PPCS Safety and Environmental Analysis

presented by

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Work performed by EURATOM Associations and European industry, including:

ENEA, UKAEA, VR Studsvik, FZK, TEKES
EFET-Belgatom, EFET-Framatome ANP,
EFET-Ibertef, University of Pisa

Outline

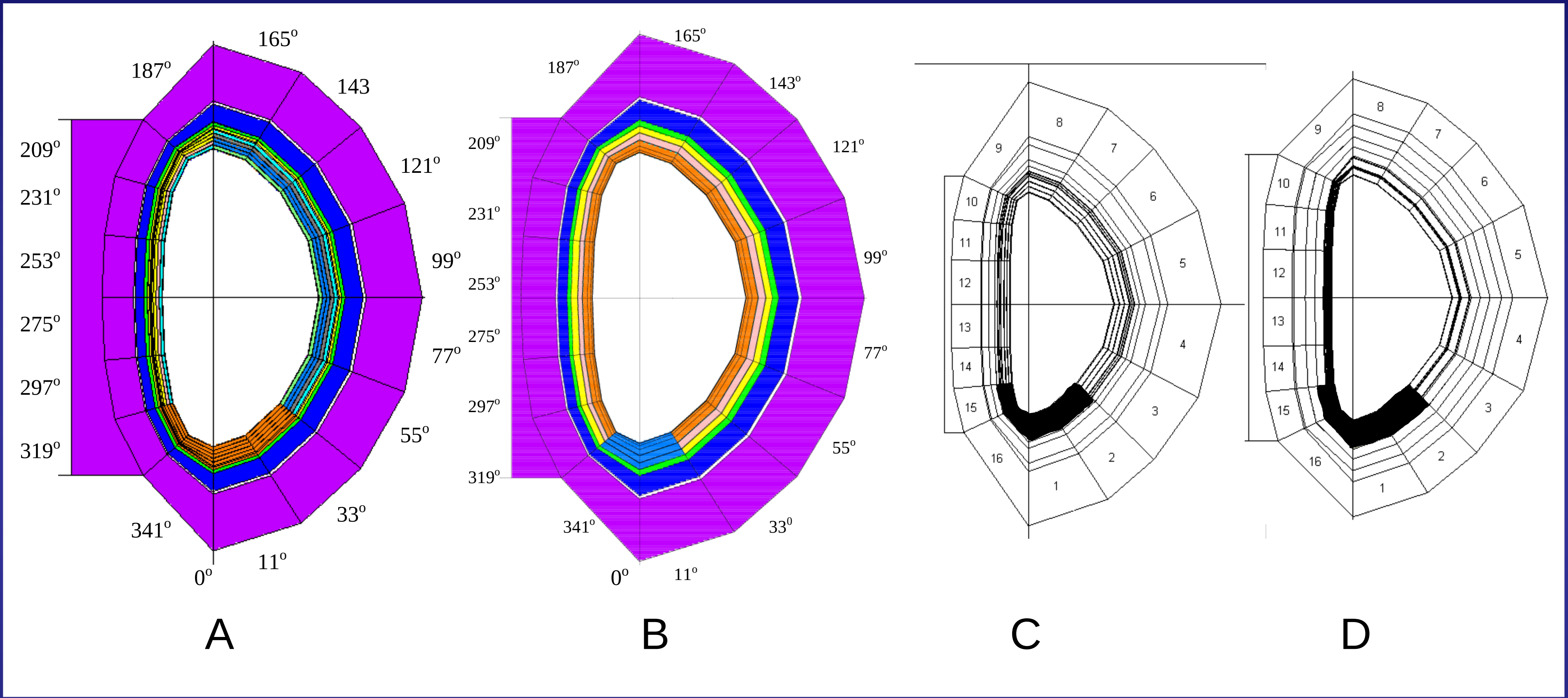
- Neutron activation calculations
- Accident analyses
- Bounding accident modelling
- Maximum public doses
- Waste categorisation
- Conclusions

Neutron activation modelling

- Complete knowledge of activation and related quantities essential for S&E assessment
  - source term for potential release in accident scenarios
  - origin of occupational radiation exposure
  - long-term activation and waste management issues

- Computation for all PPCS models using 3-D geometry
  - MCNP4C3 calculations of 175-group neutron spectra in all cells of 3-D model out to TF coils
  - FISPACT calculation of inventories, activation and derived quantities (dose rates, decay heat, clearance index etc.)
    - in every cell
    - at various decay times up to 10,000 years
  - Calculations supervised by in-house code system HERCULES

Geometry of activation models

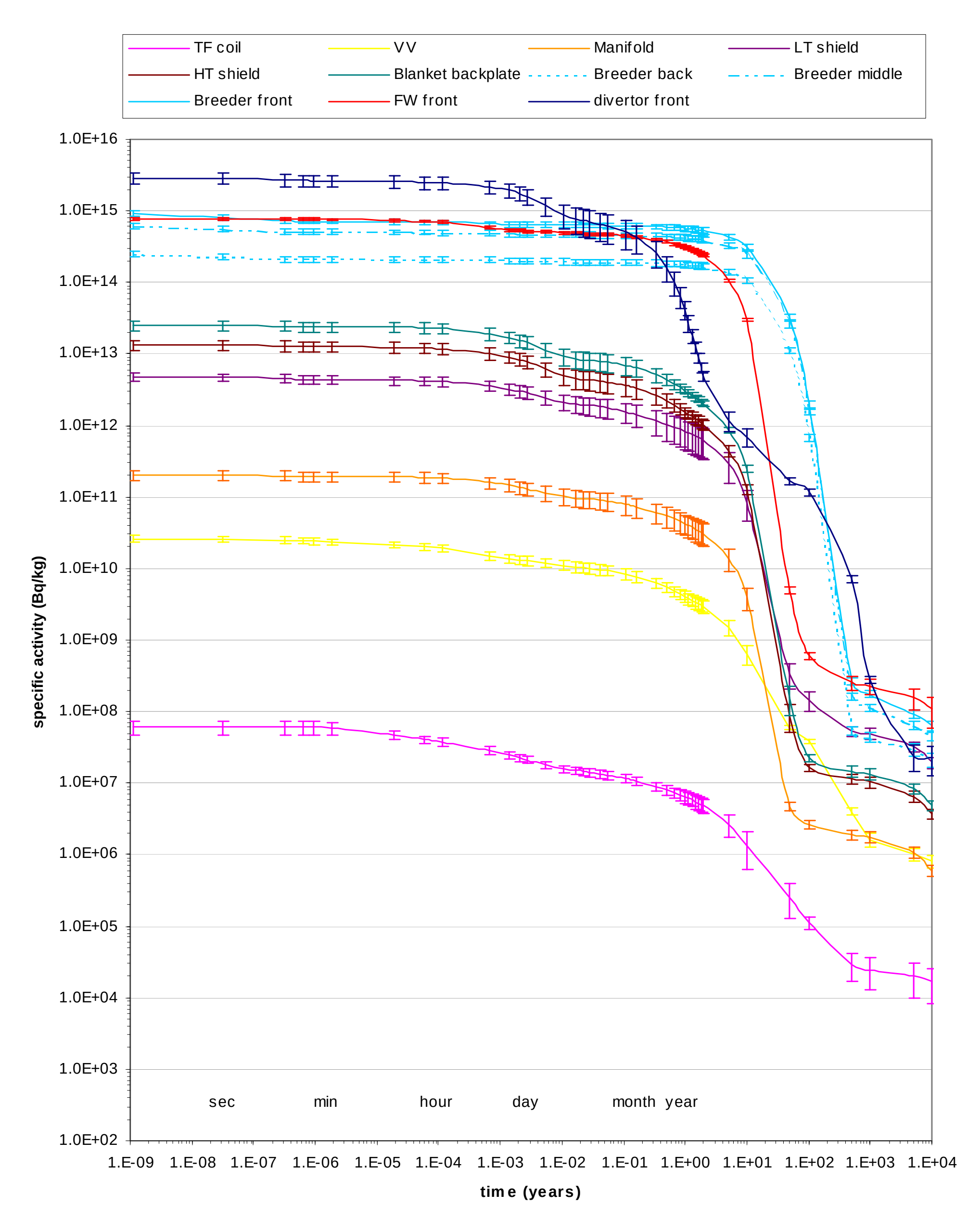


Divertor regions represented only by homogenised composition

Assumed operation history for activation

- 2.5 years at full power, then 2 months for divertor replacement
- Further 2.5 years full power, then 10 months blanket + divertor replacement
- Repeated 5 times (25 full-power years total)

Model B activation - inboard midplane



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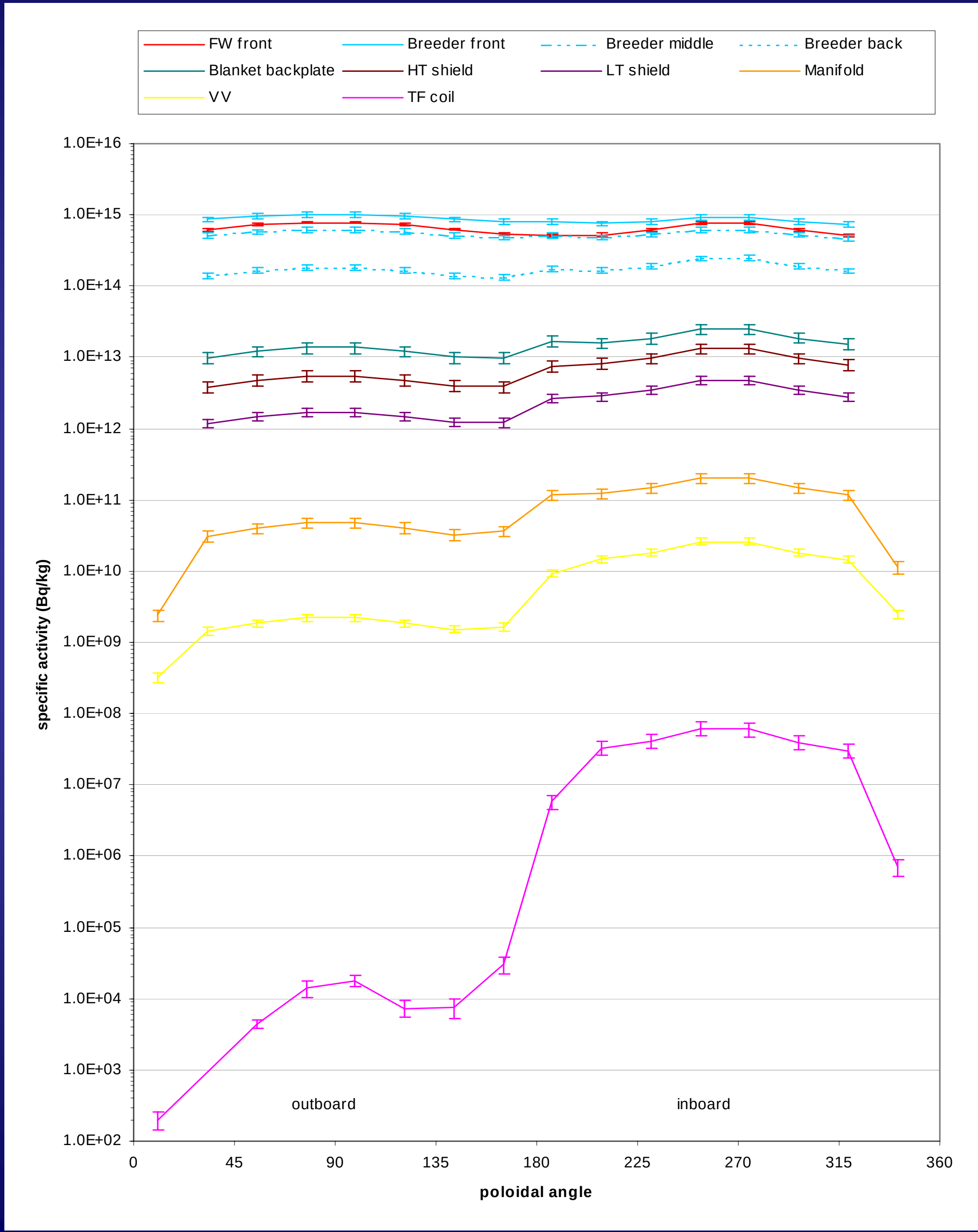
Model B activation - poloidal variation



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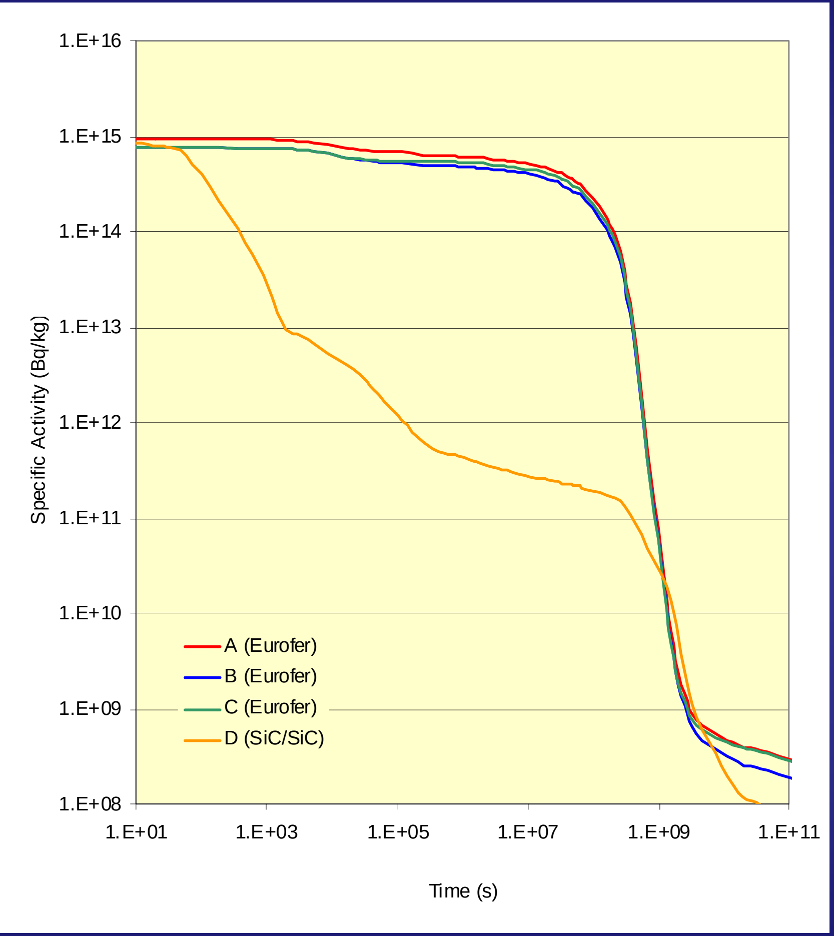
First wall (outboard) activation - all models



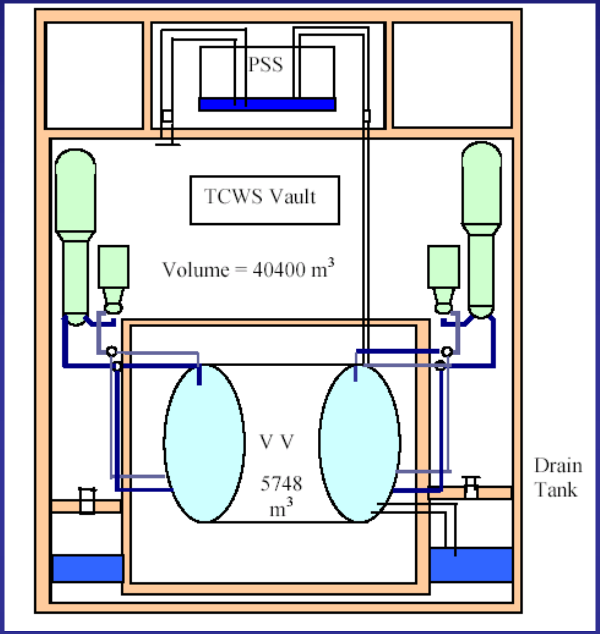
Accident sequence analyses

- Functional Failure Modes and Effects Analysis (FFMEA) used to identify sequences for study
  - functional breakdown of plant
  - loss of functions identified : Postulated Initiating Events (PIEs)
  - PIEs grouped according to type and their impact
- Four scenarios selected for deterministic analyses:
  - ex-vessel LOCA
  - ex-vessel LOCA leading to in-vessel LOCA
  - loss of coolant flow, no plasma shutdown, leading to in-vessel LOCA
  - loss of heat sink, no plasma shutdown

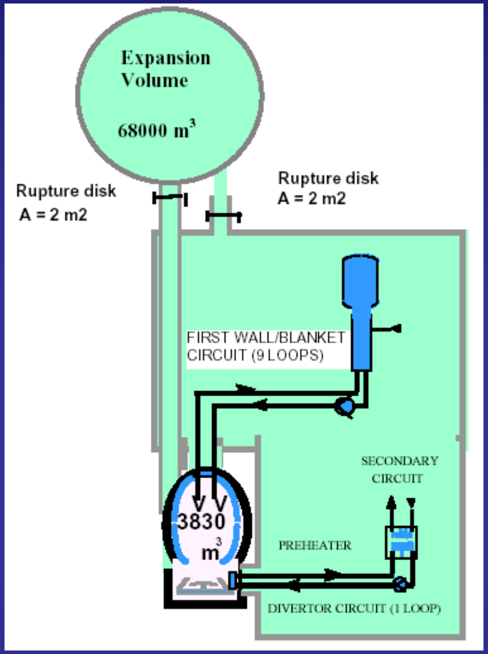
Assumed source terms for accident analyses

Source terms	Model A	Model B	Model C
Tritium in VV	1 kg	1 kg	1 kg
Dust	10 kg (7.6 kg steel + 2.4 kg W)	10 kg (7.6 kg steel + 2.4 kg W)	10 kg (8.55 kg steel + 1.45 kg W)
Tritium in coolant	15 g (per loop)	1 g (per loop)	0.003 g (per loop)
ACPs total inventory	50 kg (per loop)	-	-
ACPs mobilization fraction	1% of 50 kg (per loop)	-	-
Sputtering products		~ 0 g	~ 0 g

Outline of confinement schemes



Model A



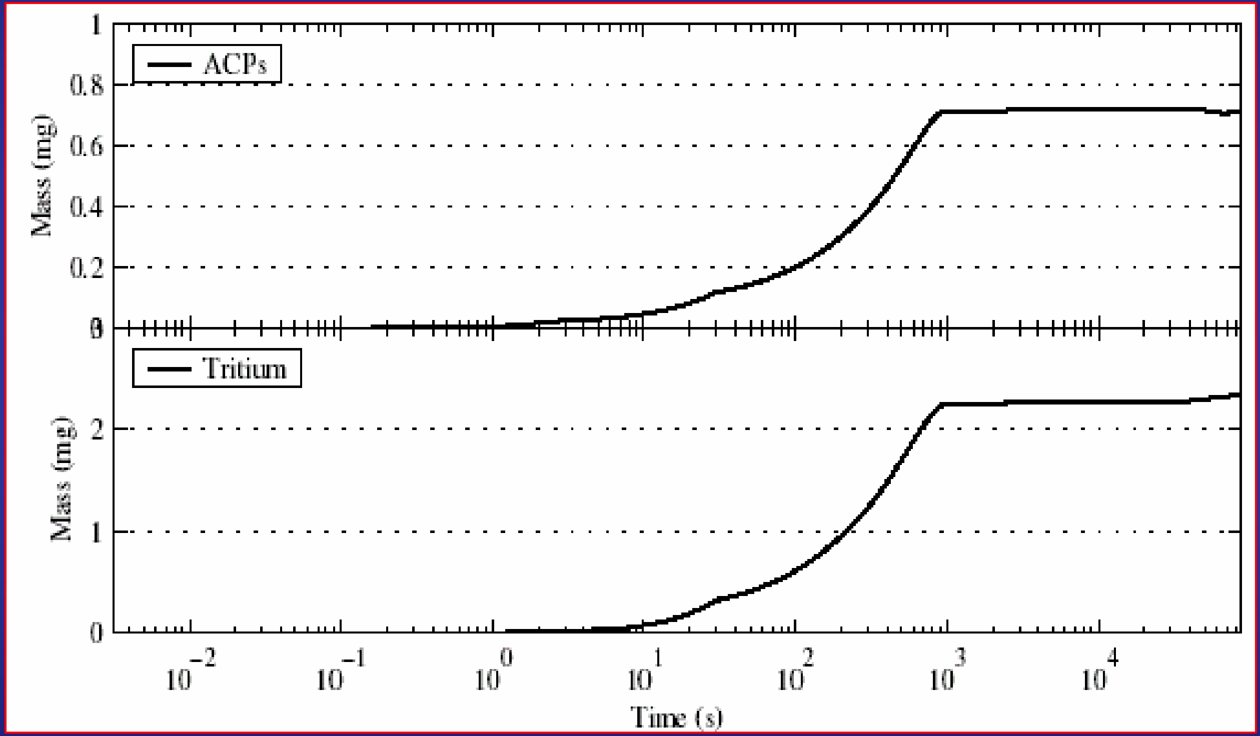
Model B

Codes used for accident analyses

Sequence	Model A	Model B	Model C
Ex-VV LOCA	MELCOR	MELCOR + COCOSYS	
Ex-VV / in-VV LOCA	MELCOR	MELCOR + COCOSYS	
LOFA / in-VV LOCA	APROS	ATHENA + ECART	MELCOR
Loss of heat sink	APROS	ATHENA + ECART	

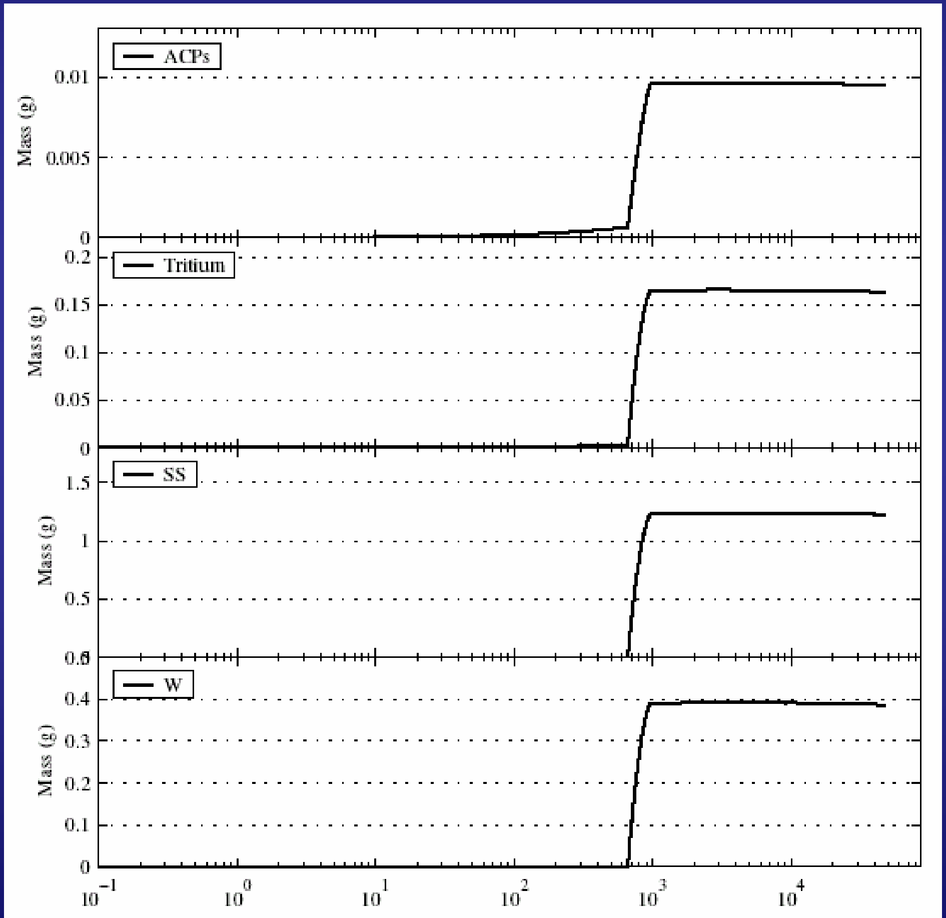
Accident analyses - example of results

- Model A ex-vessel LOCA



Accident analyses - example of results

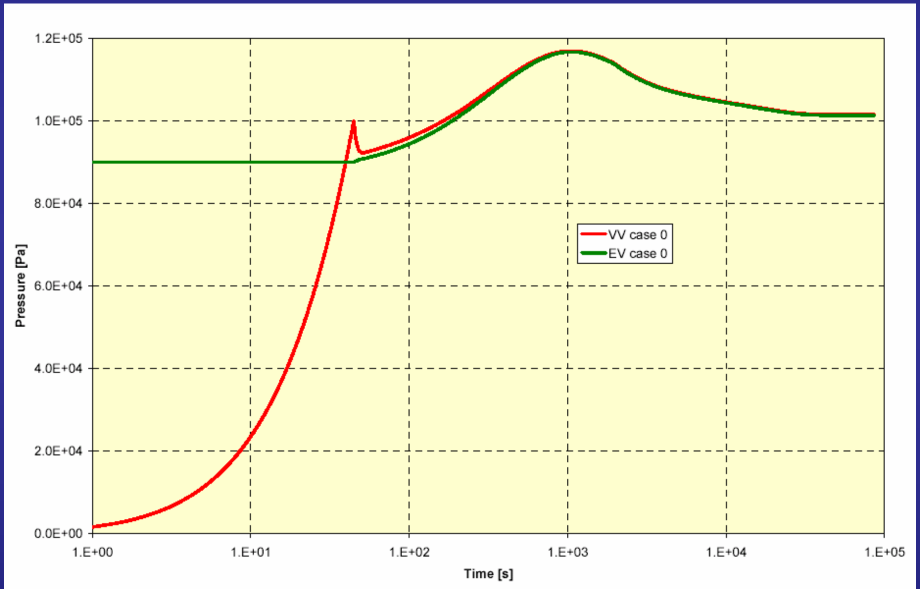
- Model A ex-vessel / in-vessel LOCA



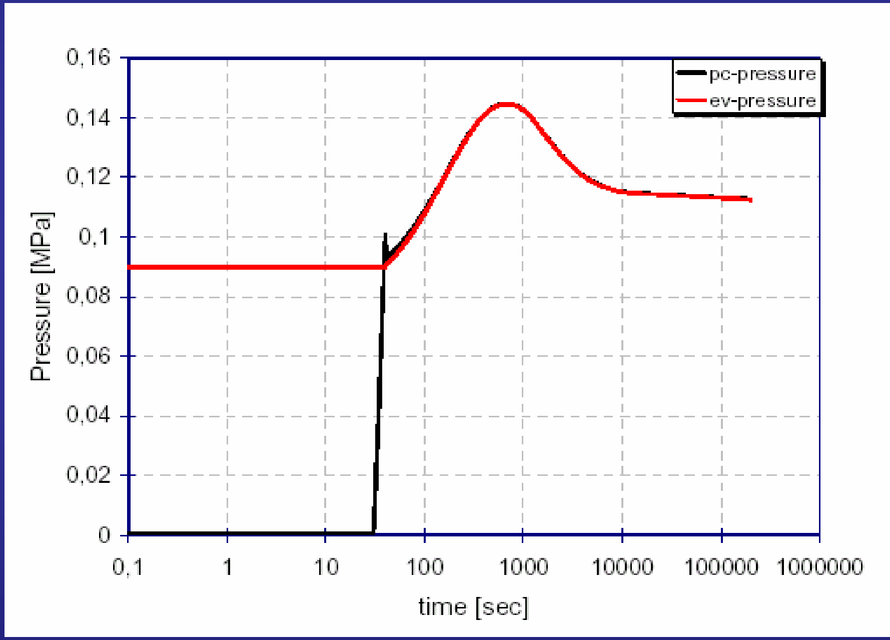
Loss of flow / ex-vessel LOCA

- Pressure in vacuum vessel

ECART results for Model B



MELCOR results for Model C



Release from loss of flow / ex-vessel LOCA

- Plant Model B

	Environmental Source Term mass [g]
Tritium	3.5
Dust (activated steel)	4.6
Dust (activated tungsten)	14.5

Assumptions in bounding accident modelling

- Complete loss of all coolant from every loop in the plant
  - all first walls, blankets, divertors, shields, vacuum vessel
  - loss is instantaneous
- No operation of any active safety system for indefinite period
- Heat rejection is only by passive transport:
  - conduction through layers of material and radiation across gaps
- Very conservative assumptions
  - to ensure result is **bounding**
  - assumptions may be non-physical or contradictory

Calculation methods for bounding accidents

3-D neutronic model
(MCNP4C)

↓

Activation calcs. in all cells
(FISPACT)

↓

2-D finite element thermal
models (COSMOS/M)

↓

Mobilisation
(APMOB)

↓

HERCULES

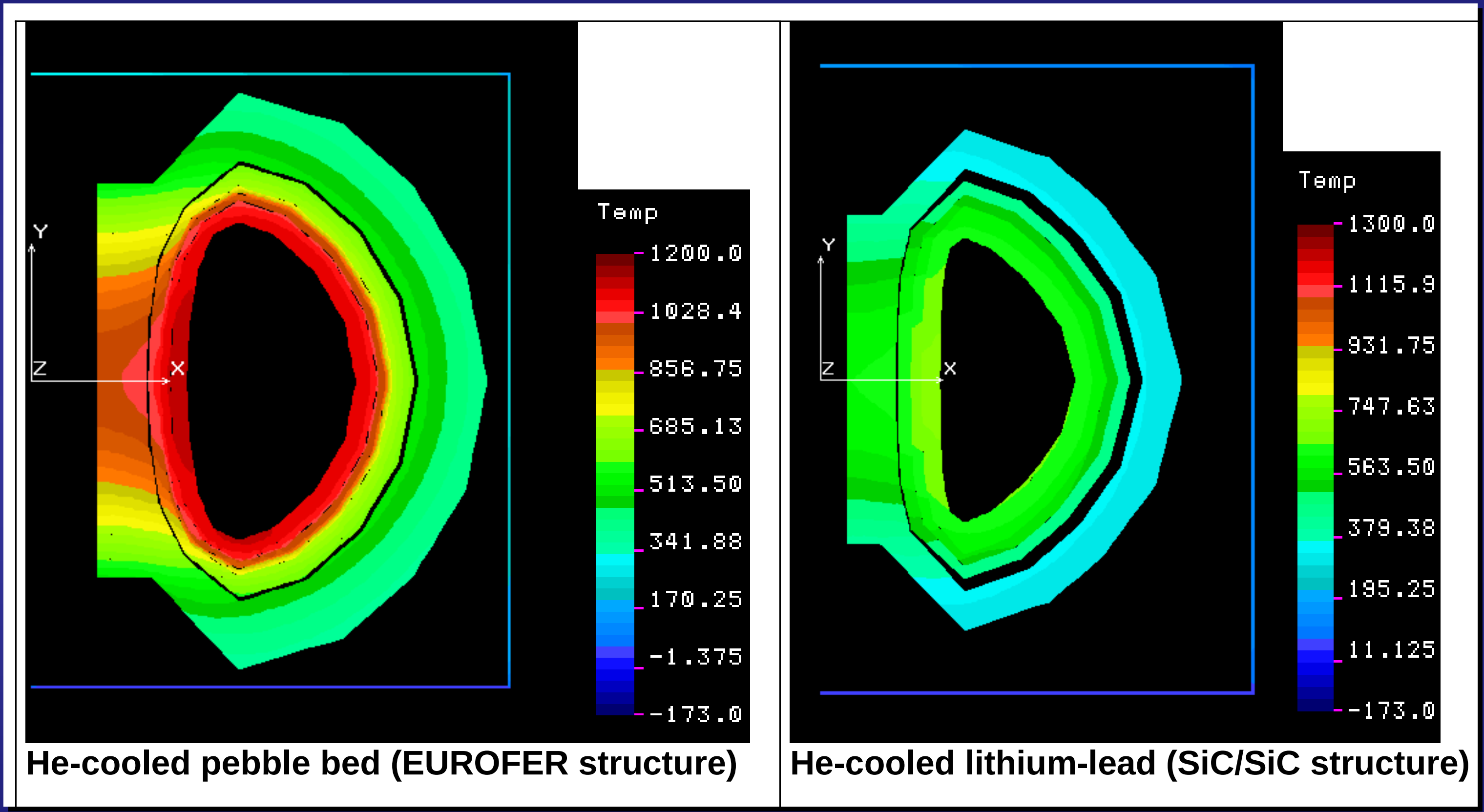
- Full 3-D models (or 2-D with toroidal symmetry)
- Automated coupling of codes (HERCULES)
- Activation and related quantities computed in all cells of model
- Temperature histories in postulated total loss of active cooling

These steps were performed for Models A and B only.

Source terms → Release, dispersion and dose calcs.

Bounding accident temperature distributions

after 100 days



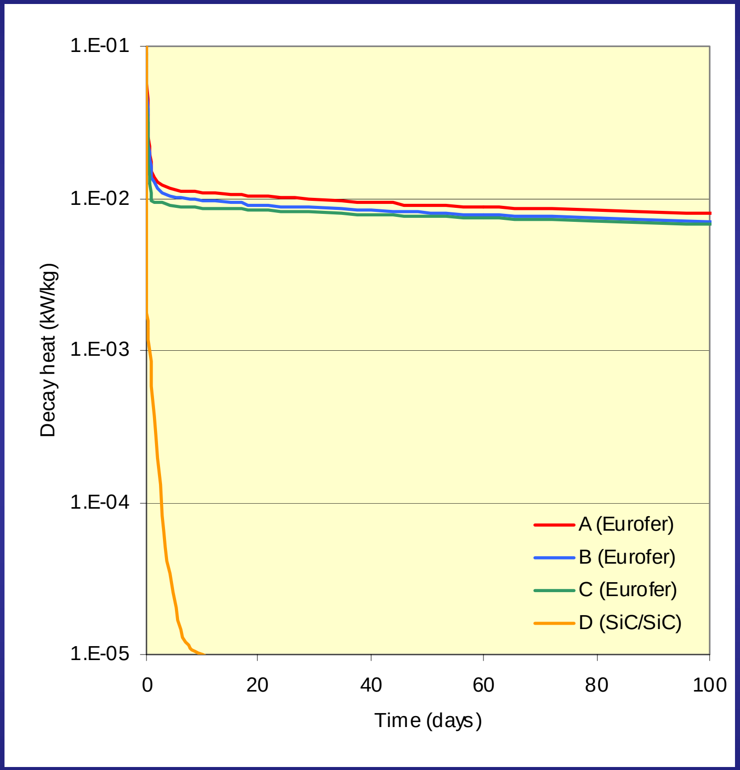
He-cooled pebble bed (EUROFER structure)

He-cooled lithium-lead (SiC/SiC structure)

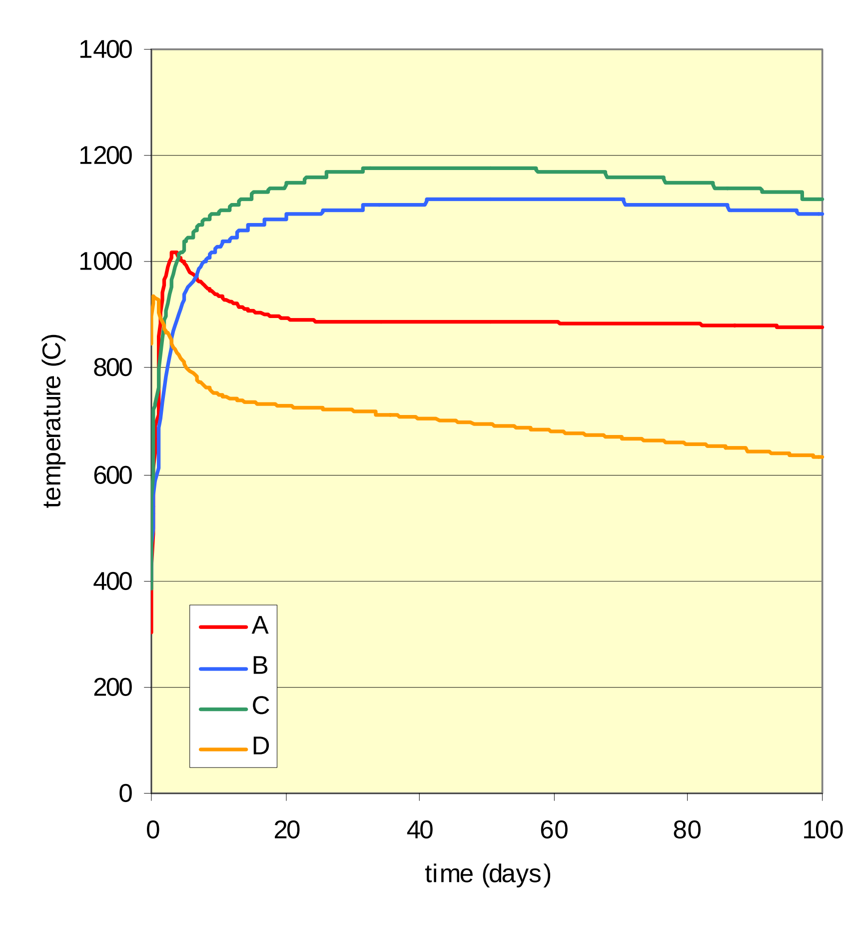
Plant Model B

Plant Model D

First wall decay heat - all models



Bounding accident first wall temperatures



Bounding accident peak temperatures

component	Model A	Model B	Model C	Model D
FW	1030^{o} C	1130^{o} C	1180^{0} C	935^{0} C
blanket	1000^{o} C	1130^{o} C	1190^{0} C	934^{0} C
shield	918^{o} C	1140^{o} C	1190^{0} C	881^{0} C
VV	836^{o} C	1040^{o} C	1150^{0} C	716^{0} C
TF coil	772^{o} C	990^{o} C	1120^{0} C	692^{0} C
cryostat	165^{o} C	216^{o} C	230^{0} C	123^{0} C
divertor	1240^{o} C	1140^{o} C	1210^{0} C	908^{0} C

Release from bounding accidents

- Assumed mobilised source term:
  - 1kg in-vessel tritium (as HTO)
  - 10 kg dust (7.6 kg steel, 2.4 kg tungsten)
  - 500g activated corrosion products (Plant Model A only)
  - volatilised activated structure as computed by APMOB (based on empirical data from INEEL tests)
- Confinement and aerosol modelling using FUSCON

Dispersion and dose

- Dispersion and dose calculations using COSYMA
  - results from 95% percentile point on real weather distributions based on Karlsruhe, Germany
  - 10m release height
  - dose commitment to Most Exposed Individual at 1km site boundary in 7-day exposure

Maximum 7-day doses to MEI

Plant Model	Dose [mSv]			
	Bounding Accident Sequence	Ex-VV LOCA	Ex-VV LOCA + in-VV LOCA	LOFA + in-VV LOCA
Water-cooled lithium-lead	1.16	0.0017	0.16	
Helium-cooled pebble bed	18.1			0.42

- Compare with target of 50 mSv maximum (trigger for public evacuation according to IAEA guidelines)

Occupational Safety

- Largest ORE doses predicted for water-cooled Plant Model A
  - (but no new detailed study)
- Study of ORE from fuel cycle system
  - vacuum pumping system largest potential contributor

Effluents from normal operation

- Earlier SEAFP results re-assessed for PPCS Power Plant

Maximum public doses μSv/year

	Model A		Model B	
	gas	liquid	gas	liquid
Tritium (HT + HTO)	0.87	0.05	0.28	0.003
Activation products	0.02	0.02	0.004	0
total	0.89	0.07	0.28	0.003

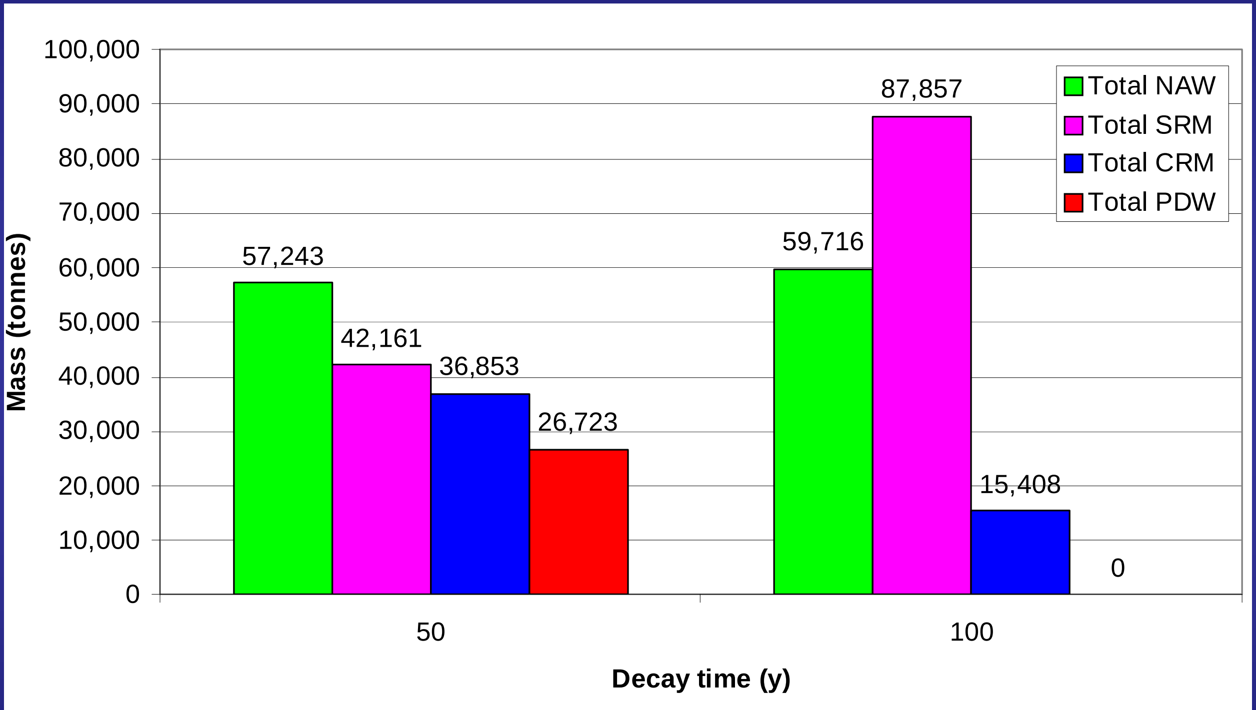
Categorisation of active material

- Summation of all active material from plant, including
  - blanket and divertor replacements during operation
  - all fixed components at end-of life
- Categories adopted as in earlier SEAFP studies:
  - Clearance according to IAEA recommendations (1996)
  - non-cleared material assessed for recycling suitability on radiological criteria:

Category	Gamma dose rate	Decay heat
Hands-on recycle	< 10 μSv/hr	
Simple recycle	< 2 mSv/hr	< 1 W/m^3
Complex recycle	2 – 20 mSv/hr	1 – 10 W/m^3
Permanent disposal	> 20 mSv/hr	> 10 W/m^3

  - Limits for "complex" (remote handling) recycling now believed to be very conservative

PPCS Plant Model A



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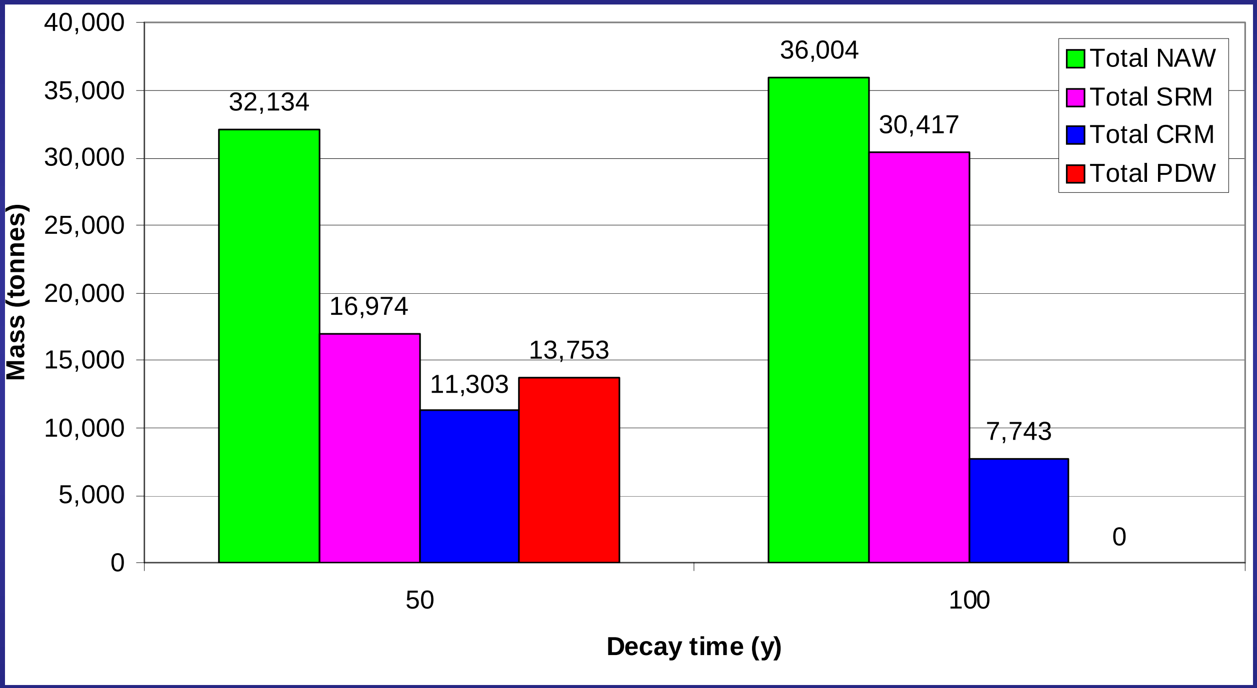
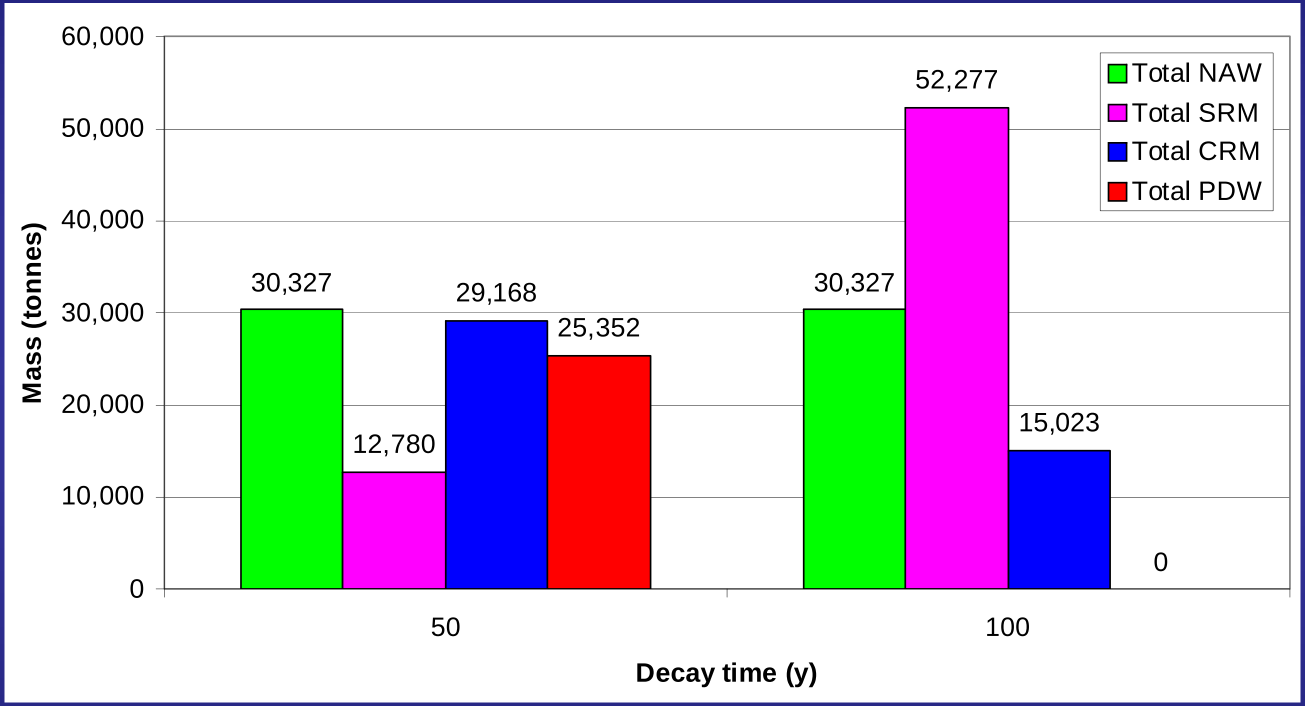
PPCS Plant Model B



PPCS Plant Model C



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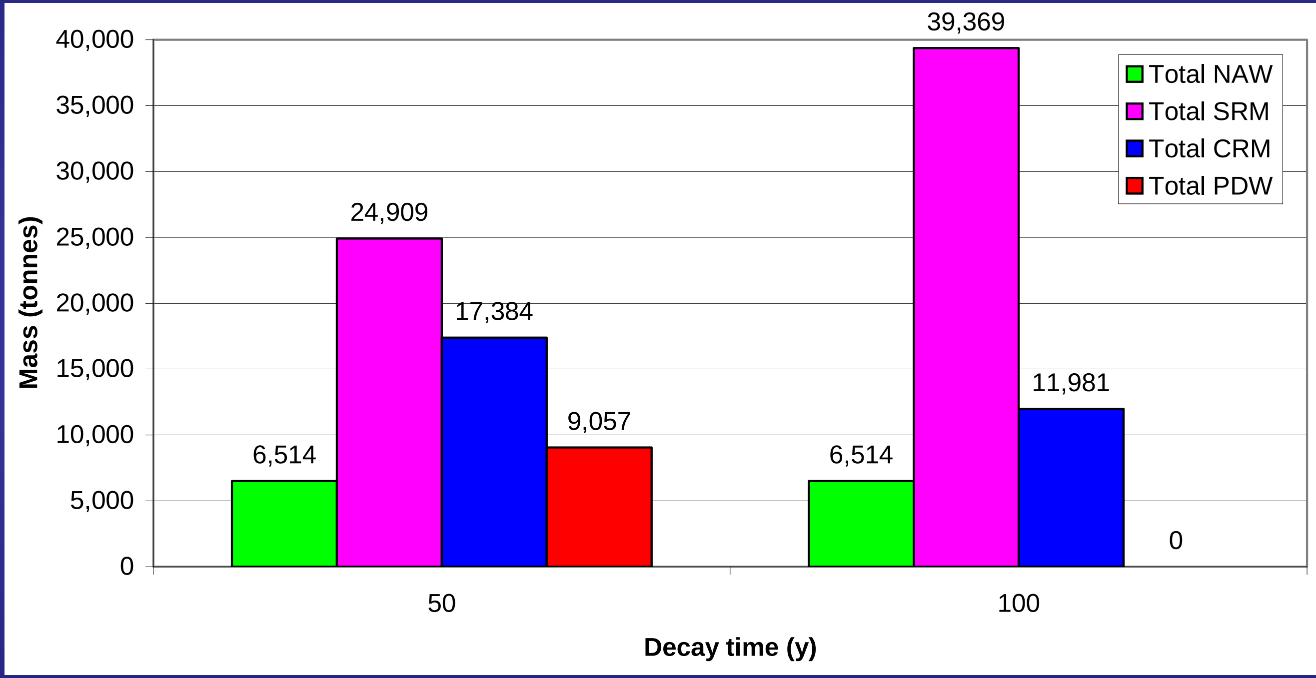
PPCS Plant Model D



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Categorisation of waste - summary

- **No** permanent disposal waste after 100 years
  - result holds for all power plant designs and variants
  - *except* when TZM molybdenum alloy used in divertor (reminder of importance of materials selection)
- Recycling of fusion material seems possible on radiological grounds
- But there are other factors
  - Will there be feasible recycling operations for the relevant materials?
  - Will the processing be economically viable?
- Proper evaluation of potential for recycling is essential

PPCS overall conclusions on S&E (1)

- The broad features of all the safety and environmental conclusions of earlier studies have been sustained, and demonstrated with increased confidence and understanding.
- If a total loss of active cooling were to occur during the burn, the plasma would switch off passively and any temperature increase due to residual decay heat cannot lead to melting of the structures. This result is achieved without any reliance on active safety systems or operator actions.
- The maximum radiological doses - assessed with deliberate pessimism - to the public arising from the most severe conceivable accident driven by in-plant energies would be below the level at which evacuation would be considered and needed and would not be much greater than typical annual doses from natural causes. This result also is achieved without reliance on active safety systems or operator actions.

PPCS overall conclusions on S&E (2)

- Confinement damage by an extremely rare (hypothetical) ultra-energetic ex-plant event, such as an earthquake of hitherto never experienced magnitude, would, on pessimistic assumptions, give rise to health effects smaller than the typical consequences of the external hazard itself. On realistic assumptions the maximum dose would be lower, and this very hypothetical scenario could be removed by design provision: essentially, this would be an economic issue.

CS overall conclusions on S&E (3)

he radiotoxicity of the materials decays relatively rapidly – very pidly at first and broadly by a factor ten thousand over a hundred ears.

uch of this material, after an adequate decay time, falls to levels o low that it could be “cleared” from regulatory control. Other aterial could be recycled or reused in further fusion power plant onstruction, with no need for repository disposal. Thus the ctivated material from fusion power stations would not constitute waste management burden for future generations. The decision n whether to actually recycle the recyclable material is a matter r future generations to determine, possibly on economic criteria.