Analysis of a PWR-type SMR using SEANAP system

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INGENIA-NUCLEAR 2021/22: Scope

- INGENIA/NUCLEAR-2021/22 is focused on:
  - **Neutronic Design of Small Modular Reactors PWR-type: NuScale – 160 MWth**
- Simulation PWR Core Analysis: **System "SEANAP"** ⇒ **"Updated" SEANAP to SMRs**

- **PWR Core Analysis System "SEANAP"**
  - **MARIA** system for lattice/assembly calculations (based on WIMSD5 code)
  - **COBAYA** system for a detailed (pin-by-pin) core calculations
  - **SIMULA** system for 3D/1-group corrected-nodal diffusion core simulation

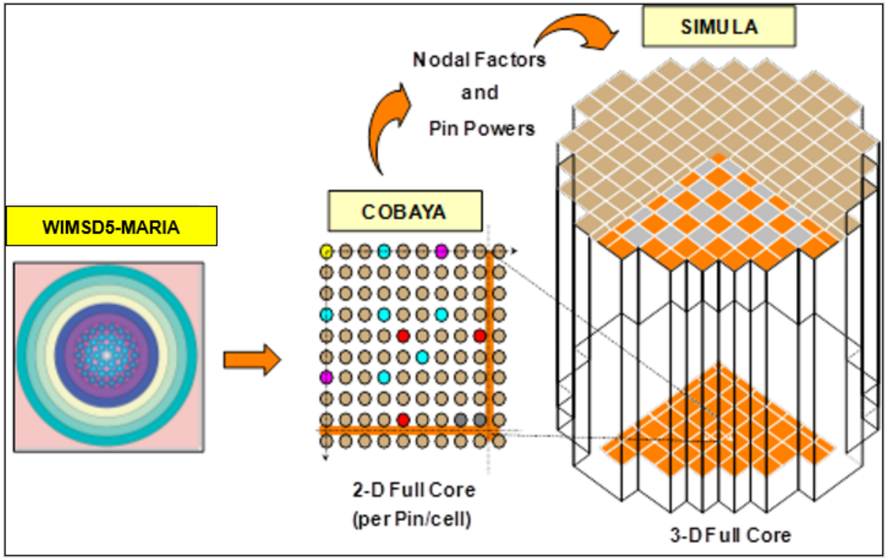


Figure. SEANAP: WIMS-D5 + COBAYA + SIMULA

References:

- *"Validation of PWR Core Analysis system SEANAP-86 with measurements in test and operation", C. Ahnert et al., M&C87*

See also, *"*upgraded SEANAP" with WIMSD5 + recent nuclear data evaluations

- **JEFDOC-1917, April 2018**
- **JEFDOC-1968, April 2019**

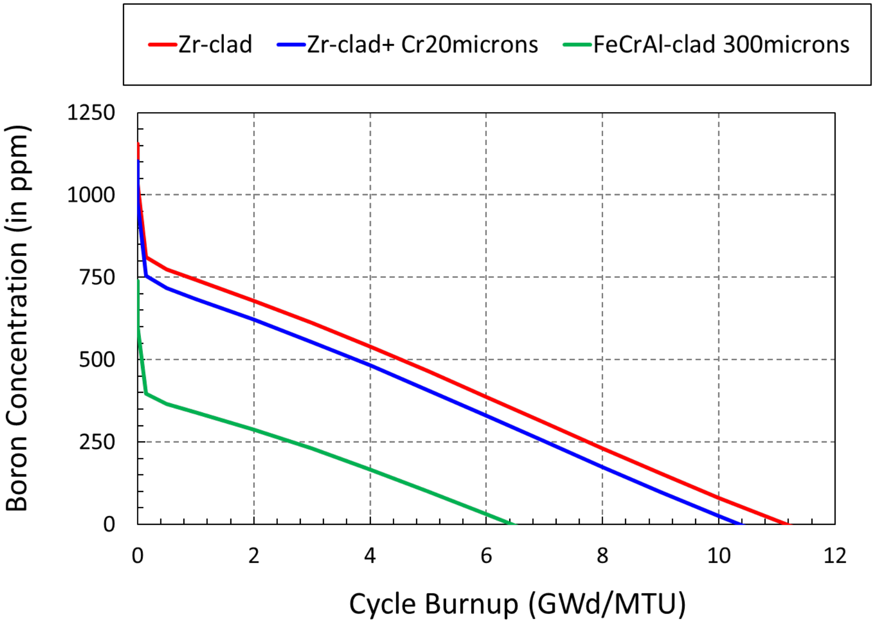
1. Testing ATFs
- cladding materials

☐ ATF – Accident Tolerant Fuels - Cladding Materials /INGENIA 2019/2020

https://ceiden.com/programas/grupo-siren-simulacion-de-reactores-nucleares/segunda-jornada-ceiden-upm-accident-tolerant-fuels-for-lwrs/

- ○ **Zr+Coatings:** Cr-20 microns coating, $\rho_{Cr} = 7.15\ g/cm^3$
- ○ **FeCrAl- clad**: Fe 80.80 wt%, Cr 13 wt%, Al 6.20 wt%, clad thickness: 300μm

☐ Boron letdown



☐ Simulations with SEANAP

- ○ Changes in clad only in fuel rods
- ○ NuSCALE core
- ○ Nuclear Data: ENDF/B-VII.1

1. Testing ATFs

- cladding materials

- ## Heat Flux Hot Channel Factor (F_Q)

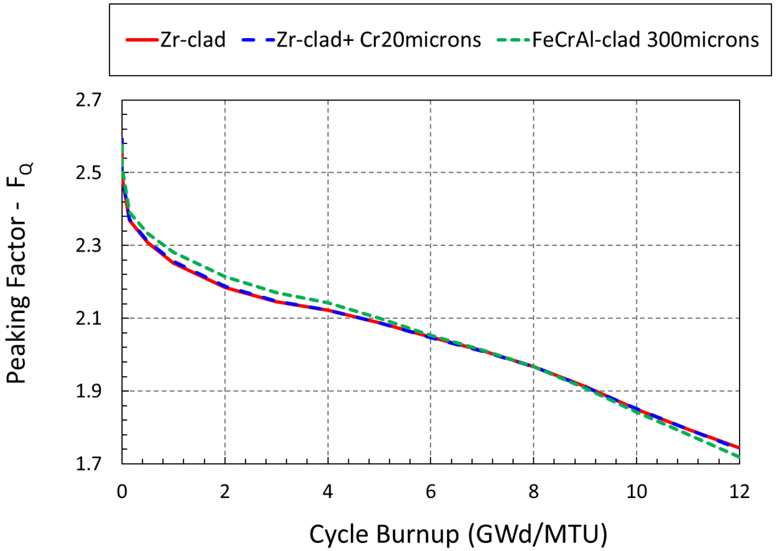
$$F_Q = \frac{max. local\ LHGR}{average\ LHGR} = \max[Q(z)]$$

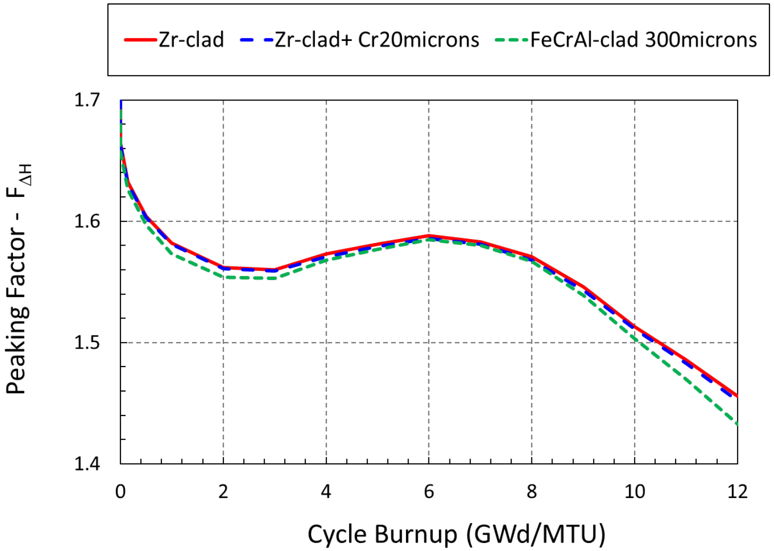
LHGR is Linear Heat Generation Rate (W/cm)

Q(z) is the maximum linear power at elevation-z

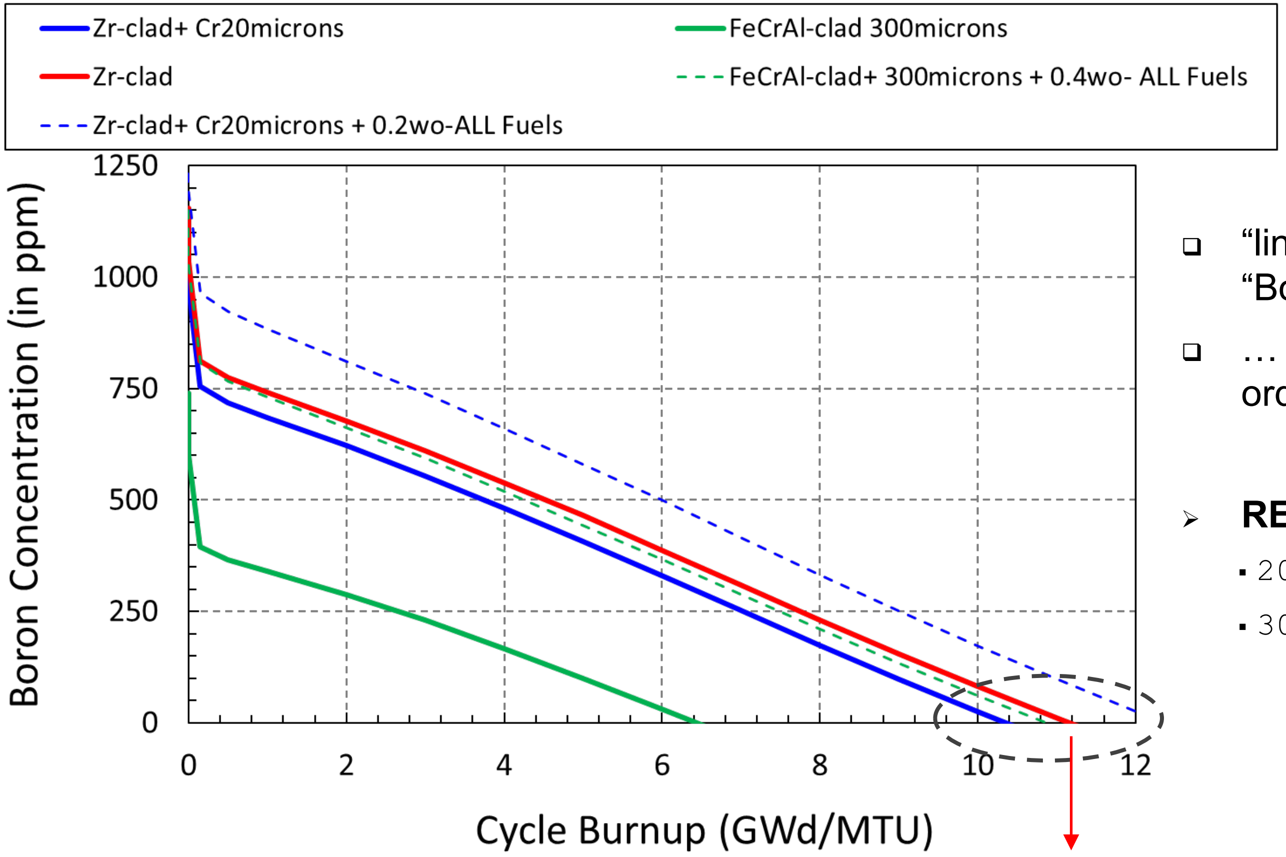
- ## Enthalpy Peaking factor ($F_{\Delta H}$)

$$F_{\Delta H} = \frac{max. chhanel\ enthalpy\ rise}{core\ average\ entalphy\ rise} = \max[\frac{\Delta H}{avg\ \Delta H}]$$





1. Testing ATFs: “Equivalent Cycle Length”



Cycle length

~ 11.16 GWd/MTU

- “linear reactivity” approach in “Boron concentration” at EOC
- … lower/higher enrichment in order to meet the cycle length

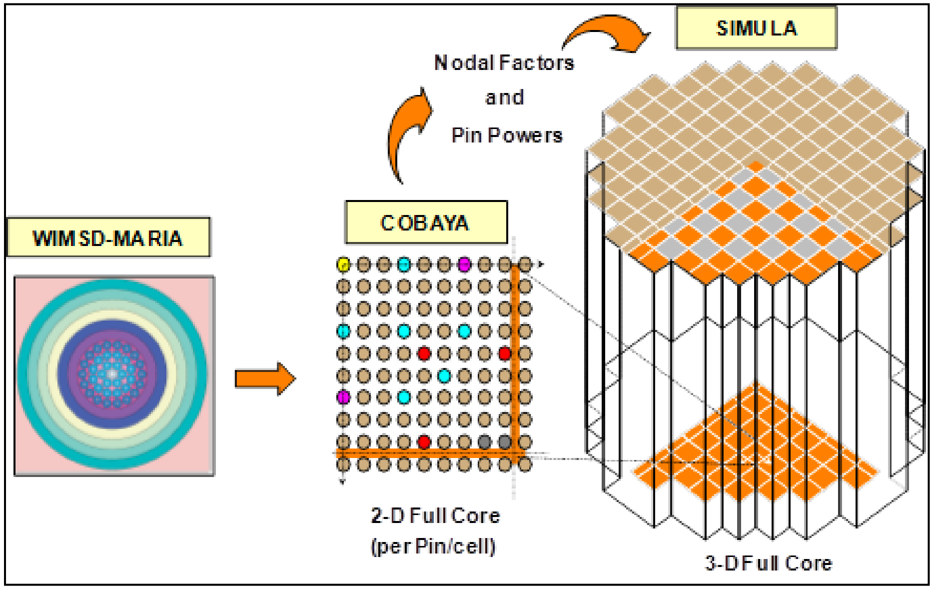
➢ **RESULTS**

- `20mmCr-coating:` **+0.08 w/o**
- `300mm FeCrAl-clad:`**+0.40 w/o**

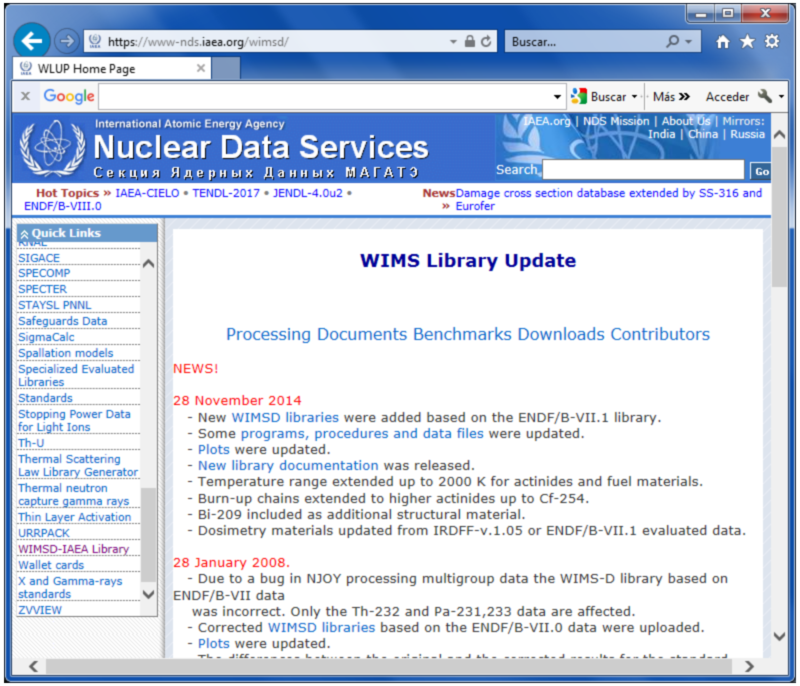
2.Impact of Nuclear Data in Simulations

❑ Upgraded SEANAP with WIMSD5:

INGENIA 2018/2019



❑ IAEA-WIMS Library Update Project



Hot Topics » IAEA-CIELO • TENDL-2017 • JENDL-4.0u2 • ENDF/B-VIII.0

News Damage cross section database extended by SS-316 and » Eurofer

WIMS Library Update

Processing Documents Benchmarks Downloads Contributors

NEWS!

28 November 2014
- New WIMSD libraries were added based on the ENDF/B-VII.1 library.
- Some programs, procedures and data files were updated.
- Plots were updated.
- New library documentation was released.
- Temperature range extended up to 2000 K for actinides and fuel materials.
- Burn-up chains extended to higher actinides up to Cf-254.
- Bi-209 included as additional structural material.
- Dosimetry materials updated from IRDFF-v.1.05 or ENDF/B-VII.1 evaluated data.

28 January 2008.
- Due to a bug in NJOY processing multigroup data the WIMS-D library based on ENDF/B-VII data was incorrect. Only the Th-232 and Pa-231,233 data are affected.
- Corrected WIMSD libraries based on the ENDF/B-VII.0 data were uploaded.
- Plots were updated.

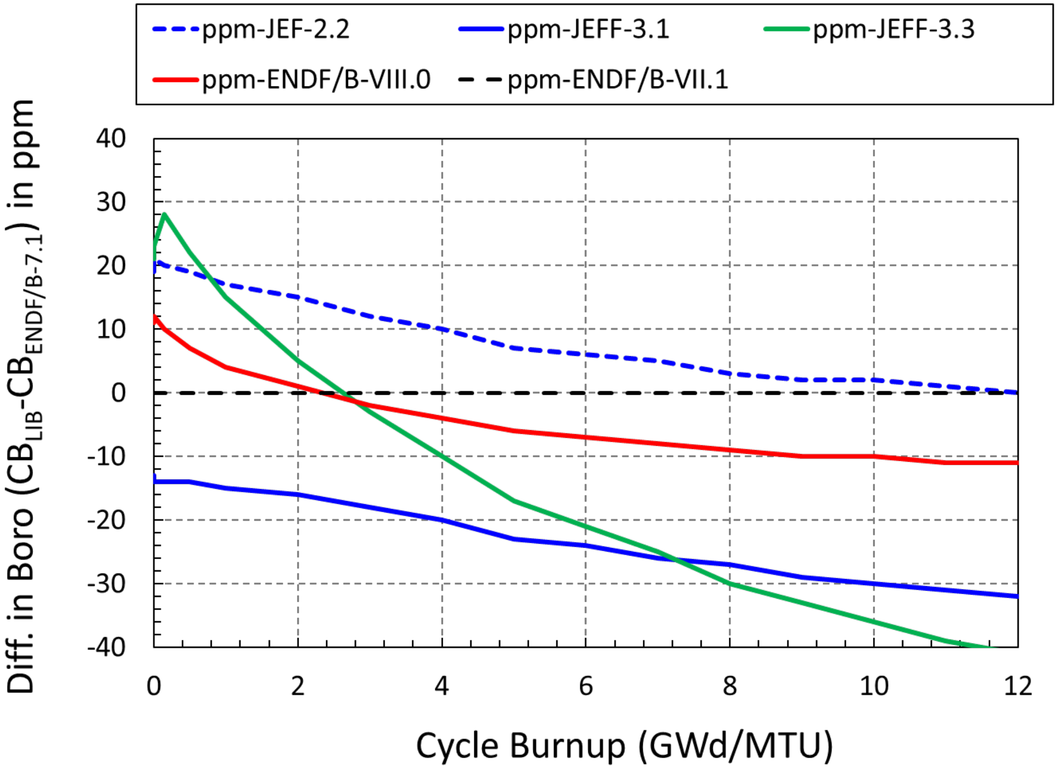
Ref: https://www-nds.iaea.org/wimsd/

SEANAP:
- WIMS-D5 (Lattice code) + COBAYA + SIMULA

Nuclear Data:
- JEF-2.2, JEFF-3.1 and ENDF/B-VII.1: https://www-nds.iaea.org/wimsd/
- JEFF-3.3 and ENDF/B-VIII.0: **INGENIA 2018/2019**

2.Impact of Nuclear Data in Simulations

❑ Boron letdown in NuScale



❑ Simulations with SEANAP

- **Reference: ENDF/B-VII.1**
- Changes only ND library
- ND libraries (Decay Data, Fission Yields and Neutron Interaction):
  - JEF-2.2, JEFF-3.1 and JEFF-3.3
  - ENDF/B-VII.1, ENDF/B-VIII.0

❑ Impact of ND libraries:

- Significant changes in JEFF-3.3

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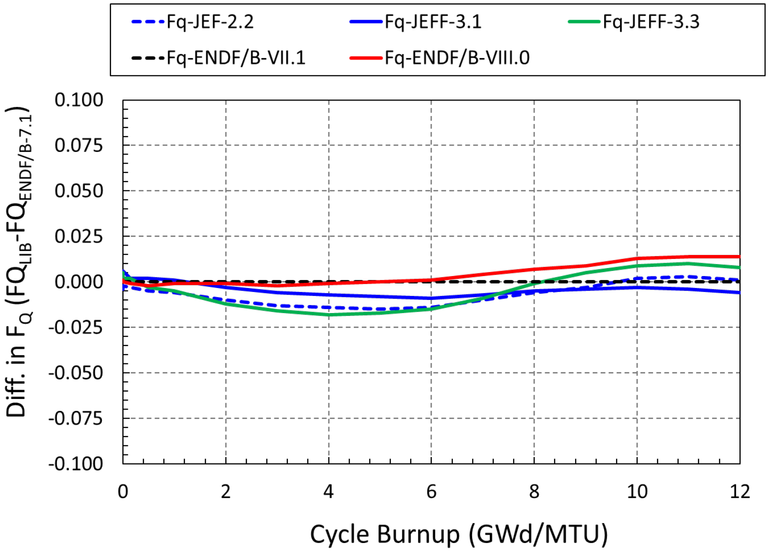
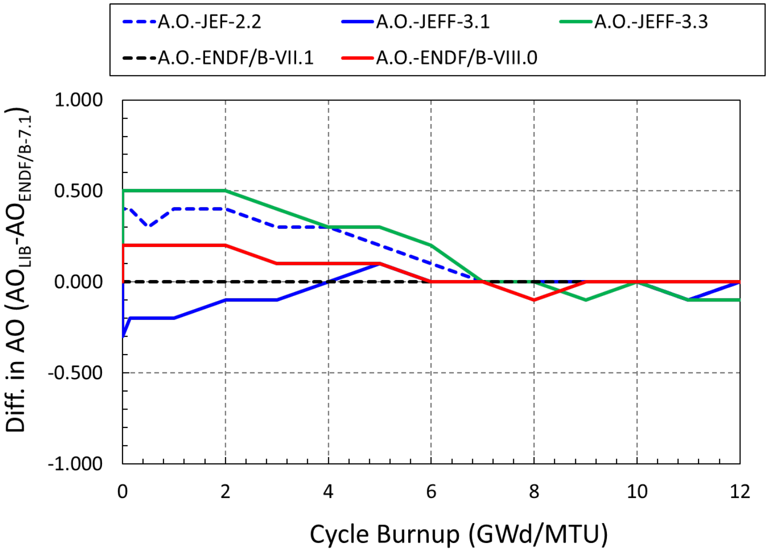
2.Impact of Nuclear Data in Simulations

❑ Heat Flux Hot Channel Factor (F_Q)



❑ Axial Offset (A.O.)

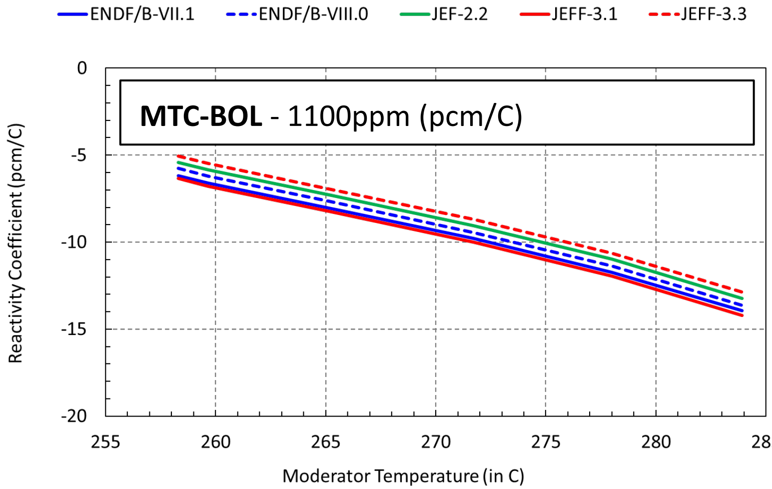


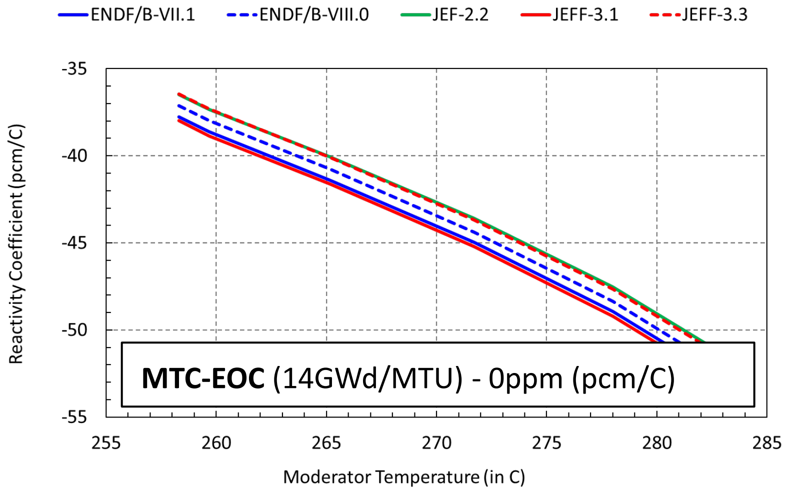
❑ Impact of ND libraries:

- Small differences in power peaking and axial distribution

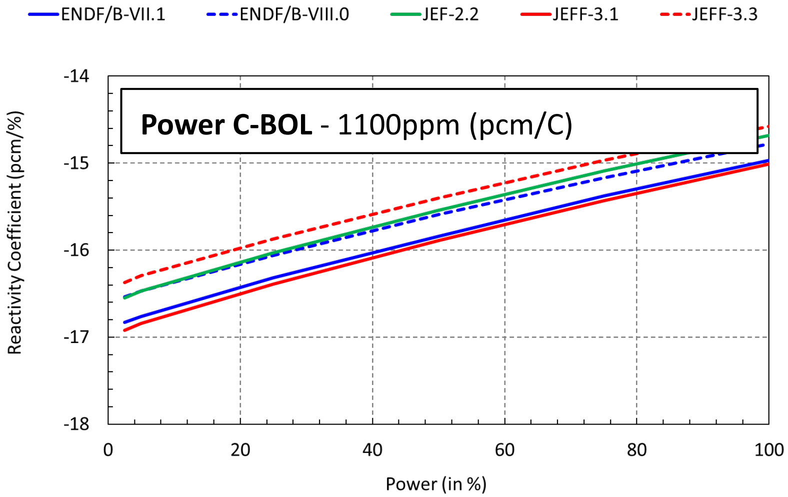
2.Impact of Nuclear Data in Simulations

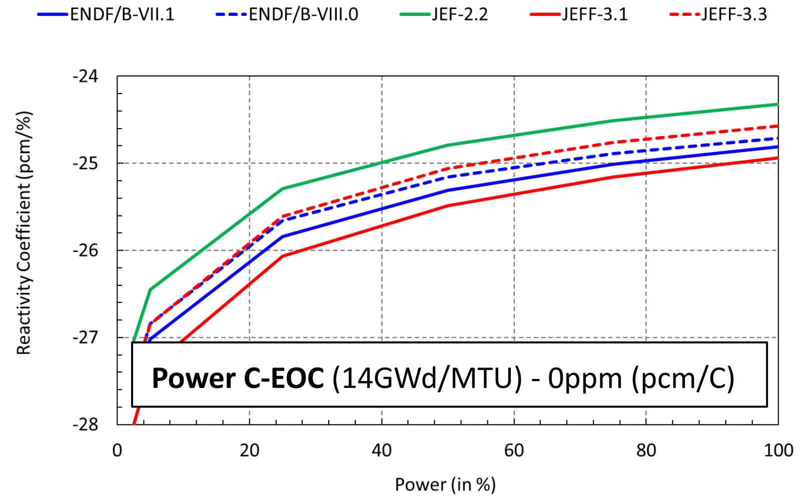
❑ Temperature Reactivity Coefficients





❑ Power Reactivity Coefficient





3. Operational Maneuvering in SMRs

- **Operational Manoeuvring: “Technical Specifications”**
  - Control Rod Insertion Limit
  - Constant Axial Offset Control (CAOC) is implemented in SEANAP

- **Example of operational manoeuvres using SEANAP**
  - **Power Maneuvering**
    - Flexible operation, return to power after a short shutdown, etc …
    - As a function of burnup: BOC, MOC, EOC
  - **Analysis of simulations**
    - Xenon Level versus time
    - Boron Concentration versus time
    - Control Rod position versus Relative Power
    - Axial Flux Difference (AO·Prel) versus Relative Power
    - P·FQ versus time

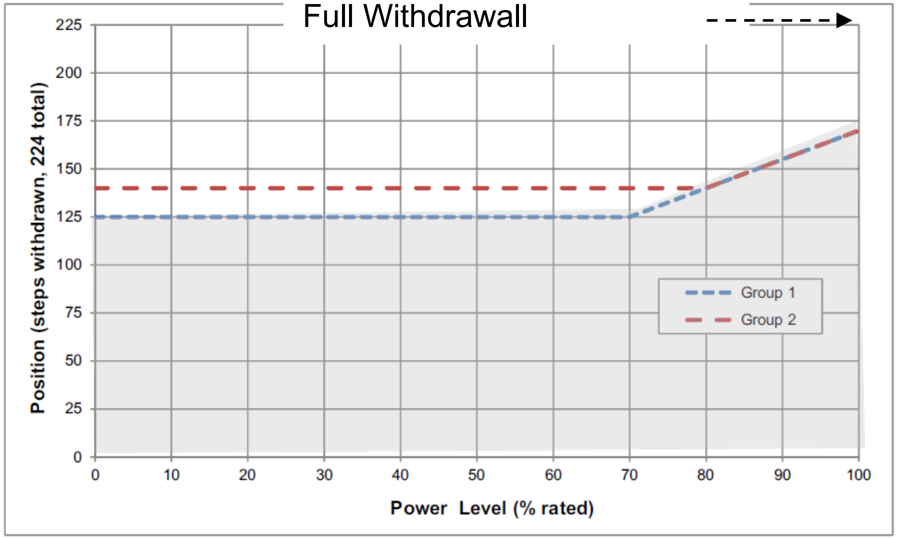
3.1 Control Rod Insertion Limit

❑ Control Rod Insertion : Limiting Condition for Operation

Rod Cluster Control Assemblies (RCCAs) are uniformly located in the core

- ○ Regulating group (RE1+RE2) for power control
- ○ Shutdown group (SH3+SH4)

Figure. Control Rod insertion limit



Ref: Fig4.3.2 at https://www.nrc.gov/docs/ML2022/ML20224A492.pdf

Control Rod insertion limit will allow to:

- reduce the decrement of reactivity worth
- limit $F_{\Delta H}$*P
- secure shutdown margin

			Regulating Bank Group 2			
	ICI 12	Shutdown Bank Group 4	ICI 1	Shutdown Bank Group 3	ICI 9	
	Shutdown Bank Group 3	ICI 8	Regulating Bank Group 1	ICI 2	Shutdown Bank Group 4	
Regulating Bank Group 2	ICI 7	Regulating Bank Group 1		Regulating Bank Group 1	ICI 3	Regulating Bank Group 2
	Shutdown Bank Group 4	ICI 6	Regulating Bank Group 1	ICI 4	Shutdown Bank Group 3	
	ICI 11	Shutdown Bank Group 3	ICI 5	Shutdown Bank Group 4	ICI 10	
			Regulating Bank Group 2			

🟨 Regulating CRA Bank Location
🟥 Shutdown CRA Bank Location
🟦 In-Core Instrumentation Location
■ Empty Fuel Assembly Location

Ref: Fig4.3-18 at https://www.nrc.gov/docs/ML2022/ML20224A492.pdf

- ○ Control Rod “Groups” are divided into **“banks”** to :
  - avoid the effect on the power distribution
  - avoid large reactivity change in control insertion

3.2 Constant Axial Offset Control (CAOC) is implemented in SEANAP

❑ **Axial Flux Difference : Limiting Condition for Operation**

Axial power distribution can be controlled by the Constant Axial Offset Control (CAOC)

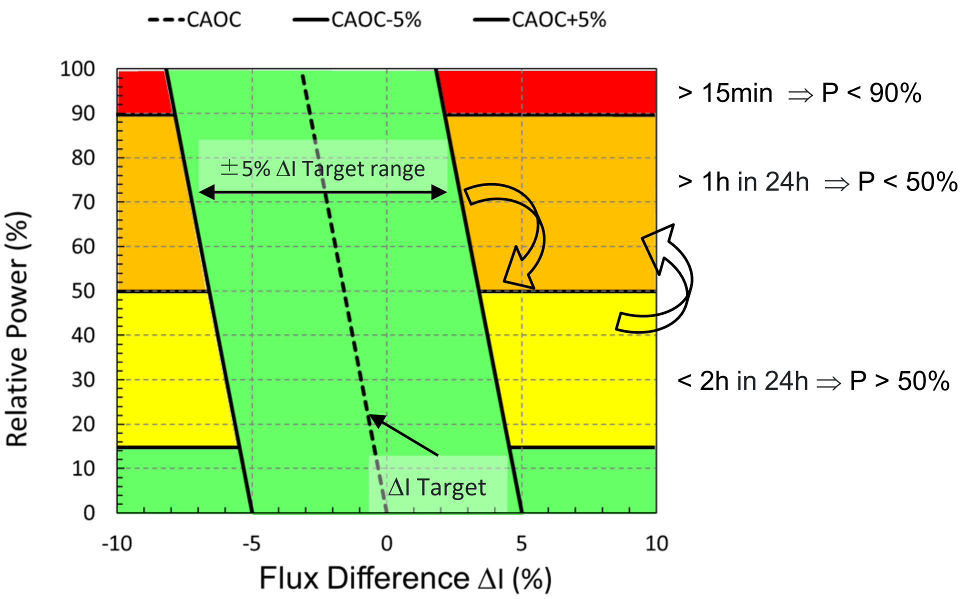
- ○ **Axial Flux Difference (AFD)**

$$\Delta I(\%) = AO \cdot P_{rel}\ (\%)$$

- ○ **ΔI Range: ±5% to:**
  - ▪ ensure peaking factor limits: Low ΔI ⇒ low F_Q
  - ▪ reduce ^{135}Xe/^{135}I axial oscillations

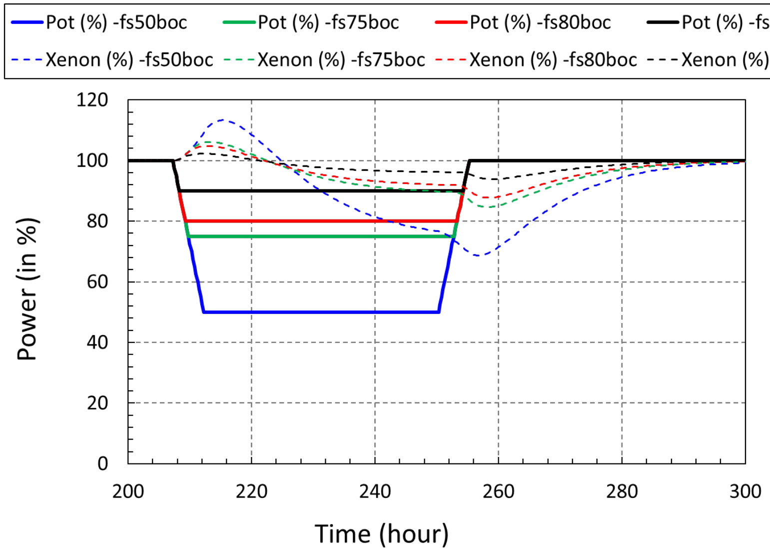
- ○ **Relaxation in CAOC** restrictions will allow to:
  - ▪ enhance load follow capability by allowing control strategies that minimize dilution/boration
  - ▪ increase the ability to return to power after shutdown

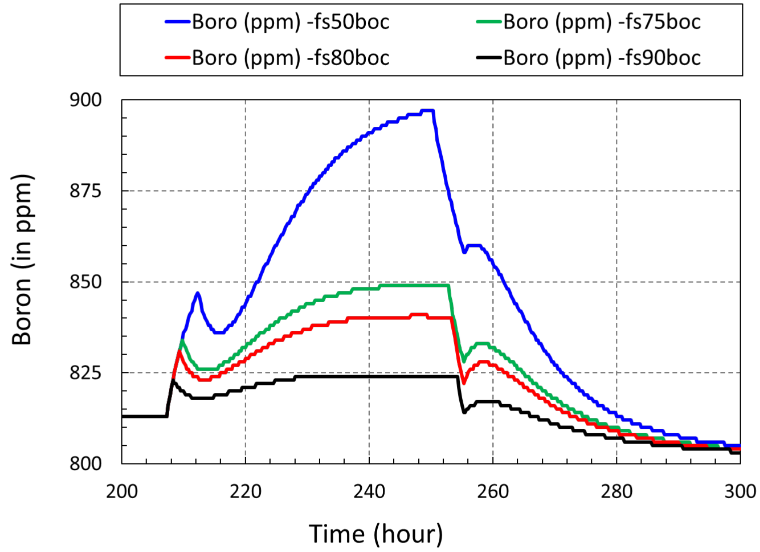
Figure. Axial Flux Difference (AFD) Limits as a functions of rated thermal power

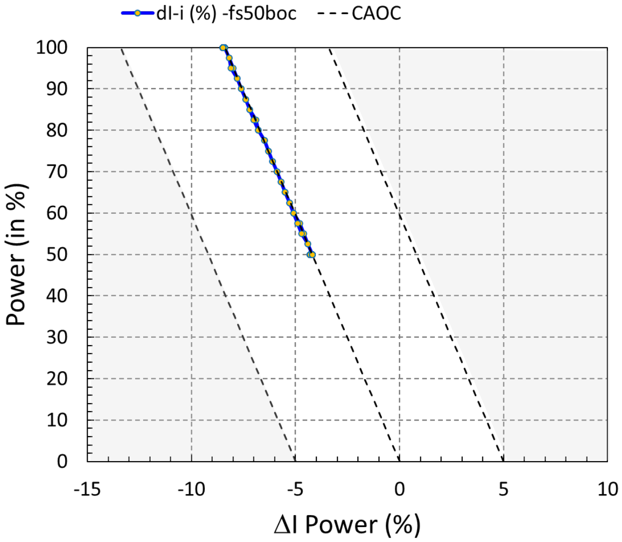


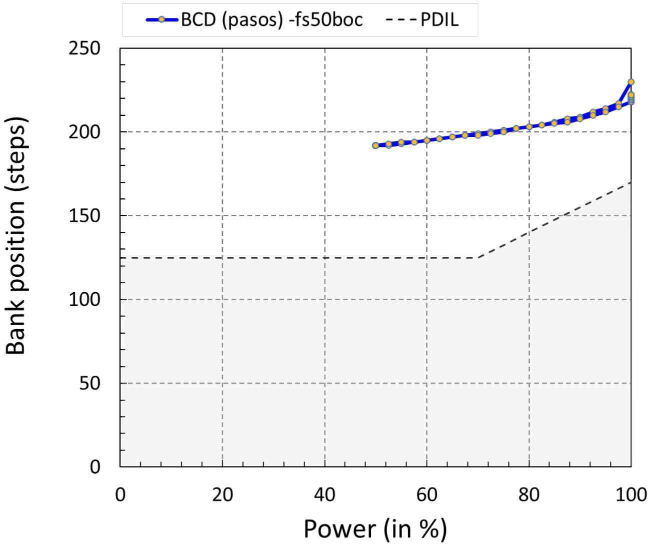
P> 90%	• AFD outside of target band, within 15 min. restore the AFD to within the target band or P < 90%
50% < P < 90%	• AFD outside of target band <1h in 24h. AFD within limits ⇒ P>90% • AFD outside of target band > 1h ⇒ P<50% within 30min.
15% < P < 50%	• AFD outside of target band < 2h in 24h ⇒P>50%
P < 15%	• No restrictions

3.3 Example of operational maneuvers in NuScale

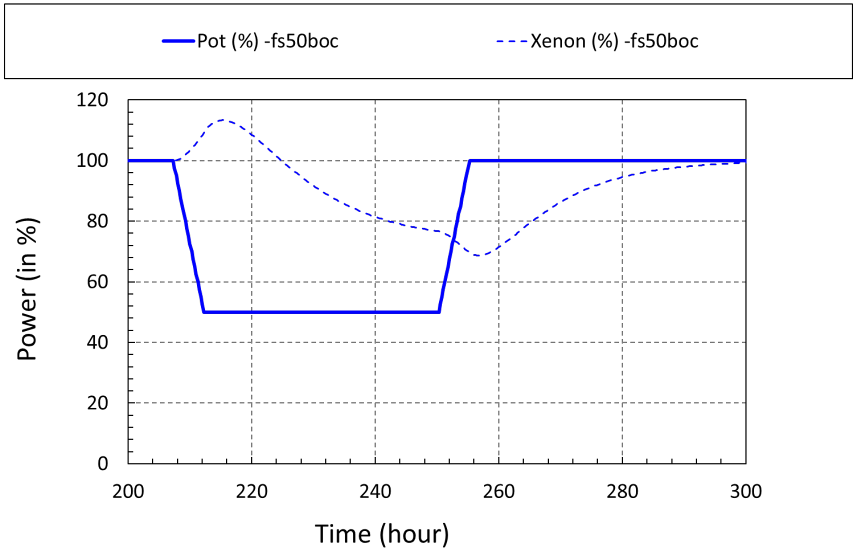


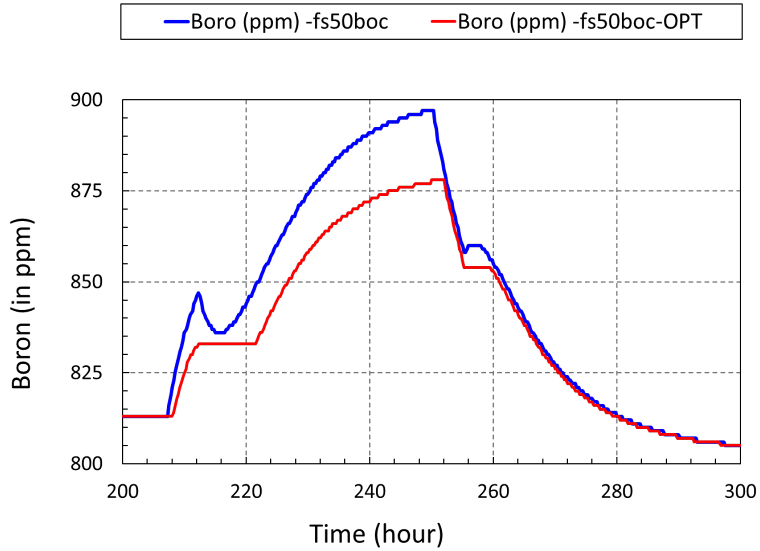


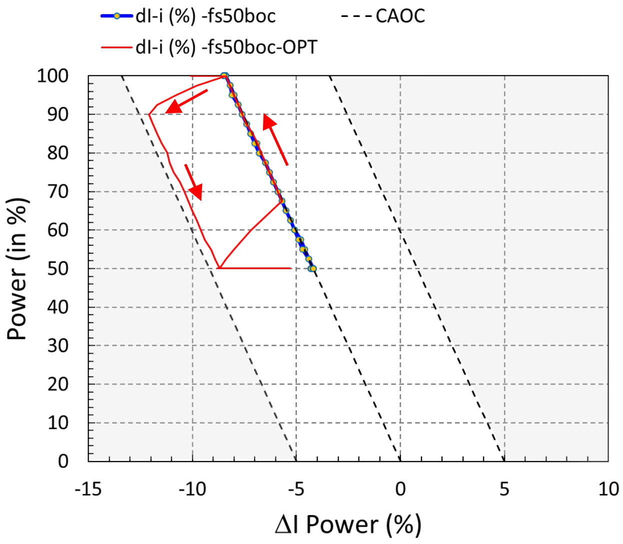


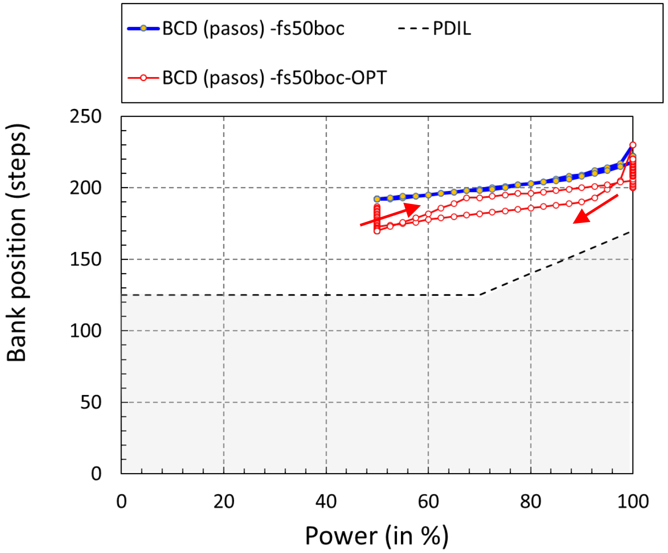


3.3 Example of maneuvers: 48h-50%-BOC: CAOC vs OPTIM

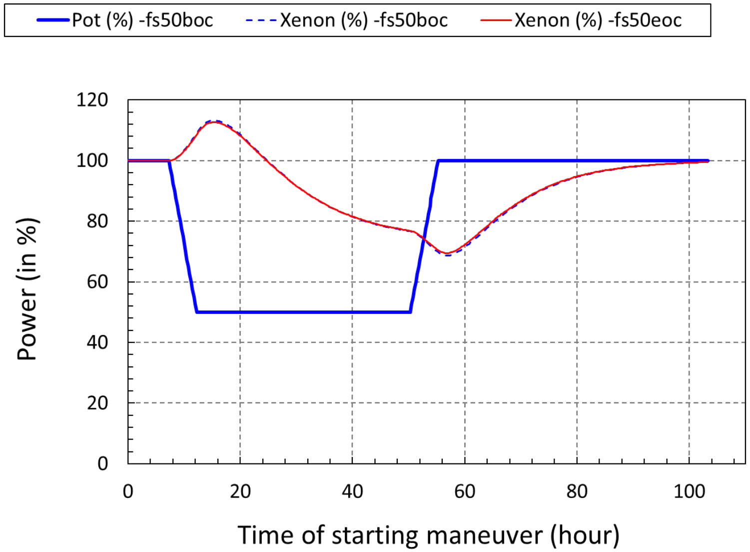


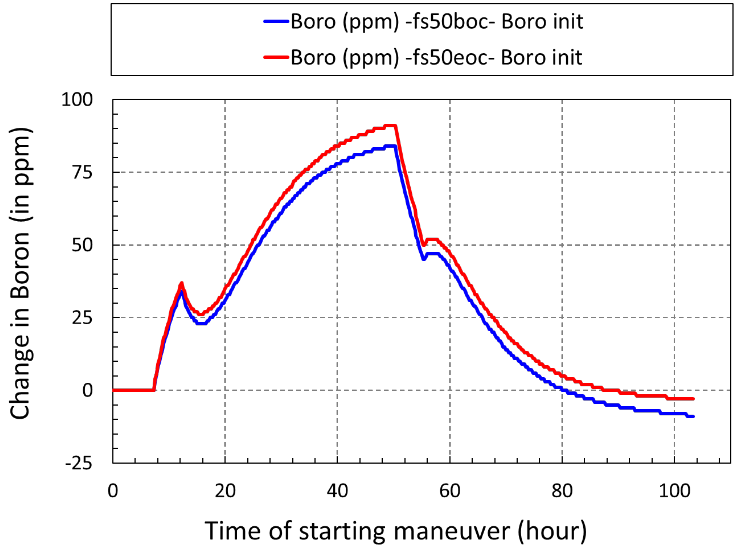


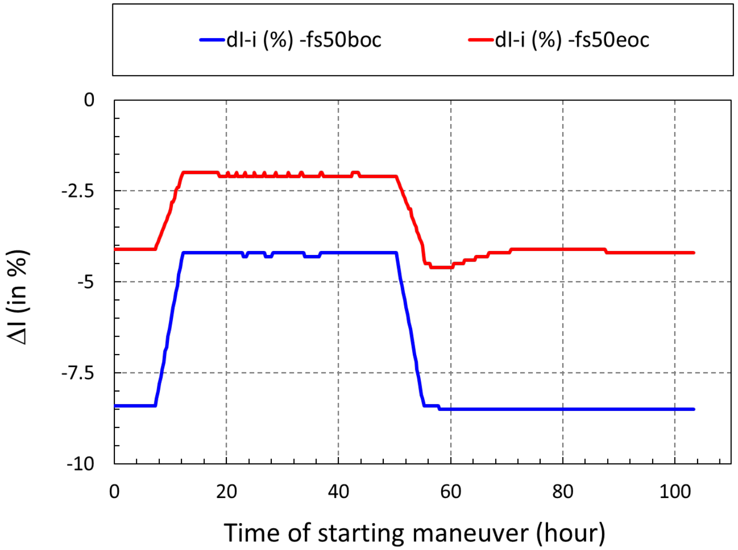


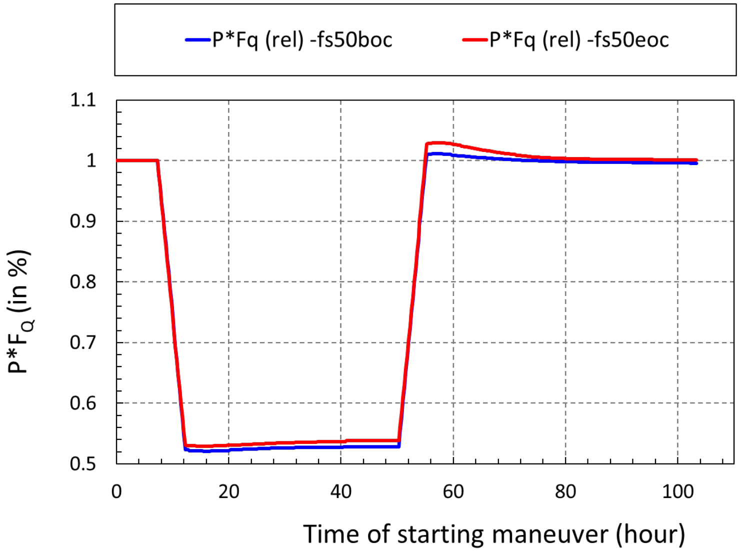


3.3 Example of maneuvers: 48h-50%-CAOC: BOC vs EOC









SUMMARY

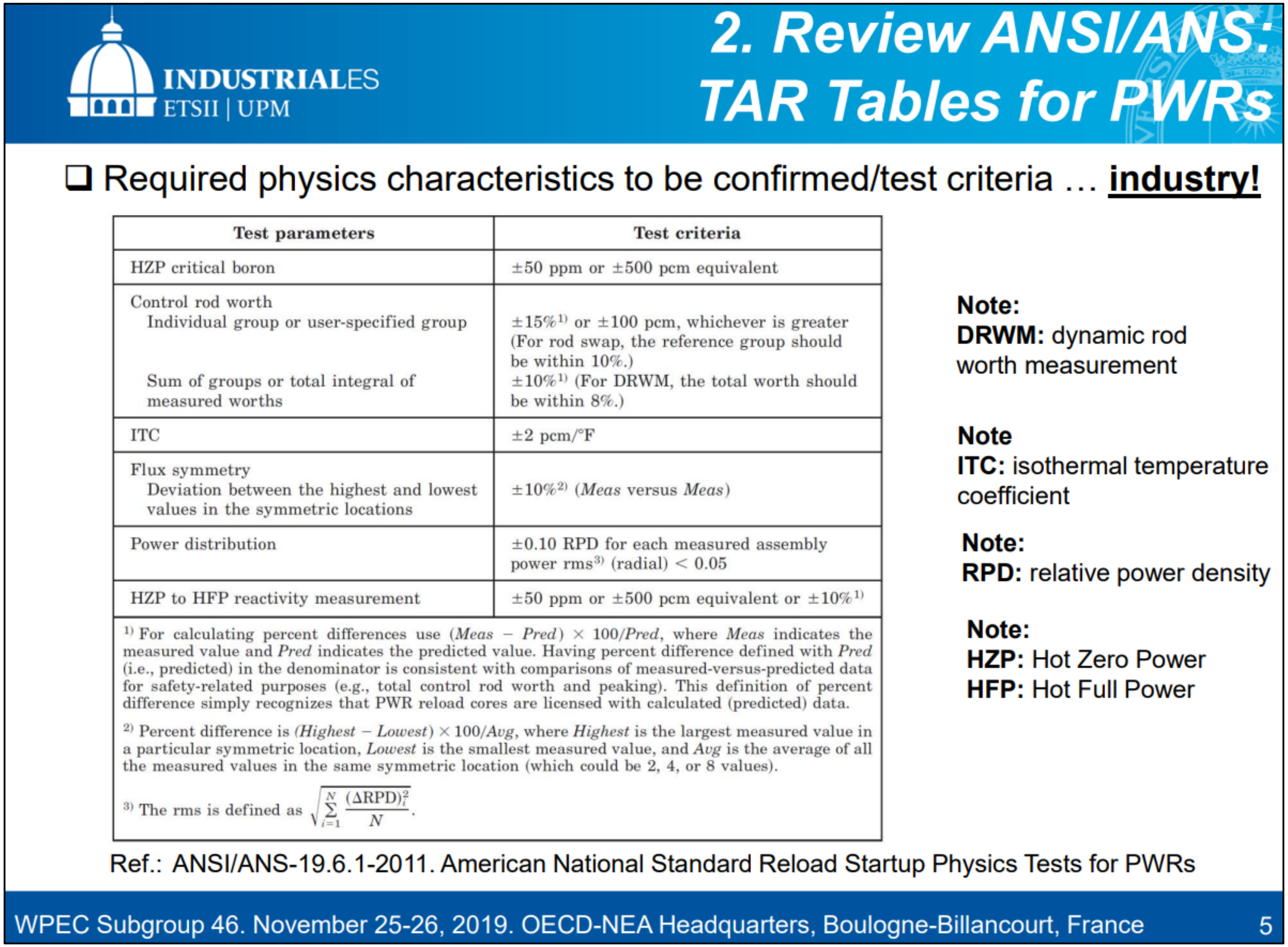
❑ The **SEANAP 3-D Core analysis** system has been used to carry out this work:

- **Testing ATFs (Accident Tolerant Fuels) cladding materials**
  - Cr- 20 μm coating
  - FeCrAl- clad with 300 μm thickness
- **The impact of different nuclear data libraries**
  - JEF-2.2, JEFF-3.1 and JEFF-3.3
  - ENDF/B-VII.1 and ENDF/B-VIII.0
- **Simulation of Maneuvers/Operation in NuScale with SEANAP system**

Acknowledgments

This work is part of the ***McSAFER project*** *(**IMPROVING SAFETY ANALYSIS METHODOLOGIES AND MOVING FROM TRADITIONAL TO HIGH-FIDELITY SAFETY ANALYSIS TOOLS FOR SMALL MODULAR REACTORS**) that has received funding from the European Union's H2020/Euratom under grant agreement No. 945063*

Back-up Slides



2. Review ANSI/ANS: TAR Tables for PWRs

- ☐ Required physics characteristics to be confirmed/test criteria … **industry!**

Test parameters	Test criteria
HZP critical boron	±50 ppm or ±500 pcm equivalent
Control rod worth Individual group or user-specified group	±15%[1] or ±100 pcm, whichever is greater (For rod swap, the reference group should be within 10%.)
Sum of groups or total integral of measured worths	±10%[1] (For DRWM, the total worth should be within 8%.)
ITC	±2 pcm/°F
Flux symmetry Deviation between the highest and lowest values in the symmetric locations	±10%[2] (*Meas* versus *Meas*)
Power distribution	±0.10 RPD for each measured assembly power rms[3] (radial) < 0.05
HZP to HFP reactivity measurement	±50 ppm or ±500 pcm equivalent or ±10%[1]

[1] For calculating percent differences use (*Meas* − *Pred*) × 100/*Pred*, where *Meas* indicates the measured value and *Pred* indicates the predicted value. Having percent difference defined with *Pred* (i.e., predicted) in the denominator is consistent with comparisons of measured-versus-predicted data for safety-related purposes (e.g., total control rod worth and peaking). This definition of percent difference simply recognizes that PWR reload cores are licensed with calculated (predicted) data.

[2] Percent difference is (*Highest* − *Lowest*) × 100/*Avg*, where *Highest* is the largest measured value in a particular symmetric location, *Lowest* is the smallest measured value, and *Avg* is the average of all the measured values in the same symmetric location (which could be 2, 4, or 8 values).

[3] The rms is defined as $\sqrt{\sum_{i=1}^{N}\frac{(\Delta \mathrm{RPD})_i^2}{N}}$.

Note:
DRWM: dynamic rod worth measurement

Note
ITC: isothermal temperature coefficient

Note:
RPD: relative power density

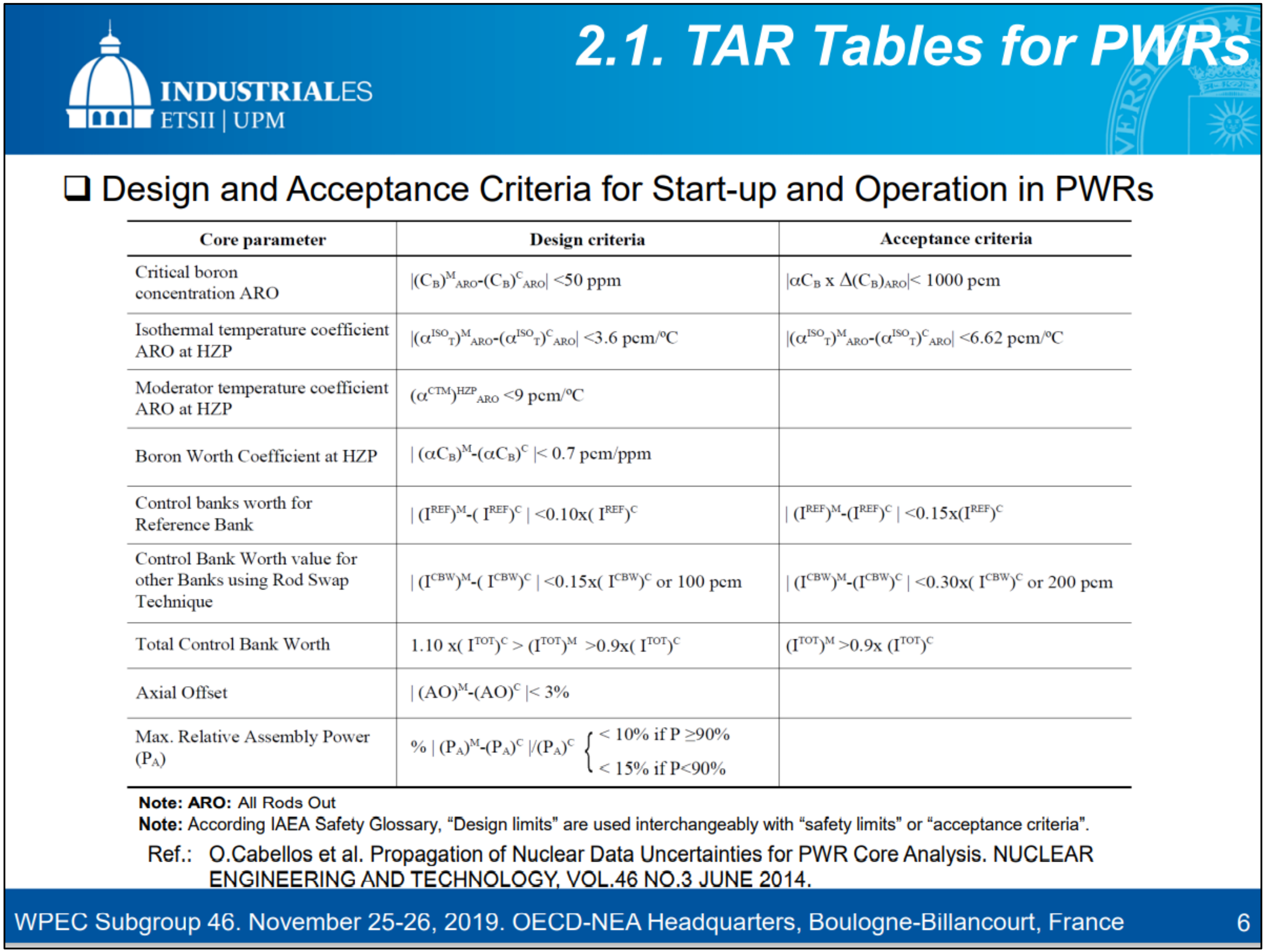
Note:
HZP: Hot Zero Power
HFP: Hot Full Power

Ref.: ANSI/ANS-19.6.1-2011. American National Standard Reload Startup Physics Tests for PWRs

WPEC Subgroup 46. November 25-26, 2019. OECD-NEA Headquarters, Boulogne-Billancourt, France 5

https://www.oecd-nea.org/download/wpec/sg46/meetings/2019-11/documents/15.1-Cabellos.pdf

Back-up Slides



2.1. TAR Tables for PWRs

❑ Design and Acceptance Criteria for Start-up and Operation in PWRs

Core parameter	Design criteria	Acceptance criteria
Critical boron concentration ARO	$\lvert (C_B)^M_{ARO}-(C_B)^C_{ARO}\rvert < 50$ ppm	$\lvert \alpha C_B \ \mathrm{x}\ \Delta(C_B)_{ARO}\rvert < 1000$ pcm
Isothermal temperature coefficient ARO at HZP	$\lvert (\alpha^{ISO}_T)^M_{ARO}-(\alpha^{ISO}_T)^C_{ARO}\rvert < 3.6$ pcm/ºC	$\lvert (\alpha^{ISO}_T)^M_{ARO}-(\alpha^{ISO}_T)^C_{ARO}\rvert < 6.62$ pcm/ºC
Moderator temperature coefficient ARO at HZP	$(\alpha^{CTM})^{HZP}_{ARO} < 9$ pcm/ºC	
Boron Worth Coefficient at HZP	$\lvert (\alpha C_B)^M-(\alpha C_B)^C \rvert < 0.7$ pcm/ppm	
Control banks worth for Reference Bank	$\lvert (I^{REF})^M-(I^{REF})^C \rvert < 0.10\mathrm{x}(I^{REF})^C$	$\lvert (I^{REF})^M-(I^{REF})^C \rvert < 0.15\mathrm{x}(I^{REF})^C$
Control Bank Worth value for other Banks using Rod Swap Technique	$\lvert (I^{CBW})^M-(I^{CBW})^C \rvert < 0.15\mathrm{x}(I^{CBW})^C$ or 100 pcm	$\lvert (I^{CBW})^M-(I^{CBW})^C \rvert < 0.30\mathrm{x}(I^{CBW})^C$ or 200 pcm
Total Control Bank Worth	$1.10\ \mathrm{x}(I^{TOT})^C > (I^{TOT})^M > 0.9\mathrm{x}(I^{TOT})^C$	$(I^{TOT})^M > 0.9\mathrm{x}\ (I^{TOT})^C$
Axial Offset	$\lvert (AO)^M-(AO)^C \rvert < 3\%$	
Max. Relative Assembly Power (P_A)	$\% \lvert (P_A)^M-(P_A)^C \rvert/(P_A)^C \begin{cases} < 10\% \text{ if } P \geq 90\% \\ < 15\% \text{ if } P<90\% \end{cases}$	

Note: ARO: All Rods Out
Note: According IAEA Safety Glossary, “Design limits” are used interchangeably with “safety limits” or “acceptance criteria”.

Ref.: O.Cabellos et al. Propagation of Nuclear Data Uncertainties for PWR Core Analysis. NUCLEAR ENGINEERING AND TECHNOLOGY, VOL.46 NO.3 JUNE 2014.

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https://www.oecd-nea.org/download/wpec/sg46/meetings/2019-11/documents/15.1-Cabellos.pdf