Comparison of ENDF/B-VII.1 and JEFF-3.2 in VVER-1000 operational data calculation

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Content

Introduction to Neutronic Calculations

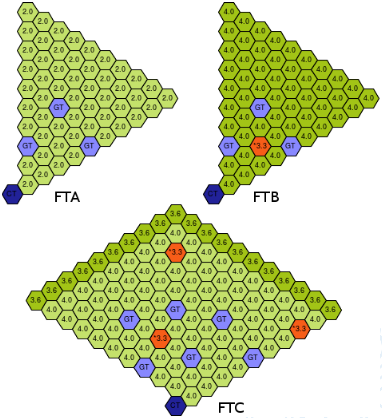
- Full-Core calculations of reactor neutronics and thermal-hydraulic characteristics is essential for safe operation
- The typical sequence of neutronic calculations consists of macroscopic data generation by microcodes and subsequent full-core calculations by macrocodes
- Macroscopic data are generated for fuel assemblies for typical range of operational characteristics
- Microcodes can utilize deterministic or Monte-Carlo methods
- Macrocodes are deterministic and they typically use diffusion method for the solution
- All calculations are affected by nuclear data uncertainty
- Full-core calculation of Czech VVER-1000 reactors are realized by ANDREA diffusion code

VVER-1000 Core Description

- VVER-1000 nuclear reactor belongs in the group of Pressurized-Water Reactors
- Its core consists of 163 hexagonal fuel assemblies with UO_2 enrichment below 5 %
- Each fuel assembly contains 312 fuel pins, 18 guide tubes, and one central tube
- Initial excess of reactivity is reduced by Gd_2O_3 burnable absorber included in fuel mixture of usually 12 or 18 fuel pins
- Core reactivity is controlled in long-term by boric acid in moderator, its concentration is expressed in g/kg

VVER-1000 Fuel Design

- Three VVER-1000 fuel assembly designs were selected for this analysis
- They cover typical range of fuel types:
  - FTA - no burnable absorber
  - FTB - burnable absorber but no enrichment profiling
  - FTC - burnable absorber and enrichment profiling
- Schematics of these fuel types are shown in figure
- 60° or 120° symmetry is utilized to show only uniqe fuel parts



Serpent and ANDREA Calculation Model

- Nuclear data for ANDREA code are by default prepared by detailed HELIOS lattice calculations
- These calculations feature elaborated fuel temperature model respecting radial temperature profile in fuel pins
- There are multiple branch-off calculations to cover changes to nominal operating conditions
- Fuel symmetry helps to reduce calculation complexity
- Generation of required macro data can be alternatively achieved by Serpent Monte-Carlo code
- Due to calculation time constraints Serpent calculates depletion depletion at average fuel characteristics, also radial temperature profile is omitted

Nuclear Data for Calculations

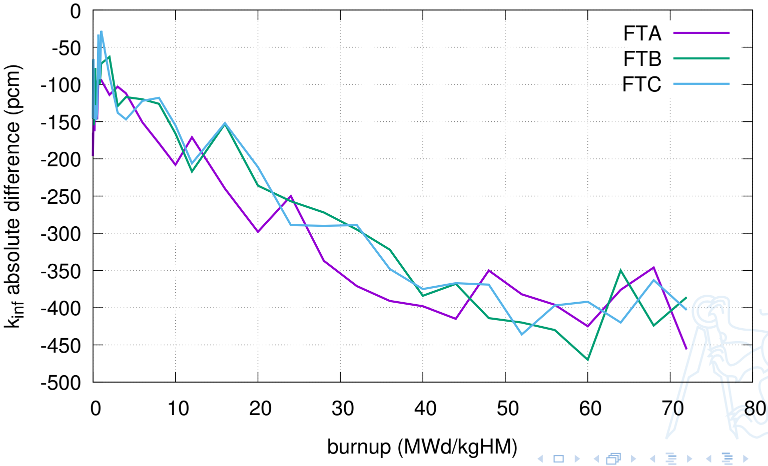
- By default ANDREA uses two-group data
- Except for usual set of macroscopic characteristics (diffusion coefficient, XS, fission yields, energy and neutron production) several microscopic data and number densities are calculated
- Cross-sections and number densities for these nuclides are treated in detail: uranium, plutonium, gadolinium isotopes and isotopes in ^{135}Xe and ^{149}Sm chains
- Calculated data are parametrized by reactor power, fuel temperature, moderator temperature, or by combination of these parameters
- HELIOS utilizes ENDF/B-VII.1 nuclear data, the same is true for Serpent code
- Serpent allows easy switching among various data sets
- Data in this analysis were calculated using ENDF/B-VII.1 and JEFF-3.2 nuclear data

Compared Characteristics

- The comparison is based both on direct macroscopic data and resulting full-core characteristics
- All results are presented by either absolute or relative differences of results obtained by JEFF-3.2 data and ENDF/B-VII.1 data
- No differences from operational data are evaluated here, but better agreement with operational characteristics is achieved with ENDF/B-VII.1 data
- Such a comparison helps to understand the level of uncertainty in the nuclear data and how they influence full-core calculations

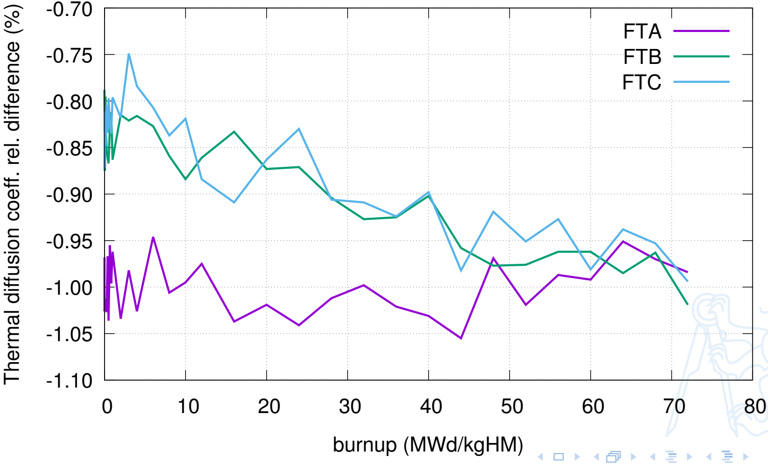
Infinite Multiplication Data

- There are absolute differences increasing during fuel depletion to up to 500 pcm



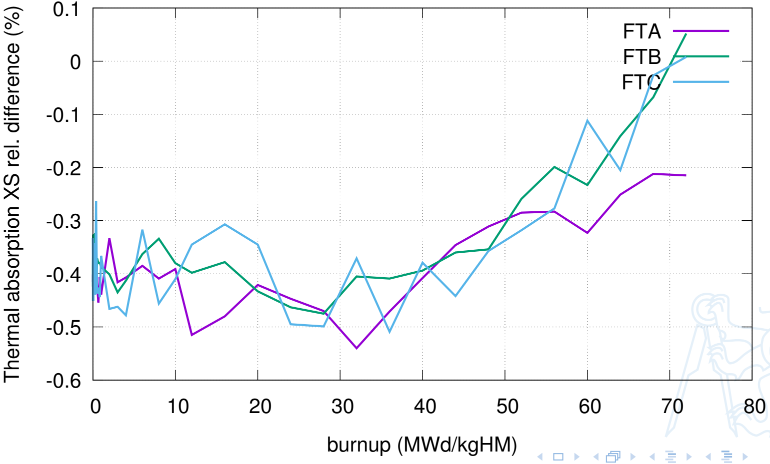
Thermal Diffusion Coefficient

- Results for JEFF-3.2 data are about 1 % lower and effect of Gd can be observed



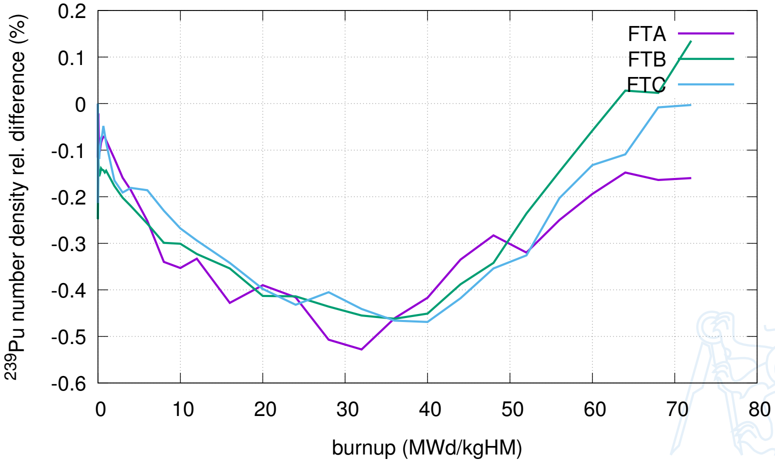
Macroscopic Thermal Absorption XS

- The cross-sections are lower for JEFF-3.2 data regardless of fuel type



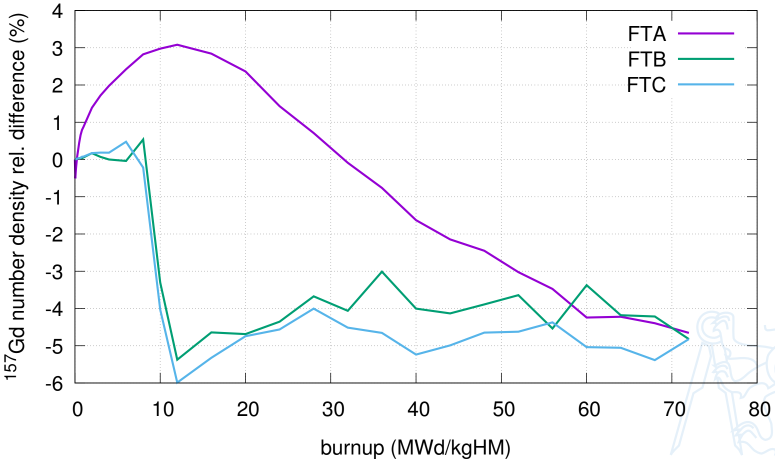
Number Density of ^{239}Pu

- Production and depletion of ^{239}Pu is different for both data sets



Number Density of ^{157}Gd

- Data for both production and depletion of ^{157}Gd are different

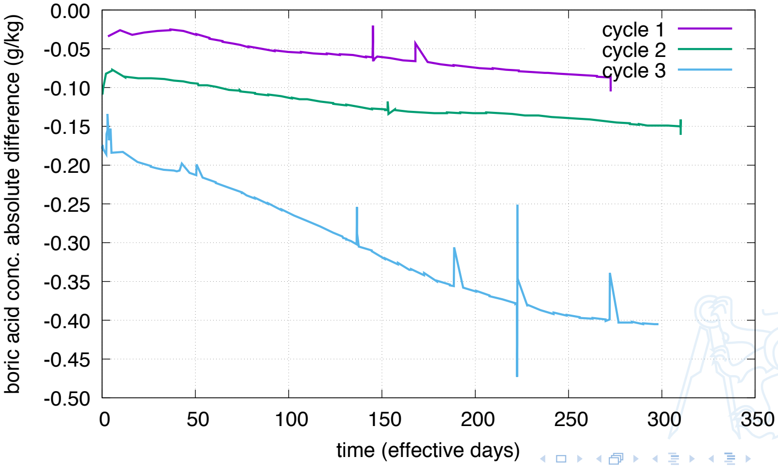


Comparison of Full-Core Characteristics

- The determined macroscopic data were applied in full-core calculations by ANDREA code
- Calculations were conducted for three cycles of first unit of Temelin NPP with the current TVSA-T fuel
- Real operational data were used, thus the data were tested for different conditions including power changes, rods movements, and stretch-out effects
- Several characteristics were selected for comparison: critical boric acid concentration, core averaged Gd concentration, and delayed neutrons fraction
- Illustrated are also average differences in calculated power distributions

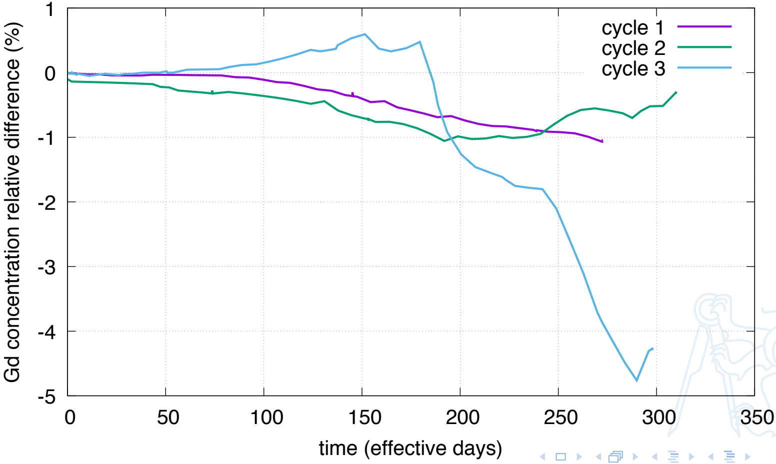
Boric Acid Concentration

- Absolute difference of boric acid concentration was calculated and JEFF-3.2 leads to lower boric acid critical concentration



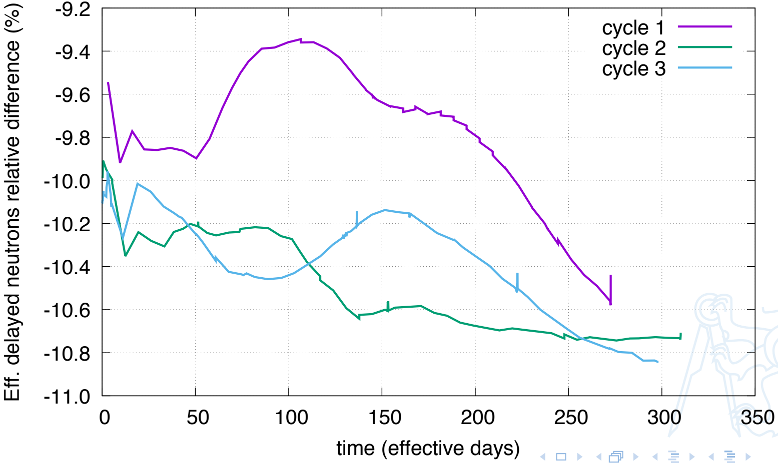
Full-Core Gadolinium Concentration

- Relative difference of Gd concentration was calculated and it is generally lower for JEFF-3.2 data

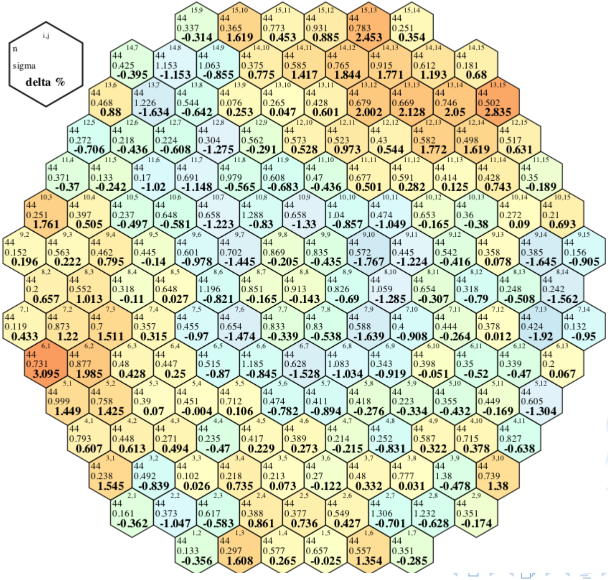


Effect Delayed Neutrons Fraction

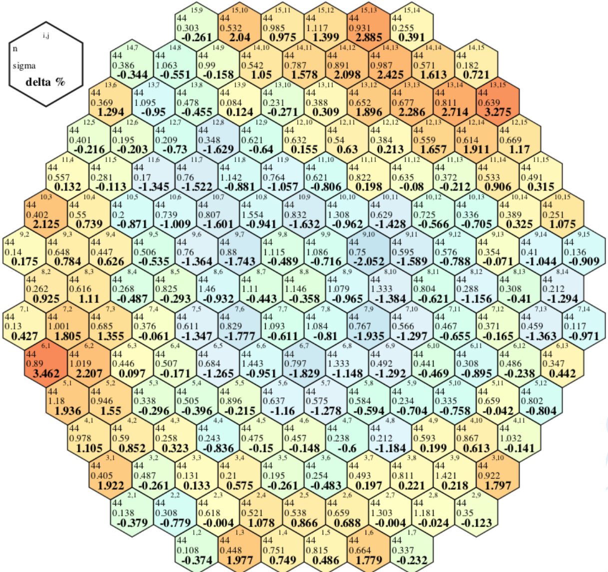
- There are fundamental differences in delayed neutrons fractions that can influence core kinetics characteristics



Core Power Distribution Calculated with ENDF/B-VII.1



Core Power Distribution Calculated with JEFF-3.2



Conclusions

- The same set of macroscopic data calculations and full-core calculations were conducted with ENDF/B-VII.1 and JEFF-3.2 data
- Absolute or relative differences of selected characteristics were determined
- Substantial differences in the calculated quantities were found
- It was determined that difference of several tenths of g/kg of boric acid concentration can be explained by effect of nuclear data
- This study helped to understand importance of nuclear data for full-core calculations and the level of uncertainty coming from nuclear data

Thank you for your attention

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