Analysis and Passive Safety Design of a Gen-IV Reactor with Nitride Fuel and Sodium Coolant

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1. Introduction

Gen-IV reactors require inherent safety through passive systems. This sodium-cooled design with nitride fuel addresses this via natural convection cooling and optimized rod geometry to manage fission gas pressure. The design ensures 20 hour survival during transients without active systems.

2. Natural Convection Cooling and Fuel Rod Design

2.1. Coolant Properties

The thermophysical properties of sodium were evaluated at the average coolant temperature of 785.15 K (512.5 °C) using empirical models:

$$\rho(T) = 1011.8 - 0.22054T - 1.9226 \times 10^{-5}T^2 + 5.6371 \times 10^{-9}T^3 \quad (1)$$

$$\mu(T) = \exp(-6.4406 - 0.3958 \ln T + 556.835/T) \quad (2)$$

$$c_p(T) = 1658.2 - 0.8479T + 4.4541 \times 10^{-4}T^2 - \frac{2.9926 \times 10^6}{T^2} \quad (3)$$

The calculated values are:

- Density: $814.4\,\mathrm{kg/m^3}$
- Dynamic viscosity: $2.13 \times 10^{-4}\,\mathrm{Pa\,s}$
- Specific heat capacity: $1255.3\,\mathrm{J/(kg\,K)}$
- Thermal expansion coefficient: $2.41 \times 10^{-1}\,\mathrm{kg/(m^3\,K)}$

These properties were used to evaluate the flow conditions and heat transfer in natural convection during a station blackout scenario.

2.2. Rod Geometry

Rod pitch and diameter were derived analytically using:

$$P_{\mathrm{rod}} = \frac{8\dot{Q}_{\mathrm{ch,res}}}{\pi \Delta T \mu c_p Re_{\mathrm{min}}} \quad (4)$$

$$D_{\mathrm{rod}} = \frac{P_{\mathrm{rod}}}{2} + \sqrt{\left(\frac{P_{\mathrm{rod}}}{2}\right)^2 \left(\frac{4\sqrt{3}}{\pi} - 1\right) - \frac{4A_{\mathrm{flow}}}{\pi}} \quad (5)$$

Using the natural convection residual heat target of 2.0 kW per channel and a minimum Reynolds number of 4000, the resulting fuel rod pitch was calculated to be 13.58 mm. The corresponding rod diameter was determined as 12.56 mm, yielding a pitch-to-diameter ratio of 1.08. This configuration supports sufficient coolant flow between the rods while maintaining good heat transfer and mechanical integrity.

Table 1: Core Design Parameters

Parameter	Description	Value
P_{rod}	Rod pitch	13.58 mm
D_{rod}	Rod diameter	12.56 mm
P/D	Pitch-to-diameter ratio	1.08
r_{clad}	Cladding radius	6.283 mm
δ_{clad}	Cladding thickness	0.628 mm
H_{fuel}	Fuel height	1.000 m
H_{plenum}	Plenum height	0.852 m
H_{ch}	Channel height	1.852 m
σ_{hoop}	Hoop stress	212.84 MPa
P_{gas}	Gas pressure	21.28 MPa
t_r	Rupture time	20.4 h

2.3. Cladding and Plenum

Using a 20-hour creep rupture constraint at 750 °C, the allowable hoop stress σ_{hoop} was derived from the Larson-Miller parameter:

$$LMP = 21100 - 1.89 \times 10^{-6}\sigma - 0.030 \times 10^{-12}\sigma^2$$

Plenum height was calculated via the ideal gas law assuming 100% Xe, Kr, and He release, with:

$$\frac{r_{\mathrm{clad}}}{\delta_{\mathrm{clad}}} = 10$$

This yields a cladding radius of 6.283 mm and a thickness of 0.628 mm. Using the Larson-Miller creep model and assuming full noble gas release (including 5 % Am), the resulting internal gas pressure was 21.28 MPa, leading to a hoop stress of 212.84 MPa.

The calculated time to rupture of the cladding under the given gas pressure and temperature was 20.4 hours, which exceeds the design requirement of 20 hours. This confirms that the cladding can survive a prolonged station blackout transient without failure.

An iterative solution to the ideal gas law provided a required plenum height of 0.852 m to safely accommodate the generated gases at 750 °C for 20 hours. The fuel column height was set to 1.000 m, resulting in a total active channel height of 1.852 m.

3. Fuel-Clad Gap

With burnup, the fuel will swell and upon contact with the cladding, the hard UN might damage it. To prevent this from happening, the fresh fuel pellets are designed with a gap, the fuel-clad gap, distancing the fuel from the cladding to accommodate the swelling.

At the temperature of interest, the swelling rate of UN is 1.5% per % burnup. As the target peak burnup is 10%, we design the gap to accommodate 15% of volumetric

swelling. This makes $(1.15)^{\frac{1}{3}} = 1.047 = 4.7\%$ of radial swelling.
Therefore,

$$r_{fuel} = \frac{r_{clad} - \delta_{clad}}{1.047} = 5.4[mm] \quad (6)$$

and

$$\delta_{gap} = (r_{clad} - \delta_{clad}) - r_{fuel} = 0.25[mm]. \quad (7)$$

4. Plutonium Fraction

With our calculated parameters, we want to find a Plutonium fraction such that the reactivity swing for a core averaged burnup of 6% FIMA is minimal.
The core contains 5% of Americium. Both the Americium and the Plutonium isotopic compositions are from 5 year cooled LWR spent fuel. The fuel has 5% of porosity and the power density is $40\left[\frac{W}{g_{HM}}\right]$.
Burnup simulations for a 1m high, infinite 2D lattice have been run (black axial BC). The results are shown on Figure 1.

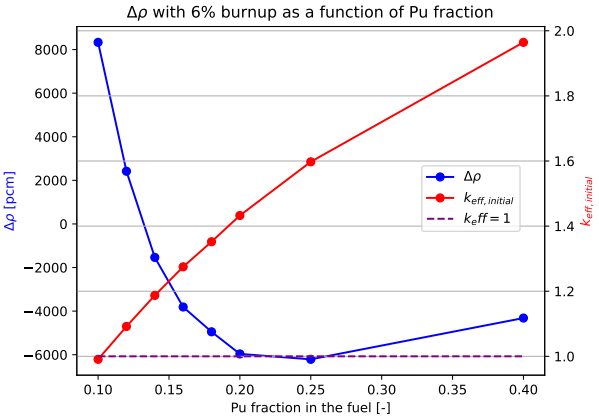

Figure 1: Reactivity swing with 6% burnup as a function of Plutonium fraction.

The Plutonium fraction yielding the lowest reactivity swing is 13.2%, with $k_{eff} = 1.408$, leaving sufficient margin for leakage.

5. Critical Mass and Number of Fuel Rods

To find the critical mass and the number of fuel rods out, multiple simulations with an increasing amount of assemblies have been run, with each assembly made of 61 fuel rods.
The assemblies pitch p_{assemb} is $11.0886[cm]$ and the whole core is immersed in a 3mx3mx3m cube of sodium with black BCs.
Results are shown in Figure 2 and 3. For 5551 fuel pins and 61.53 kg of fissile, the effective multiplication factor is $k_{eff} = 1.003$. Core configurations and their effective multiplication factor are displayed on Figure 4.

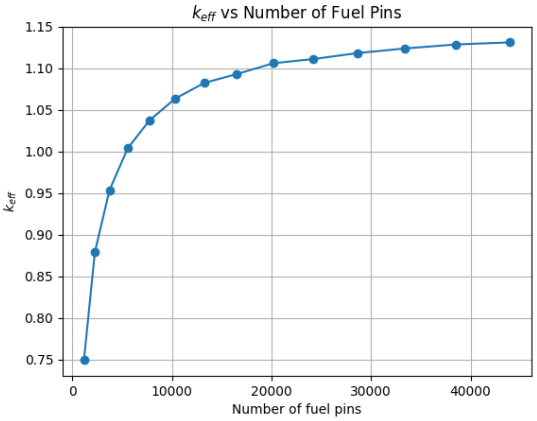

Figure 2: Effective multiplication factor vs number of fuel pins

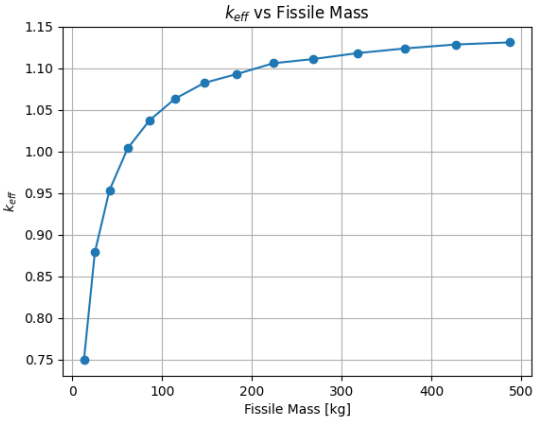

Figure 3: Effective multiplication factor vs mass of fissile

6. Reactivity Coefficients and Reactivity Insertion of going from Forced to Natural Circulation

In a SFR, there are four reactivity coefficients to investigate :

- Doppler broadening
- Axial fuel expansion
- Coolant expansion
- Diagrid expansion

Each reactivity coefficient is investigated separately. The reference input file is at 300K and for each coefficient, a modified input file, modeling one parameter at an higher temperature, has been created and ran. To obtain the reactivity coefficient, the difference in reactivity was divided by the difference in temperature.
How was each input file created ?

- Doppler broadening ($\Delta T = 900$) :
  XS libraries were changed, from .03c to .12c.
- Axial fuel expansion ($\Delta T = 700$) :
  According to online litterature, $\alpha_{UN} = 7.5 *$

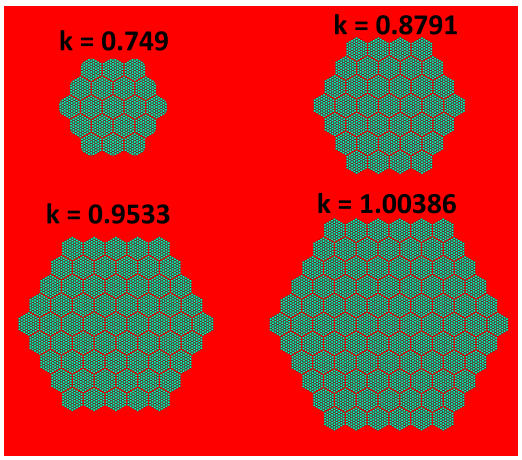

Figure 4: Core configurations and multiplication factors

$10^{-6}\left[\frac{1}{K}\right]$.
So

$$H' = H(1 + \alpha_{UN}\Delta T) = 1.00525[m] \quad (8)$$

and

$$\rho' = \rho\frac{H}{H'} = 12.706. \quad (9)$$

ρ' is a non-physical but useful parameter as it only represents the change in density due to the axial expansion because the fuel radius has been kept constant.

- Coolant expansion ($\Delta T = 700K$):
  The coolant density changes with temperature. $\rho_{Na}(T = 300K) = 0.94\left[\frac{g}{cm^3}\right]$ and $\rho_{Na}(T = 1000K) = 0.78\left[\frac{g}{cm^3}\right]$. (We neglect the fact that the sodium should freeze at 300K, and we used the course-provided correlation). Only the density of the sodium in the core has been changed, not that of the sodium far outside of it.

- Diagrid expansion ($\Delta T = 700K$):
  According to online litterature, $\alpha_{SS} \approx 13 * 10^{-6}\left[\frac{1}{K}\right]$. To account for the thermal expansion of the structure, we increase the distance between the 61 fuel pins assemblies.
  Therefore,

$$p'_{assemb} = p_{assemb}(1 + \alpha_{SS}\Delta T) = 11.189[cm]. \quad (10)$$

	$k_{eff}[-]$	ρ [pcm]	$T[K]$	$\alpha\left[\frac{pcm}{K}\right]$
Reference	1.00892	884.113	300	-
Doppler	1.00384	382.531	1200	-0.5573
Axial	1.00722	716.824	1000	-0.239
Coolant	1.00239	238.43	1000	+0.09116374
Diagrid	1.00472	469.78	1000	-0.59

Table 2: Reactivity measures and coefficients

When going from forced to natural circulation, one can estimate an overall $\Delta T = 150$, leading to

$$\Delta\rho = (\alpha_{Doppler}+\alpha_{axial}+\alpha_{Na}+\alpha_{Diagrid})\Delta T = -194.27[pcm]. \quad (11)$$

7. Conversion Ratio and Minor Actinade Burning Rate

7.1. Conversion Ratio

To evaluate the conversion ratio, the following formula is used:

$$CR_{ip} = \frac{\sum_{A,m}\sigma_c^{(^mA)}C^{(^mA)}\eta^{(^{m+1}A')}}{\sum_{A,m}\sigma_f^{(\grave{i}mA)}C^{(^mA)}\eta^{(^mA)}}$$

In which, using detectors on serpent, all the possible actinides related to the fuel are considered. The result is $CR_{ip} = 1.486$, the conversion ratio considerably larger than 1, so the reactivity produced by capture is larger than the reactivity destroyed by fission.

7.2. Minor Actinide Burning Rate

For the calculation of the minor actinide burning rate, a burn-up of 6% has been considered, as required in the project assignment.

Nuclide	Net mass produced (kg/TWh)
Np237	0.41
Np238	0.014
Am241	-19.39
Am242	0.448
Am242m	0.84
Am243	-11.046
Am244	0.014
Cm242	9.503
Cm244	9.424
Total	-9.783

Table 3: Net mass produced for Np, Am, Cm isotopes per kg/TWh.

Negative values correspond to a effective burning, while positive correspond to a production. While the net mass of Am241 and Am243 is reduced due to their high capture cross section, the net mass of Curium, not present at the beginning, is increased, as their absorption cross section is relatively lower, then a positive net production is reasonable.

8. Decay Heat

A main concern for nuclear safety is decay heat, the residual power produced by the radioactive decay of fission products and minor actinides after reactor shutdown. Although fission reactions stop, this heat remains significant and must be removed to ensure fuel and reactor safety. In this analysis, we evaluate the decay heat behavior between 2 and 20 hours and fit the data to an exponential model of the form $Ae^{-\lambda t}$ to characterize its time-dependent decrease.
Using the MATLAB function `lsqcurvefit` to fit the curve, we obtain the following exponential:

$$1.35 * 10^6 \quad e^{-0.0399\,t}$$

Where the value of λ is given in h^{-1}.
In general, the biggest contributions are shown in Figure 6.

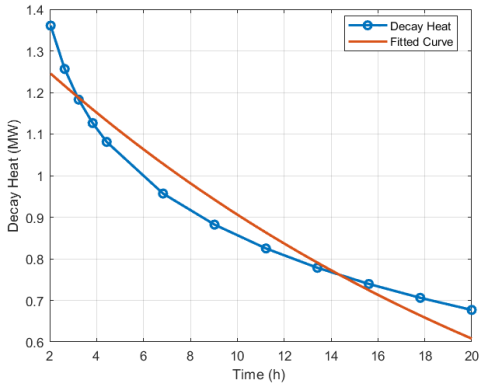

Figure 5: Comparison between decay heat and the modeled curve.

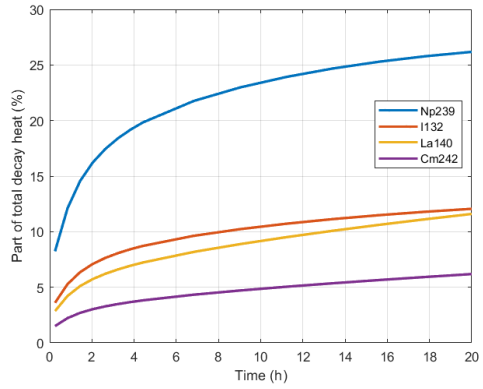

Figure 6: Biggest contribution to decay heat

9. Buffer Mass Volume

This section details the evaluation of the required buffer mass and volume for a sodium-cooled 150 MW_{th} fast reactor operating under passive safety conditions during a station blackout. The analysis is based on the semi-analytical framework introduced in [1] and adapted for sodium coolant.

9.1. Assumptions and Parameters

The thermal power of the reactor is: 150 MW_{th}. Air-side parameters are as follows: air inlet and outlet temperatures of 298 K and 398 K respectively, a buoyancy chimney height $H_{\text{air}} = 20$ m, and an air duct cross-sectional area $A_{\text{air}} = 1.5$ m^2. Sodium coolant properties are evaluated at the average temperature (575°C) between inlet (400°C) and outlet (750°C).

Table 4: Key Parameters for Buffer Calculation

Parameter	Value
β_{eff}	6.5×10^{-3}
$\Delta\rho$	-1×10^{-3}
λ_d	0.0399
a_d	0.12
ΔT_{buffer}	400 K
$c_{p,\text{Na}}$	1340 J/kg/K
ρ_{Na}	827.6 kg/m^3

9.2. Analysis

The time to reach peak temperature τ is calculated as in [1]:

$$\tau = \left(\frac{a_d \cdot \dot{Q}_{\text{nom}} \cdot \sqrt{T_{\text{in}} T_{\text{out}} (T_{\text{in}} + T_{\text{out}})}}{c_{p,\text{air}} \cdot \rho_{\text{air}} \cdot A_{\text{air}} \cdot \sqrt{g H_{\text{air}}} \cdot \Delta T_{\text{air}}^{3/2}} \right)^{1/\lambda_d} \tag{12}$$

Substituting the numerical values yields a time to peak temperature of:

$$\tau = 30.22 \text{ h}$$

The required buffer mass is estimated using the following relation:

$$m_{\text{buffer}} = \frac{\dot{Q}_{\text{nom}}}{\Delta T_{\text{buffer}} \cdot c_{p,\text{Na}}} \left(\frac{0.11 \cdot \beta_{\text{eff}}}{|\Delta\rho|} + \frac{a_d \cdot \lambda_d}{1 - \lambda_d} \cdot \tau^{1-\lambda_d} \right) \tag{13}$$

Evaluating Eq. 13 results in:

$$m_{\text{buffer}} = 74.60 \text{ tonnes}$$

The corresponding buffer volume is:

$$V_{\text{buffer}} = \frac{m_{\text{buffer}}}{\rho_{\text{Na}}} = 91.60 \text{ m}^3$$

Assuming a cylindrical reactor vessel with a diameter of 4 m, the height required to accommodate this buffer is:

$$H_{\text{vessel}} = \frac{V_{\text{buffer}}}{\pi \cdot (D_{\text{vessel}}/2)^2} \approx 7.29 \text{ m} \tag{14}$$

References

[1] Wallenius, J., 2025. Fast neutron generation iv reactors. Lecture slides, Generation IV Reactors, KTH.