NUMERICAL SIMULATIONS OF NEUTRON TRANSPORT IN SPENT LWR FUEL ASSEMBLIES FOR BURN-UP CREDIT APPLICATION TECHNIQUES

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Abstract

Numerical simulation of neutron transport in LWR FA are performed to develop confirmation methods of burn-up and reactivity depletion.

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Principle results
Neutron leakage probability P_L varies little with burn-up.

The constant P_L is preferable for
total neutron yield measurements (PWR & BWR)
source multiplication method (PWR).

Outline

I. Introduction
II. 3-D Nuclide Density Distribution
III. Total Neutron yield (Passive)
IV. Source Multiplication
V. Conclusion and Future Work

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Requirement for Burn-up Credit for LWR Operation

- Spent fuel assemblies (FA) are discharged every cycle.
  1/5~1/3 core equivalent
- A discharged spent FA is stored in a rack in spent fuel pond (SFP) until radiation heat decays out.
  capacity of SFP for storage is 4/3 core eq. in older PWR
- Well cooled FAs must be shipped steadily.

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To reduce the risk of shortage of the vacant racks,
enhancement of capacity of SFP and transport cask is preferable.

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For the enhancement, application of Burn-up credit (BUC) is effective.

Burn-up Confirmation and Current Subject

For application of BUC for the transport,
a confirmation measurement is required for every FA
whether the declared burn-up / reactivity depletion of the FA
based on reactor record is adequate.

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Confirmation methods and devices had been developed.
FORK, FORK+(US), PYTHON, NAJA(France), BUM(Japan), etc

However,
current confirmation devices are calibrated utilizing postulated burn-up or nuclide density based on reactor record.

Errors originated in utilizing reactor records have been subtracted from the credit.

Objective

To develop confirmation methods of burn-up and reactivity depletion of FA which dose not require the reactor record data for calibration in order to enhance BUC.

For the purpose,
numerical simulation studies were performed.

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FA of which nuclide densities are calculated

For the simulation study,
3-D nuclide density distributions in typical FA are evaluated.

Fuel & Core type and calculation codes

Fuel type	Burn-up (BU) (MWd/kgHM)	Core Type	Codes, assembly/core
PWR 17x17 UO_2	Max < 55	3Loop 157FAs	AEGIS /SCOPE-2
PWR 17x17 UO_2+Gd	Max < 55	3Loop 157FAs	AEGIS /SCOPE-2
BWR 9x9 UO_2+Gd	Max = 55 Ave. = 45	ABWR 872FAs	TGBLA03 /LOGOS ver.5

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Calculation Scheme

(1) Equilibrium core designing
typical history of
power, void fraction(VR), moderator density, CR, etc.
PWR: Full core calc. with pin-wised mesh (*SCOPE-2*)
BWR: Full core calc. (*LOGOS ver. 5*)

(2) Selection of typical FA
PWR: FA of max. burn-up (BU)
BWR: FA of average BU and VR.

(3) 2-D neutron transport & burn-up calc. for *assembly nodes*
(Divide the FA into 24 equal length, vertically)
PWR (*AEGIS*)
BWR (*TGBLA*)

→ 3-D pin by pin nuclide densities at the end of each cycle.

Calculations are supported by NFI, NEL, and GNF-J.

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Nuclide List

Actinides	^{234}U ^{235}U ^{236}U ^{238}U ^{237}Np ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu ^{241}Am ^{242m}Am ^{243}Am ^{242}Cm ^{243}Cm ^{244}Cm
Fission Products	^{95}Mo ^{99}Tc ^{101}Ru ^{106}Ru ^{103}Rh ^{106}Rh ^{109}Ag ^{131}Xe ^{133}Cs ^{134}Cs ^{137}Cs ^{144}Ce ^{141}Pr ^{143}Nd ^{145}Nd ^{148}Nd ^{147}Pm ^{147}Sm ^{149}Sm ^{150}Sm ^{151}Sm ^{152}Sm ^{153}Eu ^{154}Eu ^{155}Eu
Burnable Poison	^{154}Gd ^{155}Gd ^{156}Gd ^{157}Gd ^{158}Gd ^{160}Gd

(yellow) not evaluated for BWR

(orange) not evaluated for PWR

In the following analysis,
nuclide densities 10 years after discharged are used.

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Total Neutron Yield and Neutron Leakage

Total neutron yield originated in actinide neutron source accumulated in FA is considered a good indicator of BU.
Passive measurements had been performed conventionally

How to measure it without knowledge of BU?

- Total neutron yield = absorption inside FA
  + absorption outside FA
- Horizontal buckling B_h^2 of LWR FA is large
  = neutrons horizontally leak
- Neutron detection outside FA is easier than inside it

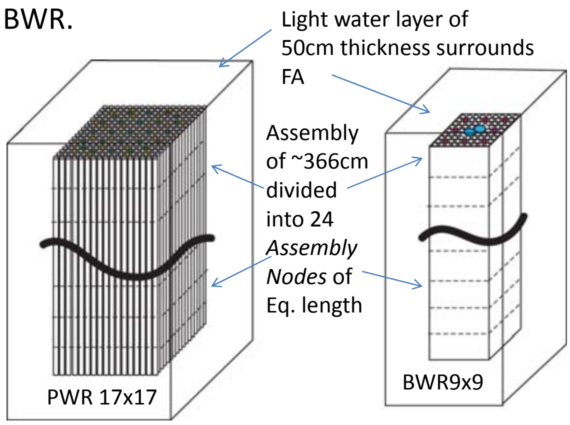
Ratio of absorption outside FA to total neutron yield
= neutron leakage probability P_L is investigated.

Fixed Source Calculation for Isolated FA

- Nuclide density for each pin divided into 24 segments
- Grid spacer, nozzle, tie-plate are neglected
- 50cm thick light water surrounds the FA
- Channel box for BWR.
- 300K, 1atm

SOURCES4C for spectra and yield from radioactive decay of actinides

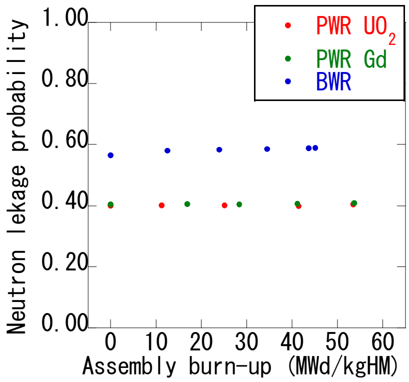
MCNP-5 for neutron multiplication & transport with ENDF/B-VI lib.



Leakage Probability

P_L of PWR FAs
- almost constant to BU.
- from 0 to 55MWd/kgHM

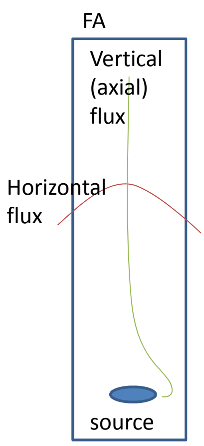
P_L of BWR FAs
- slightly increases with BU
- but max/min ratio < 1.042
- from 0 to 48 MWd/kgHM



In typical PWR and BWR FAs, P_Ls are almost constant to BU.

Mechanism of Constant P_L

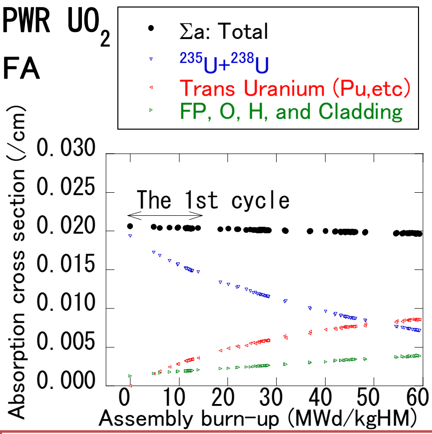
- In FA, $B_h^2 >> B_z^2$ & $B_h^2 > B_m^2$
  - B_z^2 : axial buckling
  - B_m^2 : material buckling
- FA ≈ homogenous rectangular shape core.
- Putting a point source, neutron flux
  - → $\exp(-\gamma z) \times \cos(B_h x) \times \cos(B_h y)$
  - γ: spatial decay constant
- $P_L \approx DB_h^2 / (DB_h^2 + \Sigma_a)$
  - D: diffusion constant, Σ_a: absorption cross section

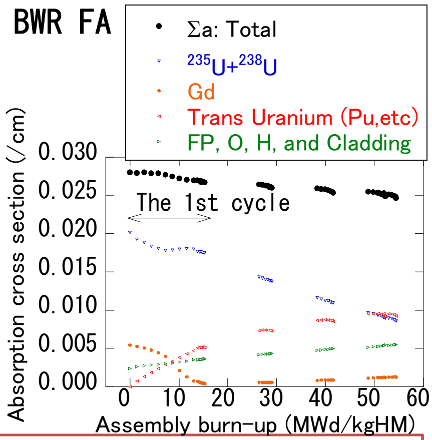


Relations of Σ_a & D to BU are key issue of constant P_L.

Macroscopic Cross Section

Calculation of Σ_a and its components for assembly node.



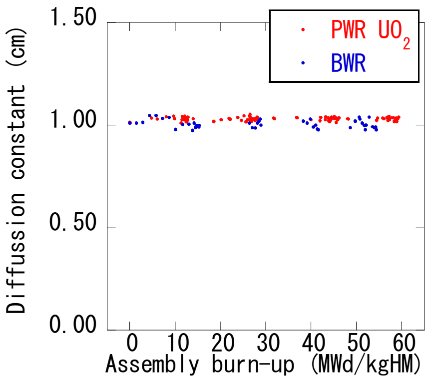


Σ_a varies little with BU.
Decrement of U component is compensated by TU and FP ones.

Diffusion Constant

D is approximated for each assembly node.

- ○ Extension of assembly node
- ○ Put a source.
- ○ Fixed source calc. for ϕ & J
  - ϕ: neutron flux
  - J: neutron current
- ○ $J = - D (d\phi / dz) \phi$

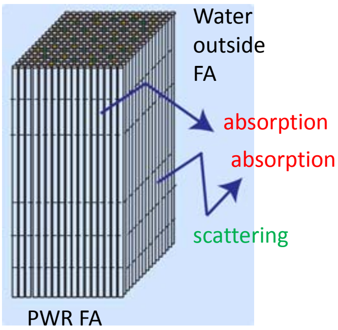


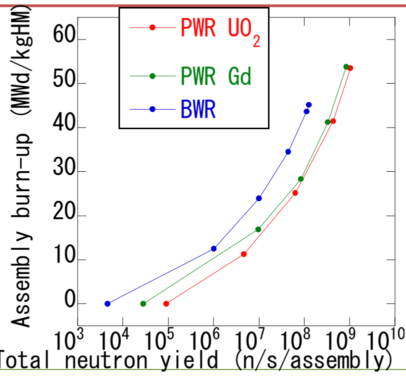
D is almost constant to BU.

Constant Σ_a and D leads to constant P_L.

P_L and Total Neutron Yield Measurements

Total neutron yield can be deduced by
  Quantification of neutron absorption outside FA and constant P_L.
Calibration can be done at 0MWd/kgHM without reactor record.
Total neutron yield is available for BU confirmation.





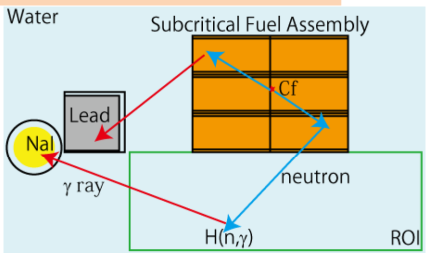
Quantification of Neutron Absorption Outside FA

~ Contents in this sheets had been published in Ann. Mtg. of AESJ (Mar. 2011)

Quantification of neutron absorption outside FA by counting the 2.2MeV γ ray.
  No perturbation to neutron flux.
  Independent to neutron spectrum

The 1st test had been performed in Kyoto Univ. Critical Assembly

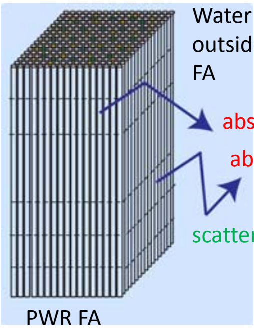
Neutron absorption in water in ROI is measured within accuracy of 7%.

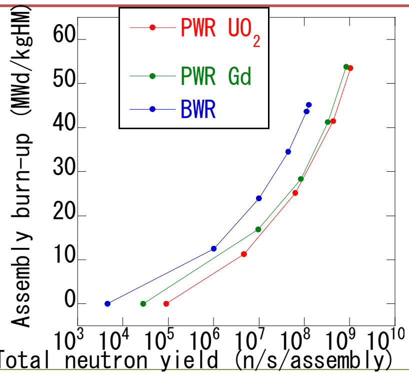


Quantification of neutrons outside FA is principally available.
The total neutron yield can be deduced with the constant P_L.

P_L and Total Neutron Yield Measurements

Total neutron yield can be deduced by
  Quantification of neutron absorption outside FA and
  constant P_L.
Calibration can be done at 0MWd/kgHM without reactor record.
Total neutron yield is available for BU confirmation.





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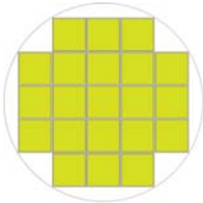
Subcriticality indices

There are several indices of burn-up reactivity depletion,

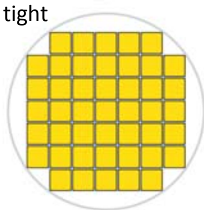
k_{sub}: subcritical multiplication factor
γ^2: axial buckling
α: prompt neutron decay constant

Among them, k_{sub} by the source multiplication method is most widely used in reactor experiments.

PWR tight Cask 21FA



BWR tight Cask 45FA



Utilizing the 3-D nuclide density distribution, k_{sub} is calculated for an isolated FA and compared to effective multiplication factor k_{eff} of *tight casks* which are filled with FAs of the same nuclide density.

Source Multiplication in FA

Source condition:
PWR: all control rod guide tubes are filled with ^{252}Cf
BWR: ^{252}Cf rod of 3.6m long is set 1cm outside channel box

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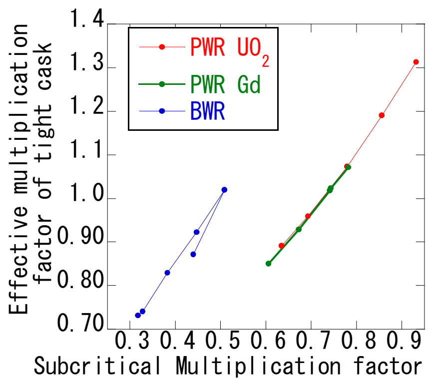
BWR:
k_{sub} can not be used for prediction of k_{eff} of tight cask.

PWR:
k_{sub} for FA uniquely relates to k_{eff} of the tight cask.



Source Multiplication and P_L of PWR FA

P_L for PWR
(UO_2 & UO_2+Gd)
in which ^{252}Cf is inserted
is calculated

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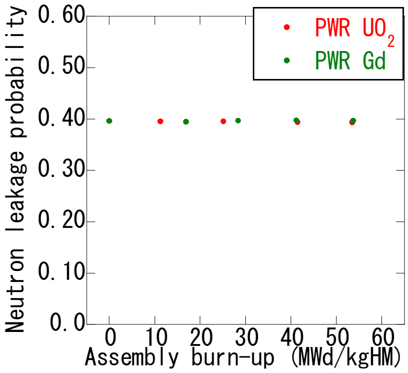
P_L varies little with BU.



= Total neutron yield & k_{sub} can be measured by quantification of neutron abs. outside FA.

Calibration (determining detector efficiency and ^{252}Cf source strength) can be done with fresh FA. It does not require reactor record.

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Conclusions

- 3-D nuclide density distributions in typical FA
- Numerical simulation of neutron transport in LWR FA
- P_L of isolated FA in water is almost constant with BU both for accumulated actinide source and inserted ^{252}Cf one.
- With the constant P_L, the total neutron yield can be deduced by measuring neutron absorption outside FA.

  Calibration can be done without reactor record.
- In passive measurements of PWR & BWR FA, total neutron yield certifies the Burn-up of the FA.
- In source-multiplication measurement of PWR FA, the total neutron yield provides k_{sub} which certifies reactivity depletion.

Future Work

Demonstration of measurement of the total neutron yield
utilizing the constant leakage probability P_L
with the proposed γ ray measurement technique
shall be done for commercial spent fuel assembly.

Thank you for your attention.

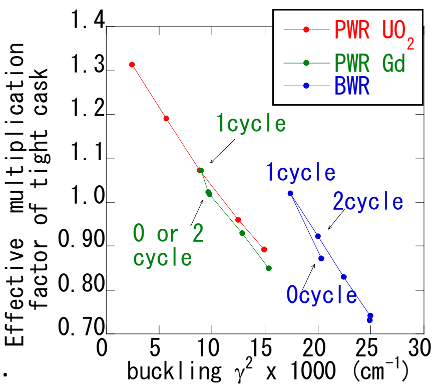
Exponential Experiments

γ^2 of the middle height of FAs are approximately calculated and compared to effective multiplication factor k_{eff} of *tight casks.*

PWR UO_2 FA:
γ^2 is good index of k_{eff}

PWR UO_2+Gd FA:
γ^2 can be index of k_{eff} of tight cask, but slope of k_{eff} vs γ^2 changes when Gd is depleted

BWR FA:
γ^2 can not be index of of k_{eff} .



The exponential experiments is more influenced by Gd contents than the source multiplication method.

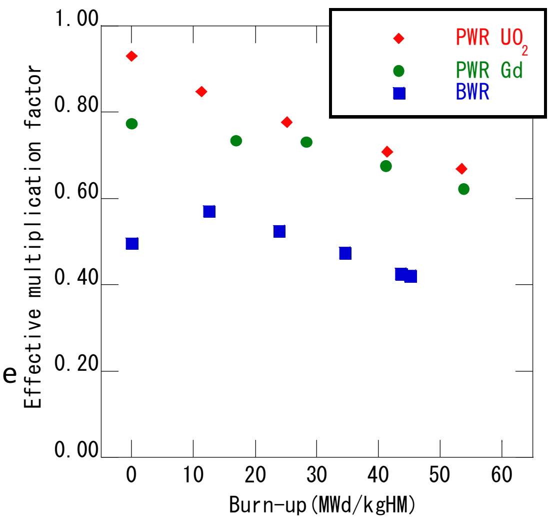
Eigenvalue of FA isolated in water

PWR-UO_2 without Gd
- k_{eff} monotonously decreases

PWR with Gd
- BU-k_{eff} swing is smaller than PWR-UO_2

BWR
- max. k_{eff} is at 1-cycle irradiated case

The trend depends on the amount of Gd.



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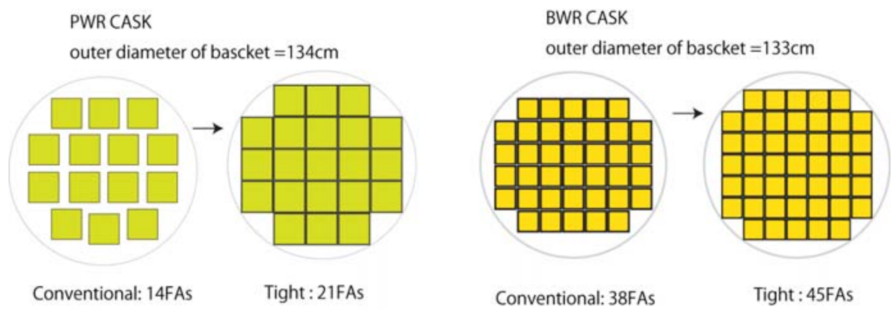
Postulated transport casks

Postulated transport casks in which only well irradiated FAs can be loaded.

PWR: 21FAs BWR: 45 FAs

Diameter of the casks are identical to conventional ones in which FAs of new fuel without Gd can be loaded



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Eigenvalue of Postulated Tight Casks in which FAs of the same nuclide densities are filled

Discharge	PWR UO_2, 21 FAs		PWR Gd, 21 FAs		BWR, 45 FAs	
Cycle	k_{eff}	error	k_{eff}	error	k_{eff}	error
0	1.3133	0.0002	1.0718	0.0003	0.8711	0.0002
1	1.1909	0.0003	1.0230	0.0003	1.0194	0.0003
2	1.0726	0.0003	1.0181	0.0003	0.9223	0.0003
3	0.9594	0.0003	0.9289	0.0003	0.8294	0.0002
4	0.8915	0.0003	0.8496	0.0002	0.7409	0.0002
5					0.7311	0.0002

We need to discriminate the FA of lower BU by measurements we will develop.