BURNUP CREDIT CRITICALITY EVALUATION OF THE ENSA ENUN 32P CASK

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This paper presents the criticality evaluation of the ENUN 32P cask which takes credit for the decrease in reactivity due to the fuel burnup. Burnup credit allows the usage of high capacity casks for storage and transportation. Sub-criticality is maintained in the ENUN 32P by a combination of neutron absorber and actinide only burnup credit. Bounding actinide isotopic inventories are computed with SAS2H, and bounding MCNP5 Monte Carlo criticality calculations are performed to generate initial enrichment versus fuel burnup loading curves for the ENUN 32P cask. An evaluation of uncredited margin due to fission product is presented.

I. INTRODUCTION

This paper presents the criticality evaluation of the ENSA ENUN 32P (ENUN 32P) cask which takes credit for the decrease in reactivity due to the fuel burnup. This approach is called burnup credit. Burnup credit allows the usage of high capacity casks for storage and transportation. Because of the high capacity, fewer casks are required for storage and transportation resulting in both economic and safety benefits.

The ENUN 32P is a metal storage/transport cask that can accommodate 32 PWR type spent fuel assemblies. The ENUN 32P cask comprises a solid steel cask body enclosed by inner and outer lids. The body is SA-350 low alloy carbon steel. The side of the body is surrounded by solid neutron shielding

The cask cavity contains an egg crate type basket which holds 32 PWR spent fuel assemblies. The basket comprises a steel egg crate structure with neutron absorbing tubes in each cell location. The cask can store or transport either KWU 16x16 or Westinghouse 17x17 fuel. Sub-criticality is maintained in the ENUN 32P by a combination of neutron absorber and actinide only burnup credit. In actinide only burnup credit, only the depletion and buildup of the uranium and plutonium isotopes is allowed in the evaluation. However, in this evaluation, subset of fission products are evaluated for uncredited margin.

II. METHODOLOGY

The actinide only burnup methodology for the ENUN 32P cask comprises the following tasks

1. Determination of bounding fuel depletion parameters
2. Depletion evaluations as a function of fuel assembly initial enrichment and discharge burnup
3. Determination of bounding axial burnup profiles for various burnup ranges
4. Criticality evaluations of the cask as a function of fuel assembly initial enrichment and discharge burnup as well as basket mechanical tolerances and moderator density

The methodology follows the guidance of U.S.NRC Interim Staff Guidance and ANSI/ANS-8.27-2008, Burnup Credit for LWR Fuel [1,2]. The depletion evaluations of the fuel assemblies are performed with the SAS2H sequence of the SCALE4.4a code system [3,4]. The 44 group ENDF/B-V neutron cross section library is utilized in all depletion calculations. The reactivity performance of this methodology has been validated against Radiochemical Assay (RCA) data from various LWRs [5]. Criticality evaluations are performed with the MCNP5 Monte Carlo code and the ENDF/B-VI neutron cross section library [8]. The reactivity performance of this methodology has been validated against Laboratory Critical Experiments (LCE) of LWR type fuel [9]. The actinides that are used in the burnup credit evaluations are shown in Table 1 along with the fission products used to determine uncredited margin.

II.A. Bounding Fuel Depletion Parameters

In order to conservatively maximize the actinide concentrations produced during fuel depletion, bounding fuel depletion parameters must be selected. The key depletion parameters are summarized in Table 2 along with their typical ranges and bounding values. The parameters include: fuel temperature, moderator temperature, moderator density, soluble boron concentration, specific power, number of operating cycles, down time between cycles, discharge cooling time, burnable poison presence and control rod presence. Parameters were chosen to produce a conservative actinide inventory typically by hardening the neutron spectrum and thus increasing the actinide production for a given burnup.

Table 1 - Actinides and Fission Products Used in the Burnup Credit Evaluations

Actinides	Fission Products
U-234	Tc-99
U-235	Cs-133
U-236	Cs-135
U-238	Nd-143
Pu-238	Nd-145
Pu-239	Sm-147
Pu-240	Sm-149
Pu-241	Sm-150
Pu-242	Sm-151
Am-241	Sm-152
	Eu-153
	Gd-155

Table 2 - Bounding PWR Depletion Parameters

Parameter	Typical Range	Value Chosen	Comment
T-Fuel (°C) at Max Power	625-800	920	Higher is conservative
Max. T-Moderator (°C)	300-328	330	Higher is conservative
P-Moderator (g/cm^3)	0.725-0.650	0.65	Lower is conservative
Boron (ppm)	0-1200	1000 const.	Higher is conservative
Specific Power (MW/assembly)	17-19	30	Higher is conservative
Cycles	3-4	1	Continuous is conservative
Days/Cycle	548-730	Derived based on burnup	Derived to match burnup at the chosen power density
Days Shutdown Between Cycles	21-30	0	None is conservative
Cool Time (y)	3-8	3.5	Shorter is conservative
Burnable Absorbers	Gad	KWU: none	None is conservative
	WABA	W: 24 WABA	Depletion with, is conservative
	IFBA	none	None is conservative
Control Rod Presence	Out of Core	None	

II.B. Depletion Evaluations

The depletion evaluations of the fuel assemblies are performed using the SAS2H sequence of the SCALE4.4a code system [3,4]. The 44 group ENDF/B-V neutron cross section library is utilized in all depletion calculations. The SAS2H depletion calculations span a range of initial enrichments and discharge burnups to cover the expected inventory of spent nuclear fuel. The specific enrichment and burnup combinations analyzed are as follows:

Initial Enrichments: 2.25, 2.75, 3.25, 3.75, 4.0, 4.25, 4.5, 4.75, 5.0 w/o U-235
Burnups: 15, 20, 25, 30, 35, 40, 45, 50 GWD/MTU

The isotopic values calculated for the pertinent actinide isotopes [Table 1] at a desired initial enrichment and burnup are extracted from the SAS2H output and reformatted into MCNP5 material input cards. These material input cards are assigned to the associated axial zones for fuel of a given total burnup in the MCNP criticality model.

II.C. Bounding Axial Burnup Profiles

Bounding axial profiles are developed from detailed discharge burnup profiles. These bounding profiles are intended to under predict the amount of burnup in the ends of the fuel while normalizing the curve by over predicting the burnup in the central region of the fuel assembly. A 7-node structure is utilized as in the criticality evaluations. This approach is based upon the work presented in Reference 5. Section 5.2.1 of Reference 5 presents a detailed justification of the 7-node approach. Simplification of the axial nodes is essential to efficiently completing the computational work, as the development of a 20-node model would take significantly longer than that of the 7-node model.

The bounding profiles for Westinghouse fuel are shown in Figure 1 and Figure 2 for fuel with burnup < 25 GWd/MtU and with burnup > 25 GWd/MtU, respectively. As can be seen in the graphs, the 3 nodes at each end of the fuel assembly are fully bounded by the axial profile developed from Westinghouse burnup data. This ensures that the ends are under burned and thus maximizes the reactivity of the ends of the active fuel zone. In addition to the bounding profile, axially uniform burnup is evaluated, and the more reactive profile between the two is utilized in the burnup loading curves.

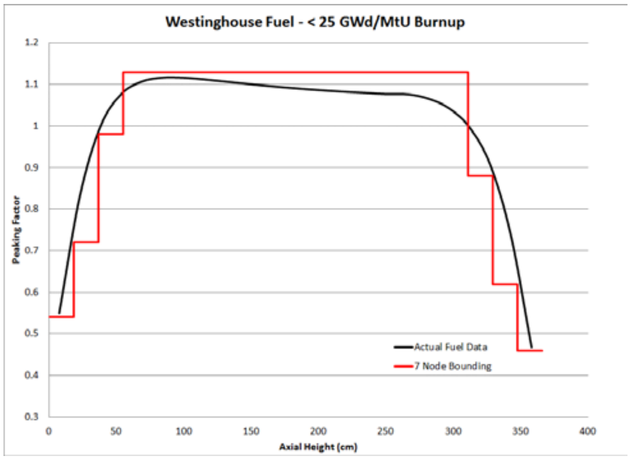

Figure 1 - Westinghouse Fuel Axial Profile – <25 GWd/MtU Burnup

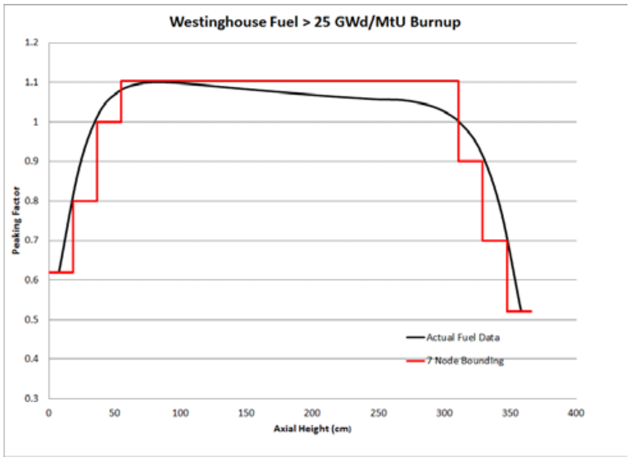

Figure 2 - Westinghouse Fuel Axial Profile – >25 GWd/MtU Burnup

II.D. Criticality Evaluations

Criticality evaluations are performed with the MCNP5 Monte Carlo code and the ENDF/B-VI neutron cross section library [8]. The MCNP5 models represent the key dimensional details of the fuel assembly and the cask, while simplifying some areas, such as the trunnions and lid closure details which are not important to the criticality evaluation.

A cross-sectional view of the cask through the fuel region is shown in Figure 3 and an X-Z plot through the fuel zones is shown in Figure 4. Together, these figures depict the general configuration of the cask body, basket, and fuel assemblies. Figure 3 illustrates the details in the fuel basket region of the model. In this model, the spacers surrounding the fuel basket structure can be seen, in addition to the stainless steel structural members and neutron absorber plates in the basket. Criticality calculations are performed using the depleted fuel data developed for each fuel type, i.e KWU 16x16-20 and Westinghouse 17x17.

Criticality computations were performed as a function of initial enrichment and burnup. These results are utilized to determine the minimum burnup required for a given initial enrichment in order to meet the criticality safety margins. In addition, sensitivity studies were performed on the critical dimensional characteristics of the spent fuel cask design. These included: the dimensions of the structural support members of the fuel basket, relative positioning of the fuel assemblies, variations in neutron poison material and moderator density variation throughout the cask cavity as well as the pellet-clad gap. The results of the sensitivity studies provided the appropriate reactivity uncertainty due to mechanical tolerances. In addition to the standard burnup credit evaluation, a misloading evaluation and an evaluation of uncredited safety margin due to fission products were performed pursuant to ISG-8, Rev. 3[1]

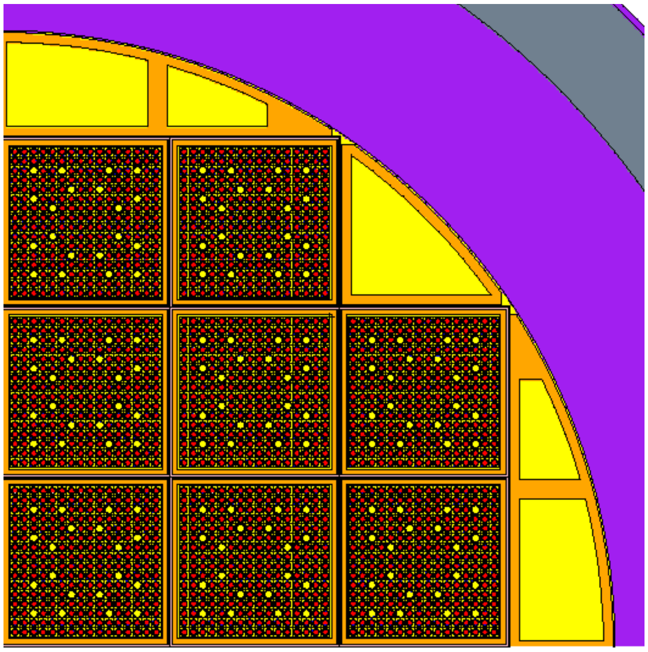

Figure 3 - MCNP Model of the ENUN 32P Cask - Quarter-Section Basket – X-Y Plot

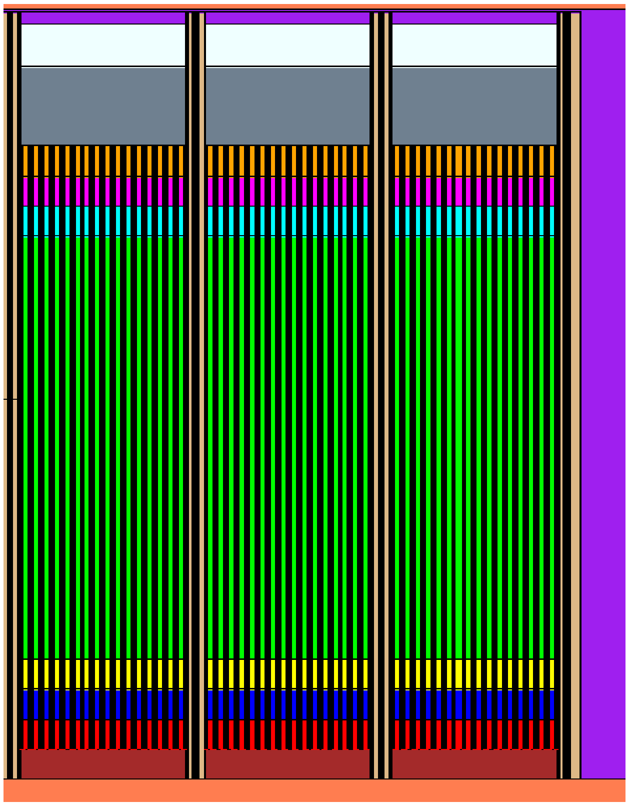

Figure 4 - MCNP Model of the ENUN 32P Cask - X-Z Plot

III. RESULTS AND DEVELOPMENT OF LOADING CURVES

Base case calculation are performed for each fuel type at the 9 enrichment and 8 burnup values specified previously. For each fuel type, a total of 288 base case calculations are performed. The 72 variations in enrichment and burnup are evaluated for average or zoned axial profiles and for actinides or actinides plus fission products (for uncredited margin). The results of these calculations for the KWU 16x16 fuel are shown in Table 3 for actinide only credit.

Table 3 - Base Case Results – KWU 16x16 Fuel – Actinide Only

Enr	Burnup							
	15000	20000	25000	30000	35000	40000	45000	50000
2.25	0.8586	0.8370	0.8177	0.8019	0.7877	0.7777	0.7674	0.7593
2.75	0.9055	0.8784	0.8595	0.8395	0.8204	0.8067	0.7929	0.7796
3.25	0.9440	0.9205	0.8975	0.8756	0.8546	0.8368	0.8198	0.8072
3.75	0.9780	0.9570	0.9332	0.9121	0.8889	0.8701	0.8507	0.8341
4.00	0.9945	0.9705	0.9488	0.9256	0.9074	0.8851	0.8647	0.8489
4.25	1.0112	0.9859	0.9624	0.9419	0.9213	0.9015	0.8813	0.8633
4.50	1.0215	0.9996	0.9782	0.9558	0.9352	0.9149	0.8955	0.8752
4.75	1.0360	1.0125	0.9904	0.9724	0.9486	0.9306	0.9081	0.8905
5.00	1.0486	1.0256	1.0033	0.9835	0.9619	0.9427	0.9216	0.9050

(K+2σ, average σ=0.0008)

Sensitivity studies were performed for a single representative set of initial enrichment and burnup for each fuel type. The enrichment and burnup selected for each fuel type was based upon the estimated burnup value that the 5.0 w/o enriched fuel assembly was expected require in order to be under the upper safety limit. The burnup was selected as 45 GWD/MTU, the approximate burnup expected to be needed for 5.0 w/o fuel to be acceptable for loading. The cases were evaluated for the actinide only nuclides and were evaluated for both average and zoned axial profiles. The variations in basket manufacturing tolerances are evaluated to ensure that any potential positive reactivity effects are included in the determination of the upper safety limit for k_{eff}. These perturbations in basket configurations are minimum stainless steel plate thickness, minimum neutron absorber plate thickness, shifting the basket components and fuel toward the center of the basket and shifting the basket components and fuel away from the center of the basket. The results of these calculations for the KWU 16x16 fuel are shown in Table 4. Additional studies of moderator density variation in the cask cavity and the pellet/clad gap showed peak reactivity between 0.9 and 1.0 g/cc.

Table 4 - Mechanical Perturbations - KWU 16x16 Fuel – 5 w/o U-235 / 45 GWD/MTU Burnup

Case	Axial Profile	Δk
minSS	avg	-3.0E-05
minSS	zone	0.00204
minMMC	avg	-0.00094
minMMC	zone	-0.00108
shiftIN	avg	0.00949
shiftIN	zone	0.00918
shiftOUT	avg	-0.00566
shiftOUT	zone	-0.00553

The results of the base case analyses are compared to the upper safety limits (USL) for each fuel type, and the amount of burnup needed for a given initial enrichment to meet the upper safety limit is determined. The upper safety limit is determined in accordance with [10]. This USL value is then utilized along with the base case results to determine the minimum burnup needed for each initial enrichment value to meet the safe loading criteria. The USL must take into account the biases and uncertainties introduced during the calculation of burnup credit isotopics and calculation of the k_{eff} of the system itself. As such, the bias and uncertainty determined in [7] for the SAS2H depletion calculations and in [9] for the MCNP calculations using burnup credit isotopics are utilized in the USL calculation. These bias values are listed in Table 5 and represent the underprediction in k_{eff} produced by each code when comparing calculated results to measured values.

Additional sources of bias are taken from the sensitivity studies. The results presented in these sections identify the worst case reactivity increase produced by each perturbation. The biases from these calculations are summarized in Table 6 for each fuel type. In this table, the perturbations that did not produce a positive reactivity increase are listed as 0.0, with the other perturbations listed with the worst case reactivity increase over the base case for each fuel type.

The sensitivity bias terms are additively combined with the code bias terms. This approach reflects the fact that these maximum reactivity changes are each essentially the worst-case biases, and it is conservative to simply combine each bias term by addition. Thus, the calculation of the USL value is as follows:

$$USL = 1.0 - 0.05 - \Delta k_{minSS} - \Delta k_{shiftIN} - \Delta k_{gapH2O} - \sqrt{{U_{SAS2H}}^2 + {U_{MCNP}}^2}$$

Where,

0.05 – Administrative Margin of 5%
Δk_{minSS} – Minimum SS Basket Plate Thickness
$\Delta k_{shiftIN}$ – Maximum Shift-In Perturbation Reactivity Increase
Δk_{gapH2O} – Maximum Fuel-Clad Gap H_2O Perturbation Reactivity Increase
U_{SAS2H} – SAS2H Bias and Uncertainty
U_{MCNP} – MCNP Bias and Uncertainty

The upper safety limit for each fuel type is determined using the equation and the bias and uncertainty terms listed in Table 5 and Table 6. The resulting upper safety limits for each fuel type are listed in Table 7. Using the upper safety limit values given in Table 7, the base case results presented in Table 3 for KWU 16x16 fuel are then evaluated for each initial enrichment value to determine the amount of burnup needed to meet the USL. The result is the loading curve for KWU 16x16 fuel as shown in Figure 5. Similar evaluation for Westinghouse 17x17 fuel results in the load curve presented in Figure 6. The loading curves represent for a given initial enrichment level, the minimum burnup the fuel requires before it can be safely stored or transported.

Table 5 - Calculational Bias Coefficients

Code	Bias
SAS2H	0.0182
MCNP	0.0127

Table 6 - Sensitivity Bias Coefficients

Perturbation	KWU Bias	West. Bias
Min. SS	0.00204	0.0
Min. MMC	0.0	0.0
Shift Out	0.0	0.0
Shift In	0.00949	0.01337
Gap H_2O	0.00796	0.00806
Mod. Density	0.0	0.0
Single Package	0.0	0.0

Table 7 Upper Safety Limits

Fuel Type	USL
KWU 16x16	0.9083
West. 17x17	0.9064

As shown in both figures, the minimum burnup required is 15,000 MWD/MTU for enrichments below 2.75 w/o U-235. As the initial enrichment increases above 2.75 w/o U-235, burnups of up to approximately 48,000 MWD/MTU for 5 w/o U-235 fuel are required in order to be acceptable for loading in the ENUN 32P cask.

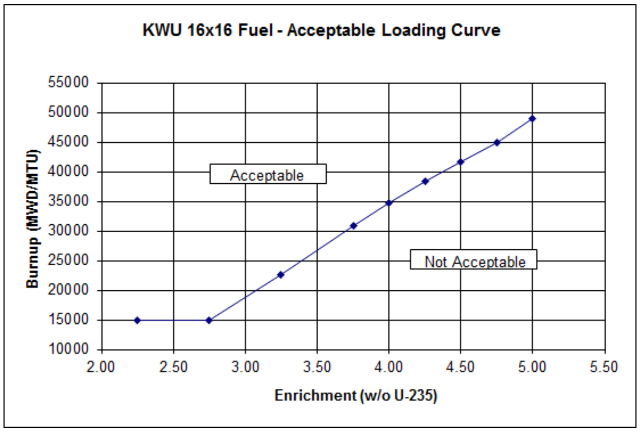

Figure 5 – ENUN 32P Loading Curve for Westinghouse 17x17 Fuel

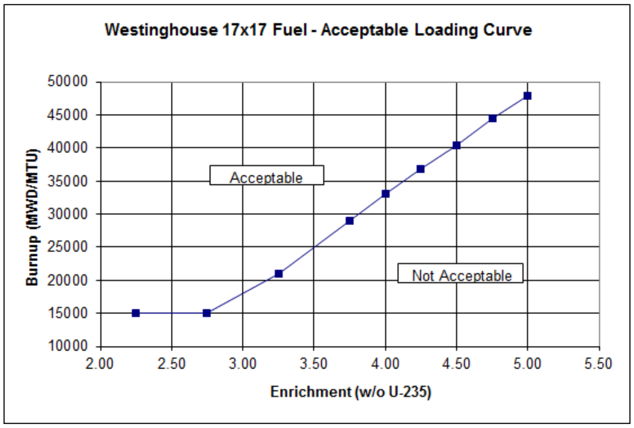

Figure 6 – ENUN 32P Loading Curve for Westinghouse 17x17 Fuel

As specified in Recommendation 6 of ISG-8 [1], an estimate of additional reactivity margin present in the burned fuel due to the presence of fission products is also made for the base case analysis. Comparisons of the loading curves for actinide only and actinide plus fission product cases are shown in Figure 7 for the KWU 16x16 fuel and in Figure 8 for the Westinghouse 17x17 fuel. As shown in these figures, the actinide only fuel cases require more than 10 GWD/MTU additional burnup at the 5.0 w/o initial enrichment level. This represents more than 30% additional burnup over the actinide plus fission product cases from initial enrichment ranges of 3.5 w/o to 5.0 w/o U-235.

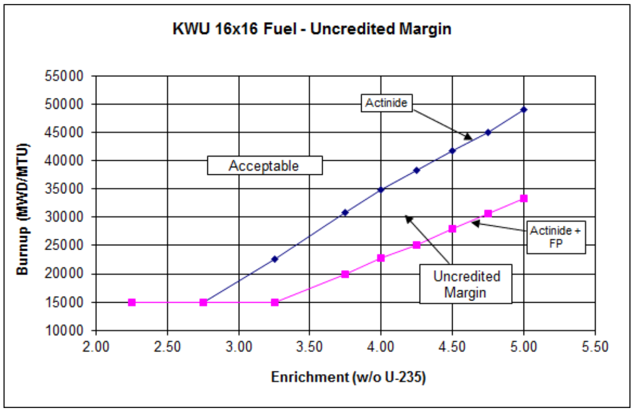

Figure 7 - Acceptable Loading Curve – KWU 16x16 Fuel – Actinide + Fission Products

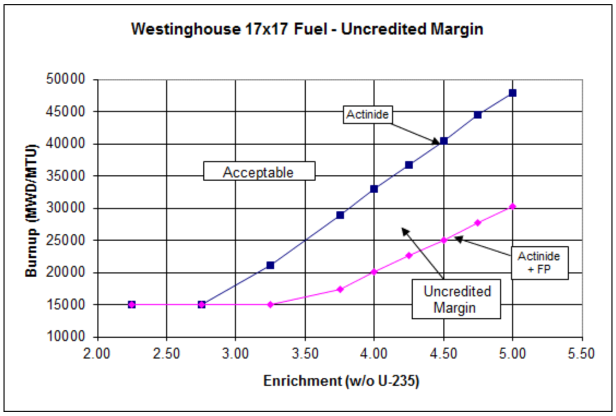

Figure 8 - Acceptable Loading Curve – Westinghouse 17x17 Fuel – Actinide + Fission Products

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