Version ES/3

EGT3
ENGINEERING TRIPOS PART IIB

Friday 22 April 2016 9.30 to 11

Module 4I11

ADVANCED FISSION & FUSION REACTOR SYSTEMS

CRIB

1

(a)

Spent nuclear fuel constituents are problematic for different reasons:

- Actinides have energy value, represent proliferation threat, some are long lived and mobile in potential repository environment
- Some fission products are long lived, others responsible for high heat load making repositories expensive.

Partitioning of FP and actinides allows directing each component to its most efficient treatment – recycling, temporary storage, transmutation or disposal.

(b)

	Advantages	Disadvantages
Pyro-processing	- Compact, high loading of actinides is possible (no moderator) - Radiation resistant - Small waste volumes	- Batch process - Salts require isolation from water and air (inert environment) - Oxide fuels require reduction to metal before processing
Aqueous processing	- Industrial experience - High purity product - Flow process - High throughput - Easy to process oxide fuels	- TBP sensitive to radiation - Large waste volumes - Combined actinide recovery is difficult

(c)

1. In fast spectrum, all fissile nuclides have larger fission neutron yield. Therefore, more neutrons are generally available (can be sacrificed) to transmutation mission.

2. Capture to fission cross section ratio for actinides is generally decreasing with energy. Therefore, production of new (minor) actinides will be less and transmutation chains will be shorter leading to fewer neutrons to be necessary for transmutation.

(d)

Determine first the fuel feed rate into the system:

Fuel inventory = Thermal power / sp. power = 1,000,000 kW /0.33/40 = 75,757 kg

Feed rate = Fuel inventory / Residence time = 75,757 / 4.5 = 16,835 kg/year

Not all of the fuel is U238 since enr. = 5%

Mass of U238 in the feed = 16,835 × 0.95 = 15,993 kg/year

Assuming that all Pu comes from neutron captures in U238 and disappears only through fission, the Pu production rate at the end of irradiation is:

$$\frac{dN^{Pu}}{dt} = N^{U238}\sigma_c^{U238}\varphi - N^{Pu}\sigma_f^{Pu}\varphi = 0$$

Assuming U238 concentration does not change much during irradiation:

$N^{U238}\sigma_c^{U238} = N^{Pu}\sigma_f^{Pu}$ or

$$N^{Pu} = \frac{N^{U238}\sigma_c^{U238}}{\sigma_f^{Pu}} = \frac{15{,}993\ \times 0.8}{60} = 213\ kg$$

(e)

Similar to PWR analysis, PRISM(Fast) Pu feed rate per year:
Feed rate = Thermal power / sp. power / Residence time
Noting that the fast burn option uses "fertile free" fuel (no U238 but 100% Pu):
Feed rate = 600,000/0.42 / 180 / 0.5 = 15,873 kgPu/year
Mission time = 140,000 kg / 15,873 kg/y = 8.82 years

(a) Internal breeding ratio.

Production of fissile = capture in fertile

= capture in 238, 240 and 242

= .1770 + .0114 + .0051 = .1935.

Consumption of fissile = (capture + fission) in fissile

= (capture + fission) in 235, 239 and 241

= .0020 + .0104 + .0297 + .1820 + .0054 + .0585 = .2880

Internal breeding ratio = .1935/.2880 = 0.672

6 (-1 if loss of 235 overlooked:

-1 if 242 capture overlooked unless production of ^{233}Am is explained)

(b) Importance of internal breeding

Internal breeding retards the loss of reactivity with burnup and therefore either allows either longer running periods between refuelling campaigns, or less excess reactivity at the start of a run and a smaller-worth absorbers to control excess reactivity (or both). The former improves the economics, the latter improves safety. **20%**

(c) Overall breeding ratio

Production of fissile = .1935 + 0.7 x .3105

= .4109.

Overall breeding ratio = .4109/.2880 = 1.427 **20%**

(d) Effect of MOx fuel on critical enrichment

The oxygen acts as a partial moderator, softening the neutron energy spectrum (i.e. reducing the mean energy of the neutrons). This reduces the fission rate in the "fertile" U238, which have fission thresholds around 1 MeV, and neutron yield per absorption in "fissile" isotopes. It also increases resonance absorption in U238. It means, more "fissile" isotopes have to be provided to compensate. This is shown by the neutron importance function φ* that falls with decreasing energy. The fall is steeper for metal fuel than for oxide, and particularly steep around and above 1 MeV.

A smaller effect is due to the lower density of oxide (because of the space taken up by the oxygen atoms). This allows more neutrons to leak from the core, again requiring more fissile material to compensate.

20% for explanation of moderation

5% for mention of leakage

3

(a) [25% total]

Some other fuels include: D-D, He^3-D, p-B **[name 1 = 5%]**

Advantages include: **[provide 2 = 15%, provide 1 = 7.5%]**

1. Stable fuels without complex handling and protection issues
2. No concerns of supply, production of fuels, generally plentiful
3. No or limited neutron production limiting damage to machine, irradiation of workers
4. Some have very large cross sections (at higher temperatures) making devices capable of burning them more efficient

These alternative fuels are not considered yet because the cross-section is too low at achievable temperatures **[5%]**

(b) [15% total]

The approximated surface area is $S = 2\pi^2 \times 2.8\ m \times 5.2\ m = 2874029\ cm^2$ **[5%]**

The estimated surface flux is then 1.0E+20 / S = 3.48×10^{13} n/s/cm^2 **[10%]**

(c) [30% total]

The reaction rate $R = 3.48 \times 10^{13}$ n/s/cm^2 x (10^{-27} cm [1mb]) x 1000 g x (1 mol / 18 g) x 6.02×10^{23}

$= 1.16 \times 10^{12}$ [^{16}N / s / L(H_2O)] **[20%]**

Fission reactor neutron spectra do not extend (significantly) to the energies required for this reaction threshold. **[10%]**

(d) [30% total]

For a 20 s pulse $N = 1.16 \times 10^{12} / 0.01 * (1 - e^{-0.2}) = 2.11 \times 10^{13}$ [^{16}N / L(H_2O)] **[10%]**

The activity is $\lambda \times N = 2.11 \times 10^{11}$ Bq/L = 211 GBq/L **[10%]**

The infinite irradiation gives $N = 1.16 \times 10^{12} / 0.01 = 1.16 \times 10^{14}$

The activity is then 1.16×10^{12} Bq/L = 1.16 TBq/L

Alternatively, for the equilibrium, they could state that the disintegration rate is equivalent to the production rate, giving the activity equal to the result from (c) but in Bq/L **[10% for equilibrium case]**

Any work in (c) or (d) based on incorrect values can receive full marks given the answers are correct for their inputs

4

(a) **[20% total]**

The alpha particles are charged and remain within the plasma, depositing their energy within the plasma. This provides plasma heating which is required to maintain adequate temperatures for fusion. **[10%]**

The neutrons travel throughout the machine and are thermalized within blanket modules to extract the fusion energy as heat for productive purposes (heat for electricity, for example). They are also used to breed fuels, for example tritium. **[10%]**

(b) **[15% total for any one of the following]**

1. Material damage through displacement, transmutations, etc. This will affect virtually any component, particularly those sensitive to material damage. Tungsten sputtering and steel fatigue are examples.
2. Activation of components with long-term radioactive decay consequences, e.g. human exposure, waste/decommissioning. Key examples are the divertor or any other example of long-term high-activity transmutations including rhenium, cobalt or others.
3. Nuclear heating of sensitive materials including superconducting magnets and optical components.

(c) **[20% total]**

Lithium is used for tritium production with both isotopes. **[5%]**

Li6 (n,t) He4 [exothermic] and Li7 (n,nt) He4 [endothermic threshold]

The Li6 is exothermic and can occur for any incident neutron energy – particularly with a high cross section at thermal energies. Li7 has a threshold between 2-3 MeV and, without a large thermal xs has a generally smaller effective cross section in typical blanket neutron spectra. It does release an additional neutron, allowing better neutron economy. **[10%]**

Li6 enrichment is standard in virtually all blanket concepts, in order to maximise tritium breeding. Heterogeneous enrichment is used to put Li7 in 14 MeV fields and Li6 in the more thermalized. **[5%]**

(d) **[25% total]**

More tritium must be produced than used as fuel since tritium is radioactive (lost over time) and isolation systems will always have less than 100% efficiency. Not all tritium will be burned and there will be various losses within exhaust, embedded within machine materials and in discharges. **[10%]**

Only a fraction of the surface area within a fusion reactor can be used for tritium breeding – there will be diagnostics, divertors, heating systems, etc. Neutrons will either be absorbed within other materials (not producing tritium) or leaked outside the boundaries of the tritium breeding modules. In order to obtain more than 1 tritium-producing reaction per fusion neutron, additional neutrons must be produced. **[15%]**

(e) **[20% total]**

Fission with typical fissile nuclides such as U235, Pu239, etc would produce plenty of neutrons. They would also produce useful heat. Fusion claims a few important points which would (with fusion/fission hybrid designs typically considered) no longer be true:

[10% for identifying fissiles and 10% for giving one of the following]

1. There is no high-level waste requiring active cooling after reactor shutdown and all wastes should return to background levels.
2. A tiny fuel inventory is loaded at any given time during operation, preventing any possibility of unwanted energy release – this would be doubly undone were fuel to exist in a form which did not require active operation to release energy.
3. Fusion is proliferation-resistant and does not offer (without the addition of fertile breeding materials) the ability to produce weapons material, such as U233 or Pu239.