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Nuclear mechanism for TRU burning oxide fueled core in Advanced Recycling Reactor

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ABSTRACT

This paper presents an approximation approach to predict the core characteristics based on parametric survey and an analysis of nuclear mechanism in a conceptual nuclear design for enhanced transuranics (TRU) burning mixed oxide fueled and sodium cooled fast reactor which can be realized in the near future. The design study of Advanced Recycling Reactor was conducted in the context of the program for the industry in Global Nuclear Energy Partnership initiatives, including a core in the first plant for demonstration and cores with enhanced TRU burning capability for the future plants. Both concepts for the first plant; low core height and large volume fraction of structure are deployed, seeking small TRU conversion ratio (CR)* and small void reactivity which are crucial in the design, but different approaches. In this paper, the TRU CR and the sodium void reactivity have been approximated with a single equation in these concepts, based on the theoretical formula related to the chain reaction in the reactor and the calculation results. Shortening the core height and increasing the structure volume fraction will enhance TRU enrichment through increased neutron leakage and capture, which will reduce a ratio of U-238 to sum of Pu-239 and Pu-241 so that TRU CR decreases from 0.78 to 0.57. A small ratio of sodium loss to plutonium fissile will be effective also in the reduction of positive reactivity effect by spectral hardening. On the other hand, when this ratio and geometrical buckling of flux reduce, negative contribution by the neutron leakage becomes small. Theses relations related to the positive void reactivity have been formularized by the approximation with few parameters within several percent respectively as well as the TRU CR, indicating that one of dominating parameters is the ratio of sodium loss to plutonium fissile in the void reactivity at large fast reactors. * = (1 − net loss of TRU/loss of heavy metal).

1. Introduction

New actions have appeared among several nations to develop and deploy advanced nuclear fuel cycle and reactor technologies. Japan decided the project of Options Making Extra Gains from Actinides and fission products (OMEGA) in 1988, and France started the project of Separation and Incineration (SPIN) in 1991. These projects aimed to develop Minor Actinide (MA) nuclides burning related technology. On the other hand, the US promoted the Advanced Fuel Cycle Initiative (AFCI) to develop advanced fuel cycle technologies since 2003. Furthermore, the Global Nuclear Energy Partnership (GNEP) initiative tried to provide reliable and emission-free energy with less of the waste burden of older technologies and with less proliferation risks so that Yucca Mountain becomes the only repository in the US in this century (DOE, 2007). In the GNEP systems, the mission of Advanced Recycling Reactor (ARR) is to reduce the risk of nuclear proliferation by burning plutonium, and the burden on the geological repository by transmuting transuranics (TRU) while generating energy through system with sodium cooled fast reactor (SFR) and fuel cycle (Ikeda et al., 2009a; Mito et al., 2009).

The approach of design chosen is steady and practical to harmonize several aspects; TRU burning, fuel integrity, manufacturing, reprocessing, economy, and safety because the first priority is to start the operation in 2025 based on the proven technology with well-defined innovations. A concept of the ARR is presented based on the experience of SFR development in Japan and discussion with specialists of France and the US, and low core height and large volume fraction of structure are chosen to be deployed. This design is not special but practicable, which is able to come true surely and is an appropriate choice for the industry that is responsible for design, safety assessment, construction and support of operation.

It must be important and desirable that core concept as a base of design is not only undoubted, but also well-understood and easily forecastable because it will lead us to smooth agreement and smart judgment in the design works. Actually in Japan, equivalent fissile coefficients that are relative ratios of reactivity worth in actinide nuclides are used in the nuclear designs in PWR mixed uranium–plutonium oxide (MOX) and SFR in place of detailed calculation or before. Further, it contributes toward simplifying the specification of MOX fuel including several kinds of plutonium.

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Nomenclature

Abbreviations

AFCI	Advanced Fuel Cycle Initiative
ARR	Advanced Recycling Reactor
ARR_1	the first plant of ARR
BCR	backup control rod
CR	conversion ratio
EFPD	Effective Full Power Days
FAIDUS	Fuel Assembly with Inner Duct Structure
GNEP	Global Nuclear Energy Partnership (initiative)
GWd/t	combustion ratio of generated thermal energy per 1 ton of actinide
HM	heavy metal
INL	Idaho National Laboratory
JSFR	Japan Sodium Cooled Fast Reactor
MA	Minor Actinide
MLHR	maximum linear heat rate
MOX	mixed (U, Pu) oxide
MW_{th}	thermal power unit of mega watt
OMEGA	Options Making Extra Gains from Actinides and fission products
PCR	primary control rod
SFR	sodium cooled fast reactor
SPIN	Separation and Incineration program
TD	theoretical density
TRU	transuranics

List of symbols

all	uranium and TRU
B^2	$\Delta\varphi/\varphi$
B^{*2}	$\Delta\varphi^*/\varphi^*$
B_r	radial buckling weighted with the real flux
B_r^*	radial buckling weighted with the adjoint flux
B_z	axial buckling weighted with the real flux
B_z^*	axial buckling weighted with the adjoint flux
C	numerator of elimination of sodium capture and change of self-shielding contribution to void reactivity defined in Eq. (10b)
D^g	diffusion coefficient of the gth
D_{outer}	thickness of outer core
F_{outer}	fraction of outer core in active core
H	core height
L	numerator of increased neutron leakage contribution to void reactivity defined in Eq. (8b)
M	numerator of spectral hardening contribution to void reactivity defined in Eq. (7c)
N235	number density of U-235
N238	number density of U-238
N239	number density of Pu-239
N241	number density of Pu-241
N_0235, N_0238, N_0239, N_0241	initial number densities of U-235, U-238, Pu-239, and Pu-241
N_X	number density of nuclide X
Puf	average Pu fissile enrichment over active core
Puf_o	Pu fissile enrichment at the outer core
R_{HM}	mass consumption of HM between beginning of life and end of life
R_{TRU}	mass consumption of TRU between beginning of life and end of life
s	the neutron generation from the other actinide nuclides; uranium, americium, neptunium, Pu-238, Pu-240, and Pu-242
S	source term defined in Eq. (7b)
tru	TRU; neptunium, plutonium, americium, and curium
U	weight fraction of uranium per heavy metal in fuel
T_D	theoretical density ratio of fuel smear density
V_c	volume fraction of coolant
V_f	volume fraction of fuel
V_s	volume fraction of structure

Greek symbols

δ	extrapolation distance
δ_r	$\chi_g \sum_{g'} \delta f (\nu\Sigma_f)_{g'} \phi_{g'} - \delta f \Sigma_{ag} \phi_g$ in Eq. (10b)
δN_{Na}	change of sodium number density at sodium loss
$\delta f(\nu\Sigma_f)_{g'}$	change of macro fission cross section through self-shielding at the sodium loss
$\delta f \Sigma_{ag}$	change of macro absorption cross section through self-shielding at the sodium loss
ΔV_v	volume fraction of sodium loss
ν	number of neutrons emitted per fission
$\bar{\nu}\bar{\sigma}_f$	average of $\nu\sigma_f$ over Pu-239 and Pu-241
σ_a^{238}	absorption cross section of U-238
σ_c^{238}	capture cross section of U-238
$\sigma_{cg,Na}$	sodium capture cross section
σ_f^X	fission cross section of nuclide X
σ_f^{gX}	fission cross section at the group gth of nuclide X
σ_f^{235}	fission cross section of U-235
σ_f^{239}	fission cross section of Pu-239
σ_f^{241}	fission cross section of Pu-241
σ_{sr}^g	scattering removal cross section of sodium
σ_{tr}^{gX}	micro transport cross section of nuclide X
σ_{tr}^{gNa}	micro transport cross section of sodium
$\sigma^{g'\to g}$	scattering matrix of sodium
ϕ_g	neutron flux
ϕ_g^*	adjoint flux
χ_g	fission neutron spectrum

It will be accepted among the specialists in the nuclear design of SFR that two concepts are appropriate for small TRU conversion ratio (CR) and small void reactivity core from their experiences and designed cores in the past. However, the related mechanism has not been analyzed and the relation with specifications has not been examined yet. Therefore, the present study has been undertaken in order to generalize the relations of the TRU CR remarked afterward and the sodium void reactivity using dominating parameters and make their mechanism clear for nuclear design.

At first Section 2 shows an overview of the conceptual design for ARR. Theoretical relations are examined to seek ruling parameters to reduce the TRC CR and the sodium void reactivity in Section 3, the approximation equations are derived from results of parametric survey and these equations are tested by other cases in Section 4. From these examinations, mechanism and new findings are discussed in Section 5. Finally conclusions are remarked.

2. Presented core concept of Advanced Recycling Reactor

The first to be explained is an overview of the nuclear design of ARR; suppositions, the definition of TRU CR, and the final case of parametric survey.

2.1. Requirements and suppositions

Requirements in the first plant of ARR (ARR_1) are set in the design study as follows. The ARR is designed to burn a large