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Request from Nuclear Fuel Cycle & Criticality Safety Design

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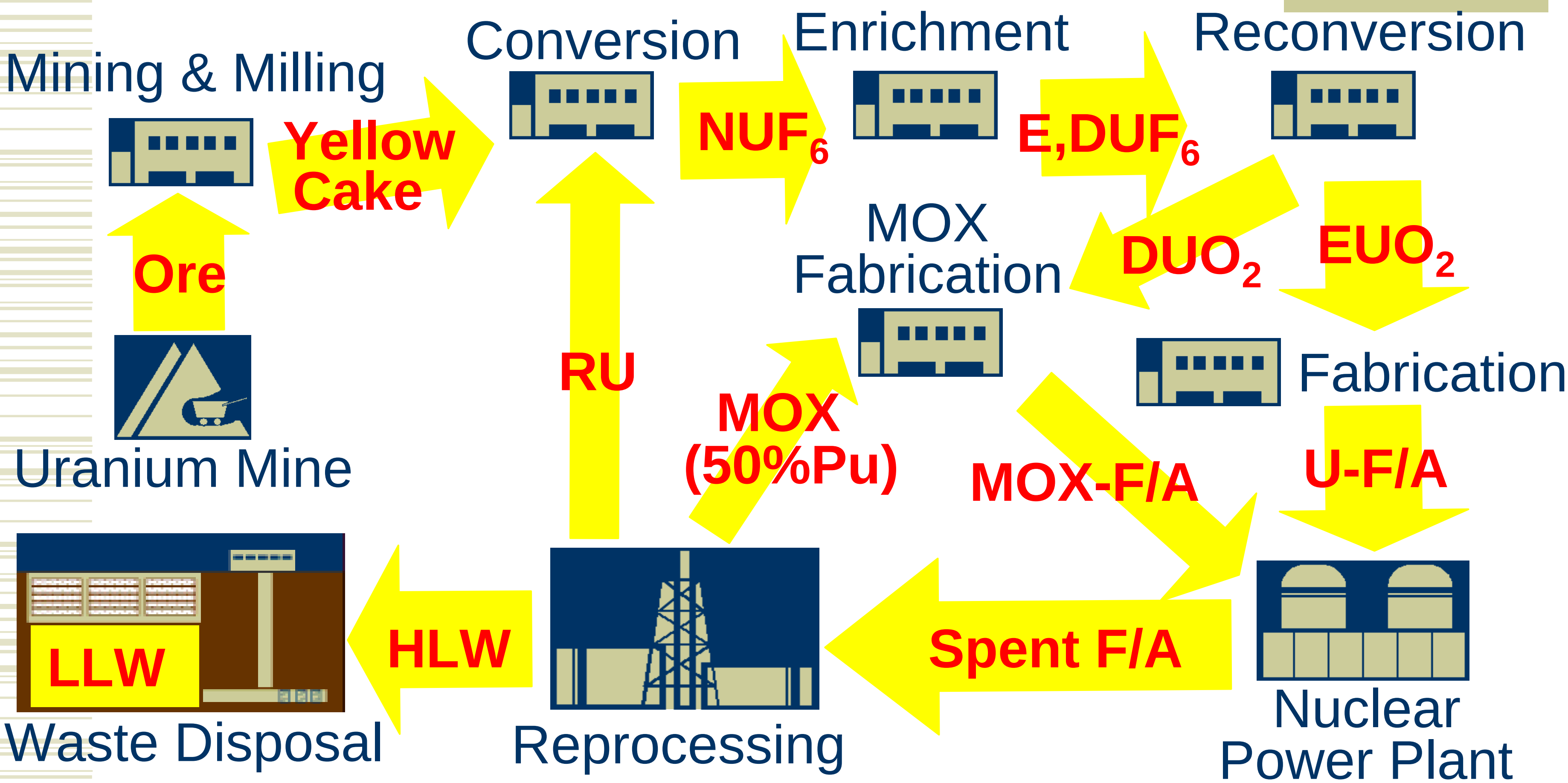
Nuclear Fuel Cycle (LWR)



Variety of Fissile Materials (1)

Uranium "Cradle to Grave" and Rebirth

- Ore – Uranium Compounds (ADU, UF_6, Oxide)
- Gas – Suspension – Precipitation – Powder – Pellets – Rods – Assemblies

- Spent Fuel (U, Pu, FPs, MAs) Assemblies
- Assemblies –Chopped Oxide (U/Pu/FP/MA)

  – Nitric Solution (U/Pu/FP/MA) – Nitric Solution (U/Pu)
- Powder (RUO_3,MOX) – Pellets (MOX) – MOX Assemblies

Variety of Fissile Materials (2)

Gaseous

- Hexa-Fluoride : UF_6

Fluidal

- Fluoride Solution : UO_2F_2aq
- Nitric Solution : $UO_2(NO_3)_2aq$, $Pu(NO_3)_4aq$
- Homo. Powder : UO_2/ UO_3-H_2O, MOX-H_2O
- Slurry : ADU

Solid & Mixture

- Hetero. Pellets/Rods/FAs : UO_2-H_2O, MOX-H_2O
- Dissolving Hetero. Mixture : Spent UO_2 – Nitric Solution

Variety of Fissile Materials (3)

- Indeterminate Form
  : Heterogeneity, Non-uniformity, Randomness
  => Modeling Capability of Criticality Codes
- Dry, Wet, Aqueous Solution
  : Various Moderation Condition (H/HM)
  => Variety of Neutron Spectra
- LEU, Plutonium, MA, Their Mixture
  : Various Resonance Nuclides
  => Accurate Resonance Data & Treatment

Criticality Design Practices in Japanese Industries(1)

- Standardized Criticality Calculation Codes Systems with Group Constants & Monte Carlo Code (ex.KENO family)
- JACS (JAPAN)
  MGCL (Multi-Group Constants Library) :
  ENDF/B-IV based 137G / 26G
  => Applied to Many Licensing Cases
- SCALE (USA)
  CSRL (Criticality Safety Reference Library) :
  ENDF/B-V based 238G / 44G
  ENDF/B-IV based 218G / 27G etc

Criticality Design Practices in Japanese Industries(2)

- Reactor Core Design Codes (ex. PHOENIX-P/HIDRA)
  - F/A Storage at Power Plant Site
  - ENDF-B/V based 42G (PWR Fuels Storage)
  - Consistency with Reactor Core Design

- Continuous Energy Monte Carlo Codes
  - Used for Many Studies without Licensing Purpose
  - MCNP(USA) – ENDF-B/VI , JENDL-3.3 etc
  - MVP(Japan) – ENDF-B/VI, JENDL-3.3 etc

Criticality Safety Design Criteria(1)

- The Most Common : 0.95 Criterion

  k-eff* ≤ *0.95

  where *k-eff* is calculated with the **"well qualified"** criticality code system.

- The Estimated Critical Lower Limit *k-eff* (usu. ≥ 0.95)

  Nuclear Data – Criticality Code – Fissile Systems Specific

  Reliability Limit Derived from Criticality Benchmarks

  <u>Bare Computational Uncertainty w/o Engineered Margin</u>

Criticality Safety Design Criteria(2)

- JACS with ENDF-B/IV based 137G-MGCL

Fissile System	critical *keff*	lower limit *keff*
Homo. LEU	0.989	0.958
Hetero. LEU	0.996	0.978
Pu Solution	1.004	0.973
Homo. MOX	1.018	0.980
Hetero. MOX	0.998	0.980
U/Pu Mixed Solution	0.991	0.950

Ref. "Nuclear Criticality Safety Handbook Version2",JAERI-1340 (1999)

Logistics in Nuclear Fuel Cycle – Fuel Casks

- Casks
  - Versatile Logistic Components for Nuclear Fuel Cycle.
  - Transportation, Interim/Long-Term Storage
- Global Market

Performance of Casks

- Payloads – No. of Fuel Assemblies/ cask
- Affordable Material – Neutron Absorber
  Gamma/Neutron Shielding
- Mechanical Strength, Heat Resistance and Radiation Resistance
- Easy to Handle – Light Weight
- Total Cost Performance

Criticality Safety of Current Spent Fuel Casks

- Subcritical when immersed in water
- Subcritical when piled up in numbers
- Neutron Absorber : B-SUS, B-Al, B-Resin, Cd-Alloy etc

- Unirradiated Fuel with Initial ^{235}U enrich.

- $keff \leq 0.95$ (usu.)

Challenge for Burnup Credit (BUC)

- "A Guide Introducing BU Credit, Preliminary Version", JAERI-Tech 2001-055
- Two Levels of Conservatism,
  Level 1 : Actinides Only (U, Pu, (+MA))
  Level 2 : Actinides and Fission Products
- Uncertainty of **Isotopic Composition** of SF :
  Nuclear Data, Depletion Code, Depletion Conditions
  as well as Cooling Time after Reactor Shutdown
- Uncertainty of **Criticality Prediction** of SF Systems :
  Nuclear Data, Criticality Code
  as well as Isotopic Composition

Validation Requirement for BUC

- Isotopic Composition
  - Post-Irradiation Examination (PIE) Data
  - PIE Data Base : SFCOMPO (JAERI)
- Criticality with Spent Fuel
  - Criticality Tests with Actual SF
    ex. REBUS International Program (Bergonuclaire)
  - Power Reactor Simulation with Criticality Codes
- Integral Tests are with Difficulty by Nature
  - To Cover Whole Spectrum of Depletion Environment,
  - To Fund High Cost to Handle Irradiated Fuels.

Level-1 : Actinide BU Credit

- Already Applied to Many Examples
- Rokkasho Reprocessing Plant (Japan)
  SF Storage Pool (Residual ^{235}U Enrich. Control)
  Dissolver and Head End
  U+Pu Only
- Dry Storage (Germany)
- Transport Casks (France, Germany, Netherlands, Switzerland and USA)

Level-2: FP Credit (1)

- Already Applied to Several Examples
- NPP Site Spent Fuel Pit (USA)
- Long-lived and Chemically Stable (Non-Gaseous nor Non-Volatile) Nuclides are Preferred

Level-2: FP Credit (2)

- Selected Important Nuclides :

6FPs (CEA-France at early stage)

Sm-149, Rh-103, Gd-155, Nd-143, Cs-133, Sm-152

12FPs (JAERI-Tech 2001-055)

+ Tc-99, Eu-153, Nd-145, Sm-147, Mo-95, Sm-150

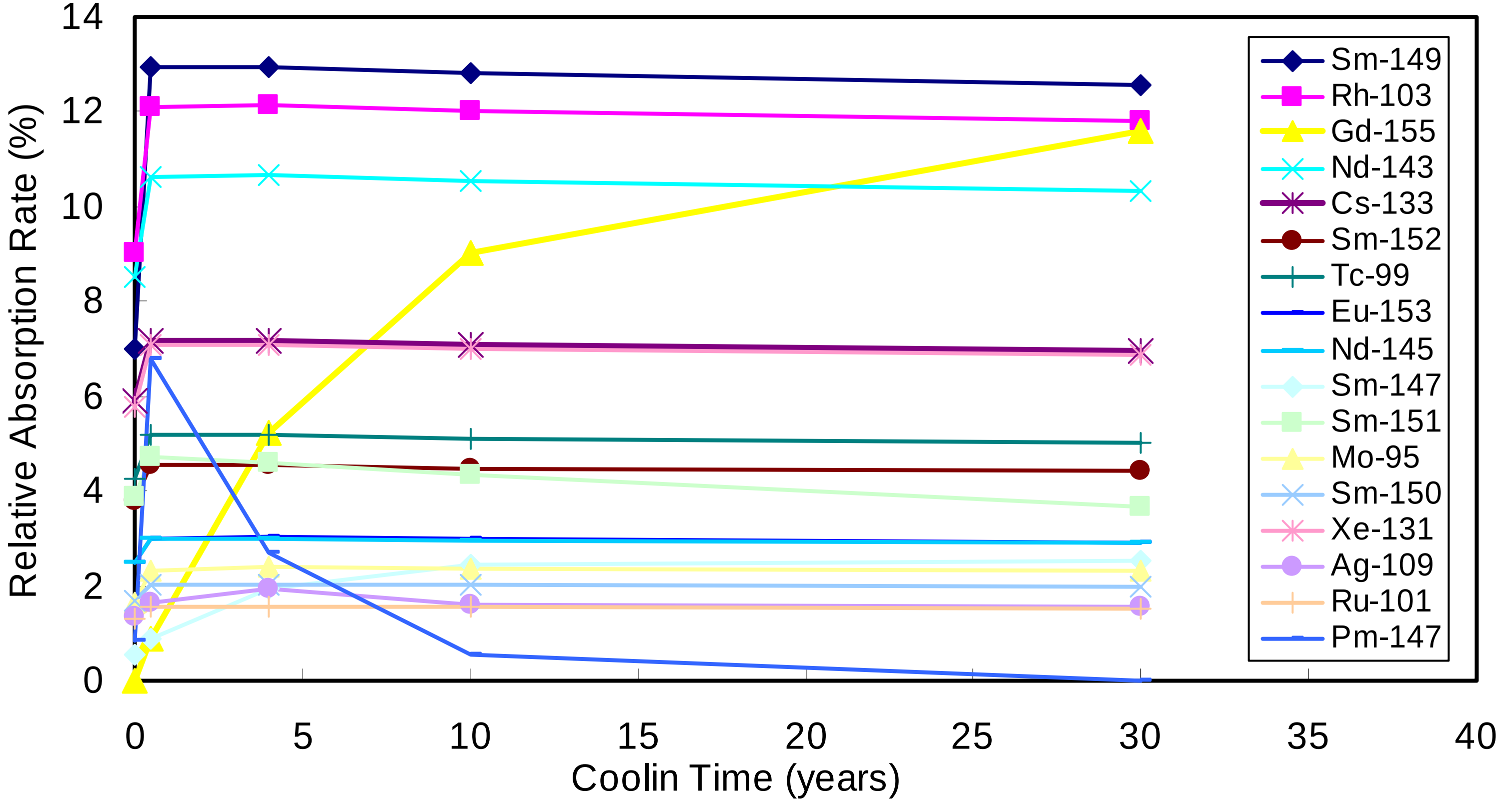
15FPs (OECD BUC W.G.)

+ Sm-151, Ag-109, Ru-101

13FPs (SAND87-0151) for Casks

Tc-99, Rh-103, *Xe-131*, Cs-133, Nd-143, Nd-145, Pm-147, Sm-147, Sm-149, Sm-151, Sm-152, Eu-153, Gd-155

Level-2: FP Credit (3)



Ref. JAERI-Tech 2001-055

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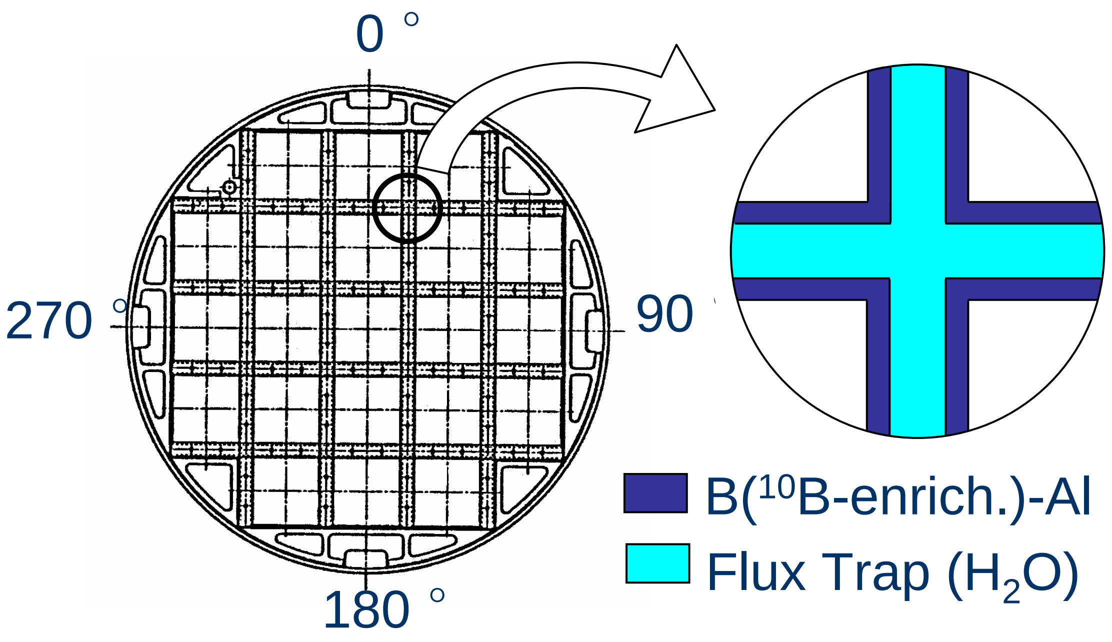
Level-2: FP Credit (4)

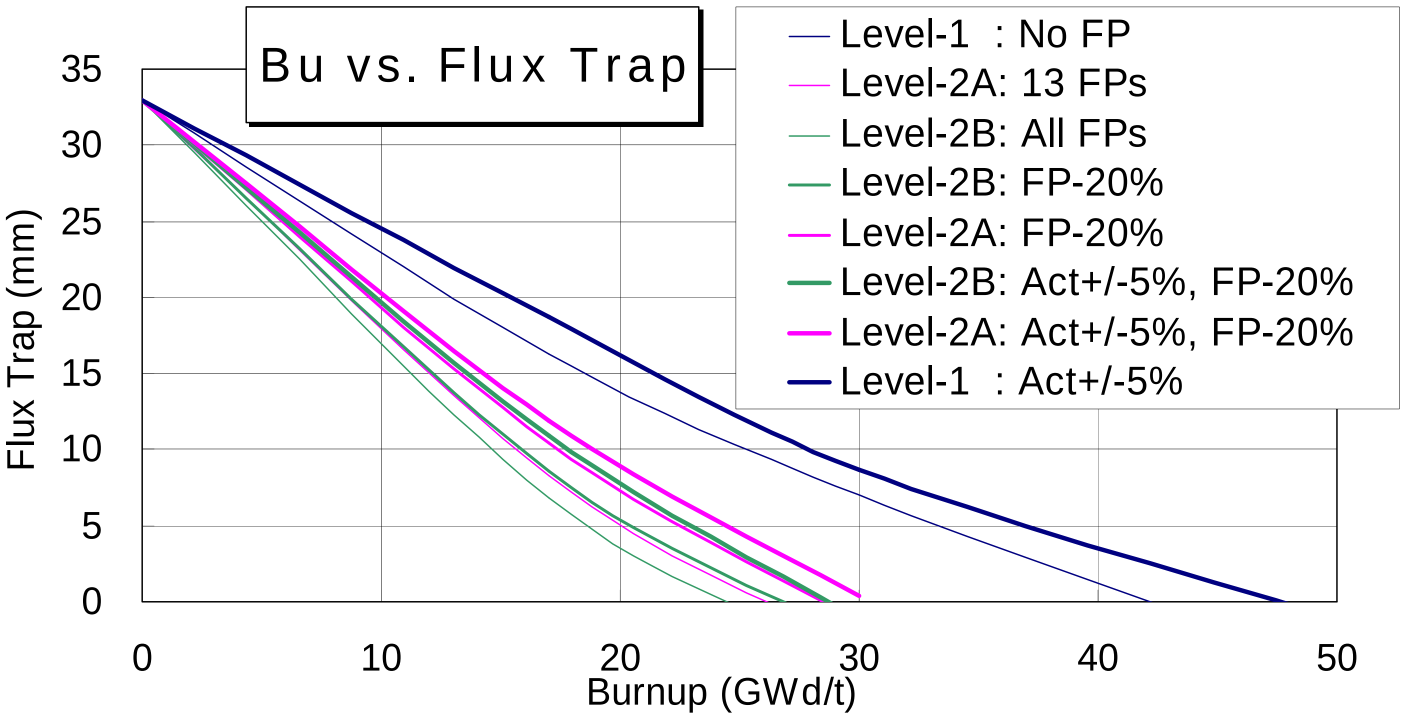
- Case Study : PWR Spent Fuel Cask
- Boronated Aluminum(B-Al) Spacer + Flux Trap Design
- Reference Fuel : PWR 4.1wt% Intact => 4.8wt% BUC
- Flux Trap / B-10 Content Surveyed with No BUC, Level-1,
  Level-2A : 13 FP Nuclides(SAND87-0151) or
  Level-2B : All FP Nuclides Considered BUC.
- Assumed Uncertainty in Isotopic Composition
  U, Pu, Actinides : +5% for fissile, -5% for fertile
  FPs : -20 %

Level-2: FP Credit (5)

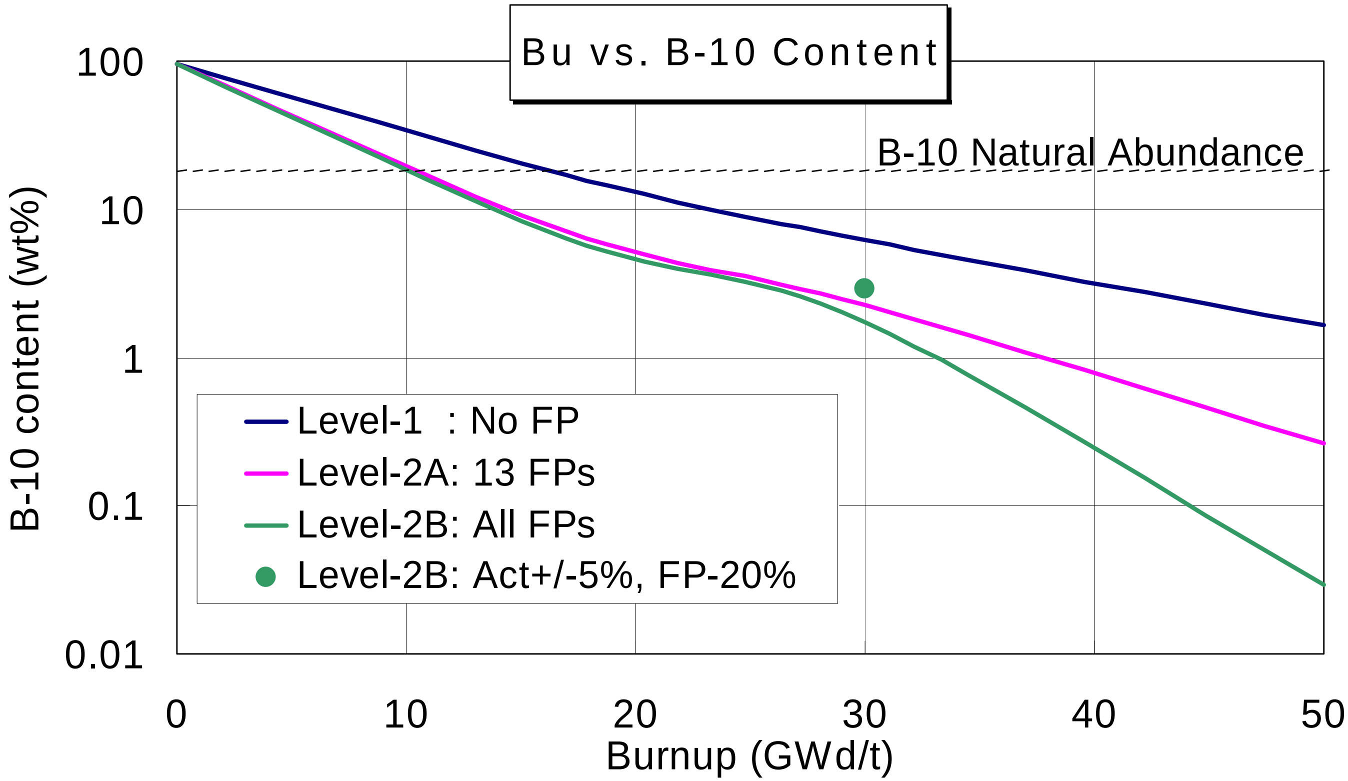


Cask Model for Case Study

Level-2: FP Credit (6)



Level-2: FP Credit (7)



Recent Issue : SF Disposal

- What if Reprocessing Project is Given Up?
- Economic Study Associated with Periodic Revision of "Long Term Plan" by Japan Atomic Energy Commission.
- Major Technical and Non-Technical Challenges were Studied and Assessed.
- Criticality Issue was Identified As One of the Major Uncertainties of the Study, since No Safety Evaluation Criteria for Pu etc's Criticality Prevention had not yet been established.

Uncertainty of Disposed SF Criticality

- No Safety Evaluation Criteria.
- Modeling, Scenario and Phenomenology are not Well Established.

From the Viewpoint of Nuclear Data :

- Very Long Term (~10^3y) Transient of Isotopics.
- Integral Validation Difficulty

Conclusions (1)

- Nuclear fuel cycle consists of wide spectrum of fissile systems. Variety of resonance nuclides and neutron spectra are to be covered.
- Better qualified codes and nuclear data could improve criticality safety design criteria, and give more competitive edge to nuclear fuel cycle.

Conclusions (2)

- Burnup credit is the major front of criticality safety design of spent fuel(SF) systems.
- Level-2 burnup or FP credit is promising, whose efficiency depends on uncertainty of SF characteristics and their prediction.

Conclusions (3)

- Integral tests for SF systems are difficult by nature, microscopic validation would be of more importance.

- In SF disposal study, criticality issue was identified as one of the major uncertainties. If its reduction was necessitated, improvement of FP and actinides data would play a certain role.