



August 13, 2026

NUREG/CR-7303 (2023)

**Validating Actinides and Fission Products for
Burnup Credit Criticality Safety Analyses –
Nuclide Compositions Prediction with
Extended Validation Basis**

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NRC RES/NMSS Hybrid Technical Workshop on Fuel Cycle
Impacts of High Burnup, Extended Enrichment, and HALEU
Fuel Designs, Featuring an Overview of ORNL's Capabilities
August 11–13, 2026



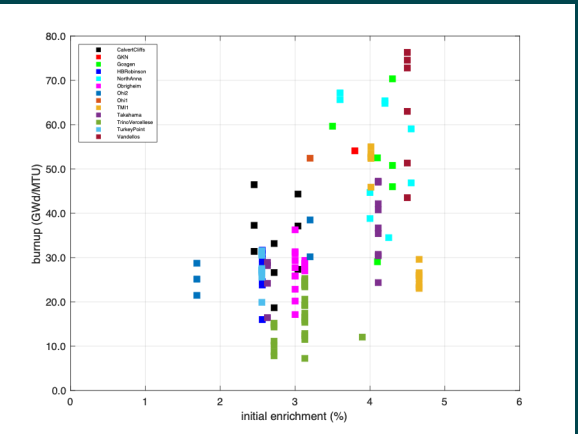
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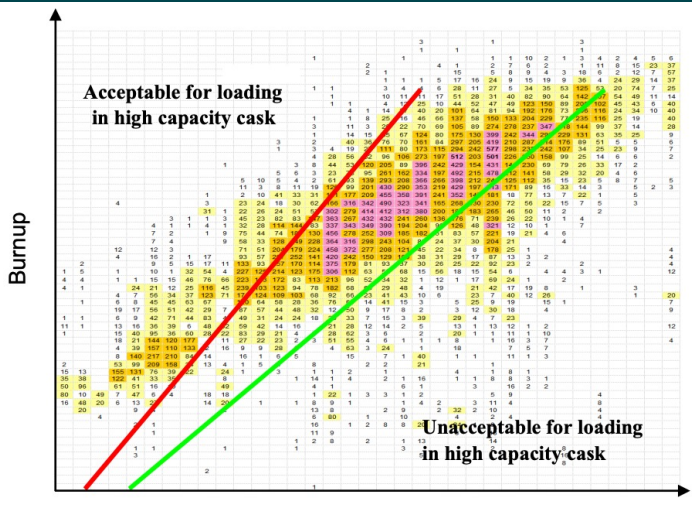


Goal: Estimate bias and uncertainty in k_{eff} for a criticality safety model resulting from bias and uncertainty in calculated nuclide concentrations

Demonstrate approach using an extended validation basis of radiochemical assay (RCA) measurement data for important actinides and fission products in PWR spent nuclear fuel



Improve the technical basis to support future updates of current NRC guidance for criticality safety analyses with burnup credit of PWR spent fuel for transportation and storage casks, the Interim Staff Guidance (ISG)-8, Rev 3 (2012)



Terminology

k_{eff} bias

Average error in calculated k_{eff} of the considered criticality safety model, which results from bias and bias uncertainty in calculated nuclide concentrations in spent fuel

k_{eff} bias uncertainty

Variance of the error in calculated k_{eff} for the considered criticality safety model, at a 95%/95% probability/confidence level, which results from bias and bias uncertainty in calculated nuclide concentrations

Depletion code

In this context, it refers to the computational tool and associated nuclear data used to simulate the nuclide transmutations in fuel during irradiation and decay

What is new in NUREG/CR-7303 vs. NUREG/CR-7108 ?

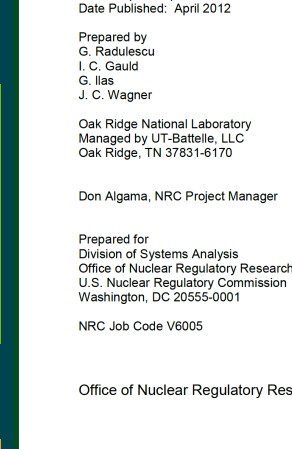
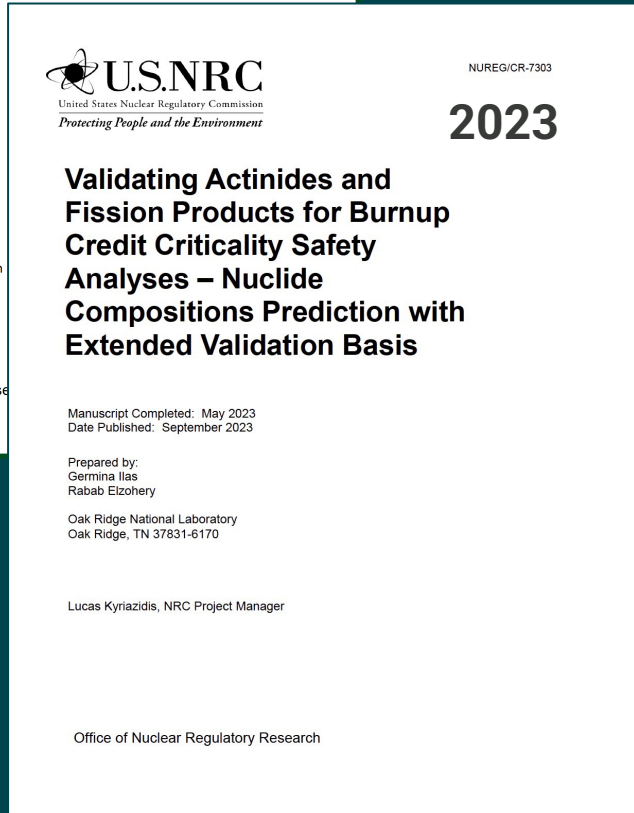
Significantly improved validation basis for PWR spent fuel:

129 measured fuel samples (vs. 100 samples)

Extended burnup coverage to 80 GWd/tU (vs. 60 GWd/tU)

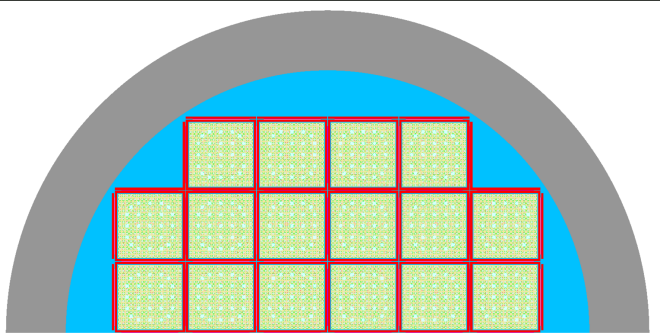
Updated loading curves, derived using new methodology using ORIGAMI fast depletion computational tool in SCALE 6.2

New implementation of the stochastic sampling method for k_{eff} bias and uncertainty determination, using the Sampler uncertainty quantification tool in SCALE 6.2

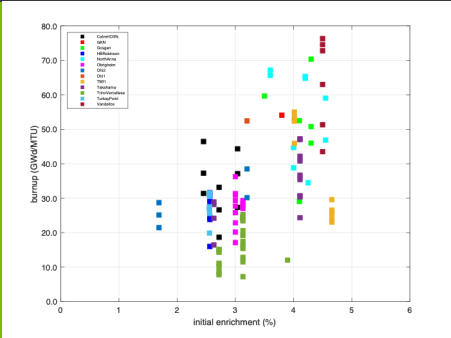


NUREG/CR-7303: summary of approach and outcomes

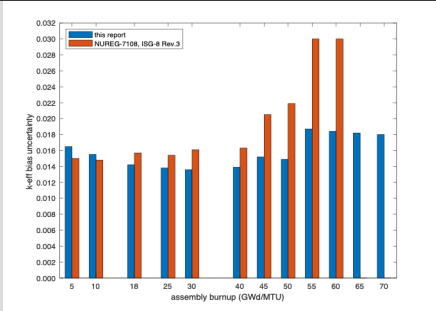
Models
and
methods



Nuclide
inventory
validation
basis



Results and
significance



NUREG/CR-7303

Validating Actinides and Fission Products for Burnup Credit Criticality Safety Analyses – Nuclide Compositions Prediction with Extended Validation Basis

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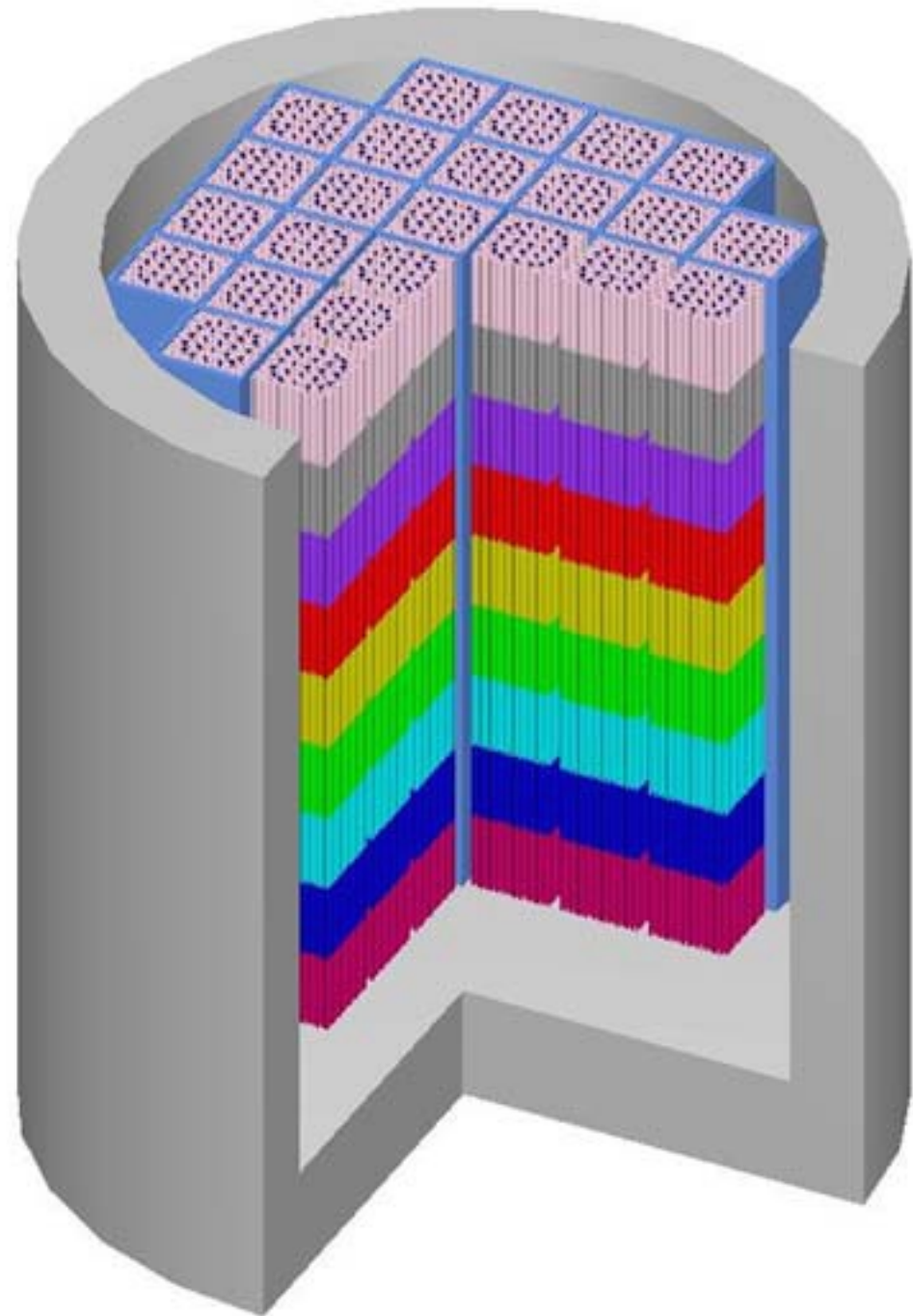
Oak Ridge National Laboratory
Oak Ridge, TN 37831-6170

Lucas Kyriazidis, NRC Project Manager

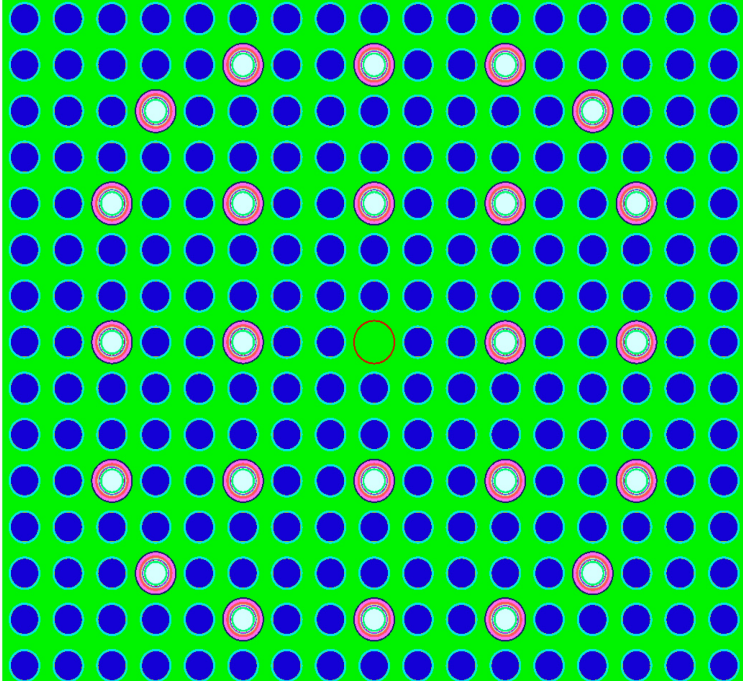
Office of Nuclear Regulatory Research

Models and methods

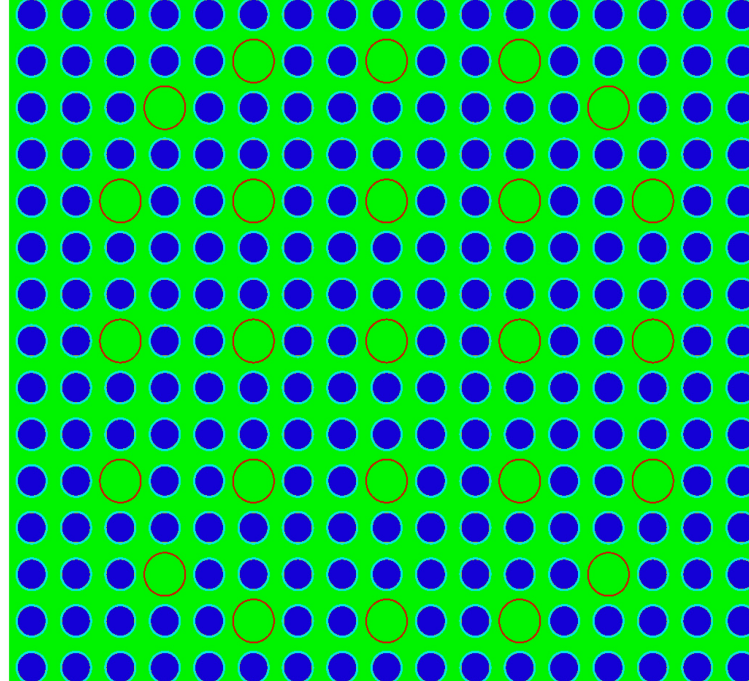
**GBC-32 cask is the
criticality safety model
used as basis for
assessment**



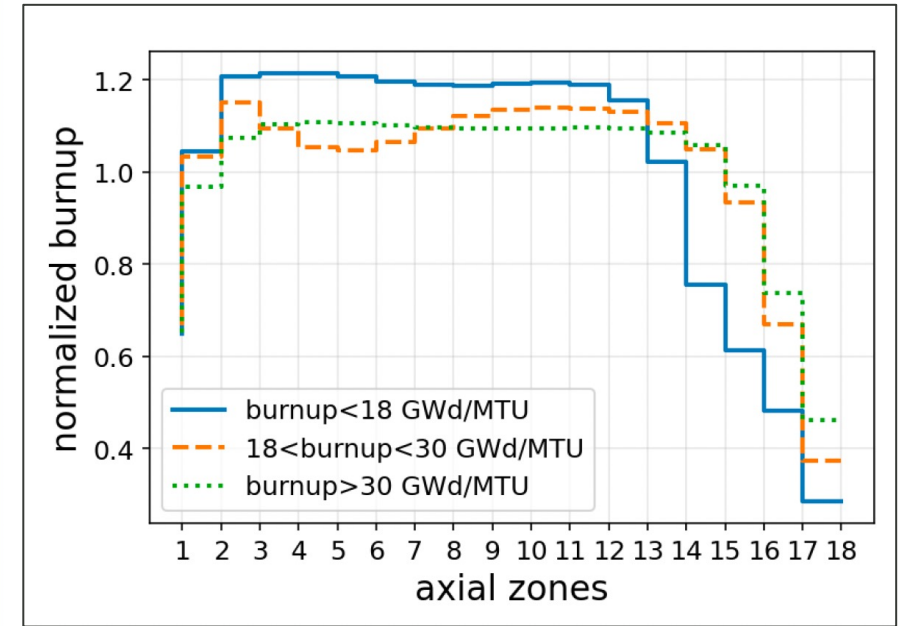
Fuel assembly models and other modeling data are consistent with previous NUREGs assessments



Assembly model (with WABA)
for depletion calculations



Assembly model (no WABA) for
cask criticality calculations



Assembly axial burnup profiles

Ref: NUREG/CR-6801 (2003)

Computational tools and associated nuclear data: SCALE 6.2.4

Criticality simulations

3D Monte Carlo criticality code **KENO-VI** under the **CSAS6** criticality safety analysis sequence

Fuel depletion and decay simulations

TRITON
ORIGAMI
ORIGEN

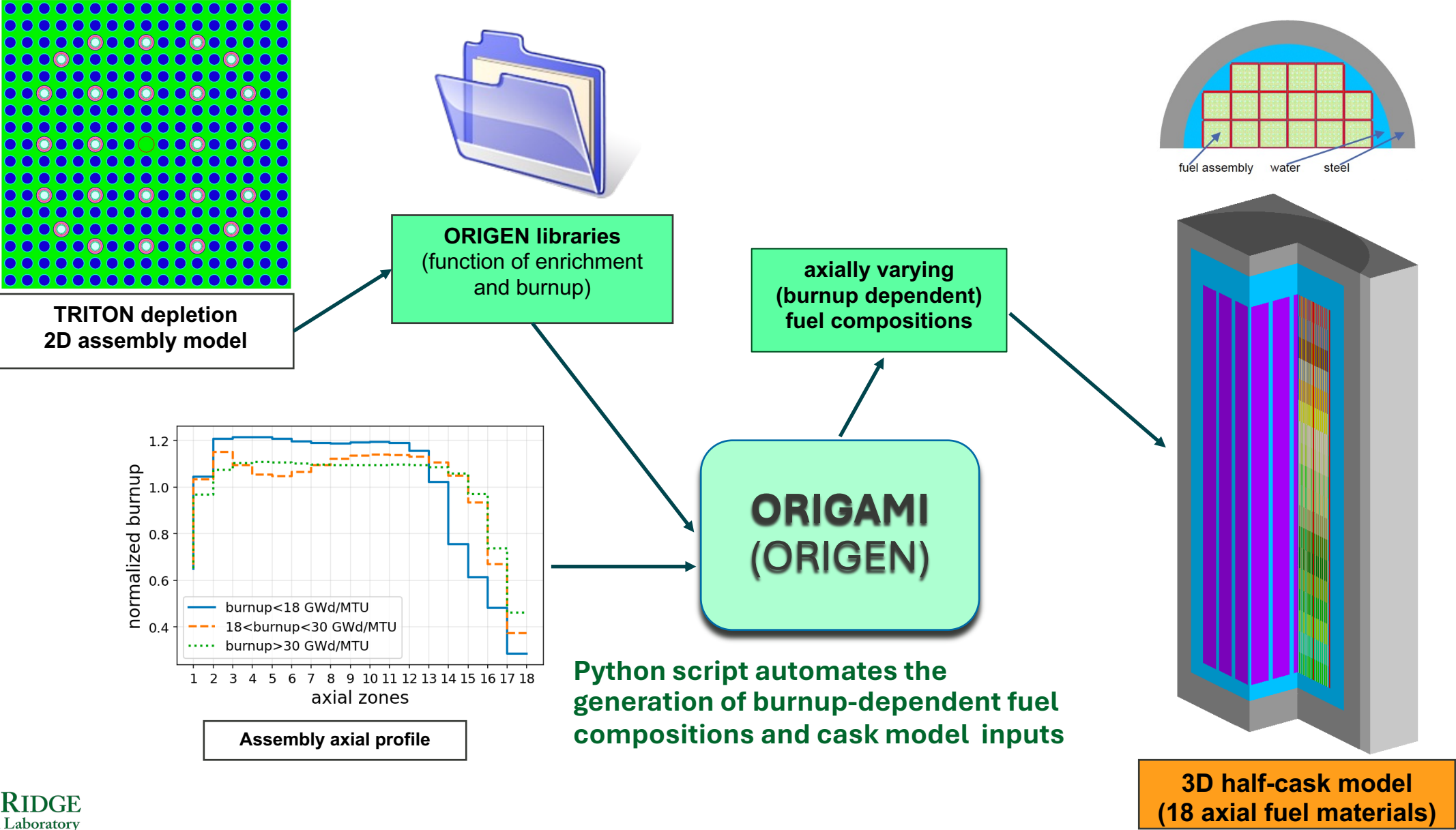
Uncertainty propagation

Sampler uncertainty quantification tool
1,000 depletion runs for each assembly average burnup value

Nuclear data libraries and other parameters

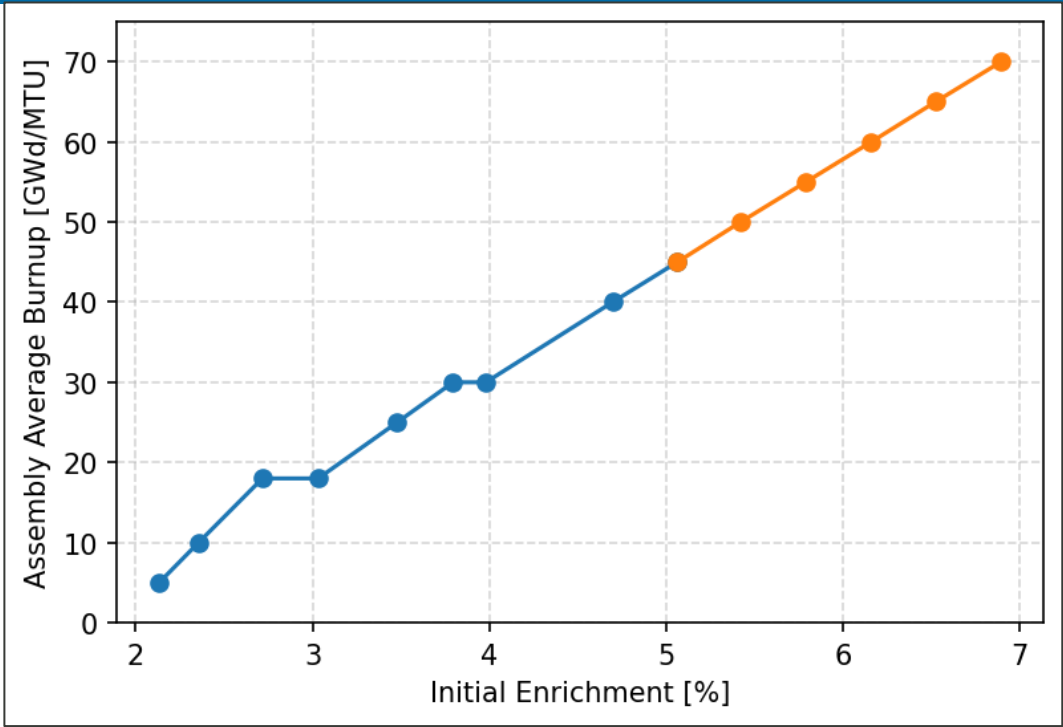
252-gr ENDF/B-VII.1
All fuel compositions for criticality model at 5-year cooling time

How we calculated axially-dependent fuel compositions at different burnups for the reference cask criticality model



New loading curve was generated for the reference cask model to get consistent basis $k_{eff}=0.94$ for assessment

Loading curve for the GBC-32 cask model, SCALE 6.2.4, CSAS6, 252-gr ENDB/VII.1 libs, extended to higher burnups



Comparison of loading curve enrichments NUREG/CR-7303 (new) vs NUREG/CR-7108 (old)

Assembly burnup (GWd/MTU)	Old enrichment (%)	New enrichment (%)
5	2.12	2.13
10	2.33	2.36
18	3.00	3.03
25	3.44	3.48
30	3.92	3.98
40	4.64	4.70
45	5.00	5.06

New methodology for loading curve generation used ORIGAMI and CSAS6 <https://www.ans.org/pubs/transactions/article-53400/>

Cask reference model: description of input and underlying fuel composition data generation

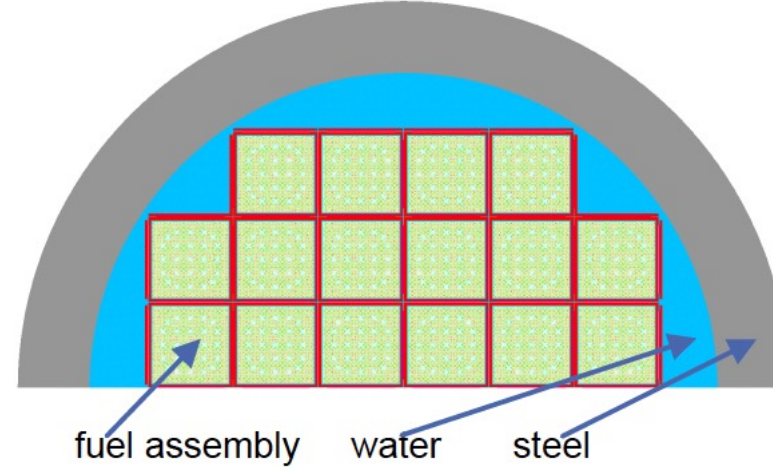
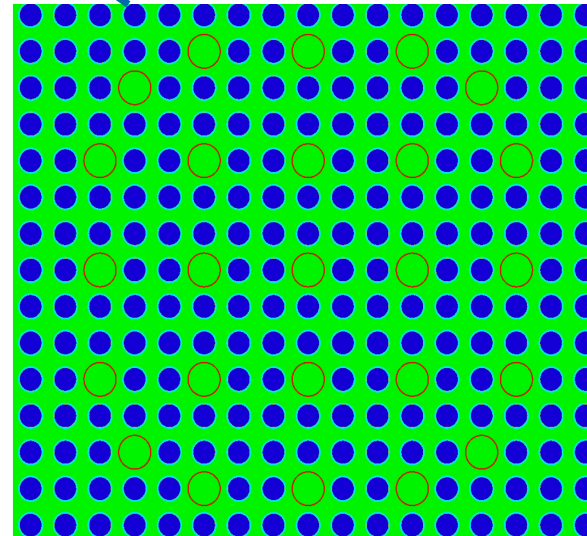
Demo follows here:

3D CSAS6 cask model/input

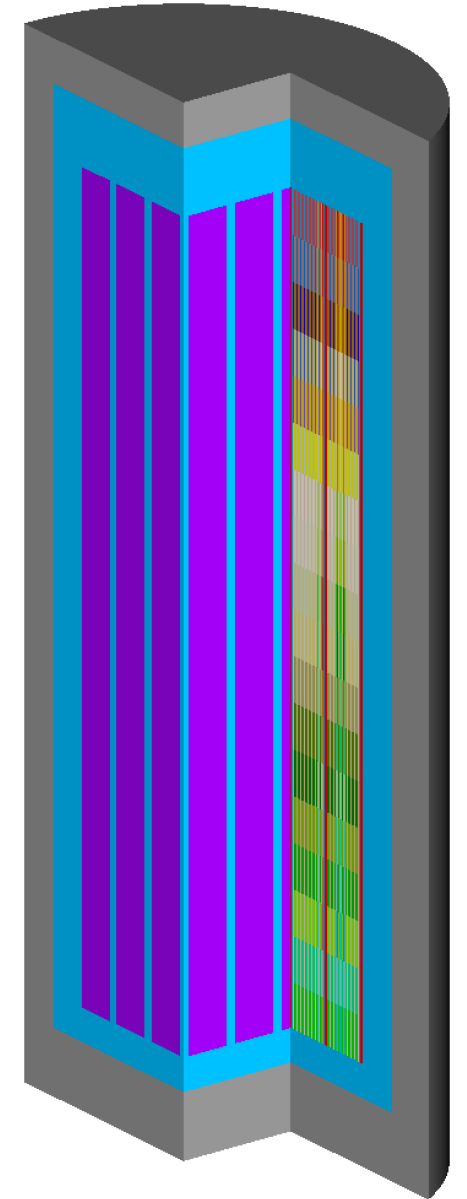
Origami model/input

TRITON model for burnup-dependent
cross section library generation

2D cross section of cask model at the axial midline

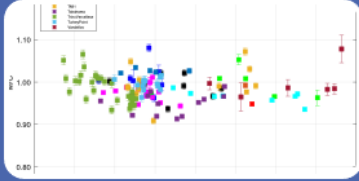


2D cross section of assembly



Cutaway view of 3D model with a quarter removed to show 18 axial fuel materials

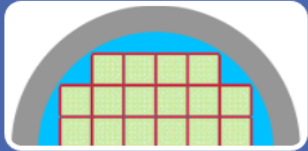
Stochastic sampling method to estimate the k_{eff} bias and bias uncertainty that are associated with calculated nuclide concentrations



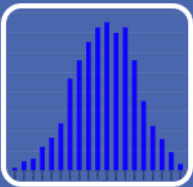
Establish bias and uncertainty in calculated nuclide concentrations based on comparison to measurement data

$$C_{i,Bu}^m = C_{i,Bu}(\bar{X}_{i,Bu} + \sigma_{i,Bu}R_i^m)$$

Randomly sample the calculated nuclide concentrations in the criticality model, resulting in M perturbed models



Calculate k_{eff} for each of the perturbed criticality models



Estimate k_{eff} bias and uncertainty based on resulting distribution of k_{eff} values

What are the relevant terms in stochastic uncertainty sampling (1)

Measured-to-calculated (M/C) nuclide concentration ratio

mean

$$\bar{X}_i = \frac{1}{N_i} \sum_{j=1}^{N_i} X_i^j = \frac{1}{N_i} \sum_{j=1}^{N_i} \frac{M_i^j}{C_i^j},$$

standard deviation s_i

$$s_i = \sqrt{\frac{\sum_{j=1}^{N_i} (X_i^j - \bar{X}_i)^2}{N_i - 1}}.$$

Expected concentration for a nuclide i in fuel material of burnup Bu

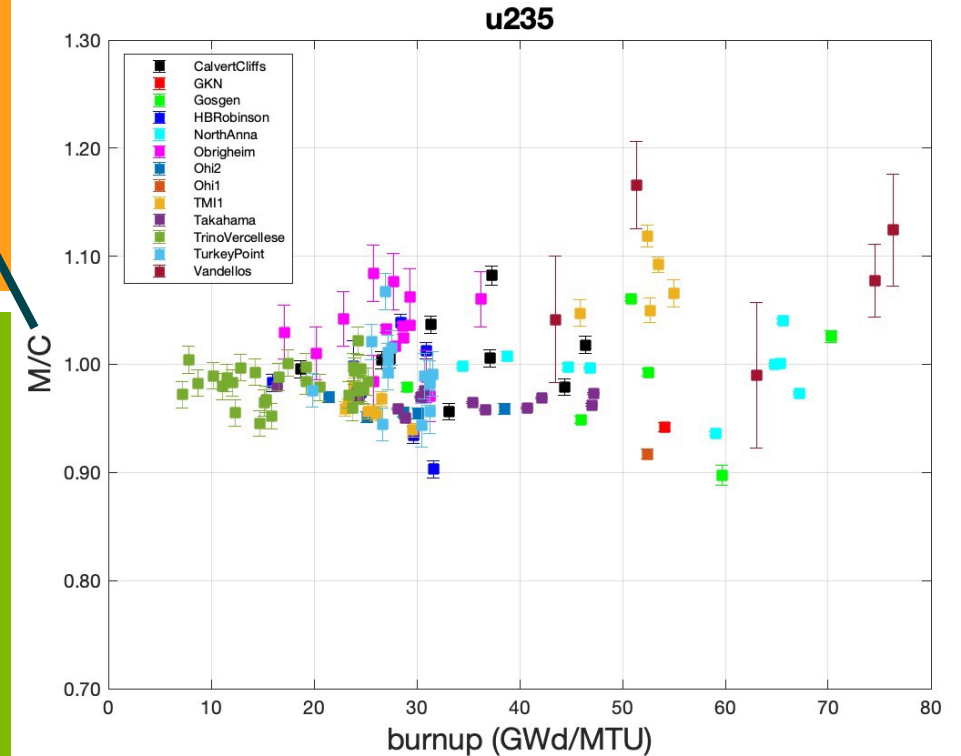
$$C_{i,Bu}^m = C_{i,Bu} (\bar{X}_{i,Bu} + \sigma_{i,Bu} R_i^m)$$

$$\sigma_i = \begin{cases} s_i \cdot tf_2^i & \text{if } N_i \geq 10 \\ s_i \cdot tf_1^i & \text{if } N_i < 10 \end{cases}$$

tf_2 = two-sided tolerance-limit factor for the normal distribution to bound 68.3% of the population at a 95% confidence level

tf_1 = one-sided tolerance limit factor for the normal distribution to bound 95% of the population at a 95% confidence level

R_i^m = random number selected from the standard normal distribution



What are the relevant terms in stochastic uncertainty sampling (2)

k_{eff} mean and standard deviation:

$$\bar{k}_{eff} = \frac{1}{M} \sum_{m=1}^M k_{eff}^m$$

$$\sigma_{k_{eff}} = \sqrt{\frac{\sum_{m=1}^M (k_{eff}^m - \bar{k}_{eff})^2}{M-1}}$$

M is number of the calculated k_{eff} values (number of perturbed cases in Sampler)

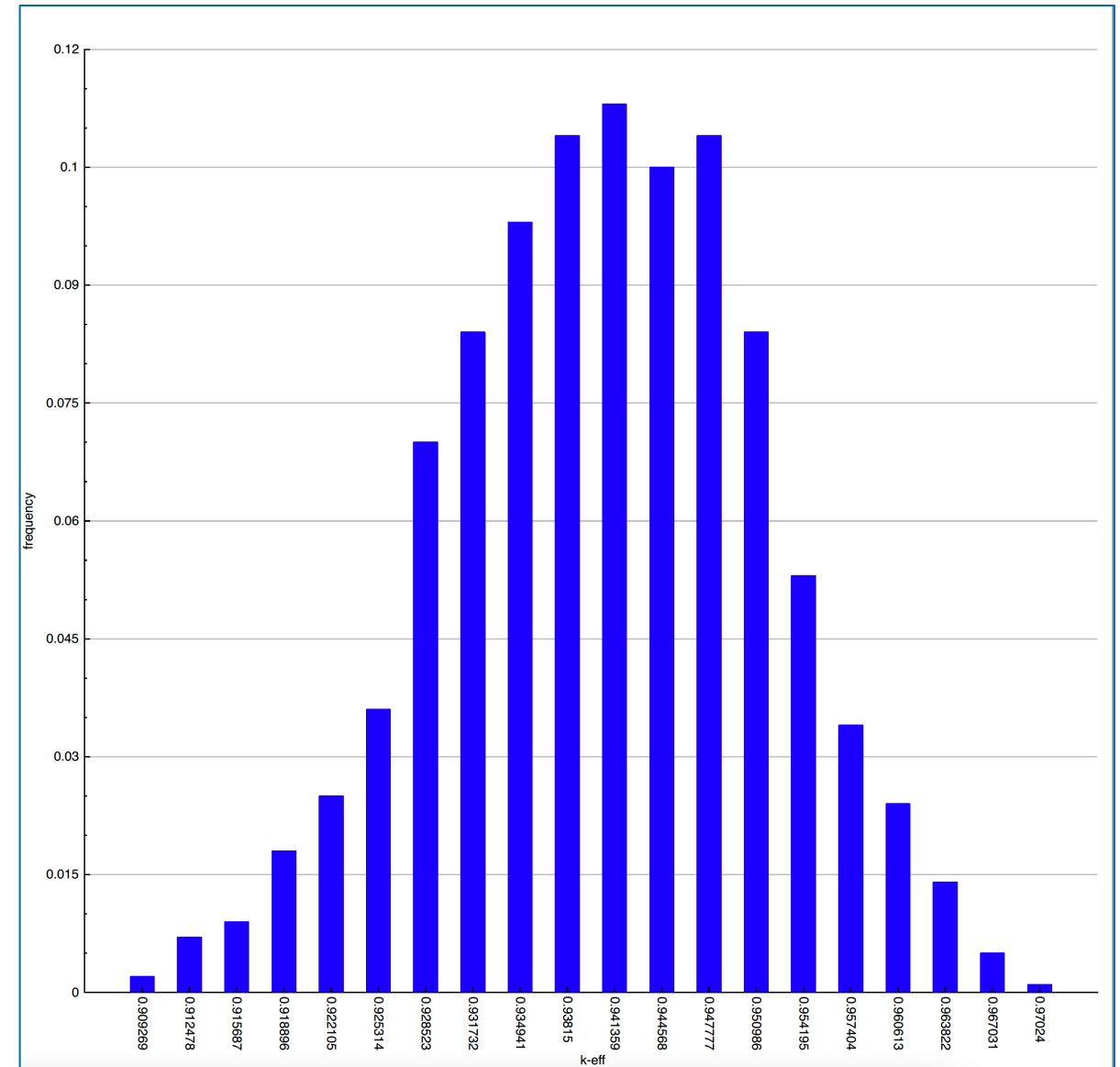
k_{eff} bias and bias uncertainty:

$$k_{eff} \text{ bias} = k_{eff}^{REF} - \bar{k}_{eff}$$

$$k_{eff} \text{ bias uncertainty} = \sigma_{k_{eff}} tf_1^M$$

k_{eff}^{REF} is k_{eff} calculated for the reference criticality model

tf_1^M is one-sided tolerance-limit factor for the normal distribution corresponding to M number of calculated k_{eff} values, at 95% probability, 95% confidence level



Example of k_{eff} distribution plot generated by Sampler

Stochastic sampling method in SCALE/Sampler is straightforward for the purpose here, when bias and uncertainty in calculated nuclide compositions are available

Demo follows here

Sampler model for cask with assembly at 60 GWt/tU burnup

```
=sampler

read parameters
  n_samples=1000
  library="v7-252"
  perturb_xs=no
  perturb_geometry=yes
  run_cases=yes
  force_run=no
end parameters

read case [PWR_cask]

sequence=csas6
gbc-32
v7-252
read comp
'burnup= 55.0 gwd/mtu; enrichment=5.79 %
' #{variable id}placeholder for each burnup credit
' nuclide in each of the 18 fuel materials
'node[01]
o-16      101  0  4.68370E-02  293.0 end
u-234     101  0  #{u2341}    293.0 end
u-235     101  0  #{u2351}    293.0 end
u-236     101  0  #{u2361}    293.0 end
u-238     101  0  #{u2381}    293.0 end
np-237     101  0  #{np2371}    293.0 end
pu-238     101  0  #{pu2381}    293.0 end
pu-239     101  0  #{pu2391}    293.0 end
pu-240     101  0  #{pu2401}    293.0 end
pu-241     101  0  #{pu2411}    293.0 end
pu-242     101  0  #{pu2421}    293.0 end
am-241     101  0  #{am2411}    293.0 end
am-243     101  0  #{am2431}    293.0 end
mo-95      101  0  #{mo951}     293.0 end
tc-99      101  0  #{tc991}     293.0 end
ru-101     101  0  #{ru1011}    293.0 end
rh-103     101  0  #{rh1031}    293.0 end
ag-109     101  0  #{ag1091}    293.0 end
cs-133     101  0  #{cs1331}    293.0 end
nd-143     101  0  #{nd1431}    293.0 end
nd-145     101  0  #{nd1451}    293.0 end
sm-147     101  0  #{sm1471}    293.0 end
sm-149     101  0  #{sm1491}    293.0 end
sm-150     101  0  #{sm1501}    293.0 end
sm-151     101  0  #{sm1511}    293.0 end
sm-152     101  0  #{sm1521}    293.0 end
eu-151     101  0  #{eu1511}    293.0 end
eu-153     101  0  #{eu1531}    293.0 end
```

```
' random numbers drawn from standard normal distribution,
' one random number per burnup credit nuclide, per perturbed case
read variable [R_u235]
  distribution=normal
  stddev=1
  value=0
end variable

.....

read variable [R_gd155]
  distribution=normal
  stddev=1
  value=0
end variable

.....

' variable for the concentrations  $C_{i,Bu}^m$  defined in Eq.(5)
' if value is less than 0, is set to 0

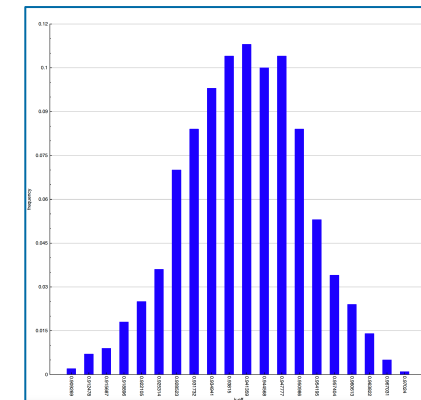
read variable [u2351]
  distribution=expression
  expression="if((6.01776E-04+R_u235*2.33209E-05<0),0,(6.01776E-04+R_u235*2.33209E-05))"
end variable
read variable [u2352]
  distribution=expression
  expression="if((3.85104E-04+R_u235*3.15434E-05<0),0,(3.85104E-04+R_u235*3.15434E-05))"
end variable

.....

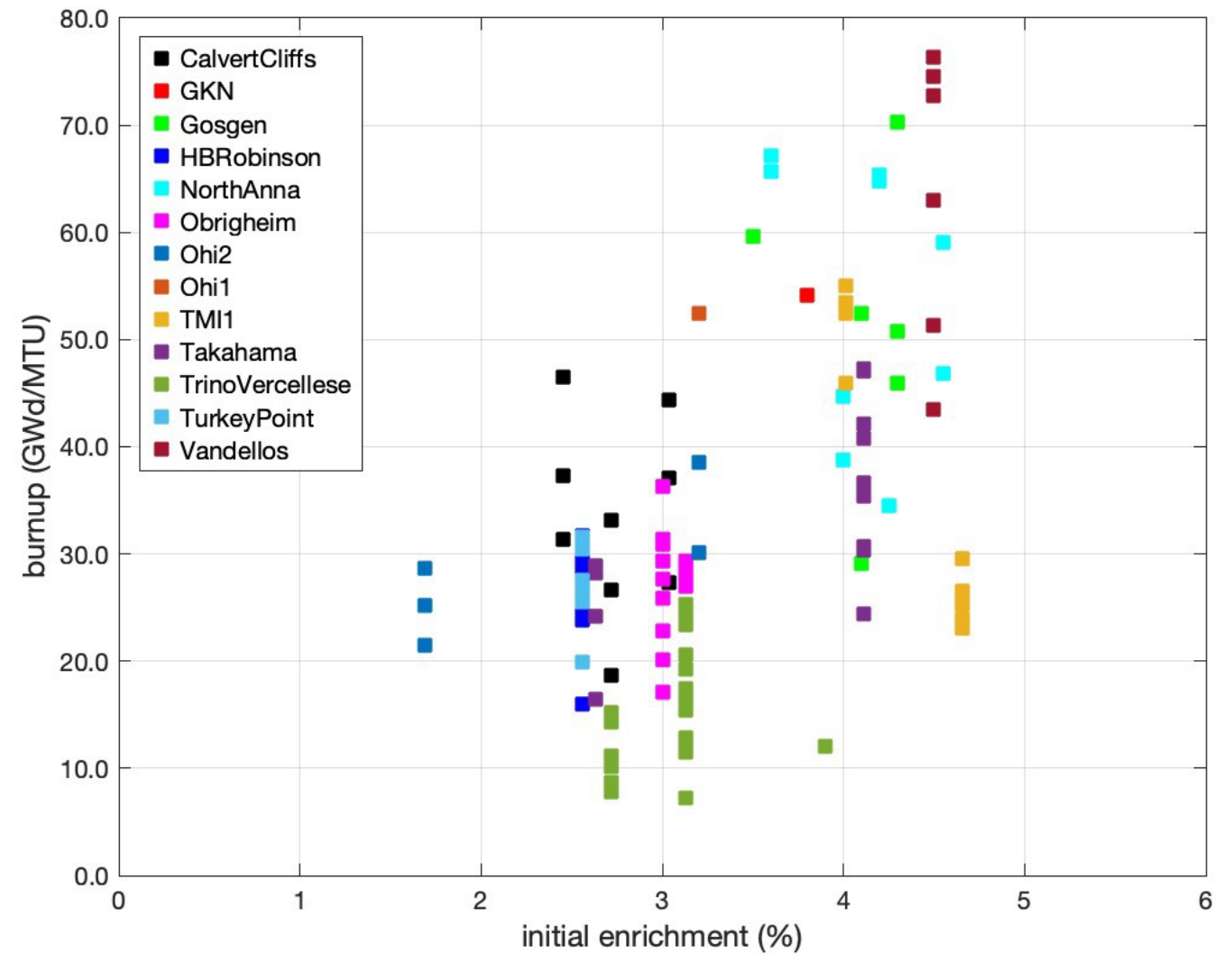
end case

' desired calculated response
read response[keff]
type = grep
regexp = ":kenovi.keff:"
end response

end
```



Nuclide inventory validation basis



RCA nuclide inventory validation basis: 129 PWR fuel samples

Reactor	Country	Measurement laboratory	Experimental program	Assembly lattice	Enrichment (%)	No. of samples / fuel rods	Burnup (GWd/MTU)
Calvert Cliffs-1	US	PNL, KRI	ATM-103, ATM-104, ATM-106	14 × 14	2.453, 2.72, 3.038	9/3	19–47
GKN II	Germany	SCK·CEN	REBUS	18 × 18	3.8	1/1	54
Gösgen	Switzerland	SCK·CEN, ITU, CEA, PSI	ARIANE, MALIBU	15 × 15	3.5, 4.1, 4.3	6/3	29–70
H. B. Robinson	US	PNL, LANL	ATM-101	15 × 15	2.561	7/3	16–32
North Anna	US	ORNL	NRC/DOE/ORNL	17 × 17	3.6, 4.0, 4.2, 4.5	9/5	33 – 67
Obrigheim	Germany	JRC, ITU, IRCh,WAK	EUR, ICE	14 × 14	2.83, 3.00, 3.13	15/7	17–31
Ohi-1	Japan	JAERI	JAERI	17 × 17	3.2	1/1	52
Ohi-2	Japan	JAERI	JAERI	17 × 17	1.6874, 3.2	5/3	22–39
Takahama-3	Japan	JAERI	JAERI	17 × 17	2.63, 4.11	13/3	17–47
TMI-1	US	GE-VNC, ORNL	YMP, ORNL	15 × 15	4.013, 4.657	13/5	23–55
Trino Vercellese	Italy	JRC Ispra, Karlsruhe	EUR	15 × 15	2.719, 3.13, 3.897	31/10	7–25
Turkey Point-3	US	Battelle-Columbus	NWTS	15 × 15	2.556	13/7	20–32
Vandellos II	Spain	Studsvik	CSN/ENUSA	17 × 17	4.5	6/2	44–78

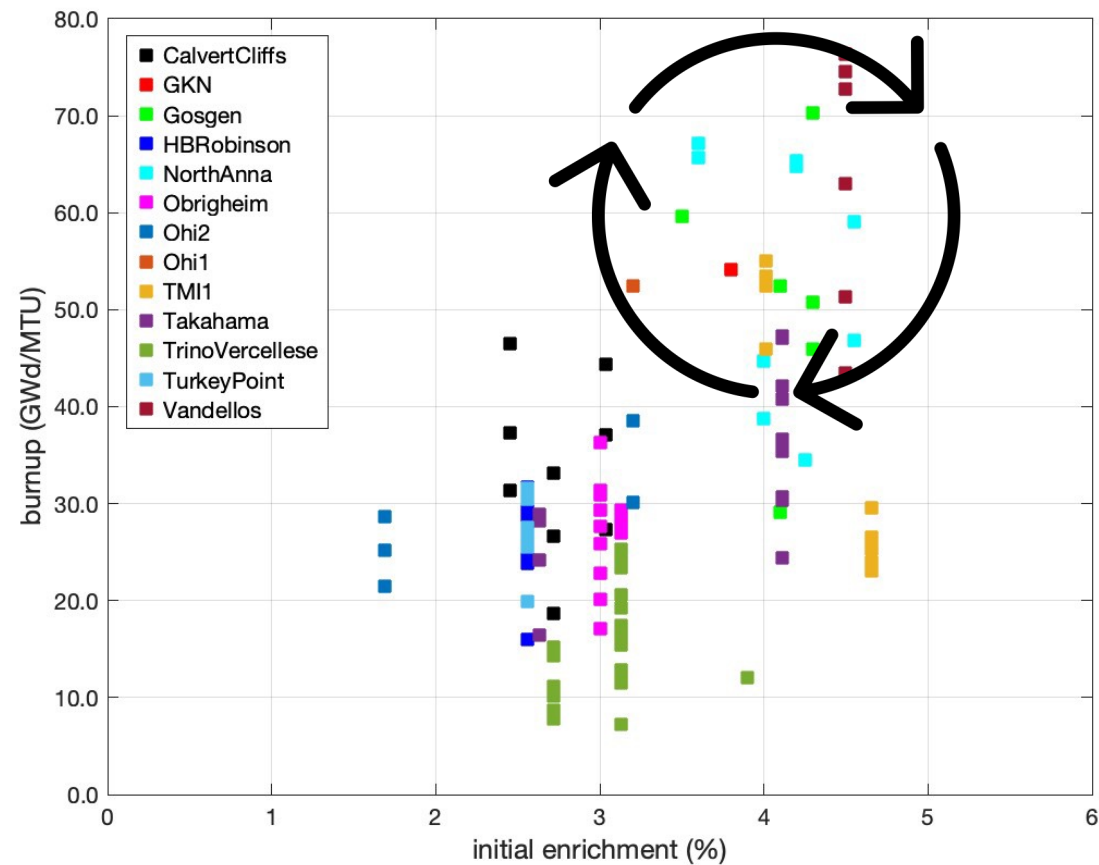
Validation basis for PWR fuel was improved in NUREG/CR-7303:

129 vs 100 fuel samples in NUREG/CR-7108

Additional **37 fuel samples** were included

- **9 North Anna** fuel samples measured by ORNL through joint funding from NRC and DOE (2020-2022)
- **6 Vandellos** fuel samples
- **3 Gosgen** samples from MALIBU program
- **5 TMI1** fuel samples measured at ORNL in 2011
- **3 H.B. Robinson** fuel samples (via OECD/NEA SFCOMPO)
- **8 Turkey Point** fuel samples (via OECD/NEA SFCOMPO)
- **6 Ohi** fuel samples (via OECD/NEA SFCOMPO)

11 TMI-1 fuel samples applied previously were not included, given lower precision of the experiment and huge uncertainty of 7% in the measured sample burnup



Burnup vs enrichment for 129 PWR fuel samples

SCALE/TRITON depletion models for nuclide inventory validation

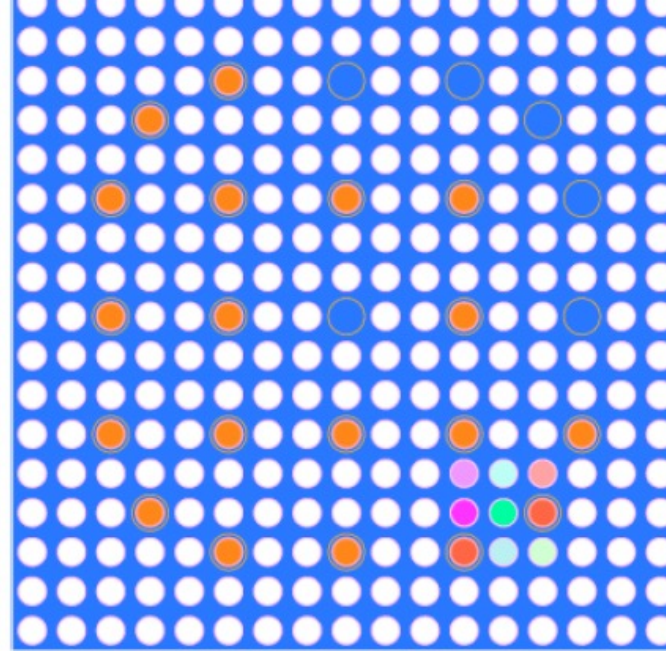
Example:

2D assembly models for simulating the irradiation history of fuel samples selected from rods in North Anna fuel assemblies

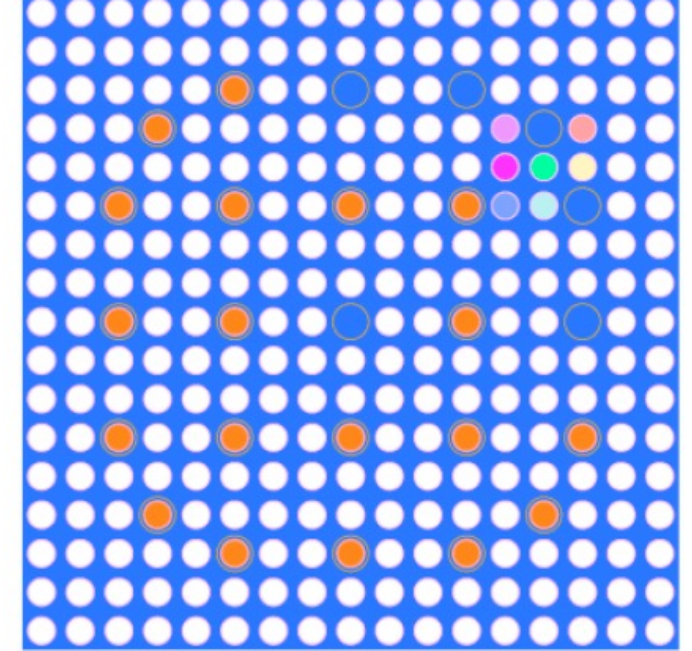
Each measured fuel rod and its nearest neighbors are independently depleted (identified by different colors)

Assemblies are characterized by different BPR patterns (orange color)

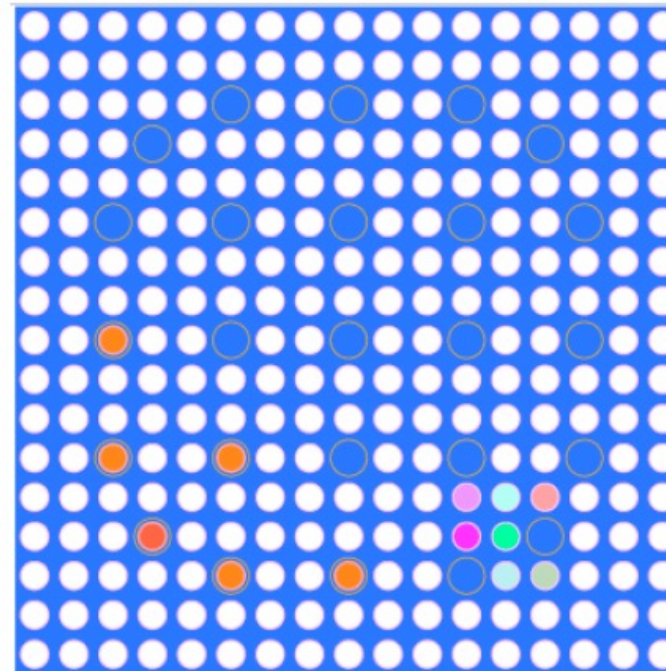
Ref: <https://doi.org/10.1080/00295639.2025.2527491>



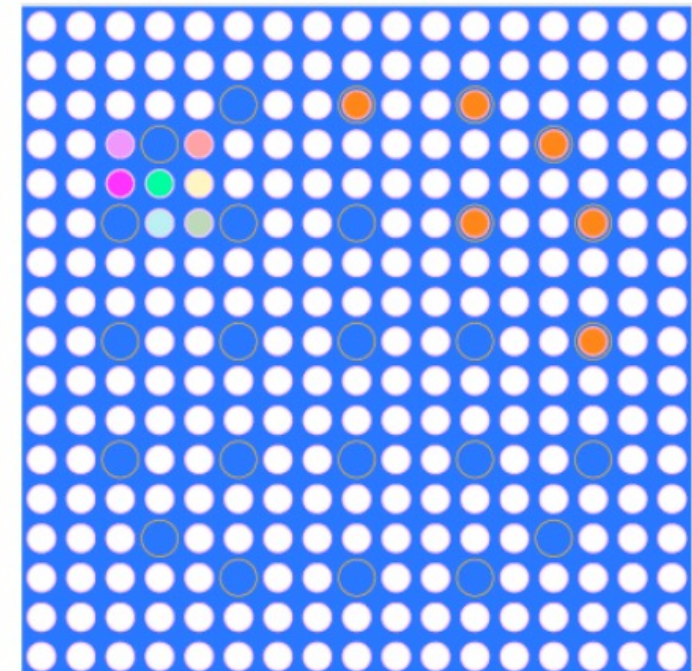
30AE14



30AD05



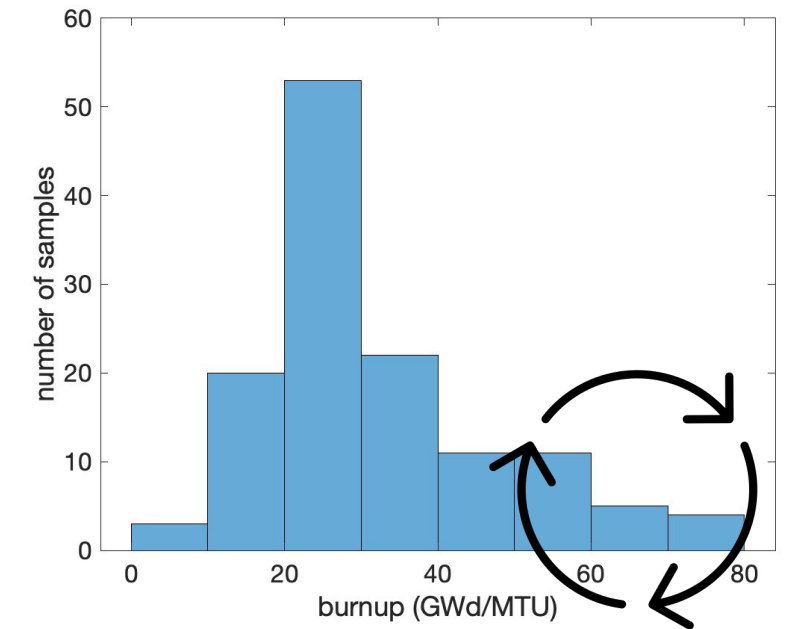
3D8E14



3F9F05

Improvement in validation basis resulted in improved coverage at high burnup and increase in number of measurements for 12 BUC actinides

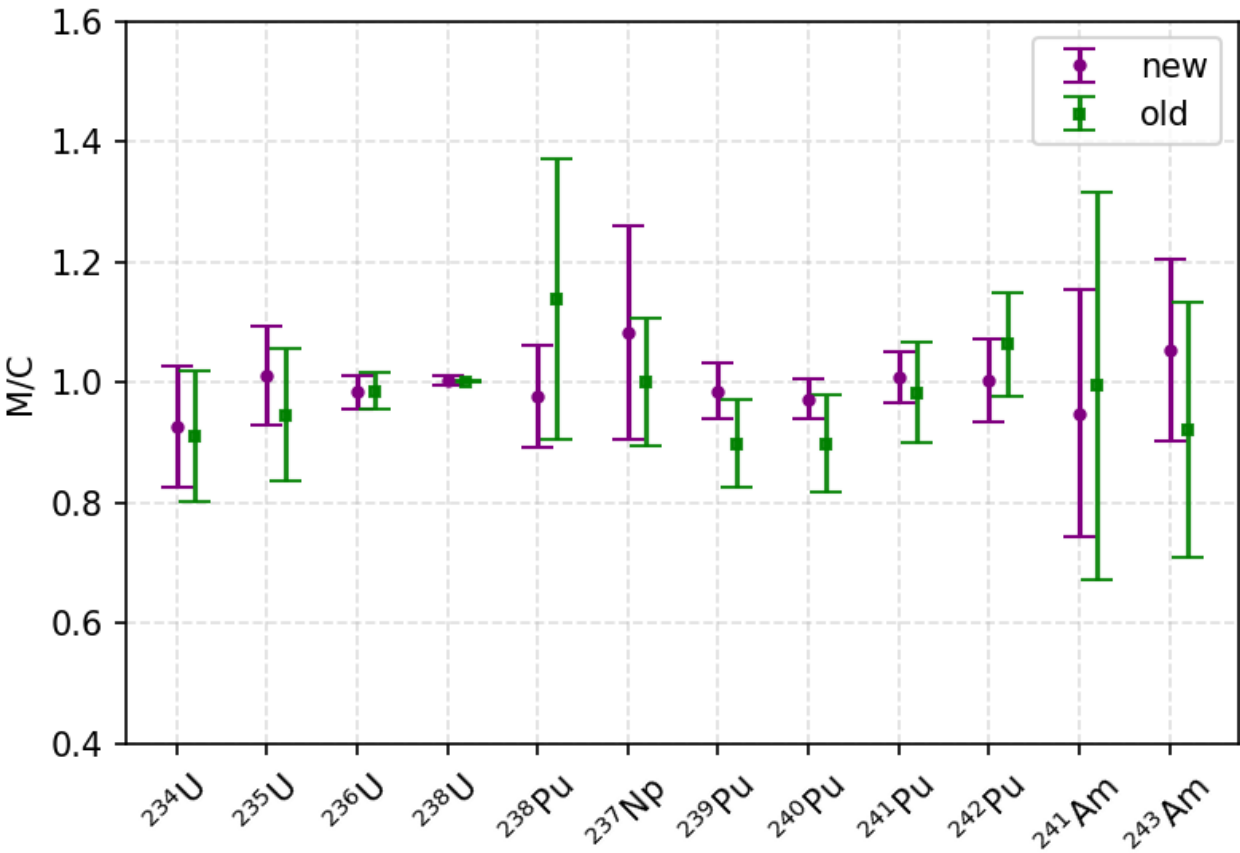
Actinide	Half-life	Number of measurements	Number of measurements	% increase of number of measurements
		NUREG/CR-7303	NUREG/CR-7108	
²³⁴ U	2.45 × 10 ⁵ yr	86	63	37
²³⁵ U	7.04 × 10 ⁸ yr	128	100	28
²³⁶ U	2.34 × 10 ⁷ yr	113	85	33
²³⁸ U	4.47 × 10 ⁹ yr	123	100	23
²³⁷ Np	2.14 × 10 ⁶ yr	49	44	11
²³⁸ Pu	87.71 yr	112	85	32
²³⁹ Pu	2.41 × 10 ⁴ yr	127	100	27
²⁴⁰ Pu	6.56 × 10 ³ yr	127	100	27
²⁴¹ Pu	14.29 yr	127	100	27
²⁴² Pu	3.75 × 10 ⁵ yr	126	99	27
²⁴¹ Am	433 yr	58	47	23
²⁴³ Am	7,370 yr	65	48	35



129 fuel samples burnup distribution

Bias and bias uncertainty for BUC actinides was improved at high BUs

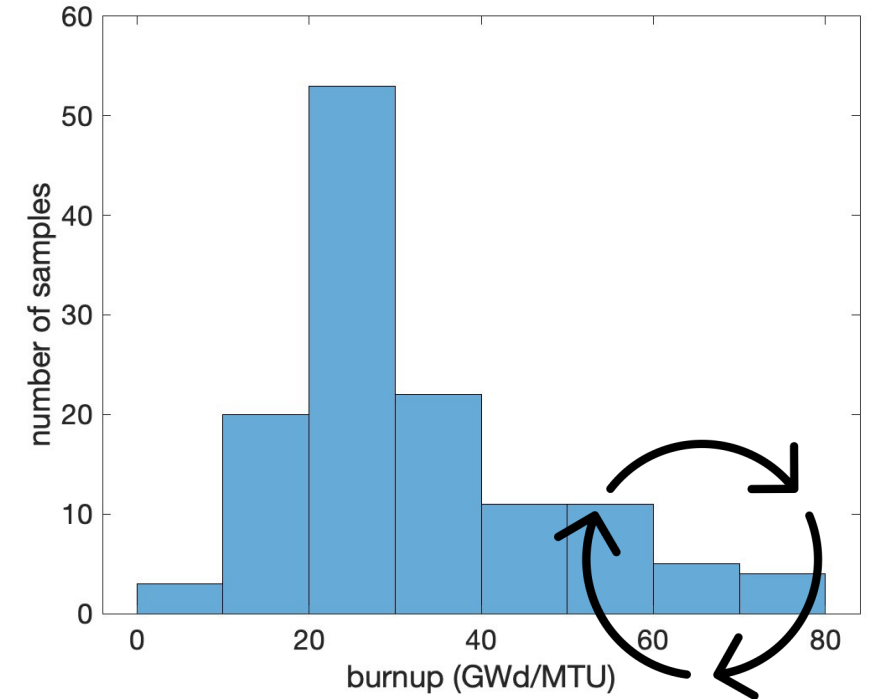
Burnup range	< 40 GWd/t			40 GWd/t - 80 GWd/t		
Nuclide	no. of meas.	bias	bias uncertainty	no. of meas.	bias	bias uncertainty
²³⁴ U	61	0.9136	0.1496	25	0.9276	0.1001
²³⁵ U	98	0.9883	0.0383	30	1.0121	0.0829
²³⁶ U	83	1.0194	0.0407	30	0.9841	0.0274
²³⁸ U	98	1.0010	0.0044	25	1.0029	0.0079
²³⁷ Np	28	1.0634	0.1456	21	1.0828	0.1778
²³⁸ Pu	83	1.0538	0.1012	29	0.9765	0.0848
²³⁹ Pu	98	0.9855	0.0371	29	0.9864	0.0452
²⁴⁰ Pu	98	1.0020	0.0370	29	0.9727	0.0337
²⁴¹ Pu	98	1.0334	0.0614	29	1.0101	0.0424
²⁴² Pu	97	1.0504	0.0826	29	1.0035	0.0691
²⁴¹ Am	28	0.9442	0.1997	30	0.9493	0.2054
²⁴³ Am	36	1.1024	0.2246	29	1.0542	0.1511



Comparison between **NUREG/CR-7303** (Bu > 40 GWd/t) and **NUREG/CR-7108** (Bu > 40 GWd/t)

Improvement in validation basis resulted in improved coverage

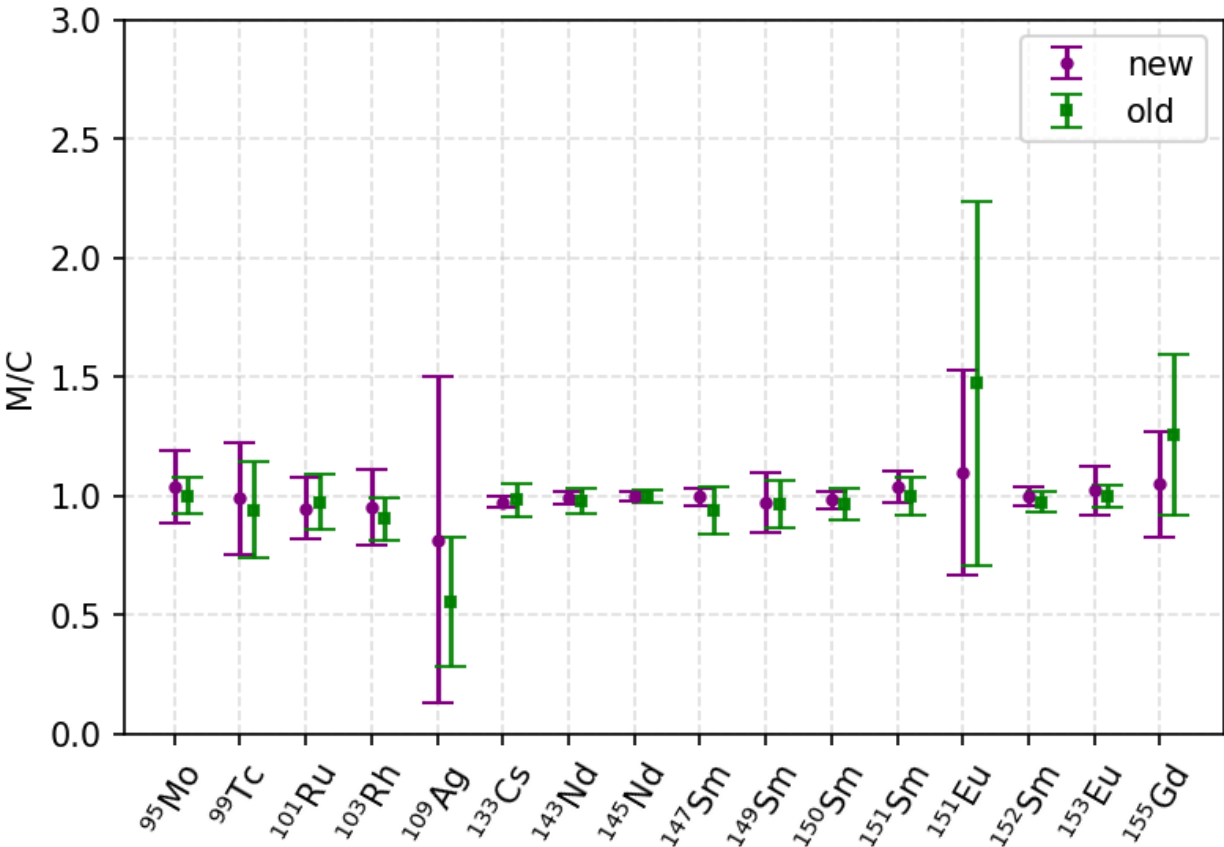
Fission product	Half-life	No. measurements NUREG/CR-7303	No. measurements NUREG/CR-7108	% increase
⁹⁵ Mo	stable	20	15	33
⁹⁹ Tc	2.11 × 10 ⁵ yr	33	25	32
¹⁰¹ Ru	stable	15	15	0
¹⁰³ Rh	stable	21	16	31
¹⁰⁹ Ag	stable	9	14	-36
¹³³ Cs	stable	26	7	271
¹⁴³ Nd	stable	62	44	41
¹⁴⁵ Nd	stable	62	44	41
¹⁴⁷ Sm	1.06 × 10 ¹¹ yr	42	32	31
¹⁴⁹ Sm	stable	38	28	36
¹⁵⁰ Sm	stable	44	32	38
¹⁵¹ Sm	90 yr	43	32	34
¹⁵² Sm	stable	44	32	38
¹⁵¹ Eu	stable	26	21	24
¹⁵³ Eu	stable	38	27	41
¹⁵⁵ Gd	stable	38	27	41



129 fuel samples **burnup distribution**

Bias and bias uncertainty for BUC fission products was improved

Burnup < 80 GWd/t			
	no. of meas.	bias	bias uncertainty
⁹⁵ Mo	20	1.0353	0.1522
⁹⁹ Tc	33	0.9877	0.2343
¹⁰¹ Ru	15	0.9468	0.1314
¹⁰³ Rh	21	0.9524	0.1607
¹⁰⁹ Ag	9	0.8146	0.6870
¹³³ Cs	26	0.9732	0.0234
¹⁴³ Nd	62	0.9906	0.0288
¹⁴⁵ Nd	62	0.9969	0.0186
¹⁴⁷ Sm	42	0.9943	0.0370
¹⁴⁹ Sm	38	0.9715	0.1238
¹⁵⁰ Sm	44	0.9812	0.0386
¹⁵¹ Sm	43	1.0360	0.0658
¹⁵² Sm	44	0.9974	0.0395
¹⁵¹ Eu	26	1.0964	0.4285
¹⁵³ Eu	38	1.0212	0.1034
¹⁵⁵ Gd	38	1.0471	0.2243

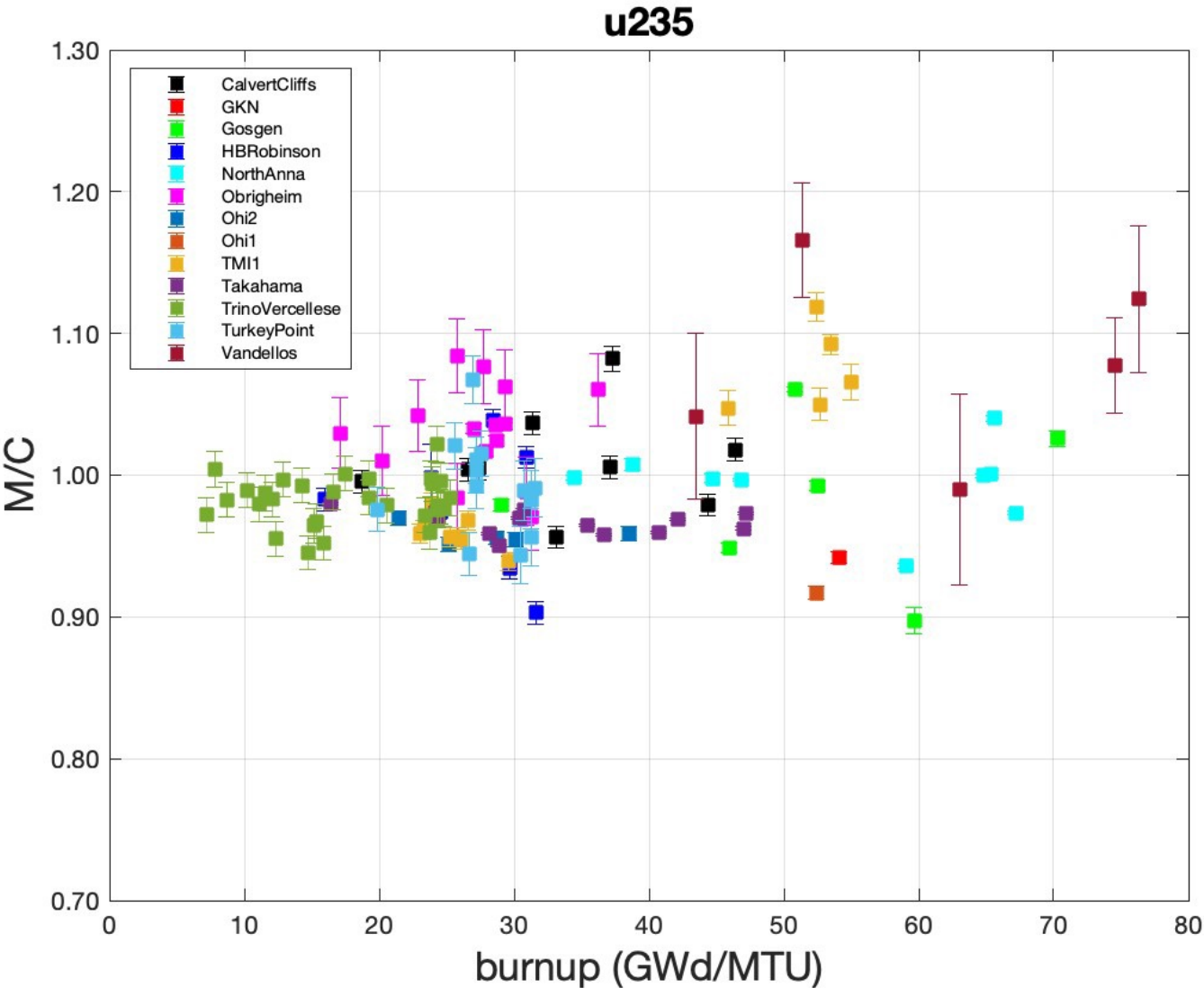


Comparison between **NUREG/CR-7303** (Bu < 80 GWd/t)
and **NUREG/CR-7108** (Bu < 60 GWd/t)

Where do bias and bias uncertainty for BUC actinides come from ?

Ex: M/C ratio vs. burnup for U-235

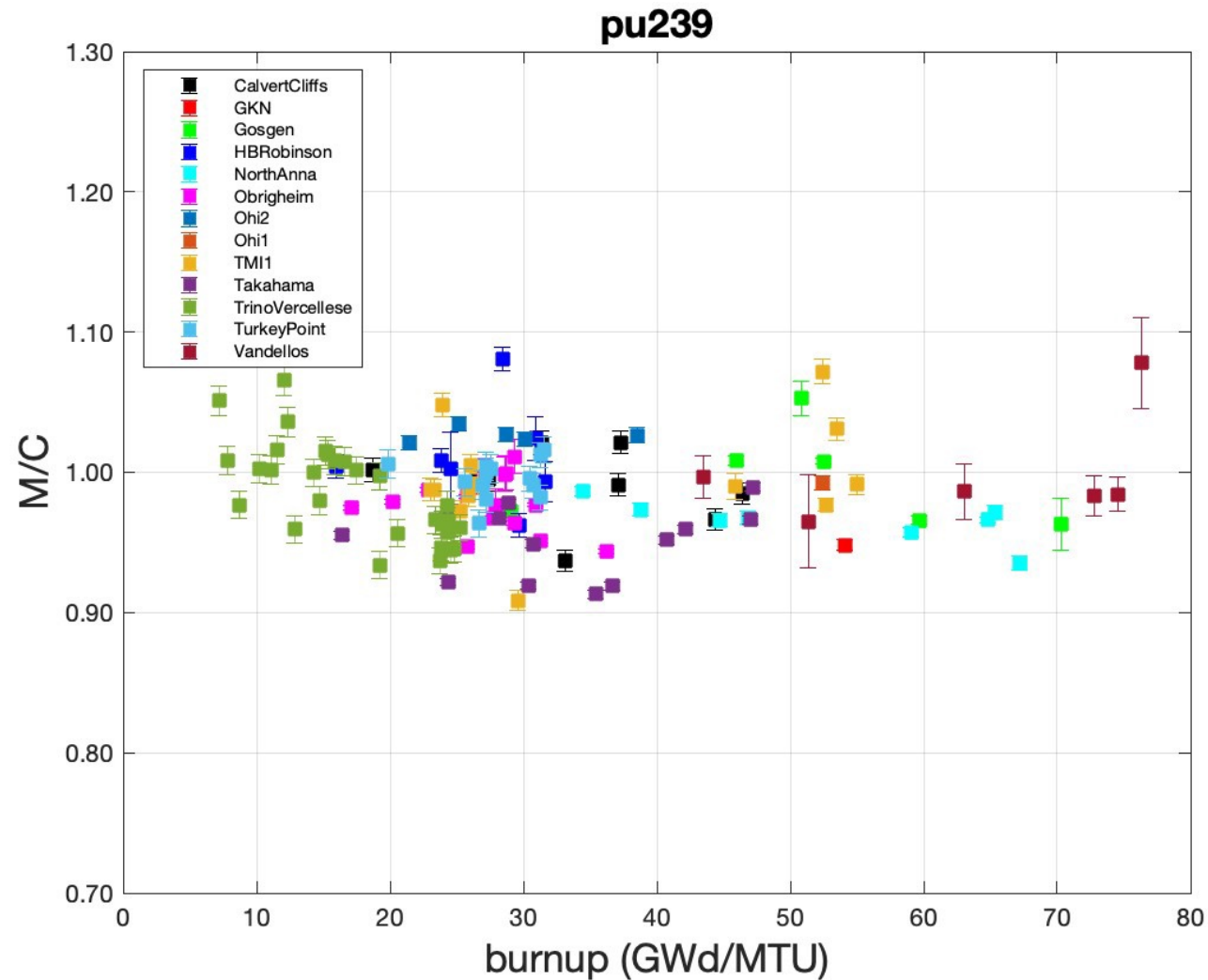
Bu < 40 GWd/t	40 < Bu < 80 GWd/t
98 measurements	30 measurements
(M/C)_{avg} = 0.9883	(M/C)_{avg} = 1.0121
σ = 0.0383	σ = 0.0829



Where do bias and bias uncertainty for BUC actinides come from ?

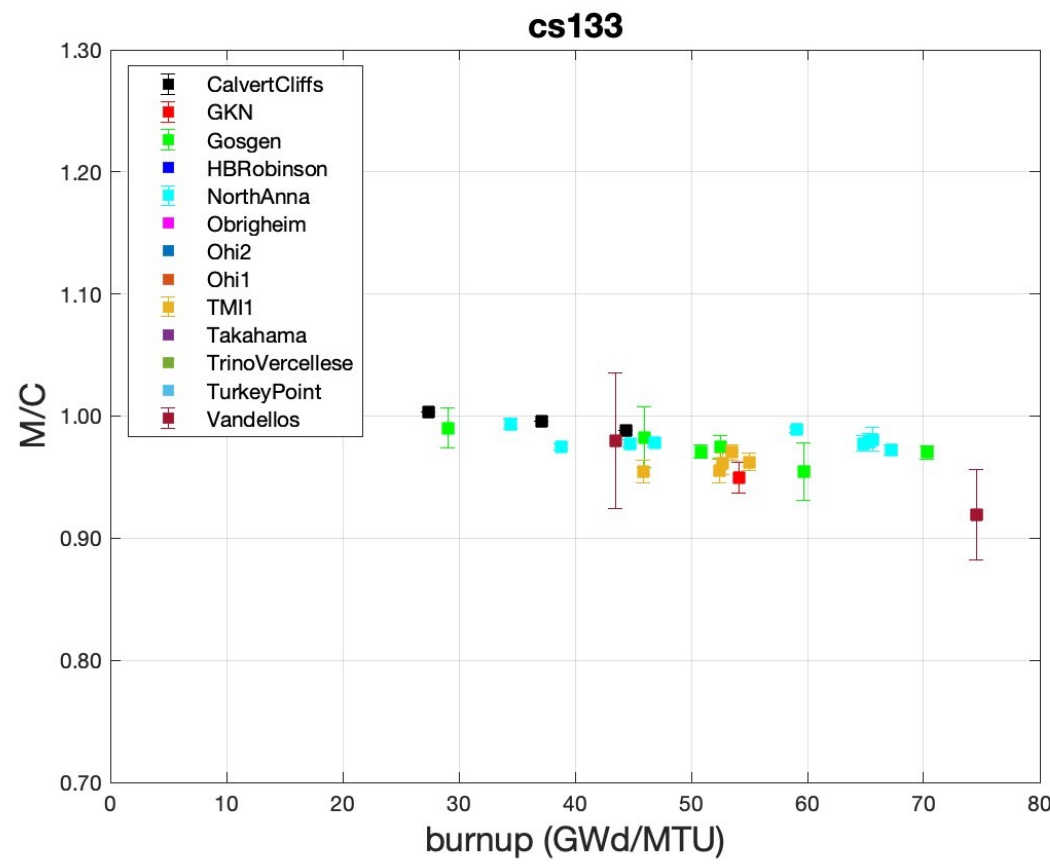
Ex: M/C ratio vs. burnup for Pu-239

Bu < 40 GWd/t	40 < Bu < 80 GWd/t
98 measurements	29 measurements
(M/C)_{avg} = 0.9855	(M/C)_{avg} = 0.9864
σ = 0.0371	σ = 0.0452

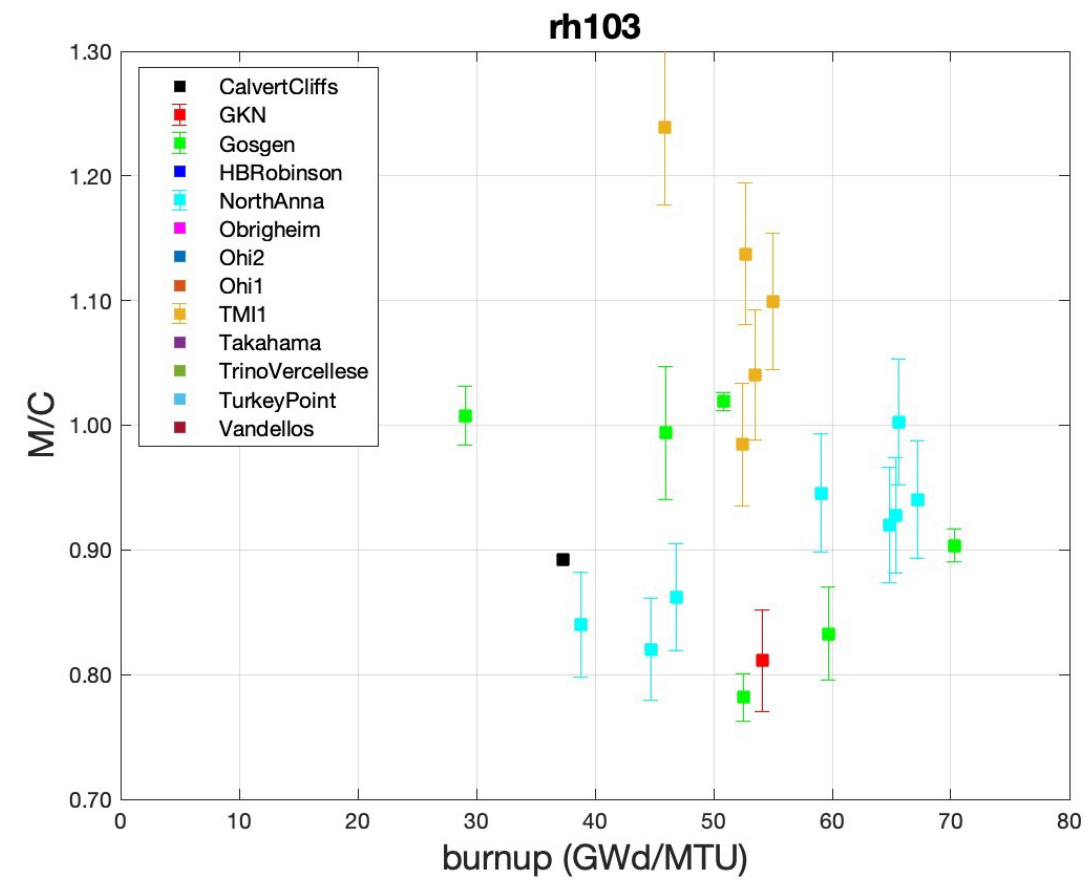


Where do bias and uncertainty for BUC fission products come from ?

Ex: M/C ratio vs. burnup for Cs-133 and Rh-103

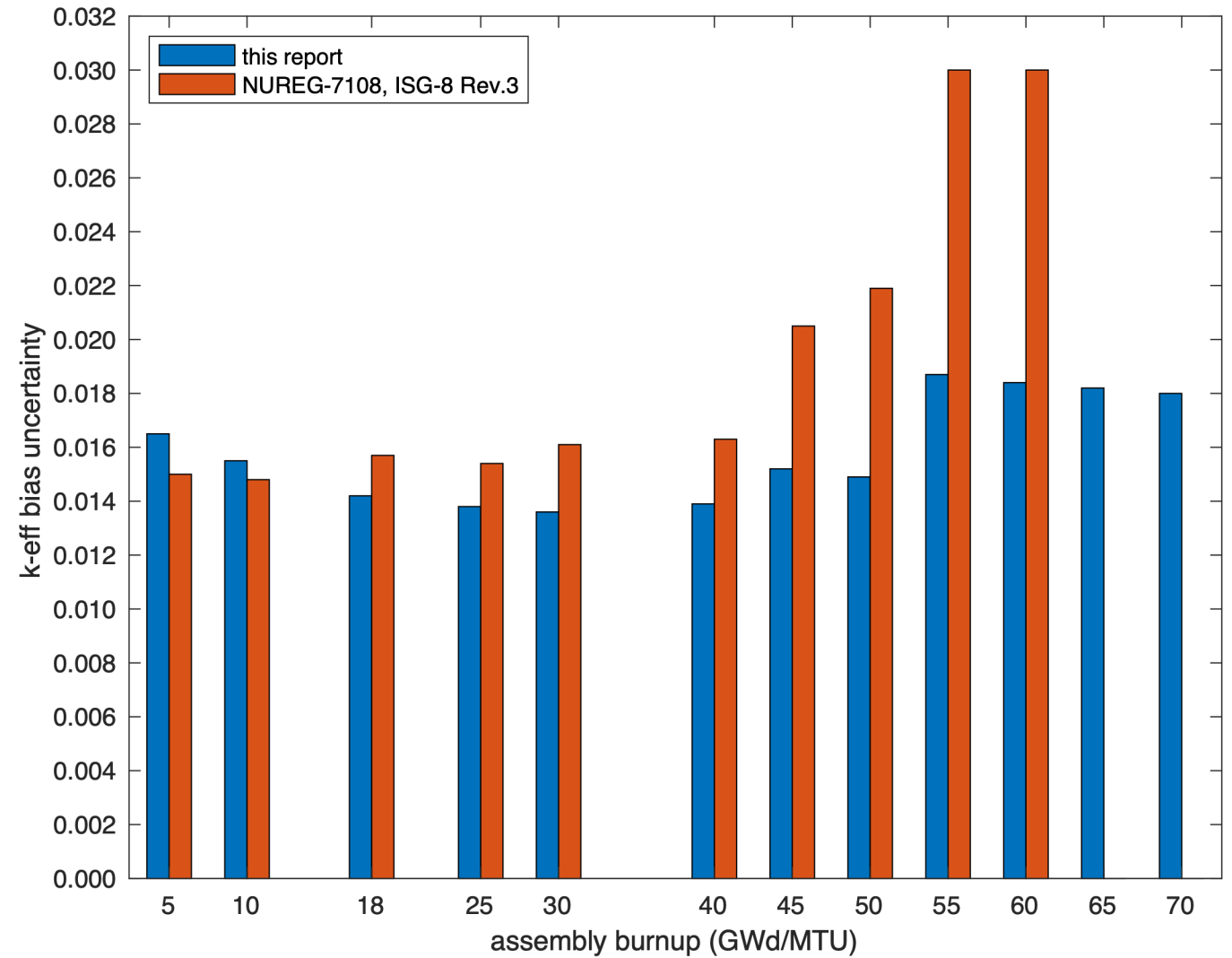


Bu < 80 GWd/
26 measurements
 $(M/C)_{avg} = 0.9732$
 $\sigma = 0.0234$



Bu < 80 GWd/
21 measurements
 $(M/C)_{avg} = 0.9524$
 $\sigma = 0.1607$

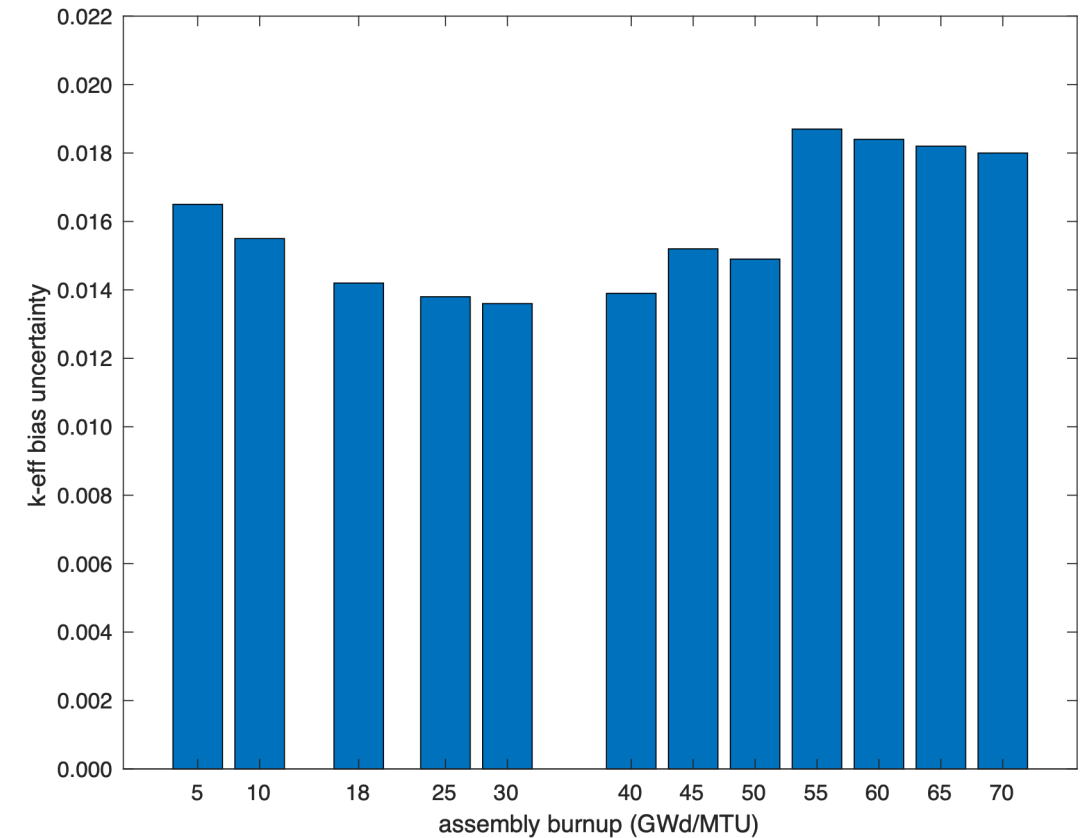
Results and significance



Bias and bias uncertainty in calculated k_{eff} was extended to higher burnups in NUREG/CR-7303

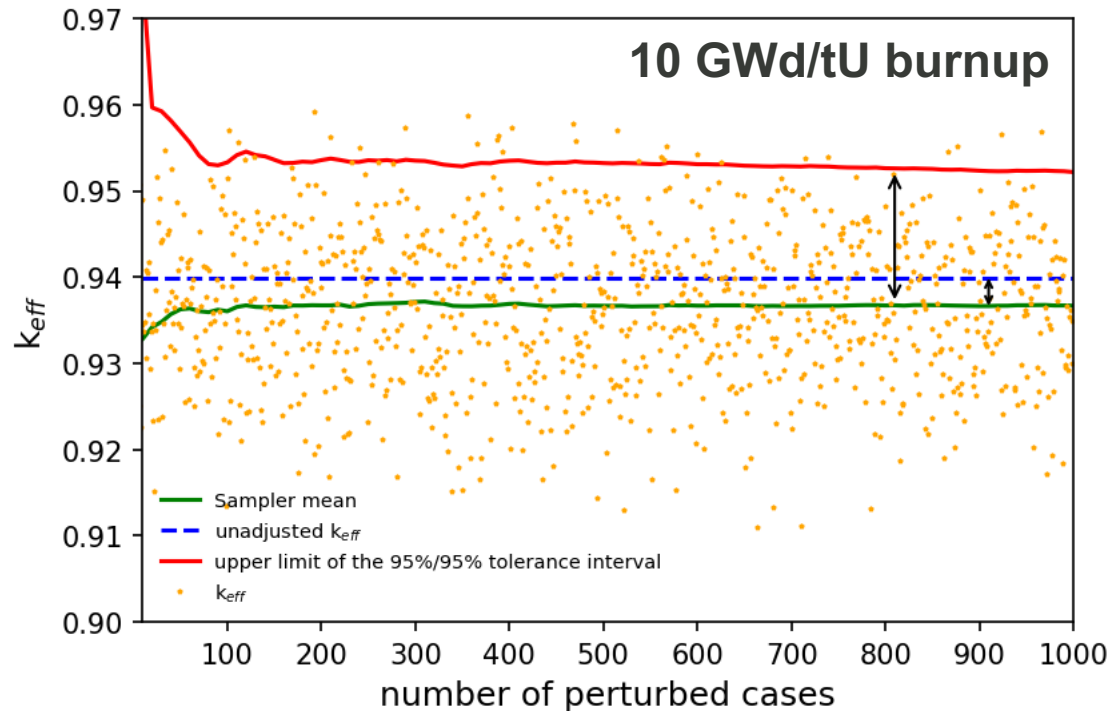
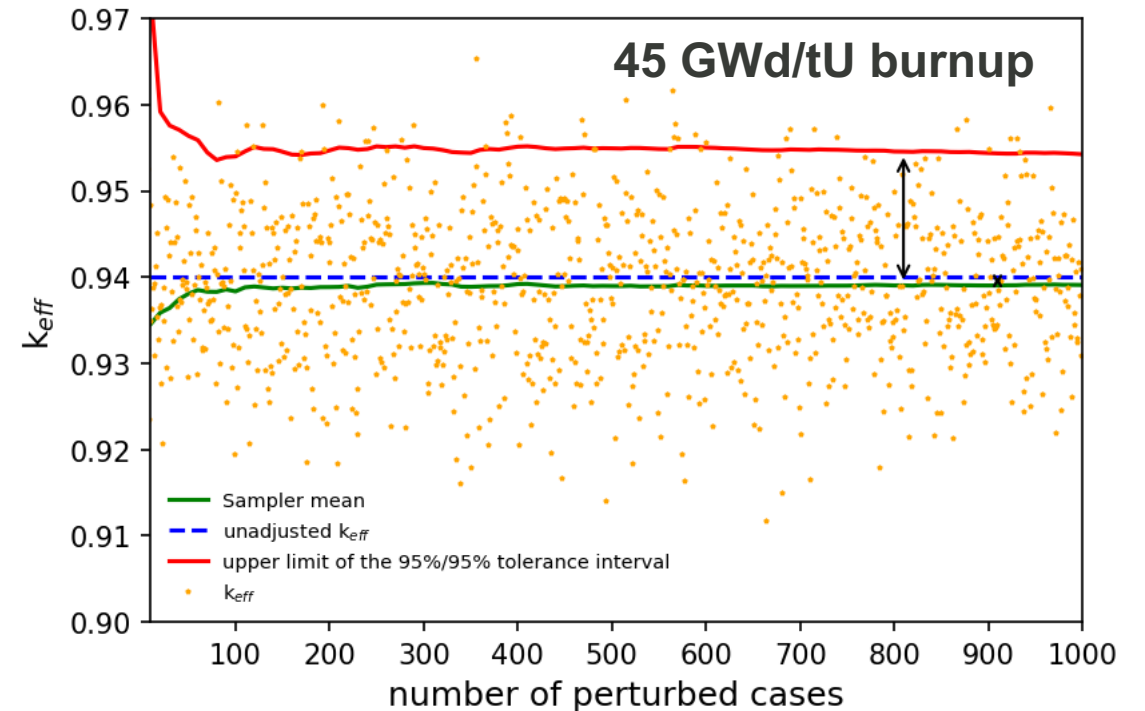
Assembly burnup (GWd/MTU)	Initial enrichment (%)	k_{eff} bias	k_{eff} bias uncertainty
5	2.13	0.0033	0.0165
10	2.36	0.0032	0.0155
18	3.03	0.0028	0.0142
25	3.48	0.0023	0.0138
30	3.98	0.0020	0.0136
40	4.70	0.0017	0.0139
45	5.06	0.0010	0.0152
50	5.42	0.0011	0.0149
55	5.79	-0.0005	0.0187
60	6.16	-0.0007	0.0184
65	6.52	-0.0006	0.0182
70	6.89	-0.0003	0.0180

bias that reduces the calculated k_{eff} is not considered and is set to 0



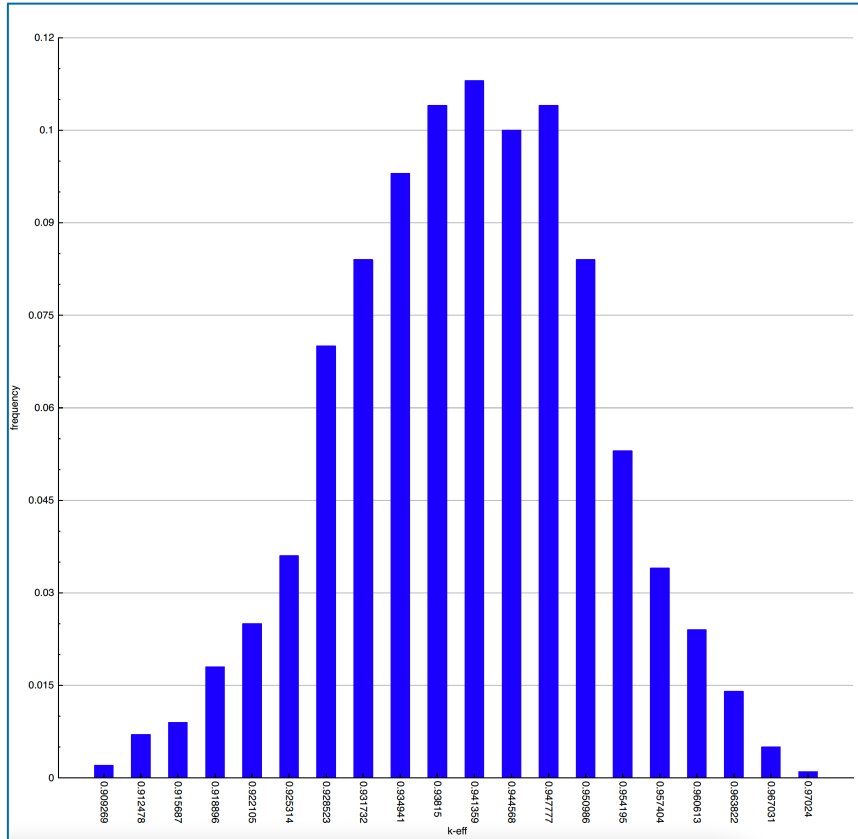
k-eff bias uncertainty vs. assembly burnup

Convergence of the k_{eff} bias and bias uncertainty estimate how does it vary with number of perturbed cases and burnup ?

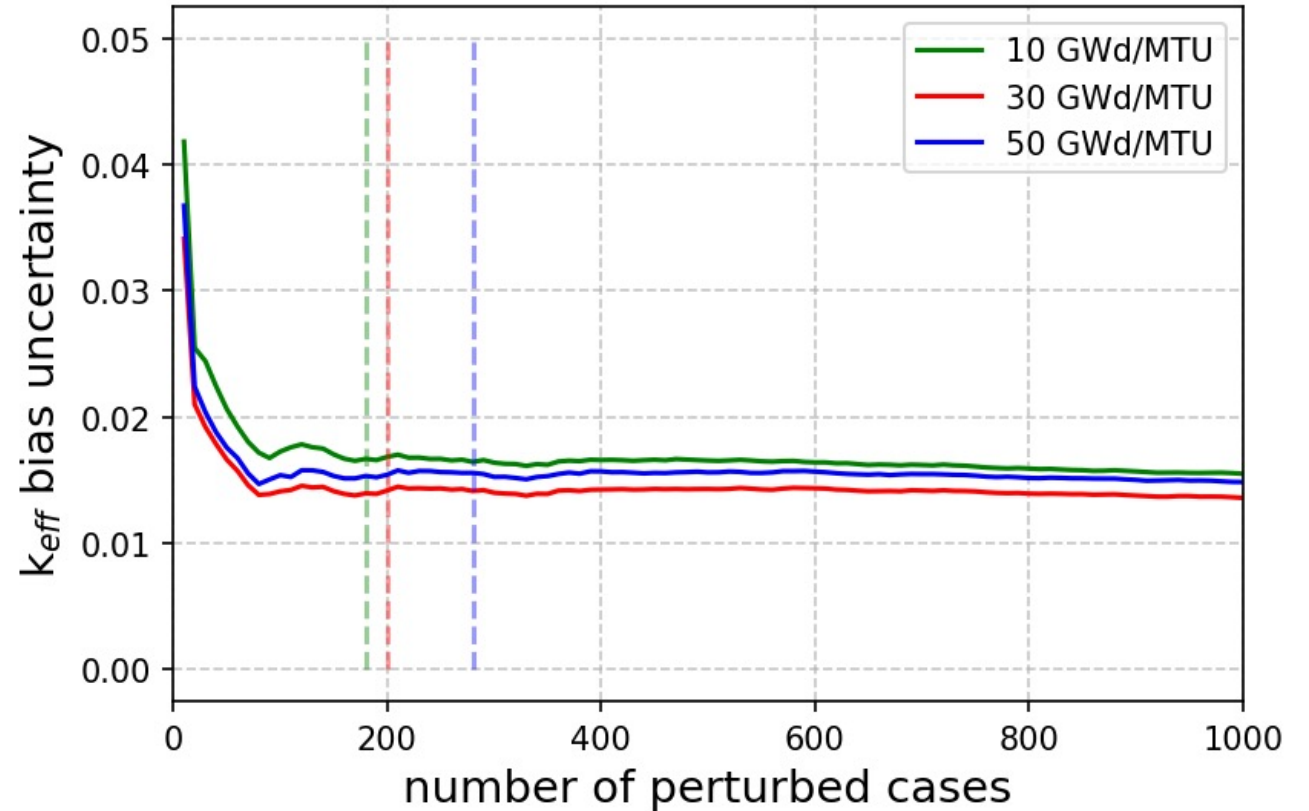


minimum number of perturbed cases to reach convergence can vary slightly with burnup
applied convergence criterion: $k_{N+1} - k_N < 0.0001$

Convergence of the k_{eff} bias uncertainty estimate: **how many perturbed cases (run times) are needed to reach convergence ?**



Histogram of k_{eff} perturbed values: illustration for 55 GWd/tU burnup



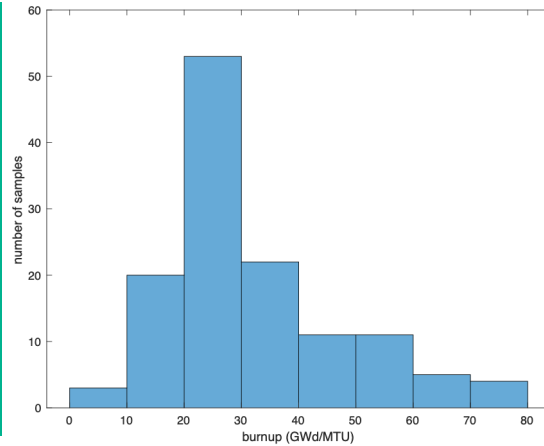
k_{eff} bias uncertainty vs. number of perturbed cases

300 perturbed cases are sufficient for achieving convergence [$k_{N+1} - k_N < 0.0001$] for this model

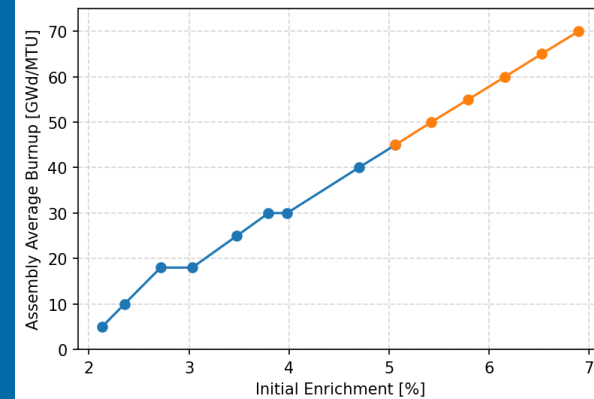
NUREG/CR-7303 primary achievements:

Validation basis and the modeling approach were enhanced

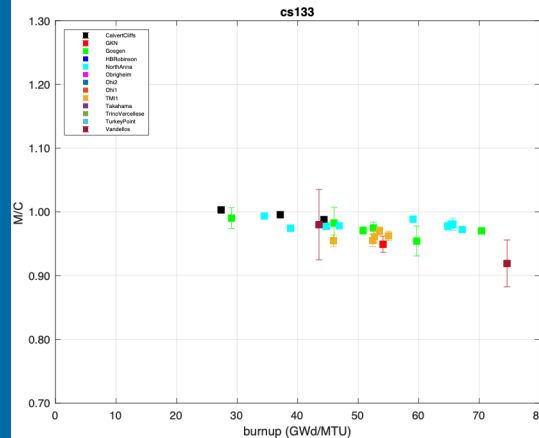
Validation basis was significantly improved: 129 fuel samples, burnups to 80 GWd/tU, more measurements for BUC nuclides



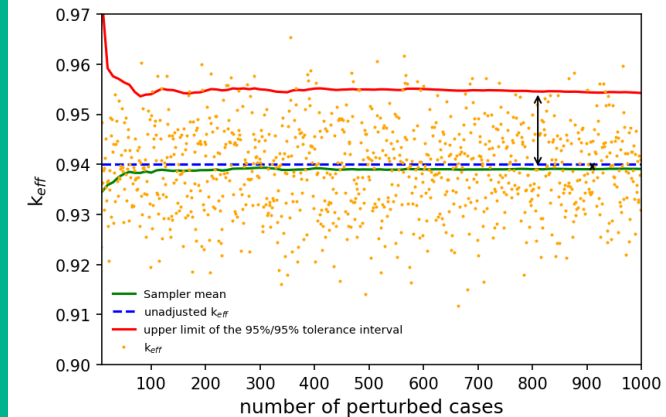
A new loading curve was generated to ensure the reference model used to determine k_{eff} bias and bias uncertainty met the 0.94 target value



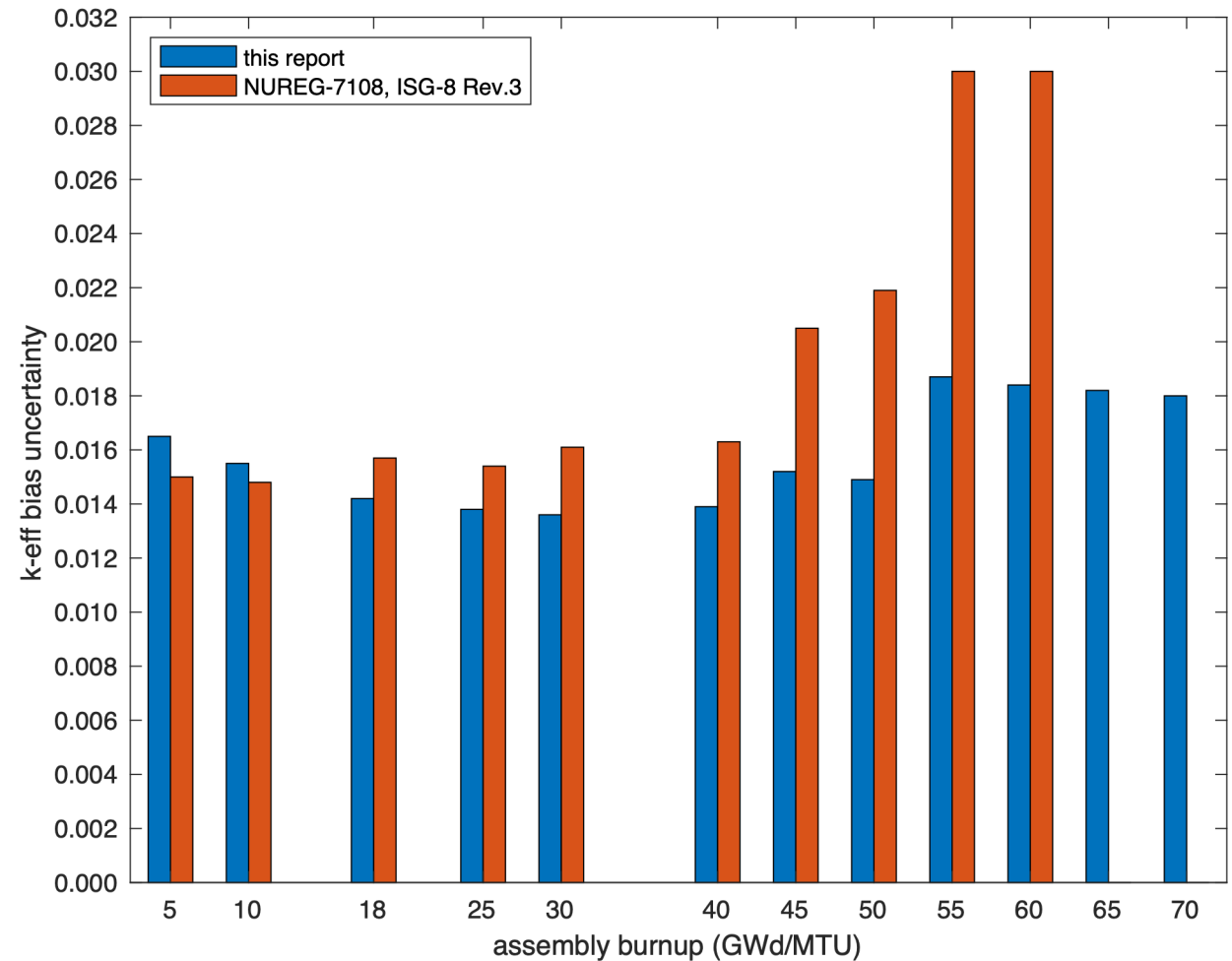
Data gaps for BUC nuclides were reduced and uncertainty in calculated nuclide concentration for most of these nuclides was reduced



New implementation of the stochastic sampling method was using the Sampler uncertainty quantification tool in SCALE 6.2



Improvement in the validation basis led to an overall decrease in k_{eff} bias uncertainty for the GBC-32 cask criticality model compared to corresponding values shown in ISG-8 Rev.3



NUREG/CR-7303 (2023)

Applicability

Path forward and opportunities

Validating Actinides and Fission Products for Burnup Credit Criticality Safety Analyses – Nuclide Compositions Prediction with Extended Validation Basis

Manuscript Completed: May 2023
Date Published: September 2023

Prepared by:
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What did NUREG/CR-7303 bring of value to NRC and industry ?

Provides estimates of bias and uncertainty in calculated nuclide concentrations in PWR spent fuel over an extended burnup range

How well can we predict actinides and fission products of interest for a specific application ?

Provides an improved technical basis to support future updates of current NRC guidance for BUC criticality safety of PWR SNF storage and transportation casks

Interim Staff Guidance (ISG)-8 Rev. 4 (202X) ?

Bias and uncertainty in calculated nuclide concentrations (e.g., how well we can predict U-235 in spent fuel) were estimated independently of the criticality safety model

The BUC nuclides considered are important players for other metrics and applications – decay heat, radioactivity, reactor operation

Provides practical approach for stochastic sampling method use with the Sampler uncertainty quantification tool under a QA-ed system (SCALE)

Can be easily adapted for similar assessments of other, different casks designs

Path forward and opportunities (1)

Modern discharged LWR fuel has different characteristics and operating history (ATF, HBU, EE) than that considered for generating axial burnup profiles in NUREG/CR-6801 (2003)

To improve k_{eff} bias and uncertainty, develop and apply burnup profiles that are representative of current and future discharged LWR fuel

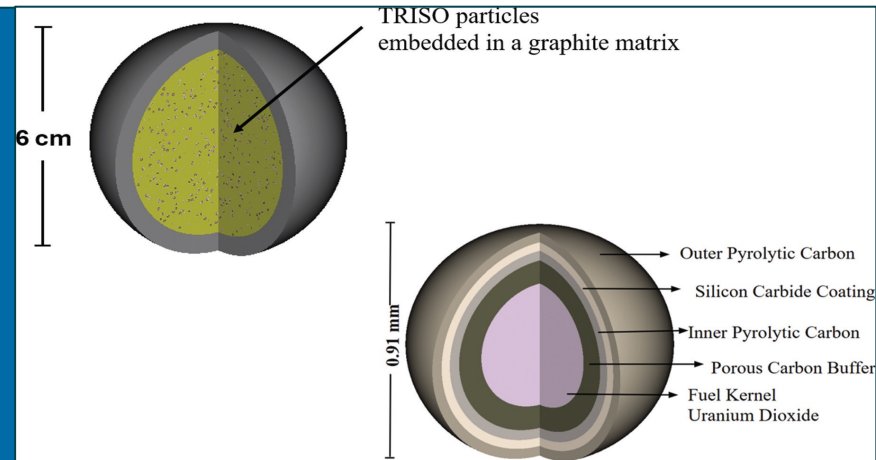
Profiles in NUREG/CR-6801 reflect PWRs operation up to ~ 1995 and assemblies with much lower burnups and less complex designs than in modern PWRs. Axial profile for $\text{Bu} > 30 \text{ GWd/tU}$ was based on 1225 assemblies' data: 40% of them with $30 < \text{Bu} < 35 \text{ GWd/tU}$, and 90% of them with $30 < \text{Bu} < 45 \text{ GWd/tU}$

Extend the validation basis at high burnups with high-quality RCA data to improve coverage and to increase the number of measurements for important nuclides, and consequently to reduce bias uncertainty in predicted concentrations

New RCA experiments are ongoing at ORNL in FY26-28 under NRC funding to add new data for 9 fuel samples: 3 PWR fuel samples with 70-80 GWd/tU , 3 PWR ATF/ADOPT fuel, 3 BWR Gd-doped fuel

Path forward and opportunities (2)

Validation basis for advanced reactor TRISO spent fuel is practically non-existent. We can take advantage of past and ongoing AGR program data and fuels and pursue new experiments for LEU HBU TRISO fuel



Back-end of fuel cycle for advanced reactors is a largely unknown space, with a huge diversity of fuel types, materials, neutron spectra, reactor concepts or operation than current LWRs

We can develop scientifically-grounded, practical methods to extrapolate LWR data and approaches to advanced reactors for which similar knowledge does not yet exist