Oral presentation | III. Fission Energy Engineering：301-1 Reactor Physics, Utilization of Nuclear Data, Criticality Safety

Thu. Mar 13, 2025 11:20 AM - 11:55 AM JST | Thu. Mar 13, 2025 2:20 AM - 2:55 AM UTC Room C(Zoom room 3)

[2C07-08] JENDL-5 Validation

Chair:Irwan Simanullang(Kyushu Univ.)

11:20 AM - 11:35 AM JST | 2:20 AM - 2:35 AM UTC

[2C07]

Effect of JENDL-5 nuclear data library on criticality and fission rate distributions of full MOX BWR core mockup experiments FUBILA

*Toru Yamamoto[1] (1. No affiliation)

11:35 AM - 11:50 AM JST | 2:35 AM - 2:50 AM UTC

[2C08]

Evaluation of impact of O-in-UO2 Thermal Scattering Law (TSL) on BWR depletion calculations with MCNP6.3

*Shinichi Namizono[1], Michitaka Ono[1], Masayuki Tojo[1] (1. GNF-J)

11:50 AM - 11:55 AM JST | 2:50 AM - 2:55 AM UTC

Time reserved for Chair

Effect of JENDL-5 nuclear data library on criticality and fission rate distributions of full MOX BWR core mockup experiments FUBILA

*Toru YAMAMOTO

Using the MVP3 Monte Carlo code with the JENDL-5 base neutron library, analysis of criticality and fission rate distributions was performed for the full MOX BWR mockup cores in the FUBILA program. The analysis results were compared with those with JENDL-4.0. As a result, the trend in k_{eff}s with the numbers of the 7.0 wt% Put MOX fuel rods in the driver region was improved while the updated thermal neutron scattering law (TSL) of water in JENDL-5 brought worse C/Es in comparing the calculated core radial fission rate distributions with the measurements.

Keywords: EOLE, FUBILA, Full MOX BWR mockup core, MVP3, JENDL-5, JENDL-4.0, Neutron multiplication factor, Fission rate distribution, Thermal neutron scattering law.

1. Introduction. To extend the validation data of JENDL-5 [1], analysis of criticality and fission rate distributions was performed with MVP3 [2] for the nine full MOX BWR mockup cores in the FUBILA program [3,4] and the k_{eff}s and fission rate distributions were compared with those with JENDL-4.0 [5].

2. Full MOX BWR mockup cores. The references [3,4] reported the detailed specifications of the full MOX BWR mockup cores in the FUBILA program implemented in the EOLE critical facility. The test regions of the mockup cores were composed of four 9x9 full MOX assemblies in a hot operating condition with a 0% void simulation (9x9Ref), those with a 40% void (40v), those with a 70% void (70v), those with an axially distributing void (axial v), those partially loaded with UO_2 fuel rods (UO_2), those loaded with Gd_2O_3-UO_2 fuel rods (Gd), those loaded with a B_4C control blade (B_4C), four 10x10 full MOX assemblies (10x10), and the 513-day-elapsed 9x9ref core (9x9Ref-te). The driver region was composed of the 7.0 wt% Put MOX rods with a ^{241}Am content of 6 wt% in Pu+Am.

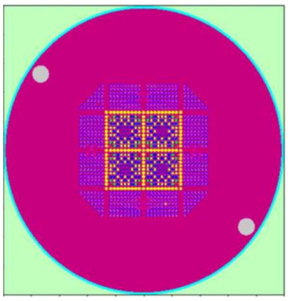

Fig. 1 Radial plane of 9x9ref core

3. Analysis models and calculation conditions. The calculations modeled the inside of the core tank wall. **Fig.1** illustrates a radial plane of the 9x9ref core. The k_{eff}s and fission rate distributions were calculated with 40-million and 100-million neutron histories, respectively.

4. Analysis results and discussions. The calculated k_{eff}s of the cores with JENDL-5 and JENDL-4.0 are illustrated in **Fig. 2**. The horizontal axis is the numbers of the 7.0 wt% Put MOX rods in the driver region. The effects of the updated cross-sections of main nuclides on the k_{eff}s of the 9x9ref core were analyzed by replacing the JENDL-5 cross-sections with the JENDL-4.0 cross-sections. They were 497 pcm for H in H_2O, −208 pcm for ^{16}O, −25 pcm for ^{235}U, 12 pcm for ^{238}U, −158 pcm for ^{238}Pu, −173 pcm for ^{239}Pu, 14 pcm for ^{240}Pu, −42 pcm for ^{241}Pu, −50 pcm for ^{242}Pu, and −235 pcm for ^{241}Am. Their calculation errors were 17 pcm. As seen in Fig. 2, the trend in k_{eff}s with the numbers of the 7.0 wt% Put MOX fuel rods in the driver region was improved by the updated cross-section of ^{241}Am in JENDL-5. The C/E−1s of the fission rate distributions with JENDL-5 were compared with those with JENDL-4.0. **Fig. 3** shows the differences in the C/E−1s (%) with JENDL-5 and those with JENDL-4.0 for the left-bottom 9x9 assembly in the 9x9ref core.

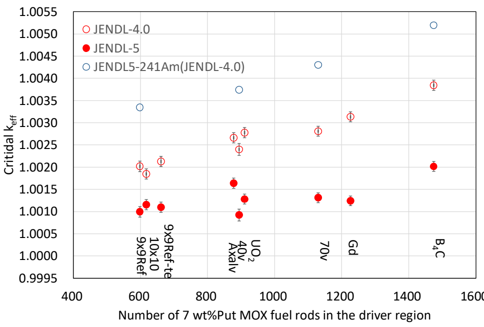

Fig. 2 Comparison of k_{eff}s with JENDL-5 and JENDL-4.0.

The calculated fission rates with JENDL-5 were smaller for the MOX rods adjoining the water gaps and larger for those not adjoining the water gaps than those with JENDL-4.0, which brought worse C/Es with JENDL-5. This trend was the same for the other cores. An additional calculation by replacing the cross-sections of H_2O with those in JENDL-4.0 indicated that the main cause was the updated TSL of H_2O in JENDL-5.

−1.266	0.078	−0.401	−0.867	−0.206	−0.325	0.048	−0.748	−1.693
−0.482	0.215	0.609	1.173	0.749	1.032	1.078	1.177	−0.977
−0.036	1.322	0.674	0.568	−0.298	−0.099	1.178	1.178	−0.209
−0.281	1.073	−0.640				−1.332	0.064	−0.153
−0.174	0.578	−0.549				−0.587	0.395	−0.168
0.315	0.976	−0.133				−0.387	1.938	0.182
−0.353	1.400	0.554	−0.187	0.079	0.108	0.421	0.755	0.220
−0.946	−0.578	0.761	0.473	−0.030	0.548	0.553	0.812	−0.434
−1.407	−0.334	−0.716	−0.345	−0.088	0.850	0.285	−0.818	−1.906

Fig. 3 Comparison of C/E−1s (%) of fission rate distributions for 9x9ref core (JENDL-5−JENDL-4.0).

References. [1] Iwamoto O, Iwamoto N, Kunieda S, et al. Japanese evaluated nuclear data library version 5: JENDL-5. J Nucl Sci Technol. 2023;60:1-60. [2] Nagaya Y, Okumura K, Mori T, et al. MVP/GMVP II: general purpose Monte Carlo codes for neutron and photon transport calculations based on continuous-energy and multigroup methods. JAERI-1348 (2005). [3] Yamamoto T, Sakai T, Ando Y. et al. Neutronics analysis of full MOX BWR core simulation experiments FUBILA, J Nucl Sci Technol. 2011;48:398-420. [4] Yamamoto T, Sakai T, Ando Y. Neutronics analysis of full MOX BWR core simulation experiments FUBILA: part 2. J Nucl Sci Technol. 2012;49:103-120. [4] Shibata K, Iwamoto O, Nakagawa T, et al. JENDL-4.0: a new library for nuclear science and engineering. J Nucl Sci Technol. 2011;48:1-30.

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MCNP6.3 を用いた沸騰水型軽水炉燃焼計算における O-in-UO_2 の熱中性子散乱則 (TSL)の影響の評価

Evaluation of impact of O-in-UO_2 Thermal Scattering Law (TSL) on BWR depletion calculations with MCNP6.3

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抄録　FRENDY コードを用いて評価済み核データである JENDL-5.0 または ENDF/B-VIII.0 に基づいた MCNP 用の ACE ファイルを作成し、BWR の代表的な集合体計算で O-in-UO_2 の熱中性子散乱則の影響を評価した。結果として、無限増倍率の差は 0.25% 、可燃性吸収材である Gd の数密度の差は 8% となった。

キーワード：沸騰水型軽水炉核計算、燃焼計算、熱中性子散乱則、集合体計算、O-in-UO_2、JENDL-5.0、FRENDY

1. 緒言　核データ処理コード FRENDY、連続エネルギーモンテカルロ計算コード MCNP6.3、核データとして JENDL-5.0 (J5) と ENDF/B-VIII.0 (E8) を用いて O-in-UO_2 熱中性子散乱則が BWR 集合体の燃焼計算に及ぼす影響について検討した。調査したパラメータは、BOL 計算における無限増倍率、ボイド係数と感度係数（摂動理論)、燃焼計算における無限増倍率と Gd 数密度などである。NEA BWR Phase III-C ベンチマークに対する結果を以下に示す[1]。

2. 結果

2-1. BOL 計算　無限増倍率の結果を**表 1** に示す。BOL におけるこの差は Gd の吸収反応率が関係しており、後述する燃焼計算に対しては、初期に添加した Gd が燃え尽きるまで両者の差をさらに拡大させるような影響を及ぼすことが予想できる。

表 1 ：BOL 無限増倍率の結果

	O-in-UO2 非考慮	O-in-UO2 考慮	差
J5	1.009491 ± 0.000199	1.010035 ± 0.000196	+0.054%
E8	1.007535 ± 0.000199	1.007611 ± 0.000192	+0.0075%

2-2. 燃焼計算　結果を**図 1** に示す。BOL の結果から予想されたように O-in-UO_2 の利用により Gd の燃焼が早くなり、基準ケース（O-in-UO_2 非考慮）との Gd 数密度の差は最大 8% となった。これが無限増倍率に影響し、Gd が燃え尽きるタイミングで無限増倍率として最大 0.25%の差が確認された。

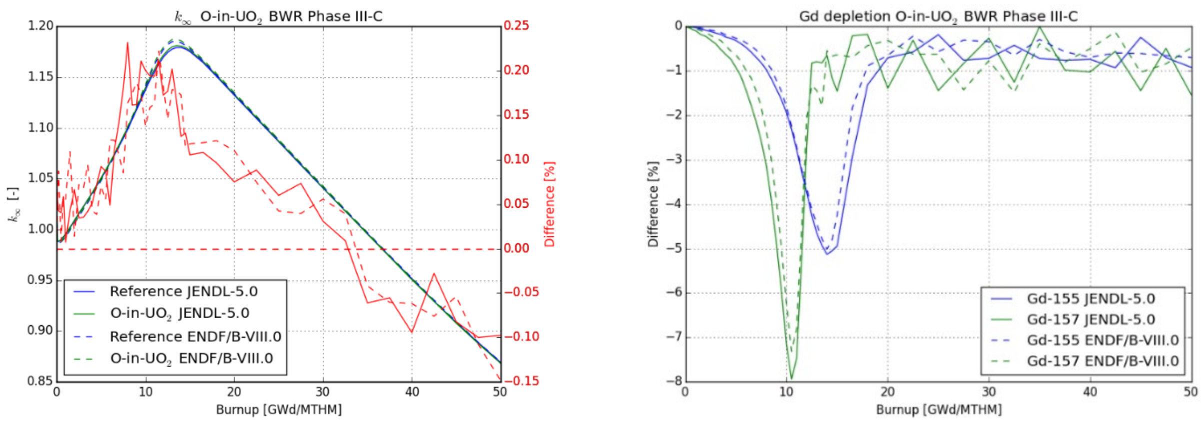

図 1 ：MCNP6.3 による燃焼計算の結果：無限増倍率（左)、Gd の数密度（右）

3. 結論　燃焼モンテカルロ計算を用いて BWR 集合体の燃焼計算における O-in-UO_2 の熱中性子散乱則の影響を評価した。その結果、無限増倍率として最大 0.25%、Gd 数密度で最大 8%程度の差が確認できた。

参考文献

[1] Nuclear Science Committee, “Burn-up Credit Criticality Safety Benchmark Phase III-C”, NEA/NSC/R (2015)6 (2016)

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