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Accuracy Evaluation of the Available Fission Yields Data and Updating

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Contents

Introduction (1/2)

- Reliability of nuclear reactor calculations and waste management depends on the accuracy of fission yield data.
- Calculations of system criticality and core cycle length are strongly dependent on the generated neutron poisons such as Xe-135 and Sm-149.
- Evaluated delayed-neutron precursors such as Br-87, Cs-142, I-137,….etc affect the calculations of delayed neutron fraction.
- Evaluation of fissile nuclides (such as residual U-235 and generated Pu nuclides) and/or recovery of valuable fission nuclides (such as Mo-99, I-131, Cs-137,…ext.) is important process before the processing of the spent fuel.
- Evaluation of spent fuel issues such as criticality safety, decay heat output and neutron emissions and radio-toxicity and environmental impact for transportation and storage is strongly dependent on the fission yield data.
- Assessment of nuclear accident needs, as a first step evaluation of the source terms of radioactivities and decay heats from radioactive materials in spent nuclear fuels.

Introduction (2/2)

- The International Atomic Energy Agency (IAEA) has launched a Coordinated Research Project (CRP) titled: “updating fission yield data for applications” with main objective of updating the evaluated independent and cumulative fission yield data for U-235, U-238, and Cf-252.

- The Egyptian Atomic Energy Authority (EAEA) has participated in this project with a research contract under the title of “accuracy evaluation of available fission yield data and updating”. One of the main objectives in this CRP is to evaluate the accuracy of the available fission yield data by simulation of a series of benchmarks in the areas of nuclear reactors calculations.

Aim of the presentation

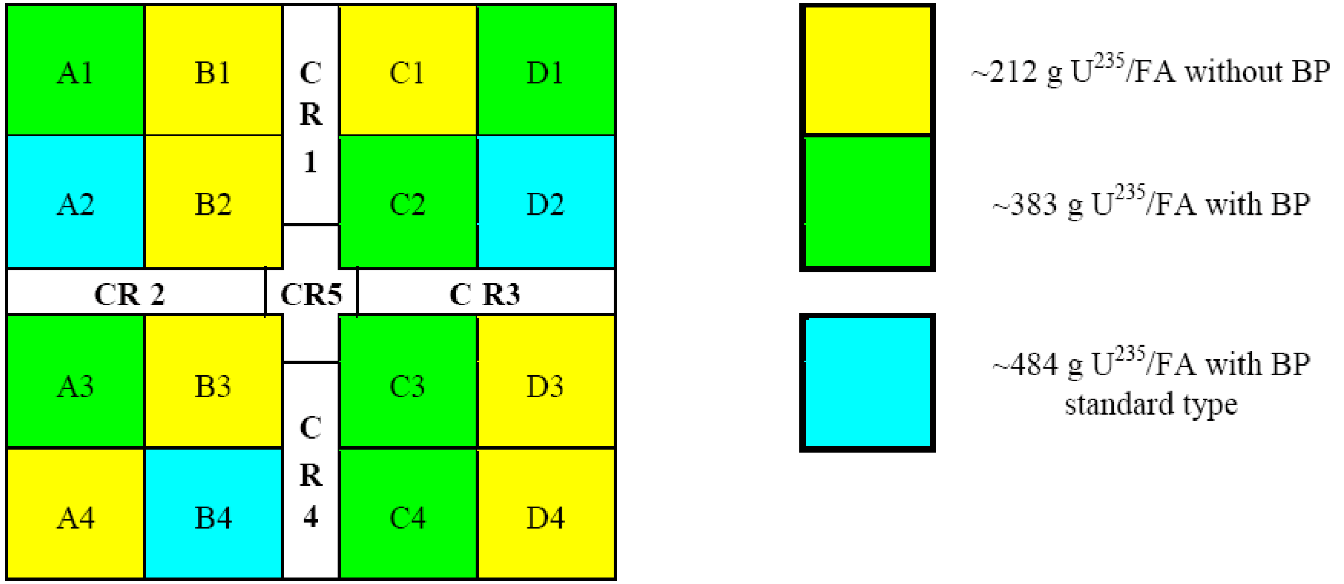
- As part of the project, the latest ENDF/B-VIII.0 and JEFF3.3 data libraries that released on 2018 and 2017, respectively, as well as the ENDF/B-VII.1 data library were tested on two research reactors (ETRR-2 and OPAL) using two different classes of computational codes: MCNPX V2.7.0 and WIMS-5B/CITVAP codes.

- Since the reactor criticality calculations are very sensitive to the data library accuracy, criticality benchmarks were selected in the work for the evaluation of these data libraries.

- This presentation focuses on the results of the OPAL research reactor benchmark.

OPAL Benchmark (1/2)

- The Open Pool Australian Light Water (OPAL) Reactor is a 20 MW, open pool type research reactor. It is composed of a compact core surrounded by a heavy water reflector.
- The reactor core contains 16 (4 × 4) fuel assemblies and four control plates and one regulating plate (CRPs).
- The fuel is flat plate U_3Si_2-Al dispersion with aluminum clad and enrichment of 19.8wt% U-235. Each fuel assembly has 2 aluminum side plates, each with 21 slots to hold 21 fuel plates. Three different types of fuel assembly with different uranium densities were used for the first operating cycle.
- Cadmium wires are provided in some fuel elements as burnable poisons.



OPAL Benchmark (2/2)

- For the criticality and burnup calculations, the benchmark has provided: (1) The power history and the control rod positions, (2) The heavy water reflector purities during the operational cycles and (3) fuel management strategy and loading for the operational cycles: 07 -12.
- The following data is the Cycle 7 operational data.

Time (days)	Reactor power (MW)	Critical and regulating plate position (%)				
		CR-1	CR-2	CR-3	CR-4	CR-5
0	0.00	85.07	23.47	23.10	84.97	49.90
0.58	0.69	85.06	23.47	23.10	85.17	67.54
0.92	0.00	85.05	40.06	40.02	85.06	23.31
1.02	2.41	85.05	40.06	40.02	84.91	22.10
1.08	0.00	85.00	40.00	39.91	85.01	25.41
1.80	5.81	85.00	40.00	39.91	85.01	66.20
1.92	0.00	85.04	45.99	55.02	85.06	62.62
2.79	9.99	85.04	70.60	69.04	85.06	20.09
2.83	0.00	85.00	84.23	84.75	84.96	35.21
2.96	14.36	85.00	62.32	63.95	84.96	21.01
3.00	0.00	84.98	78.05	79.87	85.07	22.34
4.04	19.26	84.98	77.54	81.56	84.93	21.00
5.08	18.74	84.98	84.08	84.55	84.93	21.31
6.13	18.80	84.98	84.08	84.55	84.93	28.33
7.51	18.78	84.98	84.08	84.55	84.93	31.74
9.42	0.00	85.02	85.03	84.87	84.97	26.59
9.92	1.10	85.02	62.12	64.18	84.97	21.03
9.96	8.26	85.02	57.27	62.72	84.97	21.03
10.21	14.76	85.02	72.65	63.33	84.97	21.03

Time (days)	Reactor power (MW)	Critical and regulating plate position (%)				
		CR-1	CR-2	CR-3	CR-4	CR-5
11.29	17.94	85.02	83.96	84.05	84.97	30.29
13.33	0.00	85.08	75.03	75.49	85.01	20.30
13.38	8.63	85.08	65.59	64.42	85.01	20.30
14.38	19.19	85.08	84.01	84.00	85.01	31.92
14.63	14.49	85.08	84.01	84.00	85.01	33.08
17.32	0.00	84.97	58.67	58.16	85.12	20.06
18.13	18.73	84.97	83.99	84.09	85.12	29.85
19.08	18.75	84.97	83.99	84.09	85.12	37.51
20.17	18.75	84.97	83.99	84.09	85.12	39.13
21.25	18.75	84.97	83.99	84.09	85.12	40.44
22.42	18.77	84.97	83.99	84.09	85.12	42.00
23.27	18.76	84.97	83.99	84.09	85.12	42.86
25.29	0.00	85.17	85.13	84.90	85.05	29.14
25.33	13.52	85.17	70.97	71.01	85.05	20.12
26.42	18.97	85.17	83.91	83.97	84.91	40.92
27.58	18.93	85.17	84.06	83.97	84.91	42.97
28.75	18.99	85.17	84.06	83.97	84.91	43.08
29.92	19.54	85.17	84.06	83.97	84.91	43.19
31.25	19.60	85.17	84.06	84.11	84.91	43.10
32.46	19.59	85.17	84.06	84.11	84.91	43.19
33.71	19.55	85.17	84.06	84.11	84.91	43.57
34.92	19.43	85.17	84.06	84.11	84.91	44.20
36.13	19.37	85.17	84.06	84.11	84.91	44.95
37.46	19.40	85.17	84.06	84.11	84.91	46.73
38.83	18.70	85.17	84.06	84.11	84.91	49.03

Computational Code

- The Monte Carlo N-Particle (MCNPX2.7.0) code is used in madling the benchmark and calculations.

- Till MCNP5 version released in 2003, MCNP code has not had the capability of material depletion/burnup calculations.

- By the release of MCNPX2.7.0 version in 2011, the user has the capability of depletion/burnup calculations.

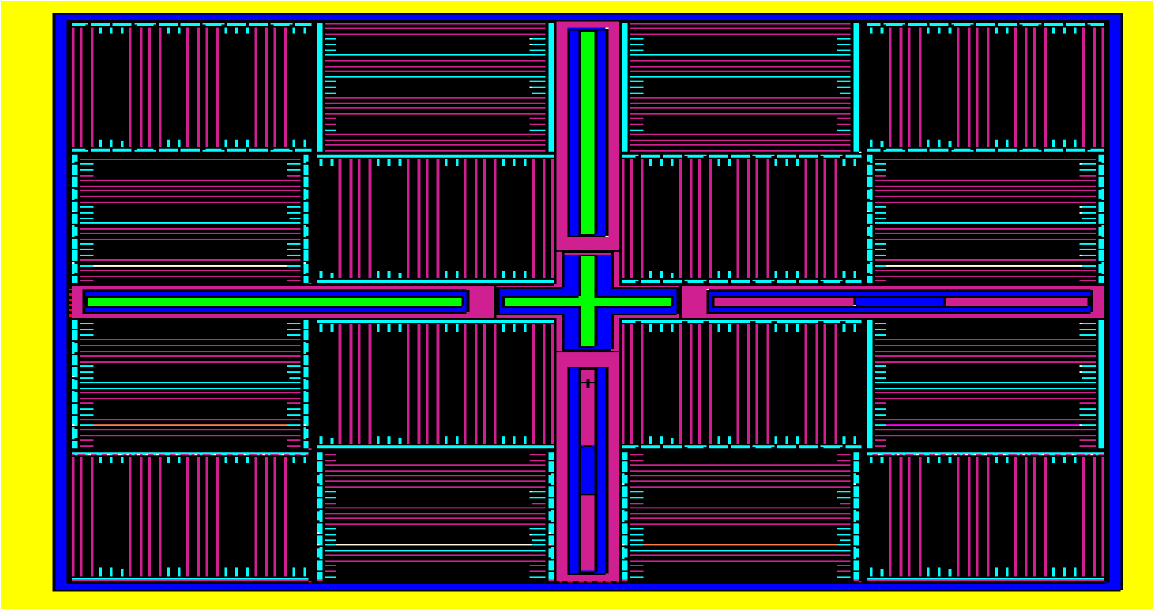
- Two years later MCNP6™ has been released which provides new options of calculating the point-kinetics parameters: the neutron generation time, the effective delayed neutron fraction and Rossi-alpha.

Data Libraries

- The nuclear data cross section libraries can be classified into two main divisions. The first one is continuous energy cross section libraries, which is mainly used for stochastic calculation i.e. Monte Carlo transport calculations.
- The other is multi-group cross section libraries, generally in WIMS format, which are used for deterministic calculations.
- A continuous energy cross section library gives results of higher accuracy than multi-group cross section libraries.
- However, stochastic calculations using continuous energy cross section libraries are intensely time-consuming.
- Nuclear data originally described in the Evaluated Nuclear Data File (ENDF) with versions: ENDF/B-VII.1 and ENDF/B-VIII.0, and Joint Evaluated Fission and Fusion (JEFF) with version: JEFF 3.3 were used in computational code: MCNPX2.7.0.

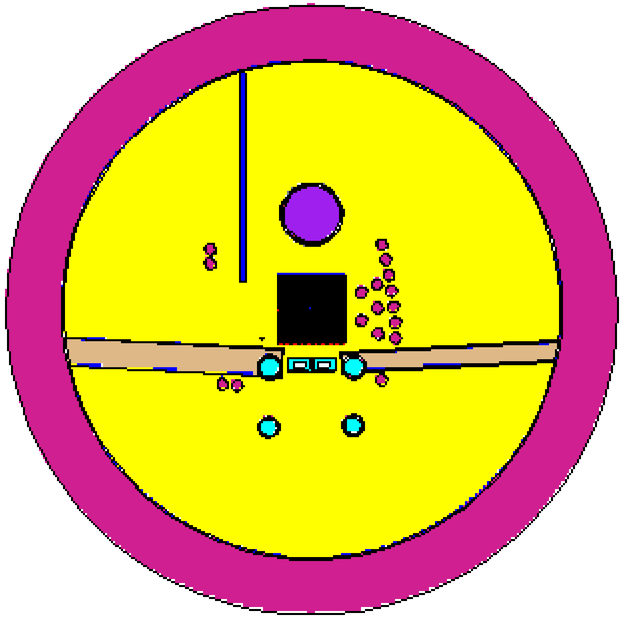
Modeling

- The reactor have been modeled using the reactor specifications and data given in the benchmark.
- In MCNPXV2.7.0 code, BURN card were used with a criticality calculation (KCODE calculations) to calculate the system criticality and the fuel burnup and material inventories after each time interval (defined in the BURN cards).
- The decay time between core cycles were considered. All nuclear cross sections were recalled at cold state (~20 °C).
- A total of about 300 cycles of which 50 were skipped and 10000 histories per cycle were employed.
- The material inventories from cycle n were inputted in the simulation of cycle n+1

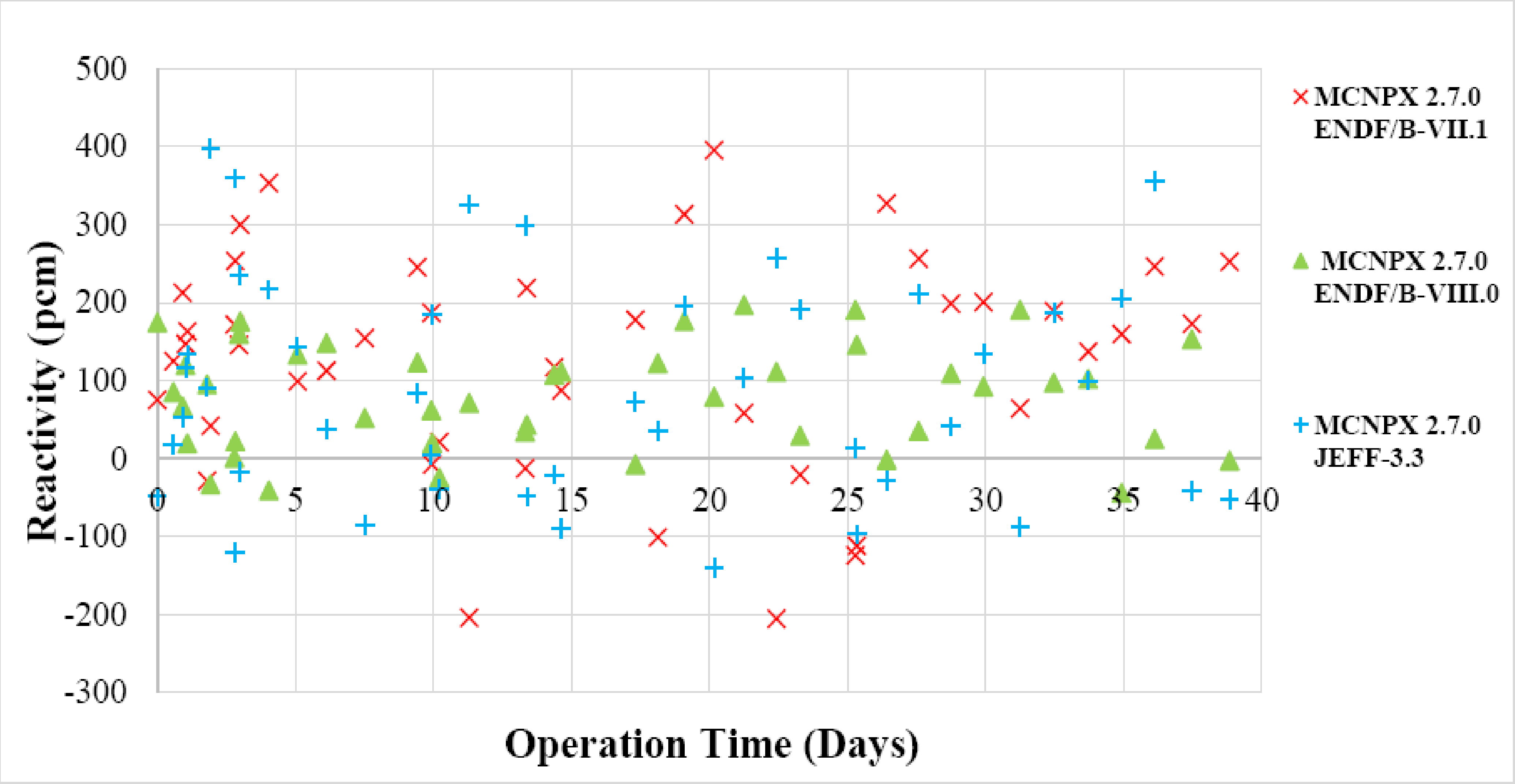


MCNP model of the reactor core

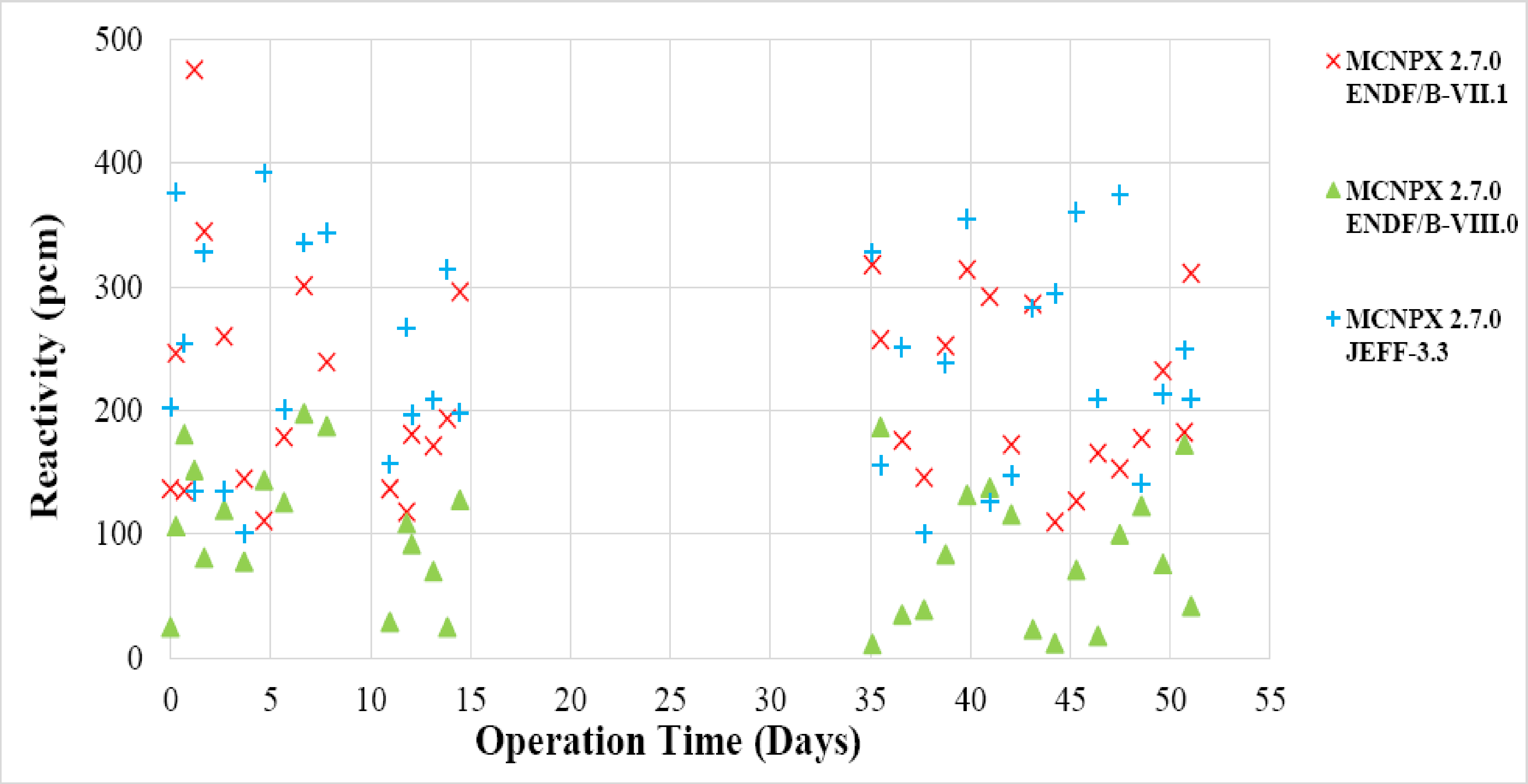
MCNP model of the reactor



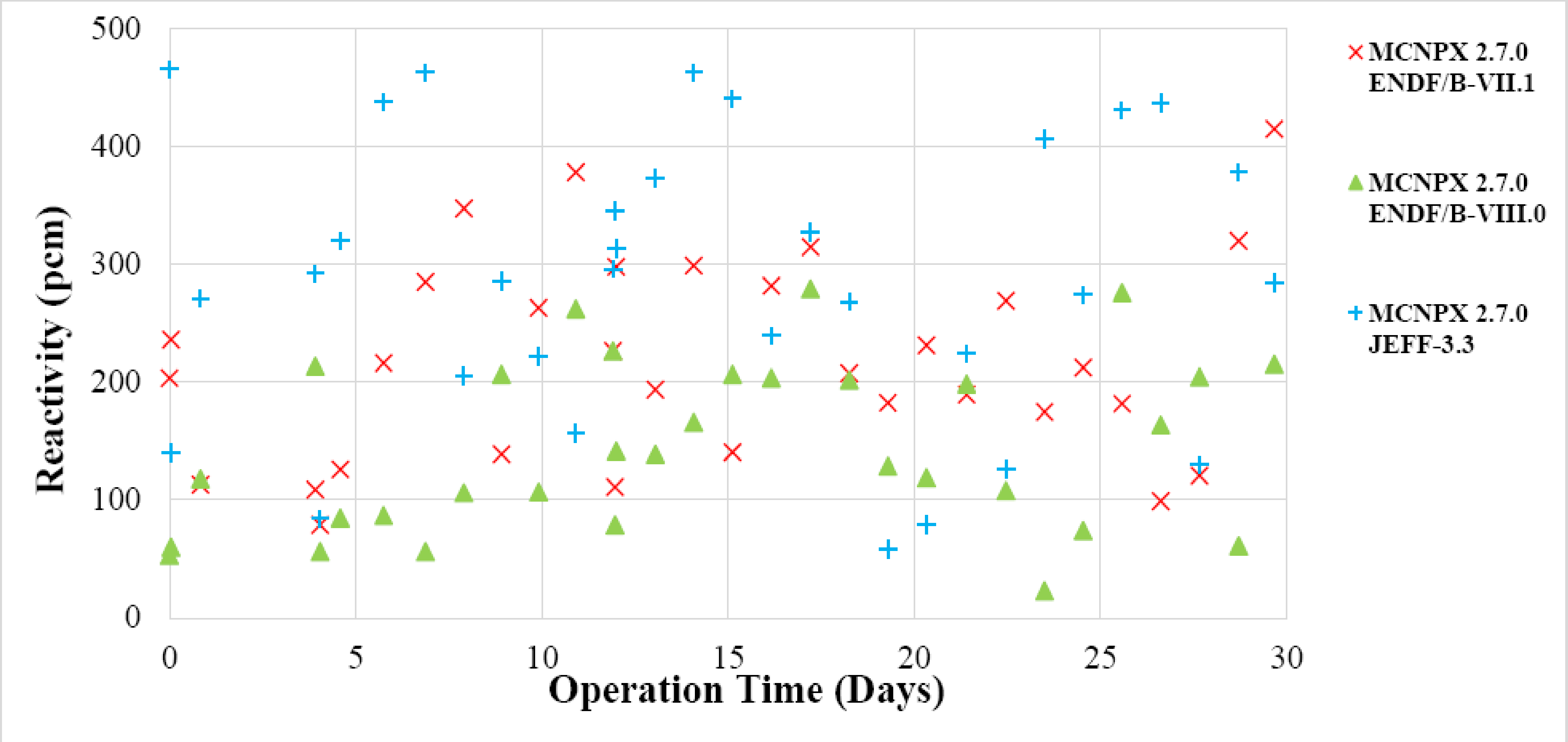
Results: Cycle 07



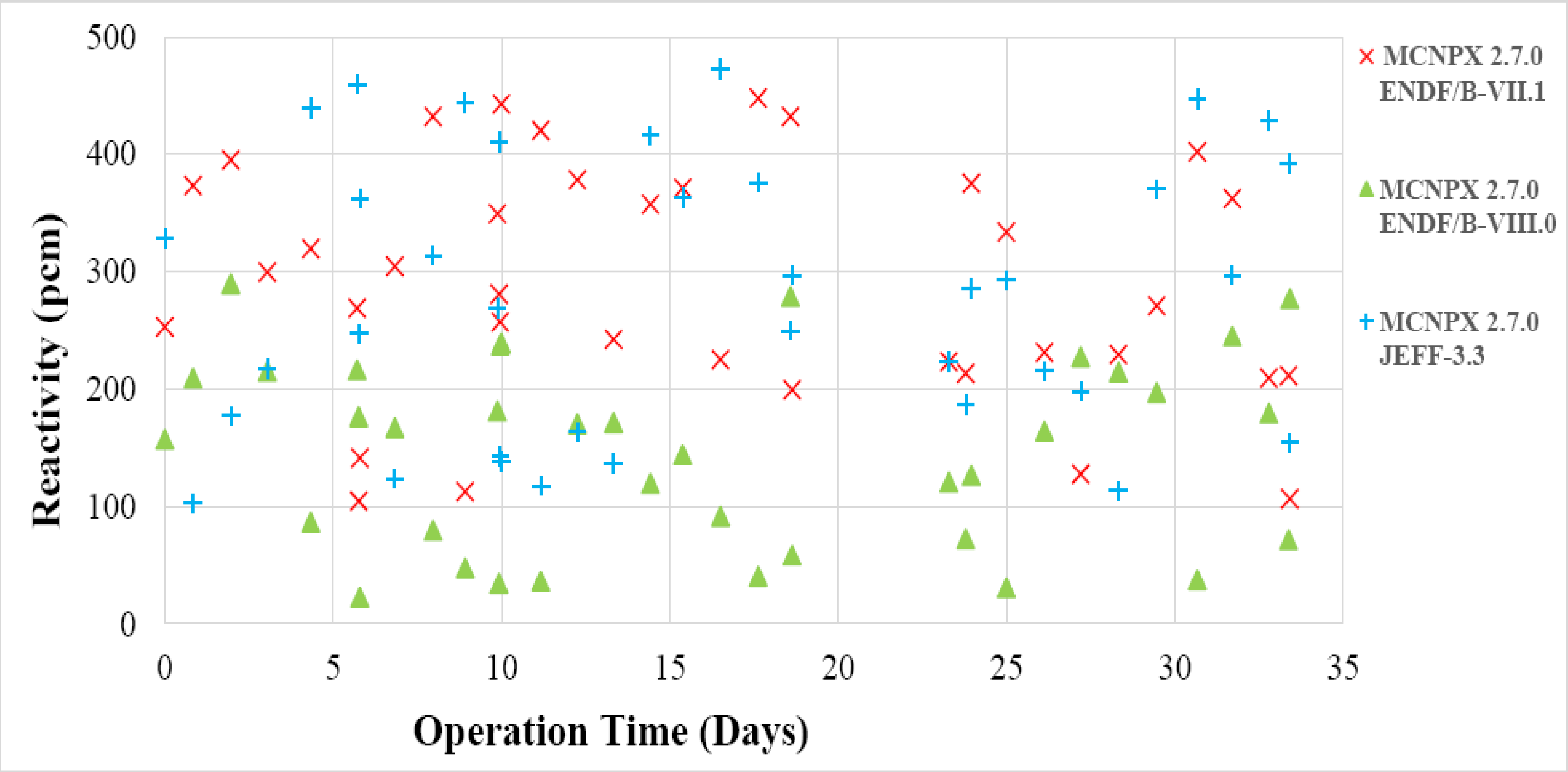
Results: Cycle 08



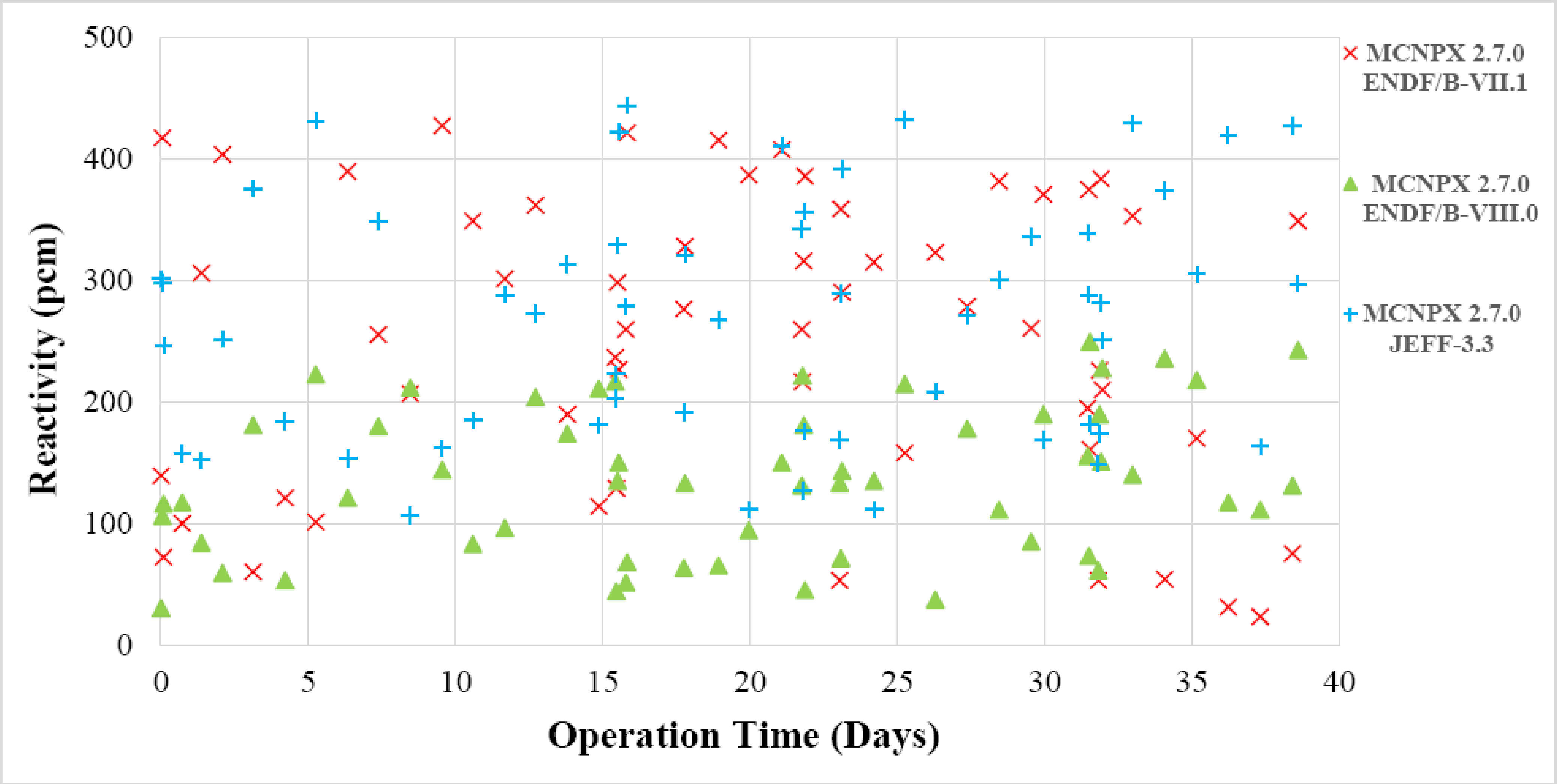
Results: Cycle 09



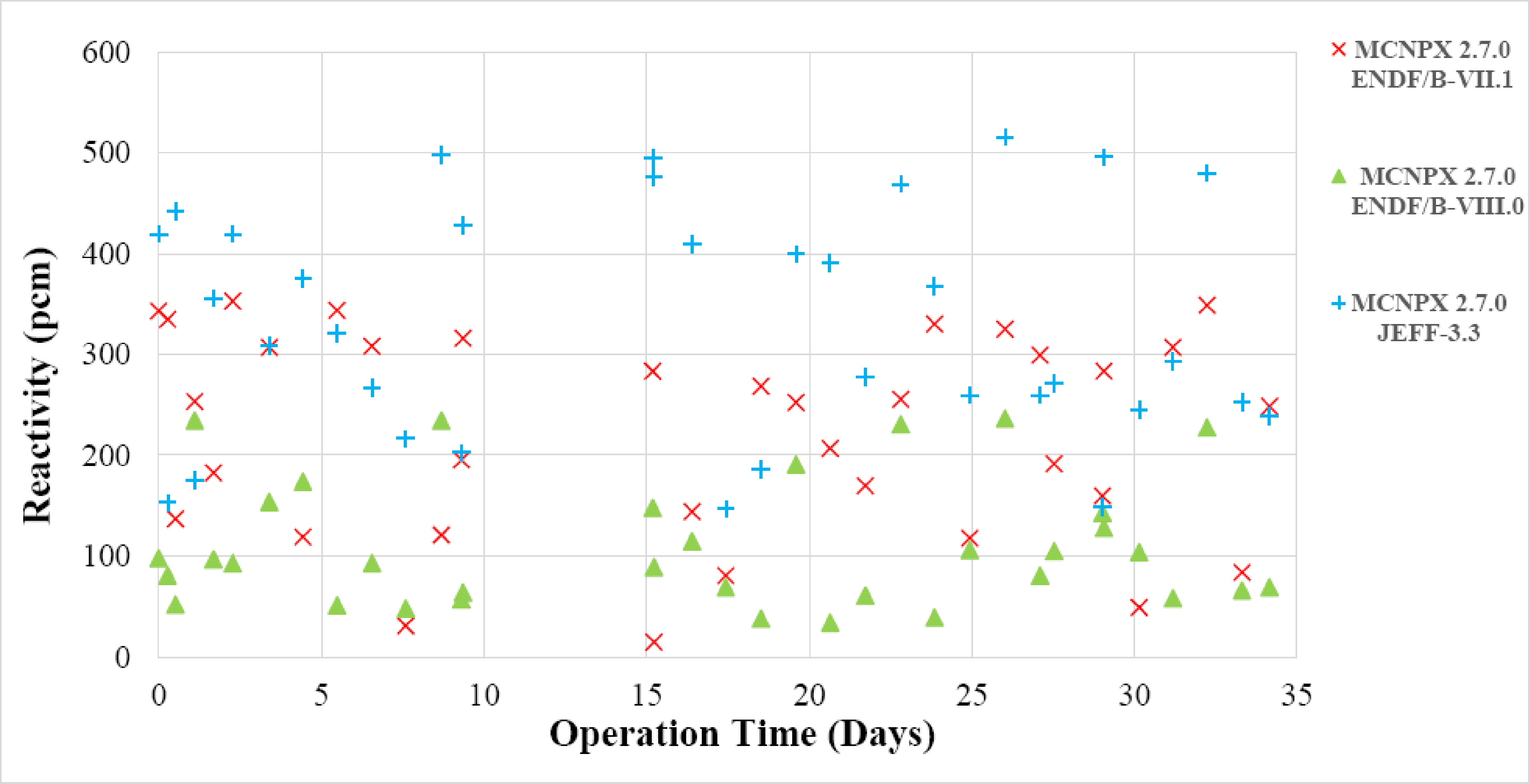
Results: Cycle 10



Results: Cycle 11



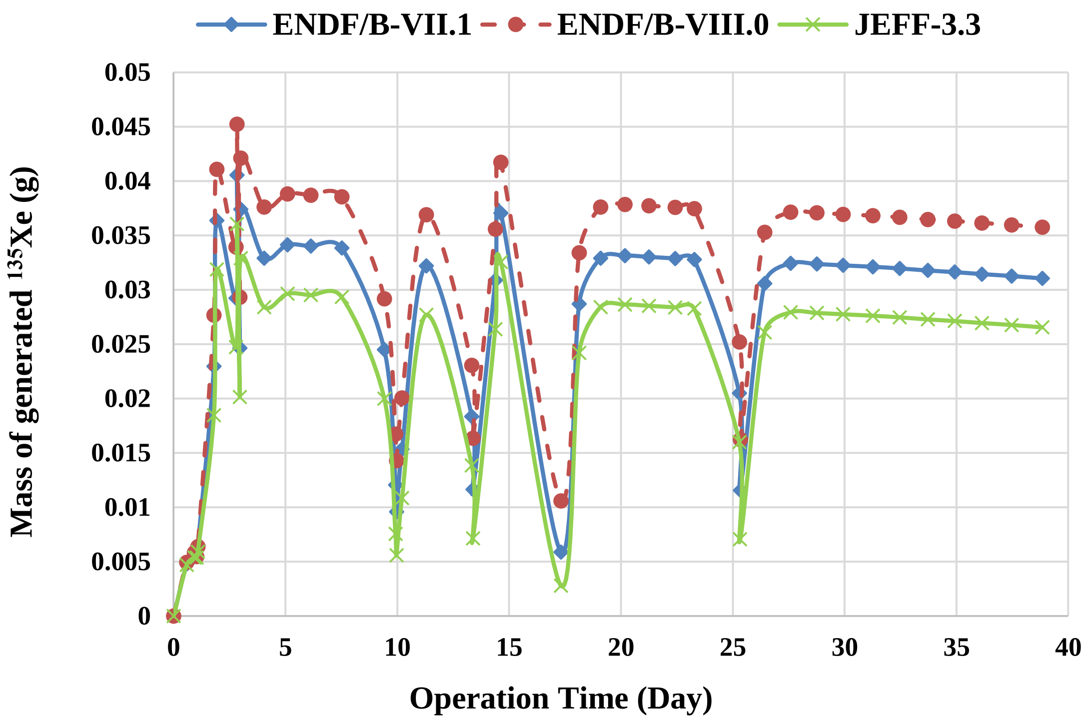
Results: Cycle 12



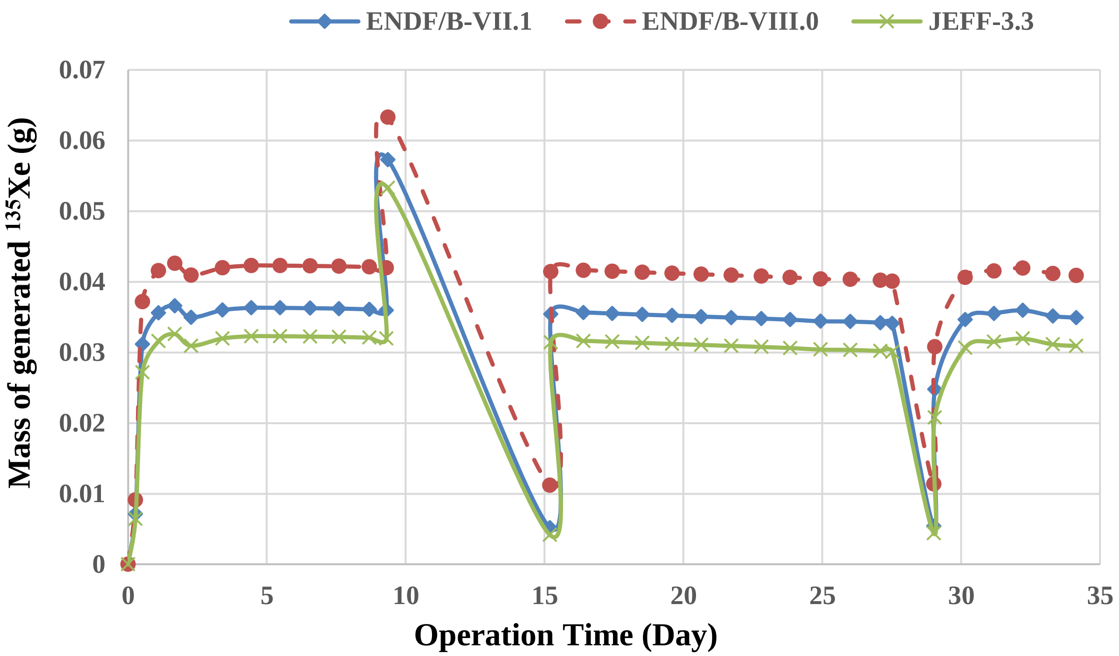
Results: Average discrepancies between calculations and measurements

Cycle No.	Average discrepancy (pcm)		
	ENDF/B-VII.1	ENDF/B-VIII.0	JEFF 3.3
007	120	80	90
008	210	100	240
009	215	150	290
010	290	150	270
011	250	130	270
012	220	110	330
Average	218	120	248

Results: Generated Xe-135

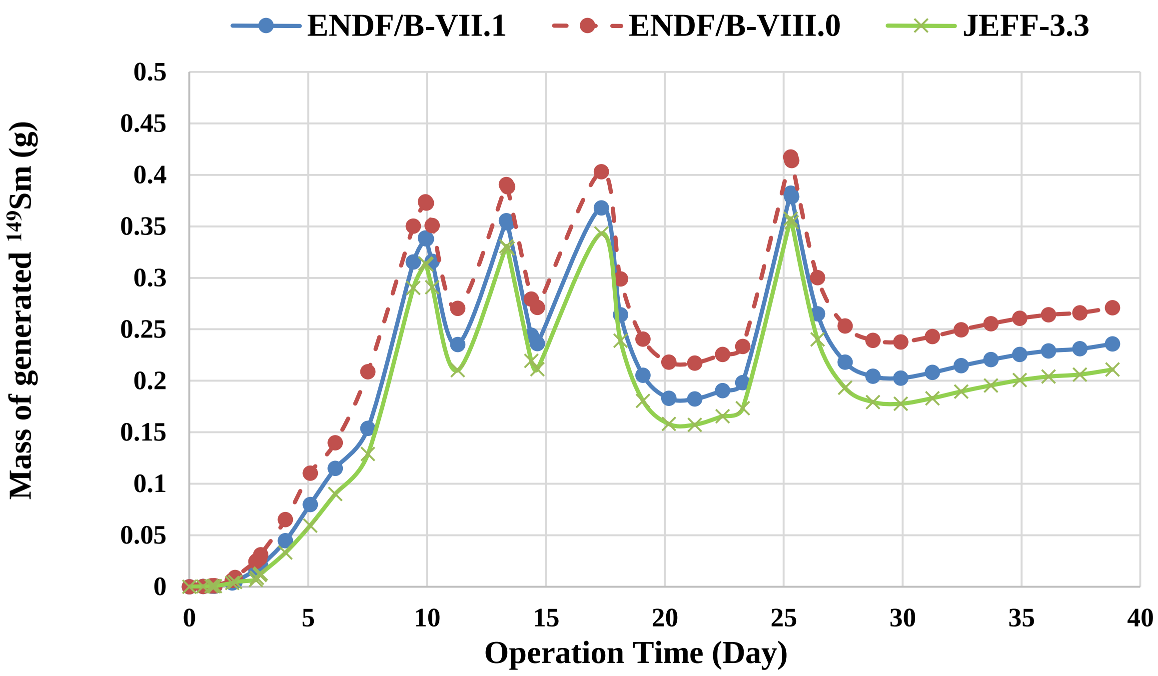


Cycle: 07

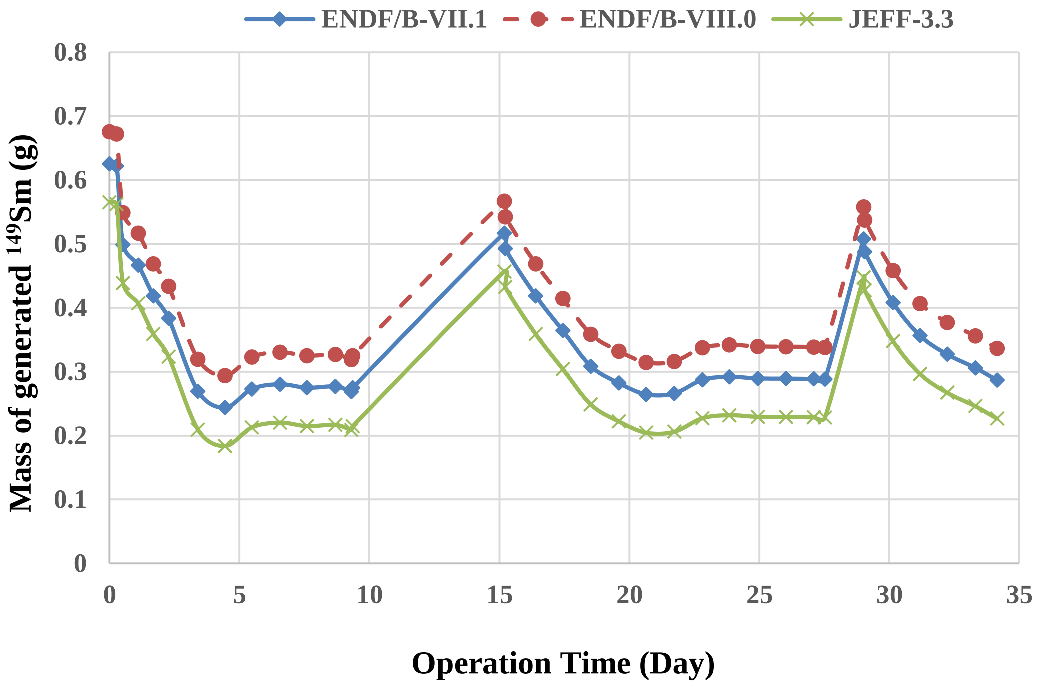


Cycle: 12

Results: Generated Sm-129



Cycle: 07



Cycle: 12

Results: Generated Xe-135 and Sm-149 at the end of core cycles.

Cycle	End of cycle power (MWth)	Mass of generated Xe-135 (g)			Mass of generated Sm-149 (g)		
		ENDF/B-VII.1	ENDF/B-VIII.0	JEFF 3.3	ENDF/B-VII.1	ENDF/B-VIII.0	JEFF 3.3
007	18.70	0.0311	0.0358	0.0266	0.236	0.271	0.211
008	19.79	0.0317	0.0367	0.0267	0.249	0.299	0.221
009	19.58	0.0339	0.0389	0.0289	0.262	0.322	0.232
010	18.21	0.035	0.0400	0.0300	0.275	0.331	0.238
011	18.19	0.0349	0.0409	0.0309	0.280	0.335	0.242
012	18.79	0.0349	0.0409	0.0309	0.287	0.339	0.250

Conclusions

- MCNPX V2.7.0 calculations based on ENDF/B-VII.1, ENDF/B-VIII.0, or JEFF 3.3 data libraries gave good agreements with the measurements of OPAL multi-cycles criticality benchmark in which ENDF/B-VIII.0 library resulted higher accuracy.

- The average deviations between ENDF/B-VIII.0 library and ENDF/B-VII.1 and JEFF 3.3 libraries were around 100 and 130 pcm, respectively.

- From the material inventories of the three data libraries calculations, considerable differences in the concentrations of Xe-135 and Sm-149 (the dominant generated poisons) resulted from the three data libraries were observed.

- ENDF/B-VIII.0 resulted in higher concentrations of the two poisons than the other two libraries while ENDF/B-VII.1 resulted in higher concentrations of the two poisons than that of JEFF 3.3.

Thank you for you attention