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Original Article

Effect of the new ENDF/B-VIII.1 Nuclear Data Library on Criticality Calculations of the DNRR with LEU Fuel

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Abstract: This paper reports the criticality calculations of the Dalat Nuclear Research Reactor (DNRR) with low enriched uranium fuel using the MCNP6.3 code and the newly released ENDF/B-VIII.1 data library. The work aims to evaluate the effect of the ENDF/B-VIII.1 data library on the criticality analysis of the DNRR and compared to previous libraries ENDF/B-VIII.0 and JENDL-5. The calculations were conducted based on ten criticality conditions of the core consisting of 92 LEU fuel bundles, and compared against measurement and with that obtained with previous versions of data libraries. A tendency was observed that ENDF/B-VIII.1 yields higher k_{eff} values compared to ENDF/B-VIII.0, with the largest discrepancy of 150 pcm, but closer to JENDL-5. The deviation compared to the experiments is from -111 to +145 pcm. This discrepancy indicates a generally good agreement between calculations and measurements, and among the three latest libraries.

Keywords: DNRR, LEU fuel, criticality calculation, MCNP6.3, ENDF/B-VIII.1.

1. Introduction

The Dalat Nuclear Research Reactor (DNRR) is a unique research reactor operated by Nuclear Research Institute in Dalat, Vietnam. It has a thermal output of 500 kW. In the early 1980s, the reactor core was upgraded from a TRIGA Mark II design (250 kW) and adopted the Russian VVR-M2 fuel type. From 1983 to 2007, DNRR operated with high enriched uranium (HEU) fuel (U-235 enrichment ~36 wt%). Beginning in 2007 and completed by early 2012, the core was converted to low enriched

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uranium (LEU) fuel (U-235 enrichment ~19.75 wt%) [1-5]. To support the DNRR operation, core physics analyses have used a range of deterministic and Monte Carlo codes for neutronics, burnup, safety analyses, and in core fuel management (ICFM). Notable activities include:

- The WIMSD/CITATION code system was used to evaluate HEU burnup distributions and benchmark against measurements [6]. The code system also contributed to generating a cross section data set for HEU-DNRR, which underpinned subsequently developed ICFM methods, including genetic algorithms, differential evolution, and advanced variants [7].

- Neutronics analyses of the HEU fueled DNRR core with MCNP5 and SRAC showed the effective multiplication factor (k_{eff}) agreeing within 55 pcm between the codes, and within 119 pcm compared to experiments. Relative power density deviations were under 4% in the middle core and about 7% in the outer core [8].

- A comprehensive criticality study was performed for 49 startup phase configurations using SRAC and MCNP5 with multiple nuclear data libraries (ENDF/B and JENDL). Criticality differences reached up to 770 pcm for ENDF/B-VII.0, JENDL-3.3, and JENDL-4.0. For the 88 bundle working core, k_{eff} deviations from experiments were below 330 pcm. Reactivity worth analyses of the automatic regulating rod showed consistency across libraries, with discrepancies under 5%. In most cases, relative discrepancies versus measurements remained below ~12% [9].

- The Serpent 2 code was used to process a cross section set for core physics calculations using PARCS. The PARCS/Serpent model provided a potential applicability for steady state and transient analyses [10].

The operation data of the DNRR with both HEU and LEU fueled configurations were extensively deployed for qualification analysis of nuclear data libraries [3, 9]. A comprehensive study was performed for sensitivity/uncertainty analyses of the HEU fueled DNRR using MCNP6.2 in combination with ENDF/B-VII.1, ENDF/B-VIII.0, JENDL-4.0, and JENDL-5 libraries. Subsequently, criticality and sensitivity/uncertainty analyses were extended for the LEU fueled DNRR using MCNP6.3 with various updated libraries including ENDF/B, JENDL, JEFF, and CENDL [3]. Thirty critical configurations were analyzed and compared against measurements and across libraries. The agreement between calculations and experiments was within 354 pcm, while discrepancies among the libraries themselves were within 172 pcm. The k_{eff} uncertainties arising from the nuclear data libraries ranged from 363 to 588 pcm. These evaluations identified ENDF/B-VIII.0 and JENDL-5 as reliable choices for core physics and safety analyses of the DNRR.

With ongoing international efforts to improve the evaluated nuclear data for reactor analyses, notable sources include ENDF/B, JENDL, JEFF, CENDL, BROND, and TENDL [11-16]. ENDF/B-VIII.0 (released in 2018) provides cross sections for 557 nuclides and features substantial updates for light nuclei, structural materials, actinides, fission energy release, prompt fission neutrons, and thermal scattering data [11]. In 2021, JAEA's Nuclear Data Center released JENDL-5, offering comprehensive evaluated data for 795 nuclides along with a thermal scattering sub library for 37 materials, marking a significant advancement over its previous version. OECD/NEA released JEFF-3.3 in 2017, containing evaluated data for 562 nuclides [12, 13]. CENDL-3.2, issued by the Chinese Evaluated Nuclear Data Library in 2020, includes cross sections for 240 isotopes. Russia released BROND-3.1 in 2016, providing data for 372 nuclides, while TENDL-2015 covers about 2800 isotopes. Recently, the new ENDF/B-VIII.1 library (2024) was released and recommended by CSEWG for nuclear science and technology applications [17]. Notable updates relative to ENDF/B-VIII.0 include a reevaluated Pu-239 data file, revised data for several nuclei (O, F, Si, Cr, Mn, Fe, Cu, La, U, Pu, etc.), new thermal neutron scattering kernels, and updated covariance data. Issues previously identified for nuclear power reactor analyses at high burnup in ENDF/B-VIII.0 have been addressed. However, benchmarking and qualification of the newly released ENDF/B-VIII.1 library for reactor analysis have not been reported.

Thus, it is valuable to assess the performance of the ENDF/B-VIII.1 library through a qualification evaluation based on DNRR core analysis.

In the present study, a preliminary evaluation of the impact of ENDF/B-VIII.1 on the criticality analysis of the DNRR core consisting of 92 LEU fuel bundles was conducted using the MCNP6.3 code [17-19]. Calculations were performed for 10 criticality conditions observed during the startup of the LEU fueled configuration. Comparison with the measurements and with the previous nuclear data libraries was also conducted and presented.

2. The DNRR with LEU Fuel

2.1. Description of the DNRR

Figure 1 illustrates the DNRR core arrangement, which includes 92 LEU fuel bundles, 12 beryllium rods, a central neutron trap, and three irradiation channels. The active region is cylindrical, with a height of 60 cm and a diameter of approximately 44.2 cm. The control rod system comprises a stainless steel automatic regulating rod (AR), four shim rods (ShR), and two boron carbide safety rods (SR). The LEU fuel bundle follows the Russian VVR-M2 design and features three concentric annular tubes: an outer hexagonal tube and two inner cylindrical fuel tubes. The beryllium moderator rod shares the same external dimensions as the fuel assembly. Figure 2 depicts the geometry and design parameters of the LEU fuel assembly. More detailed, comprehensive design parameters of the DNRR core and the LEU bundle can be found in Ref. [1-5].

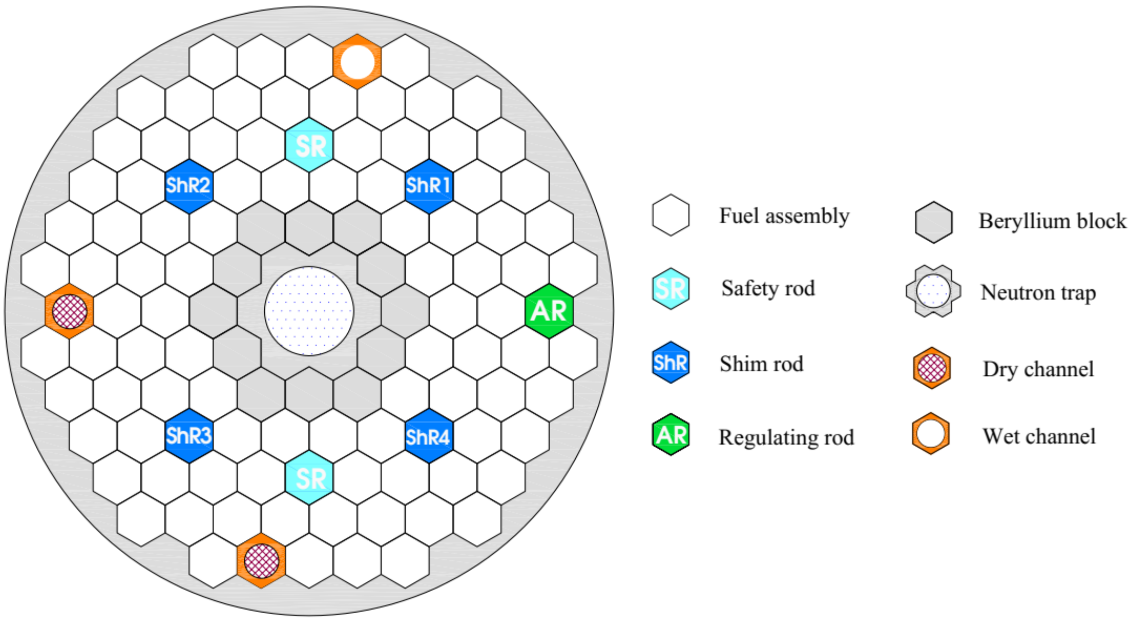

Figure 1. Configuration of the DNRR core consisting of 92 LEU fuel bundles.

The initial LEU fueled configuration of the DNRR consisted of 92 fuel bundles, including 80 fresh and 12 partially burnt elements, with burnup in the approximate range of 1.5–3.5% of U-235 loss. Criticality was achieved by fully withdrawing the two SRs, partially inserting the four ShRs, and adjusting the AR. Reactor power and reactivity changes were monitored with two detectors at beam ports and three fission chambers integrated into the control system. The detectors were mounted in waterproof aluminum housings embedded in the dry channels within the water reflection region between the graphite reflector and the reactor tank. Signals from these detectors were sent to a current neutron display for real time power monitoring. The initial detector readings were recorded. As the AR position was incrementally adjusted, corresponding readings were recorded, respectively. The critical configuration was identified when these ratios approached unity, corresponding to near zero reactivity (i.e., ratios approaching zero).

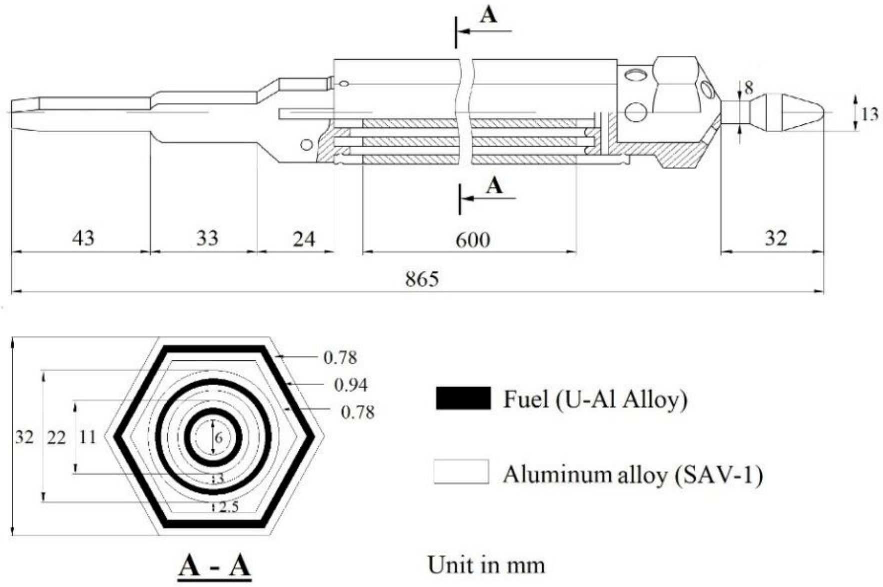

Figure 2. Cross sectional views and design parameters of the VVR-M2 LEU fuel bundle.

Table 1. Summary of the control rod positions (the four ShRs and the AR) of ten criticality conditions of the DNRR. The SR rods were completely withdrawn

Case	ShR1	ShR2	ShR3	ShR4	AR
1	430*	430	430	430	390
2	420	420	420	420	530
3	431	431	431	431	430
4	435	435	435	435	375
5	438	438	438	438	310
6	442	442	442	442	225
7	390	460	460	460	256
8	358	477	477	477	259
9	325	498	498	498	253
10	290	527	527	527	252
*unit in mm.					

Table 1 lists the detailed positions of the four ShRs and the AR corresponding to ten criticality conditions established experimentally. In this work, MCNP6.3 calculations were performed for these criticality conditions using the ENDF/B-VIII.1 library, in comparison with the measurements and the calculation results using the ENDF/B-VIII.0 and JENDL-5 data libraries.

2.2. Core Model Using MCNP6.3

MCNP6.3 represents the latest public version of the MCNP code, enabling precise modeling of a wide range of DNRR core components. To reduce model complexity, homogenization was applied to the upper and lower fuel regions, aluminum structures, beryllium rods, and irradiation channels, following approaches used in prior works. The impact of homogenization on k_{eff} was estimated to be in the range of 10–20 pcm [9]. Figure 3 illustrates cross-sectional views of the core model as implemented in MCNP6.3. For the present calculations, MCNP6.3 was run for 500 cycles, comprising 50 inactive cycles and 1,000,000 neutron histories per cycle, yielding a statistical uncertainty in k_{eff} of approximately 6 pcm.

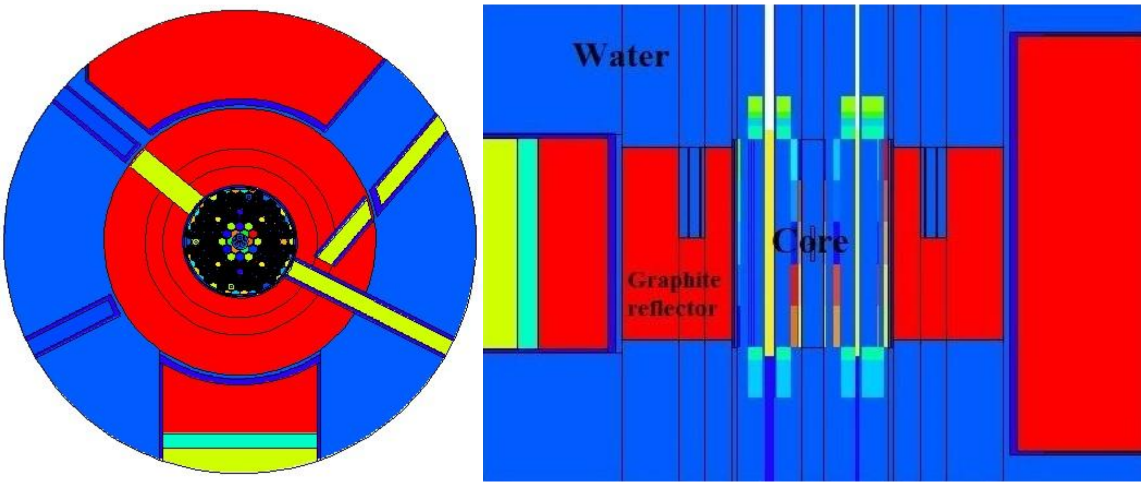

Figure 3. Horizontal (left) and vertical (right) views of the DNRR core model using the MCNP6.3 code.

3. Results and Discussion

The MCNP6.3 calculations were first performed using the ENDF/B-VIII.1 library for three conditions of the LEU fueled DNRR core: critical, supercritical, and subcritical. Table 2 presents the three calculation cases, and the reported uncertainties for the calculated k_{eff} values are 1σ statistical uncertainties. Case 1 corresponds to the experimentally established critical condition, the position of the AR at 390 mm and that of the four ShRs at 430 mm. Case 2 represents the supercritical state with all control rods withdrawn, and Case 3 represents the subcritical state with all control rods fully inserted. In all cases, the ShRs were assumed to be fully withdrawn. Table 2 presents the predicted k_{eff} values for the three cases calculated with MCNP6.3 and ENDF/B-VIII.1, and compares them with the values obtained using the same code with ENDF/B-VIII.0 and JENDL-5 as reported in a previous work [3]. The results are interpreted as below:

- Case 1 (critical): the k_{eff} predicted with ENDF/B-VIII.1 is 1.00145. The value is greater than that obtained with ENDF/B-VIII.0 by about 124 pcm, and is lower than JENDL-5 by 17 pcm.
- Case 2 (supercritical, all CRs out): the k_{eff} value is obtained as 1.07847. The values is about 124 pcm greater than that obtained with ENDF/B-VIII.0, and is about 3 pcm lower than JENDL-5.

- Case 3 (subcritical, all CRs in): the k_{eff} value is 0.97140. The value is 115 pcm greater than that obtained with ENDF/B-VIII.0 and lower than JENDL-5 by 56 cm.

It can be seen from Table 2 that across the three cases, the MCNP6.3 calculations with ENDF/B-VIII.1 provide a good agreement compared to the measurement (for the critical case) and the calculations using other libraries. The largest discrepancy is about 171 pcm. Table 2 also shows that the k_{eff} results obtained with ENDF/B-VIII.1 tend to be greater than that of the previous version ENDF/B-VIII.0 by about 115-124 pcm. This tendency agrees with the values obtained with JENDL-5 library, which reports the k_{eff} greater than ENDF/B-VIII.0 by 123-171 pcm. Thus, the ENDF/B-VIII.1 and JENDL-5 libraries demonstrate an agreement with the discrepancy of 3-56 pcm. Further calculations to evaluate the effect of the ENDF/B-VIII.1 have been conduted based on ten criticality conditions of the DNRR mentioned in Table 1.

Table 2. Calculations of the critical, supercritical and subcritical conditions of the LEU-fueled DNRR

Case	Reactor condition	Configuration	ENDF/B-VIII.0 [3]	JENDL-5 [3]	ENDF/B-VIII.1 This work
1	Critical	Experimental	1.00021 ± 0.00005 -	1.00162 ± 0.00005 141 pcm*	1.00145 ± 0.00004 124 pcm
2	Supercritical	All CRs out	1.07708 ± 0.00005 -	1.07851 ± 0.00005 123 pcm	1.07847 ± 0.00004 120 pcm
3	Subcritical	All CRs in	0.97032 ± 0.00005 -	0.97193 ± 0.00005 171 pcm	0.97140 ± 0.00004 115 pcm
*Reactivity discrepancy compared to ENDF/B-VIII.0.					

Table 3 summarizes the results for ten criticality conditions of the DNRR LEU fueled core. The results obtained with ENDF/B-VIII.0 and JENDL-5 data libraries were taken from the previous work for comparison [3]. Figure 4 displays the comparison of k_{eff} predictions among the three data libraries and compares them with the measurements. One can see that in all cases, the k_{eff} values calculated with ENDF/B-VIII.1 are higher than that obtained with ENDF/B-VIII.0 by about 115-149 cm. The observed deviation from the experiments is within the range of from -111 pcm to 145 pcm. The discrepancy is considerablly a good agreement between calculations and measurements compared to the discrepancies reported in previous works [3]. ENDF/B-VIII.1 tends to yield higher k_{eff} values, with the largest observed discrepancy around 145 pcm. When comparing among the three libraries, the spread in k_{eff} predictions remains under 172 pcm. Specifically, the k_{eff} values obtained with ENDF/B-VIII.1 are matched with that of JENDL-5, with a maximum difference of 50 pcm. The uncertainty in calculations is primarily driven by nuclear data [3]. The effect of homogenization on k_{eff} is estimated in Ref. [9]. The experimental uncertainty was estimated to be less than 1%, and arose mainly from determining criticality conditions, reactivity measurement techniques, uncertainties in control rod positioning, detector noise at low power levels, fission product poisoning, and temperature feedback effects [9].

To better understand the discrepancies of the new library, a review of major revisions of ENDF/B-VIII.1 that could effect the criticality analysis is essential. In ENDF/B-VIII.1, many neutron reaction data files (neutron sublibraries) were majorly updated and newly added compared to ENDF/B-VIII.0 [11, 17]. In which, about one third of the new neutron data were reavaluated within the International Nuclear Data Evaluation Network (INDEN) collaboration. Among these updates, remarkable revisions include a new data file for Pu-239 which was developed with major changes in the fission neutron multiplicity, prompt fission neutron spectrum, resonance and fast regions. New sublibraries of U-233, U-235, U-238 and Pu-240, Pu-241 were also reaveluated. The number of thermal n-scattering sublibraries was increaded to 114 in ENDF/B-VIII.1, compared to 33 of ENDF/B-VIII.0. An important update of Pu-241 neutron fission yield file was made. Additionally, many erroneously large uncertainties in previous version were corrected. Consequently, the criticality analysis of thermal reactors was affected.

Table 3. The k_{eff} calculations of ten criticality conditions of the DNRR core

Case	ENDF/B-VIII.0 [3]	JENDL-5 [3]		ENDF/B-VIII.1	
	k_{eff}	k_{eff}	$\Delta\rho=\rho^{J5}-\rho^{E8.0}$ (pcm)*	k_{eff}	$\Delta\rho=\rho^{E8.1}-\rho^{E8.0}$ (pcm)
1	1.00021	1.00162	141	1.00145	124
2	1.00130	1.00271	140	1.00253	123
3	0.99943	1.00095	152	1.00070	127
4	0.99892	1.00054	162	1.00034	142
5	0.99902	1.00063	161	1.00039	137
6	0.99881	1.00053	172	1.00030	149
7	0.99922	1.00076	154	1.00042	120
8	0.99885	1.00021	136	1.00012	127
9	0.99844	0.99996	152	0.99961	117
10	0.99774	0.99938	164	0.99889	115
*Reactivity discrepancy compared to ENDF/B-VIII.0.					

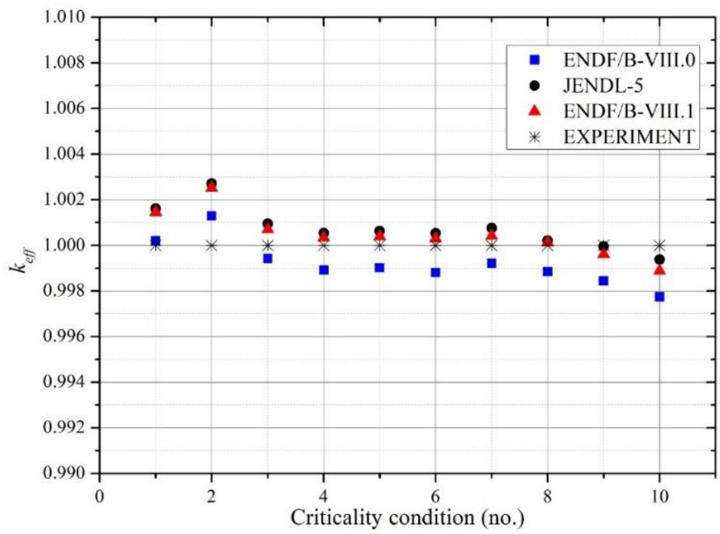

Figure 4. Comparison of the k_{eff} calculations using the ENDF/B-VIII.1 library with other data libraries (ENDF/B-VIII.0 and JENDL-5) and the measurements.

Based on the criticality analyses, it is beneficial to note that the new ENDF/B-VIII.1 provides a high-quality prediction of the k_{eff} for the LEU fueled DNRR core. In comparison, the ENDF/B-VIII.1 results show better agreement with JENDL-5 than with ENDF/B-VIII.0. However, this preliminary assessment of criticality calculations allows only a comparison of differences in a global reactor core parameter, i.e., the k_{eff} value; it is not able to provide further insight into the reasons for these differences or into differences in local parameters. To understand the reasons for the discrepancies between the two versions of the ENDF/B-VIII libraries, a comprehensive sensitivity/uncertainty analysis should be conducted. Thus, a more comprehensive analysis based on the DNRR core configurations for both HEU and LEU fuel cycles is being planned.

4. Conclusions

Criticality calculations were carried out for the DNRR LEU fueled core using MCNP6.3 to evaluate the effect of the new ENDF/B-VIII.1 data library on the criticality analysis. The calculations were conducted based on ten criticality conditions, which were established experimentally during the LEU fuel conversion of the DNRR. The results were compared against the measurements and with those obtained with previous data libraries (ENDF/B-VIII.0 and JENDL-5). There is a tendency for ENDF/B-VIII.1 to yield higher k_{eff} values compared to ENDF/B-VIII.0, with the largest observed discrepancy around 150 pcm, but closer to JENDL-5. The observed deviation from the experiments ranges from -111 to +145 pcm. This discrepancy represents a generally good agreement between calculations and measurements, as well as among the three latest libraries. Further analysis is planned to investigate the uncertainties of the new data library for the DNRR core physics analysis.

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