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The Thorium Fuel Cycle

ThEC13

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Outline

Content:

- Background
- Sustainability, proliferation resistance, economics, radiotoxicity
- Advantages and disadvantages
- Fuel Cycles

Fertile thorium

- Th-232 is the only naturally occurring thorium nuclide
- It is a *fertile* nuclide that generates *fissile* U-233 on capturing a neutron
  - Th-232 is *fissionable* in that it fissions on interacting with fast neutrons > 1 MeV kinetic energy
  - Fertile conversion occurs with thermal neutron captures:

$$\text{Th-232 } (n,\gamma) \text{ Th-233 } (\beta^-) \text{ Pa-233 } (\beta^-) \text{ U-233}$$

A Thorium Fuel Cycle needs Uranium or Plutonium to initiate a fission reaction

Thorium fuel cycles

Options for a thermal reactor are:

- Once-through fuel cycle with Th-232 as alternative fertile material to U-238 with U-235 or Pu-239 driver
  - U-233 fissioned in-situ without reprocessing/recycle
  - Modest reduction in uranium demand and sustainability
- Recycle strategy with reprocessing/recycle of U-233
  - Much improved sustainability analogous to U/Pu breeding cycle
  - But some technical difficulties to overcome

Options for a fast reactor are:

- MSFR and other Gen IV concepts (Sodium cooled fast reactors, ADS systems)
  - All require U / Pu to initiate fission reaction

Sustainability / Inherent Proliferation Resistance

Sustainability

- Thorium abundance higher than uranium
- Thorium demand lower because no isotopic enrichment
- Thorium economically extractable reserves not so well defined
- Rate of expansion of thorium fuel cycle will be limited by the slow conversion rate

Inherent proliferation resistance

- U-233 is a viable weapons usable material
  - High U-232 inventory implies high doses unless shielded
  - Low inherent neutron source suggests that U-233 weapon design may be simplified and potentially more accessible
  - U-233 fissile quality hardly changes with irradiation

Economics and Radiotoxicity

Economics

- U-233 recycle has lower demand on thorium than uranium because there is no isotopic enrichment process
- U-233 recycle potentially reduces the ore procurement cost and eliminates the enrichment cost
- Future uranium and thorium market prices unknown
- Short term economic barrier presented by need for R&D to demonstrate satisfactory fuel performance

Radiotoxicity

- Spent fuel activity/radiotoxicity dominated by fission products for 500 years after discharge
  - U/Pu long term fuel activity determined by activity of Np, Pu, Am and Cm
  - Th/U-233 long term fuel activity has only trace quantities of transuranics and therefore lower radiotoxicity after 500 years
  - However, this only applies to the long term equilibrium condition with self-sustained U-233 recycle
  - In a practical scenario, the reduction in radiotoxicity is more modest than the long term equilibrium would indicate

It is too soon to say whether the thorium fuel cycle will be economically advantageous

Need to compare radiotoxicity over range of timeframes

Advantages of Th fuel cycle

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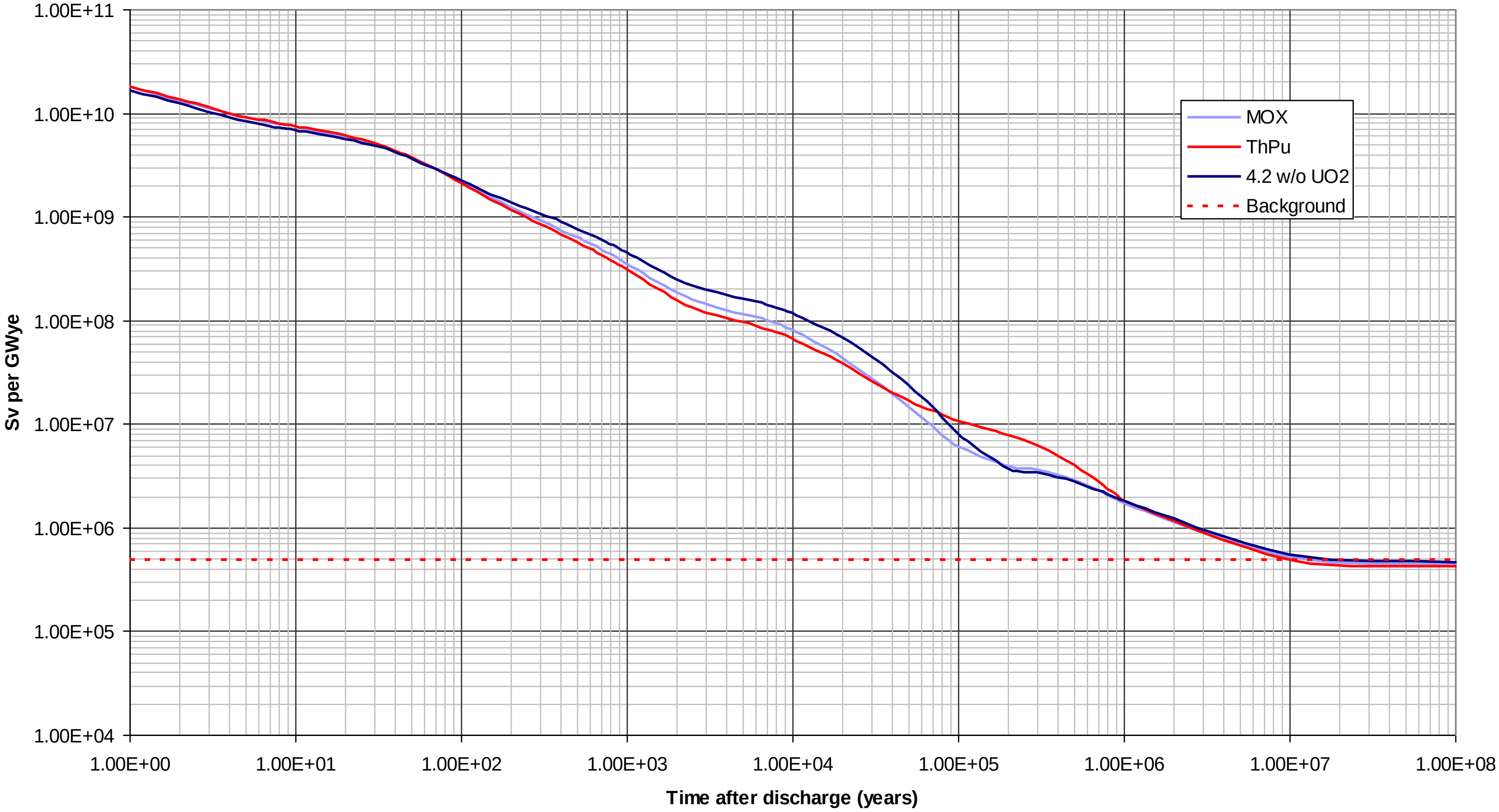
- Thorium more abundant than uranium and combined with a breeding cycle is potentially a major energy resource
- Low inventories of transuranics and low radiotoxicity after 500 years’ cooling
- Almost zero inventory of weapons usable plutonium
- Theoretical low cost compared with uranium fuel cycle
- ThO_2 properties generally favourable compared to UO_2 (thermal conductivity; single oxidation state)
- ThO_2 is potentially a more stable matrix for geological disposal than UO_2

Void coefficient mitigation

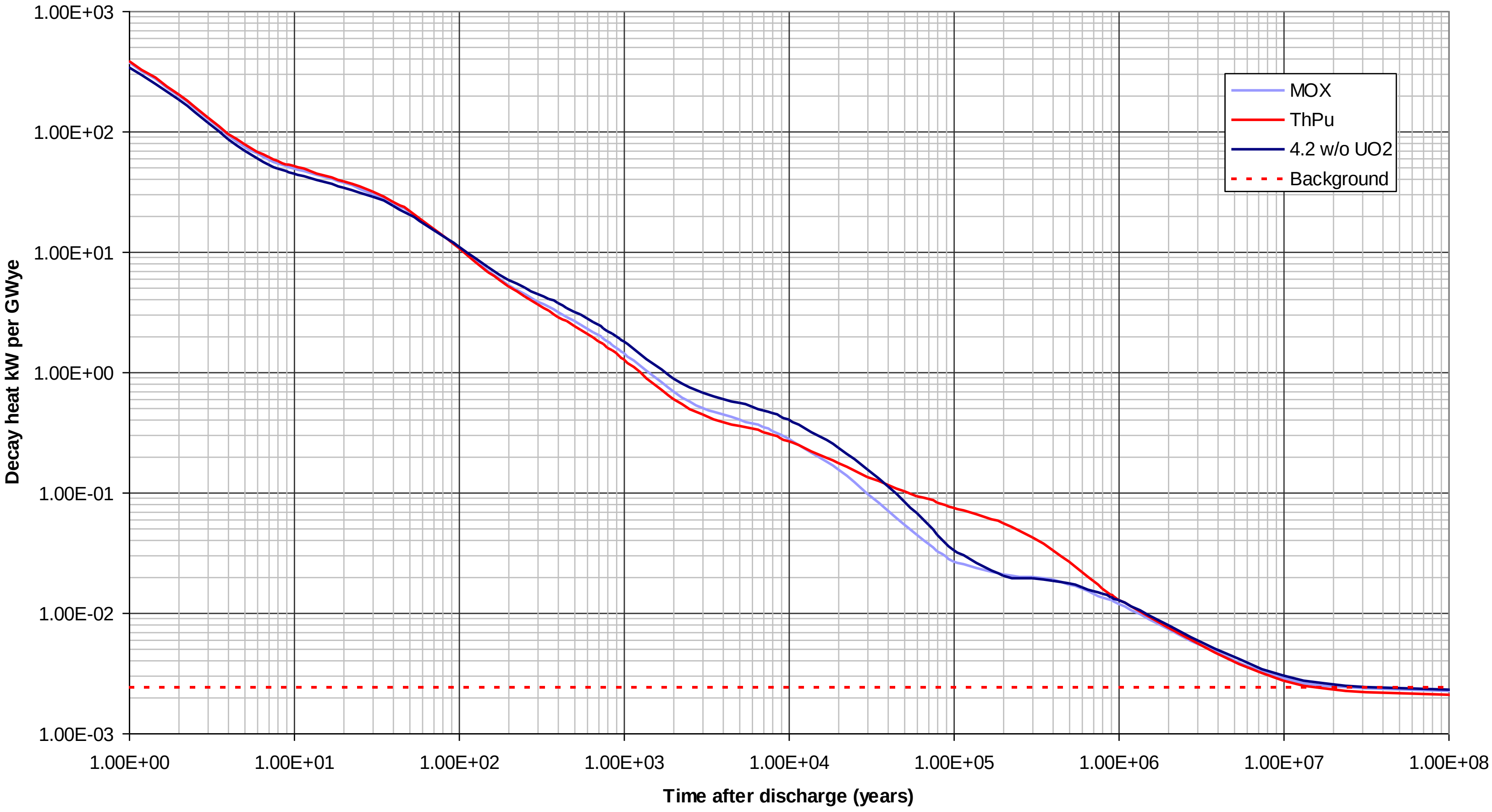
- Supplementing a U/Pu recycle strategy
- Thorium fuels drive the void coefficient more negative in thermal and fast systems
  - A positive void coefficient is an undesirable in-core positive feedback effect unless counteracted by other feedback effects
  - In LWRs a positive void coefficient is usually considered unacceptable and limits the total plutonium load in MOX fuel to <12 w/o
    - This is a potential restriction with poor fissile quality plutonium
  - Thorium-plutonium fuel could allow significantly higher total plutonium loads (up to ~18 w/o), giving more flexibility for plutonium re-use in LWRs

A Possible way to manage plutonium stocks with poorer fissile quality and to allow time for thorium plutonium MOX qualification

Radiotoxicity

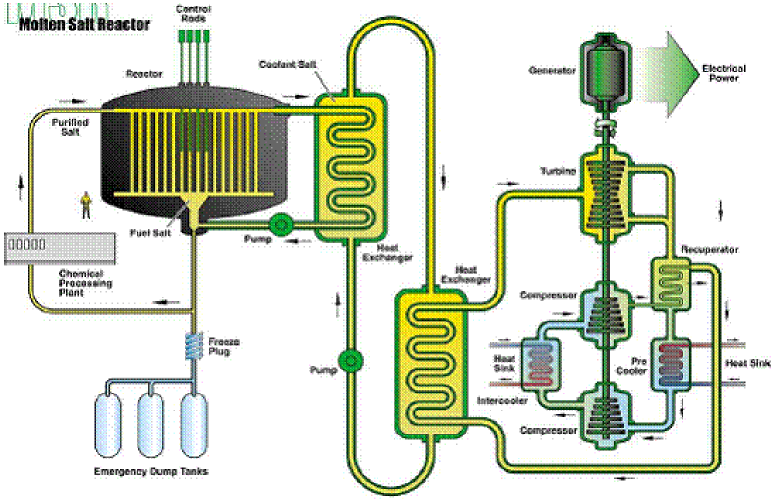


Decay heat


1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1.00E-03
Decay heat kW per GWye
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
1.00E+07
1.00E+08
Time after discharge (years)
MOX
ThPu
4.2 w/o UO2
Background

Molten salt reactor

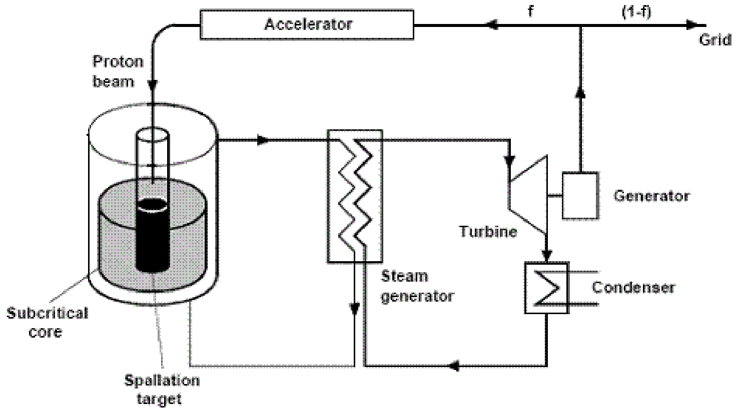
- Molten Salt Reactor (MSR)
  - Generation IV International Project is researching MSR
  - Gen IV MSR will be a fast spectrum system
  - Molten salt fuel circulates through core and heat exchangers
  - On-line reprocessing to remove fission products
  - <u>Ideally suited to thorium fuel as fuel fabrication is avoided</u>
  - <u>Equilibrium fuel cycle will have low radiotoxicity (fission products only)</u>



Many technological issues to address - MSR is a long term option

Accelerator driven system

- Accelerator driven system (ADS)
  - Sub-critical reactor core
  - Proton beam provides neutron source in spallation target
  - Neutron source multiplied by sub-critical core



Disadvantages of Th fuel cycle

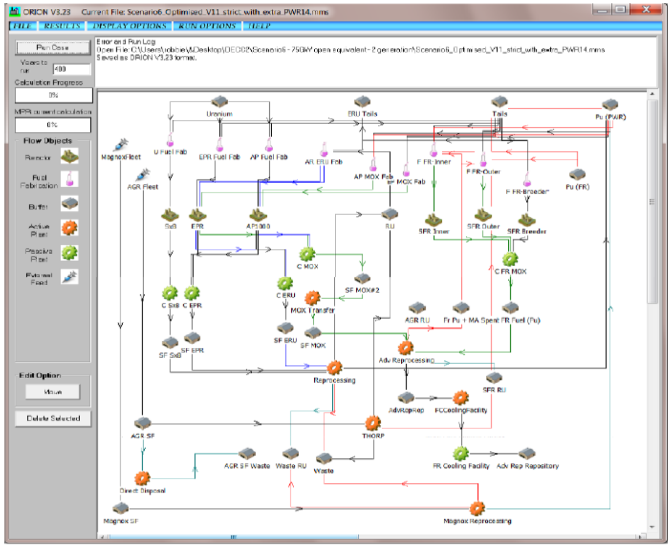
- Th-232 needs to be converted to U-233 using neutrons from another source
  - Neutrons are expensive to produce
  - The conversion rate is very low, so the time taken to build up usable amounts of U-233 are very long
- Reprocessing thorium fuel is less straightforward than with the uranium-plutonium fuel cycle
- The THOREX process has been demonstrated at small scale, but will require R&D to develop it to commercial readiness
- U-233 recycle is complicated by presence of ppm quantities of U-232 (radiologically significant for fuel fabrication operations at ppb levels)
- U-233 is weapons useable material with a low fissile mass and low spontaneous neutron source
  - U-233 classified by IAEA in same category as High Enriched Uranium (HEU) with a Significant Quantity in terms of Safeguards defined as 8 kg compared with 32 kg for HEU

R&D requirements

- Fuel materials properties
- Fuel irradiation behaviour
- THOREX reprocessing
- Waste management /disposal
- U-233 fuel fabrication
- Systems development
- Scenario modelling

Fuel cycle scenario modelling

- Fuel cycle simulation computer programs are used to assess the impacts that different fuel cycle scenarios may have on:
  - Uranium or Thorium ore requirements,
  - Time and resources needed to create sufficient fertile material to start a Thorium ‘only’ reactor
  - Ability to start a sustainable fast reactor fleet,
  - Time at which feed of natural uranium is no longer required
  - Packing density and inventory of a geological repository
  - The practicalities of handling fresh nuclear fuel
  - Processing of spent nuclear fuel
  - Requirements for high level waste immobilisation technologies



Error and Run Log
Open File: C:\Users\robbie\Desktop\OECD2\Scenario6 - 75GW open equivalent - 2 generation\Scenario6_Optimised_V11_strict_with_extra_PWR14.mms
Saved as ORION V3.23 format.

Years to run: 400
Calculation Progress: 0%
MPR current calculation: 0%

Fuel cycle

- Building up a fleet with the aim of reducing dependency on U/Pu will take time.
- Reactor doubling time is an important consideration
- Some contention that alternative systems might give a different result
- But these underlying equations give confidence that the same limitations will apply to all workable systems

Relevance to thorium

- Long doubling times are relevant to:
  - Initial build-up of U-233 inventory to get thorium fuel cycle to equilibrium
    - For practical systems this timescale is very long and this will govern strategic analysis of transition to thorium fuel cycle using enriched uranium or plutonium/transuranic fuels
    - Important for strategic assessments to account for impact of transition effects
  - Subsequent expansion of thorium reactor fleet and rate at which thorium systems can expand to meet increasing demand

Breeding ratio

- The breeding ratio (BR) is defined as:

$$\mathrm{BR} = \frac{\text{Mass of fissile material produced by fertile neutron captures}}{\text{Mass of fissile material consumed}}$$

EXAMPLE:

- 1 GWth breeder reactor operating at 90% load factor would consume approximately 330 kg of fissile material per year – equivalent to 1 kg per full power day
- If a breeder reactor produces 1.3 kg of new fissile material by fertile captures per full power day, the breeding ratio is 1.3/1.0 = 1.3 and the breeding gain (BG) defined as BG = BR-1 is (1.3-1.0)/1.0 = +0.3

Doubling time

- This is the time in which a breeder reactor would take to generate enough surplus fissile material to start off an identical reactor system
  - The doubling time (T_D) is the time needed to replace the total fissile inventory of the core M_C (kg) plus the out of core fissile inventory M_O (kg)
  - For a system which consumes μ kg of fissile material and has a net gain γ kg of fissile material per full power day, the doubling time is:

$$\begin{aligned} T_D \text{ (full power days)} &= [M_C + M_O]/\gamma \\ &= [M_C + M_O]/[(BR-1).\mu] \\ &= [M_C + M_O]/BG.\mu \end{aligned}$$

GOVERNING PARAMETERS:

- ➢ μ is governed by the thermal power output only – 1 kg per full power day for 1 GWth output
- ➢ $[M_C + M_O]$ and BG are dependent on the specific reactor design
- ➢ $[M_C + M_O]$ typically a few thousand kg
- ➢ Large positive BG very difficult to achieve and 0.3 to 0.4 is about the highest claimed for any system

Application to MSR

THERMAL SPECTRUM MSR

- Based on simplistic scale-up of ORNL Molten Salt Reactor Experiment:
  - 1.0 GWth; μ = 1.0 kg/full power day; M_C = 1500 kg U-233; M_O = 3000 kg U-233; BG = +0.06 (estimated)
  - $T_D = [M_C + M_O]/BG.\mu = (4500/0.06 \times 1.0)$ = 75000 full power days **(200 full power years)**
- Probable scope for optimisation, but doubling time still likely to be very long

FAST SPECTRUM MSR

- Based on Delpech/Merle-Lucotte et al TMSR-NM (non-moderated thorium molten salt reactor) core:
  - 2.5 GWth; μ = 2.5 kg/full power day; M_C+M_O = 5700 kg; BG = +0.12
  - $T_D = [M_C + M_O]/BG.\mu = (5700/0.12 \times 2.5)$ = 19000 full power days **(52 full power years)**

Hypothetical profile of installed capacity versus time for a breeder system

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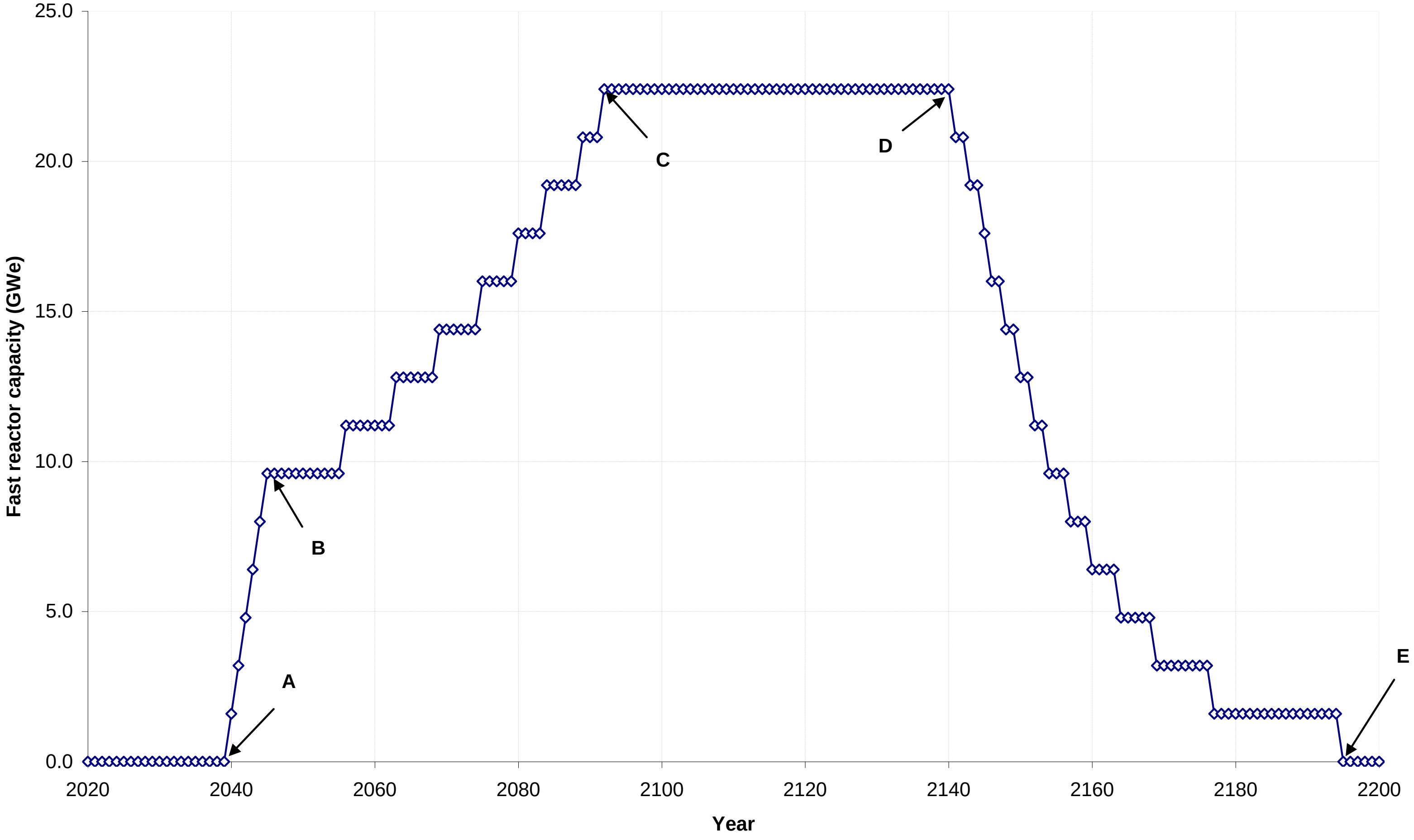

Reactor parameters

Parameter	Value	Units
Unit size	1.6	GWe
Initial core fissile loading	10.0	tHM
Dwell time	4.0	years
Recycle time	5.0	years
Net breeding gain (in breeding mode)	+0.3	-
Net breeding gain (in self-sufficient mode)	0.0	-
Net breeding gain (in burner mode)	-0.40	-
Earliest fast reactor deployment	2040	
Maximum fast reactor capacity	22.4	GWe

Generating capacity - transition from LWRs to fast reactors

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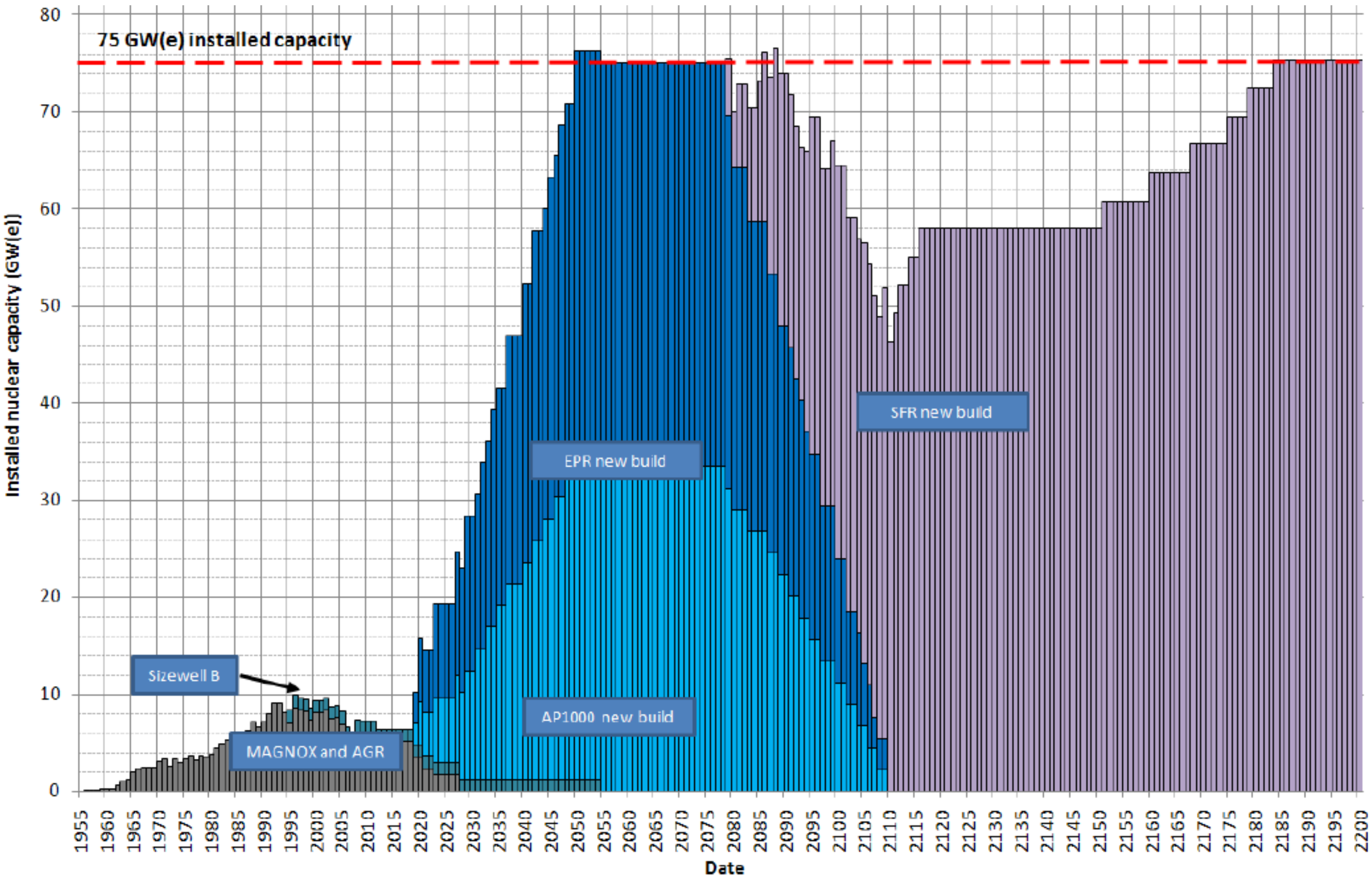
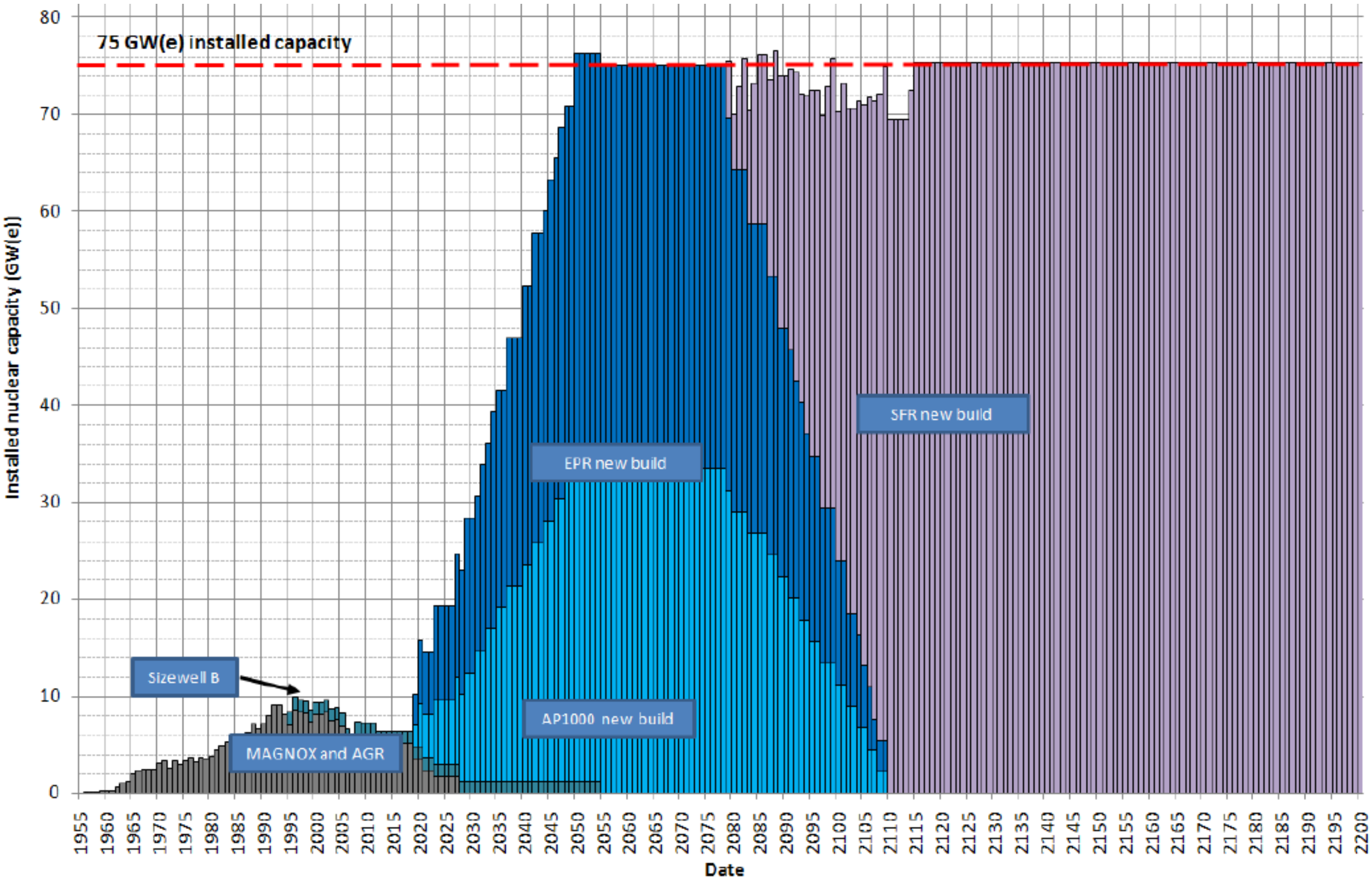


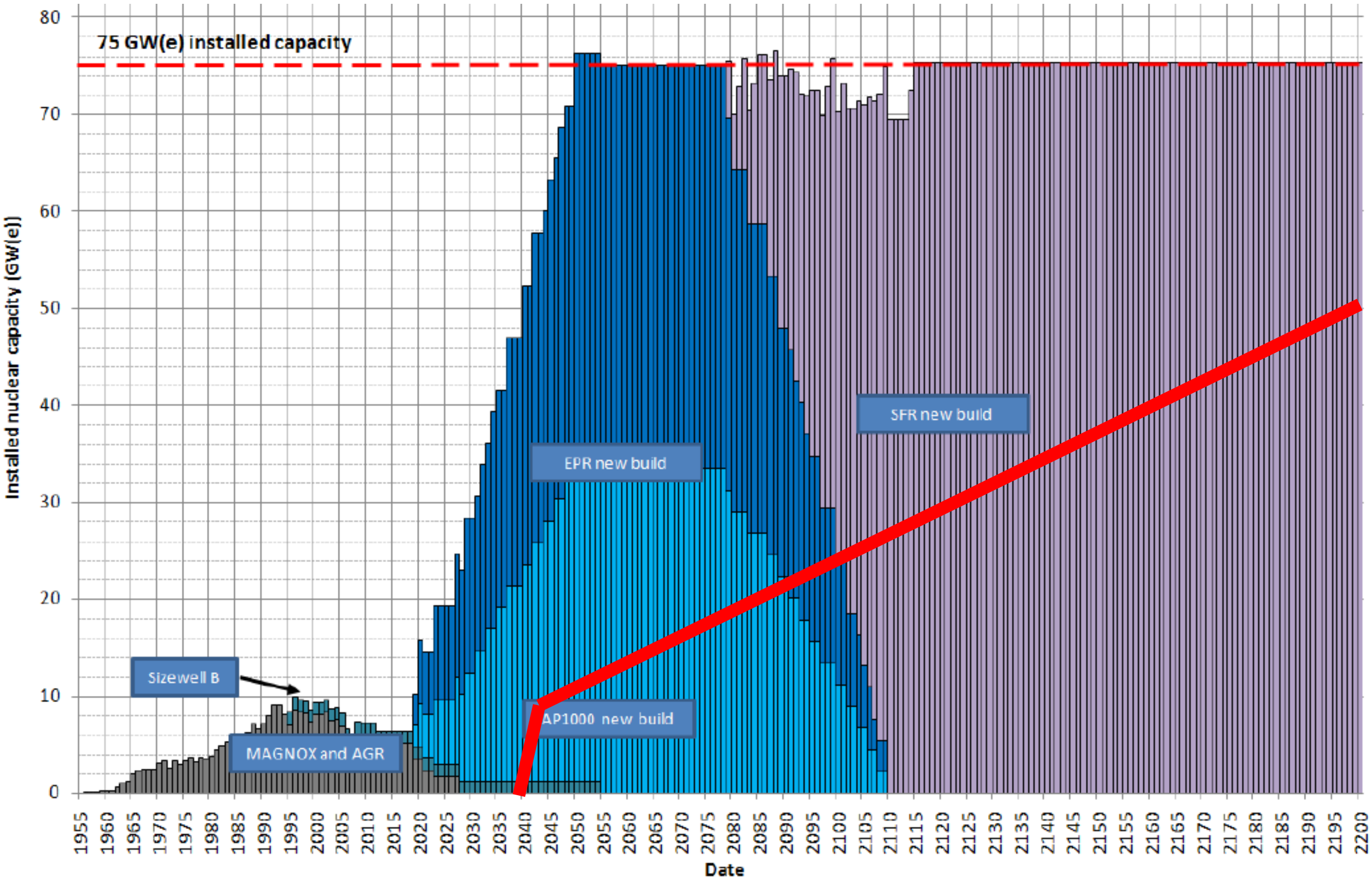
- 75 GWe target installed capacity
- FRs introduced at same rate as LWRs retire
- LWRs fuelled with UO_2

Generating capacity - transition from LWRs to fast reactors



- 75 GWe target installed capacity
- FRs introduced at same rate as LWRs retire
- FR Fuel dwell time is reduced

Generating capacity transition from LWRs to fast reactors - Scenario (b)



- 10GWe of Th breeding FR’s introduced ~2040
- FR breeders fuelled only with U-233 introduced ~2045

Conclusions

- Thorium is a valuable strategic alternative to uranium
  - Sustainability remains one of the main drivers
- Radiotoxicity benefit is real, but modest
  - Long term equilibrium radiotoxicity a simplistic measure
- Inherent proliferation resistance not proven for thorium
- Economics of thorium not known at present
- Minimum 15-20 year timeframe for commercial deployment (thermal systems) and longer timeframes for fast reactors
- Significant R&D programme required to progress technical maturity

Acknowledgements

- Kevin Hesketh
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Further information

Thorium history

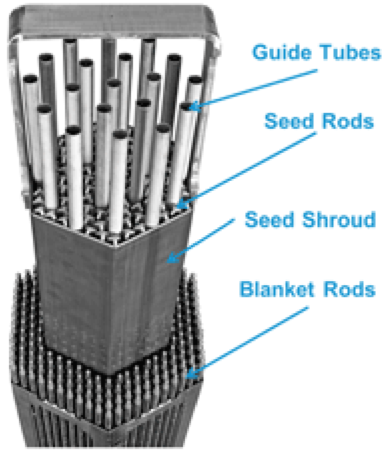
- In the 1950s through to the 1980s, there were thorium research programmes for:
  - Pressurised water reactors (PWR)
    - Shippingport breeder core
    - Germany-Brazil collaboration
  - High temperature gas reactors (HTR)
    - DRAGON (UK), Fort St Vrain (USA), Peach Bottom (USA), AVR (Germany)
  - Molten salt reactors (MSR)
    - Molten Salt Reactor Experiment (USA)
- The common driver for all these plants was to decouple nuclear expansion from uranium availability

Why did thorium research stall?

- Thorium cycle requires neutrons from uranium or plutonium fissions to get started
- U/Pu fuel cycle already established
  - Large barrier to entry for a new system
- Technological issues
  - THOREX reprocessing and fabrication of U-233 fuels

India/Lightbridge

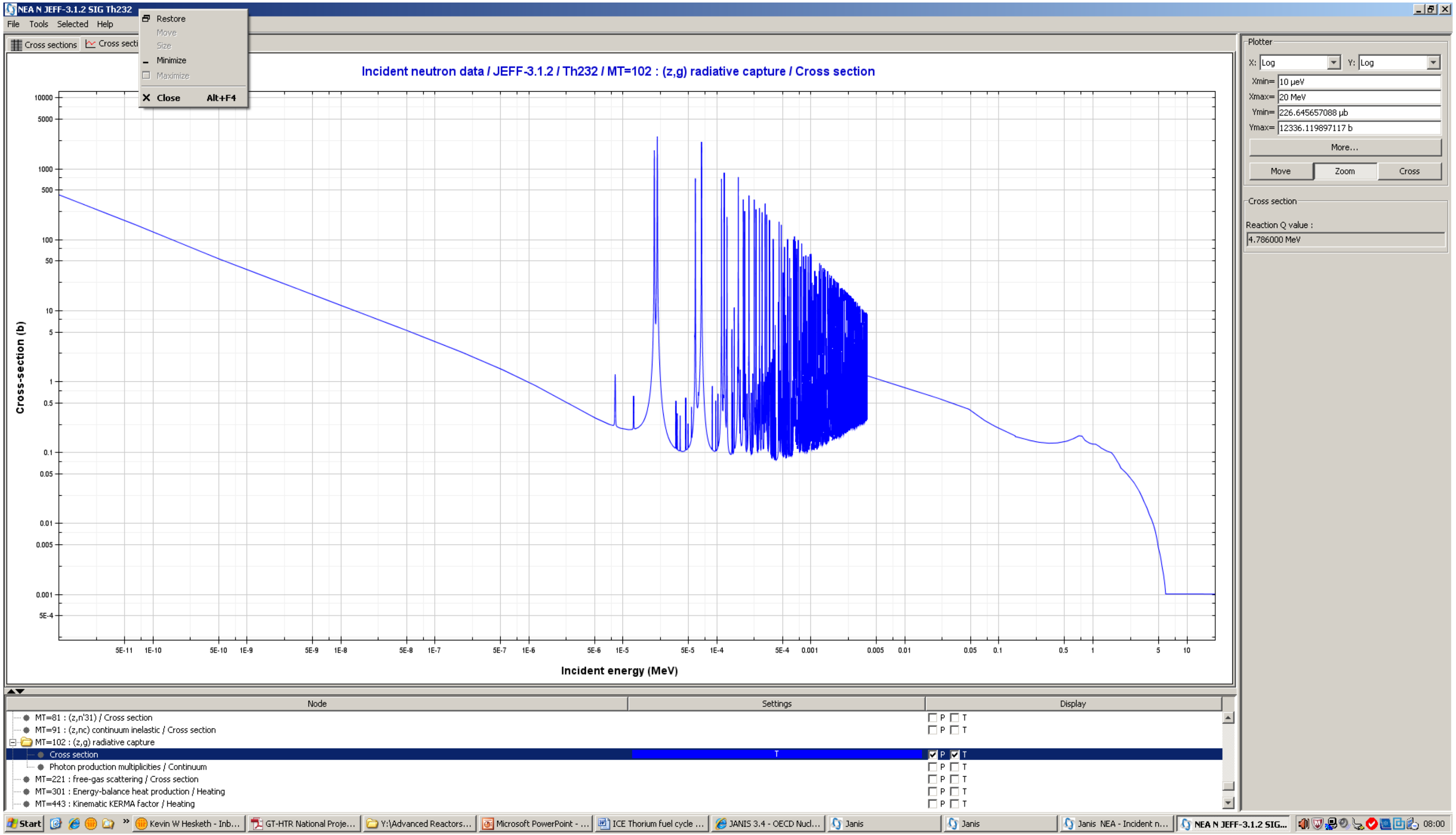
- India
  - Synergistic fuel cycle involving fast reactor and Advanced Heavy Water Reactors (AHWR)
  - Fast reactor will breed U-233 in a thorium blanket
  - U-233 will be recycled into AHWR fuel
- Lightbridge
  - Seed/blanket assembly design for PWRs
  - Low enriched uranium (LEU) seed region provides spare neutrons
  - ThO_2 blanket breeds U-233
  - Seed and blanket regions have different in-core dwell times



Pu/Th MOX

- AREVA are investigating PuO_2/ThO_2 MOX fuel for the eventual disposition of PWR MOX fuel assemblies
  - PWR MOX fuel currently not reprocessed in France
  - Held in long term storage pending eventual recycle in SFR fleet
  - Requirement to cover all contingency that SFR fleet is not built
    - Recycle of Pu from MOX fuel preferred over disposal
    - PuO_2/ThO_2 MOX is presumed to be another option with potential advantage of low development cost and high stability as a final waste form
- Thor Energy undertaking PuO_2/ThO_2 MOX fuel qualification programme through a international consortium

Th-232 radiative capture cross-section



Incident neutron data / JEFF-3.1.2 / Th232 / MT=102 : (z,g) radiative capture / Cross section

Plotter

X: Log Y: Log

Xmin= 10 µeV

Xmax= 20 MeV

Ymin= 226.645657088 µb

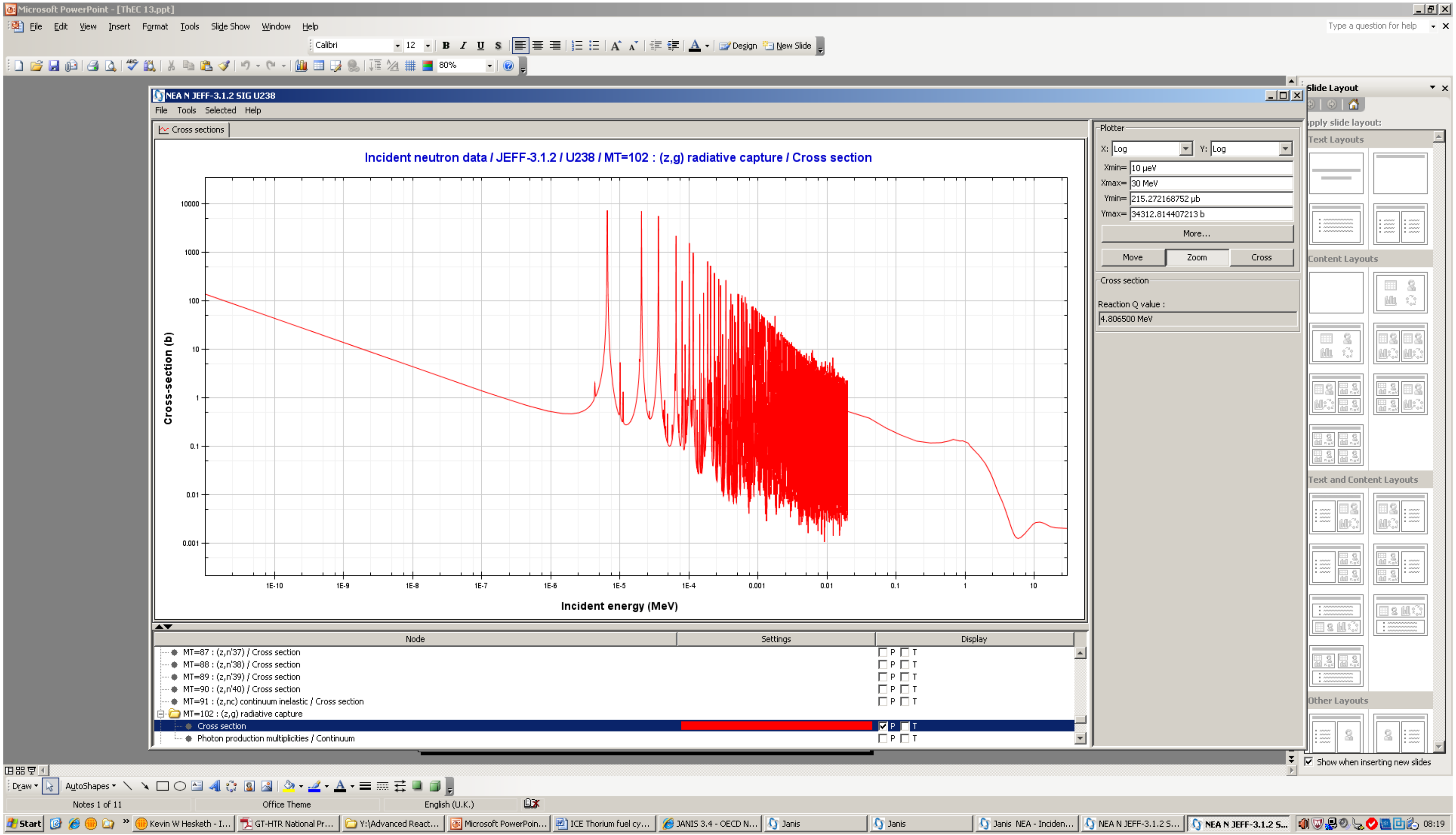
Ymax= 12336.119897117 b

Cross section

Reaction Q value : 4.786000 MeV

Node	Settings	Display
MT=81 : (z,n'31) / Cross section		☐ P ☐ T
MT=91 : (z,nc) continuum inelastic / Cross section		☐ P ☐ T
MT=102 : (z,g) radiative capture		
Cross section	T	☑ P ☑ T
Photon production multiplicities / Continuum		☐ P ☐ T
MT=221 : free-gas scattering / Cross section		☐ P ☐ T
MT=301 : Energy-balance heat production / Heating		☐ P ☐ T
MT=443 : Kinematic KERMA factor / Heating		☐ P ☐ T

U-238 radiative capture cross-section



Incident neutron data / JEFF-3.1.2 / U238 / MT=102 : (z,g) radiative capture / Cross section

Plotter

X: Log
Y: Log
Xmin= 10 µeV
Xmax= 30 MeV
Ymin= 215.272168752 µb
Ymax= 34312.814407213 b

Cross section

Reaction Q value :
4.806500 MeV

Node	Settings	Display
MT=87 : (z,n'37) / Cross section		☐ P ☐ T
MT=88 : (z,n'38) / Cross section		☐ P ☐ T
MT=89 : (z,n'39) / Cross section		☐ P ☐ T
MT=90 : (z,n'40) / Cross section		☐ P ☐ T
MT=91 : (z,nc) continuum inelastic / Cross section		☐ P ☐ T
MT=102 : (z,g) radiative capture		
Cross section	🟥	☑ P ☐ T
Photon production multiplicities / Continuum		☐ P ☐ T

Decay heat and radiotoxicity

- Thorium-plutonium MOX fuel theoretically could be advantageous for UK plutonium disposition
  - Detailed assessment by NNL of decay heat load and radiotoxicity per *GWye* shows there is only a marginal difference between Th-Pu MOX and U-Pu MOX
  - This is a holistic calculation that accounts for the total decay heat outputs of different scenarios
    - In the Th-Pu and U-Pu MOX cases, the decay heat is concentrated in the MOX assemblies, whereas in the UO_2 reference case it is distributed over a larger number of UO_2 assemblies

Core fissile inventory M_C

- The fissile inventory of the core depends on a number of factors:
  - Minimum critical mass for the system
  - Thermal power output
  - Specific rating of the core in MW/tonne
  - Refuelling interval
  - Refuelling strategy – single batch or multiple batch core loading
- KEY POINTS:
  - The minimum critical mass can range over 3 orders of magnitude for different configurations (for example from 5 kg for a HEU research reactor core to several 1000 kg for a typical 1 GWe power plant)
  - Workable designs typically nearer the upper end of the mass range and therefore M_C is practically constrained to a few x 1000 kg
  - Very important distinction between M_C and μ, which are orders of magnitude different for any practical system

Illustration that large power reactors have a large fissile inventory based on survey or world reactors

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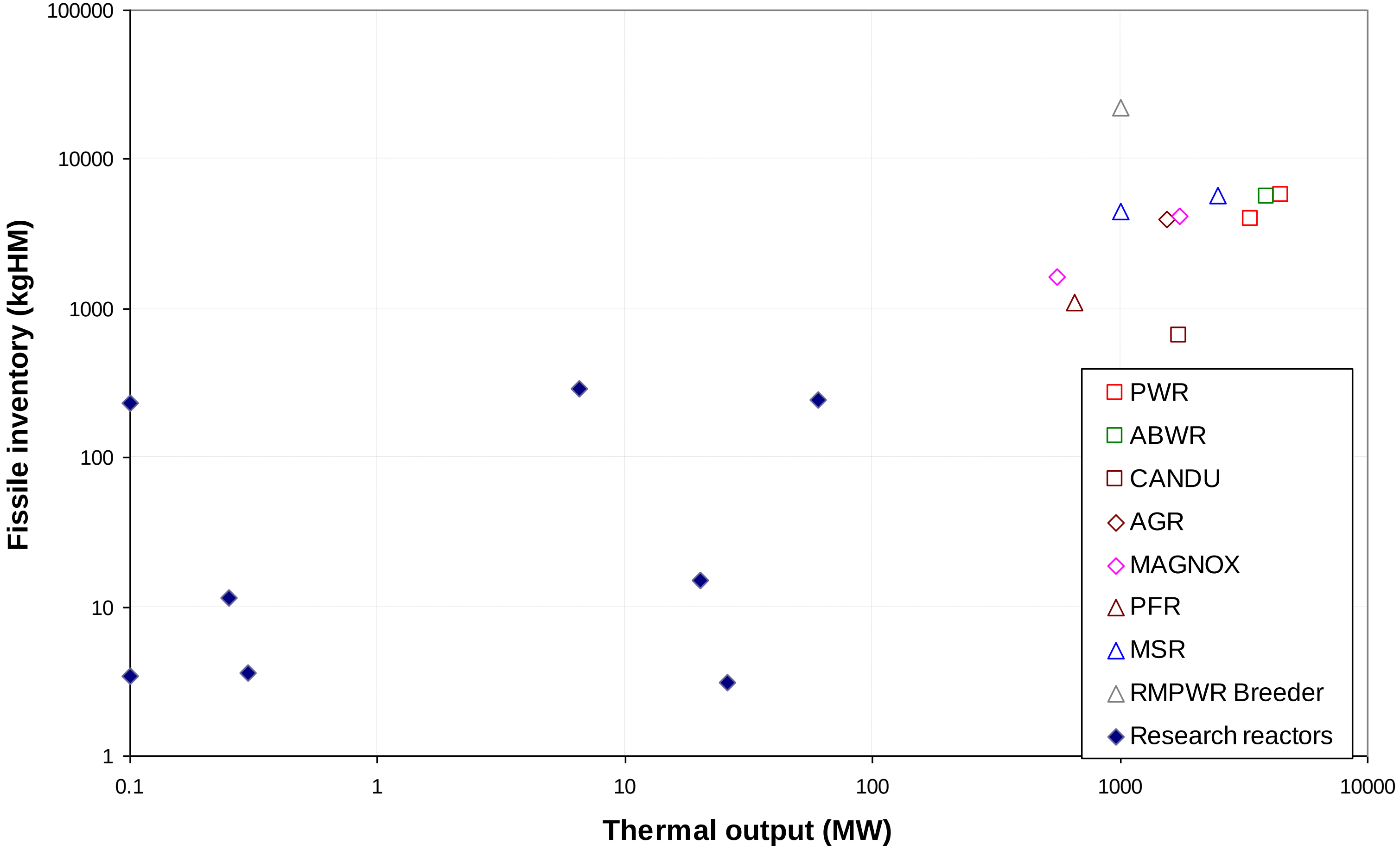

Out of core fissile inventory M_O

- For a conventional solid fuel reactor, this is the inventory in spent fuel awaiting reprocessing or being reprocessed, plus the inventory of fuel under fabrication, which depends on:
  - Spent fuel cooling time t_c
  - Reprocessing time t_r
  - Fuel fabrication time t_f
- For a fuel dwell time T, M_O scales with M_C:

$$M_O = M_C \times (t_c+t_r+t_f)/ T$$

- For a liquid fuel system such as Molten Salt Reactor (MSR), there is an out-of-core inventory, which is the mass of fuel circulating through the heat exchangers