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Transmutation of americium in a large sized sodium-cooled fast reactor loaded with nitride fuel

Youpeng Zhang *, Janne Wallenius, Mikael Jolkkonen

Royal Institute of Technology, Department of Reactor Physics, AlbaNova University Center, 10691 Stockholm, Sweden

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ABSTRACT

We have performed the transient analysis of a large sized sodium-cooled reactor loaded with nitride fuel modified by different fractions of americium. Unprotected loss-of-offsite power, unprotected loss-of-flow and unprotected transient-over-power accidents were considered as the postulated transient and simulated with the SAS4A/SASSYS code based on the geometrical model of a BN1200 type sodium-cooled fast reactor (SFR) with the rated power of 2900 MW_{th}. Safety parameters used in the transient simulation model were obtained from the SERPENT Monte Carlo calculations.

If 100 K margin to fuel melting was maintained, 2 wt.% and 6 wt.% Am content could be permitted for the nitride fuel loaded BN1200 operating with the maximum linear power rating of 37 kW/m and 33 kW/m respectively, providing Am transmutation rates of 3.9 kg/TW h_{th} and 13.7 kg/TW h_{th}. These transmutation rates are more than 6 times higher than the ones previously reported for oxide fuel loaded BN600 operating with the maximum linear power ratings of 37 kW/m and metallic fuel loaded Integral Fast Reactor (IFR) operating with the maximum linear power rating of 33 kW/m respectively, being their reference cases with permitting 1 wt.% Am content.

1. Introduction

Although multi-recycling the transuranic isotopes in light water reactors may reduce the radio-toxic inventory in spent fuel discharged from light water reactor (LWR) by a factor of 200 (Delpech et al., 1999), it also leads to accumulation of the strong neutron emittor – ^{252}Cf, since fission probabilities of minor actinides (e.g., americium, curium) are very low in thermal neutron spectrum.

Instead, in a fast neutron spectrum, the equilibrium concentration of Cf and the corresponding neutron activity of spent fuel will be 2–3 orders of magnitude lower than in thermal reactors, which enables the fast reactors to be better suited for minor actinides transmutation (Salvatores et al., 2005). The introduction of americium will degrade core safety parameters such as the Doppler effect, coolant temperature coefficient and the effective delayed neutron fraction, and the transmutation performance of fast critical reactors is highly sensitive to the exact content of americium in the fuel. Hence, a consistent methodology and criteria have been established to determine the maximum fraction of americium that may be loaded into the fuel for an arbitrary fast neutron Gen-IV system (Zhang et al., 2010).

In this paper, the upper limit of americium introduction into a large sized sodium cooled reactor loaded with nitride fuels having a fixed Pu fraction but different americium contents were assessed by applying these criteria and methodology.

Geometrical dimensions of the BN1200 design was adopted and implemented in the SAS4A/SASSYS code in order to investigate transient responses of different fuel cases with various Am concentrations. Safety parameters needed for the transient analysis were calculated with the Monte Carlo SERPENT code.

2. Tools for simulation

2.1. SERPENT

SERPENT is a continuous energy Monte Carlo burnup code developed by VTT Technical Research Centre in Finland. As in other Monte Carlo codes, such as MCNP, configurations of a fuel assembly or a full core will be described by universe based geometrical models. Neutron transport simulations was performed by applying the Woodcock Delta-tracking method. The burnup calculation is realized by applying the Transmutation Trajectory (TTA) Method and is defined using time or burn rate steps.

Comparing to existing Monte-Carlo codes, such as MCNP, the SERPENT Monte-Carlo code has two major merits: first of all, the calculation time needed to provide a given statistical uncertainty has been reduced by a factor of 5–15, which is due to evenly subdividing the cross section energy grid. Secondly, homogenized constants, e.g., diffusion rates and effective delayed neutron fractions,

* Corresponding author. Tel.: +46 07 3893 1124; fax: +46 08 5537 8216.
E-mail address: youpeng@neutron.kth.se (Y. Zhang).

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can be calculated and provided in the output file of SERPENT, including statistical error estimates (Leppänen, 2005).

2.2. SAS4A/SASSYS

The Safety Analysis System (SAS) was initially composed by D.R. Macfarlane from the Argonne National Laboratory (ANL). The version we have used in this paper is 3.1. This version SAS code is eligible to verify reactor designs loaded with oxide and metallic fuels, based on simulations of transients, such as Loss of Flow (LOF), Transient over Power (TOP) and Loss of Heat Sink (LOHS).

$$\frac{dp(t)}{dt} = \frac{\rho(t) - \beta(t)}{\Lambda(t)} \times p(t) + \sum_k \lambda_k c_k(t) + s(t) \quad (1)$$

SAS4A consists of several interconnected calculation modules, which are used to perform thermal dynamic simulations (thermal conduction and thermal expansion), thermal hydraulic simulations (pressure drop and heat transfer simulation), failure estimations, a simple model of the first loop (including pump, IHX, connection pipe, etc.) and reactivity feedback calculations. The point kinetic model, as expressed in Eq. (1), is applied to calculate reactivity feedbacks from different mechanisms. In Eq. (1), $p(t)$, ρ(t), $\beta(t)$, Λ(t), λ_k, $c_k(t)$ and $s(t)$ stand for the power, the reactivity, the delayed neutron fraction, the prompt neutron life time, group-wise decay constants, group-wise decay precursor fractions and the source term of core as the function of time (Cahalan, 2002).

Thermal hydraulic and thermal dynamic modules calculate temperature changes of core components, such as fuel, cladding and coolant, leading to reactivity feedbacks implemented into the point kinetic model. For deep subcritical cores, deviations due to the usage of the point kinetic method for 163 transient simulations was reported to be less than 5% (Eriksson et al., 2005). Meanwhile, it has also been reported that the precision of the point-kinetic model for local perturbations might decrease significantly when the core is approaching criticality. This is mainly due to the slower migration of spatial perturbation in the critical system, comparing to the deep sub-critical system. However, if ULOF and UTOP transients were assumed to cause non-local power perturbations, simulation results provided by the SAS4A code can still be considered to be reliable (Cherny et al., 2000).

3. Implementation of the BN1200 model

3.1. 3-D SERPENT model of BN1200

The reference BN1200 design is a breeder core with 2900 MW_{th} power rating and 1.33 breeding ratio. Both oxide and nitride fuels have been recommended for this design. The power density and maximum linear power rating of it has been set at 230 MW/m^3 and 42 kW/m (Poplavsky et al., 2009). In the present work, the Pu content in the nitride fuel was increase from 12.4 wt.% to 13.0 wt.% in order to achieve a conversion ratio close to unity and still maintain the reactivity loss during burnup less than total shim rod worth, equalling to approximately 5000 pcm. Radius of fuel pin was set at 9.3 mm. Fuel pellet radius was adjusted to provide a smear density of 90%. Since the reported fission gas release rate of nitride fuel irradiated in fast spectrum is more than 5 times smaller than those have been reported for oxide fuel and metallic fuel, the gas plenum length could be reduced from 800 mm to 200 mm, aiming at a lower pressure drop in fuel pin bundle and also higher mechanical strength of fuel pin structure. Other geometrical details were adopted from the reference IFR design and summarized into Table 1.

I.D. and O.D. in Table 1 represent the inner diameter and outer diameter respectively. Shielding material, consisting of 10 vol.% B_4C and 90 vol.% concrete, was loaded above the core active zone in order to reduce the irradiation damage on upper structures. The fuel zone is generally in hexagonal shape loaded with 456 subassemblies. Several drive fuel assemblies at outer rings were substituted by reflector assemblies to reduce the radial form factor. Depleted uranium loaded into the axial blanket was substituted by the same stainless steel used as structural material. Accessories, e.g., assembly diagrids and wire spacers between fuel pins, were neglected in our SERPENT model as shown in Fig. 1.

Table 1
Dimensions of the present design, in mm (Wade and Hill, 1997, Poplavsky et al., 2009).

Fuel pin dimensions, mm			
Pellet diameter	7.70	Active length	850
Clad I.D.	8.10	Lower Blanket	400
Clad O.D.	9.30	Plenum position	Below
Pin pitch	10.85	Plenum length	200
Upper shielding	650	Plug length	55
Assembly dimensions, mm			
No. of SA	432	Nozzle length	870
Flat-to-flat	181	Total length	3895
Duct pitch	190	Duct thickness	3.50
Head length	760	No. of pins	271

The isotopic vectors of Pu and Am, listed in Table 2, were adopted from the spent fuel discharged from LWR after 50 MW d/kg HM burnup and 10 years' cooling.

Among all minor actinides, only transmutation of Am is studied in this paper, since only negligible radiotoxicity in spent fuel discharged from LWRs is contributed by Np nuclides (Westlén, 2007). Besides, the concentration of Cm in spent LWR fuel is only 1/6 of that of Am, leading to negligibly small impact on safety parameters. Hence, it is more valuable to parameterize the impact solely from Am introduction into the fuel, excluding Np and Cm.

3.2. SAS4A/SASSYS model of the primary circuit

In the SAS4A/SASSYS code, the reactor core was simplified to be one hottest fuel channel containing the hottest fuel pins, one average fuel channel containing the rest fuel pins, one control rod channel and one channel containing shielding assemblies.

As shown in Fig. 2, the primary loop of IFR could be generally divided into 5 liquid segments, including coolant inlet section, core active region, coolant outlet section, intermediate heat exchanger (IHX) and primary pump. In each segment, single phase coolant flow exists. The primary pump is assumed to be located at the cold leg as BN600 and IFR designs (Nuclear Power Technology Development Section, 1996), in order to reduce the thermal creep effect on pump blades.

The core inlet temperature and the primary coolant velocity in fuel pin bundle were set at 623 K and 9.0 m/s respectively (Nuclear Power Technology Development Section, 1996). Then the coolant flowrate per fuel pin can be calculated as 0.28 kg/s.

In order to make a consistent study, pressure drop through all components in the nitride fuel loaded BN1200 type core was assumed to be 1.0 MPa, including 0.35, 0.5 and 0.15 MPa in coolant inlet section, core active region and coolant outlet section respectively, which are the same as the IFR design (Nuclear Power Technology Development Section, 1996).

4. Neutronic parameters

The impact from Am introduction on core performance parameters and reactivity coefficients were calculated with the SERPENT Monte-Carlo code. In this paper's study, the effective delayed neutron fraction (β_{eff}), the Doppler constant of fuel (K_D), the coolant