03
li 6	2.85E-05	2.85E-05	2.85E-05	2.85E-05	2.85E-05	2.85E-05	2.85E-05
li 7	1.92E-06	1.92E-06	1.92E-06	1.92E-06	1.92E-06	1.92E-06	1.92E-06
be 9	3.71E-06	3.71E-06	3.71E-06	3.71E-06	3.71E-06	3.71E-06	3.71E-06
be 10	2.47E-05	2.47E-05	2.47E-05	2.47E-05	2.47E-05	2.47E-05	2.47E-05
c 14	5.00E-06	5.00E-06	4.99E-06	4.99E-06	4.99E-06	4.99E-06	4.99E-06
zn 70	1.80E-06	1.80E-06	1.80E-06	1.80E-06	1.80E-06	1.80E-06	1.80E-06
ga 71	1.71E-05	1.71E-05	1.71E-05	1.71E-05	1.71E-05	1.71E-05	1.71E-05
ge 72	1.01E-03	1.02E-03	1.02E-03	1.02E-03	1.02E-03	1.02E-03	1.02E-03
ge 73	2.84E-03	2.84E-03	2.84E-03	2.84E-03	2.84E-03	2.84E-03	2.84E-03
ge 74	2.49E-03	2.49E-03	2.49E-03	2.49E-03	2.49E-03	2.49E-03	2.49E-03
as 75	2.11E-02	2.11E-02	2.11E-02	2.11E-02	2.11E-02	2.11E-02	2.11E-02
ge 76	6.28E-02	6.28E-02	6.28E-02	6.28E-02	6.28E-02	6.28E-02	6.28E-02
se 76	8.14E-04	8.16E-04	8.16E-04	8.16E-04	8.16E-04	8.16E-04	8.16E-04
se 77	1.41E-01	1.41E-01	1.41E-01	1.41E-01	1.41E-01	1.41E-01	1.41E-01
se 78	4.93E-01	4.93E-01	4.93E-01	4.93E-01	4.93E-01	4.93E-01	4.93E-01
se 79	9.11E-01	9.11E-01	9.11E-01	9.11E-01	9.11E-01	9.11E-01	9.11E-01
br 79	4.79E-06	9.08E-06	1.34E-05	1.77E-05	2.20E-05	2.62E-05	3.05E-05
se 80	2.48E+00	2.48E+00	2.48E+00	2.48E+00	2.48E+00	2.48E+00	2.48E+00
kr 80	1.24E-05	1.24E-05	1.24E-05	1.24E-05	1.24E-05	1.24E-05	1.24E-05
br 81	3.66E+00	3.66E+00	3.66E+00	3.66E+00	3.66E+00	3.66E+00	3.66E+00
kr 81	1.05E-06	1.05E-06	1.05E-06	1.05E-06	1.05E-06	1.05E-06	1.05E-06
se 82	5.87E+00	5.87E+00	5.87E+00	5.87E+00	5.87E+00	5.87E+00	5.87E+00
kr 82	1.42E-01	1.42E-01	1.42E-01	1.42E-01	1.42E-01	1.42E-01	1.42E-01
kr 83	6.83E+00	6.83E+00	6.83E+00	6.83E+00	6.83E+00	6.83E+00	6.83E+00
kr 84	2.11E+01	2.11E+01	2.11E+01	2.11E+01	2.11E+01	2.11E+01	2.11E+01

NUCLIDE INVENTORY OF SPENT BWR FUEL **APPENDIX A**

kr 85	3.76E+00	3.25E+00	2.81E+00	2.43E+00	2.11E+00	1.82E+00	1.58E+00
rb 85	1.67E+01	1.72E+01	1.76E+01	1.80E+01	1.83E+01	1.86E+01	1.88E+01
kr 86	3.25E+01	3.25E+01	3.25E+01	3.25E+01	3.25E+01	3.25E+01	3.25E+01
sr 86	8.13E-02	8.42E-02	8.42E-02	8.42E-02	8.42E-02	8.42E-02	8.42E-02
rb 87	4.23E+01	4.23E+01	4.23E+01	4.23E+01	4.23E+01	4.23E+01	4.23E+01
sr 87	4.80E-04	4.80E-04	4.80E-04	4.80E-04	4.80E-04	4.80E-04	4.80E-04
sr 88	6.05E+01	6.05E+01	6.05E+01	6.05E+01	6.05E+01	6.05E+01	6.05E+01
y 89	7.77E+01	8.04E+01	8.04E+01	8.04E+01	8.04E+01	8.04E+01	8.04E+01
sr 90	9.21E+01	8.72E+01	8.25E+01	7.81E+01	7.39E+01	6.99E+01	6.62E+01
y 90	2.48E-02	2.27E-02	2.14E-02	2.03E-02	1.92E-02	1.82E-02	1.72E-02
zr 90	6.41E+00	1.14E+01	1.60E+01	2.05E+01	2.47E+01	2.86E+01	3.24E+01
zr 91	1.01E+02	1.06E+02	1.06E+02	1.06E+02	1.06E+02	1.06E+02	1.06E+02
zr 92	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02	1.14E+02
zr 93	8.51E+01	8.52E+01	8.52E+01	8.52E+01	8.52E+01	8.52E+01	8.52E+01
nb 93	9.17E-06	2.17E-05	4.11E-05	6.66E-05	9.77E-05	1.34E-04	1.75E-04
nb 93m	9.31E-05	1.67E-04	2.34E-04	2.95E-04	3.50E-04	4.01E-04	4.46E-04
zr 94	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02	1.41E+02
nb 94	9.77E-05	9.77E-05	9.77E-05	9.77E-05	9.77E-05	9.77E-05	9.77E-05
mo 95	1.25E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02
zr 96	1.49E+02	1.49E+02	1.49E+02	1.49E+02	1.49E+02	1.49E+02	1.49E+02
mo 96	9.51E+00	9.51E+00	9.51E+00	9.51E+00	9.51E+00	9.51E+00	9.51E+00
mo 97	1.42E+02	1.42E+02	1.42E+02	1.42E+02	1.42E+02	1.42E+02	1.42E+02
mo 98	1.56E+02	1.56E+02	1.56E+02	1.56E+02	1.56E+02	1.56E+02	1.56E+02
tc 98	1.35E-03	1.35E-03	1.35E-03	1.35E-03	1.35E-03	1.35E-03	1.35E-03
tc 99	1.45E+02	1.45E+02	1.45E+02	1.45E+02	1.45E+02	1.45E+02	1.45E+02
ru 99	6.28E-03	7.36E-03	8.43E-03	9.50E-03	1.06E-02	1.16E-02	1.27E-02
mo100	1.78E+02	1.78E+02	1.78E+02	1.78E+02	1.78E+02	1.78E+02	1.78E+02
ru100	2.48E+01	2.48E+01	2.48E+01	2.48E+01	2.48E+01	2.48E+01	2.48E+01
ru101	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02	1.46E+02
ru102	1.53E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02
rh102	2.16E-04	1.26E-04	7.40E-05	4.33E-05	2.54E-05	1.48E-05	8.69E-06
rh103	7.98E+01	8.57E+01	8.57E+01	8.57E+01	8.57E+01	8.57E+01	8.57E+01
ru104	1.13E+02	1.13E+02	1.13E+02	1.13E+02	1.13E+02	1.13E+02	1.13E+02
pd104	5.44E+01	5.44E+01	5.44E+01	5.44E+01	5.44E+01	5.44E+01	5.44E+01
pd105	8.20E+01	8.22E+01	8.22E+01	8.22E+01	8.22E+01	8.22E+01	8.22E+01
ru106	2.68E+01	5.82E+00	1.26E+00	2.75E-01	5.97E-02	1.30E-02	2.82E-03
pd106	5.41E+01	7.51E+01	7.96E+01	8.06E+01	8.08E+01	8.09E+01	8.09E+01
pd107	5.02E+01	5.02E+01	5.02E+01	5.02E+01	5.02E+01	5.02E+01	5.02E+01
ag107	9.41E-06	2.14E-05	3.34E-05	4.54E-05	5.74E-05	6.94E-05	8.14E-05
pd108	3.31E+01	3.31E+01	3.31E+01	3.31E+01	3.31E+01	3.31E+01	3.31E+01
ag108m	4.74E-05	4.68E-05	4.62E-05	4.57E-05	4.51E-05	4.46E-05	4.40E-05
cd108	4.88E-05	4.89E-05	4.89E-05	4.89E-05	4.90E-05	4.90E-05	4.91E-05
ag109	1.89E+01	1.89E+01	1.89E+01	1.89E+01	1.89E+01	1.89E+01	1.89E+01
pd110	9.91E+00	9.91E+00	9.91E+00	9.91E+00	9.91E+00	9.91E+00	9.91E+00
ag110m	1.62E-01	1.67E-02	1.72E-03	1.78E-04	1.83E-05	1.89E-06	1.95E-07
cd110	1.08E+01	1.09E+01	1.10E+01	1.10E+01	1.10E+01	1.10E+01	1.10E+01
cd111	5.12E+00	5.17E+00	5.17E+00	5.17E+00	5.17E+00	5.17E+00	5.17E+00
cd112	2.62E+00	2.62E+00	2.62E+00	2.62E+00	2.62E+00	2.62E+00	2.62E+00
cd113	1.45E-02	1.49E-02	1.49E-02	1.49E-02	1.49E-02	1.49E-02	1.49E-02
cd113m	2.85E-02	2.55E-02	2.29E-02	2.05E-02	1.83E-02	1.64E-02	1.47E-02
in113	2.39E-03	5.36E-03	8.01E-03	1.04E-02	1.25E-02	1.44E-02	1.61E-02
cd114	2.57E+00	2.57E+00	2.57E+00	2.57E+00	2.57E+00	2.57E+00	2.57E+00
sn114	1.66E-04	1.85E-04	1.85E-04	1.85E-04	1.85E-04	1.85E-04	1.85E-04
in115	2.57E-01	2.61E-01	2.61E-01	2.61E-01	2.61E-01	2.61E-01	2.61E-01
sn115	3.60E-02	3.61E-02	3.61E-02	3.61E-02	3.61E-02	3.61E-02	3.61E-02
cd116	1.01E+00	1.01E+00	1.01E+00	1.01E+00	1.01E+00	1.01E+00	1.01E+00
sn116	5.19E-01	5.19E-01	5.19E-01	5.19E-01	5.19E-01	5.19E-01	5.19E-01
sn117	9.77E-01	9.78E-01	9.78E-01	9.78E-01	9.78E-01	9.78E-01	9.78E-01
sn118	7.71E-01	7.71E-01	7.71E-01	7.71E-01	7.71E-01	7.71E-01	7.71E-01
sn119	8.07E-01	8.10E-01	8.11E-01	8.11E-01	8.11E-01	8.11E-01	8.11E-01
sn120	7.88E-01	7.88E-01	7.88E-01	7.88E-01	7.88E-01	7.88E-01	7.88E-01
sn121m	9.07E-03	8.82E-03	8.57E-03	8.33E-03	8.10E-03	7.87E-03	7.66E-03
sb121	7.70E-01	7.71E-01	7.71E-01	7.71E-01	7.72E-01	7.72E-01	7.72E-01
sn122	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.02E+00	1.02E+00
te122	6.37E-02	6.41E-02	6.41E-02	6.41E-02	6.41E-02	6.41E-02	6.41E-02
sb123	9.04E-01	9.13E-01	9.13E-01	9.13E-01	9.13E-01	9.13E-01	9.13E-01
te123	5.90E-04	7.60E-04	7.61E-04	7.61E-04	7.61E-04	7.61E-04	7.61E-04
sn124	1.71E+00	1.71E+00	1.71E+00	1.71E+00	1.71E+00	1.71E+00	1.71E+00
te124	5.28E-02	5.97E-02	5.97E-02	5.97E-02	5.97E-02	5.97E-02	5.97E-02
sb125	1.31E+00	7.44E-01	4.21E-01	2.38E-01	1.35E-01	7.64E-02	4.33E-02
te125	7.87E-01	1.36E+00	1.69E+00	1.88E+00	1.98E+00	2.04E+00	2.07E+00
te125m	1.70E-02	1.06E-02	5.98E-03	3.39E-03	1.92E-03	1.09E-03	6.15E-04
sn126	4.06E+00	4.05E+00	4.05E+00	4.05E+00	4.05E+00	4.05E+00	4.05E+00
te126	8.15E-02	8.22E-02	8.23E-02	8.23E-02	8.24E-02	8.25E-02	8.25E-02
i127	8.50E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00	8.73E+00
te128	1.79E+01	1.79E+01	1.79E+01	1.79E+01	1.79E+01	1.79E+01	1.79E+01

APPENDIX A

xe128	6.55E-01	6.55E-01	6.55E-01	6.55E-01	6.55E-01	6.55E-01	6.55E-01
i129	3.51E+01	3.54E+01	3.54E+01	3.54E+01	3.54E+01	3.54E+01	3.54E+01
xe129	4.93E-03	5.01E-03	5.02E-03	5.02E-03	5.02E-03	5.03E-03	5.03E-03
te130	6.98E+01	6.98E+01	6.98E+01	6.98E+01	6.98E+01	6.98E+01	6.98E+01
xe130	1.84E+00	1.84E+00	1.84E+00	1.84E+00	1.84E+00	1.84E+00	1.84E+00
xe131	7.18E+01	7.28E+01	7.28E+01	7.28E+01	7.28E+01	7.28E+01	7.28E+01
xe132	2.18E+02	2.19E+02	2.19E+02	2.19E+02	2.19E+02	2.19E+02	2.19E+02
ba132	4.45E-05	4.51E-05	4.51E-05	4.51E-05	4.51E-05	4.51E-05	4.51E-05
cs133	2.09E+02	2.11E+02	2.11E+02	2.11E+02	2.11E+02	2.11E+02	2.11E+02
xe134	2.86E+02	2.86E+02	2.86E+02	2.86E+02	2.86E+02	2.86E+02	2.86E+02
cs134	2.28E+01	1.07E+01	5.05E+00	2.38E+00	1.12E+00	5.28E-01	2.48E-01
ba134	1.32E+01	2.53E+01	3.09E+01	3.36E+01	3.49E+01	3.55E+01	3.57E+01
cs135	7.53E+01	7.53E+01	7.53E+01	7.53E+01	7.53E+01	7.53E+01	7.53E+01
ba135	1.26E-01	1.27E-01	1.27E-01	1.27E-01	1.27E-01	1.27E-01	1.27E-01
xe136	4.30E+02	4.30E+02	4.30E+02	4.30E+02	4.30E+02	4.30E+02	4.30E+02
ba136	4.30E+00	4.40E+00	4.40E+00	4.40E+00	4.40E+00	4.40E+00	4.40E+00
cs137	2.31E+02	2.19E+02	2.08E+02	1.98E+02	1.88E+02	1.78E+02	1.69E+02
ba137	1.30E+01	2.47E+01	3.58E+01	4.63E+01	5.62E+01	6.57E+01	7.47E+01
ba137m	3.55E-05	3.35E-05	3.18E-05	3.02E-05	2.87E-05	2.72E-05	2.58E-05
ba138	2.41E+02	2.41E+02	2.41E+02	2.41E+02	2.41E+02	2.41E+02	2.41E+02
la138	1.68E-03	1.68E-03	1.68E-03	1.68E-03	1.68E-03	1.68E-03	1.68E-03
la139	2.26E+02	2.26E+02	2.26E+02	2.26E+02	2.26E+02	2.26E+02	2.26E+02
ce140	2.42E+02	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.45E+02
pr141	2.02E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02	2.08E+02
ce142	2.11E+02	2.11E+02	2.11E+02	2.11E+02	2.11E+02	2.11E+02	2.11E+02
nd142	4.82E+00	4.83E+00	4.83E+00	4.83E+00	4.83E+00	4.83E+00	4.83E+00
nd143	1.33E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02
ce144	4.19E+01	5.72E+00	7.81E-01	1.07E-01	1.46E-02	1.99E-03	2.72E-04
nd144	2.13E+02	2.49E+02	2.54E+02	2.55E+02	2.55E+02	2.55E+02	2.55E+02
nd145	1.21E+02	1.21E+02	1.21E+02	1.21E+02	1.21E+02	1.21E+02	1.21E+02
pm145	4.00E-06	3.93E-06	3.65E-06	3.36E-06	3.08E-06	2.82E-06	2.58E-06
nd146	1.33E+02	1.33E+02	1.33E+02	1.33E+02	1.33E+02	1.33E+02	1.33E+02
pm146	1.24E-03	9.40E-04	7.09E-04	5.36E-04	4.05E-04	3.06E-04	2.31E-04
sm146	1.73E-03	1.83E-03	1.91E-03	1.97E-03	2.01E-03	2.05E-03	2.07E-03
pm147	2.49E+01	1.43E+01	7.90E+00	4.37E+00	2.42E+00	1.34E+00	7.40E-01
sm147	1.70E+01	2.85E+01	3.49E+01	3.84E+01	4.04E+01	4.15E+01	4.20E+01
nd148	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01	6.92E+01
sm148	2.72E+01	2.75E+01	2.75E+01	2.75E+01	2.75E+01	2.75E+01	2.75E+01
sm149	3.08E-01	4.80E-01	4.80E-01	4.80E-01	4.80E-01	4.80E-01	4.80E-01
nd150	3.49E+01	3.49E+01	3.49E+01	3.49E+01	3.49E+01	3.49E+01	3.49E+01
sm150	6.11E+01	6.11E+01	6.11E+01	6.11E+01	6.11E+01	6.11E+01	6.11E+01
sm151	2.66E+00	2.64E+00	2.60E+00	2.55E+00	2.51E+00	2.47E+00	2.43E+00
eu151	3.31E-03	4.93E-02	9.46E-02	1.39E-01	1.83E-01	2.26E-01	2.68E-01
sm152	2.41E+01	2.41E+01	2.41E+01	2.41E+01	2.41E+01	2.41E+01	2.41E+01
eu152	8.08E-03	7.19E-03	6.40E-03	5.70E-03	5.07E-03	4.51E-03	4.02E-03
gd152	1.47E-02	1.50E-02	1.52E-02	1.54E-02	1.56E-02	1.57E-02	1.59E-02
eu153	2.46E+01	2.48E+01	2.48E+01	2.48E+01	2.48E+01	2.48E+01	2.48E+01
sm154	7.71E+00	7.71E+00	7.71E+00	7.71E+00	7.71E+00	7.71E+00	7.71E+00
eu154	4.94E+00	4.12E+00	3.44E+00	2.87E+00	2.40E+00	2.00E+00	1.67E+00
gd154	6.12E-01	1.43E+00	2.11E+00	2.68E+00	3.15E+00	3.55E+00	3.88E+00
eu155	1.15E+00	8.24E-01	5.91E-01	4.24E-01	3.05E-01	2.19E-01	1.57E-01
gd155	1.10E-02	3.35E-01	5.68E-01	7.35E-01	8.55E-01	9.41E-01	1.00E+00
gd156	1.98E+01	2.05E+01	2.05E+01	2.05E+01	2.05E+01	2.05E+01	2.05E+01
gd157	1.75E-02	2.03E-02	2.03E-02	2.03E-02	2.03E-02	2.03E-02	2.03E-02
gd158	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00	4.16E+00
tb159	5.14E-01	5.15E-01	5.15E-01	5.15E-01	5.15E-01	5.15E-01	5.15E-01
gd160	2.33E-01	2.33E-01	2.33E-01	2.33E-01	2.33E-01	2.33E-01	2.33E-01
dy160	5.29E-02	6.82E-02	6.83E-02	6.83E-02	6.83E-02	6.83E-02	6.83E-02
dy161	7.49E-02	7.63E-02	7.63E-02	7.63E-02	7.63E-02	7.63E-02	7.63E-02
dy162	6.25E-02	6.25E-02	6.25E-02	6.25E-02	6.25E-02	6.25E-02	6.25E-02
dy163	5.52E-02	5.52E-02	5.52E-02	5.52E-02	5.52E-02	5.52E-02	5.52E-02
dy164	1.19E-02	1.19E-02	1.19E-02	1.19E-02	1.19E-02	1.19E-02	1.19E-02
ho165	2.08E-02	2.08E-02	2.08E-02	2.08E-02	2.08E-02	2.08E-02	2.08E-02
ho166m	8.19E-05	8.18E-05	8.17E-05	8.16E-05	8.15E-05	8.14E-05	8.13E-05
er166	5.18E-03	5.20E-03	5.20E-03	5.20E-03	5.20E-03	5.20E-03	5.20E-03
er167	1.06E-04	1.06E-04	1.06E-04	1.06E-04	1.06E-04	1.06E-04	1.06E-04
er168	1.70E-04	1.70E-04	1.70E-04	1.70E-04	1.70E-04	1.70E-04	1.70E-04
yb171	4.49E-07	7.71E-07	9.14E-07	9.78E-07	1.01E-06	1.02E-06	1.03E-06
total	6.45E+03	6.45E+03	6.45E+03	6.45E+03	6.45E+03	6.45E+03	6.45E+03

NUCLIDE INVENTORY OF SPENT VVER-440 FUEL

APPENDIX B

vver440 fuel bu= 36 mwd/kgu
actinides
decay, following reactor irradiation identified by:
power= 4.96mw, burnup= 4464.mwd, flux=4.08E+13n/cm**2-sec
nuclide concentrations, grams
basis = single reactor assembly

	initial	60.8 d	121.7 d	182.5 d	243.3 d	304.2 d	365.0 d
he 4	3.77E-02	4.55E-02	5.18E-02	5.69E-02	6.11E-02	6.47E-02	6.76E-02
th230	6.58E-07	7.69E-07	8.93E-07	1.03E-06	1.18E-06	1.34E-06	1.52E-06
th232	2.74E-05	3.01E-05	3.28E-05	3.55E-05	3.82E-05	4.09E-05	4.36E-05
th234	1.69E-06	1.69E-06	1.69E-06	1.69E-06	1.69E-06	1.69E-06	1.69E-06
pa231	3.30E-06	3.50E-06	3.69E-06	3.88E-06	4.08E-06	4.27E-06	4.47E-06
pa233	1.98E-06	2.07E-06	2.09E-06	2.09E-06	2.09E-06	2.09E-06	2.09E-06
u232	1.76E-05	2.13E-05	2.49E-05	2.83E-05	3.15E-05	3.46E-05	3.76E-05
u233	6.82E-05	7.14E-05	7.47E-05	7.79E-05	8.12E-05	8.45E-05	8.77E-05
u234	2.27E-01	2.54E-01	2.82E-01	3.10E-01	3.38E-01	3.66E-01	3.95E-01
u235	1.17E+03	1.17E+03	1.17E+03	1.17E+03	1.17E+03	1.17E+03	1.17E+03
u236	5.58E+02	5.58E+02	5.58E+02	5.58E+02	5.58E+02	5.58E+02	5.58E+02
u237	1.47E+00	2.85E-03	1.08E-05	5.21E-06	5.16E-06	5.12E-06	5.08E-06
u238	1.17E+05	1.17E+05	1.17E+05	1.17E+05	1.17E+05	1.17E+05	1.17E+05
np236	8.70E-05	8.70E-05	8.70E-05	8.70E-05	8.70E-05	8.70E-05	8.70E-05
np237	6.02E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01	6.17E+01
np239	1.23E+01	1.27E-05	1.25E-05	1.25E-05	1.25E-05	1.25E-05	1.25E-05
pu236	9.69E-05	9.35E-05	8.99E-05	8.64E-05	8.30E-05	7.98E-05	7.66E-05
pu238	2.06E+01	2.12E+01	2.15E+01	2.17E+01	2.19E+01	2.20E+01	2.21E+01
pu239	7.19E+02	7.31E+02	7.31E+02	7.31E+02	7.31E+02	7.31E+02	7.31E+02
pu240	2.96E+02	2.96E+02	2.96E+02	2.96E+02	2.96E+02	2.96E+02	2.96E+02
pu241	1.76E+02	1.74E+02	1.73E+02	1.72E+02	1.70E+02	1.69E+02	1.67E+02
pu242	6.75E+01	6.75E+01	6.75E+01	6.75E+01	6.75E+01	6.75E+01	6.75E+01
am241	5.09E+00	6.50E+00	7.89E+00	9.28E+00	1.06E+01	1.20E+01	1.34E+01
am242m	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01
am242	1.40E-02	1.38E-06	1.38E-06	1.38E-06	1.38E-06	1.38E-06	1.37E-06
am243	1.45E+01	1.45E+01	1.45E+01	1.45E+01	1.45E+01	1.45E+01	1.45E+01
cm242	1.77E+00	1.38E+00	1.06E+00	8.20E-01	6.33E-01	4.89E-01	3.77E-01
cm243	4.70E-02	4.68E-02	4.66E-02	4.64E-02	4.62E-02	4.60E-02	4.58E-02
cm244	4.36E+00	4.35E+00	4.32E+00	4.29E+00	4.27E+00	4.24E+00	4.21E+00
cm245	1.55E-01	1.55E-01	1.55E-01	1.55E-01	1.55E-01	1.55E-01	1.55E-01
cm246	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.07E-02
cm247	3.53E-04	3.53E-04	3.53E-04	3.53E-04	3.53E-04	3.53E-04	3.53E-04
cm248	3.16E-05	3.16E-05	3.16E-05	3.16E-05	3.16E-05	3.16E-05	3.16E-05
total	1.20E+05	1.20E+05	1.20E+05	1.20E+05	1.20E+05	1.20E+05	1.20E+05

vver440 fuel bu= 36 mwd/ kgu
fission products
decay, following reactor irradiation identified by:
power=4.96mw, burnup=4464.mwd, flux=4.08E+13n/cm** 2-sec
nuclide concentrations, grams
basis = single reactor assembly

	initial	60.8 d	121.7 d	182.5 d	243.3 d	304.2 d	365.0 d
h 3	6.71E-03	6.64E-03	6.58E-03	6.52E-03	6.46E-03	6.40E-03	6.34E-03
li 6	2.58E-05	2.58E-05	2.58E-05	2.58E-05	2.58E-05	2.58E-05	2.58E-05
li 7	1.37E-06	1.37E-06	1.37E-06	1.37E-06	1.37E-06	1.37E-06	1.37E-06
be 9	2.64E-06	2.64E-06	2.64E-06	2.64E-06	2.64E-06	2.64E-06	2.64E-06
be 10	1.76E-05	1.76E-05	1.76E-05	1.76E-05	1.76E-05	1.76E-05	1.76E-05
c 14	3.57E-06	3.57E-06	3.57E-06	3.57E-06	3.57E-06	3.57E-06	3.57E-06
zn 70	1.09E-06	1.09E-06	1.09E-06	1.09E-06	1.09E-06	1.09E-06	1.09E-06
ga 71	1.06E-05	1.06E-05	1.06E-05	1.06E-05	1.06E-05	1.06E-05	1.06E-05
ge 72	6.85E-04	6.89E-04	6.89E-04	6.89E-04	6.89E-04	6.89E-04	6.89E-04
ge 73	1.97E-03	1.97E-03	1.97E-03	1.97E-03	1.97E-03	1.97E-03	1.97E-03
ge 74	1.72E-03	1.72E-03	1.72E-03	1.72E-03	1.72E-03	1.72E-03	1.72E-03
as 75	1.55E-02	1.55E-02	1.55E-02	1.55E-02	1.55E-02	1.55E-02	1.55E-02
ge 76	4.72E-02	4.72E-02	4.72E-02	4.72E-02	4.72E-02	4.72E-02	4.72E-02
se 76	4.87E-04	4.89E-04	4.89E-04	4.89E-04	4.89E-04	4.89E-04	4.89E-04
se 77	1.06E-01	1.06E-01	1.06E-01	1.06E-01	1.06E-01	1.06E-01	1.06E-01
se 78	3.48E-01	3.48E-01	3.48E-01	3.48E-01	3.48E-01	3.48E-01	3.48E-01
se 79	6.60E-01	6.60E-01	6.60E-01	6.60E-01	6.60E-01	6.60E-01	6.60E-01
br 79	2.12E-06	2.36E-06	2.59E-06	2.82E-06	3.05E-06	3.28E-06	3.51E-06
se 80	1.83E+00	1.83E+00	1.83E+00	1.83E+00	1.83E+00	1.83E+00	1.83E+00
kr 80	7.41E-06	7.42E-06	7.42E-06	7.42E-06	7.42E-06	7.42E-06	7.42E-06
br 81	2.70E+00	2.70E+00	2.70E+00	2.70E+00	2.70E+00	2.70E+00	2.70E+00
se 82	4.40E+00	4.40E+00	4.40E+00	4.40E+00	4.40E+00	4.40E+00	4.40E+00
kr 82	8.52E-02	8.56E-02	8.56E-02	8.56E-02	8.56E-02	8.56E-02	8.56E-02

kr 83	5.61E+00	5.61E+00	5.61E+00	5.61E+00	5.61E+00	5.61E+00	5.61E+00
kr 84	1.56E+01	1.56E+01	1.56E+01	1.56E+01	1.56E+01	1.56E+01	1.56E+01
kr 85	3.09E+00	3.06E+00	3.03E+00	3.00E+00	2.97E+00	2.93E+00	2.90E+00
rb 85	1.26E+01	1.27E+01	1.27E+01	1.27E+01	1.28E+01	1.28E+01	1.28E+01
kr 86	2.52E+01	2.52E+01	2.52E+01	2.52E+01	2.52E+01	2.52E+01	2.52E+01
sr 86	4.80E-02	5.07E-02	5.10E-02	5.10E-02	5.10E-02	5.10E-02	5.10E-02
rb 87	3.28E+01	3.28E+01	3.28E+01	3.28E+01	3.28E+01	3.28E+01	3.28E+01
sr 87	2.48E-04	2.48E-04	2.48E-04	2.48E-04	2.48E-04	2.48E-04	2.48E-04
sr 88	4.70E+01	4.70E+01	4.70E+01	4.70E+01	4.70E+01	4.70E+01	4.70E+01
sr 89	3.99E+00	1.73E+00	7.53E-01	3.27E-01	1.42E-01	6.17E-02	2.68E-02
y 89	5.88E+01	6.11E+01	6.21E+01	6.25E+01	6.27E+01	6.28E+01	6.28E+01
sr 90	7.39E+01	7.36E+01	7.33E+01	7.30E+01	7.27E+01	7.24E+01	7.21E+01
y 90	2.00E-02	1.91E-02	1.90E-02	1.90E-02	1.89E-02	1.88E-02	1.87E-02
zr 90	2.93E+00	3.23E+00	3.53E+00	3.83E+00	4.13E+00	4.43E+00	4.73E+00
y 91	6.19E+00	3.03E+00	1.47E+00	7.17E-01	3.49E-01	1.70E-01	8.25E-02
zr 91	7.52E+01	7.84E+01	8.00E+01	8.07E+01	8.11E+01	8.13E+01	8.14E+01
zr 92	8.68E+01	8.68E+01	8.68E+01	8.68E+01	8.68E+01	8.68E+01	8.68E+01
zr 93	6.39E+01	6.40E+01	6.40E+01	6.40E+01	6.40E+01	6.40E+01	6.40E+01
nb 93	3.18E-06	3.50E-06	3.85E-06	4.23E-06	4.64E-06	5.08E-06	5.56E-06
nb 93m	4.19E-05	4.64E-05	5.09E-05	5.53E-05	5.98E-05	6.41E-05	6.85E-05
zr 94	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02
nb 94	6.30E-05	6.30E-05	6.30E-05	6.30E-05	6.30E-05	6.30E-05	6.30E-05
zr 95	9.97E+00	5.16E+00	2.67E+00	1.38E+00	7.15E-01	3.70E-01	1.92E-01
nb 95	5.36E+00	4.22E+00	2.62E+00	1.48E+00	8.07E-01	4.29E-01	2.26E-01
nb 95m	6.25E-03	3.42E-03	1.77E-03	9.17E-04	4.74E-04	2.46E-04	1.27E-04
mo 95	8.75E+01	9.35E+01	9.76E+01	1.00E+02	1.01E+02	1.02E+02	1.02E+02
zr 96	1.09E+02	1.09E+02	1.09E+02	1.09E+02	1.09E+02	1.09E+02	1.09E+02
mo 96	5.08E+00	5.08E+00	5.08E+00	5.08E+00	5.08E+00	5.08E+00	5.08E+00
mo 97	1.03E+02	1.03E+02	1.03E+02	1.03E+02	1.03E+02	1.03E+02	1.03E+02

vver440 fuel bu=36 mwd/kgu
fission products
decay, following reactor irradiation identified by:
power=4.96mw, burnup=4464.mwd, flux=4.08E+13n/cm**2-sec
nuclide concentrations, grams
basis = single reactor assembly

	initial	60.8 d	121.7 d	182.5 d	243.3 d	304.2 d	365.0 d
mo 98	1.12E+02	1.12E+02	1.12E+02	1.12E+02	1.12E+02	1.12E+02	1.12E+02
tc 98	8.46E-04	8.46E-04	8.46E-04	8.46E-04	8.46E-04	8.46E-04	8.46E-04
tc 99	1.06E+02	1.07E+02	1.07E+02	1.07E+02	1.07E+02	1.07E+02	1.07E+02
ru 99	4.21E-03	4.28E-03	4.34E-03	4.40E-03	4.46E-03	4.52E-03	4.57E-03
mo100	1.26E+02	1.26E+02	1.26E+02	1.26E+02	1.26E+02	1.26E+02	1.26E+02
ru100	1.45E+01	1.45E+01	1.45E+01	1.45E+01	1.45E+01	1.45E+01	1.45E+01
ru101	1.03E+02	1.03E+02	1.03E+02	1.03E+02	1.03E+02	1.03E+02	1.03E+02
ru102	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02	1.04E+02
rh102	1.44E-04	1.39E-04	1.33E-04	1.28E-04	1.23E-04	1.18E-04	1.14E-04
ru103	6.72E+00	2.30E+00	7.85E-01	2.68E-01	9.16E-02	3.13E-02	1.07E-02
rh103	5.67E+01	6.11E+01	6.26E+01	6.31E+01	6.33E+01	6.34E+01	6.34E+01
rh103m	6.67E-03	2.28E-03	7.77E-04	2.66E-04	9.07E-05	3.10E-05	1.06E-05
ru104	7.21E+01	7.21E+01	7.21E+01	7.21E+01	7.21E+01	7.21E+01	7.21E+01
pd104	2.92E+01	2.92E+01	2.92E+01	2.92E+01	2.92E+01	2.92E+01	2.92E+01
pd105	5.03E+01	5.05E+01	5.05E+01	5.05E+01	5.05E+01	5.05E+01	5.05E+01
ru106	2.20E+01	1.97E+01	1.76E+01	1.57E+01	1.40E+01	1.25E+01	1.12E+01
rh106	2.32E-05	1.83E-05	1.63E-05	1.45E-05	1.30E-05	1.16E-05	1.03E-05
pd106	2.64E+01	2.88E+01	3.09E+01	3.28E+01	3.45E+01	3.60E+01	3.73E+01
pd107	2.93E+01	2.93E+01	2.93E+01	2.93E+01	2.93E+01	2.93E+01	2.93E+01
ag107	3.25E-06	3.77E-06	4.29E-06	4.81E-06	5.33E-06	5.85E-06	6.37E-06
pd108	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01
ag108m	2.40E-05	2.40E-05	2.39E-05	2.39E-05	2.39E-05	2.39E-05	2.39E-05
cd108	2.47E-05	2.47E-05	2.47E-05	2.47E-05	2.47E-05	2.47E-05	2.47E-05
ag109	1.16E+01	1.16E+01	1.16E+01	1.16E+01	1.16E+01	1.16E+01	1.16E+01
pd110	5.71E+00	5.71E+00	5.71E+00	5.71E+00	5.71E+00	5.71E+00	5.71E+00
ag110m	1.05E-01	8.85E-02	7.47E-02	6.31E-02	5.33E-02	4.50E-02	3.80E-02
cd110	5.08E+00	5.10E+00	5.11E+00	5.12E+00	5.13E+00	5.14E+00	5.15E+00
cd111	2.92E+00	2.98E+00	2.98E+00	2.98E+00	2.98E+00	2.98E+00	2.98E+00
cd112	1.53E+00	1.53E+00	1.53E+00	1.53E+00	1.53E+00	1.53E+00	1.53E+00
cd113	1.24E-02	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.28E-02
cd113m	1.71E-02	1.69E-02	1.68E-02	1.67E-02	1.65E-02	1.64E-02	1.63E-02
in113	8.78E-04	1.02E-03	1.16E-03	1.29E-03	1.43E-03	1.56E-03	1.70E-03
cd114	1.59E+00	1.59E+00	1.59E+00	1.59E+00	1.59E+00	1.59E+00	1.59E+00
sn114	4.62E-05	5.12E-05	5.33E-05	5.43E-05	5.47E-05	5.48E-05	5.49E-05
cd115m	2.08E-03	8.09E-04	3.14E-04	1.22E-04	4.74E-05	1.84E-05	7.16E-06
in115	1.94E-01	1.97E-01	1.98E-01	1.98E-01	1.98E-01	1.98E-01	1.98E-01
sn115	2.34E-02	2.35E-02	2.35E-02	2.35E-02	2.35E-02	2.35E-02	2.35E-02

cd116	6.59E-01	6.59E-01	6.59E-01	6.59E-01	6.59E-01	6.59E-01	6.59E-01
sn116	2.99E-01	2.99E-01	2.99E-01	2.99E-01	2.99E-01	2.99E-01	2.99E-01
sn117	6.16E-01	6.16E-01	6.16E-01	6.16E-01	6.16E-01	6.16E-01	6.16E-01
sn118	4.96E-01	4.96E-01	4.96E-01	4.96E-01	4.96E-01	4.96E-01	4.96E-01
sn119	5.22E-01	5.23E-01	5.23E-01	5.23E-01	5.24E-01	5.24E-01	5.24E-01
sn119m	2.72E-03	2.36E-03	2.04E-03	1.77E-03	1.53E-03	1.33E-03	1.15E-03
sn120	5.12E-01	5.12E-01	5.12E-01	5.12E-01	5.12E-01	5.12E-01	5.12E-01
sn121m	5.69E-03	5.68E-03	5.67E-03	5.65E-03	5.64E-03	5.63E-03	5.62E-03
sb121	5.09E-01	5.11E-01	5.11E-01	5.11E-01	5.11E-01	5.11E-01	5.11E-01
sn122	6.65E-01	6.65E-01	6.65E-01	6.65E-01	6.65E-01	6.65E-01	6.65E-01
te122	3.41E-02	3.45E-02	3.45E-02	3.45E-02	3.45E-02	3.45E-02	3.45E-02
sn123	1.00E-02	7.23E-03	5.22E-03	3.76E-03	2.71E-03	1.96E-03	1.41E-03
sb123	5.96E-01	5.99E-01	6.01E-01	6.03E-01	6.04E-01	6.04E-01	6.05E-01
te123	2.82E-04	3.13E-04	3.36E-04	3.51E-04	3.62E-04	3.70E-04	3.76E-04
te123m	1.07E-04	7.50E-05	5.28E-05	3.71E-05	2.61E-05	1.83E-05	1.29E-05

vver440 fuel bu=36 mwd/kgu fission products
decay, following reactor irradiation identified by:
power=4.96mw, burnup=4464.mwd, flux=4.08E+13n/cm**2-sec
nuclide concentrations, grams
basis = single reactor assembly

	initial	60.8 d	121.7 d	182.5 d	243.3 d	304.2 d	365.0 d
sn124	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00	1.11E+00
sb124	5.97E-03	2.96E-03	1.47E-03	7.30E-04	3.62E-04	1.80E-04	8.92E-05
te124	2.58E-02	2.88E-02	3.03E-02	3.10E-02	3.14E-02	3.16E-02	3.17E-02
sb125	1.01E+00	9.74E-01	9.34E-01	8.95E-01	8.58E-01	8.23E-01	7.89E-01
te125	3.38E-01	3.79E-01	4.20E-01	4.59E-01	4.96E-01	5.32E-01	5.67E-01
te125m	1.24E-02	1.29E-02	1.28E-02	1.25E-02	1.21E-02	1.16E-02	1.12E-02
sn126	2.55E+00	2.55E+00	2.55E+00	2.55E+00	2.55E+00	2.55E+00	2.55E+00
te126	4.63E-02	4.70E-02	4.70E-02	4.70E-02	4.70E-02	4.70E-02	4.70E-02
te127	4.54E-03	5.02E-04	3.41E-04	2.32E-04	1.57E-04	1.07E-04	7.26E-05
te127m	2.03E-01	1.44E-01	9.75E-02	6.62E-02	4.50E-02	3.05E-02	2.07E-02
i127	5.46E+00	5.57E+00	5.61E+00	5.64E+00	5.66E+00	5.68E+00	5.69E+00
te128	1.20E+01	1.20E+01	1.20E+01	1.20E+01	1.20E+01	1.20E+01	1.20E+01
xe128	3.38E-01	3.38E-01	3.38E-01	3.38E-01	3.38E-01	3.38E-01	3.38E-01
te129m	2.79E-01	8.00E-02	2.28E-02	6.50E-03	1.85E-03	5.28E-04	1.51E-04
i129	2.39E+01	2.42E+01	2.42E+01	2.42E+01	2.42E+01	2.42E+01	2.42E+01
xe129	1.96E-03	2.01E-03	2.01E-03	2.01E-03	2.01E-03	2.01E-03	2.01E-03
te130	4.89E+01	4.89E+01	4.89E+01	4.89E+01	4.89E+01	4.89E+01	4.89E+01
xe130	9.52E-01	9.54E-01	9.54E-01	9.54E-01	9.54E-01	9.54E-01	9.54E-01
xe131	5.47E+01	5.59E+01	5.59E+01	5.59E+01	5.59E+01	5.59E+01	5.59E+01
xe132	1.48E+02	1.49E+02	1.49E+02	1.49E+02	1.49E+02	1.49E+02	1.49E+02
ba132	2.79E-05	2.86E-05	2.86E-05	2.86E-05	2.86E-05	2.86E-05	2.86E-05
cs133	1.53E+02	1.55E+02	1.55E+02	1.55E+02	1.55E+02	1.55E+02	1.55E+02
xe134	2.05E+02	2.05E+02	2.05E+02	2.05E+02	2.05E+02	2.05E+02	2.05E+02
cs134	1.62E+01	1.53E+01	1.45E+01	1.37E+01	1.30E+01	1.23E+01	1.16E+01
ba134	5.36E+00	6.24E+00	7.08E+00	7.87E+00	8.61E+00	9.32E+00	9.99E+00
cs135	4.77E+01	4.78E+01	4.78E+01	4.78E+01	4.78E+01	4.78E+01	4.78E+01
ba135	4.07E-02	4.07E-02	4.07E-02	4.07E-02	4.07E-02	4.07E-02	4.07E-02
xe136	3.09E+02	3.09E+02	3.09E+02	3.09E+02	3.09E+02	3.09E+02	3.09E+02
ba136	2.27E+00	2.36E+00	2.36E+00	2.36E+00	2.36E+00	2.36E+00	2.36E+00
cs137	1.67E+02	1.67E+02	1.66E+02	1.65E+02	1.65E+02	1.64E+02	1.64E+02
ba137	5.56E+00	6.21E+00	6.85E+00	7.49E+00	8.12E+00	8.75E+00	9.38E+00
ba137m	2.58E-05	2.55E-05	2.54E-05	2.53E-05	2.52E-05	2.51E-05	2.50E-05
ba138	1.74E+02	1.74E+02	1.74E+02	1.74E+02	1.74E+02	1.74E+02	1.74E+02
la138	1.15E-03	1.15E-03	1.15E-03	1.15E-03	1.15E-03	1.15E-03	1.15E-03
la139	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02	1.64E+02
ce140	1.73E+02	1.76E+02	1.76E+02	1.76E+02	1.76E+02	1.76E+02	1.76E+02
ce141	7.63E+00	2.10E+00	5.73E-01	1.57E-01	4.28E-02	1.17E-02	3.19E-03
pr141	1.43E+02	1.49E+02	1.50E+02	1.51E+02	1.51E+02	1.51E+02	1.51E+02
ce142	1.53E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02	1.53E+02
nd142	2.58E+00	2.58E+00	2.58E+00	2.58E+00	2.58E+00	2.58E+00	2.58E+00
nd143	1.08E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02	1.11E+02
ce144	4.87E+01	4.20E+01	3.63E+01	3.13E+01	2.70E+01	2.33E+01	2.01E+01
pr144	2.07E-03	1.77E-03	1.53E-03	1.32E-03	1.14E-03	9.80E-04	8.45E-04
pr144m	1.20E-05	1.03E-05	8.91E-06	7.68E-06	6.63E-06	5.71E-06	4.93E-06
nd144	1.27E+02	1.34E+02	1.40E+02	1.45E+02	1.49E+02	1.53E+02	1.56E+02
nd145	9.07E+01	9.07E+01	9.07E+01	9.07E+01	9.07E+01	9.07E+01	9.07E+01
pm145	2.88E-06	2.88E-06	2.89E-06	2.89E-06	2.88E-06	2.88E-06	2.87E-06
nd146	9.35E+01	9.35E+01	9.35E+01	9.35E+01	9.35E+01	9.35E+01	9.35E+01
pm146	1.02E-03	9.97E-04	9.76E-04	9.56E-04	9.36E-04	9.17E-04	8.98E-04
sm146	7.40E-04	7.47E-04	7.54E-04	7.61E-04	7.67E-04	7.74E-04	7.80E-04
pm147	2.33E+01	2.33E+01	2.23E+01	2.14E+01	2.05E+01	1.96E+01	1.87E+01
sm147	9.27E+00	1.03E+01	1.13E+01	1.23E+01	1.32E+01	1.41E+01	1.49E+01

APPENDIX B

NUCLIDE INVENTORY OF SPENT VVER-440 FUEL

vver440 fuel bu= 36 mwd/ kgu fission products
decay, following reactor irradiation identified by:
power=4.96mw, burnup=4464.mwd, flux=4.08E+13n/cm**2-sec
nuclide concentrations, grams
basis = single reactor assembly

	initial	60.8 d	121.7 d	182.5 d	243.3 d	304.2 d	365.0 d
nd148	4.94E+01	4.94E+01	4.94E+01	4.94E+01	4.94E+01	4.94E+01	4.94E+01
pm148	1.46E-01	4.78E-04	1.52E-04	5.47E-05	1.97E-05	7.09E-06	2.56E-06
pm148m	1.70E-01	6.12E-02	2.20E-02	7.94E-03	2.86E-03	1.03E-03	3.71E-04
sm148	1.56E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01	1.59E+01
sm149	3.31E-01	5.45E-01	5.45E-01	5.45E-01	5.45E-01	5.45E-01	5.45E-01
nd150	2.38E+01	2.38E+01	2.38E+01	2.38E+01	2.38E+01	2.38E+01	2.38E+01
sm150	4.10E+01	4.10E+01	4.10E+01	4.10E+01	4.10E+01	4.10E+01	4.10E+01
sm151	2.10E+00	2.13E+00	2.13E+00	2.13E+00	2.13E+00	2.12E+00	2.12E+00
eu151	2.04E-03	4.78E-03	7.52E-03	1.02E-02	1.30E-02	1.57E-02	1.84E-02
sm152	1.70E+01	1.70E+01	1.70E+01	1.70E+01	1.70E+01	1.70E+01	1.70E+01
eu152	5.01E-03	4.97E-03	4.93E-03	4.89E-03	4.84E-03	4.80E-03	4.76E-03
gd152	7.12E-03	7.14E-03	7.15E-03	7.16E-03	7.17E-03	7.19E-03	7.20E-03
eu153	1.58E+01	1.60E+01	1.60E+01	1.60E+01	1.60E+01	1.60E+01	1.60E+01
gd153	5.75E-04	4.83E-04	4.06E-04	3.41E-04	2.86E-04	2.40E-04	2.02E-04
sm154	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00
eu154	3.18E+00	3.14E+00	3.10E+00	3.05E+00	3.01E+00	2.97E+00	2.93E+00
gd154	2.31E-01	2.73E-01	3.15E-01	3.56E-01	3.97E-01	4.37E-01	4.77E-01
eu155	7.23E-01	7.06E-01	6.89E-01	6.72E-01	6.55E-01	6.39E-01	6.24E-01
gd155	5.35E-03	2.30E-02	4.02E-02	5.69E-02	7.33E-02	8.93E-02	1.05E-01
gd156	9.49E+00	1.01E+01	1.01E+01	1.01E+01	1.01E+01	1.01E+01	1.01E+01
gd157	1.39E-02	1.69E-02	1.69E-02	1.69E-02	1.69E-02	1.69E-02	1.69E-02
gd158	2.39E+00	2.39E+00	2.39E+00	2.39E+00	2.39E+00	2.39E+00	2.39E+00
tb159	3.02E-01	3.03E-01	3.03E-01	3.03E-01	3.03E-01	3.03E-01	3.03E-01
gd160	1.33E-01	1.33E-01	1.33E-01	1.33E-01	1.33E-01	1.33E-01	1.33E-01
tb160	1.11E-02	6.20E-03	3.46E-03	1.93E-03	1.08E-03	6.02E-04	3.36E-04
dy160	2.33E-02	2.82E-02	3.09E-02	3.24E-02	3.33E-02	3.38E-02	3.40E-02
dy161	4.46E-02	4.60E-02	4.60E-02	4.60E-02	4.60E-02	4.60E-02	4.60E-02
dy162	3.48E-02	3.48E-02	3.48E-02	3.48E-02	3.48E-02	3.48E-02	3.48E-02
dy163	2.77E-02	2.77E-02	2.77E-02	2.77E-02	2.77E-02	2.77E-02	2.77E-02
dy164	6.28E-03	6.28E-03	6.28E-03	6.28E-03	6.28E-03	6.28E-03	6.28E-03
ho165	8.71E-03	8.72E-03	8.72E-03	8.72E-03	8.72E-03	8.72E-03	8.72E-03
ho166m	2.93E-05	2.93E-05	2.93E-05	2.93E-05	2.93E-05	2.93E-05	2.93E-05
er166	1.88E-03	1.89E-03	1.89E-03	1.89E-03	1.89E-03	1.89E-03	1.89E-03
er167	3.58E-05	3.58E-05	3.58E-05	3.58E-05	3.58E-05	3.58E-05	3.58E-05
er168	4.63E-05	4.63E-05	4.63E-05	4.63E-05	4.63E-05	4.63E-05	4.63E-05
total	4.59E+03	4.59E+03	4.59E+03	4.59E+03	4.59E+03	4.59E+03	4.59E+03

NEUTRON SOURCE OF SPENT BWR FUEL

APPENDIX C

neutron source intensity as a function of time
bu=37816, bwr, 5 cyc, e=2.746, vf=0.426, 8x8-e2-275-1-200-10
alpha-n neutron source, neutrons/sec/basis
basis = single reactor assembly

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
pb210	6.12E-16	6.24E-16	6.14E-16	6.60E-16	8.02E-16	1.09E-15	1.59E-15
bi210m	1.37E-16	1.37E-16	1.37E-16	1.37E-16	1.37E-16	1.37E-16	1.37E-16
bi210	1.57E-13	1.59E-13	1.57E-13	1.68E-13	2.05E-13	2.79E-13	4.07E-13
bi211	4.32E-05	6.67E-05	9.65E-05	1.33E-04	1.76E-04	2.25E-04	2.79E-04
bi212	1.38E-01	5.10E-01	9.21E-01	1.24E+00	1.45E+00	1.57E+00	1.64E+00
bi213	7.53E-07	1.26E-07	1.34E-07	1.44E-07	1.57E-07	1.73E-07	1.92E-07
bi214	6.99E-12	2.91E-11	9.20E-11	2.20E-10	4.38E-10	7.68E-10	1.24E-09
po210	1.23E-07	1.95E-07	1.91E-07	2.01E-07	2.37E-07	3.15E-07	4.54E-07
po211	1.71E-07	2.64E-07	3.82E-07	5.27E-07	6.97E-07	8.90E-07	1.10E-06
po212	7.07E-01	2.61E+00	4.72E+00	6.34E+00	7.41E+00	8.05E+00	8.38E+00
po213	9.93E-05	1.67E-05	1.77E-05	1.90E-05	2.08E-05	2.29E-05	2.53E-05
po214	1.91E-05	2.60E-07	8.19E-07	1.96E-06	3.90E-06	6.85E-06	1.10E-05
po215	6.10E-05	9.42E-05	1.36E-04	1.88E-04	2.49E-04	3.18E-04	3.94E-04
po216	5.51E-01	2.04E+00	3.68E+00	4.94E+00	5.78E+00	6.28E+00	6.54E+00
po218	2.96E-08	1.23E-07	3.90E-07	9.33E-07	1.85E-06	3.26E-06	5.24E-06
at217	6.44E-05	1.08E-05	1.15E-05	1.23E-05	1.35E-05	1.48E-05	1.64E-05
rn218	1.54E-05	2.21E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
rn219	4.85E-05	7.49E-05	1.08E-04	1.50E-04	1.98E-04	2.53E-04	3.13E-04
rn220	4.37E-01	1.62E+00	2.92E+00	3.92E+00	4.58E+00	4.97E+00	5.18E+00
rn222	2.16E-08	9.02E-08	2.85E-07	6.81E-07	1.35E-06	2.38E-06	3.82E-06
fr221	4.70E-05	7.88E-06	8.36E-06	9.00E-06	9.82E-06	1.08E-05	1.20E-05
fr223	1.88E-11	2.84E-11	4.10E-11	5.66E-11	7.48E-11	9.56E-11	1.19E-10
ra222	1.19E-05	1.71E-17	3.58E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ra223	2.81E-05	4.34E-05	6.28E-05	8.66E-05	1.15E-04	1.46E-04	1.81E-04
ra224	3.09E-01	1.14E+00	2.06E+00	2.77E+00	3.24E+00	3.52E+00	3.66E+00
ra226	1.27E-08	5.27E-08	1.66E-07	3.98E-07	7.92E-07	1.39E-06	2.24E-06
ac225	3.38E-05	5.67E-06	6.01E-06	6.47E-06	7.06E-06	7.77E-06	8.60E-06
ac227	2.11E-07	3.19E-07	4.62E-07	6.37E-07	8.42E-07	1.08E-06	1.33E-06
ac228	2.07E-17	5.00E-17	9.04E-17	1.39E-16	1.95E-16	2.55E-16	3.19E-16
th226	1.08E-05	1.54E-17	3.23E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00
th227	3.13E-05	4.79E-05	6.93E-05	9.55E-05	1.26E-04	1.61E-04	2.00E-04
th228	2.59E-01	9.63E-01	1.74E+00	2.33E+00	2.73E+00	2.96E+00	3.08E+00
th229	3.18E-06	3.31E-06	3.51E-06	3.78E-06	4.12E-06	4.54E-06	5.03E-06
th230	1.73E-05	6.60E-05	1.57E-04	2.91E-04	4.66E-04	6.81E-04	9.36E-04
th232	1.33E-09	2.31E-09	3.28E-09	4.25E-09	5.22E-09	6.19E-09	7.16E-09
pa231	1.17E-04	1.66E-04	2.15E-04	2.64E-04	3.13E-04	3.62E-04	4.10E-04
u230	8.48E-06	1.21E-17	2.54E-29	0.00E+00	0.00E+00	0.00E+00	0.00E+00
u231	1.05E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
u232	7.71E-01	1.83E+00	2.42E+00	2.74E+00	2.89E+00	2.96E+00	2.97E+00
u233	4.03E-04	7.20E-04	1.04E-03	1.35E-03	1.67E-03	1.99E-03	2.32E-03
u234	1.45E+00	3.66E+00	5.87E+00	8.05E+00	1.02E+01	1.23E+01	1.44E+01
u235	6.26E-01	6.26E-01	6.26E-01	6.26E-01	6.26E-01	6.26E-01	6.26E-01
u236	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01	1.55E+01
u238	1.51E+01	1.51E+01	1.51E+01	1.51E+01	1.51E+01	1.51E+01	1.51E+01
np235	2.87E-05	6.86E-06	1.64E-06	3.91E-07	9.35E-08	2.23E-08	5.33E-09
np237	2.75E+01	2.79E+01	2.80E+01	2.81E+01	2.82E+01	2.83E+01	2.84E+01
pu236	8.44E+01	4.96E+01	2.90E+01	1.70E+01	9.94E+00	5.82E+00	3.41E+00
pu237	3.45E-02	1.21E-07	4.27E-13	1.51E-18	5.30E-24	1.87E-29	0.00E+00

neutron source intensity as a function of time
bu=37816, bwr, 5 cyc, e=2.746, vf=0.426, 8x8-e2-275-1-200-10
alpha-n neutron source, neutrons/sec/basis
basis = single reactor assembly

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
pu238	5.66E+05	6.08E+05	5.99E+05	5.89E+05	5.79E+05	5.68E+05	5.58E+05
pu239	3.84E+04	3.89E+04	3.89E+04	3.89E+04	3.89E+04	3.89E+04	3.89E+04
pu240	6.94E+04	6.96E+04	6.97E+04	6.98E+04	6.99E+04	7.00E+04	7.01E+04
pu241	3.40E+02	3.05E+02	2.74E+02	2.46E+02	2.21E+02	1.98E+02	1.78E+02
pu242	2.84E+02	2.84E+02	2.84E+02	2.84E+02	2.84E+02	2.84E+02	2.84E+02
pu244	1.53E-12	8.95E-12	1.64E-11	2.38E-11	3.12E-11	3.87E-11	4.61E-11
am239	8.45E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
am240	1.29E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
am241	3.25E+04	1.07E+05	1.74E+05	2.34E+05	2.87E+05	3.35E+05	3.77E+05
am242m	7.74E+00	7.66E+00	7.57E+00	7.49E+00	7.41E+00	7.33E+00	7.25E+00
am243	4.88E+03	4.88E+03	4.88E+03	4.88E+03	4.88E+03	4.87E+03	4.87E+03
cm241	4.19E-02	1.30E-09	4.00E-17	1.24E-24	0.00E+00	0.00E+00	0.00E+00
cm242	1.40E+07	4.35E+05	1.57E+04	2.74E+03	2.31E+03	2.28E+03	2.25E+03

APPENDIX C

NEUTRON SOURCE OF SPENT BWR FUEL

cm243	6.39E+03	6.05E+03	5.73E+03	5.43E+03	5.14E+03	4.87E+03	4.61E+03
cm244	1.09E+06	9.98E+05	9.16E+05	8.40E+05	7.71E+05	7.08E+05	6.50E+05
cm245	7.04E+01	7.03E+01	7.03E+01	7.03E+01	7.03E+01	7.03E+01	7.03E+01
cm246	2.28E+01	2.28E+01	2.28E+01	2.28E+01	2.27E+01	2.27E+01	2.27E+01
cm247	1.04E-04	1.04E-04	1.04E-04	1.04E-04	1.04E-04	1.04E-04	1.04E-04
cm248	5.74E-04	5.74E-04	5.74E-04	5.74E-04	5.74E-04	5.74E-04	5.74E-04
cm250	2.94E-11	2.94E-11	2.94E-11	2.94E-11	2.94E-11	2.94E-11	2.94E-11
bk249	7.29E-05	1.24E-05	2.11E-06	3.58E-07	6.08E-08	1.03E-08	1.76E-09
cf249	2.97E-03	1.74E-02	1.98E-02	2.01E-02	2.01E-02	2.00E-02	1.99E-02
cf250	8.67E-02	7.73E-02	6.86E-02	6.10E-02	5.41E-02	4.81E-02	4.27E-02
cf251	7.31E-04	7.29E-04	7.28E-04	7.27E-04	7.25E-04	7.24E-04	7.23E-04
cf252	2.18E-01	1.21E-01	6.74E-02	3.75E-02	2.08E-02	1.16E-02	6.44E-03
cf253	5.97E-05	8.84E-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cf254	1.46E-08	1.24E-12	1.05E-16	8.91E-21	7.55E-25	6.40E-29	0.00E+00
es253	2.10E-02	1.79E-13	1.68E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00
es254	6.61E-05	8.45E-06	1.08E-06	1.38E-07	1.76E-08	2.25E-09	2.88E-10
es255	1.06E-06	5.14E-13	2.48E-19	1.20E-25	0.00E+00	0.00E+00	0.00E+00
total	1.58E+07	2.27E+06	1.82E+06	1.79E+06	1.76E+06	1.73E+06	1.71E+06

neutron source intensity as a function of time
bu=37816, bwr, 5 cyc, e=2.746, vf=0.426, 8x8-e2-275-1-200-10
spontaneous fission neutron source, neutrons/sec/basis
basis = single reactor assembly

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
th230	4.43E-10	1.69E-09	4.04E-09	7.46E-09	1.19E-08	1.75E-08	2.40E-08
pa231	1.47E-08	2.08E-08	2.70E-08	3.31E-08	3.92E-08	4.53E-08	5.15E-08
u232	4.74E-05	1.13E-04	1.49E-04	1.68E-04	1.78E-04	1.82E-04	1.82E-04
u234	3.13E-03	7.86E-03	1.26E-02	1.73E-02	2.19E-02	2.64E-02	3.09E-02
u235	7.65E-03	7.66E-03	7.66E-03	7.66E-03	7.66E-03	7.66E-03	7.66E-03
u236	2.32E+00	2.32E+00	2.32E+00	2.32E+00	2.32E+00	2.32E+00	2.33E+00
u237	2.02E-06	1.12E-11	1.00E-11	9.00E-12	8.08E-12	7.25E-12	6.51E-12
u238	2.12E+03	2.12E+03	2.12E+03	2.12E+03	2.12E+03	2.12E+03	2.12E+03
u239	1.01E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
np236	2.52E-08	2.52E-08	2.52E-08	2.52E-08	2.52E-08	2.52E-08	2.52E-08
np238	1.16E-05	2.25E-12	2.22E-12	2.20E-12	2.17E-12	2.15E-12	2.13E-12
np239	2.06E-02	5.06E-08	5.06E-08	5.05E-08	5.05E-08	5.05E-08	5.05E-08
pu236	5.71E+00	3.36E+00	1.96E+00	1.15E+00	6.73E-01	3.94E-01	2.31E-01
pu238	1.05E+05	1.12E+05	1.11E+05	1.09E+05	1.07E+05	1.05E+05	1.03E+05
pu239	2.06E+01	2.08E+01	2.08E+01	2.08E+01	2.08E+01	2.08E+01	2.08E+01
pu240	4.62E+05	4.62E+05	4.63E+05	4.64E+05	4.65E+05	4.65E+05	4.66E+05
pu241	1.22E+01	1.10E+01	9.86E+00	8.85E+00	7.94E+00	7.13E+00	6.39E+00
pu242	2.21E+05	2.21E+05	2.21E+05	2.21E+05	2.21E+05	2.21E+05	2.21E+05
pu243	1.86E-03	4.48E-15	4.48E-15	4.48E-15	4.48E-15	4.48E-15	4.48E-15
pu244	3.66E-07	2.14E-06	3.92E-06	5.69E-06	7.47E-06	9.25E-06	1.10E-05
am241	1.25E+01	4.13E+01	6.71E+01	9.01E+01	1.11E+02	1.29E+02	1.45E+02
am242m	3.68E+01	3.64E+01	3.60E+01	3.56E+01	3.52E+01	3.48E+01	3.44E+01
am242	2.97E+02	3.95E-02	3.91E-02	3.87E-02	3.82E-02	3.78E-02	3.74E-02
am243	2.24E+01	2.25E+01	2.25E+01	2.25E+01	2.24E+01	2.24E+01	2.24E+01
am244	4.18E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
cm242	6.98E+07	2.17E+06	7.82E+04	1.37E+04	1.15E+04	1.14E+04	1.12E+04
cm243	1.39E+02	1.31E+02	1.24E+02	1.18E+02	1.12E+02	1.06E+02	1.00E+02
cm244	1.42E+08	1.30E+08	1.19E+08	1.10E+08	1.01E+08	9.23E+07	8.48E+07
cm245	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01
cm246	8.21E+05	8.21E+05	8.20E+05	8.20E+05	8.20E+05	8.20E+05	8.19E+05
cm248	9.39E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03	9.40E+03
cm250	1.25E-02	1.25E-02	1.25E-02	1.25E-02	1.25E-02	1.25E-02	1.25E-02
bk249	4.02E-01	6.84E-02	1.16E-02	1.97E-03	3.35E-04	5.70E-05	9.68E-06
cf249	1.80E-03	1.06E-02	1.20E-02	1.22E-02	1.22E-02	1.22E-02	1.21E-02
cf250	7.26E+03	6.47E+03	5.75E+03	5.11E+03	4.53E+03	4.03E+03	3.58E+03
cf252	7.64E+05	4.24E+05	2.36E+05	1.31E+05	7.29E+04	4.05E+04	2.25E+04
cf254	6.28E+02	5.32E-02	4.51E-06	3.83E-10	3.24E-14	2.75E-18	2.33E-22
es253	1.56E-01	1.33E-12	1.25E-24	0.00E+00	0.00E+00	0.00E+00	0.00E+00
es255	3.06E-03	1.48E-09	7.15E-16	3.45E-22	1.66E-28	0.00E+00	0.00E+00
total	2.14E+08	1.34E+08	1.21E+08	1.11E+08	1.02E+08	9.40E+07	8.64E+07
total	2.30E+08	1.37E+08	1.23E+08	1.13E+08	1.04E+08	9.58E+07	8.81E+07

NEUTRON SOURCE OF SPENT BWR FUEL

APPENDIX C

alpha-n neutron source spectrum as a function of time
(using reaction spectra for uranium dioxide)
bu=37816, bwr, 5 cyc, e=2.746, vf=0.426, 8x8-e2-275-1-200-10
alpha-n neutron spectra, neutrons/sec/basis
basis = single reactor assembly

	boundaries, mev	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
1	6.43E+00 - 2.00E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
2	3.00E+00 - 6.43E+00	6.765E+06	6.296E+05	4.227E+05	4.086E+05	4.007E+05	3.930E+05	3.855E+05
3	1.85E+00 - 3.00E+00	7.498E+06	1.178E+06	9.720E+05	9.528E+05	9.394E+05	9.261E+05	9.129E+05
4	1.40E+00 - 1.85E+00	1.049E+06	2.728E+05	2.492E+05	2.455E+05	2.423E+05	2.392E+05	2.361E+05
5	9.00E-01 - 1.40E+00	3.829E+05	1.423E+05	1.357E+05	1.340E+05	1.324E+05	1.308E+05	1.293E+05
6	4.00E-01 - 9.00E-01	7.087E+04	3.946E+04	3.887E+04	3.845E+04	3.800E+04	3.756E+04	3.712E+04
7	1.00E-01 - 4.00E-01	1.305E+04	6.656E+03	6.506E+03	6.421E+03	6.334E+03	6.247E+03	6.161E+03
8	1.70E-02 - 1.00E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9	3.00E-03 - 1.70E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
10	5.50E-04 - 3.00E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
11	1.00E-04 - 5.50E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
12	3.00E-05 - 1.00E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
13	1.00E-05 - 3.00E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
14	3.05E-06 - 1.00E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
15	1.77E-06 - 3.05E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
16	1.30E-06 - 1.77E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
17	1.13E-06 - 1.30E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
18	1.00E-06 - 1.13E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
19	8.00E-07 - 1.00E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
20	4.00E-07 - 8.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
21	3.25E-07 - 4.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
22	2.25E-07 - 3.25E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
23	1.00E-07 - 2.25E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
24	5.00E-08 - 1.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
25	3.00E-08 - 5.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
26	1.00E-08 - 3.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
27	1.00E-11 - 1.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
		1.578E+07	2.269E+06	1.825E+06	1.786E+06	1.759E+06	1.733E+06	1.707E+06

spontaneous fission neutron source spectrum as a function of time
bu=37816, bwr, 5 cyc, e=2.746, vf=0.426, 8x8-e2-275-1-200-10
spontaneous fission neutron spectra, neutrons/ sec/ basis
basis = single reactor assembly

	boundaries, mev	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
1	6.43E+00 - 2.00E+01	4.042E+06	2.530E+06	2.285E+06	2.097E+06	1.926E+06	1.770E+06	1.626E+06
2	3.00E+00 - 6.43E+00	4.478E+07	2.816E+07	2.544E+07	2.335E+07	2.145E+07	1.971E+07	1.811E+07
3	1.85E+00 - 3.00E+00	4.854E+07	3.045E+07	2.751E+07	2.524E+07	2.319E+07	2.130E+07	1.958E+07
4	1.40E+00 - 1.85E+00	2.785E+07	1.758E+07	1.588E+07	1.457E+07	1.339E+07	1.230E+07	1.130E+07
5	9.00E-01 - 1.40E+00	3.817E+07	2.407E+07	2.175E+07	1.996E+07	1.833E+07	1.684E+07	1.548E+07
6	4.00E-01 - 9.00E-01	4.209E+07	2.642E+07	2.386E+07	2.190E+07	2.011E+07	1.848E+07	1.699E+07
7	1.00E-01 - 4.00E-01	8.265E+06	5.173E+06	4.673E+06	4.288E+06	3.939E+06	3.619E+06	3.326E+06
8	1.70E-02 - 1.00E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9	3.00E-03 - 1.70E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
10	5.50E-04 - 3.00E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
11	1.00E-04 - 5.50E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
12	3.00E-05 - 1.00E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
13	1.00E-05 - 3.00E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
14	3.05E-06 - 1.00E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
15	1.77E-06 - 3.05E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
16	1.30E-06 - 1.77E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
17	1.13E-06 - 1.30E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
18	1.00E-06 - 1.13E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
19	8.00E-07 - 1.00E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
20	4.00E-07 - 8.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
21	3.25E-07 - 4.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
22	2.25E-07 - 3.25E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
23	1.00E-07 - 2.25E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
24	5.00E-08 - 1.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
25	3.00E-08 - 5.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
26	1.00E-08 - 3.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
27	1.00E-11 - 1.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
		2.137E+08	1.344E+08	1.214E+08	1.114E+08	1.023E+08	9.402E+07	8.641E+07

APPENDIX C

NEUTRON SOURCE OF SPENT BWR FUEL

total (alpha-n plus spon. fission) neutron source spectrum as a function of time
(using reaction spectra for uranium dioxide)
bu=37816, bwr, 5 cyc, e=2.746, vf=0.426, 8x8-e2-275-1-200-10
neutron spectra, neutrons/sec/basis
basis = single reactor assembly

	boundaries, mev	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
1	6.43E+00 - 2.00E+01	4.042E+06	2.530E+06	2.285E+06	2.097E+06	1.926E+06	1.770E+06	1.626E+06
2	3.00E+00 - 6.43E+00	5.154E+07	2.879E+07	2.587E+07	2.376E+07	2.185E+07	2.010E+07	1.850E+07
3	1.85E+00 - 3.00E+00	5.603E+07	3.163E+07	2.848E+07	2.620E+07	2.413E+07	2.223E+07	2.049E+07
4	1.40E+00 - 1.85E+00	2.890E+07	1.785E+07	1.613E+07	1.482E+07	1.363E+07	1.254E+07	1.154E+07
5	9.00E-01 - 1.40E+00	3.855E+07	2.421E+07	2.189E+07	2.009E+07	1.847E+07	1.698E+07	1.561E+07
6	4.00E-01 - 9.00E-01	4.216E+07	2.646E+07	2.390E+07	2.194E+07	2.015E+07	1.852E+07	1.702E+07
7	1.00E-01 - 4.00E-01	8.278E+06	5.180E+06	4.679E+06	4.294E+06	3.945E+06	3.625E+06	3.332E+06
8	1.70E-02 - 1.00E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
9	3.00E-03 - 1.70E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
10	5.50E-04 - 3.00E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
11	1.00E-04 - 5.50E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
12	3.00E-05 - 1.00E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
13	1.00E-05 - 3.00E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
14	3.05E-06 - 1.00E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
15	1.77E-06 - 3.05E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
16	1.30E-06 - 1.77E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
17	1.13E-06 - 1.30E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
18	1.00E-06 - 1.13E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
19	8.00E-07 - 1.00E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
20	4.00E-07 - 8.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
21	3.25E-07 - 4.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
22	2.25E-07 - 3.25E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
23	1.00E-07 - 2.25E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
24	5.00E-08 - 1.00E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
25	3.00E-08 - 5.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
26	1.00E-08 - 3.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
27	1.00E-11 - 1.00E-08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
		2.295E+08	1.367E+08	1.232E+08	1.132E+08	1.041E+08	9.576E+07	8.812E+07

Gamma source of spent BWR fuel

APPENDIX D

principal photon sources in group 1, photons/sec
mean energy = 0.0100 mev. nuclides exceeding 1.0E-02 of total group release rate (3.08E+14) at 4091.7 d
nuclide time after discharge

nuclide	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
kr 85	7.78E+12	6.73E+12	5.83E+12	5.04E+12	4.36E+12	3.77E+12	3.26E+12
sr 90	5.30E+13	5.01E+13	4.75E+13	4.49E+13	4.25E+13	4.02E+13	3.81E+13
y 90	2.70E+14	2.46E+14	2.33E+14	2.21E+14	2.09E+14	1.98E+14	1.87E+14
cs137	7.09E+13	6.73E+13	6.39E+13	6.07E+13	5.76E+13	5.47E+13	5.20E+13
eu154	8.77E+12	7.32E+12	6.11E+12	5.10E+12	4.26E+12	3.55E+12	2.97E+12

principal photon sources in group 2, photons/sec
mean energy = 0.0300 mev. nuclides exceeding 1.0E-02 of total group release rate (1.41E+14) at 4091.7 d
nuclide time after discharge

nuclide	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
sr 90	1.50E+13	1.42E+13	1.34E+13	1.27E+13	1.20E+13	1.14E+13	1.08E+13
y 90	8.79E+13	8.03E+13	7.59E+13	7.19E+13	6.80E+13	6.44E+13	6.09E+13
sb125	2.52E+13	1.43E+13	8.10E+12	4.59E+12	2.60E+12	1.47E+12	8.32E+11
cs137	1.98E+13	1.88E+13	1.78E+13	1.69E+13	1.61E+13	1.53E+13	1.45E+13
ba137m	5.75E+13	5.42E+13	5.15E+13	4.89E+13	4.64E+13	4.41E+13	4.19E+13

principal photon sources in group 3, photons/sec
mean energy = 0.0550 mev. nuclides exceeding 1.0E-02 of total group release rate (6.70E+13) at 4091.7 d
nuclide time after discharge

nuclide	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
kr 85	1.39E+12	1.20E+12	1.04E+12	9.01E+11	7.79E+11	6.74E+11	5.83E+11
sr 90	8.85E+12	8.38E+12	7.93E+12	7.50E+12	7.10E+12	6.72E+12	6.36E+12
y 90	6.08E+13	5.55E+13	5.25E+13	4.97E+13	4.70E+13	4.45E+13	4.21E+13
cs137	1.15E+13	1.09E+13	1.04E+13	9.84E+12	9.35E+12	8.88E+12	8.43E+12
eu154	1.13E+13	9.40E+12	7.84E+12	6.55E+12	5.46E+12	4.56E+12	3.81E+12
eu155	4.37E+12	3.13E+12	2.25E+12	1.61E+12	1.16E+12	8.31E+11	5.96E+11

principal photon sources in group 4, photons/sec
mean energy = 0.0850 mev. nuclides exceeding 1.0E-02 of total group release rate (3.54E+13) at 4091.7 d
nuclide time after discharge

nuclide	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
sr 90	4.22E+12	3.99E+12	3.78E+12	3.58E+12	3.38E+12	3.20E+12	3.03E+12
y 90	3.52E+13	3.21E+13	3.04E+13	2.88E+13	2.72E+13	2.57E+13	2.44E+13
cs137	5.38E+12	5.11E+12	4.86E+12	4.61E+12	4.38E+12	4.16E+12	3.95E+12
eu155	6.63E+12	4.76E+12	3.41E+12	2.45E+12	1.76E+12	1.26E+12	9.05E+11

principal photon sources in group 5, photons/sec
mean energy = 0.1200 mev. nuclides exceeding 1.0E-02 of total group release rate (3.20E+13) at 4091.7 d
nuclide time after discharge

nuclide	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
sr 90	2.40E+12	2.27E+12	2.15E+12	2.04E+12	1.93E+12	1.82E+12	1.72E+12
y 90	2.47E+13	2.26E+13	2.14E+13	2.02E+13	1.91E+13	1.81E+13	1.71E+13
cs137	3.02E+12	2.87E+12	2.72E+12	2.58E+12	2.45E+12	2.33E+12	2.21E+12
eu154	2.09E+13	1.75E+13	1.46E+13	1.22E+13	1.02E+13	8.48E+12	7.08E+12
eu155	3.78E+12	2.72E+12	1.95E+12	1.40E+12	1.00E+12	7.20E+11	5.17E+11

principal photon sources in group 6, photons/sec
mean energy = 0.1700 mev. nuclides exceeding 1.0E-02 of total group release rate (2.24E+13) at 4091.7 d
nuclide time after discharge

nuclide	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
sr 90	1.71E+12	1.62E+12	1.53E+12	1.45E+12	1.37E+12	1.30E+12	1.23E+12
y 90	2.55E+13	2.33E+13	2.20E+13	2.09E+13	1.97E+13	1.87E+13	1.77E+13
cs137	2.13E+12	2.02E+12	1.92E+12	1.82E+12	1.73E+12	1.65E+12	1.56E+12

principal photon sources in group 7, photons/sec
mean energy = 0.3000 mev. nuclides exceeding 1.0E-02 of total group release rate (2.39E+13) at 4091.7 d
nuclide time after discharge

nuclide	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
sr 90	7.77E+11	7.35E+11	6.96E+11	6.58E+11	6.23E+11	5.89E+11	5.58E+11
y 90	2.83E+13	2.59E+13	2.45E+13	2.32E+13	2.19E+13	2.07E+13	1.96E+13
cs137	1.08E+12	1.02E+12	9.73E+11	9.23E+11	8.77E+11	8.33E+11	7.91E+11
eu154	3.20E+12	2.67E+12	2.23E+12	1.86E+12	1.55E+12	1.30E+12	1.08E+12

principal photon sources in group 8, photons/sec
mean energy = 0.6500 mev. nuclides exceeding 1.0E-02 of total group release rate (5.76E+14) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
y 90	1.20E+13	1.09E+13	1.03E+13	9.79E+12	9.26E+12	8.76E+12	8.29E+12
cs134	2.49E+15	1.17E+15	5.52E+14	2.60E+14	1.22E+14	5.76E+13	2.71E+13
ba137m	6.47E+14	6.10E+14	5.79E+14	5.50E+14	5.22E+14	4.96E+14	4.71E+14
eu154	2.68E+13	2.24E+13	1.87E+13	1.56E+13	1.30E+13	1.09E+13	9.06E+12

principal photon sources in group 9, photons/sec
mean energy = 1.1250 mev. nuclides exceeding 1.0E-02 of total group release rate (1.54E+13) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
y 90	1.56E+12	1.43E+12	1.35E+12	1.28E+12	1.21E+12	1.14E+12	1.08E+12
cs134	3.05E+13	1.43E+13	6.75E+12	3.18E+12	1.50E+12	7.05E+11	3.32E+11
eu154	3.32E+13	2.77E+13	2.31E+13	1.93E+13	1.61E+13	1.35E+13	1.12E+13

principal photon sources in group 10, photons/sec
mean energy = 1.5750 mev. nuclides exceeding 1.0E-02 of total group release rate (1.31E+12) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
y 90	1.99E+11	1.82E+11	1.72E+11	1.63E+11	1.54E+11	1.46E+11	1.38E+11
cs134	2.87E+13	1.35E+13	6.37E+12	3.00E+12	1.41E+12	6.66E+11	3.13E+11
eu154	1.20E+12	1.00E+12	8.37E+11	6.99E+11	5.83E+11	4.87E+11	4.06E+11

principal photon sources in group 11, photons/sec
mean energy = 2.0000 mev. nuclides exceeding 1.0E-02 of total group release rate (1.40E+10) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
y 90	1.19E+10	1.09E+10	1.03E+10	9.75E+09	9.22E+09	8.73E+09	8.26E+09
rh106	6.62E+12	1.33E+12	2.89E+11	6.28E+10	1.36E+10	2.97E+09	6.45E+08
pr144	4.36E+13	5.92E+12	8.09E+11	1.11E+11	1.51E+10	2.06E+09	2.82E+08
eu154	6.44E+08	5.37E+08	4.49E+08	3.74E+08	3.12E+08	2.61E+08	2.18E+08

principal photon sources in group 12, photons/sec
mean energy = 2.4000 mev. nuclides exceeding 1.0E-02 of total group release rate (1.68E+09) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
rh106	3.70E+12	7.43E+11	1.61E+11	3.51E+10	7.63E+09	1.66E+09	3.60E+08
pr144	4.12E+11	5.60E+10	7.65E+09	1.04E+09	1.43E+08	1.95E+07	2.66E+06

principal photon sources in group 13, photons/sec
mean energy = 2.8000 mev. nuclides exceeding 1.0E-02 of total group release rate (2.81E+08) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
rh106	6.24E+11	1.25E+11	2.72E+10	5.92E+09	1.29E+09	2.80E+08	6.08E+07

principal photon sources in group 14, photons/sec
mean energy = 3.2500 mev. nuclides exceeding 1.0E-02 of total group release rate (4.81E+07) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
rh106	1.07E+11	2.16E+10	4.68E+09	1.02E+09	2.21E+08	4.81E+07	1.05E+07

GAMMA SOURCE OF SPENT BWR FUEL APPENDIX D

principal photon sources in group 15, photons/sec
mean energy = 3.7500 mev. nuclides exceeding 1.0E-02 of total group release rate (2.12E+04) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
rh106	4.73E+07	9.50E+06	2.07E+06	4.49E+05	9.76E+04	2.12E+04	4.61E+03

principal photon sources in group 16, photons/sec
mean energy = 4.2500 mev. nuclides exceeding 1.0E-02 of total group release rate (6.91E-06) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
ce142	5.77E-06	5.77E-06	5.77E-06	5.77E-06	5.77E-06	5.77E-06	5.77E-06
sm147	4.44E-07	7.46E-07	9.13E-07	1.01E-06	1.06E-06	1.08E-06	1.10E-06

principal photon sources in group 17, photons/sec
mean energy = 4.7500 mev. nuclides exceeding 1.0E-02 of total group release rate (3.47E-06) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
ce142	2.89E-06	2.89E-06	2.89E-06	2.89E-06	2.89E-06	2.89E-06	2.89E-06
sm147	2.23E-07	3.74E-07	4.58E-07	5.04E-07	5.30E-07	5.44E-07	5.52E-07

principal photon sources in group 18, photons/sec
mean energy = 5.5000 mev. nuclides exceeding 1.0E-02 of total group release rate (2.57E-06) at 4091.7 d
nuclide time after discharge

	initial	818.3 d	1636.7 d	2455.0 d	3273.3 d	4091.7 d	4910.0 d
ce142	2.15E-06	2.15E-06	2.15E-06	2.15E-06	2.15E-06	2.15E-06	2.15E-06
sm147	1.65E-07	2.78E-07	3.40E-07	3.74E-07	3.93E-07	4.04E-07	4.09E-07