

# NUEN 647 Final Project

Uncertainty quantification of depletion calculations for specific isotopes using ORIGEN2.

## I Introduction

Determining composition of irradiated fuel is of importance for a myriad of reasons. Whether for flux calculations, reprocessing, or irradiation history verification, calculating fuel composition requires a Bateman solver, and a means for building a sparse matrix.

Applications using these compositions rarely report the uncertainty associated with results, even when inputs, such as flux shape, fission yield, cross sections, and half-lives have varying degrees of uncertainty. Further sources of error in this calculation are due to the multi-group approximation, and single point approximation, but will not be explored here.

Several isotope concentrations, shown in Table 1, were calculated as a function of burnup with the depletion code ORIGEN2 for a PWR system with 3 Wt% enriched uranium. ORIGEN2 solves the bateman equations with the matrix exponential method and requires a library with decay and cross section information. Cross sections and fission product yields are reduced to single group through flux averaging before execution of the code, with the assumption that the flux has the same shape as a typical PWR.

The uncertainty of concentrations were determined by varying the absorption and fission cross sections for  $^{235}\text{U}$ ,  $^{238}\text{U}$ ,  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$ , and  $^{241}\text{Pu}$ . Originally, absorption cross sections and yields for the fission products were to be varied, but variance information for the lighter nuclides is either difficult to acquire or not available.

The uncertainties on cross sections were determined by calculating the range of the single group cross section, taking the mid point as a mean, the range as a standard deviation, and assuming a Gaussian distribution.

**Table 1:** Isotope solve list.

$^{133}\text{Cs}$	$^{136}\text{Ba}$	$^{153}\text{Eu}$
$^{134}\text{Cs}$	$^{138}\text{Ba}$	$^{154}\text{Eu}$
$^{135}\text{Cs}$	$^{149}\text{Sm}$	$^{239}\text{Pu}$
$^{137}\text{Cs}$	$^{150}\text{Sm}$	$^{242}\text{Pu}$
$^{148}\text{Nd}$	$^{106}\text{Rh}$	$^{125}\text{Sb}$

## II Objectives

- ✓ Build ORIGEN2 model for thermal system which calculates concentrations of isotopes shown in Table 1.

**Listing 1:** PWR Input Deck

-1
-1
-1

```

5  RDA  Irradiation of 1 MT of PWR fuel
   RDA  Fuel enrichment is 3.0 w/o U-235
   RDA
   LIB  0  1  2  3  601  602  603  9  50  0  1  38
   PHO      101 102 103  10
   INP  1  1  -1  -1  1  1
10  BUP
   IRP 100.0 37.5 1 2 4 2 BURNUP=3,750 MWd/MT
   IRP 200.0 37.5 2 2 4 0 BURNUP=7,500 MWd/MT
   IRP 300.0 37.5 2 2 4 0 BURNUP=11,250 MWd/MT
   IRP 400.0 37.5 2 2 4 0 BURNUP=15,000 MWd/MT
15  DEC 500.0      2 3 4 0 DECAY FOR 100.0 DAYS
   DEC 4150.0      3 4 4 0 DECAY FOR 10 YEARS
   DEC 73500.0     4 5 4 0 DECAY FOR 200.0 YEARS
   BUP
   OPTL 24*8
20  OPTA 4*8 5 19*8
   OPTF 4*8 5 19*8
   OUT  5  1  -1  0
   END
2  922340 270. 922350 30000. 922380 969730. 0 0.0
25 0

```

The model irradiates 1 metric ton of US PWR fuel for a single cycle (15,000 MWd/Mt). The calculations use a constant power assumption of 37.5 W/g. The model does not include the oxygen because we are not interested in the activation of oxygen. Cross section modification throughout the calculation use the changing flux associated with a US PWR.

Initial verification of the model analyzed the end concentration of  $^{137}\text{Cs}$  and calculated the burn-up from that value. This calculation does not have an exact value for the yield of  $^{137}\text{Cs}$  and is used qualitatively as a sanity check.

$$\frac{552.8 \text{ g } ^{137}\text{Cs}}{\text{Mt}} \cdot \frac{6.022E23 \text{ atoms}}{137 \text{ g } ^{137}\text{Cs}} \cdot \frac{\text{Fission}}{0.06 \text{ atoms}} \cdot \frac{200 \text{ MeV}}{\text{Fission}} \cdot \frac{1.602E-19 \text{ MJ}}{1 \text{ MeV}} \cdot \frac{1 \text{ day}}{86400 \text{ s}} = 15,018 \frac{\text{MWd}}{\text{Mt}}$$

- ✓ Determine how to vary cross section and or flux spectrum inputs for calculation

ORIGEN2 reads in cross section information through a file named “TAPE9.inp”, specified by the 8th input on the LIB card. “TAPE9.inp” needs at least 3 cross section libraries. These are specified by the 5th 6th and 7th inputs on the LIB card as 601, 602, and 603 for the activation products, actinides, and fission products, respectively. The input for  $^{235}\text{U}$  from library 601 is shown in the listing below, with a corresponding key shown in Table 2 [1].

**Listing 2:**  $^{235}\text{U}$  cross section library 602 input

602	922350	1.068E+01	2.338E-03	8.049E-07	4.752E+01	0.0	0.0	-1.0
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**Table 2:** Key to parameters in cross section library

LIB	NUCLID	(n, $\gamma$ )	(n,2n)	(n,3n)	(n,f)	(n, $\gamma^*$ )	(n,2n <sup>*</sup> )	YYN
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The cross sections,  $\sigma_\gamma$  and  $\sigma_f$  will be modified based on the locations in the cross section libraries shown above. Half-life information is contained in the decay libraries 1, 2, and 3 for activation products, actinides, and fission products, respectively. These values will not be modified to reduce the scope of the project.

A program was written to modify  $\sigma_\gamma$ , yield, or half-life based on lines from a text file. This code is shown below but will only be used to modify cross section.

**Listing 3:** Script for modifying ORIGEN2 input.

```
#!/usr/bin/env python3

#Please note, might have to run command
# sed -i 's/E /E\+/g' TAPE9_BANK.inp
5 # on file to make sure there are no spaces after E's

#####
##### Import Packages #####
#####

10
import time
start_time = time.time()
import numpy as np
import Functions as Fun

15
#####
##### Class For Change #####
#####

20 class ChangeClass:
    def __init__(self):
        self.XSec      = False #X Section
        self.Y          = False #Yield
        self.HL         = False #Half Life
        25 self.LIB       = ""
        self.ID         = ""
        self.Mod        = 0

#####
30 ##### Calculations #####
#####

#Open Input File
with open('../Origen2/TAPE9_BANK.inp') as f:
35     content=f.readlines()

#Open output file
output=open("../Origen2/TAPE9.inp", "w")

40 #Grab all the changes you want to make from
#the 'ChangesToMake.txt' file C will be a list with
#Each item being a change to make (all items are the same class)
C=[]
with open('ChangesToMake.txt') as f:
45     Lines=f.readlines()
    for i in Lines:
```

```

i=i.split()
C.append(ChangeClass())
C[-1].XSec = eval(i[0])
50 C[-1].Y = eval(i[1])
C[-1].HL = eval(i[2])
C[-1].LIB = i[3]
C[-1].ID = i[4]
C[-1].Mod = i[5]
55 if C[-1].HL:
    C[-1].Mod2 = i[6] #Change time scale too for half life

#Loop through the TAPE9 file, make changes, and write to output
60 SecondLine=False
for i in content:
    hold=i.split()
    for M in C: #Look through all the Mods
        if M.LIB in hold[0] and "-" not in hold[0]:
            if M.ID in hold[1]:
                if M.XSec: #Replace the gamma x-section
                    i=i.replace(hold[2],M.Mod)
                if SecondLine: #If on second line,replaceU235yield
                    #Iftherewerea third line, thenthiswouldn't work
                    i=i.replace(hold[3],M.Mod)
70 SecondLine=False
                if M.Y: #Look to see if there is a second line
                    if hold[8]>0:
                        SecondLine=True #if so, then change yield
                        #next tim
75 if M.HL: #Change Half-life
                    #Count occurrences of integer to replace (before
                    #our occurrence), so
                    #we only replace the one we want to
                    Count=hold[0].count(hold[2])+hold[1].count(hold[2])+1
                    i=Fun.nth_repl(i,hold[2],M.Mod2,Count)
                    i=i.replace(hold[3],M.Mod)
80 i=i.replace("\n","")
    print(i,file=output)
85

#####
##### Time to Execute #####
#####
90 print("--- %s seconds ---" % (time.time() - start_time))

```

✓ Determine variance of Cross-sections

Both  $\sigma_\gamma$  and  $\sigma_f$  were determined by flux averaging via:

$$\sigma = \frac{\int \sigma(E) \phi(E) dE}{\int \phi(E) dE}$$

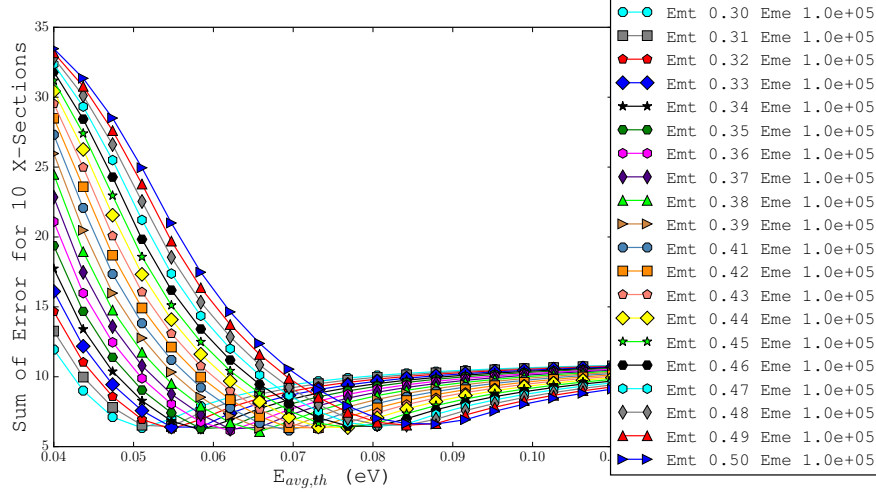
where:

$$\begin{aligned}
\phi(E) &= C_1 \cdot \frac{E}{E_0^2} \cdot \exp\left(-\frac{E}{E_0}\right) & E < E_{max,th} \\
&= \frac{C_2}{E} & E_{max,th} < E < E_{max,epi} \\
&= C_3 \cdot \frac{\sqrt{\frac{E}{E_f}}}{E_f} \cdot \exp\left(-\frac{E}{E_f}\right) & E > E_{max,epi}
\end{aligned}$$

and:

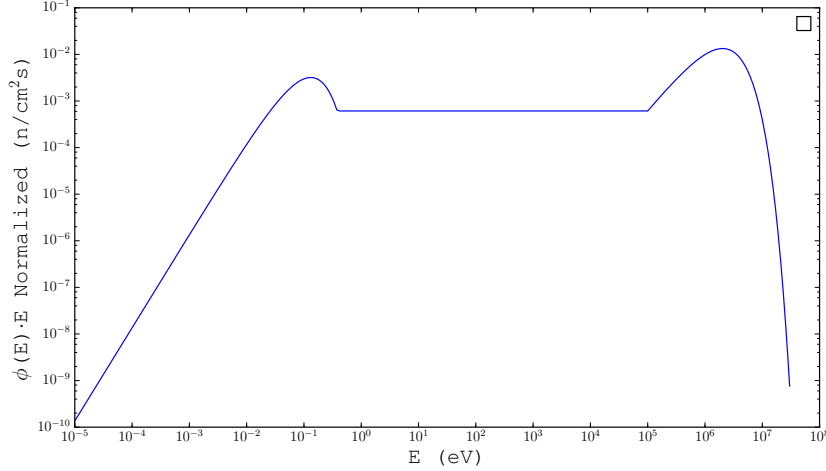
$$\begin{aligned}
C_1 &= \frac{E_0^2}{E_{max,th}^2} e^{E_{max,th}/E_0} \\
C_2 &= 1 \\
C_3 &= \frac{E_f}{E_{max,epi}} \cdot e^{\frac{E_{max,epi}}{E_f}} \frac{1}{\sqrt{\frac{E_{max,epi}}{E_f}}}
\end{aligned}$$

Where:  $E_{max,th} = 0.50$  eV,  $E_{max,epi} = 1E5$  eV,  $\theta_{th} = 0.09$  eV (764 K), and  $\theta_{fis} = 1.35E6$  eV. These values were picked because they minimized the difference between the cross-sections in the TAPE9 file, and those calculated with ENDF-VII and the above method. This is highlighted in figure 1. Most of this error is from  $^{238}\text{U}$  and  $^{240}\text{Pu}$ .



**Figure 1:** Minimized error for 10 cross section calculations

The flux spectrum is shown as below:



**Figure 2:** Flux Spectra used for weighting x-sections and yields

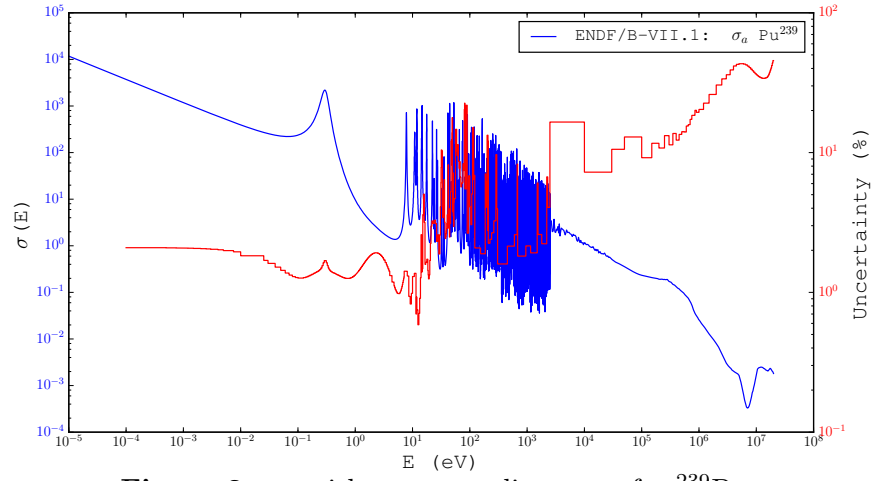
Table 3 shows the difference between my calculated cross section and the cross sections provided in ORIGEN2. The ratios will be used as a correction factor for the cross sections so that the input to ORIGEN2 will be constant. Differences have been attributed to the fact that the flux spectrum used does not capture resonances. The cross sections were calculated with ENDF VII, and ORIGEN2 could have been processed with ENDF V, which would further account for differences, but has not been verified. The reason ENDF VII is being used for the current analysis is because the variance information is included in ENDF VII and not ENDF V.

**Table 3:** Comparison of one-group cross sections

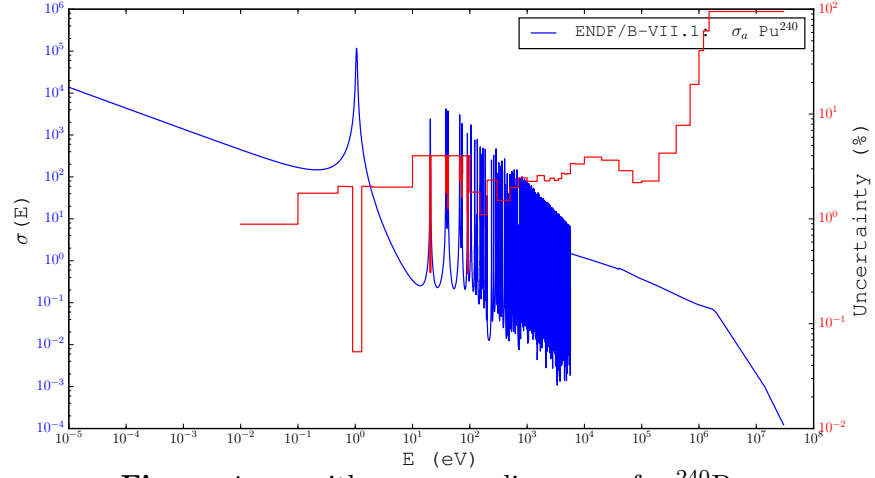
Isotope <sup>Rxn</sup>	ENDF VII	ORIGEN2	Ratio
<sup>239</sup> Pu <sup>γ</sup>	6.544e+01	6.909E+01	1.06
<sup>240</sup> Pu <sup>γ</sup>	1.521e+02	2.228E+02	1.46
<sup>241</sup> Pu <sup>γ</sup>	4.518e+01	4.202E+01	0.93
<sup>235</sup> U <sup>γ</sup>	9.387e+00	1.068E+01	1.14
<sup>238</sup> U <sup>γ</sup>	4.098e+00	8.872E-01	0.22
<sup>239</sup> Pu <sup>f</sup>	1.179e+02	1.211E+02	1.03
<sup>240</sup> Pu <sup>f</sup>	9.609e-01	5.787E-01	0.60
<sup>241</sup> Pu <sup>f</sup>	1.253e+02	1.259E+02	1.01
<sup>235</sup> U <sup>f</sup>	4.621e+01	4.752E+01	1.03
<sup>238</sup> U <sup>f</sup>	2.091e-01	9.281E-02	0.44

Both  $\sigma_{\gamma}^{error}$  and  $\sigma_f^{error}$  were determined by calculating the single group cross section with error subtracted, and then added. These values would constitute a mean with error.

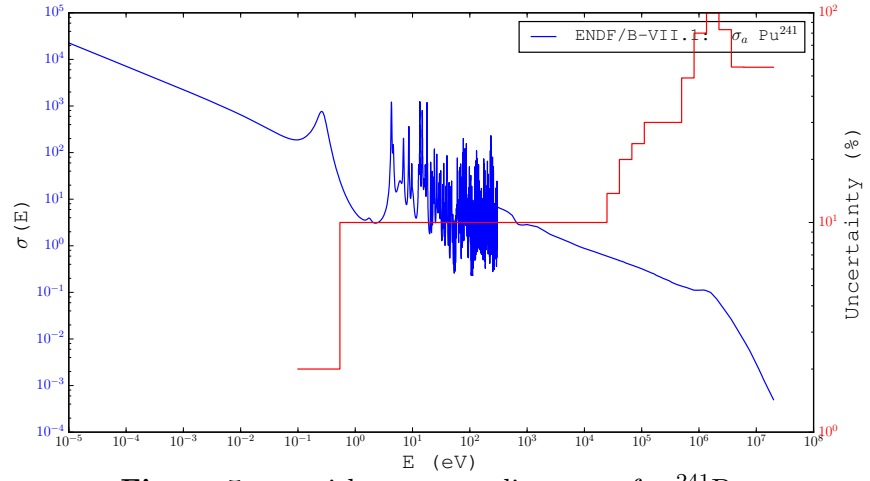
Cross section versus flux for several isotopes are shown in the following figures.



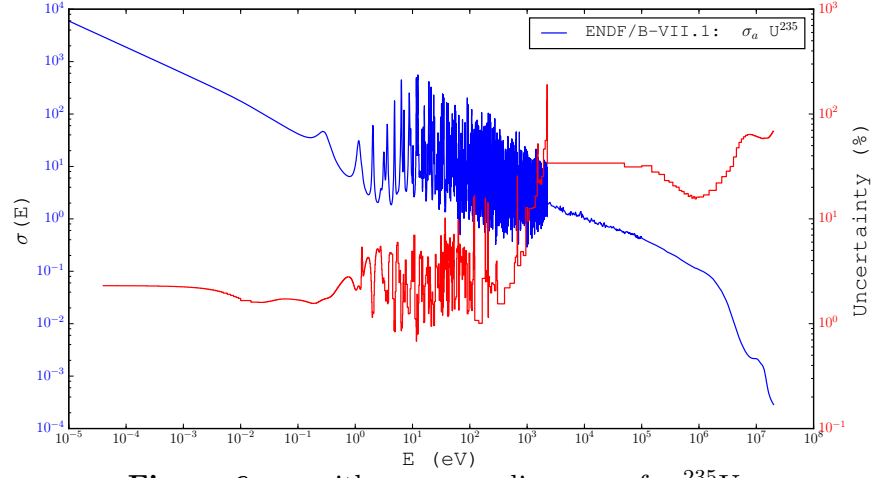
**Figure 3:**  $\sigma_a$  with corresponding error for  $^{239}\text{Pu}$



**Figure 4:**  $\sigma_a$  with corresponding error for  $^{240}\text{Pu}$

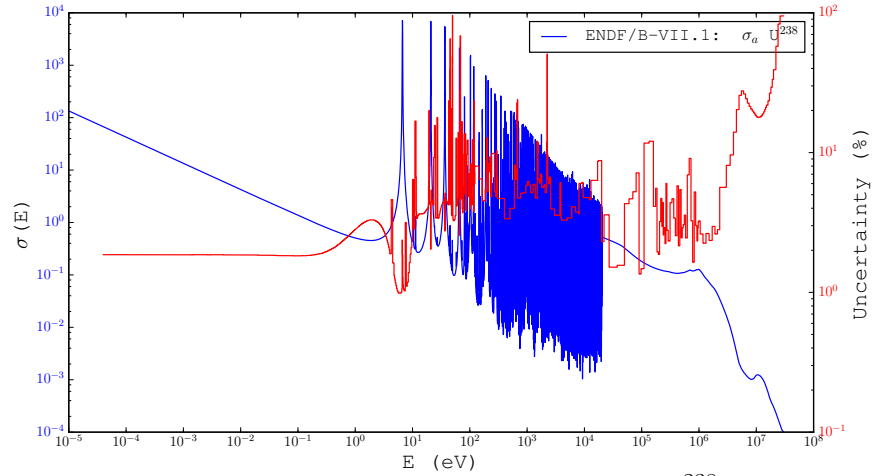


**Figure 5:**  $\sigma_a$  with corresponding error for  $^{241}\text{Pu}$



**Figure 6:**  $\sigma_a$  with corresponding error for  $^{235}\text{U}$





**Figure 7:**  $\sigma_a$  with corresponding error for  $^{238}\text{U}$

- ☐ Create a sampling space for all possible variations of calculations

With the following codes the sampling space was determined to be...

- ☐ Determine importance of various uncertain parameters by running the code a number of times randomly sampling the sample space (still not 100% sure how to do this - not even 50% sure how to do this)

### III Quantities of Interest and Uncertain Parameters

Quantities of interest are shown in Table 1 above. Uncertain parameters are listed below:

- Fission yield
- Cross sections
- Half-lives

### IV Prediction

The first major prediction for this project is that half-lives will not have a large impact on results because they are relatively well known. Secondly,  $^{125}\text{Sb}$  is notorious for being difficult to calculate correctly, I would predict that there would be large uncertainties due to uncertainties in the cross section data.

### References

- [1] Allen G Croff. User's manual for the origen2 computer code. Technical report, Oak Ridge National Lab., 1980.