

# Specification for Phase VII Benchmark

## UO<sub>2</sub> Fuel: Study of spent fuel compositions for long-term disposal

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### 1. Introduction

The concept of taking credit for a reduction in reactivity due to fuel burnup is commonly referred to as *burnup credit*. The reduction in reactivity that occurs with fuel burnup is due to the change in concentration (net reduction) of fissile nuclides and the production of actinide and fission-product neutron absorbers. After spent nuclear fuel (SNF) is discharged from a reactor, the reactivity continues to vary as a function of time due to the decay of unstable isotopes.

Burnup credit analyses for storage and transport consider timeframes that are extremely short (typically less than 100 years), as compared to the timeframe of interest to long-term disposal (e.g., 10,000 years after closure in the US). This benchmark proposes to study the ability of relevant computer codes and associated nuclear data to predict spent fuel isotopic compositions and corresponding  $k_{eff}$  values in a cask configuration over the time duration relevant to SNF disposal. It is recognized that the benefits of this proposed benchmark are largely confined to revealing differences in nuclear data for decay constants (half-lives, branching fractions), which are widely considered to be well-known. However, the results of this exercise may serve to reveal differences in international nuclear data sets and/or improve understanding and confidence in our ability to predict  $k_{eff}$  and source terms for timeframes relevant to long-term disposal of SNF.

The benchmark is divided into two sets of calculations:

- Decay calculations for provided PWR UO<sub>2</sub> discharged fuel compositions
- Criticality ( $k_{eff}$ ) calculations for a representative cask model

Participants are requested to perform decay calculations, using the provided PWR fuel compositions as a starting point, and criticality calculations for the PWR fuel in the OECD cask model for 30 post-irradiation time steps, out to 1,000,000 years. Decay calculations will be performed for nuclides that are relevant to burnup credit and to public dose from nuclear waste repositories. Although it is acknowledged that the physical condition of the fuel will change over such a long time period, there is interest in the change in isotopic compositions over this duration, as well as interest in the relative behavior of  $k_{eff}$  over this duration. Analysis of the results will involve comparison of participant's isotopic compositions and  $k_{eff}$  values as a function of time.

## 2. Decay Calculations

The isotopes that are considered relevant to burnup-credit criticality calculations and those that are potential contributors to radiation dose to the public from nuclear waste repositories are listed in the “Benchmark nuclide” column in Table 1. Utilizing the discharge fuel composition provided in Table 1 for a representative PWR assembly of 4.5-wt% <sup>235</sup>U initial enrichment and 50-GWd/MTU burnup, participants are requested to perform decay calculations and report atom densities for the Benchmark nuclides designated in Table 1 and times listed in Table 2. The discharge fuel composition provided contains only the Benchmark nuclides designated in Table 1 and their precursors that are relevant to the decay calculations. The nuclide atom densities for the discharge fuel composition, in atom/barn-cm, are provided with four significant digits in Table 1 and in a text file attached to this benchmark specification.

**Table 1. Discharge fuel composition (4.5 initial wt% <sup>235</sup>U, 50-GWd/MTU) for calculating time-dependent spent fuel compositions**

Isotope	Atom density (atom/barn-cm)	Benchmark nuclide <sup>a</sup>	Area of applicability		
			Actinide-only burnup credit	Actinide + FP burnup credit	Public dose
<sup>14</sup> C	1.8462E-09	X			X
<sup>16</sup> O <sup>b</sup>	4.7923E-02		X	X	
<sup>36</sup> Cl <sup>c</sup>	1.0000E-06	X			X
<sup>41</sup> Ca <sup>c</sup>	1.0000E-06	X			X
<sup>59</sup> Ni <sup>c</sup>	1.0000E-06	X			X
<sup>79</sup> Se	5.0582E-07	X			X
<sup>93</sup> Zr	6.3637E-05	X			X
<sup>93</sup> Rb	1.6072E-12				
<sup>90</sup> Sr	4.8584E-05	X			X
<sup>93</sup> Sr	2.3719E-10				
<sup>93</sup> Y	1.9886E-08				
<sup>95</sup> Y	3.8958E-10				
<sup>93m</sup> Nb	6.6305E-11	X			X
<sup>94</sup> Nb	6.2143E-11	X			X
<sup>95</sup> Nb	1.9348E-06				
<sup>93</sup> Mo	1.1478E-14	X			X
<sup>95</sup> Mo	6.0803E-05	X		X	

<sup>a</sup> Nuclides that are relevant to either burnup credit or public dose. Benchmark nuclides selected based on a review of the following references and other preliminary European studies.

Ref. 1: J. C. Wagner and C. E. Sanders, *Assessment of Reactivity Margins and Loading Curves for PWR Burnup Credit Cask Analyses*, NUREG/CR-6800 (ORNL/TM-2002/6), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 2003.

Ref. 2: *Radionuclide Screening*, ANL-WIS-MD-000006 REV 02, Sandia National Laboratories, Las Vegas, Nevada (2007).

Ref. 3: *Project Opalinus Clay Safety Report: Demonstration of Disposal Feasibility for Spent Fuel, Vitrified High-Level Waste and Long-Lived Intermediate-Level Waste (Entsorgungsnachweis)*, NAGRA Technical Report NTB 02-05, NAGRA, Wettingen, Switzerland (2002).

<sup>b</sup> <sup>16</sup>O concentration is provided for criticality calculations only.

<sup>c</sup> The nuclide does not exist in the calculated discharge inventory for the PWR assembly.

**Table 1. Discharge fuel composition (4.5 initial wt% <sup>235</sup>U, 50-GWd/MTU) for calculating time-dependent spent fuel compositions (continued)**

Isotope	Atom density (atom/barn-cm)	Benchmark nuclide <sup>a</sup>	Area of applicability		
			Actinide- only burnup credit	Actinide + FP burnup credit	Public dose
<sup>99</sup> Mo	1.7898E-07				
<sup>101</sup> Mo	6.2291E-10				
<sup>99m</sup> Tc	1.4716E-08				
<sup>99</sup> Tc	6.6733E-05	X		X	X
<sup>101</sup> Tc	6.0649E-10				
<sup>101</sup> Ru	6.5615E-05	X		X	
<sup>103</sup> Ru	2.3665E-06				
<sup>103</sup> Rh	3.4702E-05	X		X	
<sup>107</sup> Pd	1.8503E-05	X			X
<sup>109</sup> Pd	8.5605E-09				
<sup>109</sup> Ag	6.0729E-06	X		X	
<sup>126</sup> Sn	1.2852E-06	X			X
<sup>126</sup> Sb	2.3835E-10	X			X
<sup>126m</sup> Sb	3.1762E-13	X			X
<sup>129</sup> Sb	1.9663E-09				
<sup>129m</sup> Te	6.4572E-08				
<sup>129</sup> I	1.0251E-05	X			X
<sup>133</sup> I	6.1772E-08				
<sup>135</sup> I	1.8776E-08				
<sup>133</sup> Xe	3.7597E-07				
<sup>135</sup> Xe	1.0156E-08				
<sup>133</sup> Cs	6.9874E-05	X		X	
<sup>135</sup> Cs	3.4626E-05	X			X
<sup>137</sup> Cs	7.5114E-05	X			X
<sup>143</sup> Pr	6.9646E-07				
<sup>147</sup> Pr	2.1435E-10				
<sup>149</sup> Pr	2.0197E-11				
<sup>143</sup> Nd	4.6567E-05	X		X	
<sup>145</sup> Nd	3.8049E-05	X		X	
<sup>147</sup> Nd	2.5090E-07				
<sup>149</sup> Nd	9.9358E-10				
<sup>147</sup> Pm	7.4886E-06				
<sup>149</sup> Pm	4.4831E-08				
<sup>151</sup> Pm	8.6483E-09				
<sup>147</sup> Sm	4.7086E-06	X		X	
<sup>149</sup> Sm	1.3806E-07	X		X	
<sup>150</sup> Sm	1.6254E-05	X		X	
<sup>151</sup> Sm	9.7106E-07	X		X	X
<sup>152</sup> Sm	6.3220E-06	X		X	
<sup>153</sup> Sm	3.8127E-08				
<sup>155</sup> Sm	2.3852E-11				
<sup>151</sup> Eu	1.5639E-09	X		X	

<sup>a</sup> See footnote "a" on page 2 of 17.

**Table 1. Discharge fuel composition (4.5 initial wt% <sup>235</sup>U, 50-GWd/MTU) for calculating time-dependent spent fuel compositions (continued)**

Isotope	Atom density (atom/barn-cm)	Benchmark nuclide <sup>a</sup>	Area of applicability		
			Actinide- only burnup credit	Actinide + FP burnup credit	Public dose
<sup>152</sup> Eu	3.2421E-09				
<sup>153</sup> Eu	6.6248E-06	X		X	
<sup>155</sup> Eu	3.4461E-07				
<sup>155</sup> Gd	5.4622E-09	X		X	
<sup>210</sup> Pb	3.8862E-18	X			X
<sup>222</sup> Rn	9.3947E-21				
<sup>226</sup> Ra	1.4422E-15	X			X
<sup>228</sup> Ra	9.3958E-22	X			X
<sup>227</sup> Ac	3.2593E-16	X			X
<sup>226</sup> Th	1.9447E-22				
<sup>229</sup> Th	5.9651E-14	X			X
<sup>230</sup> Th	5.5098E-11	X			X
<sup>232</sup> Th	1.2690E-11	X			X
<sup>231</sup> Th	4.8649E-14				
<sup>231</sup> Pa	1.8739E-11	X			X
<sup>230</sup> U	1.8514E-19				
<sup>232</sup> U	2.3106E-11	X			X
<sup>233</sup> U	8.7082E-11	X	X	X	X
<sup>234</sup> U	4.5729E-06	X	X	X	X
<sup>235</sup> U	2.4950E-04	X	X	X	X
<sup>236</sup> U	1.5044E-04	X		X	X
<sup>237</sup> U	2.7902E-07				
<sup>238</sup> U	2.1947E-02	X	X	X	X
<sup>239</sup> U	1.2843E-08				
<sup>240</sup> U	1.3550E-20				
<sup>235</sup> Np	6.7582E-13				
<sup>236</sup> Np	1.2422E-11				
<sup>236m</sup> Np	2.8487E-13				
<sup>237</sup> Np	1.9889E-05	X		X	X
<sup>238</sup> Np	4.8303E-08				
<sup>239</sup> Np	1.8489E-06				
<sup>240</sup> Np	5.2420E-11				
<sup>236</sup> Pu	3.5918E-11				
<sup>237</sup> Pu	2.0813E-11				
<sup>238</sup> Pu	9.3508E-06	X	X	X	X
<sup>239</sup> Pu	1.8344E-04	X	X	X	X
<sup>240</sup> Pu	7.2862E-05	X	X	X	X
<sup>241</sup> Pu	4.7994E-05	X	X	X	X
<sup>242</sup> Pu	1.9005E-05	X	X	X	X
<sup>243</sup> Pu	4.6387E-09				
<sup>244</sup> Pu	6.7468E-10				

<sup>a</sup> See footnote "a" on page 2 of 17.

**Table 1. Discharge fuel composition (4.5 initial wt% <sup>235</sup>U, 50-GWd/MTU) for calculating time-dependent spent fuel compositions (continued)**

Isotope	Atom density (atom/barn-cm)	Benchmark nuclide <sup>a</sup>	Area of applicability		
			Actinide-only burnup credit	Actinide + FP burnup credit	Public dose
<sup>245</sup> Pu	3.2226E-14				
<sup>246</sup> Pu	2.2173E-16				
<sup>239</sup> Am	1.8587E-16				
<sup>240</sup> Am	8.0708E-14				
<sup>241</sup> Am	2.2311E-06	X	X	X	X
<sup>242</sup> Am	3.8342E-09				
<sup>242m</sup> Am	5.2630E-08	X		X	X
<sup>243</sup> Am	5.6091E-06	X		X	X
<sup>242</sup> Cm	5.9799E-07				
<sup>243</sup> Cm	2.2037E-08				
<sup>244</sup> Cm	2.3542E-06				
<sup>245</sup> Cm	1.3952E-07	X			X
<sup>246</sup> Cm	1.3896E-08	X			X

<sup>a</sup>See footnote “a” on page 2 of 17.

**Table 2: Times for calculating and reporting isotopic compositions**

Time case number	Time (y)	Time case number	Time (y)
1	0	16	1000
2	1	17	2000
3	2	18	5000
4	5	19	8000
5	10	20	10,000
6	20	21	15,000
7	40	22	20,000
8	60	23	25,000
9	80	24	30,000
10	100	25	40,000
11	120	26	45,000
12	150	27	50,000
13	200	28	100,000
14	300	29	500,000
15	500	30	1,000,000

### 3. *k<sub>eff</sub>* Calculations

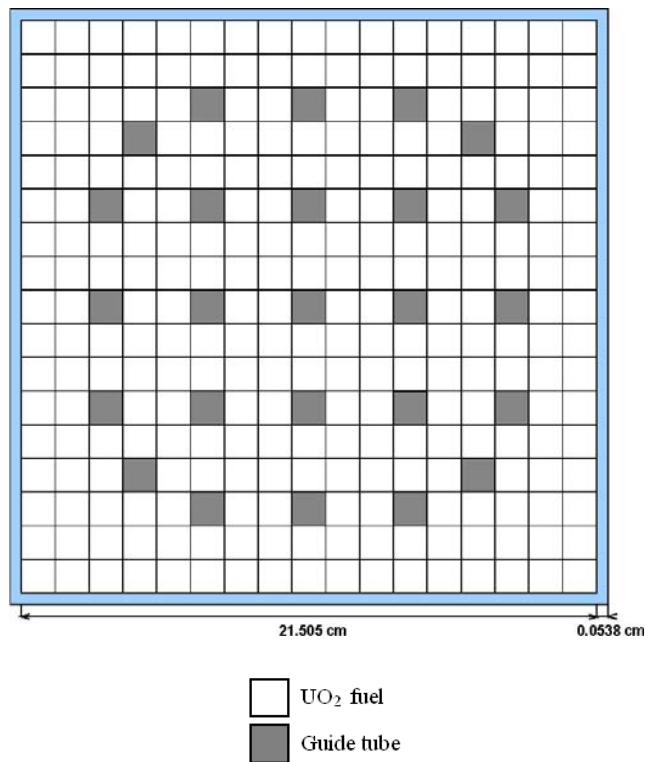
Criticality calculations are to be performed for a representative PWR cask model utilizing the PWR spent fuel isotopic compositions from the decay calculations for nuclides relevant to burnup credit corresponding to the times listed in Table 2. The cask model to be used is described below and is identical to the cask model used in Phase IID of the Expert Group on Burn-up Credit Benchmarks. *k<sub>eff</sub>* values will be calculated for both actinide only and actinide and fission product cases. The actinide only cases should include <sup>16</sup>O and the nuclides identified as “Set 1” in Table 3. The actinide and fission product case should include <sup>16</sup>O and the nuclides identified as “Set 2” in Table 3.

**Table 3: Nuclide sets to be used in  $k_{eff}$  calculations**

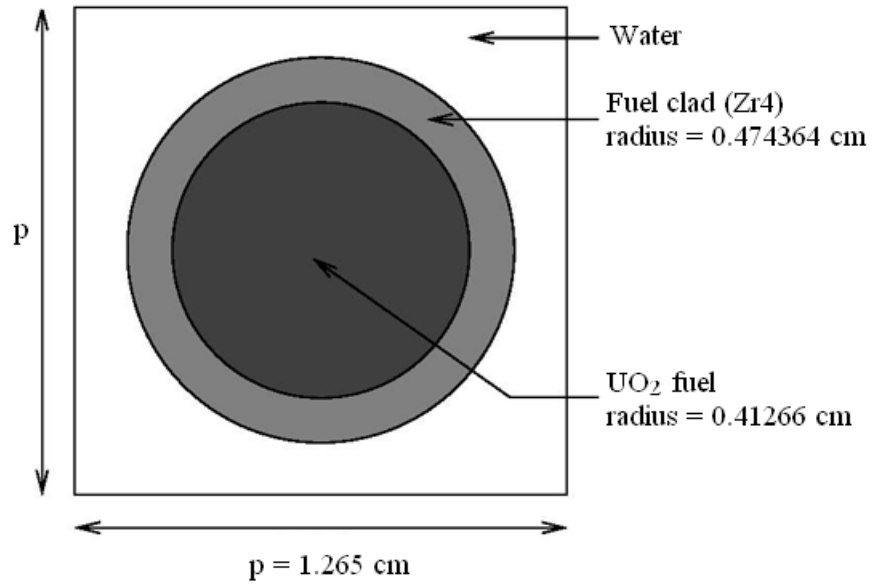
<i>Set 1: Actinide-only burnup-credit nuclides (11 total)</i>
$^{233}\text{U}$ , $^{234}\text{U}$ , $^{235}\text{U}$ , $^{236}\text{U}$ , $^{238}\text{U}$ , $^{238}\text{Pu}$ , $^{239}\text{Pu}$ , $^{240}\text{Pu}$ , $^{241}\text{Pu}$ , $^{242}\text{Pu}$ , and $^{241}\text{Am}$
<i>Set 2: Actinide + fission product burnup-credit nuclides (30 total)</i>
$^{233}\text{U}$ , $^{234}\text{U}$ , $^{235}\text{U}$ , $^{236}\text{U}$ , $^{238}\text{U}$ , $^{237}\text{Np}$ , $^{238}\text{Pu}$ , $^{239}\text{Pu}$ , $^{240}\text{Pu}$ , $^{241}\text{Pu}$ , $^{242}\text{Pu}$ , $^{241}\text{Am}$ , $^{242m}\text{Am}$ , $^{243}\text{Am}$ , $^{95}\text{Mo}$ , $^{99}\text{Tc}$ , $^{101}\text{Ru}$ , $^{103}\text{Rh}$ , $^{109}\text{Ag}$ , $^{133}\text{Cs}$ , $^{143}\text{Nd}$ , $^{145}\text{Nd}$ , $^{147}\text{Sm}$ , $^{149}\text{Sm}$ , $^{150}\text{Sm}$ , $^{151}\text{Sm}$ , $^{152}\text{Sm}$ , $^{151}\text{Eu}$ , $^{153}\text{Eu}$ , and $^{155}\text{Gd}$

### 3.1 Geometry data

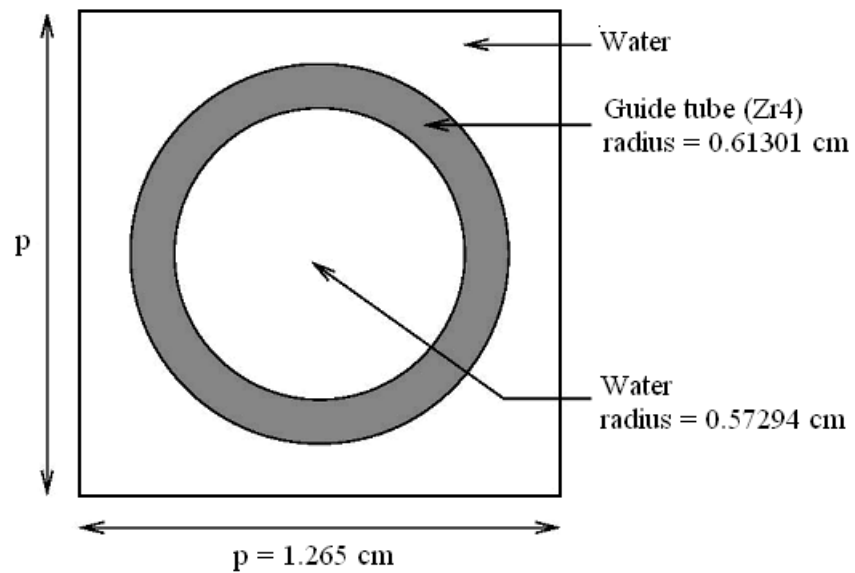
The representative OECD cask loaded with intact  $\text{UO}_2$  17 × 17 assemblies is the criticality model for  $k_{eff}$  calculations. The  $\text{UO}_2$  assembly geometry and the locations for 25 guide tubes are illustrated in Figure 1. Fuel rod and guide tube radial dimensions are shown in Figures 2 and 3, respectively. Cross-section views of the cask model for use in criticality calculations are provided in Attachment I, Figures 4 through 6.



**Figure 1 :  $\text{UO}_2$  assembly geometry and guide tube locations**



**Figure 2 : Fuel rod geometry**



**Figure 3 : Guide tube geometry**

## **Material and geometrical descriptions**

### **Fuel assembly**

#### *Fuel rod data*

Rod pitch	1.265 cm
Rod length	365.7 cm (active fuel and guide tube)
Endplug material	Zircaloy 4
Endplug height	1.75 cm
Full rod length	369.2 cm (fuel + 2 endplug)
Radial dimensions	See Figure 2.

#### *Assembly data*

Lattice	17 × 17 (264 fuel rods, 25 guide tubes) (See Figure 1.)
Dimensions	21.505 × 21.505 × 409.2 cm <sup>3</sup>
Moderator	Water
Guide tube radial dimensions	See Figure 3.
Upper and lower end hardware	50% stainless steel, 50% water (by volume) (Note: The assembly upper and lower end hardware will be modeled as a region of smeared water and stainless steel; other hardware, such as grid spacers, is ignored).
Upper hardware height	30.0 cm
Lower hardware height	10.0 cm
Upper water region height	7.0 cm
Lower water region	0.0 cm

### **Cask**

#### *Cask shell*

Inner diameter	136.0 cm
Outer diameter	196.0 cm
Material	Stainless steel (SS304)
Total height	476.2 cm
Inner cavity height	416.2 cm

#### *Assembly basket*

Inner basket compartment dimensions	22 × 22 × 416.2 cm <sup>3</sup>
Material	Borated stainless steel (1 wt% boron)
Basket wall thickness	1 cm

#### *Configuration*

21 assemblies positioned in a 5 × 5 array (without assembly in corner).  
Fuel assemblies are centered within basket region.  
Cask is completely flooded with water. The temperature for cask components is 293K.

### 3.2 Material compositions

The criticality calculations will use spent fuel compositions from decay calculations (see Section 2) and the nuclide densities, in atom/barn-cm, for the other assembly and cask materials provided in this section.

<i>Fresh fuel</i>	<sup>234</sup> U	9.5013E-06
	<sup>235</sup> U	1.0916E-03
	<sup>236</sup> U	5.0726E-06
	<sup>238</sup> U	2.2859E-02
	<sup>16</sup> O	4.7923E-02

*Spent fuel* Spent fuel compositions corresponding to the time cases listed in Table 2 will include:

- 1) actinide-only cases: <sup>16</sup>O and the nuclides identified as “Set 1” in Table 3.
- 2) actinides + fission products cases: <sup>16</sup>O and the nuclides identified as “Set 2” in Table 3.

<i>Fuel Clad</i>	Fe	1.383E-04
	Cr	7.073E-05
	O	2.874E-04
	Zr	3.956E-02

<i>End plug</i>	Cr	7.589E-05
	Fe	1.484E-04
	Zr	4.298E-02

<i>Guide tube</i>	Fe	1.476E-04
	Cr	7.549E-05
	O	3.067E-04
	Zr	4.222E-02

<i>Water</i>	H	6.662E-02
	O	3.331E-02

<i>Stainless steel</i>	Cr	1.743E-02
	Mn	1.736E-03
	Fe	5.936E-02
	Ni	7.721E-03

<i>Borated (1 wt%) stainless steel</i>	Cr	1.691E-02
	Mn	1.684E-03
	Fe	5.758E-02
	Ni	7.489E-03
	<sup>10</sup> B	7.836E-04
	<sup>11</sup> B	3.181E-03

<i>50/50 stainless steel/ water mixture</i>	Cr	8.714E-03
	Mn	8.682E-04
	Fe	2.968E-02
	Ni	3.860E-03
	H	3.338E-02
	O	1.669E-02

#### 4. Parameters required

##### 4.1 Fuel compositions

Provide atom densities, in atom/barn-cm, for the light element, actinide, and fission product nuclides designated as Benchmark nuclides in Table 1 for each of the time case numbers listed in Table 2. The reported values will contain four significant digits.

##### 4.2 $k_{eff}$ calculations

Provide  $k_{eff}$  values for fresh fuel and isotopic compositions from the decay calculations for cases involving only the actinides and cases involving both the actinides and the fission products. The total number of  $k_{eff}$  calculation cases is 61 (1 fresh fuel composition + 30 decay-time steps  $\times$  2 burnup-credit nuclide sets). Standard deviation values will be reported for the  $k_{eff}$  values calculated using a Monte Carlo transport code. The reported values should contain four significant digits.

Criticality calculation cases 1 will use fresh fuel isotopic composition.

Criticality calculation cases 2 through 31 (see Set 1 in Table 3) will use isotopic compositions that contain actinides only and correspond to decay time case numbers 1 through 30, respectively.

Criticality calculation cases 32 through 61 will use isotopic compositions that contain actinides and fission products (see Set 2 in Table 3) and correspond to decay time case numbers 1 through 30, respectively.

#### 5. Requested Information and Results

Forward the results via e-mail to the benchmark coordinator, John Wagner ([wagnerjc@ornl.gov](mailto:wagnerjc@ornl.gov)). The results should be provided in two files according to the format instructions provided below.

##### 5.1 Spent fuel composition results

The "spent fuel composition results" file must be composed of:

<u>Line No.</u>	<u>Contents</u>
1	"PWR assembly: 4.5 wt% <sup>235</sup> U enrichment and 50 GWd/MTU burnup"
2	Date
3	Institute
4	Contact Person
5	E-mail address or Telefax Number of the contact person
6	Computer Code

7 \*Time case 2\*

8 Nuclide density (atom/barn-cm) of  $^{14}\text{C}$

9 Nuclide density (atom/barn-cm) of  $^{36}\text{Cl}$

10 Nuclide density (atom/barn-cm) of  $^{41}\text{Ca}$

11 Nuclide density (atom/barn-cm) of  $^{59}\text{Ni}$

12 Nuclide density (atom/barn-cm) of  $^{79}\text{Se}$

13 Nuclide density (atom/barn-cm) of  $^{93}\text{Zr}$

14 Nuclide density (atom/barn-cm) of  $^{90}\text{Sr}$

15 Nuclide density (atom/barn-cm) of  $^{93\text{m}}\text{Nb}$

16 Nuclide density (atom/barn-cm) of  $^{94}\text{Nb}$

17 Nuclide density (atom/barn-cm) of  $^{93}\text{Mo}$

18 Nuclide density (atom/barn-cm) of  $^{95}\text{Mo}$

19 Nuclide density (atom/barn-cm) of  $^{99}\text{Tc}$

20 Nuclide density (atom/barn-cm) of  $^{101}\text{Ru}$

21 Nuclide density (atom/barn-cm) of  $^{103}\text{Rh}$

22 Nuclide density (atom/barn-cm) of  $^{107}\text{Pd}$

23 Nuclide density (atom/barn-cm) of  $^{109}\text{Ag}$

24 Nuclide density (atom/barn-cm) of  $^{126}\text{Sn}$

25 Nuclide density (atom/barn-cm) of  $^{126}\text{Sb}$

26 Nuclide density (atom/barn-cm) of  $^{126\text{m}}\text{Sb}$

27 Nuclide density (atom/barn-cm) of  $^{129}\text{I}$

28 Nuclide density (atom/barn-cm) of  $^{133}\text{Cs}$

29 Nuclide density (atom/barn-cm) of  $^{135}\text{Cs}$

30 Nuclide density (atom/barn-cm) of  $^{137}\text{Cs}$

31 Nuclide density (atom/barn-cm) of  $^{143}\text{Nd}$

32 Nuclide density (atom/barn-cm) of  $^{145}\text{Nd}$

33 Nuclide density (atom/barn-cm) of  $^{147}\text{Sm}$

34 Nuclide density (atom/barn-cm) of  $^{149}\text{Sm}$

35 Nuclide density (atom/barn-cm) of  $^{150}\text{Sm}$

36 Nuclide density (atom/barn-cm) of  $^{151}\text{Sm}$

37 Nuclide density (atom/barn-cm) of  $^{152}\text{Sm}$

38 Nuclide density (atom/barn-cm) of  $^{151}\text{Eu}$

39 Nuclide density (atom/barn-cm) of  $^{153}\text{Eu}$

40 Nuclide density (atom/barn-cm) of  $^{155}\text{Gd}$

41 Nuclide density (atom/barn-cm) of  $^{210}\text{Pb}$

42 Nuclide density (atom/barn-cm) of  $^{226}\text{Ra}$

43 Nuclide density (atom/barn-cm) of  $^{228}\text{Ra}$

44 Nuclide density (atom/barn-cm) of  $^{227}\text{Ac}$

45 Nuclide density (atom/barn-cm) of  $^{229}\text{Th}$

46 Nuclide density (atom/barn-cm) of  $^{230}\text{Th}$

47 Nuclide density (atom/barn-cm) of  $^{232}\text{Th}$

48 Nuclide density (atom/barn-cm) of  $^{231}\text{Pa}$

49 Nuclide density (atom/barn-cm) of  $^{232}\text{U}$

50 Nuclide density (atom/barn-cm) of  $^{233}\text{U}$

51 Nuclide density (atom/barn-cm) of  $^{234}\text{U}$

52	Nuclide density (atom/barn-cm) of $^{235}\text{U}$
53	Nuclide density (atom/barn-cm) of $^{236}\text{U}$
54	Nuclide density (atom/barn-cm) of $^{238}\text{U}$
55	Nuclide density (atom/barn-cm) of $^{237}\text{Np}$
56	Nuclide density (atom/barn-cm) of $^{238}\text{Pu}$
57	Nuclide density (atom/barn-cm) of $^{239}\text{Pu}$
58	Nuclide density (atom/barn-cm) of $^{240}\text{Pu}$
59	Nuclide density (atom/barn-cm) of $^{241}\text{Pu}$
60	Nuclide density (atom/barn-cm) of $^{242}\text{Pu}$
61	Nuclide density (atom/barn-cm) of $^{241}\text{Am}$
62	Nuclide density (atom/barn-cm) of $^{242\text{m}}\text{Am}$
63	Nuclide density (atom/barn-cm) of $^{243}\text{Am}$
64	Nuclide density (atom/barn-cm) of $^{245}\text{Cm}$
65	Nuclide density (atom/barn-cm) of $^{246}\text{Cm}$
66	*Time case 3*
67 to 124	As for items 8 to 65
125	* Time case 4*
126 to 183	As for items 8 to 65
184	* Time case 5*
185 to 242	As for items 8 to 66
243	* Time case 6*
244 to 301	As for items 8 to 65
302	* Time case 7*
303 to 360	As for items 8 to 65
361	* Time case 8*
362 to 419	As for items 8 to 65
420	* Time case 9*
421 to 478	As for items 8 to 65
479	* Time case 10*
480 to 537	As for items 8 to 65
538	* Time case 11*
539 to 596	As for items 8 to 65
597	* Time case 12*
598 to 655	As for items 8 to 65
656	* Time case 13*
657 to 714	As for items 8 to 65
715	* Time case 14*
716 to 773	As for items 8 to 65
774	* Time case 15*
775 to 832	As for items 8 to 65
833	* Time case 16*
834 to 891	As for items 8 to 65
892	* Time case 17*
893 to 950	As for items 8 to 65
951	* Time case 18*

- 952 to 1009 As for items 8 to 65  
 1010 \* Time case 19\*
- 1011 to 1068 As for items 8 to 65  
 1069 \* Time case 20\*
- 1070 to 1127 As for items 8 to 65  
 1128 \* Time case 21\*
- 1129 to 1186 As for items 8 to 66  
 1187 \* Time case 22\*
- 1188 to 1245 As for items 8 to 66  
 1246 \* Time case 23\*
- 1247 to 1304 As for items 8 to 65  
 1305 \* Time case 24\*
- 1306 to 1363 As for items 8 to 65  
 1364 \* Time case 25\*
- 1365 to 1422 As for items 8 to 65  
 1423 \* Time case 26\*
- 1424 to 1481 As for items 8 to 65  
 1482 \* Time case 27\*
- 1483 to 1540 As for items 8 to 65  
 1541 \* Time case 28\*
- 1542 to 1599 As for items 8 to 65  
 1600 \* Time case 29\*
- 1601 to 1658 As for items 8 to 65  
 1659 \* Time case 30\*
- 1660 to 1717 As for items 8 to 65
- 1718 Please describe your analysis environment here. It will be included in the benchmark report. The description should include:  
     Institute and Country, Participants,  
     Neutron data library,  
     Neutron data processing code or method,  
     Description of your code system,  
     Omitted nuclides if any,  
     Omitted cases if any,  
     Other related information.

## 5.2 $k_{eff}$ values

The "keff results" file must be composed of:

<u>Line No.</u>	<u>Contents</u>
1	"keff calculation"
2	Date
3	Institute
4	Contact Person
5	E-mail address or Telefax Number of the contact person
6	Computer Code
7	"actinide only"

- 8  $k_{eff}$  ("±" standard deviation, if applicable) value for fresh fuel
- 9 to 38  $k_{eff}$  ("±" standard deviation, if applicable) values for cases 1 through 30 (see Section 4.2 for case description).
- 39 "actinides + fission products"
- 40  $k_{eff}$  ("±" standard deviation, if applicable) value for fresh fuel
- 41 to 70  $k_{eff}$  ("±" standard deviation, if applicable) values for cases 31 through 60 (see Section 4.2 for case description).
- 71 Please describe your analysis environment here. It will be included in the benchmark report. The description should include:
- Institute and Country,
  - Participants,
  - Description of your code system,
  - Neutron data library,
  - Neutron data processing code or method,
  - Neutron energy groups,
  - Geometry modeling (3-D, 2-D etc.),
  - Omitted nuclides if any,
  - Omitted cases if any,
  - Other related information.

## 6. Schedule

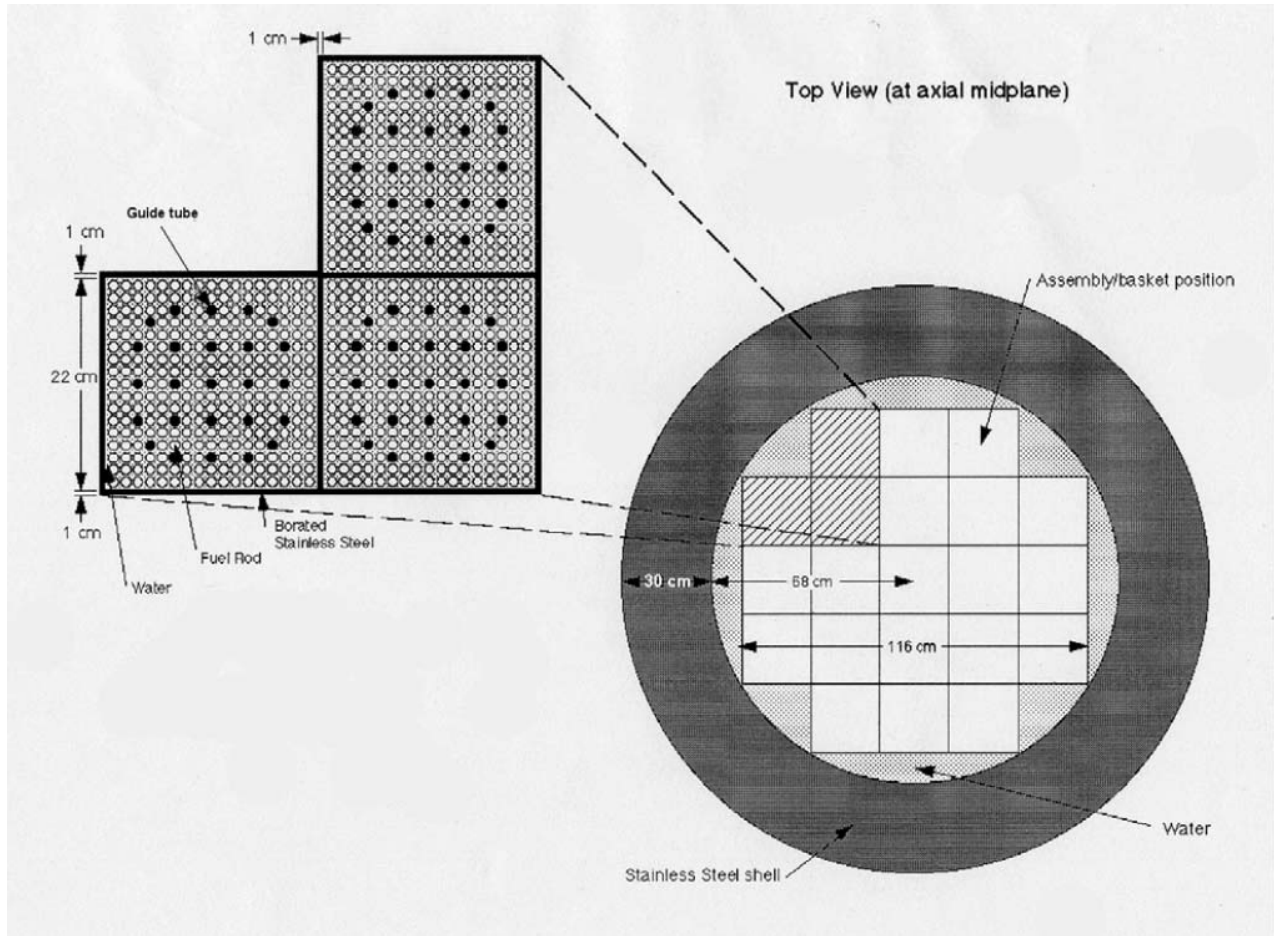
Assuming this benchmark proposal is finalized and approved by Dec. 2008

June 2009	Participants provide results to benchmark coordinator
September 2009	Distribution of draft benchmark report
December 2009	All comments on draft report received by benchmark coordinator
April 2010	Final draft of benchmark report for Nuclear Science Committee

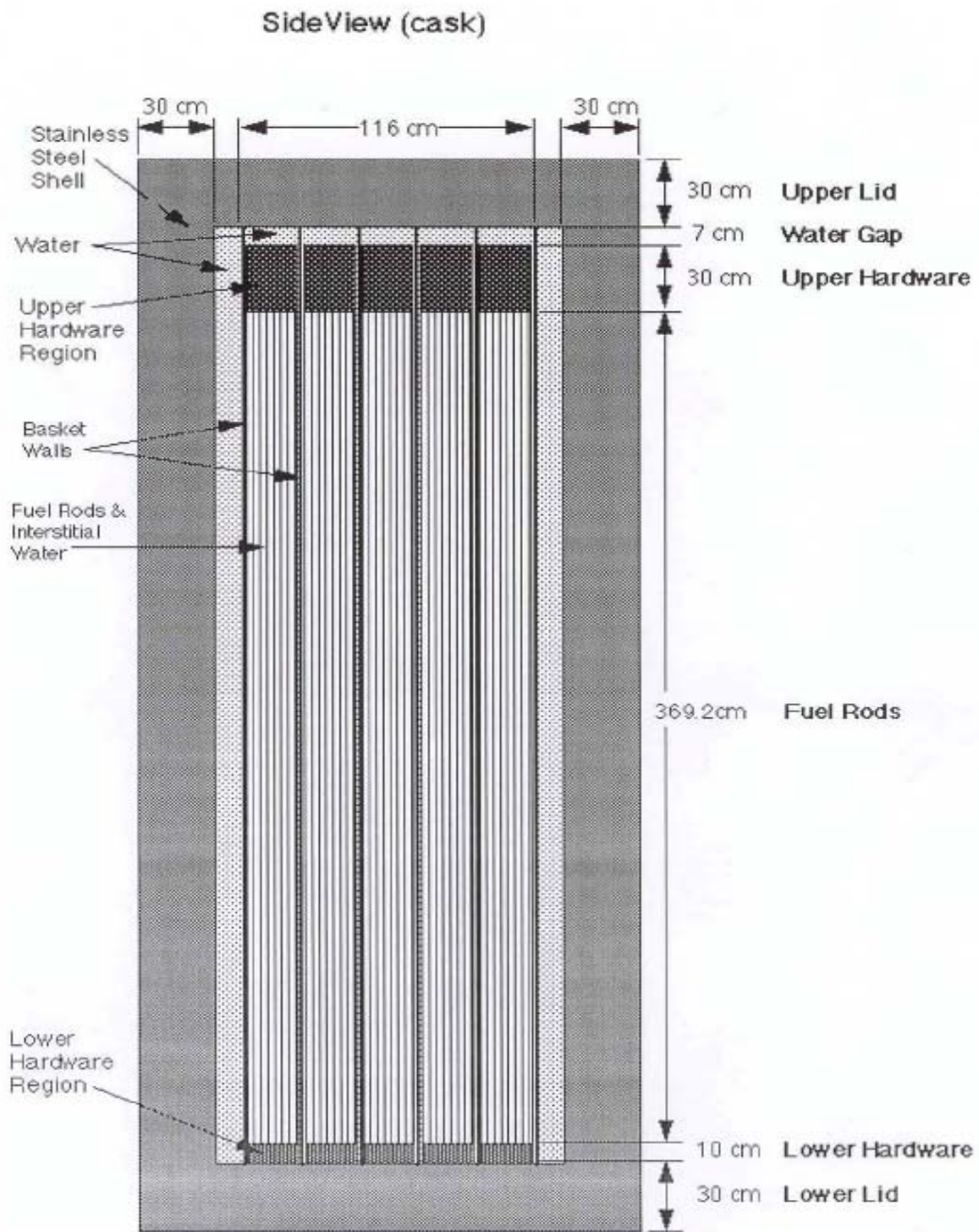
# ATTACHMENT I

## CASK MODEL CROSS-SECTION VIEWS

Figures 4 and 5 show top and side views of the cask model. A vertical cross-section through the basket compartment illustrated in Figure 6 shows the fuel rod and assembly geometry regions, including active fuel, rod endplugs, and assembly upper and lower hardware.

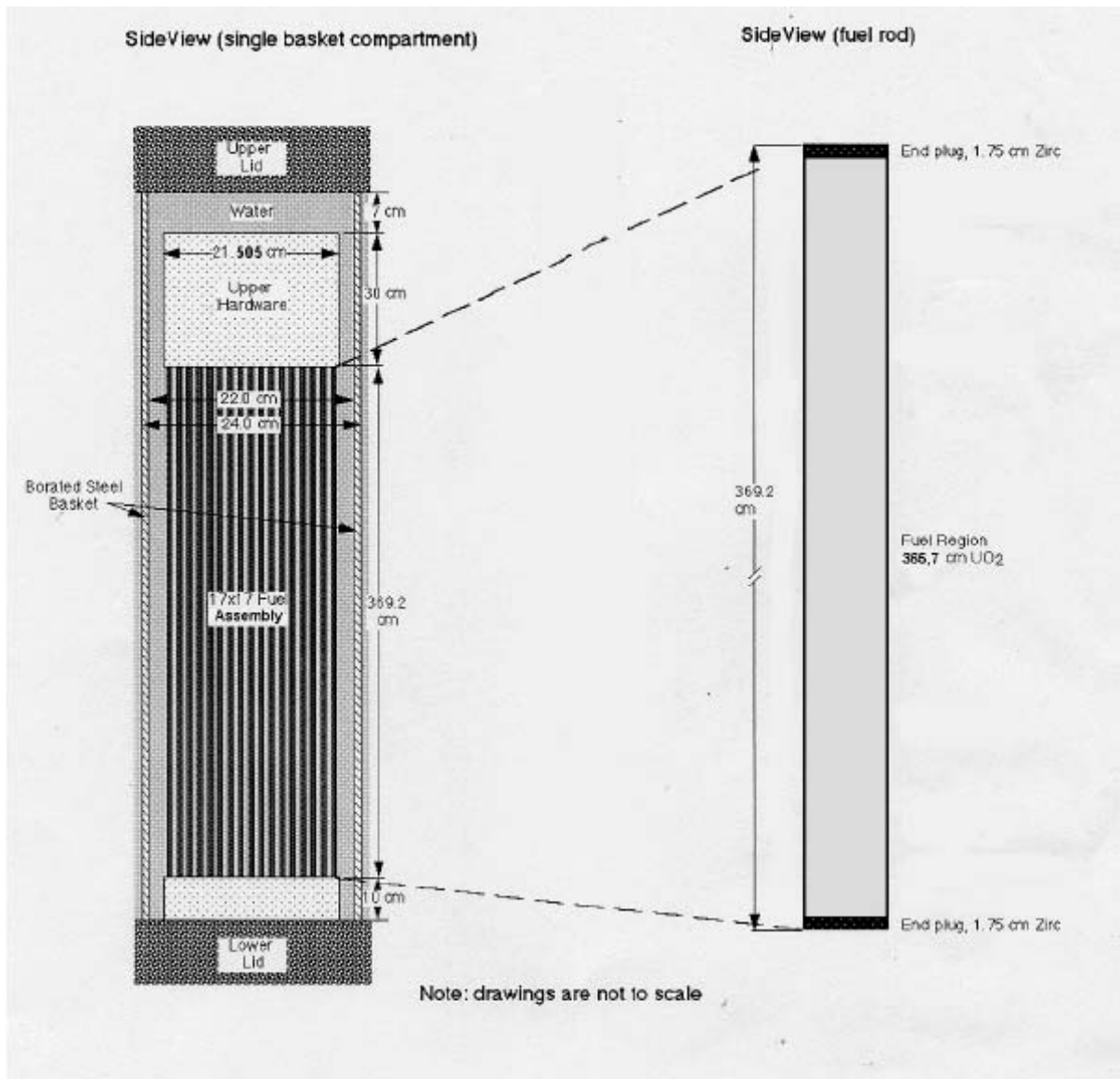


**Figure 4 : Cask model (top view)**



Note: drawing is not to scale

**Figure 5 : Cask model (side view)**



**Figure 6 : Single basket compartment**