Appendix A

LIST OF EXPERT GROUP MEMBERS

BELGIUM

Mr. Jean-Marie CUCHET	Belgonucléaire (from 11/2004)
Mr. Servais PILATE	Belgonucléaire (until 11/2004)
Mr. Guy DEMAZY	SYNATOM
Mr. Edouard Mbala MALAMBU	SCK•CEN
Dr. Jan MARIVOET	SCK•CEN

FINLAND

Dr. Markku ANTTILA	VTT

FRANCE

Mr. Marc DELPECH	SACLAY
Ms. Christine LOAËC	SACLAY
Prof. Charles MADIC	SACLAY
Mr. Alain ZAETTA	CADARACHE

GERMANY

Mr. Ludger LAMBERS	GRSKOLN
Dr. Aleksandra SCHWENK-FERRERO	FZK

ITALY

Mr. Mario DIONISI	APATSIA
Mr. Piero RISOLUTI	CASACCIA

JAPAN

Dr. Tetsuo IKEGAMI	JNC

KOREA (REPUBLIC OF)

Dr. Kwan Sik CHUN	KAERI
Dr. Jae-Hyung YOO	KAERI

RUSSIAN FEDERATION

Mr. A.N. LUKINYKH	NIIAR

SPAIN

Dr. Enrique GONZALEZ-ROMERO	CIEMAT
Dr. Armando URIARTE	ENRESA

SWEDEN

Mr. Per-Eric AHLSTRÖM	SKB

SWITZERLAND

Prof. Jean-Marc CAVEDON	PSI
Dr. Peter WYDLER*	

UNITED KINGDOM

Dr. Samantha KING	NIREX
Dr. Colin Hunter ZIMMERMAN	BNFL
Dr. Darren PARKER	BNFL

UNITED STATES OF AMERICA

Dr. James Collins BRESEE	DOE
Dr. William HALSEY	LLNL
Mr. Leroy STEWART	DOE
Dr. Abraham VAN LUIK	DOE

International Atomic Energy Agency (IAEA)

Mr. José Luis GONZALEZ GOMEZ	IAEA

European Commission

Dr. Ved BHATNAGAR	DG Research
Mr. Enric PLA CAMPANA	DG Energy and Transport

OECD/Nuclear Energy Agency

Dr. Timo HAAPALEHTO	Nuclear Development Division
Dr. Kwang-Seok LEE	Nuclear Development Division

* With support from PSI Villigen and HSK Villigen.

Appendix B

ACRONYMS

ADS	Accelerator-driven system (sub-critical)
AFCI	Advanced fuel cycle initiative
ALMR	Advanced liquid metal (-cooled) reactor
An	Actinide, i.e. U, Pu, Np, Am or Cm. "Actinides" stands for any combination of U, Pu, Np, Am and Cm.
ANL	Argonne National Laboratory (United States)
Bq	Becquerel, unit of activity, often TBq (10^{12})
BNFL	British Nuclear Fuels Limited
BWR	Boiling water reactors
CANDU	Canadian deuterium uranium Canadian reactor
CCD	Chlorinated cobalt dicarbolide
CEA	Commissariat à l'énergie atomique (France)
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas
COGEMA	Compagnie générale des matières nucléaires
CRIEPI	Combined Research Institute of the Electric Power Industries (Japan)
DAW	Dry active waste
DepU	Depleted uranium from enrichement
DOE	Department of Energy (United States)
DUPIC	Direct Use of spent PWR fuel in CANDU reactors
EBS	Engineering barrier system
EDF	Électricité de France
EFR	European fast reactor concept
ENEA	Ente per le Nuove tecnologie, l'Energia e l'Ambiente (Italy)
ENRESA	Empresa Nacional de Residuos Radioactivos S.A.
FP	Fission product
FR	Fast reactor (operating at criticality)
Framatome	Franco-Américaine de constructions atomiques
FZJ	Forschungszentrum (research centre) Jülich (Germany)

FZK	Forschungszentrum Karlsruhe (Germany)
Gen-IV	Next generation nuclear energy systems
GeV	Giga-electron-volt (10^9 electron volts)
GFR	Gas-cooled fast reactor
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit mbH
GWd	Gigawatt-day (usually thermal, i.e. without allowance for conversion losses)
GWe	Gigawatt (electrical)
HBU	High burn-up core
HEPA	High efficiency particulate absorbing
HEU	Highly enriched uranium
HIC	High integrity containers
HLLW	High-level radioactive liquid waste
HLW	High-level waste
HM	Heavy metal
HTOM	High-temperature operating mode
HWR	Heavy water reactor
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IFR	Integral fast reactor
ILW	Intermediate-level waste
JAERI	Japan Atomic Energy Research Institute
JNC	Japan Nuclear Cycle Development Institute
KAERI	Korea Atomic Energy Research Institute
LANL	Los Alamos National Laboratory (US)
LBE	Lead bismuth eutectic
LILW	Low- and intermediate-level waste
LL	Long-lived waste
LLFP	Long-lived fission products
LMFR	Liquid metal (cooled) fast reactor
Ln	Lanthanide
LLNL	Lawrence Livermoore National Laboratory (United States)
LSM	Low specific material
LWRs	Light water reactors (either pressurised or boiling)
MA	Minor actinides, principally neptunium, americium and curium

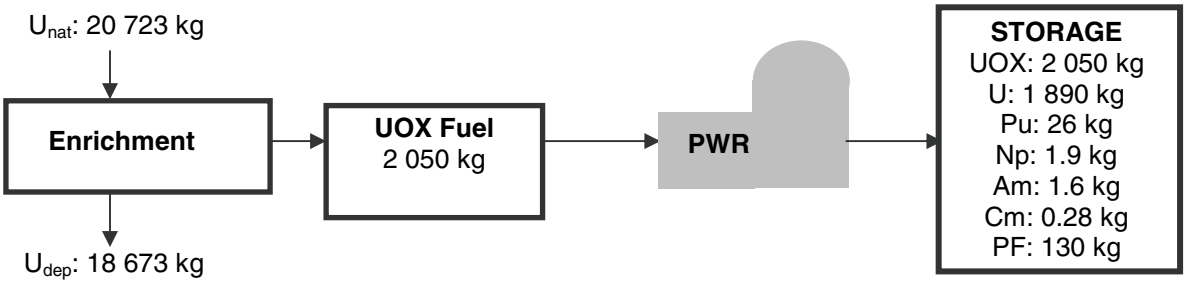
MCA	Multi-criteria analysis
MEGAPIE	MEGAwatt Pilot Experiment
Mo	molybdenum
MOX	Mixed oxide fuel, uranium and plutonium unless otherwise specified
Mt	Metric ton
MW	Megawatt (million watts)
NIRAS	Nationale instelling voor radioactief afval en verrijkte splijtstoffen
ONDRAF	Organisme national des déchets radioactifs et des matières fissiles enrichies
OREOX	Oxidation and reduction of oxide fuel
ORIENT	Optimisation by removing impedimental elements
P&T	Partitioning and transmutation
PSI	Paul Scherrer Institute (Switzerland)
PUREX	Plutonium and uranium reduction of oxide fuel
PWR	Pressurised water (cooled and moderated) reactor
RIAR	Research Institute of Atomic Reactors (Russia)
SAFIR 2	Safety Assessment and Feasibility Interim Report
SCK•CEN	Studiecentrum voor Kernenergie – Centre d'étude de l'Énergie nucléaire
SepU	Separated uranium from reprocessing
SL	Short-lived waste
SNF	Spent nuclear fuel
SWU	Separating work unit
TBP	tri-butyl-phosphate, an organic molecule to extract plutonium (Pu) and uranium (U) from dissolved spent fuel
tHM	ton of heavy metal
t.l.e.	temperature limit exceeded
TRU	Transuranic elements, i.e. Pu, Np, Am and Cm
tU	ton of uranium
TWhe	Terawatt-hour (electrical), i.e. 10^{12} watt-hours after conversion losses
UOX	Uranium oxide fuel
UREX+	Aqueous fuel reprocessing scheme
VR	Volume reduction factor

Appendix C

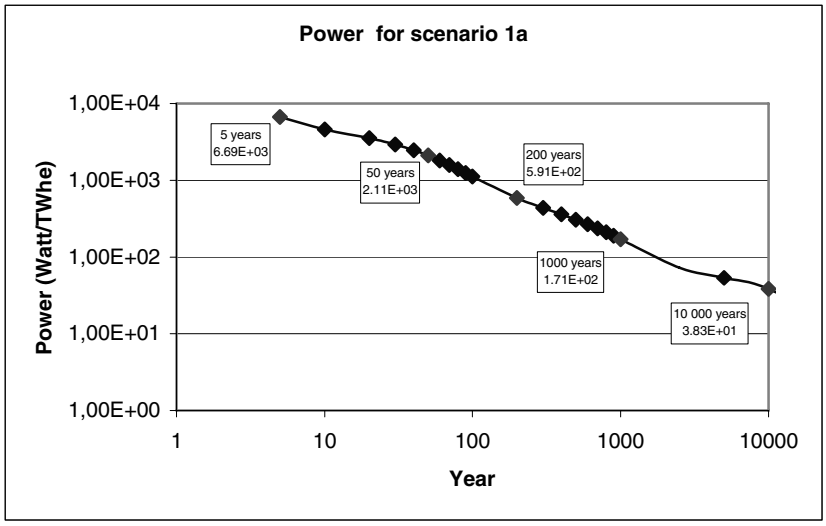
FUEL CYCLE DESCRIPTIONS AND MAIN DATA

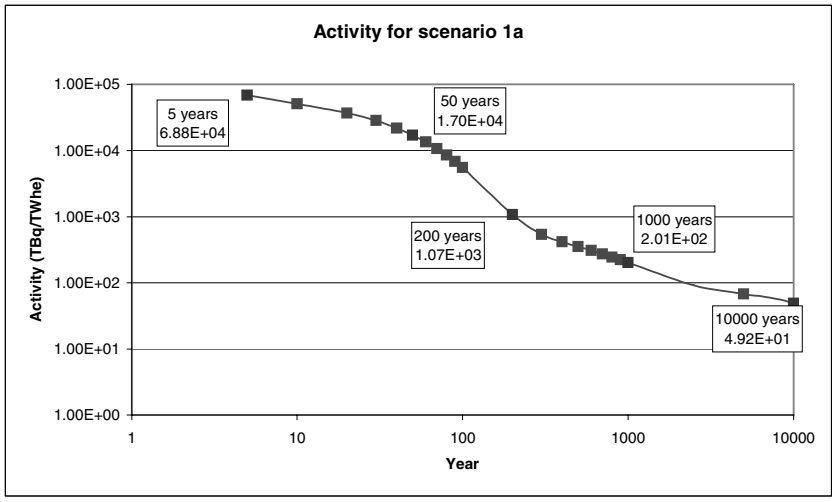
Scheme 1a
Open cycle

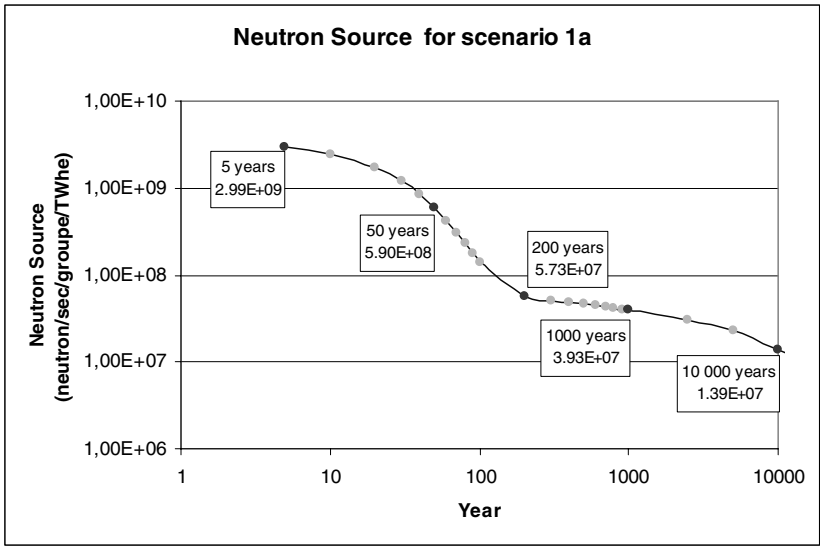
Fuel Characteristics and Mass Flows (kg/TWhe)



	PWR/UOX
ENRICHMENT	
Natural Uranium	0.71% ^{235}U
Depleted Uranium	0.25% ^{235}U
Enriched Uranium	4.90% ^{235}U
FUEL	UOX
Burn Up	60 GWd/tHM
Ageing time	2 years
Cooling time	5 years
REACTOR	PWR – EPR type
Thermal power	4 250 MWth
Electrical power	1 450 MWe

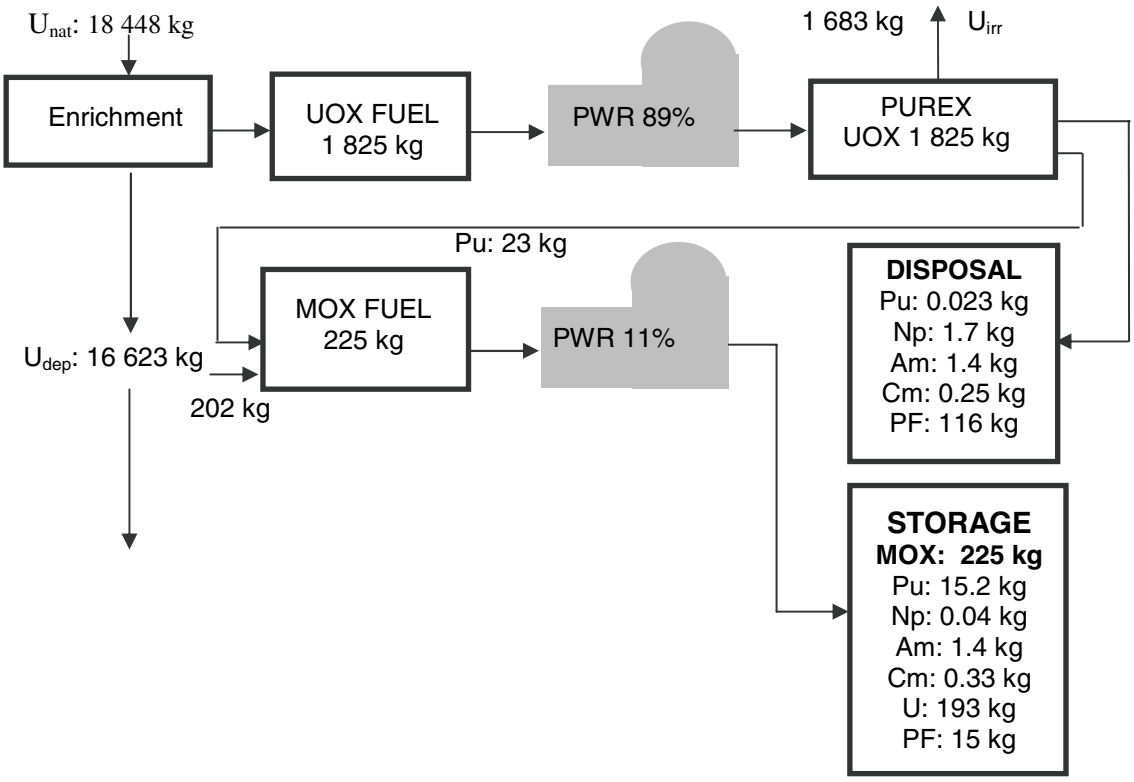

Power for scenario 1a
Power (Watt/TWhe)
1,00E+04
1,00E+03
1,00E+02
1,00E+01
1,00E+00
1
10
100
1000
10000
Year
5 years
6.69E+03
50 years
2.11E+03
200 years
5.91E+02
1000 years
1.71E+02
10 000 years
3.83E+01


Activity for scenario 1a
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
100
1000
10000
Year
5 years
6.88E+04
50 years
1.70E+04
200 years
1.07E+03
1000 years
2.01E+02
10000 years
4.92E+01

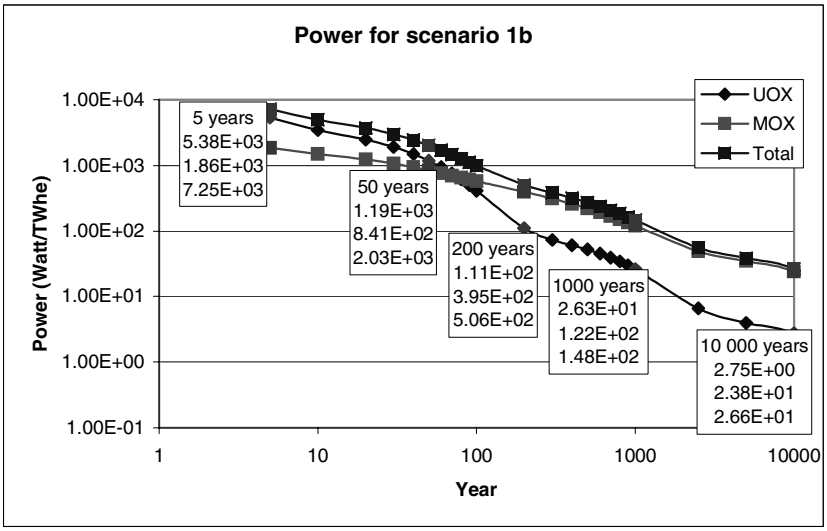

Neutron Source for scenario 1a
Neutron Source
(neutron/sec/groupe/TWhe)
1,00E+10
1,00E+09
1,00E+08
1,00E+07
1,00E+06
1
10
100
1000
10000
Year
5 years
2.99E+09
50 years
5.90E+08
200 years
5.73E+07
1000 years
3.93E+07
10 000 years
1.39E+07

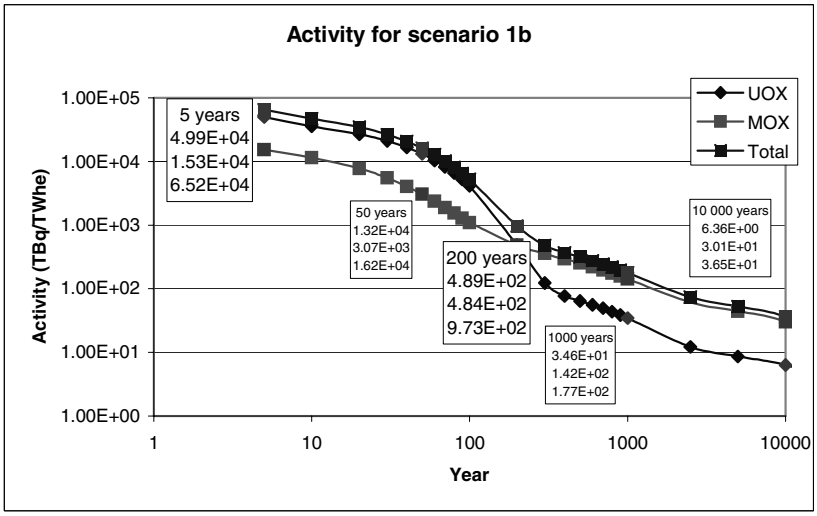
Scheme 1b
UOX reprocessed (PUREX) – Pu once cycling in MOX

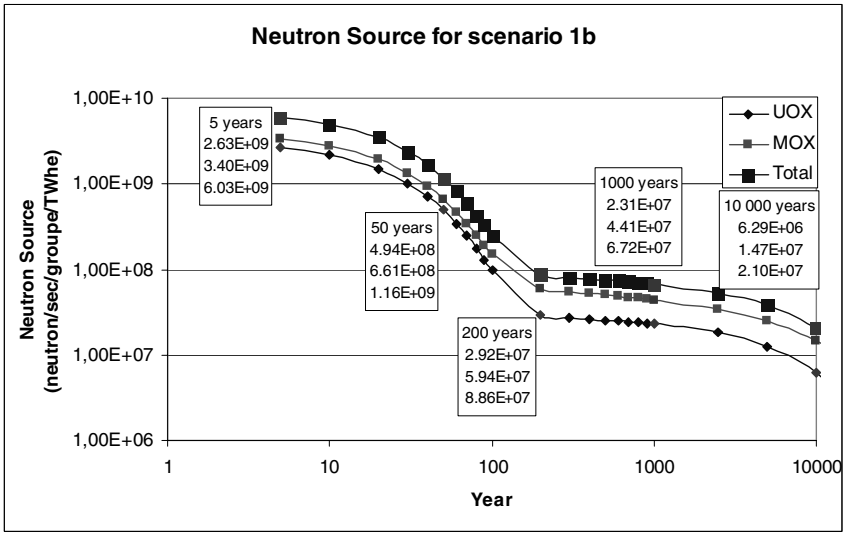
Fuel Characteristics and Mass Flows (kg/TWhe)



	PWR/UOX	PWR/MOX
ENRICHMENT Natural Uranium Depleted Uranium Enriched Uranium	 0.71% ^{235}U 0.25% ^{235}U 4.90% ^{235}U	 0.25% ^{235}U
FUEL Burn Up Ageing time Cooling time Pu Content Fissile Pu	UOX 60 GWd/tHM 2 years 5 years – 	MOX 60 GWd/tHM 2 years 5 years 10% 63%
REACTOR Thermal power Electrical power	PWR – EPR type 4 250 MWth 1 450 MWe	PWR – EPR 100% MOX 4 250 MWth 1 450 MWe
REPROCESSING Losses	PUREX 0.1% U,Pu 100% MA, FP	–

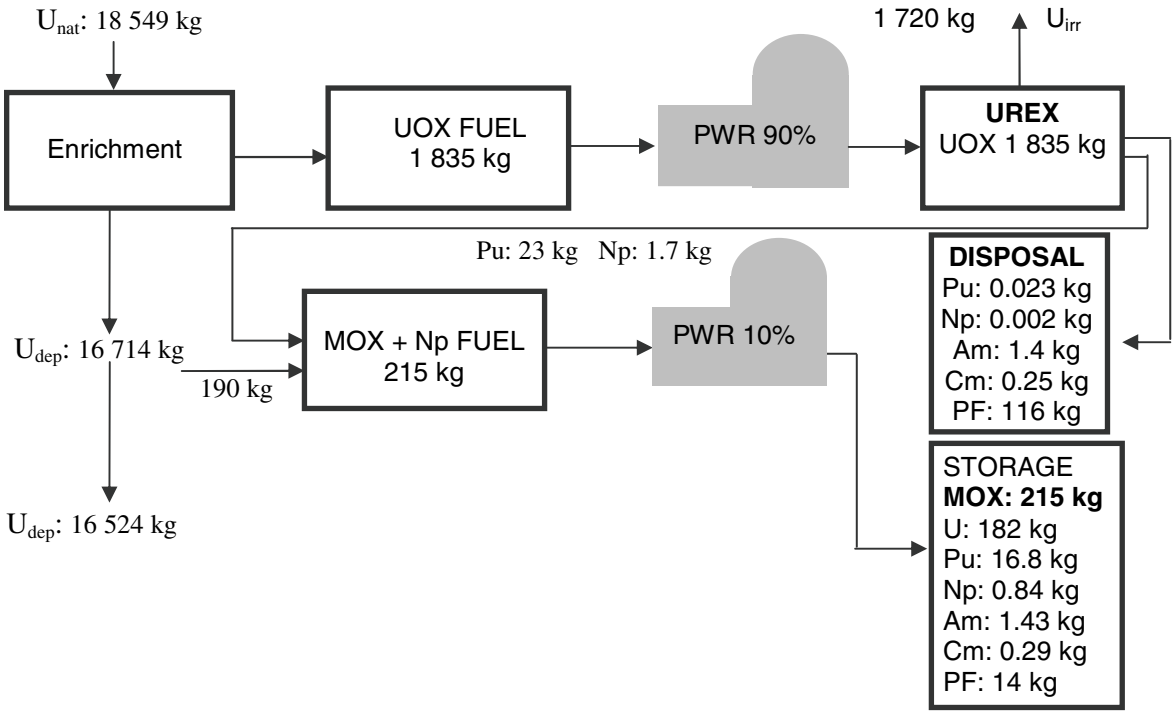

Power for scenario 1b
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
5.38E+03
1.86E+03
7.25E+03
50 years
1.19E+03
8.41E+02
2.03E+03
200 years
1.11E+02
3.95E+02
5.06E+02
1000 years
2.63E+01
1.22E+02
1.48E+02
10 000 years
2.75E+00
2.38E+01
2.66E+01


Activity for scenario 1b
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
4.99E+04
1.53E+04
6.52E+04
50 years
1.32E+04
3.07E+03
1.62E+04
200 years
4.89E+02
4.84E+02
9.73E+02
1000 years
3.46E+01
1.42E+02
1.77E+02
10 000 years
6.36E+00
3.01E+01
3.65E+01


Neutron Source for scenario 1b
Neutron Source
(neutron/sec/groupe/TWhe)
1,00E+10
1,00E+09
1,00E+08
1,00E+07
1,00E+06
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
2.63E+09
3.40E+09
6.03E+09
50 years
4.94E+08
6.61E+08
1.16E+09
200 years
2.92E+07
5.94E+07
8.86E+07
1000 years
2.31E+07
4.41E+07
6.72E+07
10 000 years
6.29E+06
1.47E+07
2.10E+07

Scheme 1c
UOX reprocessed (UREX) Pu, Np once cycling in MOX

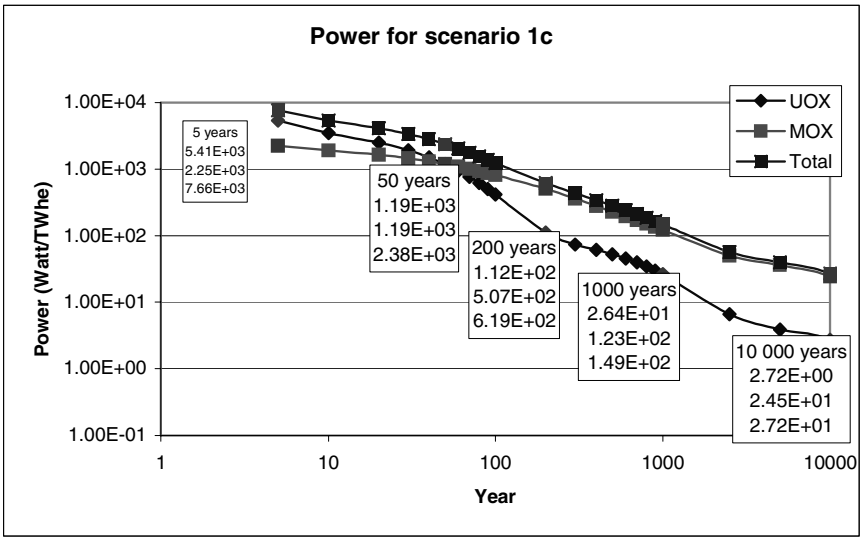
Fuel Characteristics and Mass Flows (kg/TWhe)

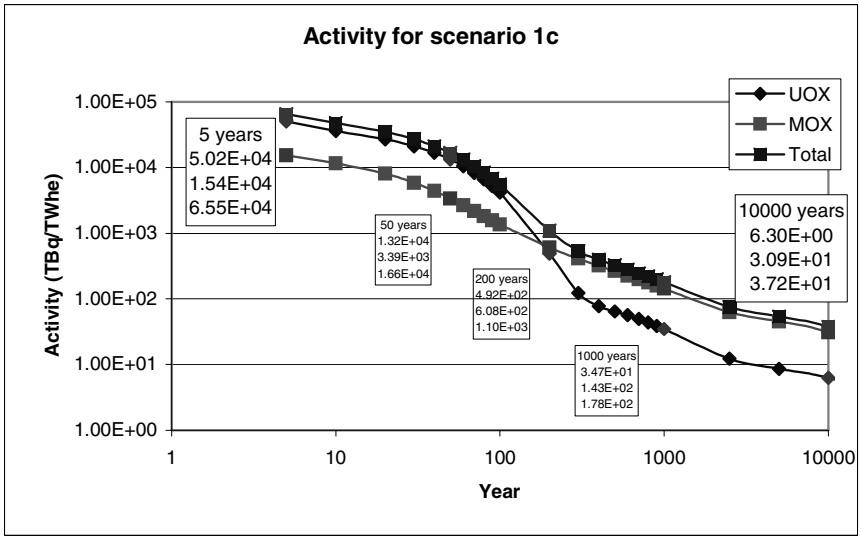


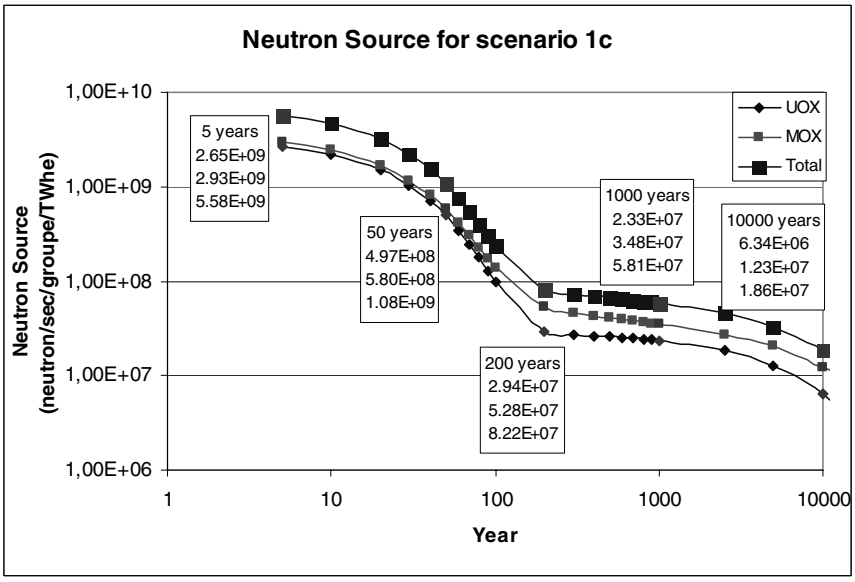
	PWR/UOX	PWR/MOX
ENRICHMENT		
Natural Uranium	0.71% ^{235}U	
Depleted Uranium	0.25% ^{235}U	0.25% ^{235}U
Enriched Uranium	4.90% ^{235}U	
FUEL	**UOX**	**MOX with Np**
Burn Up	60 GWd/tHM	60 GWd/tHM
Ageing time	2 years	2 years
Cooling time	5 years	5 years
Pu, Np Content	–	11.5%
Fissile Pu	–	63%
REACTOR	PWR – EPR type	PWR – EPR 100% MOX
Thermal power	4 250 MWth	4 250 MWth
Electrical power	1 450 MWe	1 450 MWe
REPROCESSING	UREX	
Losses	0.1% U, Pu, Np 100% other MA, FP	–

COMMENTS:

1. This scenario has been calculated by CEA on the basis of an EPR RMA for MOX fuel.
2. EPR 100% MOX has an increased moderation ratio up to 2.5.
3. For FP mass composition of irradiated MOX with Np fuel, MOX fuel has been used (same spectra).

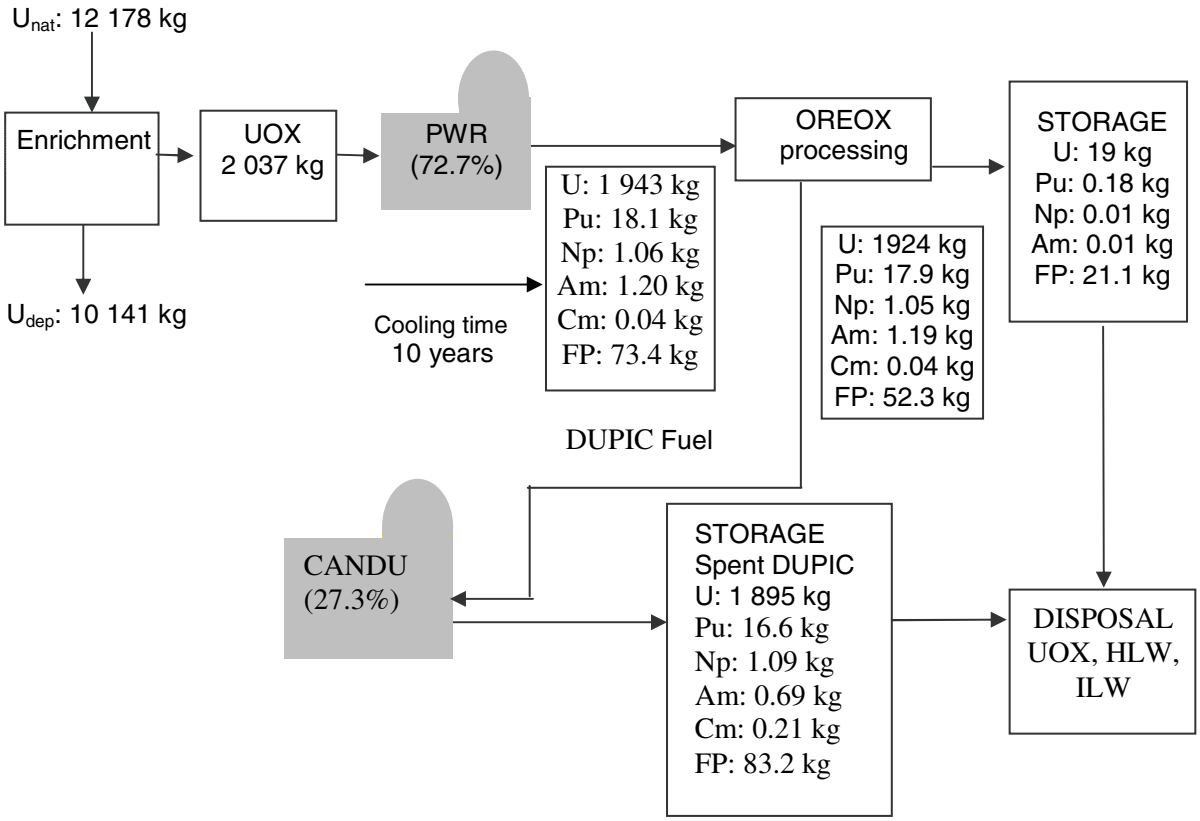

Power for scenario 1c
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
5.41E+03
2.25E+03
7.66E+03
50 years
1.19E+03
1.19E+03
2.38E+03
200 years
1.12E+02
5.07E+02
6.19E+02
1000 years
2.64E+01
1.23E+02
1.49E+02
10 000 years
2.72E+00
2.45E+01
2.72E+01


Activity for scenario 1c
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
5.02E+04
1.54E+04
6.55E+04
50 years
1.32E+04
3.39E+03
1.66E+04
200 years
4.92E+02
6.08E+02
1.10E+03
1000 years
3.47E+01
1.43E+02
1.78E+02
10000 years
6.30E+00
3.09E+01
3.72E+01

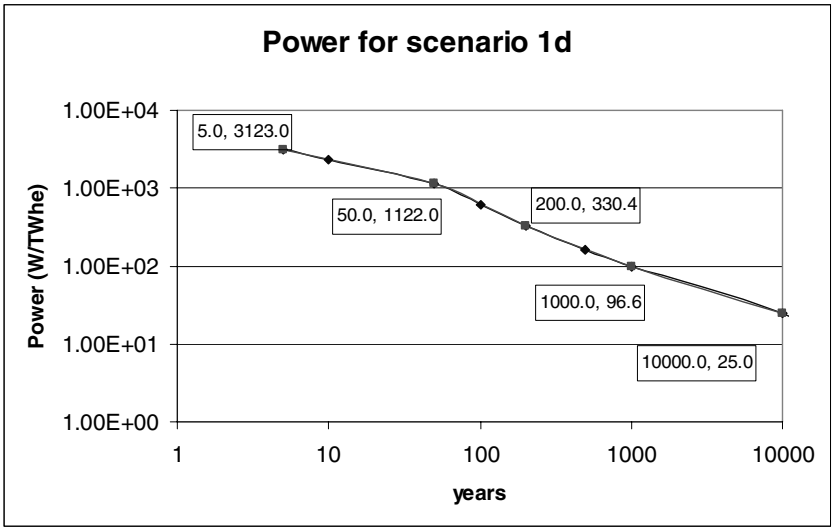

Neutron Source for scenario 1c
Neutron Source
(neutron/sec/groupe/TWhe)
1,00E+10
1,00E+09
1,00E+08
1,00E+07
1,00E+06
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
2.65E+09
2.93E+09
5.58E+09
50 years
4.97E+08
5.80E+08
1.08E+09
200 years
2.94E+07
5.28E+07
8.22E+07
1000 years
2.33E+07
3.48E+07
5.81E+07
10000 years
6.34E+06
1.23E+07
1.86E+07

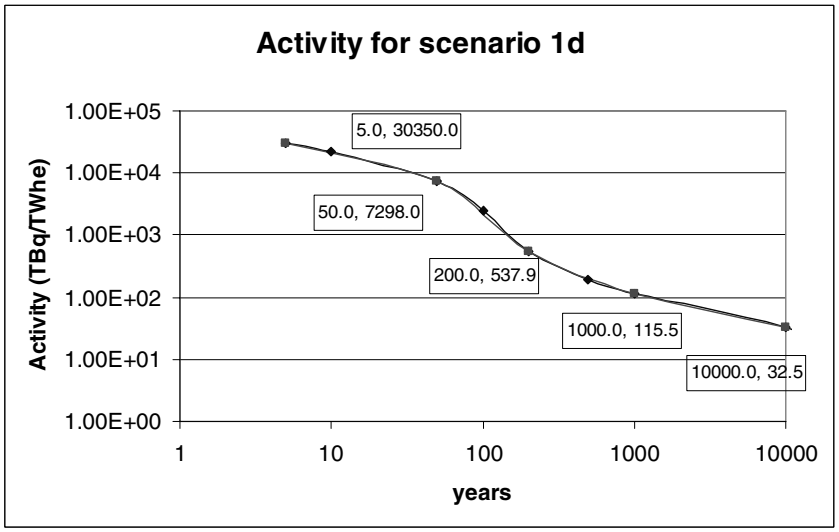
Scheme 1d
UOX processed (OREOX)

Fuel Characteristics and Mass Flows (kg/TWhe)



	PWR/UOX	CANDU/DUPIC
ENRICHMENT		
Natural Uranium	0.71% ^{235}U	
Depleted Uranium	0.25% ^{235}U	
Enriched Uranium	3.50% ^{235}U	
FUEL	**UOX**	**DUPIC**
Burn Up	35 GWd/tHM	15 GWd/tHM
Ageing time		
Cooling time	10 years	
Pu Content/SPWRF	–	0.90%
Fissile Pu	–	68.40%
REACTOR	PWR	CANDU
Thermal power	4 250 MWth	2 159 MWth
Electrical power	1 450 MWe	713 MWe
REPROCESSING	OREOX	
Losses	1% U, Pu, MA, FPs 100% Kr, Xe, Ru, Cs, I	–


Power for scenario 1d
Power (W/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
100
1000
10000
years
5.0, 3123.0
50.0, 1122.0
200.0, 330.4
1000.0, 96.6
10000.0, 25.0


Activity for scenario 1d
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
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1000
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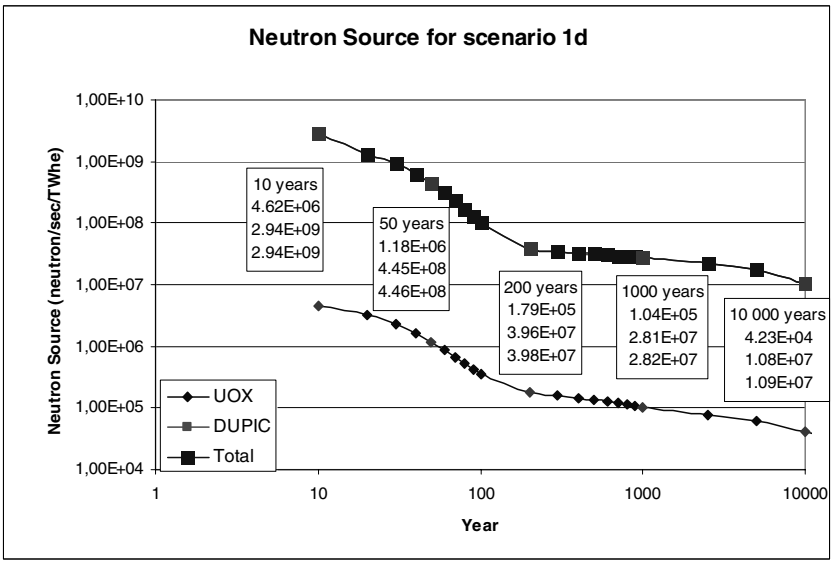
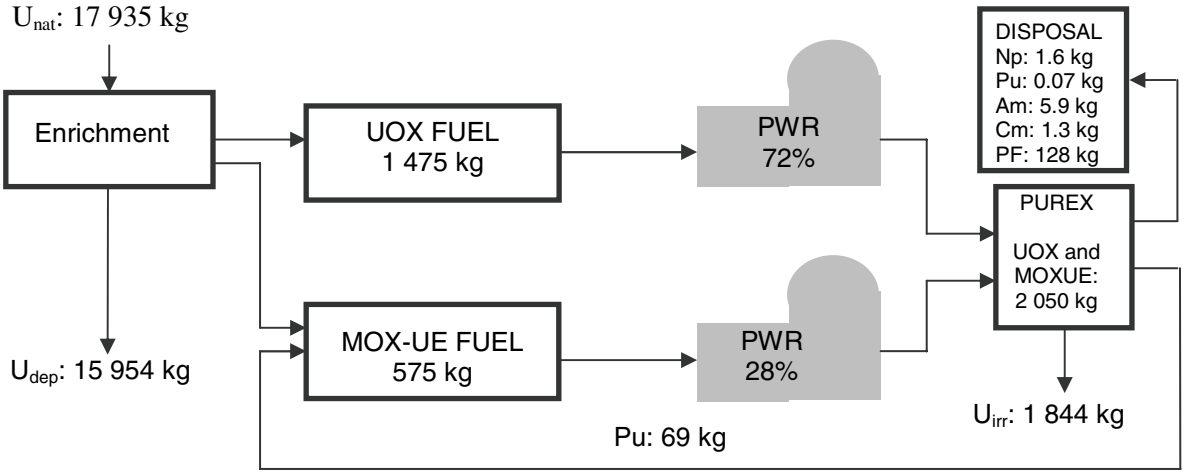
10000.0, 32.5


Neutron Source for scenario 1d
Neutron Source (neutron/sec/TWhe)
1,00E+10
1,00E+09
1,00E+08
1,00E+07
1,00E+06
1,00E+05
1,00E+04
1
10
100
1000
10000
Year
10 years
4.62E+06
2.94E+09
2.94E+09
50 years
1.18E+06
4.45E+08
4.46E+08
200 years
1.79E+05
3.96E+07
3.98E+07
1000 years
1.04E+05
2.81E+07
2.82E+07
10 000 years
4.23E+04
1.08E+07
1.09E+07
UOX
DUPIC
Total

Scheme 2a
UOX and MOX-EU reprocessed (PUREX) – Pu recycling

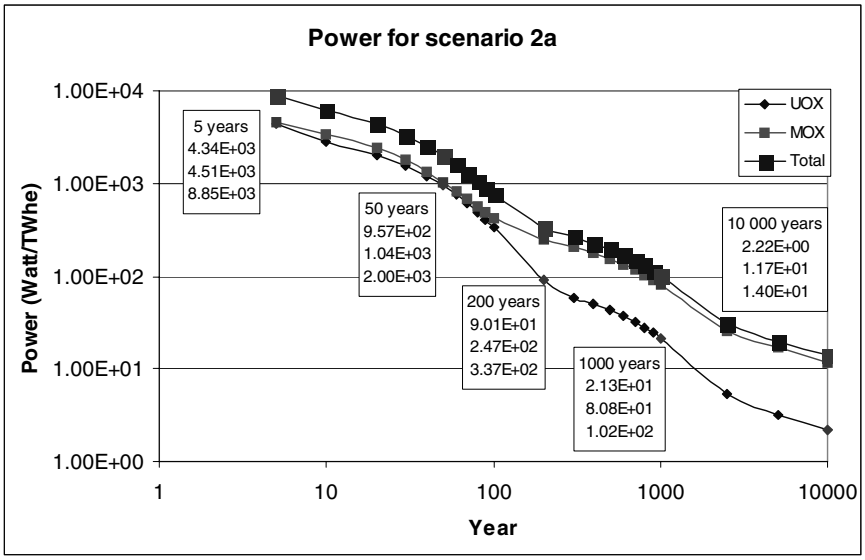
Fuel Characteristics and Mass Flows (kg/TWhe)

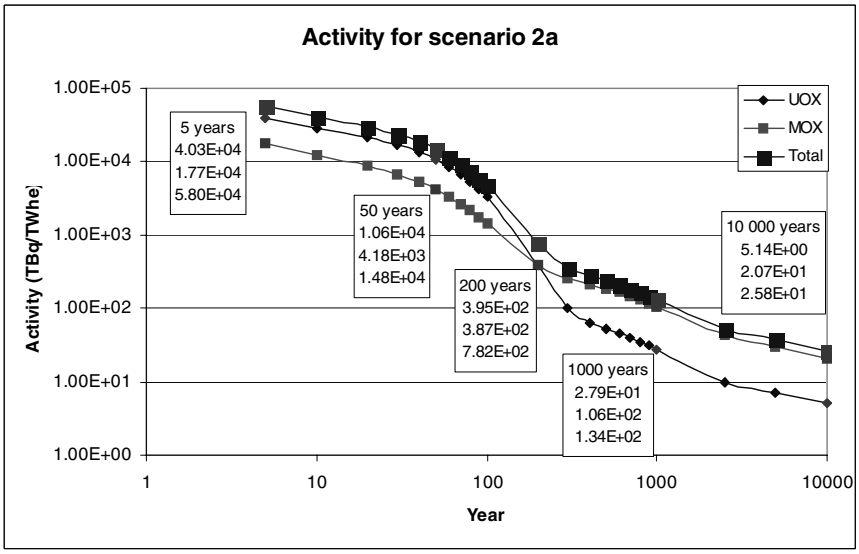


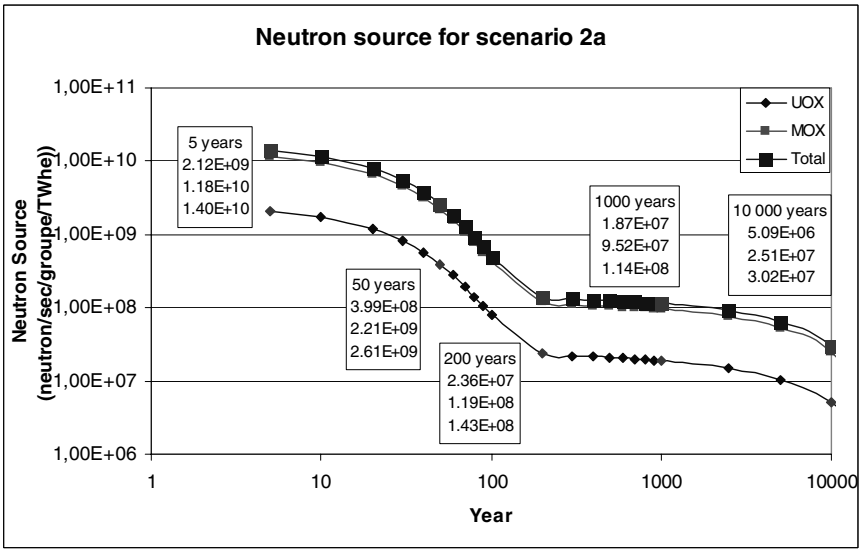
	PWR/UOX	PWR/MOX-EU
ENRICHMENT		
Natural Uranium	0.71% ^{235}U	0.71% ^{235}U
Depleted Uranium	0.25% ^{235}U	0.25% ^{235}U
Enriched Uranium	4.90% ^{235}U	3% ^{235}U
FUEL	UOX	MOX-EU
Burn Up	60 GWd/tHM	60 GWd/tHM
Ageing time	2 years	2 years
Cooling time	5 years	5 years
Pu Content	–	12%
Fissile Pu	–	56.8%
REACTOR	PWR – EPR type	PWR – EPR 100% MOX
Thermal power	4 250 MWth	4 250 MWth
Electrical power	1 450 MWe	1 450 MWe
REPROCESSING	PUREX	PUREX
Losses	0.1% U, Pu 100% MA, FP	0.1% U, Pu 100% MA, FP

COMMENTS:

1. For FP mass composition of irradiated MOX-EU fuel, MOX fuel has been used.
2. EPR MOX-EU has an increased moderation ratio up to 2.5.

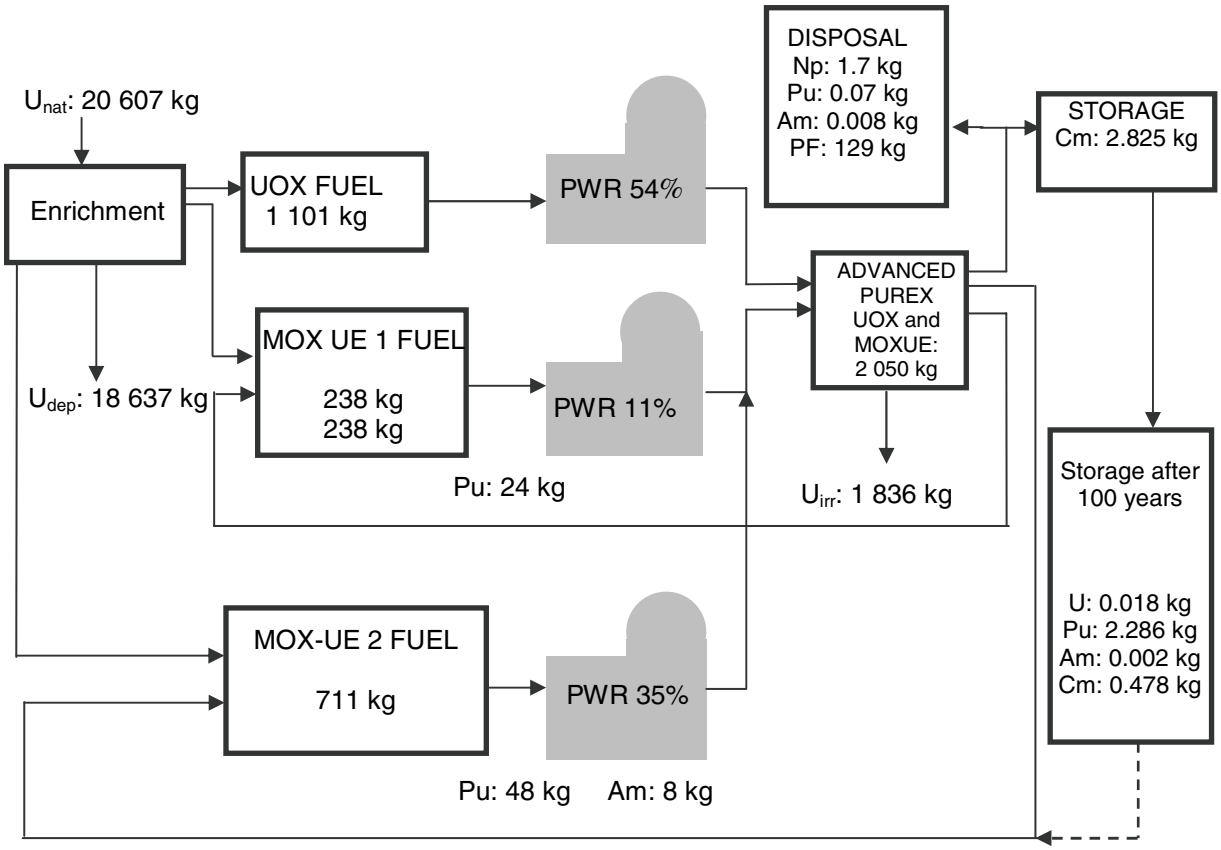

Power for scenario 2a
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
4.34E+03
4.51E+03
8.85E+03
50 years
9.57E+02
1.04E+03
2.00E+03
200 years
9.01E+01
2.47E+02
3.37E+02
1000 years
2.13E+01
8.08E+01
1.02E+02
10 000 years
2.22E+00
1.17E+01
1.40E+01


Activity for scenario 2a
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
4.03E+04
1.77E+04
5.80E+04
50 years
1.06E+04
4.18E+03
1.48E+04
200 years
3.95E+02
3.87E+02
7.82E+02
1000 years
2.79E+01
1.06E+02
1.34E+02
10 000 years
5.14E+00
2.07E+01
2.58E+01


Neutron source for scenario 2a
Neutron Source
(neutron/sec/groupe/TWhe))
1,00E+11
1,00E+10
1,00E+09
1,00E+08
1,00E+07
1,00E+06
1
10
100
1000
10000
Year
UOX
MOX
Total
5 years
2.12E+09
1.18E+10
1.40E+10
50 years
3.99E+08
2.21E+09
2.61E+09
200 years
2.36E+07
1.19E+08
1.43E+08
1000 years
1.87E+07
9.52E+07
1.14E+08
10 000 years
5.09E+06
2.51E+07
3.02E+07

Scheme 2b
UOX and MOX reprocessed (Advanced PUREX) – Am and Cm Partitioning – Cm storage – Pu and Am recycling in MOX-EU

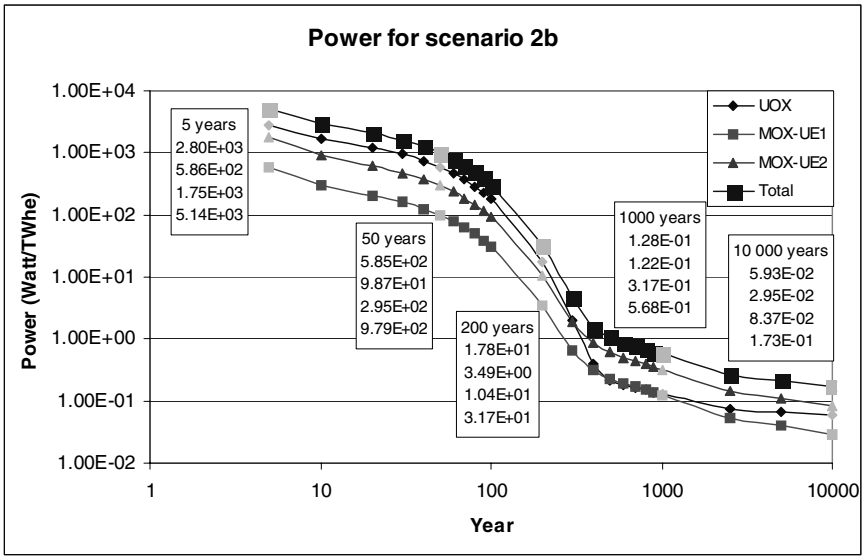
Fuel Characteristics and Mass Flows (kg/TWhe)

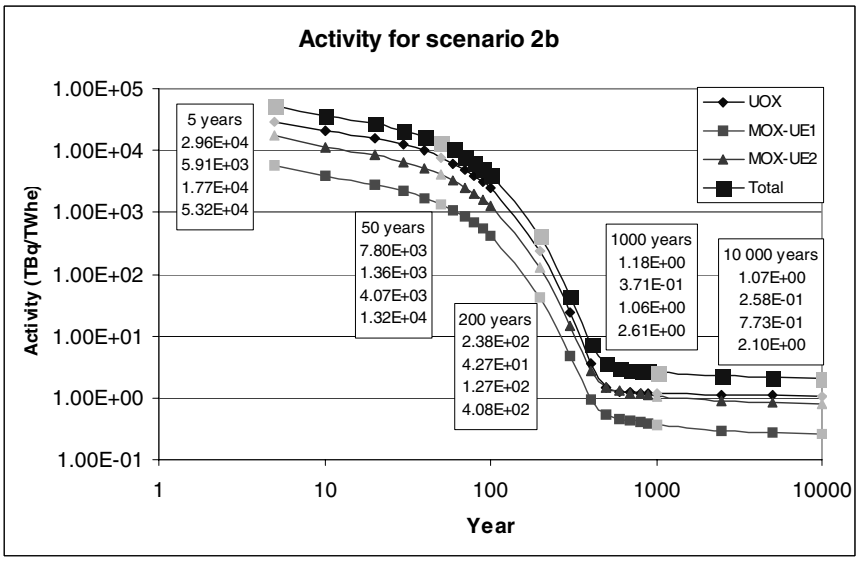


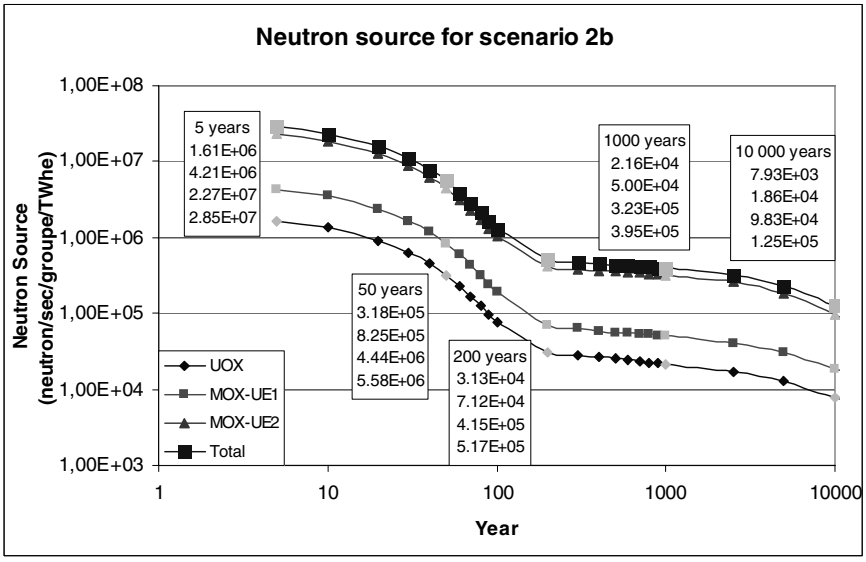
	PWR/UOX	PWR/MOX-EU 1	PWR/MOX-EU 2
ENRICHMENT			
Natural Uranium	0.71% ^{235}U	0.71% ^{235}U	0.71% ^{235}U
Depleted Uranium	0.25% ^{235}U	0.25% ^{235}U	0.25% ^{235}U
Enriched Uranium	4.90% ^{235}U	3.48% ^{235}U	5.85% ^{235}U
FUEL	UOX	MOX EU 1	MOX EU 2
Burn Up	60 GWd/tHM	60 GWd/tHM	60 GWd/tHM
Ageing time	2 years	2 years	2 years
Cooling time	5 years	5 years	5 years
Pu Content	–	10%	6.7%
Fissile Pu	–	42.8%	42.8%
Am Content	–	–	1.13%
REACTOR	PWR – EPR type	PWR – EPR type 100% MOX	PWR – EPR type 100% MOX
Thermal power	4 250 MWth	4 250 MWth	4 250 MWth
Electrical power	1 450 MWe	1 450 MWe	1 450 MWe
REPROCESSING	Advanced PUREX	Advanced PUREX	Advanced PUREX
Losses	0.1% U, Pu, Am 100% Np, Cm, FP	0.1% U, Pu, Am 100% Np, Cm, FP	0.1% U, Pu, Am 100% Np, Cm, FP

COMMENTS:

1. For FP mass composition of irradiated MOX-EU 1 and 2 fuel, MOX fuel has been used.
2. EPR MOX-EU 1 and MOX-EU 2 have an increased moderation ratio up to 2.5.

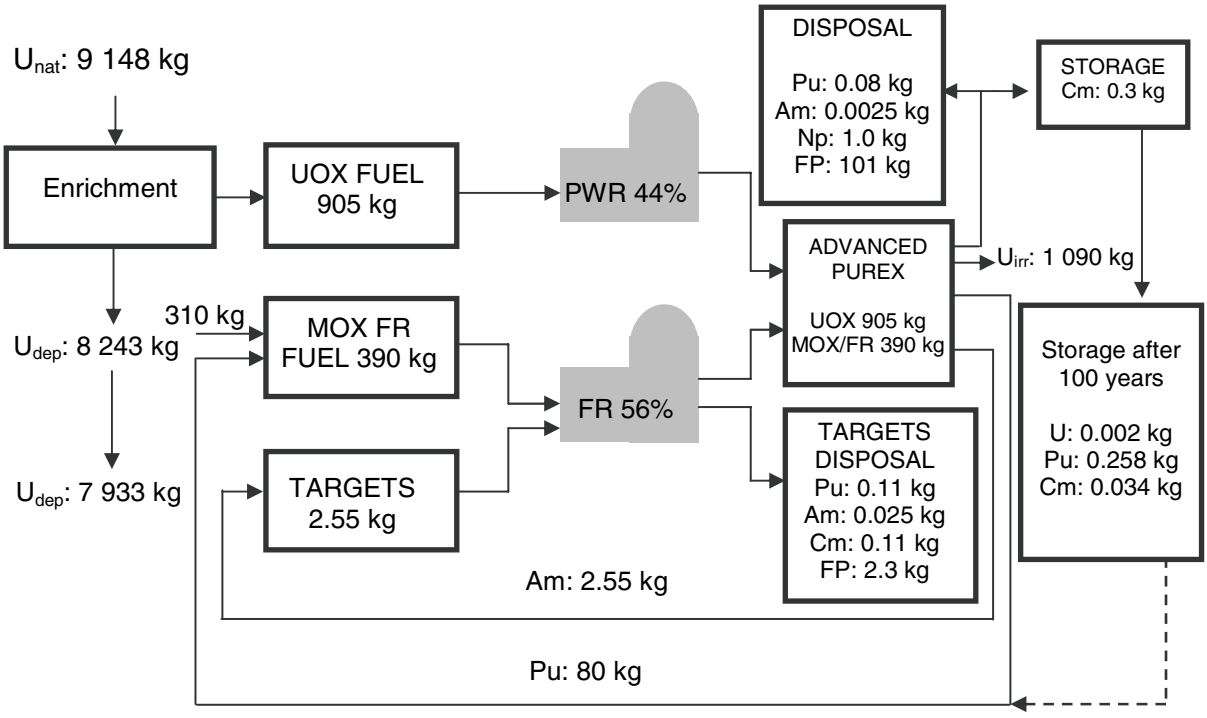

Power for scenario 2b
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1
10
100
1000
10000
Year
UOX
MOX-UE1
MOX-UE2
Total
5 years
2.80E+03
5.86E+02
1.75E+03
5.14E+03
50 years
5.85E+02
9.87E+01
2.95E+02
9.79E+02
200 years
1.78E+01
3.49E+00
1.04E+01
3.17E+01
1000 years
1.28E-01
1.22E-01
3.17E-01
5.68E-01
10 000 years
5.93E-02
2.95E-02
8.37E-02
1.73E-01


Activity for scenario 2b
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1
10
100
1000
10000
Year
UOX
MOX-UE1
MOX-UE2
Total
5 years
2.96E+04
5.91E+03
1.77E+04
5.32E+04
50 years
7.80E+03
1.36E+03
4.07E+03
1.32E+04
200 years
2.38E+02
4.27E+01
1.27E+02
4.08E+02
1000 years
1.18E+00
3.71E-01
1.06E+00
2.61E+00
10 000 years
1.07E+00
2.58E-01
7.73E-01
2.10E+00

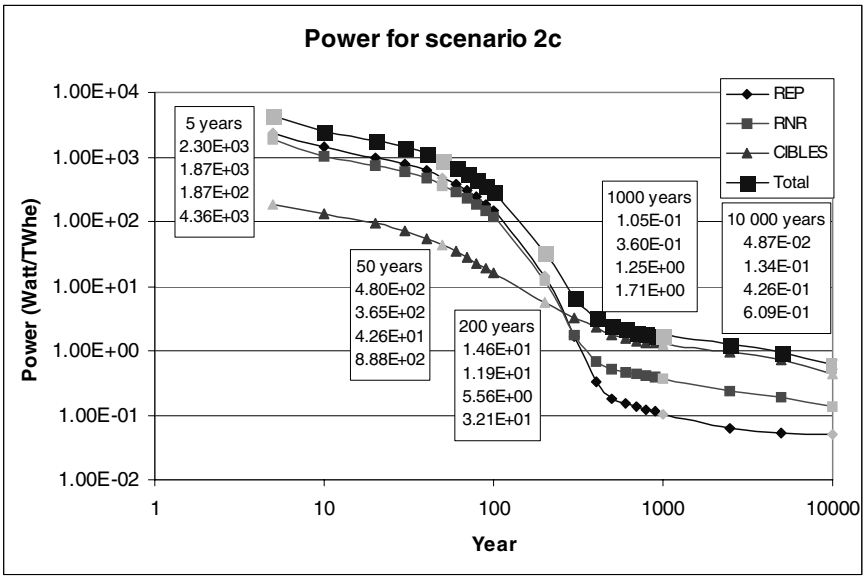

Neutron source for scenario 2b
Neutron Source
(neutron/sec/groupe/TWhe)
1,00E+08
1,00E+07
1,00E+06
1,00E+05
1,00E+04
1,00E+03
1
10
100
1000
10000
Year
UOX
MOX-UE1
MOX-UE2
Total
5 years
1.61E+06
4.21E+06
2.27E+07
2.85E+07
50 years
3.18E+05
8.25E+05
4.44E+06
5.58E+06
200 years
3.13E+04
7.12E+04
4.15E+05
5.17E+05
1000 years
2.16E+04
5.00E+04
3.23E+05
3.95E+05
10 000 years
7.93E+03
1.86E+04
9.83E+04
1.25E+05

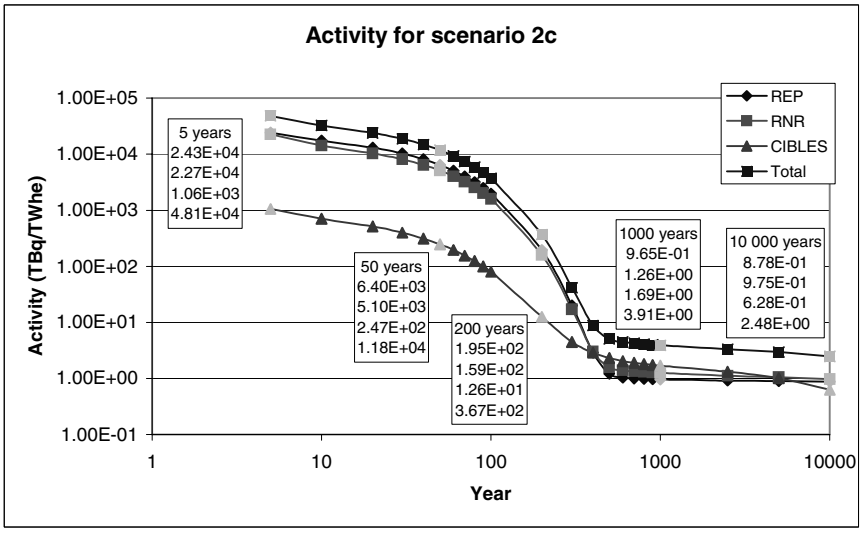
Scheme 2c
UOX and MOX reprocessed (Advanced PUREX)
Am and Cm Partitioning – Cm storage – Am once through transmutation

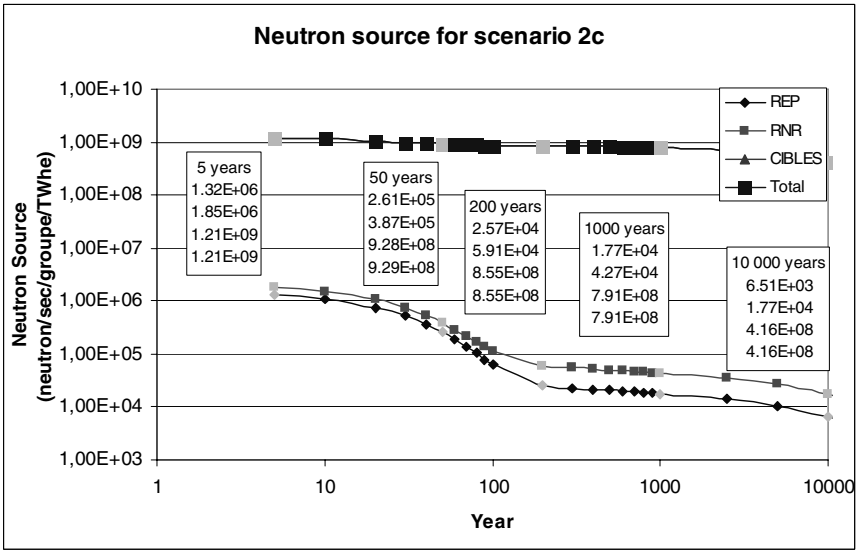
Fuel Characteristics and Mass Flows (kg/TWhe)



	PWR/UOX	FR/MOX	Am TARGET
ENRICHMENT Natural Uranium Depleted Uranium Enriched Uranium	 0.71% ^{235}U 0.25% ^{235}U 4.90% ^{235}U	 0.25% ^{235}U	
FUEL Burn Up Ageing time Cooling time Pu Content	UOX 60 GWd/tHM 2 years 5 years –	MOX 140 GWd/tHM 2 years 5 years 18.6%	Am TARGET
REACTOR Thermal power Electrical power	PWR – EPR type 4 250 MWth 1 450 MWe	FR – EFR type 3 600 MWth 1 450 MWe	FR – EFR type 3 600 MWth 1 450 MWe
REPROCESSING Losses	Advanced PUREX 0.1% U, Pu, Am 100% Np, Cm, FP	Advanced PUREX 0.1% U, Pu, Am 100% Np, Cm, FP	–

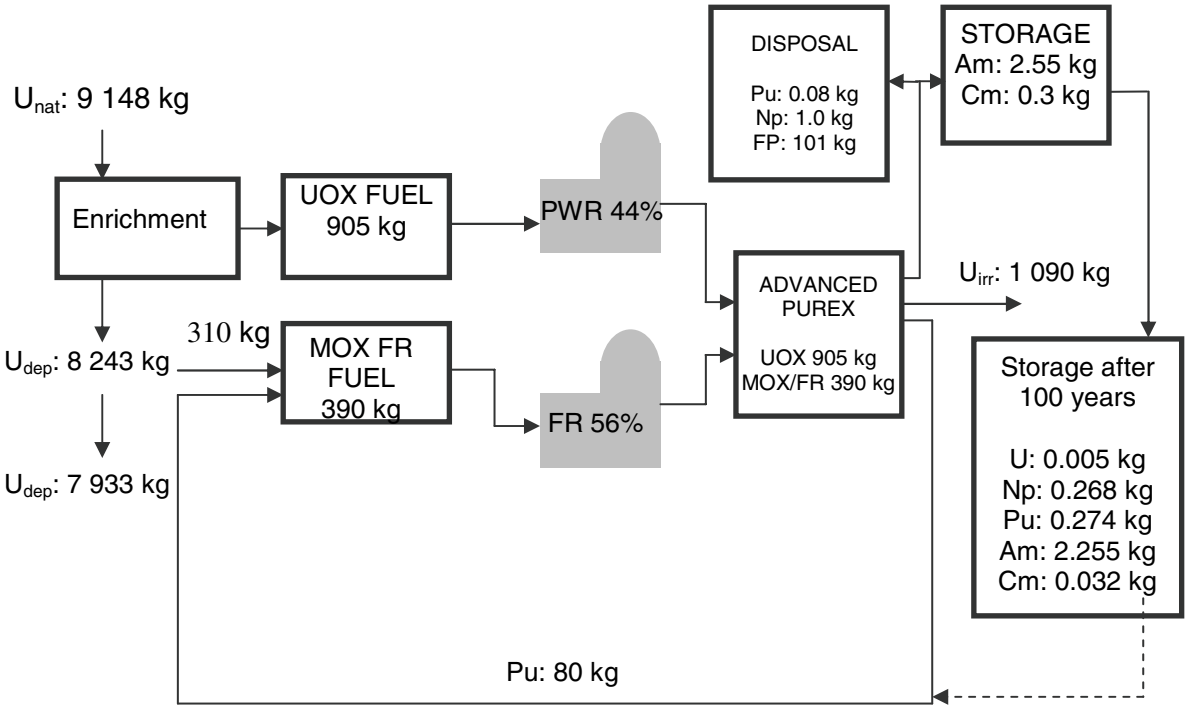

Power for scenario 2c
Power (Watt/TWhe)
Year
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1
10
100
1000
10000
REP
RNR
CIBLES
Total
5 years
2.30E+03
1.87E+03
1.87E+02
4.36E+03
50 years
4.80E+02
3.65E+02
4.26E+01
8.88E+02
200 years
1.46E+01
1.19E+01
5.56E+00
3.21E+01
1000 years
1.05E-01
3.60E-01
1.25E+00
1.71E+00
10 000 years
4.87E-02
1.34E-01
4.26E-01
6.09E-01


Activity for scenario 2c
Activity (TBq/TWhe)
Year
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1
10
100
1000
10000
REP
RNR
CIBLES
Total
5 years
2.43E+04
2.27E+04
1.06E+03
4.81E+04
50 years
6.40E+03
5.10E+03
2.47E+02
1.18E+04
200 years
1.95E+02
1.59E+02
1.26E+01
3.67E+02
1000 years
9.65E-01
1.26E+00
1.69E+00
3.91E+00
10 000 years
8.78E-01
9.75E-01
6.28E-01
2.48E+00

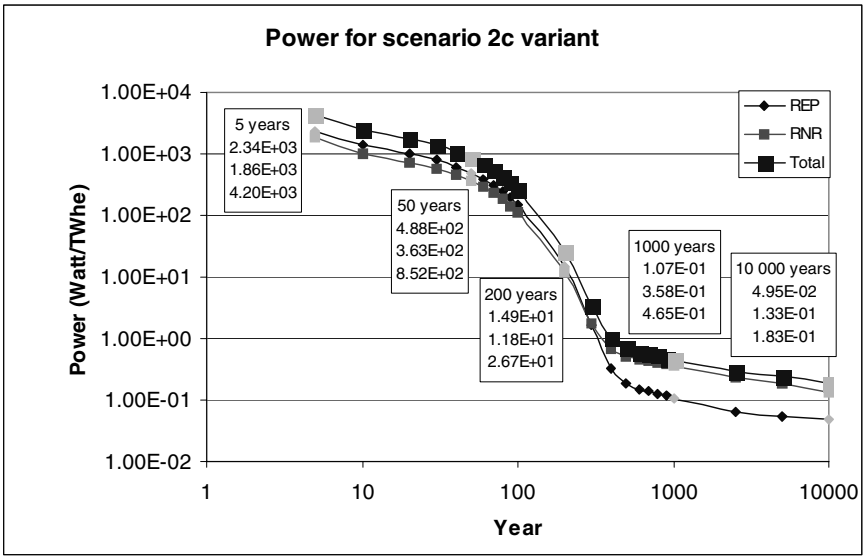

Neutron source for scenario 2c
Neutron Source
(neutron/sec/groupe/TWhe)
Year
1,00E+10
1,00E+09
1,00E+08
1,00E+07
1,00E+06
1,00E+05
1,00E+04
1,00E+03
1
10
100
1000
10000
REP
RNR
CIBLES
Total
5 years
1.32E+06
1.85E+06
1.21E+09
1.21E+09
50 years
2.61E+05
3.87E+05
9.28E+08
9.29E+08
200 years
2.57E+04
5.91E+04
8.55E+08
8.55E+08
1000 years
1.77E+04
4.27E+04
7.91E+08
7.91E+08
10 000 years
6.51E+03
1.77E+04
4.16E+08
4.16E+08

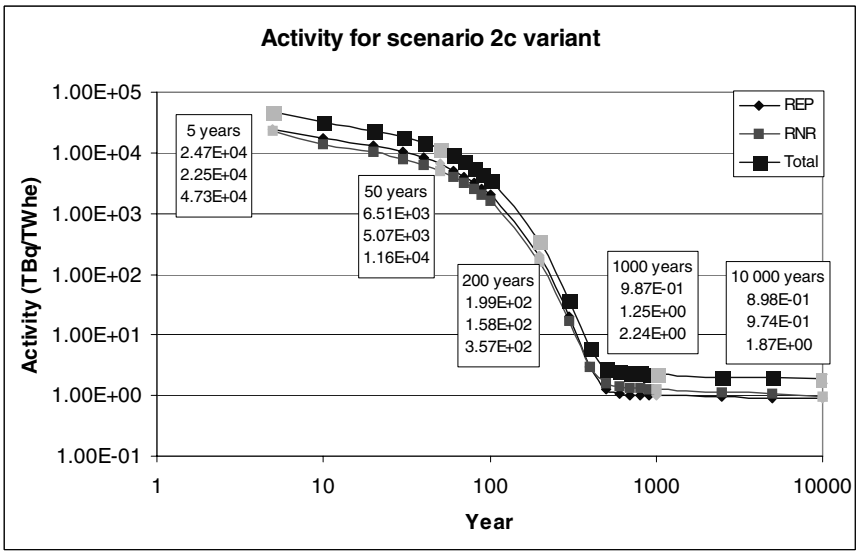
Scheme 2c Variant
UOX and MOX reprocessed (Advanced PUREX) – Am and Cm Partitioning – Am and Cm storage

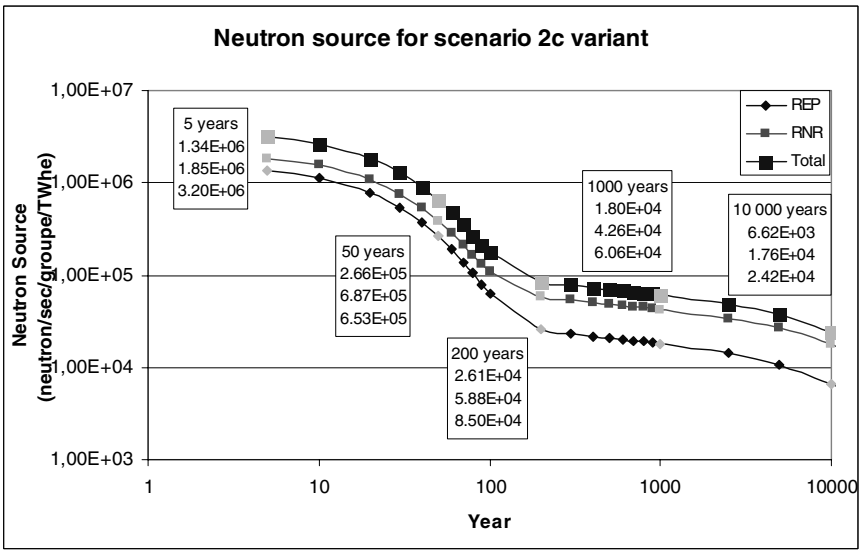
Fuel Characteristics and Mass Flows (kg/TWhe)



	PWR/UOX	FR/MOX
ENRICHMENT Natural Uranium Depleted Uranium Enriched Uranium	 0.71% ^{235}U 0.25% ^{235}U 4.90% ^{235}U	 0.25% ^{235}U
FUEL Burn Up Ageing time Cooling time Pu Content	UOX 60 GWd/tHM 2 years 5 years –	MOX 140 GWd/tHM 2 years 5 years 18.6%
REACTOR Thermal power Electrical power	PWR – EPR type 4 250 MWth 1 450 MWe	FR – EFR type 3 600 MWth 1 450 MWe
REPROCESSING Losses	Advanced PUREX 0.1% U, Pu, MA 100% FP	Advanced PUREX 0.1% U, Pu, MA 100% FP

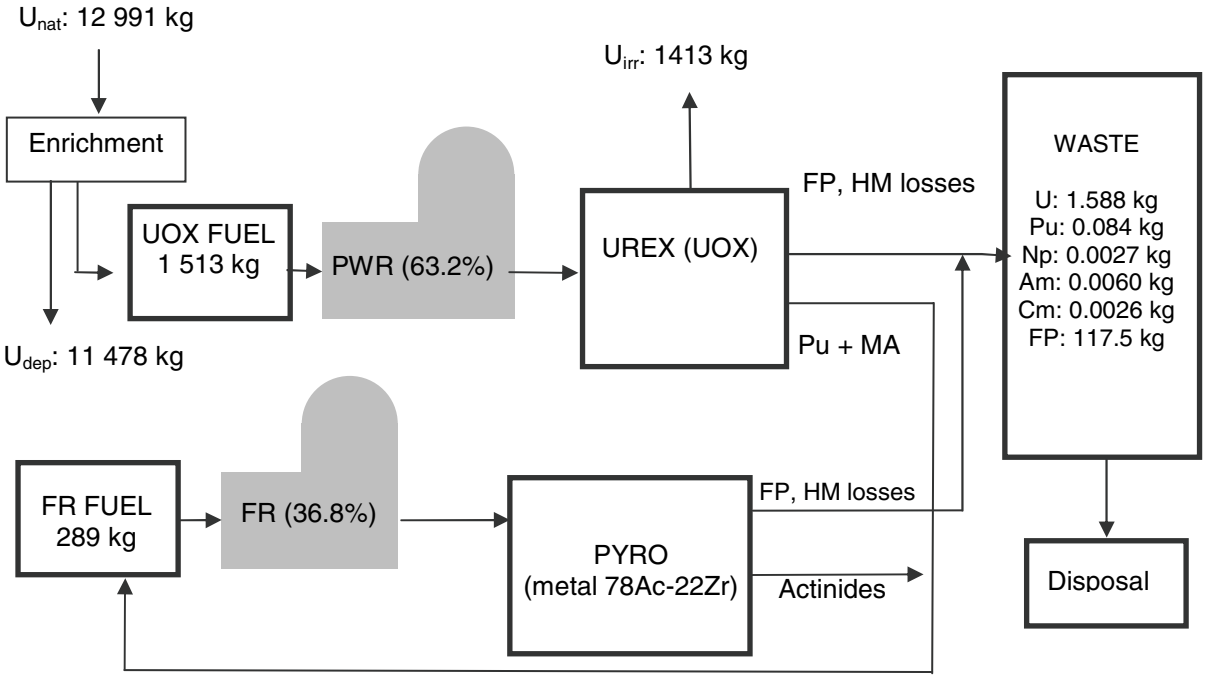

Power for scenario 2c variant
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1
10
100
1000
10000
Year
REP
RNR
Total
5 years
2.34E+03
1.86E+03
4.20E+03
50 years
4.88E+02
3.63E+02
8.52E+02
200 years
1.49E+01
1.18E+01
2.67E+01
1000 years
1.07E-01
3.58E-01
4.65E-01
10 000 years
4.95E-02
1.33E-01
1.83E-01


Activity for scenario 2c variant
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1
10
100
1000
10000
Year
REP
RNR
Total
5 years
2.47E+04
2.25E+04
4.73E+04
50 years
6.51E+03
5.07E+03
1.16E+04
200 years
1.99E+02
1.58E+02
3.57E+02
1000 years
9.87E-01
1.25E+00
2.24E+00
10 000 years
8.98E-01
9.74E-01
1.87E+00


Neutron source for scenario 2c variant
Neutron Source
(neutron/sec/groupe/TWhe)
1,00E+07
1,00E+06
1,00E+05
1,00E+04
1,00E+03
1
10
100
1000
10000
Year
REP
RNR
Total
5 years
1.34E+06
1.85E+06
3.20E+06
50 years
2.66E+05
6.87E+05
6.53E+05
200 years
2.61E+04
5.88E+04
8.50E+04
1000 years
1.80E+04
4.26E+04
6.06E+04
10 000 years
6.62E+03
1.76E+04
2.42E+04

Scheme 3a "TRU Burning in FR"
UOX reprocessed (UREX) Metal reprocessed (PYRO) – TRU partitioning and homogeneous transmutation

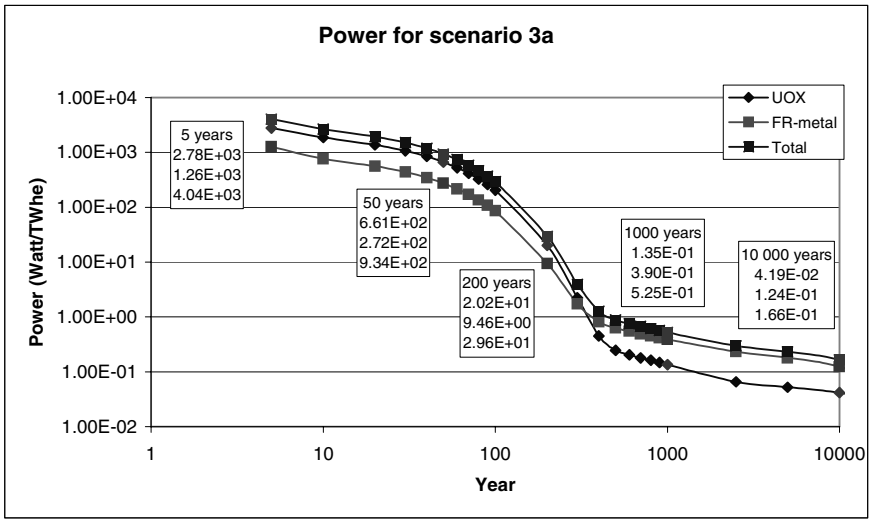
Fuel Characteristics and Mass Flows (kg/TWhe)

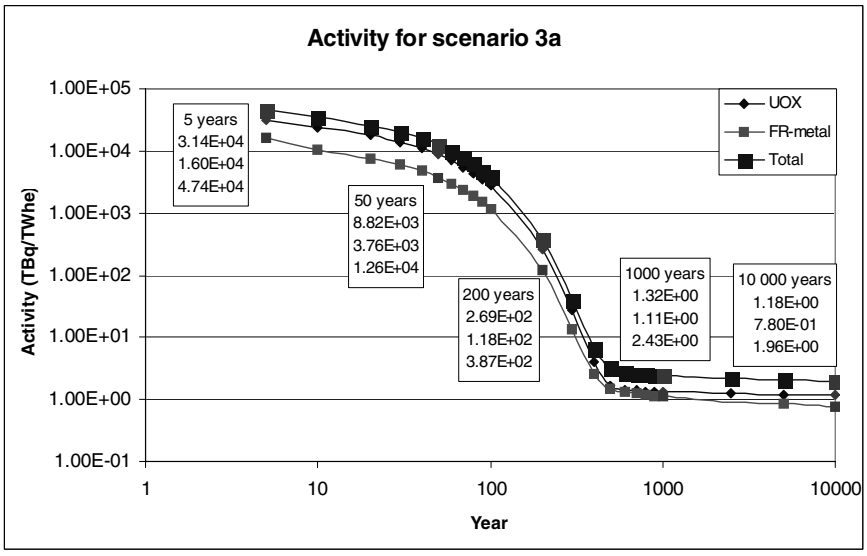


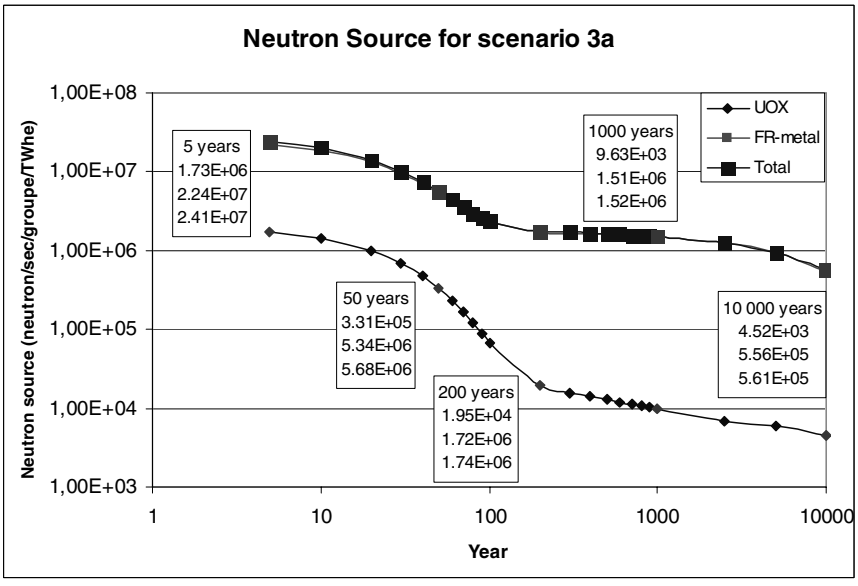
	PWR/UOX	FR/METAL
ENRICHMENT		
Natural Uranium	0.71% ^{235}U	
Depleted Uranium	0.25% ^{235}U	0.25% ^{235}U
Enriched Uranium	4.20% ^{235}U	
FUEL	UOX	FR METAL AcZr
Burn Up	50 GWd/tHM	140 GWd/tHM
Ageing time	2 years	1 year
Cooling time	4 years	2 years
Pu Content	–	29.1%
MA Content	–	3.9%
REACTOR	PWR	FR
Thermal power	4 250 MWth	1 575 MWth
Electrical power	1 450 MWe	600 MWe
REPROCESSING	UREX	PYRO
Power	3.5 W/kgHM	46 W/kgHM
Source	1. E6 n/s-kgHM	86. E6 n/s-kgHM
Losses	0.1% U, Pu, MA 100% FP	0.1% U, Pu, MA 100% FP

COMMENTS:

1. For FP mass composition of the irradiated metallic fuel, the values calculated by PSI have been used.


Power for scenario 3a
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1
10
100
1000
10000
Year
UOX
FR-metal
Total
5 years
2.78E+03
1.26E+03
4.04E+03
50 years
6.61E+02
2.72E+02
9.34E+02
200 years
2.02E+01
9.46E+00
2.96E+01
1000 years
1.35E-01
3.90E-01
5.25E-01
10 000 years
4.19E-02
1.24E-01
1.66E-01

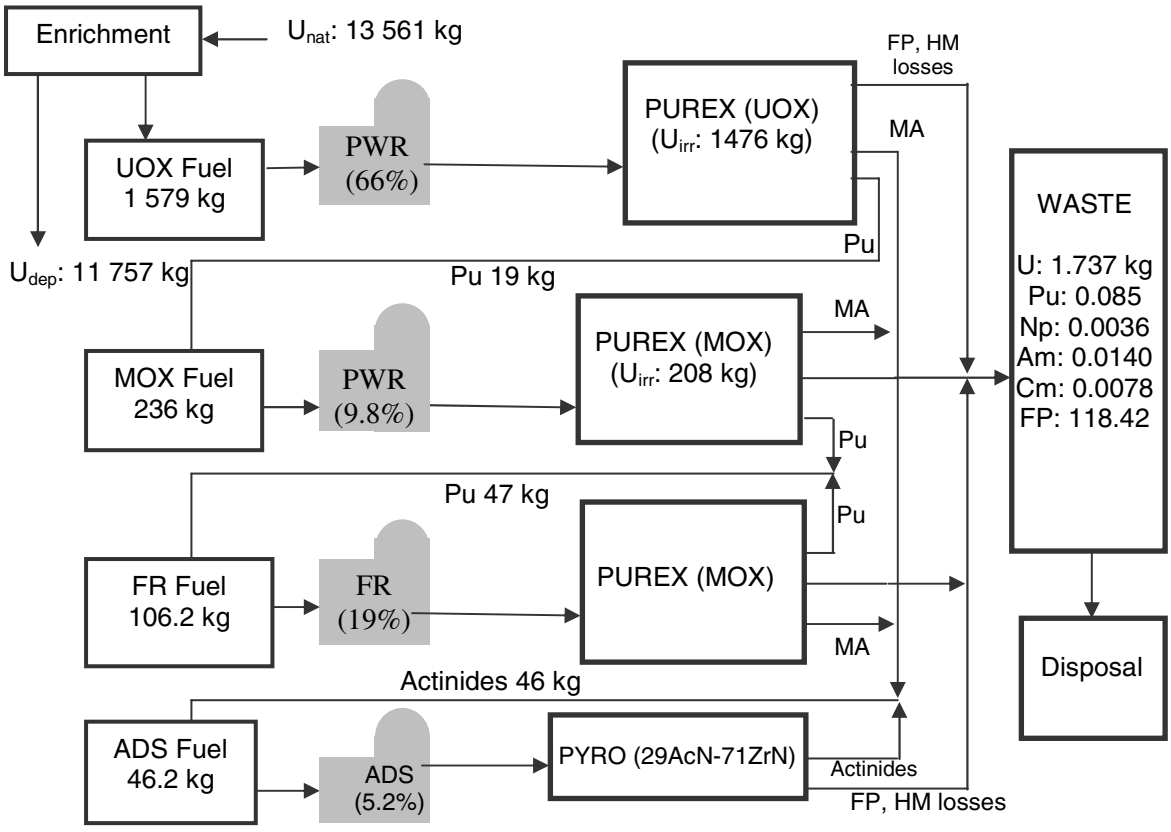

Activity for scenario 3a
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1
10
100
1000
10000
Year
UOX
FR-metal
Total
5 years
3.14E+04
1.60E+04
4.74E+04
50 years
8.82E+03
3.76E+03
1.26E+04
200 years
2.69E+02
1.18E+02
3.87E+02
1000 years
1.32E+00
1.11E+00
2.43E+00
10 000 years
1.18E+00
7.80E-01
1.96E+00


Neutron Source for scenario 3a
Neutron source (neutron/sec/groupe/TWhe)
1,00E+08
1,00E+07
1,00E+06
1,00E+05
1,00E+04
1,00E+03
1
10
100
1000
10000
Year
UOX
FR-metal
Total
5 years
1.73E+06
2.24E+07
2.41E+07
1000 years
9.63E+03
1.51E+06
1.52E+06
50 years
3.31E+05
5.34E+06
5.68E+06
200 years
1.95E+04
1.72E+06
1.74E+06
10 000 years
4.52E+03
5.56E+05
5.61E+05

Scheme 3b "Double Strata"

UOX, MOX and FR(MOX) reprocessed (PUREX) – Pu once recycling in PWR (MOX) – Pu multi recycling in FR(MOX) – Pu and MA partitioning and homogeneous transmutation – ADS (nitride) Fuel reprocessed (PYRO)

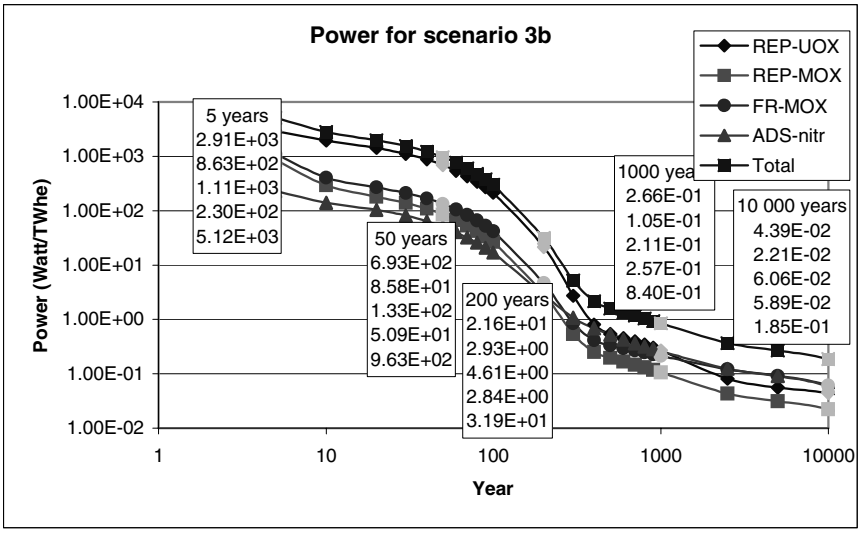
Fuel Characteristics and Mass Flows (kg/TWhe)

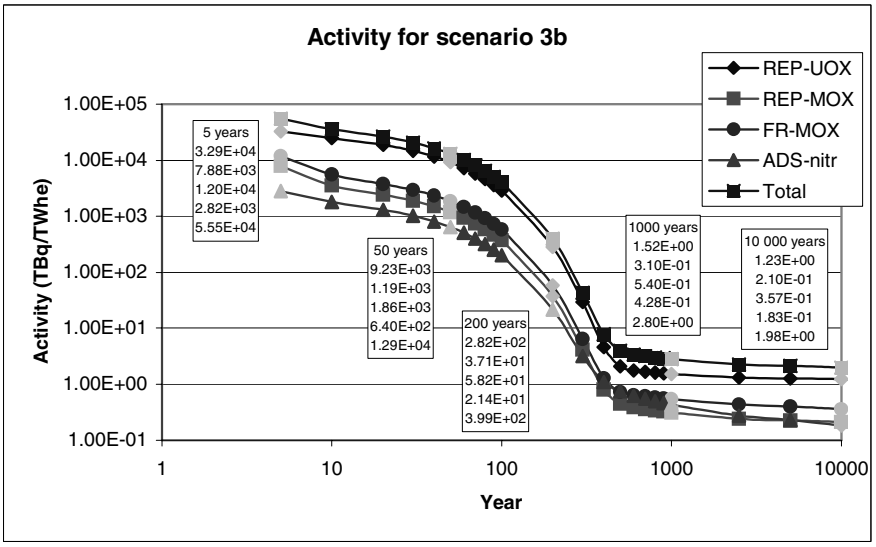


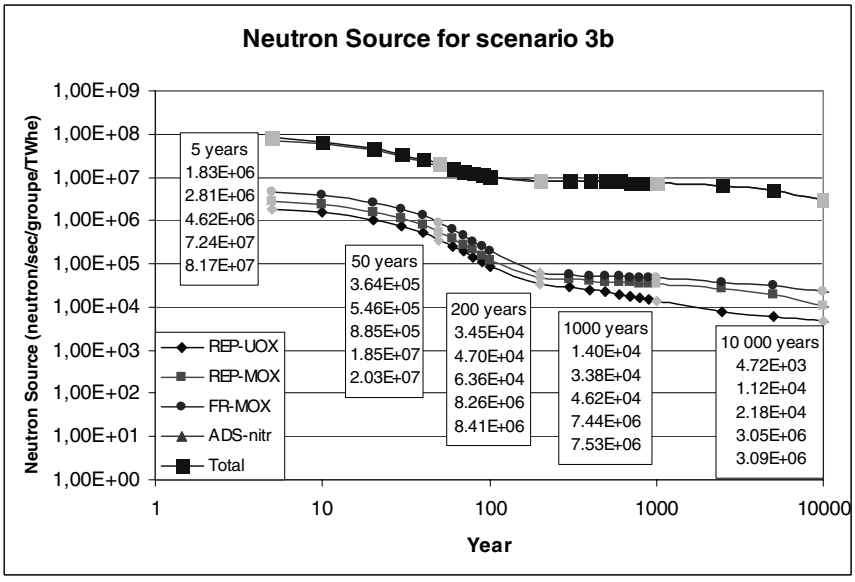
	PWR/UOX	PWR/MOX	FR/MOX	ADS Nitride
ENRICHMENT				
Natural Uranium	0.71% ^{235}U			
Depleted Uranium	0.25% ^{235}U	0.25% ^{235}U	0.25% ^{235}U	
Enriched Uranium	4.20% ^{235}U			
FUEL	UOX	MOX	FR MOX	ADS Nitride
Burn Up	50 GWd/tHM	50 GWd/tHM	185 GWd/tHM	150 GWd/tHM
Ageing time	2 years	2 years	2 years	1 years
Cooling time	4 years	5 years	5 years	2 years
Pu Content	–	8%	44.4%	40.5%
MA Content				55%
REACTOR	PWR N4 type	PWR N4 type	FR EFR Type	ADS JAERI Type
Thermal power	4 250 MWth	4 250 MWth	3 600 MWth	Lead Bismuth cooled
Electrical power	1 450 MWe	1 450 MWe	1 450 MWe	system
REPROCESSING	Advanced PUREX	Advanced PUREX	Advanced PUREX	PYRO
	3.5 W/kgHM	6.3 W/kgHM	21.8 W/kgHM	455 W/kgHM
	1. E6 n/s-kgHM	11. E6 n/s-kgHM	40 E6 n/s-kgHM	1 800 E6 n/s-kgHM
	0.1% U, Pu, AM	0.1% U, Pu, AM	0.1% U, Pu, AM	0.1% U, Pu, AM
Losses	100% FP	100% FP	100% FP	100% FP

COMMENTS:

1. For FP mass composition of the irradiated ADS fuel, PSI calculated values have been used.
2. For mass composition of the irradiated PWR-MOX and FR-MOX fuels, a "5 years cooling time" versus a "7 years cooling time" value has been recalculated.
3. For the FR, a metal fuel reprocessed by pyro could be considered as an alternative.

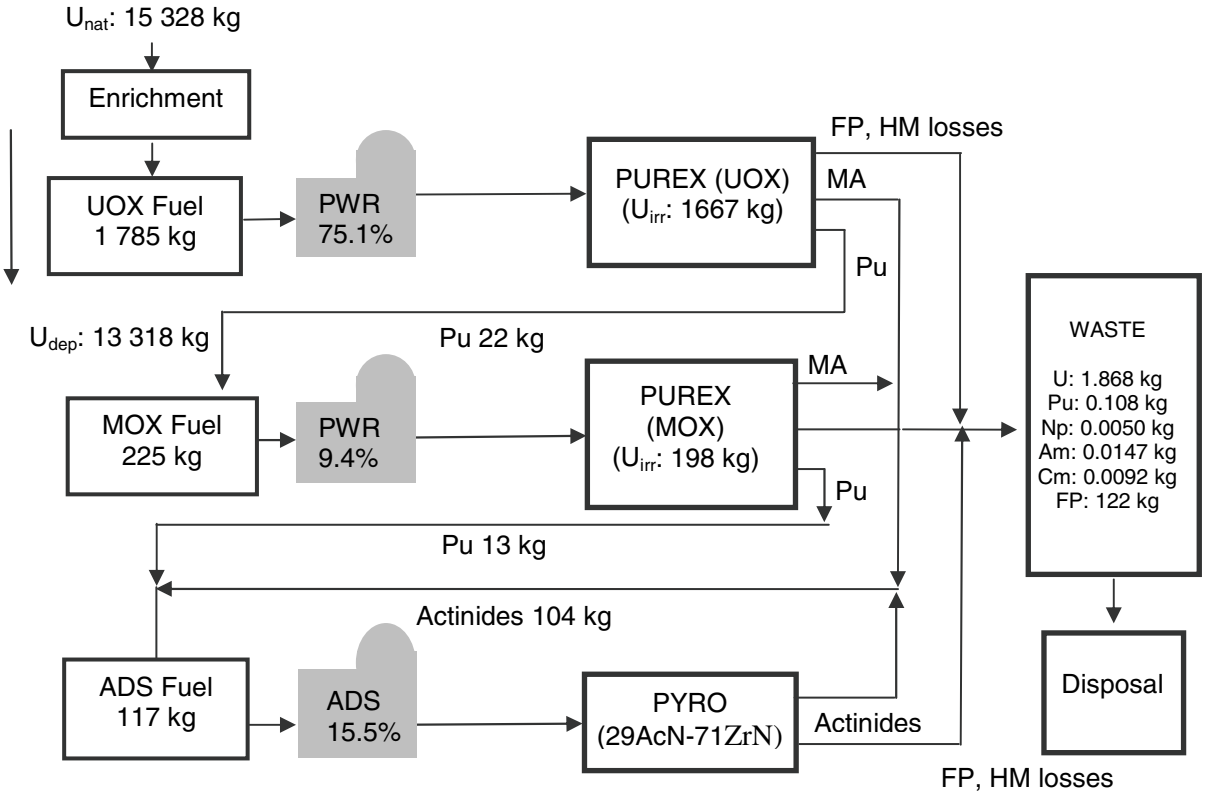

Power for scenario 3b
REP-UOX
REP-MOX
FR-MOX
ADS-nitr
Total
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1
10
100
1000
10000
Year
5 years
2.91E+03
8.63E+02
1.11E+03
2.30E+02
5.12E+03
50 years
6.93E+02
8.58E+01
1.33E+02
5.09E+01
9.63E+02
200 years
2.16E+01
2.93E+00
4.61E+00
2.84E+00
3.19E+01
1000 yea
2.66E-01
1.05E-01
2.11E-01
2.57E-01
8.40E-01
10 000 years
4.39E-02
2.21E-02
6.06E-02
5.89E-02
1.85E-01


Activity for scenario 3b
REP-UOX
REP-MOX
FR-MOX
ADS-nitr
Total
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1
10
100
1000
10000
Year
5 years
3.29E+04
7.88E+03
1.20E+04
2.82E+03
5.55E+04
50 years
9.23E+03
1.19E+03
1.86E+03
6.40E+02
1.29E+04
200 years
2.82E+02
3.71E+01
5.82E+01
2.14E+01
3.99E+02
1000 years
1.52E+00
3.10E-01
5.40E-01
4.28E-01
2.80E+00
10 000 years
1.23E+00
2.10E-01
3.57E-01
1.83E-01
1.98E+00


Neutron Source for scenario 3b
Neutron Source (neutron/sec/groupe/TWhe)
1,00E+09
1,00E+08
1,00E+07
1,00E+06
1,00E+05
1,00E+04
1,00E+03
1,00E+02
1,00E+01
1,00E+00
1
10
100
1000
10000
Year
REP-UOX
REP-MOX
FR-MOX
ADS-nitr
Total
5 years
1.83E+06
2.81E+06
4.62E+06
7.24E+07
8.17E+07
50 years
3.64E+05
5.46E+05
8.85E+05
1.85E+07
2.03E+07
200 years
3.45E+04
4.70E+04
6.36E+04
8.26E+06
8.41E+06
1000 years
1.40E+04
3.38E+04
4.62E+04
7.44E+06
7.53E+06
10 000 years
4.72E+03
1.12E+04
2.18E+04
3.05E+06
3.09E+06

Scheme 3b Variant "Double Strata"
UOX and MOX reprocessed (PUREX) – Pu once recycling in PWR(MOX) – TRU partitioning and homogeneous transmutation – ADS fuel reprocessed (PYRO)

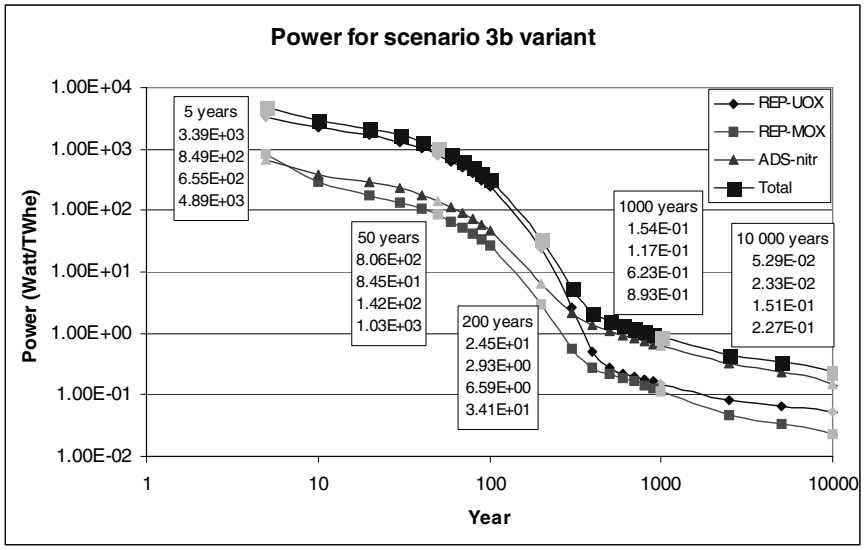
Fuel Characteristics and Mass Flows (kg/TWhe)

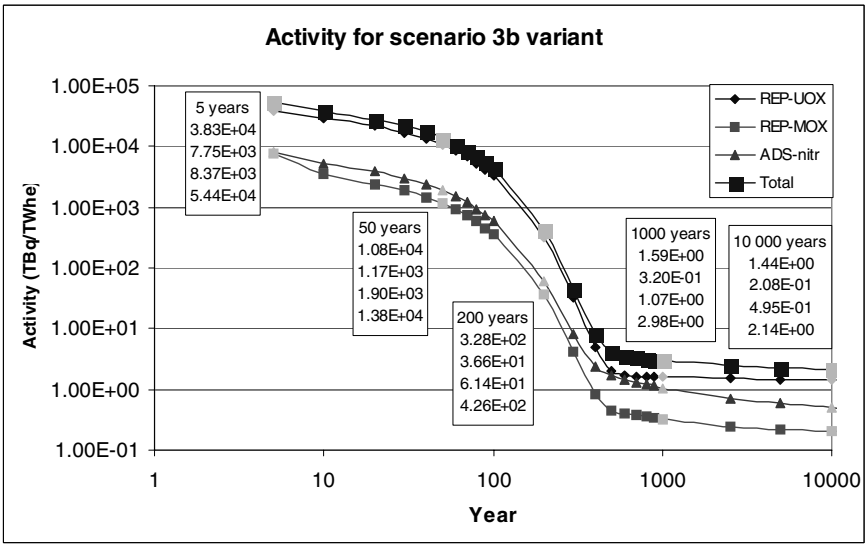


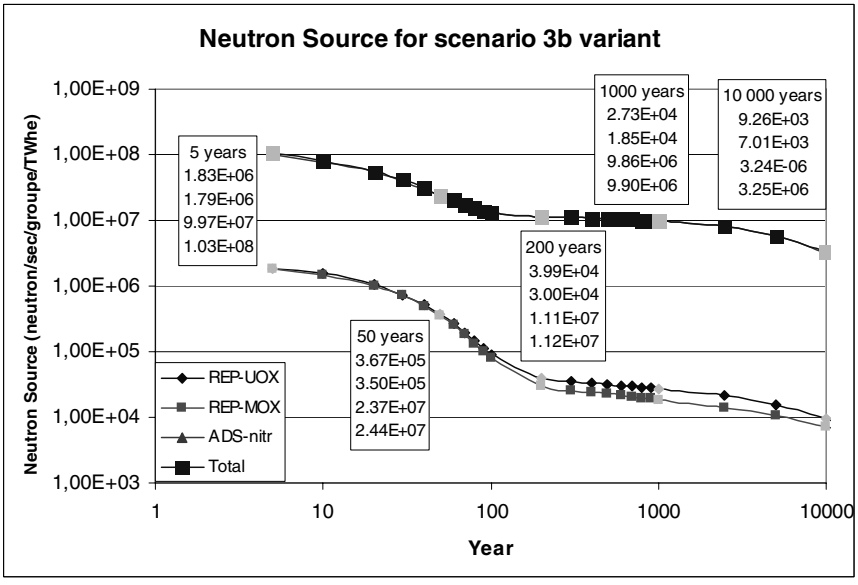
	PWR/UOX	PWR/MOX	ADS/Nitride
ENRICHMENT Natural Uranium Depleted Uranium Enriched Uranium	 0.71% ^{235}U 0.25% ^{235}U 4.20% ^{235}U	 0.25% ^{235}U	
FUEL Burn Up Ageing time Cooling time Pu Content TRU Content	UOX 50 GWd/tHM 2 years 4 years –	MOX 50 GWd/tHM 2 years 5 years 10%	ADS Nitride 150 GWd/tHM 1 years 2 years 42.3%
REACTOR Thermal power Electrical power	PWR French N4 4 250 MWth 1 450 MWe	PWR French N4 4 250 MWth 1 450 MWe	ADS JAERI Type Lead Bismuth cooled system
REPROCESSING Losses	Extended PUREX 0.1% U, Pu, MA 100% FP	Extended PUREX 0.1% U, Pu, MA 100% FP	PYRO 0.1%U, Pu, MA 100% FP

COMMENTS:

1. For FP mass composition of the irradiated ADS fuel, PSI calculated values have been used.
2. For mass composition of the irradiated PWR-MOX and FR-MOX fuels, a "5 years cooling time" versus a "7 years cooling time" value has been recalculated.

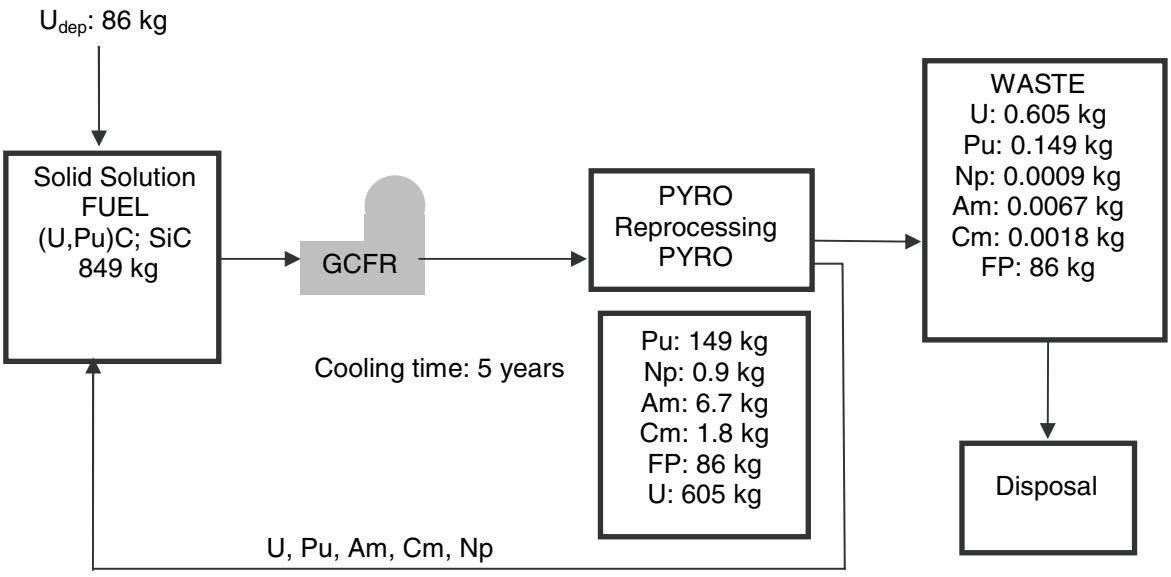

Power for scenario 3b variant
Power (Watt/TWhe)
Year
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1
10
100
1000
10000
REP-UOX
REP-MOX
ADS-nitr
Total
5 years
3.39E+03
8.49E+02
6.55E+02
4.89E+03
50 years
8.06E+02
8.45E+01
1.42E+02
1.03E+03
200 years
2.45E+01
2.93E+00
6.59E+00
3.41E+01
1000 years
1.54E-01
1.17E-01
6.23E-01
8.93E-01
10 000 years
5.29E-02
2.33E-02
1.51E-01
2.27E-01


Activity for scenario 3b variant
Activity (TBq/TWhe)
Year
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1
10
100
1000
10000
REP-UOX
REP-MOX
ADS-nitr
Total
5 years
3.83E+04
7.75E+03
8.37E+03
5.44E+04
50 years
1.08E+04
1.17E+03
1.90E+03
1.38E+04
200 years
3.28E+02
3.66E+01
6.14E+01
4.26E+02
1000 years
1.59E+00
3.20E-01
1.07E+00
2.98E+00
10 000 years
1.44E+00
2.08E-01
4.95E-01
2.14E+00


Neutron Source for scenario 3b variant
Neutron Source (neutron/sec/groupe/TWhe)
Year
1,00E+09
1,00E+08
1,00E+07
1,00E+06
1,00E+05
1,00E+04
1,00E+03
1
10
100
1000
10000
REP-UOX
REP-MOX
ADS-nitr
Total
5 years
1.83E+06
1.79E+06
9.97E+07
1.03E+08
50 years
3.67E+05
3.50E+05
2.37E+07
2.44E+07
200 years
3.99E+04
3.00E+04
1.11E+07
1.12E+07
1000 years
2.73E+04
1.85E+04
9.86E+06
9.90E+06
10 000 years
9.26E+03
7.01E+03
3.24E-06
3.25E+06

Scheme 3c Variant 1
Carbide fuel reprocessed (PYRO) – TRU partitioning and homogeneous transmutation

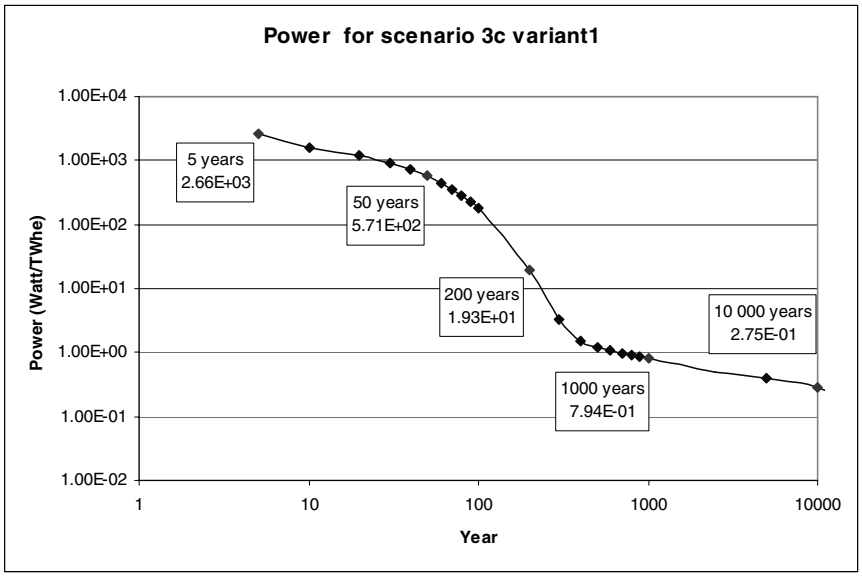
Fuel Characteristics and Mass Flows (kg/TWhe)

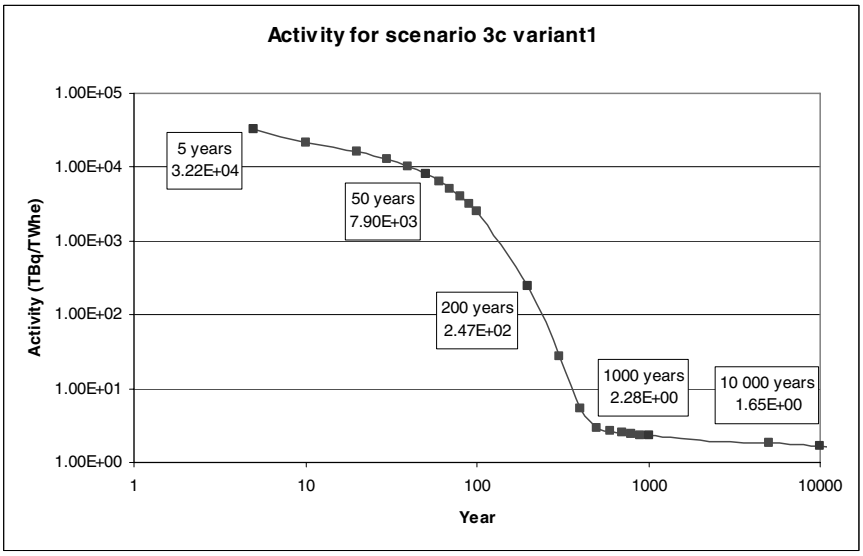


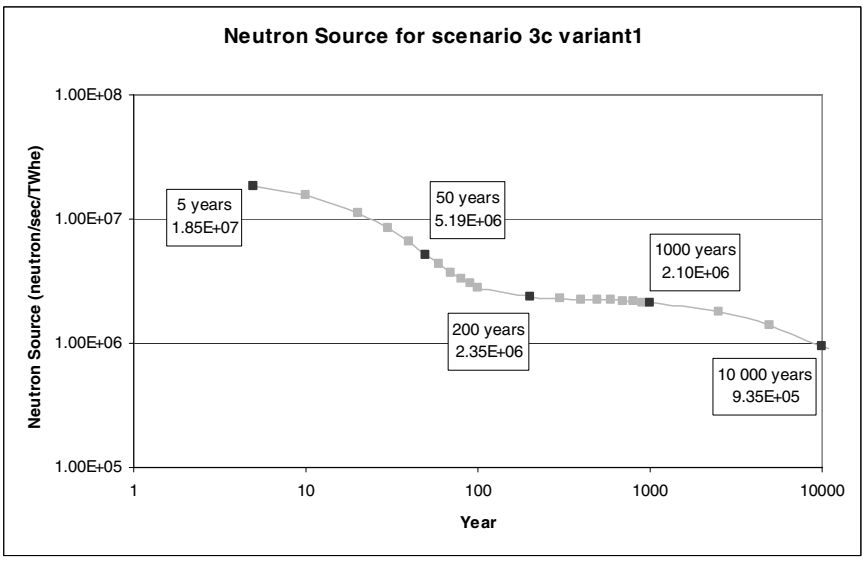
	GCFR/Carbide
ENRICHMENT Natural Uranium Depleted Uranium Enriched Uranium	 0.25% ^{235}U
FUEL Burn Up Pu Content MA Content Ageing time Cooling time	Solid Solution: (U, Pu)C; SiC 100 GWd/tHM 17.6% 1.1% 2 years 5 years
REACTOR Thermal power Electrical power	GCFR (He) 2 400 MWth 1 158 MWe
REPROCESSING Losses	PYRO 0.1% U, Pu, MA 100% FP

COMMENTS:

1. For FP mass composition of the irradiated CERCER fuel, MOX EFR fuel with a five year cooling time has been used (same spectra) with a mass normalisation.

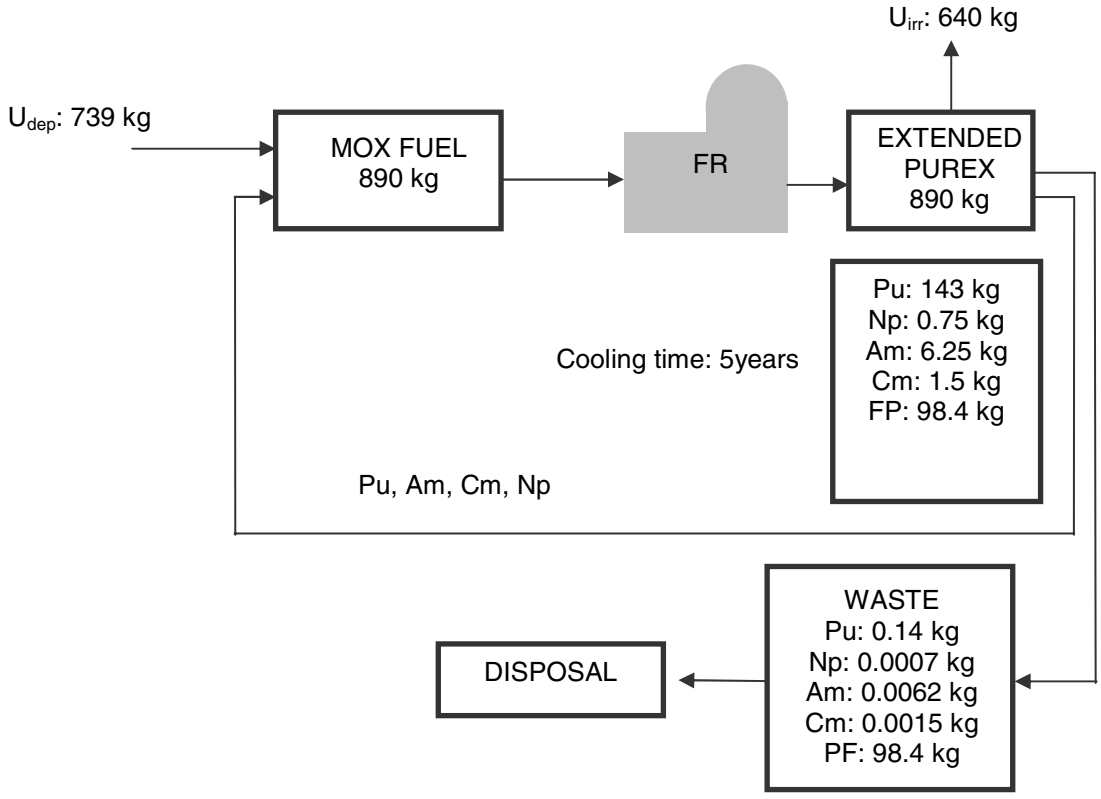

Power for scenario 3c variant1
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1
10
100
1000
10000
Year
5 years
2.66E+03
50 years
5.71E+02
200 years
1.93E+01
1000 years
7.94E-01
10 000 years
2.75E-01


Activity for scenario 3c variant1
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
100
1000
10000
Year
5 years
3.22E+04
50 years
7.90E+03
200 years
2.47E+02
1000 years
2.28E+00
10 000 years
1.65E+00

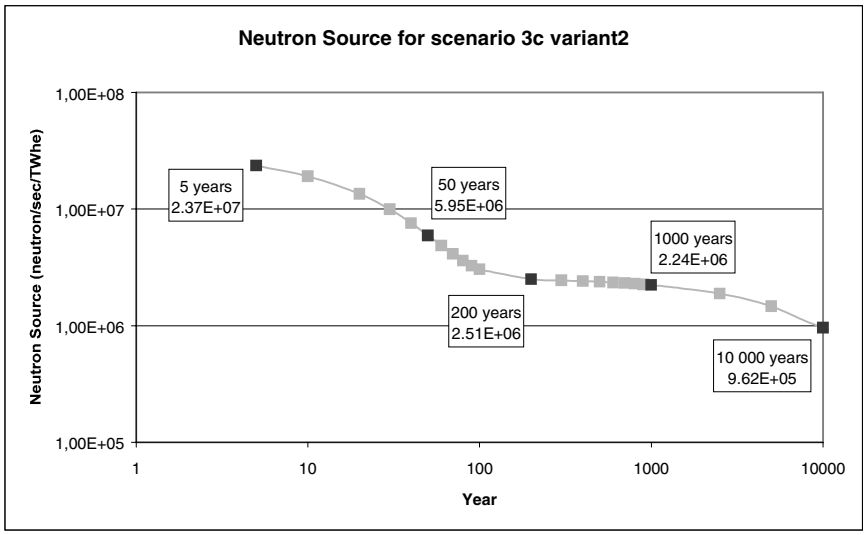

Neutron Source for scenario 3c variant1
Neutron Source (neutron/sec/TWhe)
1.00E+08
1.00E+07
1.00E+06
1.00E+05
1
10
100
1000
10000
Year
5 years
1.85E+07
50 years
5.19E+06
1000 years
2.10E+06
200 years
2.35E+06
10 000 years
9.35E+05

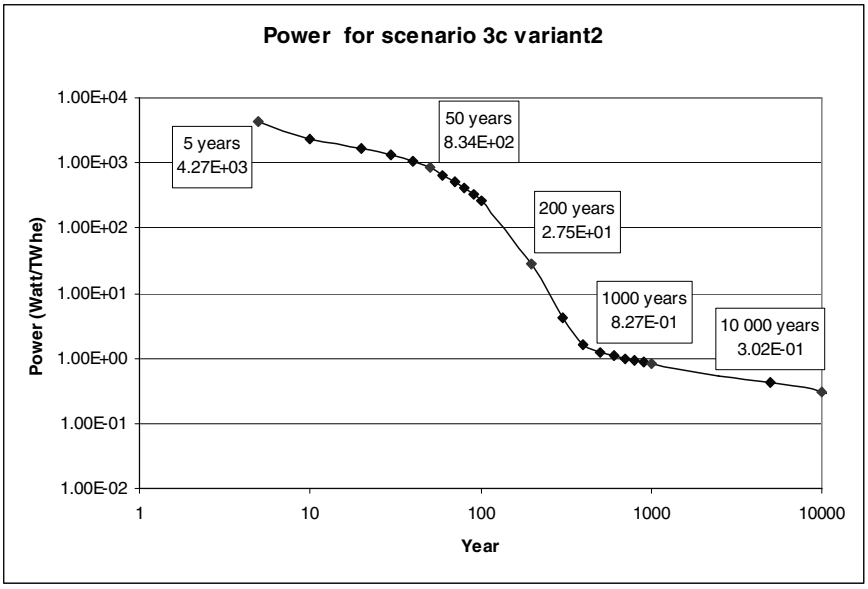
Scheme 3c Variant 2
MOX reprocessed (Extended PUREX) – TRU partitioning and homogeneous transmutation

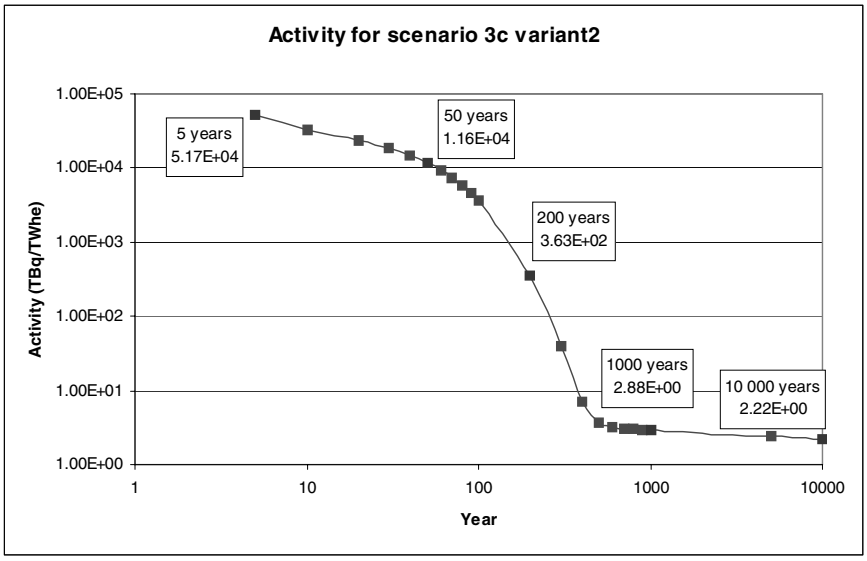
Fuel Characteristics and Mass Flows (kg/TWhe)



	FR/MOX
ENRICHMENT Natural Uranium Depleted Uranium Enriched Uranium	 0.25% ^{235}U
FUEL Burn Up Ageing time Cooling time Pu Content MA Content	MOX 140 GWd/tHM 2 years 5 years 16% 1%
REACTOR Thermal power Electrical power	FR – EFR type 3 600 MWth 1 450 MWe
REPROCESSING Losses	ADVANCED PUREX 0.1% U, Pu, MA 100% FP


Neutron Source for scenario 3c variant2
Neutron Source (neutron/sec/TWhe)
1,00E+08
1,00E+07
1,00E+06
1,00E+05
1
10
100
1000
10000
Year
5 years
2.37E+07
50 years
5.95E+06
1000 years
2.24E+06
200 years
2.51E+06
10 000 years
9.62E+05


Power for scenario 3c variant2
Power (Watt/TWhe)
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1
10
100
1000
10000
Year
5 years
4.27E+03
50 years
8.34E+02
200 years
2.75E+01
1000 years
8.27E-01
10 000 years
3.02E-01


Activity for scenario 3c variant2
Activity (TBq/TWhe)
1.00E+05
1.00E+04
1.00E+03
1.00E+02
1.00E+01
1.00E+00
1
10
100
1000
10000
Year
5 years
5.17E+04
50 years
1.16E+04
200 years
3.63E+02
1000 years
2.88E+00
10 000 years
2.22E+00

Appendix D

OREOX PROCESS FOR THE FABRICATION OF DUPIC FUEL AND THE RELATED WASTE TYPES

The DUPIC (Direct Use of Spent PWR Fuel in CANDU Reactors) fuel cycle is based upon dry thermal and mechanical processes to directly fabricate CANDU (Canada Deuterium Uranium) fuel from spent PWR fuel material without separating the fissile material and fission products. This concept was proposed and termed the "DUPIC" fuel cycle in a joint development programme involving KAERI (Korea Atomic Energy Research Institute), AECL (Atomic Energy Canada Limited) and the US Department of State in 1991 [1]. Since then, KAERI, AECL and the United States, with the participation of IAEA, have been engaged in a practical exercise to verify the concept.

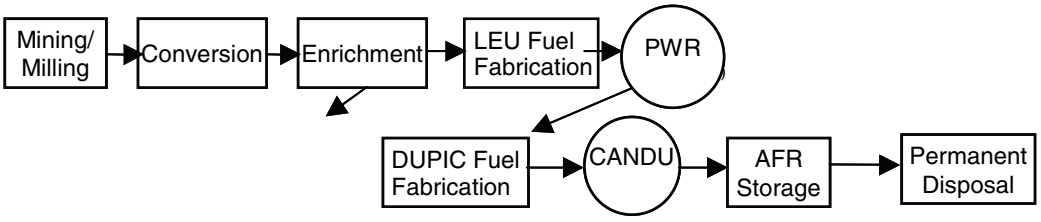
The fissile material that typically remains in spent PWR fuel exceeds the natural uranium fissile content of 0.711 wt%. The neutron economy of a CANDU reactor is sufficient to allow DUPIC fuel to be used in a CANDU reactor which is originally designed for natural uranium fuel, even though the neutron-absorbing fission products remain as impurities in the DUPIC fuel.

The DUPIC fuel cycle offers several benefits to countries operating both PWR and CANDU reactors. The benefits include the additional energy extracted from the fuel in a CANDU reactor, efficient natural uranium utilisation, and a significant reduction in spent fuel arisings through a proliferation resistant process of reusing spent PWR fuel [2].

The general compatibility and safeguardability of the DUPIC fuel cycle were confirmed in a broad feasibility study, and the OREOX (Oxidation and Reduction of Oxide fuel) process was selected as the most promising fabrication method for DUPIC fuel [3]. This OREOX process is the heart of the DUPIC fuel cycle and the key feature in the DUPIC fuel fabrication. The latter differs from the conventional CANDU fuel fabrication process in that it has to be done in a shielded facility because the fuel material remains highly radioactive throughout the process [4]. In the following, the feasibility of the concept was confirmed in an experimental verification programme. To verify the in-reactor performance of the DUPIC fuel, a comprehensive fuel development programme was carried out, including irradiation tests in the NRU reactor in Canada and in HANARO in the Republic of Korea [5].

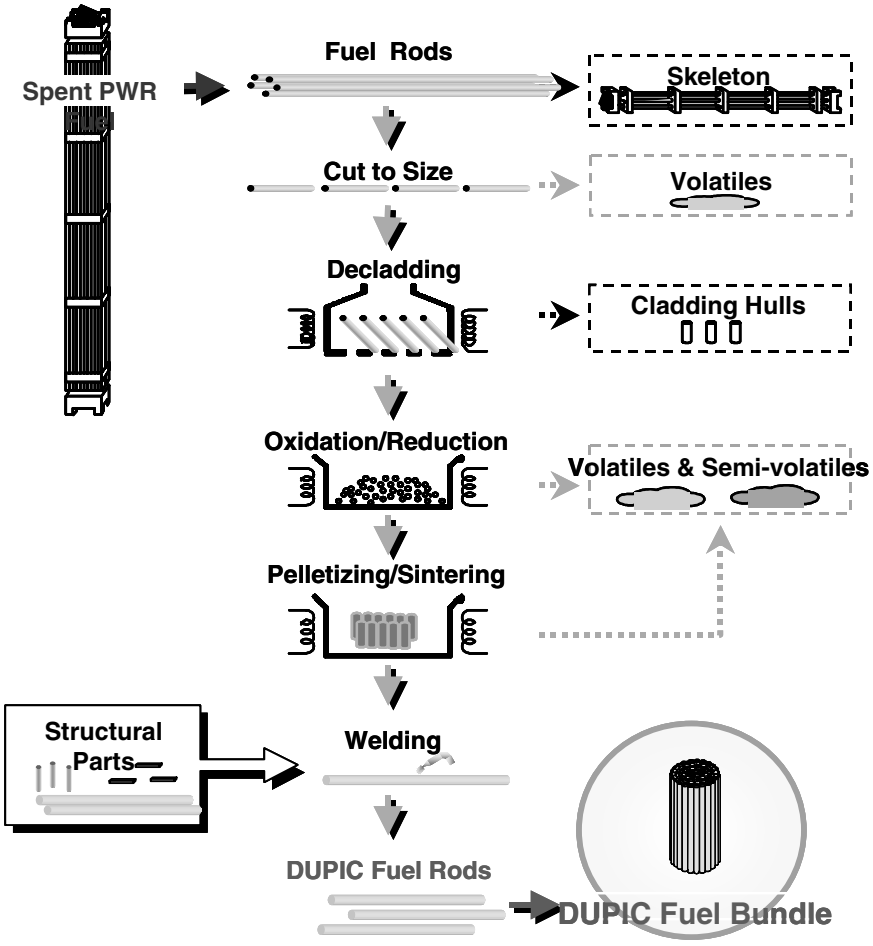
In the DUPIC fuel cycle, schematically shown in Figure D.1 [6], spent PWR fuel having a nominal burn-up of 35 MWd/kg discharged from a PWR could be stored for about 10 years prior to being processed into CANDU fuel. The spent PWR fuel re-fabricated by the OREOX process, could achieve an additional 15 MWd/kg burn-up in a CANDU reactor. The spent DUPIC fuel discharged from the CANDU reactor would be cooled for 50 years in an interim storage facility and then disposed of.

Figure D.1. Schematic flow sheet of DUPIC fuel cycle



As shown in Figure D.2, the key process of the DUPIC fuel cycle is oxidation and reduction of oxide fuel (OREOX) for preparing spent PWR fuel powder to be fabricated CANDU fuels, which is a dry process inherently having no steps to separate any fissile isotopes from the spent PWR fuel.

Figure D.2. Schematic drawing of DUPIC fuel fabrication process



As shown in this figure, the fabrication process begins with disassembling of the spent PWR fuel assembly to extract the fuel rods. The fuel pellet is recovered from the rod by decladding using a mechanical and thermal method. The fuel pellet is then subjected to several cycles of OREOX process to produce resinterable powder feedback. During the oxidation and reduction treatment, the fuel pellet is pulverised to fine powder due to crystallographic transformation in the pellet matrix which causes volume changes and develops more micro-crack in a particle. Once the resinterable powder is

prepared, the pellet and rod manufacturing processes are almost the same as the conventional powder/pellet route in CANDU fuel fabrication except all the processes must be done remotely in the shielded hot cells.

The main DUPIC fuel fabrication campaign has been performed at DUPIC fuel development facility (DFDF) using spent PWR fuel (G23 assembly discharged from Gori Unit #1 in 1986; burnup 35.5 GWD/tU) since January 2000. Until now, nearly one thousand DUPIC fuel pellets were made for the fabrication experiments and the fuel pellet irradiation tests. Twelve elements were manufactured for the irradiation tests in HANARO and for the verification of the DUPIC fuel fabrication technology [7]. A series of DUPIC fuel element fabrication experiments are being performed for the further improvement of the fabrication technology and for more irradiation tests to evaluate the DUPIC fuel performance.

The main waste stream is coming from the DUPIC fuel fabrication process, which mainly consists of metallic components from spent PWR fuel, and the fission gases and semi-volatile fission products released from the bulk fuel material treatment, in addition to the measurable discards and losses such as cladding hulls, trapped or collected fission gases, vitrified Cs/Ru and dirty scrap, and other miscellaneous waste. The projected waste generation for a 400 tHM/yr facility is summarised in Table D.1, assuming that the process loss rate is 1% [6]. Also the estimated generation rates of waste from the DUPIC fuel fabrication plant, based on the electricity of one TWhe produced by the DUPIC fuel cycle, are included in this table.

Table D.1. Projected annual waste generation in DUPIC facility (400 tHM/yr)

Waste constituent	Waste form	Volume for disposal package
^{3}H, ^{14}C	200 litres cemented drums	80.0 m^3
Kr, Xe	Compressed 50 litres cylinders	13.2 m^3
Iodine	Silver zeolites	0.5 m^3
Dirty scrap	Vitrified	9.98 m^3
Cs, Ru	Vitrified	40.44 m^3
Spent fuel H/W	Compacted	35.12 m^3
Zircalloy hulls	Cemented	30.34 m^3
Miscellaneous waste	Cemented	764.1 m^3
Total LLW (Generation rate)		844 m^3 (5.91 m^3/TWhe)
Total ILW (Generation rate)		79 m^3 (0.55 m^3/TWhe)
Total HLW (Generation rate)		50 m^3 (0.35 m^3/TWhe)

On the other hand, the generation rates of the waste coming from the decommissioning of this plant was estimated under assuming that the waste from this plant be very similar to those from a MOX fuel fabrication plant without any process for blending and co-precipitating uranium and plutonium solutions, and with the additional processes for the mechanical and thermal decladding, and for resinterable powder preparation. Based on the small scale MOX fuel fabrication facility [8] and on the additional processes, and assuming that the portion of ILW be 62.7% of the decommissioning waste [9], the rates were estimated to be about 0.61 m^3/TWhe for ILW and 0.36 m^3/TWhe for LLW.

The major R&D items in the near future will include: 1) data accumulation related to the fuel performance, 2) improvement of NDA system related to the accountability of fissile materials to reduce measurement error, 3) improvement of productivity including economics and 4) a safe management of the produced waste [10].

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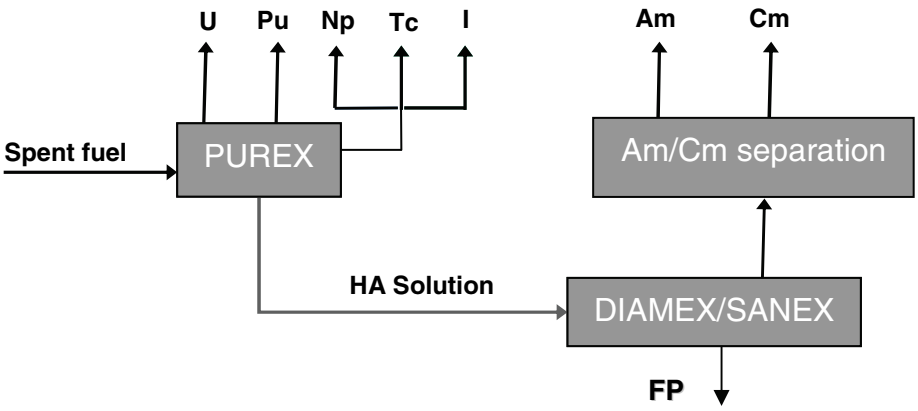
Appendix E

SIMPLIFIED BLOCK DIAGRAMS OF REPROCESSING PROCESSES

Simplified block diagram of the Extented PUREX partitioning process



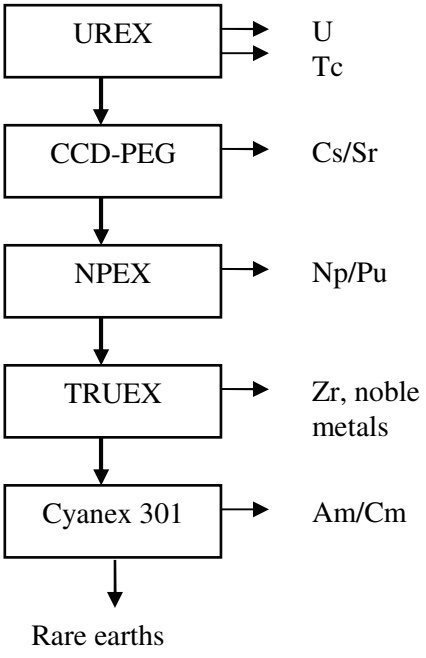
Source: NEA (2005), *Actinide and Fission Product Partitioning and Transmutation*, Boullis, B.; Lecomte, M.; Ranchoux, M.; Baron, P.; Rostaing, C.; Warin, D. and Madic, C., "CEA Actual Demonstration Programme for Actinide Partitioning", Proceedings of 8th Information Exchange Meeting on, Las Vegas, Nevada, USA, 9-11 November 2004, OECD, Paris.

PUREX process. A new flow-sheet has been designed in order to be able to extract the neptunium with a high yield, together with the uranium and plutonium.

DIAMEX process. This process, based on the use of the DMDOHEMA diamide extractant will co-extract the trivalent minor actinides, An(III) = Am and Cm, together with the lanthanides (Ln) fission products from the high active raffinate issuing the implementation of the PUREX process.

DIAMEX-SANEX process. This process, based on the use of a mixture of DMDOHEMA and an acidic organophosphorous extractant (HDEHP) will permit to separate the An(III) vs the most abundant Ln(III) from the product issued from the implementation of the DIAMEX process.

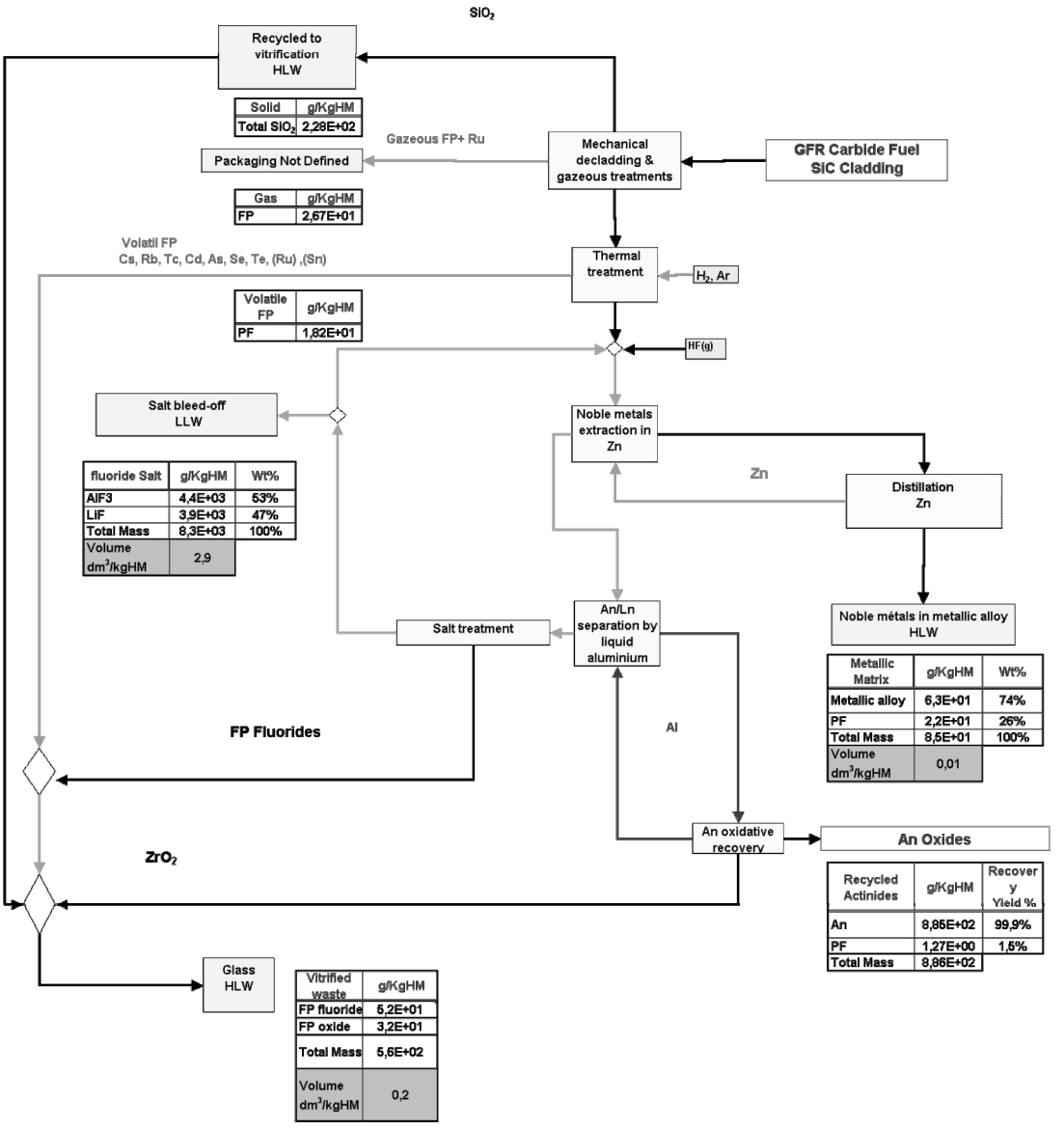
Simplified block diagram of the UREX+ partitioning process



Source: Vandegrift, G.F.; Regalbuto, Monica, C.; Aase, S.B.; Arafat, H.A.; Bakel, A.; Bowers, D.; Byrnes, J.P.; Clark, M.A.; Emery, J.W.; Falkenberg, J.R.; Gelis, A.V.; Hafenrichter, L.; Leonard, R.; Pereira, C.; Quigley, K.J.; Tsai, Y.; Vander Pol, M.H. and Laidler, J.J. (2004), "Lab-Scale Demonstration of the Urex+ Process", Waste Management Symposium, 29 February-4 March 2004, Tucson, AZ.

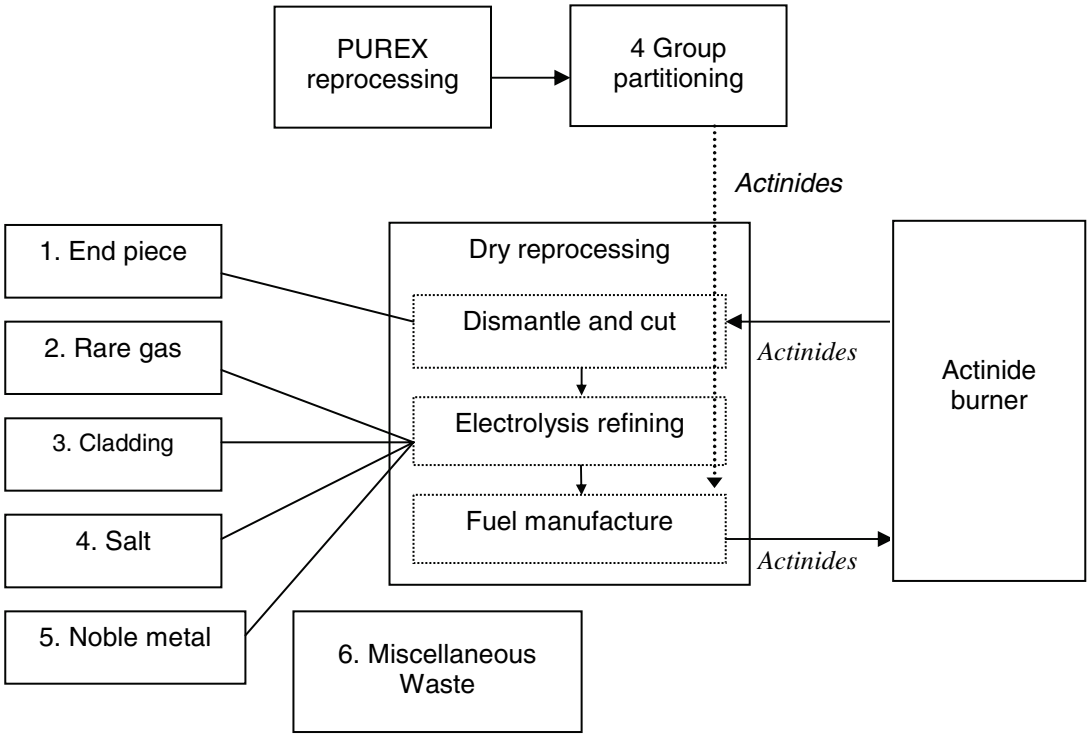
The UREX+ process is a series of five solvent-extraction flowsheets that perform the following operations: (1) recovery of Tc and U (UREX), (2) recovery of Cs and Sr (CDC-PEG), (3) recovery of Pu and Np (NPEX), (4) recovery of Am, Cm, and rare-earth fission products (TRUEX), and finally (5) separation of Am and Cm from the rare earths (Cyanex 301).

Simplified block diagram of carbide fuel reprocessing



Source: CEA.

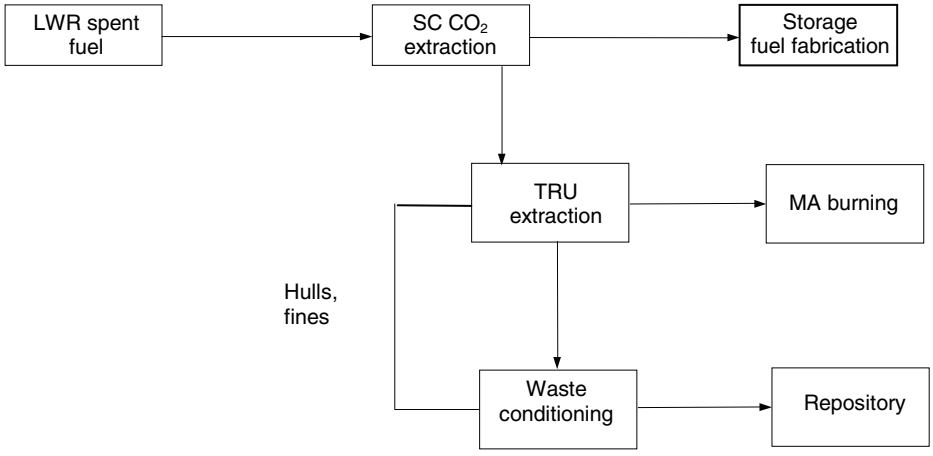
Simplified block diagram of nitride fuel reprocessing



Examples of advanced PUREX processes

* *Use of supercritical CO_2*

Supercritical separation may reduce volume of highly radioactive waste



Source: Carrott, Mike J., Waller, Brenda E., Smart Neil G. and Wai, Chien M. (2001), "Reprocessing of Spent Nuclear Fuel in SC-CO_2", *Chem. Commun.*

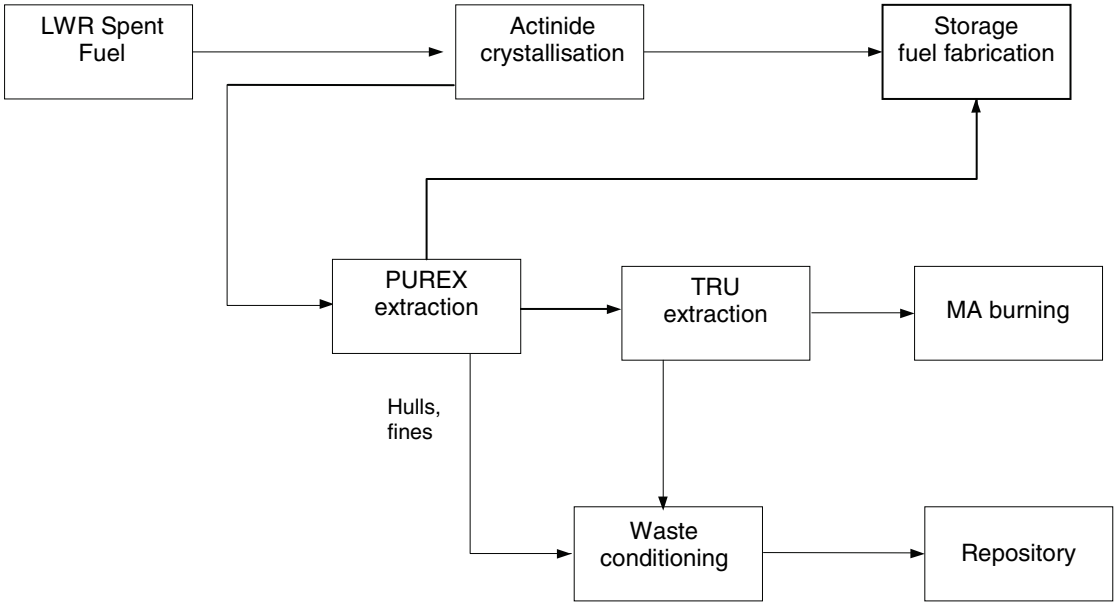
- The extractant oxidises uranium(IV) to uranium(VI) to form a complex that is very soluble (0.0025-0.4981 M) in supercritical CO_2.

- Solid uranium dioxide, a nuclear fuel, dissolves in supercritical CO_2 containing a complex of tri-n-butylphosphate (TBP) and nitric acid.

- No aqueous solution or organic solvent needed.

- The resulting uranyl-nitrate-TBP complex is recovered by reducing the fluid pressure to atmospheric pressure and then allowing the gas to expand into a collection vial.

**** U crystallisation***



Source: Jarvinen, G., Villareal, B., Ford, D., Long, K. (2003), "Actinide Crystallisation Process", LANL.

- Dissolve spent fuel in nitric acid.

- Crystallise uranyl nitrate at -20 to -40°C by evaporative concentration.

- Several recrystallisation-washing steps needed to obtain desired decontamination.

- U can be separated alone, leaving Pu and Np with the fission products.

- Under oxidising conditions U(VI), Pu(VI) and Np(VI) can be co-crystallised and separated later by changing the oxidation state.

Appendix F

MANAGEMENT OF DEPLETED AND SEPARATED URANIUM

Introduction

Depleted uranium (DU) has a ^{235}U content of less than the 0.711 wt% value found in natural uranium. Over 95% of the global stocks of DU (~1.2 million tU) are produced as uranium hexafluoride "tails" from the operation of gas diffusion and centrifuge uranium enrichment plants. The largest holders of depleted UF_6 are the United States with ~480 000 tU and the Russian Federation with ~450 000 tU (NEA and IAEA, 2001).

Minor separated uranium (SepU) stocks have arisen from reprocessing of spent nuclear fuels, commonly Magnox fuels (DU with < 0.5 wt% ^{235}U) and high burn-up gas cooled or light water reactor fuel (DU with > 0.5 wt% ^{235}U) (NEA and IAEA, 2001). Separated uranium trioxide UO_3 is commonly produced during the PUREX or tributyl phosphate plutonium uranium reduction extraction process, which separates uranium as uranyl nitrate solution. This can be concentrated and de-nitrified to form UO_3 solid product. In the United States this was typically packed into plastic bags and placed in 55-gallon steel drums with a plastic liner for storage. SepU salt precipitates e.g. ammonium diuranate, have also been produced from the reprocessing of spent fuel e.g. from fast breeder reactors. These stocks of separated uranium that come from fuel processing typically have significantly higher radioactivity levels than DU from enrichment, primarily due to ^{236}U. Because of this, the storage and disposal requirements may differ, making it important to differentiate these materials. Traces of DU may also be present in the highly radioactive liquid raffinate waste but these are not considered here.

Some properties of depleted uranium compounds from enrichment plants

The most important chemical characteristics of the various DU compounds derived from enrichment plants are their potential solubilities and reactions in water. The chemical properties of these DU compounds under ambient conditions (e.g. Biwer *et al.*, 2000; USDOE, 1999; Duerksen *et al.,* 2000) are briefly summarised below:

- DUF_6 – uranium hexafluoride is a solid at ambient pressure (m.pt. 64°C) and reacts easily with water to form uranyl fluoride UO_2F_2 and hydrofluoric acid (HF), which is highly chemotoxic and corrosive; DUF_6 is stored as a solid in air- and water-tight containers to stop hydrolysis.
- DUF_4 – very slightly soluble – this non-volatile solid compound reacts slowly with moisture to form DUO_2 and HF and eventually other oxides and minerals. Steel container corrosion can be enhanced by the presence of HF in the DUF_4.
- DUO_2 – insoluble – finely divided DUO_2 i.e. uranium dioxide powder can be pyrophoric in air; granules and pellets are not pyrophoric; DUO_2 reacts very slowly with oxygenated groundwater to yield more stable oxides e.g. DU_3O_8, and eventually, minerals; DUO_2 is ideal for disposal in reducing geological environments.

- DU_3O_8 – insoluble – uranium sesquioxide reacts very slowly with oxygenated groundwater to yield more stable uranium minerals; this oxide is very stable, non-corrosive and passively safe for long-term storage; as a conversion product from DUF_6, it tends to be a fine particulate or powder, which may not be a suitable form for waste packaging for disposal, if it is to be regarded as waste.
- DU metal – insoluble – reacts slowly with moisture to form oxides in the presence of oxygen; condensed water moisture promotes the generation of H_2 and uranium hydride, UH_3, which is highly flammable; DU may form pyrophoric surface in the absence of oxygen; oxide layers normally spall to reveal fresh metal surfaces.

Some properties of separated uranium compounds from reprocessing plants

SepU compounds derived from reprocessing plants usually contain radionuclide impurities (fission products and alpha-emitters such as americium, curium, neptunium and plutonium) which can render the compounds as intermediate-level radioactive waste. Chemical impurities are also common e.g. phosphate and sodium.

- UO_3 – insoluble – solid uranium trioxide reacts very slowly with oxygenated groundwater to yield more stable oxides e.g. U_3O_8, and eventually, minerals.
- ADU – insoluble salt – ADU i.e. ammonium diuranate $(NH_4)_2U_2O_7$ can be stored as a floc in tanks, e.g. at Dounreay, United Kingdom. It can be easily heated to expel ammonia gas to yield UO_3; also it is easy to convert it to other salts e.g. Na and Ca, for storage or cementation prior to disposal.

Management of separated uranium from reprocessing operations

United States

The USDOE has a depleted UF_6 management programme to manage the DOE's inventory of DUF_6 in a way that protects the health and safety of workers and the public, and protects the environment until the DUF_6 is either used or disposed. The programme involves three primary activities: cylinder surveillance and maintenance; conversion of DUF_6 to a more stable chemical form for use or disposal; and development of beneficial uses of depleted uranium (see http://web.ead.anl.gov/uranium/mgmtuses/index.cfm).

The final programmatic environmental impact statement (PEIS) for alternative strategies for the long-term management and use of depleted uranium hexafluoride assesses the potential impacts of alternative management strategies for DUF_6 currently stored at three DOE sites: Paducah site, Kentucky, Portsmouth site, Ohio, and K-25 site on the Oak Ridge Reservation, Tennessee. The alternatives analysed in the PEIS include no action, long-term storage as DUF_6, long-term storage as uranium oxide, use as uranium oxide, use as uranium metal, and disposal. The DOE's preferred alternative is to begin conversion of the DUF_6 inventory as soon as possible, either to uranium oxide, uranium metal, or a combination of both, while allowing for use of as much of this inventory as practicable (USDOE, 1999).

Along with the impacts from other management options, the PEIS discussed the environmental impacts from the disposal of depleted uranium oxide from conversion of DUF_6. The impacts were evaluated for three disposal facility options: shallow earthen structures, belowground vaults, and

mines. These were evaluated for a dry environmental setting representative of the western United States e.g. the Nevada Test Site (NTS). The environmental impacts of the same options for disposal of DUF_4 have also been aired (ANL, 2001). The DU products are considered to be low-level waste (LLW) under both DOE orders and Nuclear Regulatory Commission (NRC) regulations. The four potential forms of DU (DU metal, DUF_4, DUO_2, and DU_3O_8) are considered to be acceptable for near-surface disposal at sites at the NTS, although each form has a degree of uncertainty regarding acceptability for disposal with the uncertainty decreasing in the order: metal, UF_4, UO_2, U_3O_8 (Croff *et al.,* 2000).

While DU may qualify for disposal as LLW by US NRC regulations, disposal of large quantities in shallow land burial may cause the site to violate US EPA ground water standards. For this reason, there have been suggestions to create a regulatory framework for "greater confinement disposal" to accommodate DU (and separated U), such as placement in abandoned deep mines.

France

To meet general French safety requirements related to environmental protection issues, COGEMA commissioned an industrial facility in 1984 to convert DUF_6 into DU_3O_8 for safe interim storage. The by-product is high purity aqueous hydrofluoric acid (70% HF) which is sold on to other industries in Europe. A second facility was commissioned in 1993, doubling the capacity. Improvement work has been performed over the years e.g. optimising the de-fluorination output, reducing the corrosion inside the kilns, improving the uranium oxide powder compaction factor etc. Up to the end of 2000, about 220 000 tons of DUF_6 had been de-fluorinated (Hartmann *et al.*, 2001).

United Kingdom and other countries

About 60% of the United Kingdom stockpile of DU is in the DUF_6 form. BNFL in the United Kingdom has about 30 000 tU of DUF_6 and is considering a conversion plant for DU metal. Russia is also considering conversion technologies for its ~450 000 tU of DUF_6. Japan has about 10 000 tU of DUF_6 and is planning to build a conversion plant to store DU sesquioxide as a future energy resource. Urenco (Germany, Netherlands and United Kingdom) is like-minded with its ~16 000 tU inventory (NEA and IAEA, 2001).

Nirex has carried out some initial scoping performance assessment (PA) work for the disposal of cemented DU_3O_8 on the basis of the Nirex Phased Disposal Concept for ILW/LLW (Nirex, 2001). Each 2-metre box would hold 50 tU. In this concept there would be a preference for DU in the U(IV) form e.g. DUO_2, rather than U(VI), to help maintain a low redox potential.

Management of SepU from reprocessing operations

Two options have been discussed recently for managing the ~22 700 metric tonnes of separated UO_3 that are stored in 55-gallon drums stacked three high in several buildings at the Savannah River Site: the feasibility of using the separated UO_3 as a high density inert filler in grout for filling HLW storage tanks for closure; and disposing directly into existing waste facilities on site (Jones, 2002). However, the fission products present in the oxide are an issue under the WAC for the SRS LLW and ILW facilities.

A number of management options of the ADU floc at Dounreay are being considered e.g. continued safe storage, direct in-drum cementation, conversion to sodium diuranate slurry followed by direct in-drum cementation, etc.

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Appendix G

CHARACTERISTICS OF THE CONSIDERED HLW PACKAGES

The number of spent fuel assemblies or containers (in case of reprocessing) per produced TWhe is given Table G.1. For the estimation of the number of vitrified HLW containers a number of assumptions had to be made. For Scheme 1b a waste loading of 40 kg waste per 150 litre container (COGEMA type), which corresponds to the present practise, was assumed. For Scheme 2a two options for the HLW vitrification have been considered. In the first option the HLW from the reprocessing of both fuels (UOX and MOX) is assumed to be mixed before vitrification. In the second option both waste streams are assumed to be conditioned separately. Because the waste at vitrification was too hot (> 2300 W/canister), the glass matrix loading was reduced to 30 kg (option 1) and even to 19 kg for the HLW from the MOX-EU fuel (option 2). In the case of pyro-reprocessing (Scheme 3cV1) the removal of noble metals should allow a higher waste loading (55 kg). The noble metals waste is assumed to be packed also in a standard (COGEMA type) container, with the same maximum weight (412.5 kg), and a metallic matrix consisting of 26% waste and 74% inert metal. For Scheme 3cV1, considering the separation of Cs and Sr, it is assumed that the Cs is conditioned after a sufficient decay period in a glass matrix and disposed of in a geological repository, because of the presence of the very long-lived ^{135}Cs, after a total cooling time of 100 years.

Table G.1. Number of waste forms per produced TWhe for each fuel cycle scheme

Fuel cycle	Waste type	Waste form	Number/TWhe
1a	spent fuel	assemblies	3.981
1b	HLW	canister	2.592
1b	MOX spent fuel	assemblies	0.437
2a	HLW (mixed)	canister	3.987
2a	HLW (UOX)	canister	2.057
2a	HLW (MOX)	canister	1.946
3cV1	HLW	canister	1.181
3cV1	Noble metals	canister	0.085
3cV1 (Cs,Sr)	HLW	canister	0.705
3cV1 (Cs,Sr)	Noble metals	canister	0.085
3cV1 (Cs,Sr)	Cs-waste	canister	0.225

The radionuclide inventories, corresponding to a 50 years cooling time, of the considered high level waste types are given in Table G.2. As mentioned in § 5.4.1, it is assumed that 0.1% of the iodine in the spent fuel is present in the high level waste. At the bottom of the table the total amount of the elements Se and Pd is also given what allows to take into account isotopic dilution when applying solubility limits.

Table G.2. Radionuclide inventory of the considered HLW types

Isotope	Spent fuel UOX 1a (g/TWhe)	HLW UOX 1b (g/TWhe)	Spent fuel MOX 1b (g/TWhe)	HLW UOX 2a (g/TWhe)	HLW MOX 2a (g/TWhe)	HLW 3cV1 (g/TWhe)
SE 79	1.68E+01	1.49E+01	1.79E+00	1.21E+01	4.55E+00	1.30E+01
SR 90	5.69E+02	5.07E+02	3.08E+01	4.08E+02	7.85E+01	2.05E+02
Y 90	1.43E-01	1.27E-01	7.72E-03	1.02E-01	1.97E-02	5.13E-02
ZR 90	1.09E+03	9.73E+02	5.90E+01	7.82E+02	1.51E+02	3.93E+02
ZR 93	2.56E+03	2.28E+03	1.95E+02	1.85E+03	4.97E+02	1.31E+03
NB 93M	2.33E-02	2.08E-02	1.78E-03	1.68E-02	4.52E-03	1.19E-02
MO 95	2.69E+03	2.40E+03	2.54E+02	1.94E+03	6.46E+02	1.52E+03
MO 96	2.26E+02	2.01E+02	2.69E+01	1.62E+02	6.84E+01	9.58E+01
MO 97	2.90E+03	2.58E+03	2.96E+02	2.09E+03	7.52E+02	1.67E+03
MO 98	3.05E+03	2.71E+03	3.38E+02	2.20E+03	8.59E+02	2.07E+03
MO100	3.44E+03	3.07E+03	3.90E+02	2.48E+03	9.93E+02	2.36E+03
TC 99	2.89E+03	2.57E+03	3.15E+02	2.08E+03	7.99E+02	1.92E+03
RU106	7.42E-13	6.59E-13	1.36E-13	5.33E-13	3.57E-13	5.83E-13
RH106	6.88E-19	6.12E-19	1.26E-19	4.95E-19	3.31E-19	5.41E-19
PD106	1.53E+01	1.36E+01	2.81E+00	1.10E+01	7.36E+00	1.20E+01
PD107	8.29E+02	7.38E+02	2.02E+02	5.96E+02	5.19E+02	1.02E+03
AG107	3.98E-03	3.54E-03	9.69E-04	2.86E-03	2.49E-03	4.91E-03
SN126	7.95E+01	7.08E+01	1.42E+01	5.73E+01		9.18E+01
SB126	2.70E-05	2.40E-05	4.82E-06	1.94E-05		3.12E-05
TE126	2.48E-02	2.21E-02	4.42E-03	1.78E-02		2.86E-02
I 127	1.45E+02	1.29E-01	2.48E+01	1.04E-01	6.61E-02	1.82E-01
I 129	6.30E+02	5.61E-01	8.44E+01	4.53E-01	2.20E-01	5.47E-01
CS134	2.80E-05	2.50E-05	2.83E-06	2.02E-05	6.28E-06	8.49E-06
CS135	1.64E+03	1.46E+03	2.98E+02	1.18E+03	9.60E+02	3.50E+03
CS137	1.40E+03	1.25E+03	1.55E+02	1.01E+03	3.98E+02	9.33E+02
BA134	1.04E+02	9.28E+01	1.05E+01	7.50E+01	2.33E+01	3.15E+01
BA135	2.22E-02	1.98E-02	4.04E-03	1.60E-02	1.30E-02	4.74E-02
BA137	2.55E+03	2.26E+03	2.80E+02	1.83E+03	7.22E+02	2.23E+03
BA137M	2.14E-04	1.90E-04	2.36E-05	1.54E-04	6.07E-05	1.42E-04
ND143	3.15E-08	2.81E-08	3.70E-09	2.27E-08	1.09E-08	3.35E-08
PM147	8.22E-04	7.33E-04	9.66E-05	5.93E-04	2.85E-04	8.74E-04
SM147	1.20E+02	1.07E+02	1.41E+01	8.66E+01	4.16E+01	1.28E+02
EU154	2.72E+00	2.43E+00	4.54E-01	1.96E+00	1.12E+00	1.28E+00
GD154	9.16E+01	8.16E+01	1.52E+01	6.59E+01	3.75E+01	4.29E+01
TL207	3.70E-13	2.60E-13	5.38E-15	2.11E-13	1.07E-14	2,93E-18
TL208	9.41E-11	8.37E-14	1.97E-12	6.77E-14	5.00E-15	1,25E-22
PB206	2.85E-08	1.12E-08	3.76E-09	9.07E-09	1.92E-09	3,73E-11
PB207	7.15E-07	5.50E-07	1.03E-08	4.45E-07	2.25E-08	3,64E-12
PB208	1.63E-03	1.45E-06	3.40E-05	1.17E-06	8.64E-08	7,24E-16
PB209	3.86E-13	2.95E-13	8.49E-15	2.38E-13	2.21E-14	6,72E-17
PB210	7.10E-08	2.08E-08	1.13E-08	1.68E-08	3.71E-09	1,14E-10
PB211	2.86E-12	2.01E-12	4.16E-14	1.63E-12	8.27E-14	2,26E-17
PB212	5.58E-08	4.96E-11	1.16E-09	4.02E-11	2.96E-12	7,40E-20
PB214	4.52E-13	9.85E-14	8.40E-14	7.97E-14	1.82E-14	8,37E-16
BI209	2.35E-08	1.89E-08	2.94E-10	1.53E-08	7.05E-10	1,79E-12

Table G.2. Radionuclide inventory of the considered HLW types *(continued)*

Isotope	Spent fuel UOX 1a (g/TWhe)	HLW UOX 1b (g/TWhe)	Spent fuel MOX 1b (g/TWhe)	HLW UOX 2a (g/TWhe)	HLW MOX 2a (g/TWhe)	HLW 3cV1 (g/TWhe)
BI210	4.36E-11	1.28E-11	6.96E-12	1.03E-11	2.28E-12	7,04E-14
BI211	1.72E-13	1.21E-13	2.50E-15	9.79E-14	4.97E-15	1,36E-18
BI212	5.29E-09	4.71E-12	1.11E-10	3.81E-12	2.81E-13	7,02E-21
BI213	9.40E-14	7.17E-14	2.06E-15	5.80E-14	5.39E-15	1,64E-17
BI214	3.36E-13	7.31E-14	6.23E-14	5.92E-14	1.35E-14	6,22E-16
PO210	1.17E-09	3.46E-10	1.85E-10	2.80E-10	6.18E-11	1,87E-12
PO212	2.80E-19	2.49E-22	5.84E-21	2.01E-22	1.49E-23	3,71E-31
PO213	1.41E-22	1.08E-22	3.10E-24	8.72E-23	8.09E-24	2,46E-26
PO214	4.64E-20	1.01E-20	8.61E-21	8.17E-21	1.86E-21	8,59E-23
PO215	2.40E-18	1.69E-18	3.48E-20	1.36E-18	6.92E-20	1,90E-23
PO216	2.08E-13	1.85E-16	4.34E-15	1.50E-16	1.10E-17	2,75E-25
PO218	5.24E-14	1.14E-14	9.74E-15	9.24E-15	2.11E-15	9,71E-17
AT217	1.13E-18	8.63E-19	2.48E-20	6.98E-19	6.48E-20	1,97E-22
RN219	5.43E-15	3.82E-15	7.90E-17	3.09E-15	1.57E-16	4,30E-20
RN220	8.40E-11	7.48E-14	1.75E-12	6.05E-14	4.47E-15	1,11E-22
RN222	9.65E-11	2.10E-11	1.79E-11	1.70E-11	3.88E-12	1,79E-13
FR221	1.05E-14	8.00E-15	2.30E-16	6.47E-15	6.01E-16	1,82E-18
RA223	1.38E-09	9.71E-10	2.01E-11	7.85E-10	3.98E-11	1,09E-14
RA224	4.85E-07	4.32E-10	1.01E-08	3.50E-10	2.58E-11	6,44E-19
RA225	4.65E-11	3.54E-11	1.02E-12	2.87E-11	2.66E-12	8,09E-15
RA226	1.50E-05	3.27E-06	2.79E-06	2.64E-06	6.03E-07	2,78E-08
RA228	7.21E-12	1.15E-12	3.59E-14	9.30E-13	1.28E-14	4,10E-16
AC225	3.14E-11	2.39E-11	6.89E-13	1.94E-11	1.80E-12	5,46E-15
AC227	9.92E-07	6.98E-07	1.44E-08	5.64E-07	2.86E-08	7,87E-12
AC228	8.79E-16	1.40E-16	4.39E-18	1.13E-16	1.56E-18	5,01E-20
TH227	2.30E-09	1.62E-09	3.35E-11	1.31E-09	6.64E-11	1,82E-14
TH228	9.45E-05	8.41E-08	1.97E-06	6.80E-08	5.02E-09	1,25E-16
TH229	8.57E-06	6.54E-06	1.89E-07	5.29E-06	4.92E-07	1,50E-09
TH230	7.18E-02	7.51E-03	1.72E-02	6.08E-03	1.51E-03	1,62E-04
TH231	6.29E-08	5.60E-11	9.90E-10	4.54E-11	3.11E-11	2,63E-12
TH232	2.12E-02	2.87E-03	1.09E-04	2.33E-03	3.28E-05	1,27E-06
TH234	2.69E-05	2.40E-08	2.79E-06	1.94E-08	6.83E-09	8,71E-09
PA231	2.24E-03	1.39E-03	3.28E-05	1.13E-03	5.72E-05	2,61E-08
PA233	7.05E-05	5.88E-05	6.66E-06	4.75E-05	1.49E-05	5,11E-08
PA234	4.05E-10	3.61E-13	4.20E-11	2.92E-13	1.03E-13	1,31E-13
PA234M	9.06E-10	8.09E-13	9.40E-11	6.53E-13	2.30E-13	2,93E-13
U 232	3.52E-03	3.14E-06	7.36E-05	2.54E-06	1.87E-07	7,10E-06
U 233	3.42E-02	2.43E-02	1.87E-03	1.97E-02	5.07E-03	1,70E-05
U 234	6.53E+02	6.43E-01	2.20E+02	5.19E-01	1.77E+00	1,81E+00
U 235	1.55E+04	1.38E+01	2.44E+02	1.12E+01	7.66E+00	6,49E-01
U 236	1.41E+04	1.31E+01	8.81E+01	1.06E+01	4.06E+00	1,12E+00
U 237	1.12E-05	1.08E-08	7.74E-06	8.76E-09	2.70E-08	2,20E-08
U 238	1.86E+06	1.66E+03	1.93E+05	1.34E+03	4.73E+02	6,03E+02
U 240	4.53E-12	4.04E-15	3.26E-12	3.27E-15	8.31E-15	1,97E-17
NP237	2.08E+03	1.73E+03	1.96E+02	1.40E+03	4.39E+02	1,51E+00

Table G.2. Radionuclide inventory of the considered HLW types *(continued)*

Isotope	Spent fuel UOX 1a (g/TWhe)	HLW UOX 1b (g/TWhe)	Spent fuel MOX 1b (g/TWhe)	HLW UOX 2a (g/TWhe)	HLW MOX 2a (g/TWhe)	HLW 3cV1 (g/TWhe)
NP238	4.00E-07	3.55E-07	1.16E-06	2.87E-07	3.03E-06	3,94E-08
NP239	5.15E-04	4.58E-04	4.56E-04	3.70E-04	1.94E-03	1,33E-06
NP240	3.47E-13	3.10E-16	2.50E-13	2.51E-16	6.37E-16	1,51E-18
PU238	7.08E+02	9.52E-01	4.31E+02	7.69E-01	5.10E+00	2,52E+00
PU239	1.31E+04	1.58E+01	5.86E+03	1.28E+01	2.59E+01	7,93E+01
PU240	6.55E+03	1.84E+02	5.09E+03	1.49E+02	8.19E+02	5,40E+01
PU241	3.62E+02	3.50E-01	2.50E+02	2.83E-01	8.71E-01	7,11E-01
PU242	2.31E+03	2.14E+00	1.70E+03	1.73E+00	1.56E+01	6,34E+00
PU243	2.11E-12	1.88E-12	3.89E-12	1.52E-12	7.42E-12	8,17E-13
PU244	2.36E-01	2.11E-04	1.70E-01	1.71E-04	4.34E-04	1,03E-06
AM241	3.61E+03	8.42E+02	2.67E+03	6.80E+02	2.31E+03	9,72E+00
AM242M	2.25E+00	1.99E+00	6.49E+00	1.61E+00	1.70E+01	2,21E-01
AM243	6.00E+02	5.35E+02	5.32E+02	4.32E+02	2.26E+03	1,55E+00
CM242	5.68E-03	5.05E-03	1.64E-02	4.08E-03	4.30E-02	5,60E-04
CM243	1.37E+00	1.22E+00	1.05E+00	9.89E-01	2.77E+00	1,11E-02
CM244	4.45E+01	3.97E+01	5.07E+01	3.20E+01	1.79E+02	2,28E-01
CM245	2.15E+01	1.91E+01	5.10E+01	1.54E+01	1.46E+02	2,87E-01
CM246	2.99E+00	2.66E+00	3.95E+00	2.16E+00	1.12E+01	1,86E-01
CM247	5.92E-02	5.27E-02	1.09E-01	4.26E-02	2.08E-01	2,29E-02
CM248	1.01E-02	8.98E-03	2.55E-02	7.25E-03	1.00E-03	1,29E-02
BK249						8,73E-22
CF249						8,39E-04
CF250						8,08E-06
CF251						6,89E-06
CF252						2,91E-13
Se_tot	2.00E+02	1.77E+02	1.77E+01	1.44E+02	4.21E+01	1.13E+02
Pd_tot	5.45E+03	4.86E+03	1.09E+03	3.92E+03	2.76E+03	6.17E+03

Appendix H

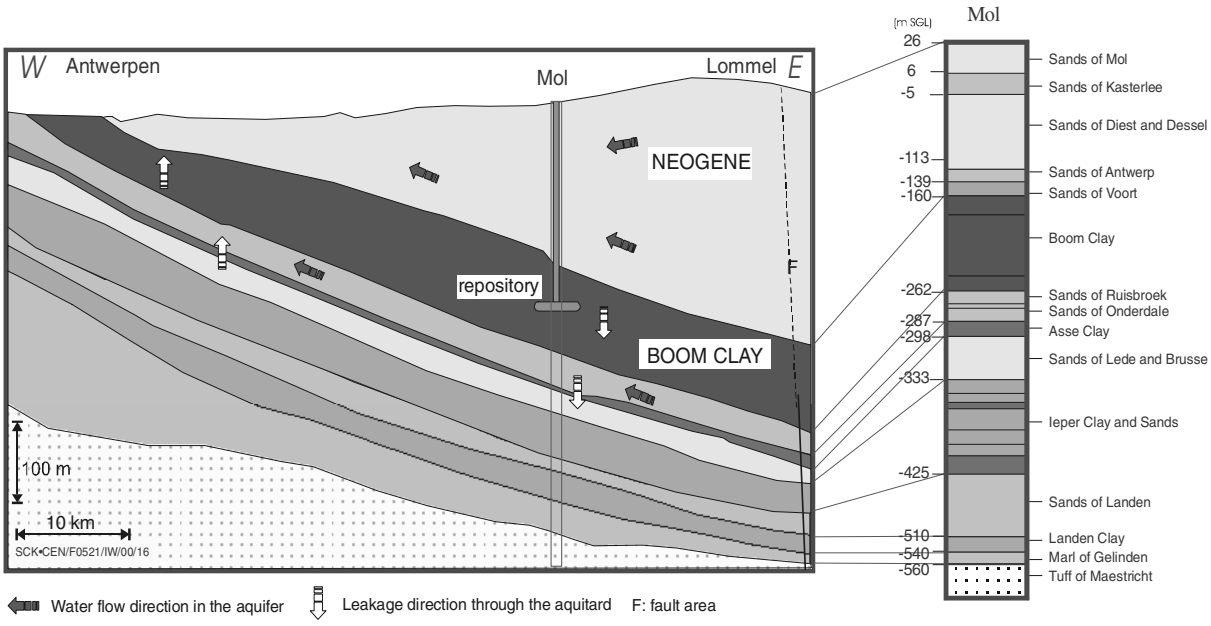
ASSESSMENT OF THE DISPOSAL OF HLW FROM ADVANCED FUEL CYCLE SCHEMES IN THE BOOM CLAY FORMATION[1]

1. Considered repository

1.1 Reference repository concept

The repository considered for this study is the reference repository that is described in detail in the Belgian safety case report SAFIR 2 [1]. This reference repository is assumed to be excavated in the Boom Clay Formation at the Mol site. A scheme of the geology of the Mol site is given in Figure H.1.

Figure H.1. Geological cross-section of Mol site



The considered repository will have a central access facility consisting of at least two vertical transport shafts and two transport galleries. The disposal galleries will be excavated perpendicular upon the transport galleries. The density of the HLW disposal in a gallery and the spacing between disposal galleries depend on the thermal output of the disposed HLW.

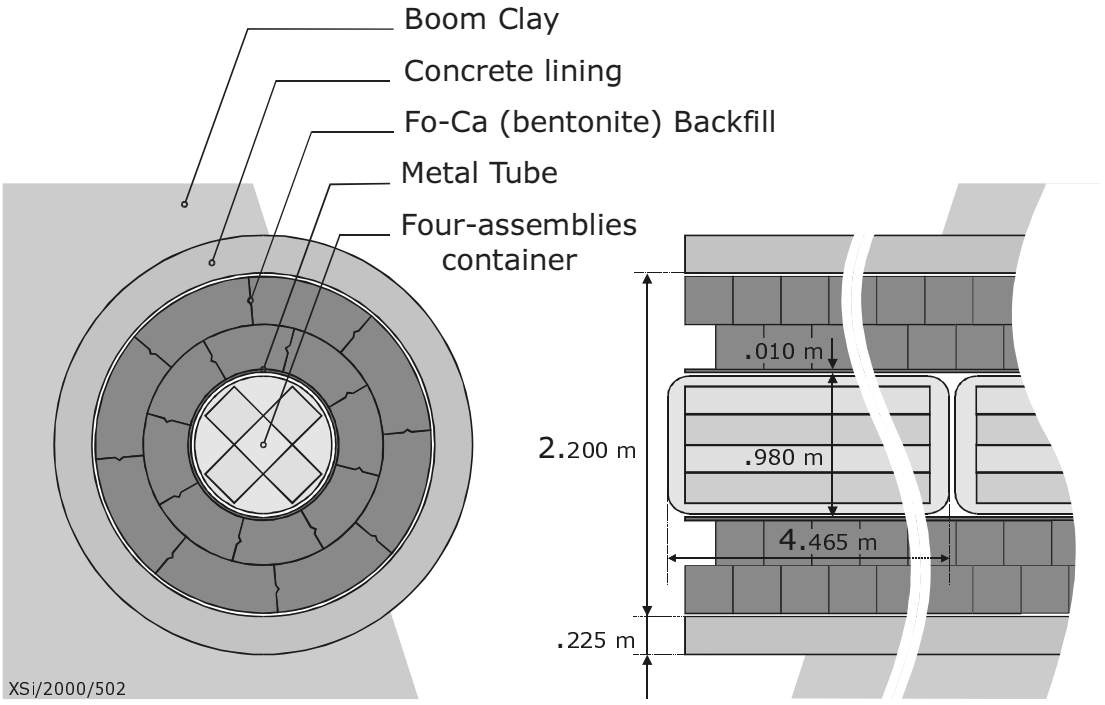
1. This work was contributed by Jan Marivoet and Eef Weetjens (SCK•CEN, Mol, Belgium).

The HLW canisters will be placed one after the other, or with a relatively small distance between two canisters (to respect temperature limitations), in the centre of the disposal galleries. A clayey backfill, e.g. a mixture of swelling Ca-bentonite and sand, will be placed between the HLW canisters and the gallery walls. Because the Boom Clay is plastic clay, a concrete lining is required to avoid convergence of the gallery walls. A scheme of a HLW gallery configuration for UOX spent fuel with a burn up of 45 GWd/tHM is given in Figure H.2.

1.2 Impact of the thermal output of the HLW on the repository configuration

The maximum allowable disposal density for the SAFIR 2 repository configuration has been determined by making detailed thermal calculations. It appeared that in the case of disposal of vitrified HLW (comparable with the HLW of fuel cycle scheme 1b) and of disposal of spent fuel (comparable with schemes 1a and 1b) the limitation for the maximum temperature at the container/backfill interface, which has to remain lower than 100°C, is the most stringent thermal limitation. The available results have shown that, as far as the thermal output of the HLW is dominated by the fission products ^{137}Cs and ^{90}Sr, which have half-lives of about 30 years, detailed thermal calculations can be approximated by applying a linear thermal limit of 300 W per metre of gallery.

Figure H.2. Gallery configuration for UOX spent fuel disposal (four assemblies per container)



For scheme 1a, three spent fuel assemblies can be packed in a disposal container (also called canister). After 50 years cooling the thermal output of one spent fuel assembly is 530 W. The length of the considered container is 5.28 m. As the linear thermal limitation is respected, no spacing between containers is needed. The minimum length of the HLW disposal galleries for scheme 1a is 7.03 m per TWhe.

For scheme 1b, only one MOX spent fuel assembly can be packed in a disposal container. After 50 years cooling the thermal output of one spent fuel assembly is 1 925 W. The length of the considered container is 5.28 m. To respect the linear thermal limitation, a spacing of 1.14 m between

containers is needed. For the vitrified high level waste, the thermal output of one canister is 459 W. The length of the considered container is 1.335 m. To respect the linear thermal limitation, a spacing of 0.20 m between containers is needed. The minimum length of the HLW disposal galleries for scheme 1b is 6.77 m per TWhe.

For scheme 2a, the thermal output of one vitrified high level waste canister is 501 W. To respect the linear thermal limitation, a spacing of 0.33 m between containers is needed. The minimum length of the HLW disposal galleries for scheme 2a is 6.66 m per TWhe.

For scheme 3cV1, it is assumed that the HLW and the noble metals are separately conditioned. The thermal output of one vitrified high level waste canister is 483 W. To respect the linear thermal limitation, a spacing of 0.27 m between containers is needed. The thermal output of the noble metals is negligible. The total length of the HLW disposal galleries for scheme 3cV1 is 2.02 m per TWhe.

When it is assumed for scheme 3cV1 that Cs and Sr will be separated from the high level waste, the thermal output of one vitrified high level waste canister is after a 50 years cooling time only 17.7 W. Several canisters can be placed in a gallery section. By assuming that 3 canisters are placed in a gallery section the length of the disposal galleries for HLW and noble metals is 0.35 m per TWhe. The thermal output of one vitrified Cs-waste canister is after a 100 years cooling time 537 W. To respect the linear thermal limitation spacing of 0.45 m between containers is needed. The length of the disposal galleries for Cs-waste is 0.40 m per TWhe. The total length of the HLW disposal galleries for the variant of scheme 3cV1 with separation of Cs and Sr is 0.75 m per TWhe.

The needed length of disposal galleries is about equal for schemes 1a, 1b and 2a. However, the advanced fuel cycle scheme 3cV1 results in a reduction of the needed length of the HLW disposal galleries with a factor 3.5 in comparison with the reference once through fuel cycle scheme 1a. Separation of Cs and Sr allows further reducing of the needed gallery length.

2. Transport model and parameters

2.1 Conceptual model

For this assessment a strongly simplified conceptual model of the repository system is used.

The actinide decay chains are simplified by not explicitly modelling the short-lived radionuclides. The decay chains considered in the simulations are:

4N+1 : $^{237}Np \rightarrow {}^{233}U \rightarrow {}^{229}Th$
4N+2 : $^{238}U \rightarrow {}^{234}U \rightarrow {}^{230}Th \rightarrow {}^{226}Ra$
4N+3 : $^{235}U \rightarrow {}^{231}Pa$
4N : $^{248}Cm \rightarrow {}^{244}Pu \rightarrow {}^{236}U \rightarrow {}^{232}Th$

After perforation of the container, water comes in contact with the waste form. The radionuclides that are not embedded in a stable waste matrix, i.e. the gap inventory or instantaneously released fraction (IRF) in case of spent fuel disposal, can immediately dissolve in the porewater. The corrosion of the waste matrix starts and consequently radionuclides are released from the waste matrix into the porewater. When the concentration of a given radionuclide should exceed its solubility limit, this radionuclide will precipitate. The contributions from the barriers provided by the buffer and the gallery lining are here conservatively neglected.

The main barrier of the repository system is the clay barrier provided by the host formation. Because of the low hydraulic conductivity of clay (2 10^{-12} m/s in the case of the Boom Clay) and the low hydraulic gradient over the host formation (0.02 at the Mol site) the radionuclide transport is essentially due to molecular diffusion; the contribution of advective transport is less than 5% of the contribution of diffusive transport. The slow diffusive transport is for many radionuclides further retarded by sorption upon clay minerals.

The radionuclides that have migrated through the host clay formation reach the surrounding aquifers. The aquifer overlying the Boom Clay layer is locally drained by a river called Kleine Nete. Retardation due to sorption on minerals present in the overlying sand formations is conservatively neglected. The transport of unretarded radionuclides through the overlying aquifer layer from the top of the host formation into the river lasts about 2 000 years.

To calculate the dose to a member of the critical group the results of the biosphere modelling made for SAFIR 2 [1] are used. The critical group considered in the SAFIR 2 biosphere model is a small self-sustaining community that uses water from the Kleine Nete river for drinking, watering of cattle and irrigation of crop fields and pastures. The considered exposure pathways are ingestion, inhalation and external irradiation.

2.2 Transport parameters

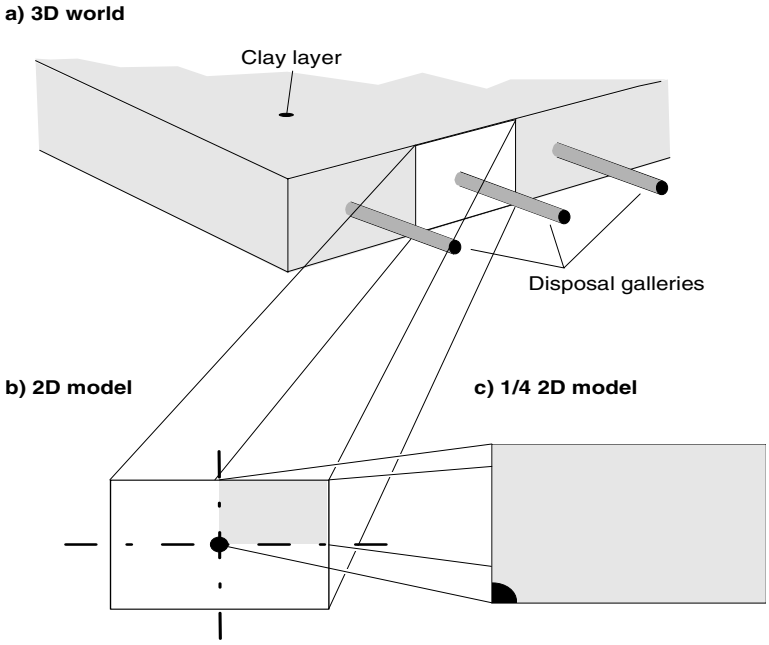
The repository is assumed to be located in the middle of the host clay formation. The thickness of the Boom Clay Formation at the Mol site is about 100 m. To take the thickness of a few thin more permeable layers and of the excavated gallery into account, the effective thickness of the clay layer is taken equal to 40 m. Table H.1 gives the values of the transport parameters in the clay layer used for the transport calculations.

Table H.1. Transport parameters for the clay layer

	Solubility limit	**η**	**D_p**	**R**			**Solubility limit**	**η**	**D_p**	**R**
	(mol/L)	(–)	(m^2/s)	(–)			(mol/L)	(–)	(m^2/s)	(–)
C	–	0.12	1.2E-10	1		**Cm**	2.0E-06	0.30	2.0E-10	1 000
Cl	–	0.12	2.0E-10	1		**Pu**	5.0E-07	0.30	2.0E-10	1 000
Ni	1.0E-03	0.30	2.0E-10	50		**Np**	1.0E-06	0.30	2.0E-10	1 000
Se	5.5E-08	0.13	2.0E-10	1		**U**	3.2E-08	0.30	2.0E-10	300
Zr	1.0E-06	0.30	2.0E-10	400		**Pa**	1.0E-05	0.30	2.0E-10	400
Nb	3.2E-06	0.30	2.0E-10	50		**Th**	5.0E-07	0.30	2.0E-10	500
Tc	3.0E-08	0.30	2.0E-10	1		**Ra**	1.0E-09	0.30	2.0E-10	50
Pd	1.0E-07	0.30	2.0E-10	20						
Sn	5.5E-07	0.30	2.0E-10	20						
I	–	0.12	2.0E-10	1						
Cs	–	0.30	3.6E-10	3 600						

(η: diffusion accessible porosity; D_p: pore diffusion coefficient; R: retardation factor.)

Figure H.3. Reduction of the 3-D real world problem to a 2-D model



3. Calculations

3.1 Computer model

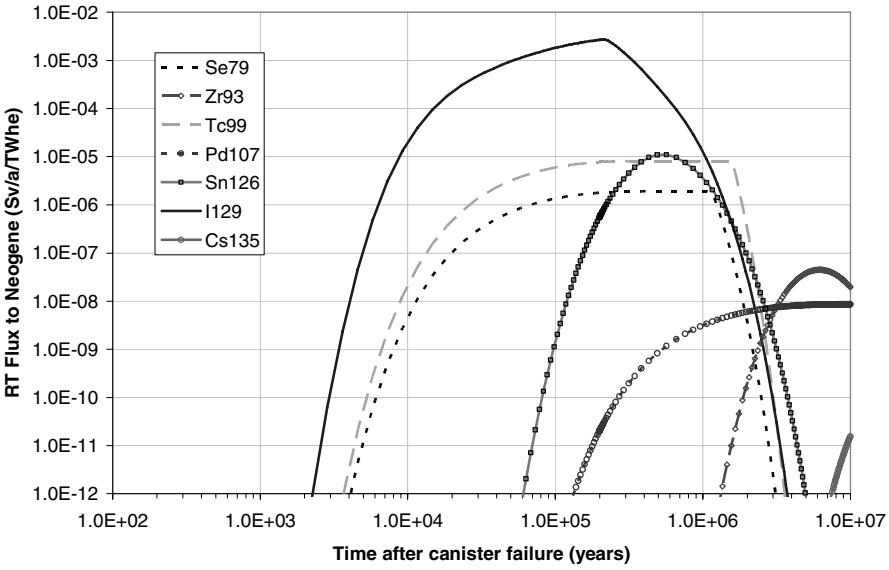
The calculations made for this assessment are carried out with the finite volume code PORFLOW 3.07 [2]. The existence of symmetry axes allows to strongly reduce the modelled domain (see Figure H.3).

3.2 Results

Fuel cycle scheme 1a

The main radiotoxicity fluxes released from the host clay formation into the overlying aquifer are shown in Figure H.4 for the fission products and the actinides of decay chain 4N+1. The consequences of the other 3 decay chains have also been calculated but are not presented here because the resulting radiotoxicity is at least one order of magnitude smaller than the one of ^{229}Th. The cumulative radiotoxicity fluxes are given in Figure H.5 and the doses calculated for the river pathway are given in Figure H.6.

Figure H.4. Radiotoxicity fluxes of fission products (a) and of the actinides from decay chain 4N+1



(b) released into the overlying aquifer (scheme 1a)

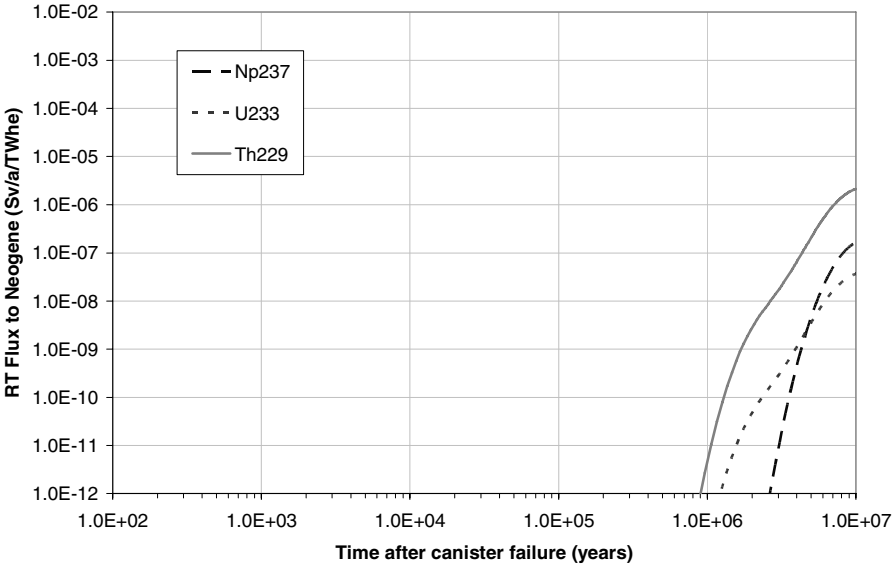


Figure H.5. Cumulative radiotoxicity fluxes of the fission products released into the overlying aquifer (scheme 1a)

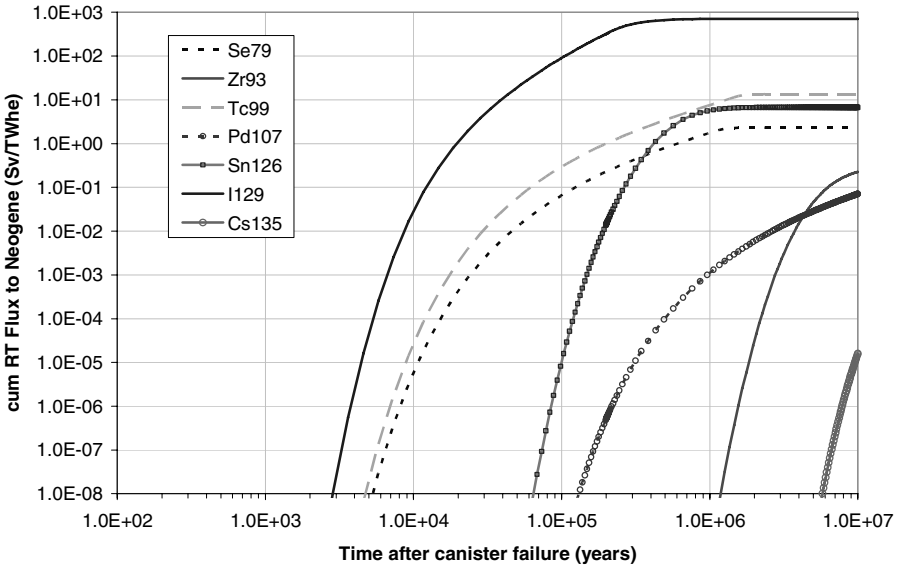
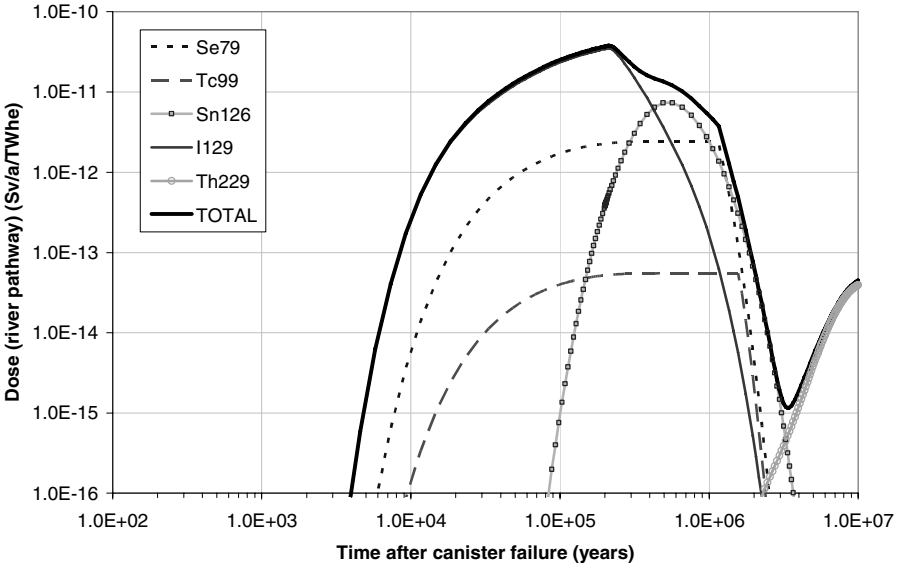


Figure H.6. Dose via the river pathway per TWhe (scheme 1a)



Fuel cycle scheme 1b

The main radiotoxicity fluxes of fission products released from the host clay formation into the overlying aquifer are shown in Figure H.7. The doses calculated for the river pathway are given in Figure H.8.

Figure H.7. Radiotoxicity fluxes of the fission products released into the overlying aquifer (scheme 1b)

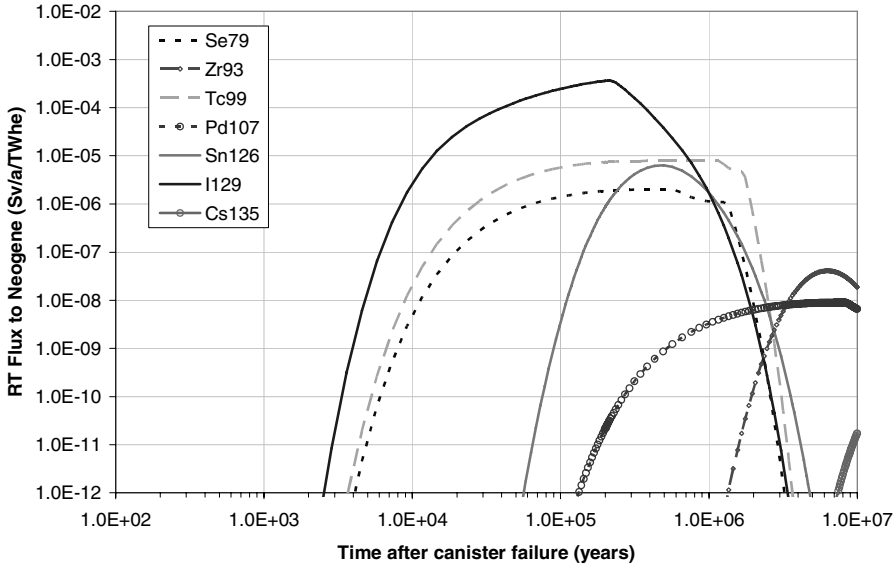
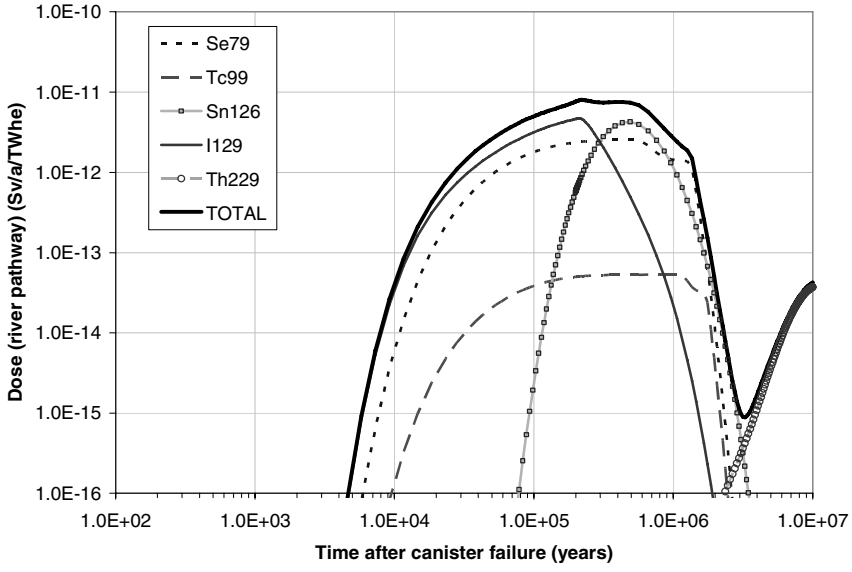


Figure H.8. Dose via the river pathway per TWhe (scheme 1b)



Fuel cycle scheme 2a

The main radiotoxicity fluxes of fission products released from the host clay formation into the overlying aquifer are shown in Figure H.9. The doses calculated for the river pathway are given in Figure H.10.

Figure H.9. Radiotoxicity fluxes of the fission products released into the overlying aquifer (scheme 2a)

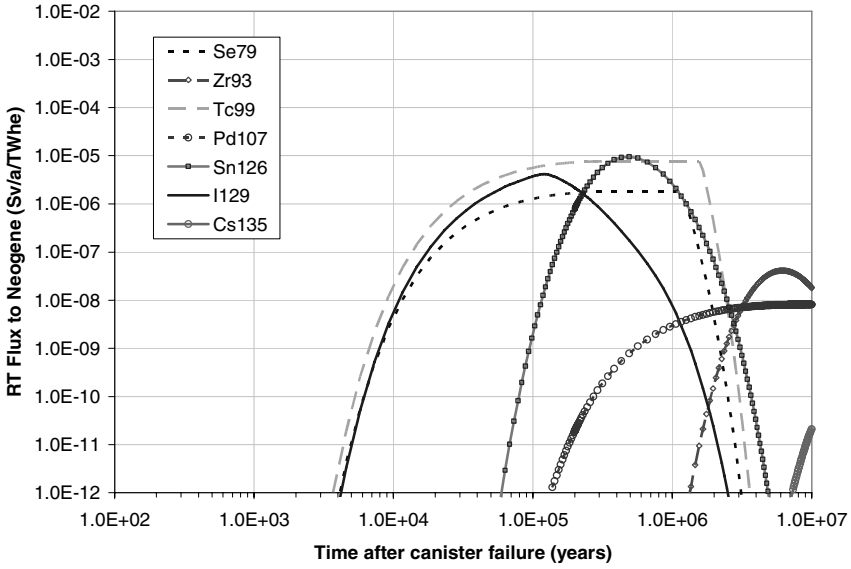
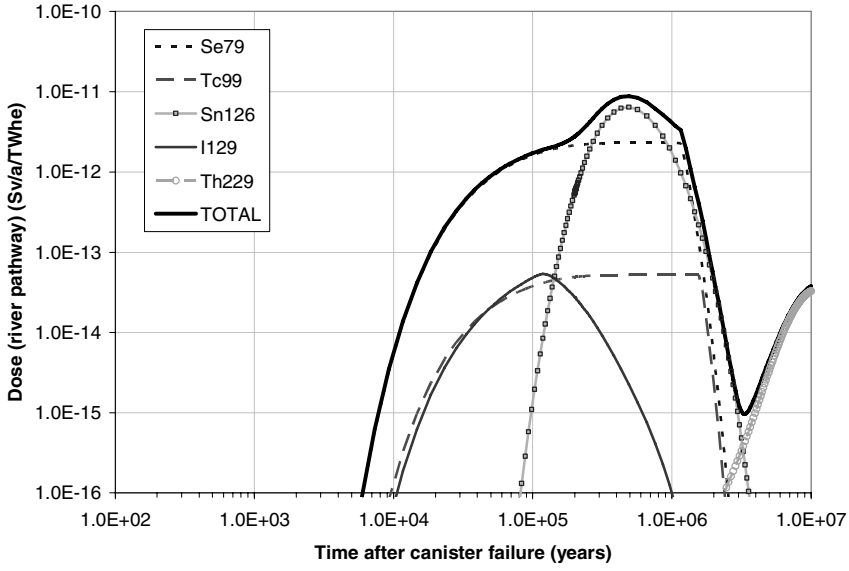


Figure H.10. Dose via the river pathway per TWhe (scheme 2a)



Fuel cycle scheme 3cV1

The main radiotoxicity fluxes of fission products released from the host clay formation into the overlying aquifer are shown in Figure H.11. The doses calculated for the river pathway are given in Figure H.12.

Figure H.11. Radiotoxicity fluxes of the fission products released into the overlying aquifer (scheme 3cV1)

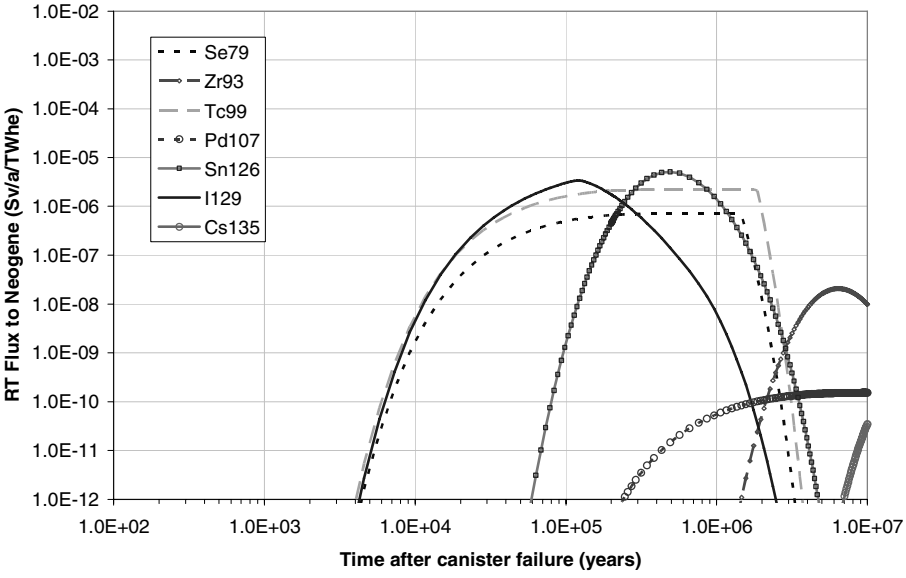
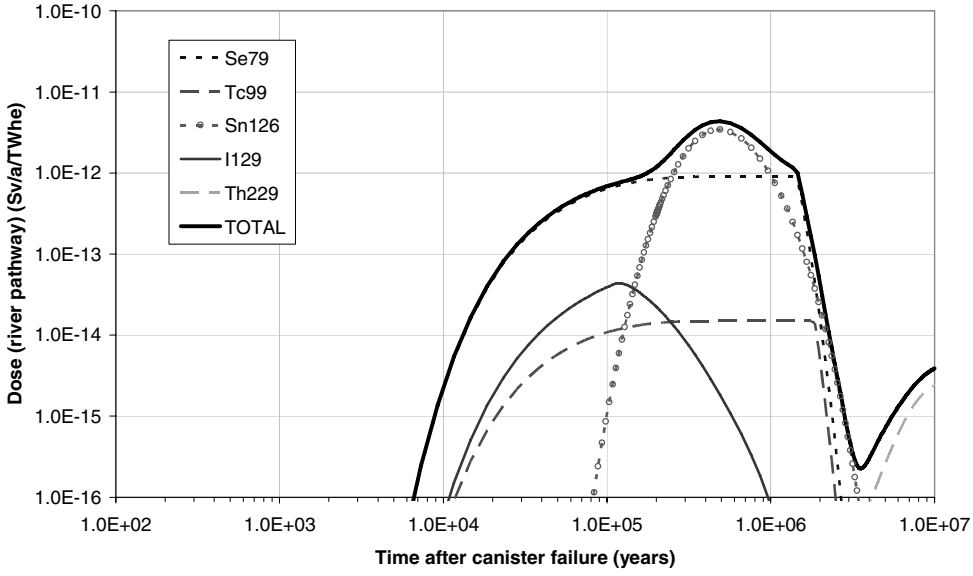


Figure H.12. Dose via the river pathway per TWhe (scheme 3cV1)



Fuel cycle scheme 3cV1 with separation of Cs and Sr

The main radiotoxicity fluxes of fission products released from the host clay formation into the overlying aquifer are shown in Figure H.13. The doses calculated for the river pathway are given in Figure H.14.

Figure H.13. Radiotoxicity fluxes of the fission products released into the overlying aquifer (scheme 3cV1, separation of Cs and Sr)

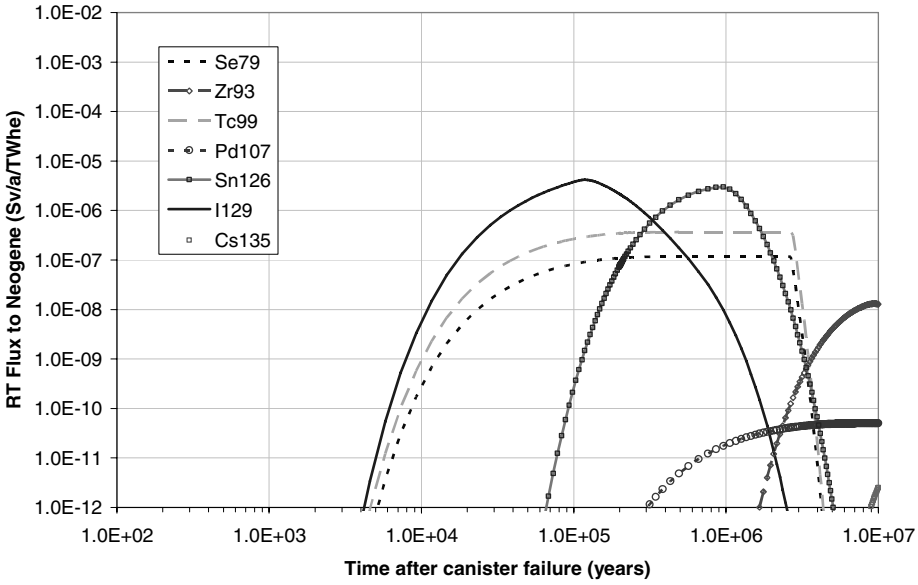
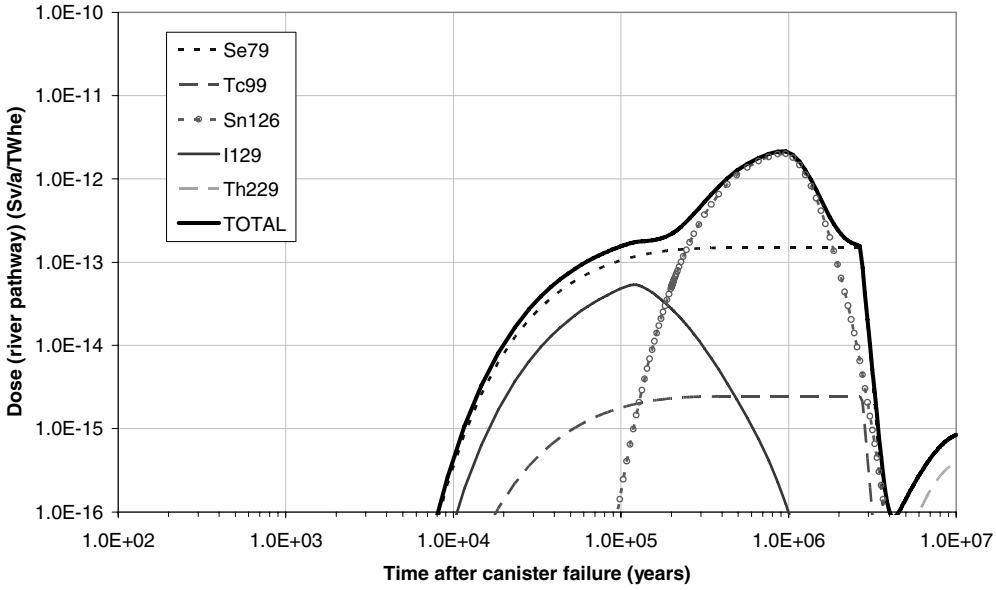


Figure H.14. Dose via the river pathway per TWhe (scheme 3cV1, separation of Cs and Sr)



Short discussion of the results

Figures H4-H8 clearly show that in case of disposal of spent fuel (scheme 1a and partially for scheme 1b) the total radiotoxicities and doses are strongly dominated by the ^{129}I present in spent fuel. Other important radionuclides are ^{79}Se and ^{126}Sn. No actinides are released from the host formation before 1 million years. At 10 million years the total dose is dominated by the actinides, mainly ^{229}Th, but this dose is 3 orders of magnitude smaller than the maximum dose due to ^{129}I or ^{79}Se and ^{126}Sn.

Figures H.10, H.12 and H.14 show that in case of disposal of HLW arising from the considered advanced fuel cycle schemes 2a, 3cV1 and 3cV1 with separation of Cs and Sr, four fission products, ^{79}Se, ^{99}Tc, ^{126}Sn, and ^{129}I (for which it is assumed that a fraction of 0.1% is present in the conditioned HLW), are the main contributors to the total dose. At 10 million years the total dose is dominated by the actinides, mainly ^{229}Th.

The total doses calculated for the five considered fuel cycle schemes are given in Figure H.15. Between 5 000 and 1 million years the difference between the calculated doses is essentially due to the different fraction of ^{129}I that is going into the repository: 100% for scheme 1a, about 11% for scheme 1b, and 0.1% for schemes 2a and 3cV1. The doses calculated for the variant of scheme 3cV1 with separation of Cs and Sr are a factor 3 lower than those calculated for scheme 3cV1 itself. In both cases the inventory of long-lived radionuclides disposed of in the repository is identical. The lower dose is explained by the denser disposal configuration for the variant with Cs and Sr separation, what results in an amplification of the release limitation by solubility limits.

Figure H.15. Total doses via the river pathway per TWhe calculated for the considered schemes

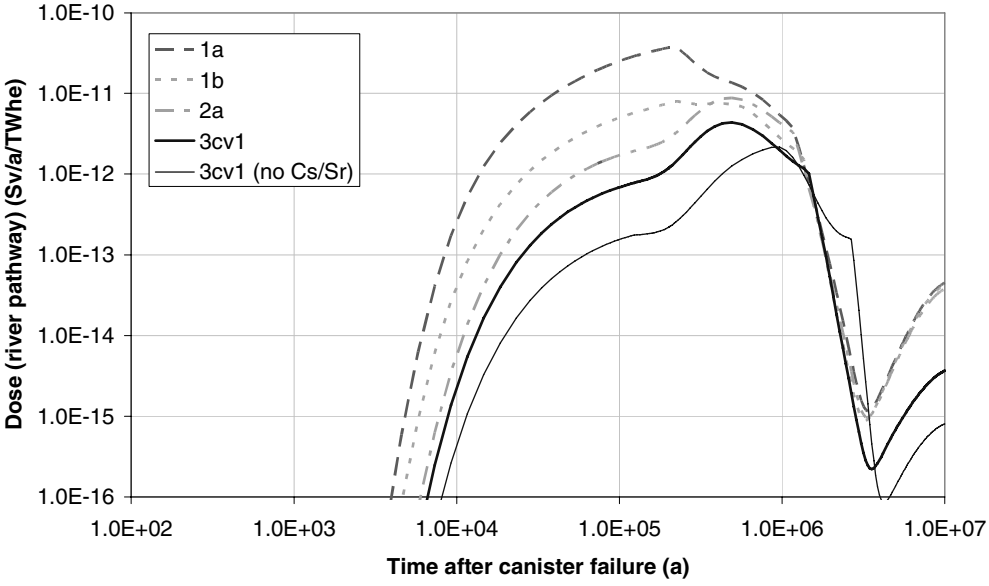


Table H.2 gives the radiotoxicity fluxes integrated per timeframe for the 5 considered fuel cycle schemes. In case of schemes 1a and 1b the radiotoxicity fluxes integrated over all the timeframes are completely dominated by ^{129}I. In case of schemes 2a and 3cV1 the radiotoxicity fluxes integrated over the timeframes 10^3-10^4y and 10^4-10^5y are dominated by ^{99}Tc and ^{129}I; for the timeframe 10^5-10^6y the

main contributors are ^{126}Sn and ^{99}Tc; for the timeframe 10^6-10^7y the main contributors are ^{99}Tc and the actinides (mainly ^{229}Th). The cumulative releases of the fission products ^{79}Se, ^{93}Zr, ^{126}Sn are about a factor 2 lower in the case of scheme 3cV1 and a factor 7 for the variant considering the separation of Cs and Sr in comparison with those calculated for the reference scheme 1a; although, for all schemes about the same amount of fission products is disposed of in the repository. This difference can be explained by the more compact disposal configuration for the advanced fuel cycle schemes, which makes that the solubility limits give a stronger contribution to the confinement of these radionuclides. This effect is even much stronger for ^{107}Pd, for which reduction factors of 56 and even 168 for the variant considering the separation of Cs and Sr are calculated; the separate conditioning of the non-heat producing noble metals allows to consider very dense disposal configurations.

4. Conclusions

The HLW arising from the advanced fuel cycle schemes generates less heat than the HLW from the reference fuel cycle 1a. For scheme 3cV1 the needed length of HLW disposal galleries is reduced with a factor 3.5, and even a factor 9 if Cs and Sr are separated from the HLW.

The maximum total dose resulting from the disposal of the HLW forms from fuel cycle schemes1b and 2a is a factor 4.4 lower than the one from the reference scheme 1a. For scheme 3cV1 the reduction of the maximum total dose is a factor 9, and even a factor 18 in the case that Cs and Sr are separated from the HLW prior to conditioning. However, the lower doses are essentially due the separation of ^{129}I from the HLW during the reprocessing of the spent fuels. If the ^{129}I has to be captured and disposed of in the same HLW repository, the doses resulting from all considered fuel cycle schemes will be about equal.

Although the amount of generated fission products is more or less proportional with the produced energy, the releases of solubility limited fission products, such as ^{79}Se, ^{99}Tc, ^{107}Pd and ^{126}Sn, are lower in the case of advanced fuel cycle schemes, because the denser repository configuration makes the contribution of the solubility limitation more efficient. This is especially the case for the variant scheme considering the separation of Cs and Sr.

References

[1] ONDRAF/NIRAS (2001), “Safety Assessment and Feasibility Interim Report 2 – SAFIR 2”, Brussels, Report NIROND 2001–06 E.

[2] Runchal, A. (1997), “PORFLOW User's Manual”, ACRi, Bel Air, USA.

Table H.2. Cumulative radiotoxicity fluxes per timeframe calculated for the considered schemes

Scheme	From	To	^{79}Se	^{93}Zr	^{99}Tc	^{107}Pd	^{126}Sn	^{129}I	^{135}Cs	Actinides	Total
1a	1	1000	2.11E-23		9.74E-23	1.06E-72	6.20E-70	6.35E-19		1.36E-85	6.36E-19
	1000	10000	1.84E-05	2.63E-78	8.43E-05	4.56E-33	2.58E-29	7.82E-02		1.92E-45	7.83E-02
	10000	100000	7.07E-02	1.01E-40	3.15E-01	6.22E-10	1.58E-05	9.63E+01	3.49E-68	9.71E-27	9.67E+01
	100000	1000000	1.80E+00	1.27E-09	7.73E+00	1.32E-03	5.91E+00	6.12E+02	6.44E-29	7.61E-07	6.27E+02
	1000000	10000000	5.12E-01	2.26E-01	5.68E+00	6.98E-02	7.94E-01	9.78E+01	1.59E-05	7.30E+00	1.12E+02
	1	10000000	2.38E+00	2.26E-01	1.37E+01	7.11E-02	6.71E+00	8.06E+02	1.59E-05	7.30E+00	8.36E+02
1b	1	1000	2.20E-23		9.37E-23	1.12E-72	8.67E-68	8.51E-20		5.05E-86	8.52E-20
	1000	10000	1.92E-05	2.53E-78	8.11E-05	4.80E-33	3.64E-28	1.05E-02		6.69E-46	1.06E-02
	10000	100000	7.37E-02	9.74E-41	3.03E-01	6.55E-10	4.08E-05	1.31E+01	1.89E-68	3.68E-27	1.34E+01
	100000	1000000	1.58E+00	1.10E-09	7.44E+00	1.38E-03	3.44E+00	8.24E+01	9.83E-29	7.30E-07	9.49E+01
	1000000	10000000	5.25E-01	2.05E-01	4.83E+00	7.13E-02	3.72E-01	1.33E+01	1.79E-05	6.88E+00	2.61E+01
	1	10000000	2.18E+00	2.05E-01	1.26E+01	7.27E-02	3.81E+00	1.09E+02	1.79E-05	6.88E+00	1.34E+02
2a	1	1000	1.99E-23		9.21E-23	1.01E-72	8.94E-70	4.91E-24		2.10E-86	1.17E-22
	1000	10000	1.74E-05	2.48E-78	7.98E-05	4.32E-33	3.72E-29	2.27E-05		2.16E-46	1.20E-04
	10000	100000	6.69E-02	9.58E-41	2.98E-01	5.89E-10	2.28E-05	1.76E-01	1.84E-68	1.73E-27	5.41E-01
	100000	1000000	1.71E+00	1.21E-09	7.32E+00	1.24E-03	5.07E+00	5.65E-01	1.27E-28	7.23E-07	1.47E+01
	1000000	10000000	5.19E-01	2.08E-01	5.50E+00	6.61E-02	5.62E-01	1.77E-01	2.20E-05	6.18E+00	1.32E+01
	1	10000000	2.29E+00	2.08E-01	1.31E+01	6.73E-02	5.63E+00	9.19E-01	2.20E-05	6.18E+00	2.84E+01
3cV1	1	1000	7.82E-24		2.63E-23	1.87E-74	4.15E-68	4.99E-24		2.84E-88	3.92E-23
	1000	10000	6.82E-06	7.10E-79	2.28E-05	8.02E-35	1.72E-28	2.30E-05		4.58E-48	5.26E-05
	10000	100000	2.62E-02	2.74E-41	8.52E-02	1.09E-11	1.83E-05	1.78E-01	3.01E-68	5.24E-28	2.89E-01
	100000	1000000	6.69E-01	3.45E-10	2.09E+00	2.31E-05	2.71E+00	5.66E-01	2.07E-28	1.08E-07	6.04E+00
	1000000	10000000	4.02E-01	1.02E-01	2.14E+00	1.23E-03	3.03E-01	1.78E-01	3.59E-05	6.08E-01	3.74E+00
	1	10000000	1.10E+00	1.02E-01	4.32E+00	1.25E-03	3.01E+00	9.22E-01	3.59E-05	6.08E-01	1.01E+01

Table H.2. Cumulative radiotoxicity fluxes per timeframe calculated for the considered schemes (*continued*)

Scheme	From	To	^{79}Se	^{93}Zr	^{99}Tc	^{107}Pd	^{126}Sn	^{129}I	^{135}Cs	Actinides	Total
3cV1	1	1000	1.29E-24		4.35E-24	6.24E-75	6.85E-69	4.99E-24		8.49E-88	1.06E-23
(Cs,Sr)	1000	10000	1.12E-06	1.17E-79	3.76E-06	2.67E-35	2.84E-29	2.30E-05		1.29E-47	2.79E-05
	10000	100000	4.33E-03	4.52E-42	1.40E-02	3.64E-12	3.01E-06	1.78E-01	2.16E-69	1.47E-28	1.96E-01
	100000	1000000	1.10E-01	5.69E-11	3.45E-01	7.71E-06	1.71E+00	5.66E-01	1.50E-29	1.78E-08	2.73E+00
	1000000	10000000	2.10E-01	5.34E-02	6.77E-01	4.09E-04	9.25E-01	1.78E-01	2.62E-06	1.00E-01	2.14E+00
	1	10000000	3.25E-01	5.34E-02	1.04E+00	4.17E-04	2.63E+00	9.22E-01	2.62E-06	1.00E-01	5.07E+00

Appendix I

HLW DISPOSAL – GRANITE CONCEPT, SPAIN

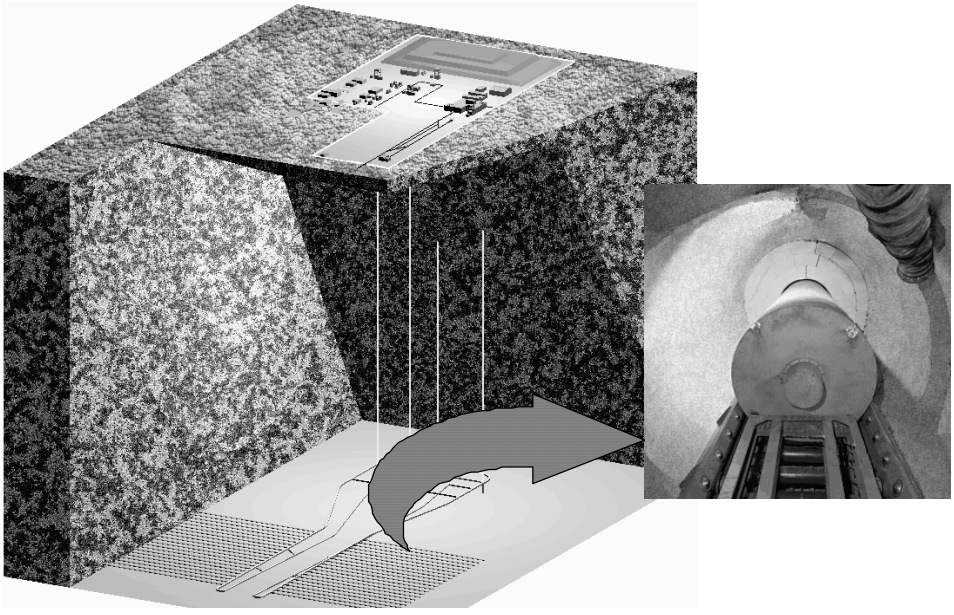
1. Introduction

In this document the results of the deterministic calculations for the 4 schemes (1a, 1b, 2a and 3cV1) of the NEA study are given. The ENRESA disposal concept for spent fuel, carbon steel canister and bentonite buffer for horizontal disposal in a granite formation has been used, and is also described. The methodology, models and input data used for performance assessment calculations are presented too.

2. Disposal system description

The reference concept in granite is based in the disposal of spent fuel in carbon steel canisters in long horizontal galleries. Canisters are surrounded by high density bentonite. The disposal galleries are grouped into two symmetrically arranged disposal panels. Access is accomplished by means of "main drifts" which run perpendicular to the tunnels. The main drifts meet at a central area, which includes the required underground infrastructure. Communications between the surface and the central underground area are accomplished by way of 4 access shafts. Figure I.1 shows a view of the underground installations.

Figure I.1. Underground installations for the ENRESA repository concept



Spent fuel elements are disposed in cylindrical carbon steel canisters measuring 4.54 m in length and 0.90 m in diameter, capable of maintaining 4 PWR elements in a subcritical configuration. The thickness of the wall of the canister is 0.10 m at the sides and 0.12 m at the ends, and is capable of

withstanding the pressures to which it is subjected under disposal conditions and of providing a minimum period of containment greater than one thousand years. After being unloaded from the reactor, the fuel elements are temporarily stored for their thermal power to decay to a level at which they may be disposed of with a total thermal power of 1 220 W per canister. For the 14 400 PWR equivalent fuel elements a total 3 600 canisters will be required.

Canisters are disposed in cylindrical receptacles, constructed with blocks of compacted bentonite inside the horizontal drifts measuring 500 m in length and 2.4 m in diameter located at a depth of 500 m in the host formation (see Figure I.2). The separation between canisters is determined mainly by thermal constrains. A separation of 2 m between canisters has been established, and of 35 m between disposal drifts, in order not to exceed a temperature of 100°C in the bentonite. The detailed dimensions of an individual "cell" are shown in Figure I.3.

Figure I.2. Longitudinal section of an emplacement tunnel

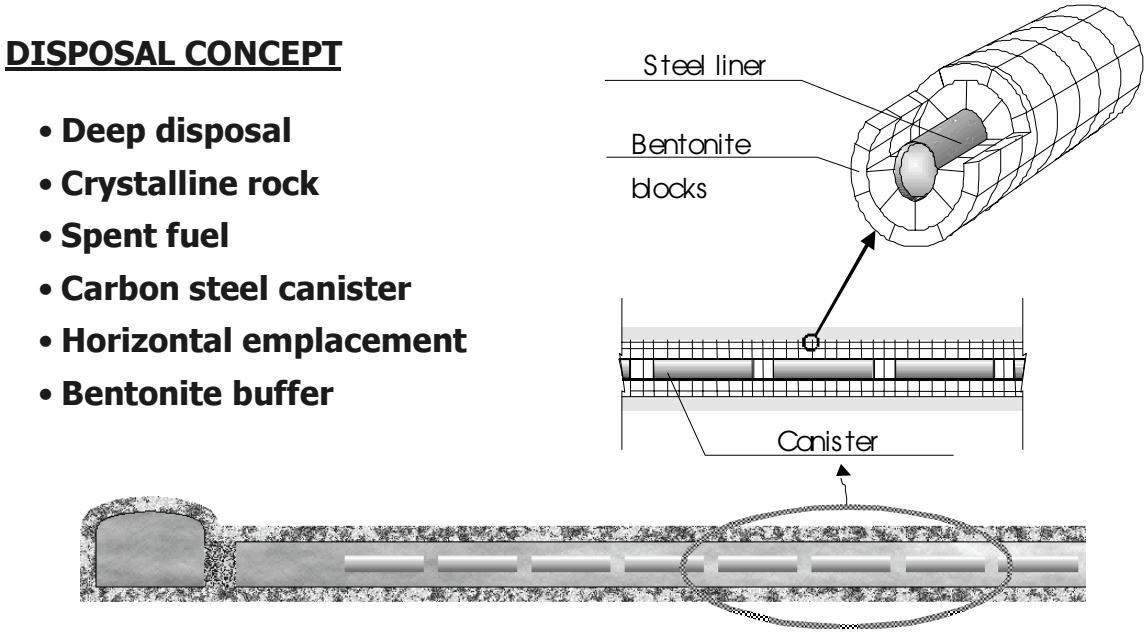
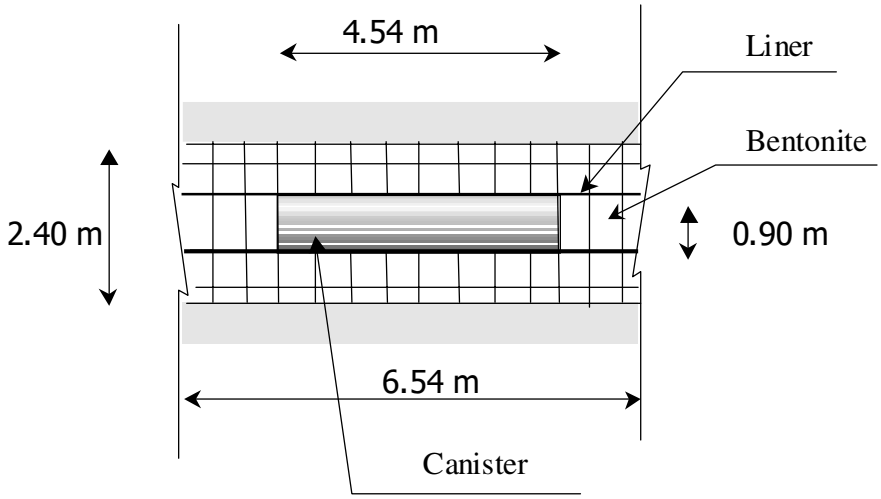


Figure I.3. Dimensions of an individual disposal cell



When a disposal drift is filled, it is sealed using blocks of bentonite and structures of concrete at the entry to the drifts. Galleries, shafts and other remaining rock cavities are backfilled, after completion of waste emplacement, with a mixture of bentonite and natural sand or an appropriate crushed material. The backfilling material will consist of 10% bentonite (increasing up to 20% at the top of the galleries) and suitably graded sand.

Sealing plugs of galleries and shafts are made of compacted bentonite blocks piled over the whole cross section.

The use of cement-based materials is limited to both plugs at the end of the disposal drifts. The uppermost parts of the shafts are, as well, plugged with concrete in order to prevent human intrusion.

3. Performance assessment

3.1 Description of the calculations

After waste emplacement, the buffer saturates in about 20 years due to bentonite high suction pressures. When the groundwater contacts the canister, general corrosion of the carbon steel starts. Minimum canister lifetime is 1 300 years after disposal, when the peak temperatures due to decay heat have passed and many fission products have decayed. In the calculations the 3 600 canisters do not fail simultaneously, but sequentially.

After canister failure, and since no credit is given to the cladding as a barrier, water reaches the irradiated UO_2 and radionuclide releases start. Three different contributions are considered: an instant release fraction (IRF) that corresponds to the radionuclide inventory in the pellet-cladding gap and the grain boundaries, a relatively fast release of the activation products in the metallic parts of the fuel elements and a slow release of the inventory in the UO_2 matrix (that contains the vast majority of the radionuclides in the waste).

The radionuclides and stable species released from the waste enter the water around the waste, where they dissolve or precipitate depending on their solubility limits. Solute concentration in the water around the waste is the boundary condition at the inner surface of bentonite for near field transport calculations.

Granite close to the disposal drift is modelled as an equivalent porous medium, with water flowing parallel to the disposal drift axis. There is an Excavation Damaged Zone (EDZ) surrounding the disposal drift, where Darcy velocity is assumed to be 10 times higher than in intact host rock. Radionuclides diffuse radially through the bentonite into the EDZ and adjacent intact granite, where flowing water transports the contaminants out of the near field. The radionuclide flux leaving the near field enters the geosphere.

Geosphere has been modelled as an equivalent porous medium with main fractures explicitly represented. With this model and using a particle tracking code a probability distribution of the water travel time in the geosphere is obtained, to serve as the main input parameter to the transport model.

Since transport in granite is controlled by fractures, in transport calculations the geosphere is modelled as a single one-dimensional planar fracture (or stream tube). Longitudinal dispersion and matrix diffusion into the wall rock are modelled, including sorption on the rock matrix. Neither solubility limits nor sorption on fracture surfaces/infill is considered.

The flux of radionuclides (Bq/a) that cross the geosphere is discharged to a river which flows at a rate of 10^6 m^3/a. Radionuclide concentrations in river water are calculated dividing the release rate from the geosphere by the river flow rate.

River water is used by the critical group, which is a self-sufficient community that produces most of its aliments using water from the river. Doses to an average member of the critical group are calculated, taking into account all the relevant exposure pathways, not only drinking water. Calculated doses are compared with a reference value of 10^{-4} Sv/y.

3.2 Probabilistic and deterministic calculations

In the assessment study of the ENRESA reference disposal concept both deterministic and probabilistic calculations are done:

- In the *deterministic calculation* the process described in the previous section is done just one time assigning to each parameter the “best estimate” value.
- In the *probabilistic calculation* the uncertainty in each parameter (Pu solubility, water travel time in the geosphere, etc.) is explicitly taken into account, because each parameter is represented using a probability distribution, not a single value.
- The process described in the previous section is done 100 times, assigning to each stochastic variable a value that has been sampled using LHS (Latin Hypercube Sampling).
- Each of the 100 runs provides a result for the time-dependent dose rate. Averaging over the 100 runs for each instant, the time-dependent mean dose is obtained. This mean dose is compared with the reference limit 10^{-4} Sv/a.

In this document only deterministic calculations are performed.

3.3 Deterministic calculations for the 4 schemes of the NEA Study

3.3.1 Doses

Figure I.4. Normalised total doses in the different schemes

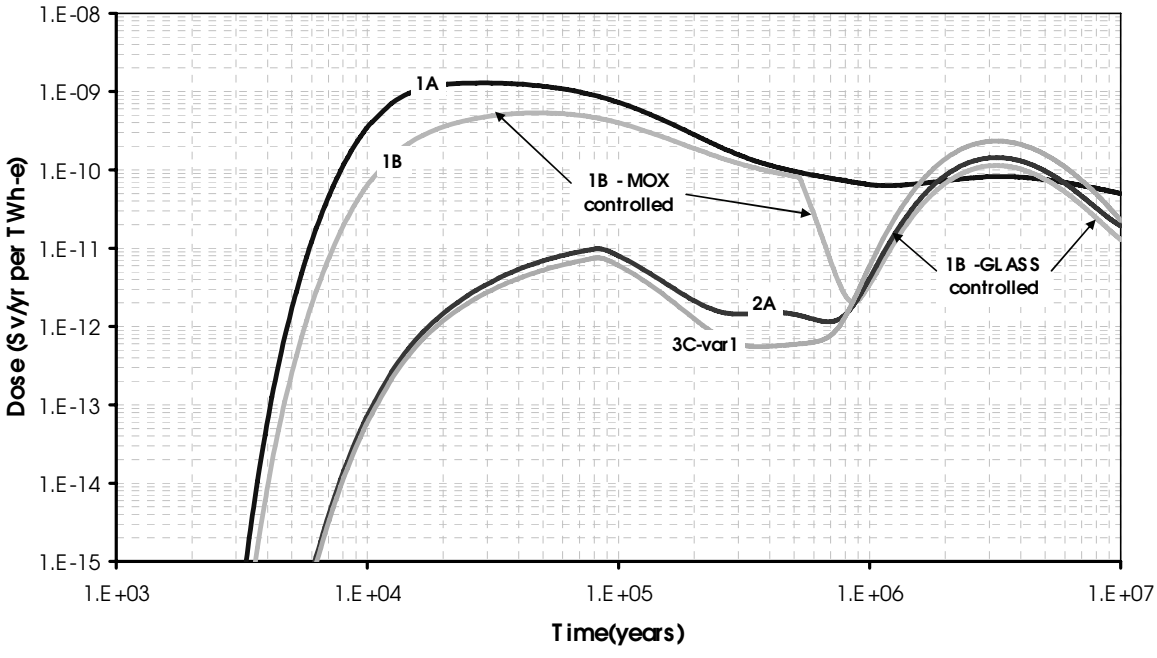


Figure I.4 shows the total doses in the 4 analysed schemes. The curve of scheme 1b is the superposition of two curves: one for the MOX fuel that controls the doses up to $8x10^5$ years, and another curve for the vitrified waste that is dominant after 10^6 years.

The dose curves for schemes 2a, 3cV1 and the vitrified waste in scheme 1b are quite similar. This is a logical result because in the 3 schemes the same model for the glass matrix degradation is used and the inventories of fission products are quite similar.

Figure I.5 shows the doses per radionuclide in scheme 1a (UOX). ^{129}I controls the peak doses and the contribution of ^{135}Cs becomes significant after 10^6 years. Doses due to actinides and daughters start only after several million years, because these radionuclides are strongly sorbed in the bentonite barrier and the geosphere.

Figure I.5. Dose per radionuclide in scheme 1a

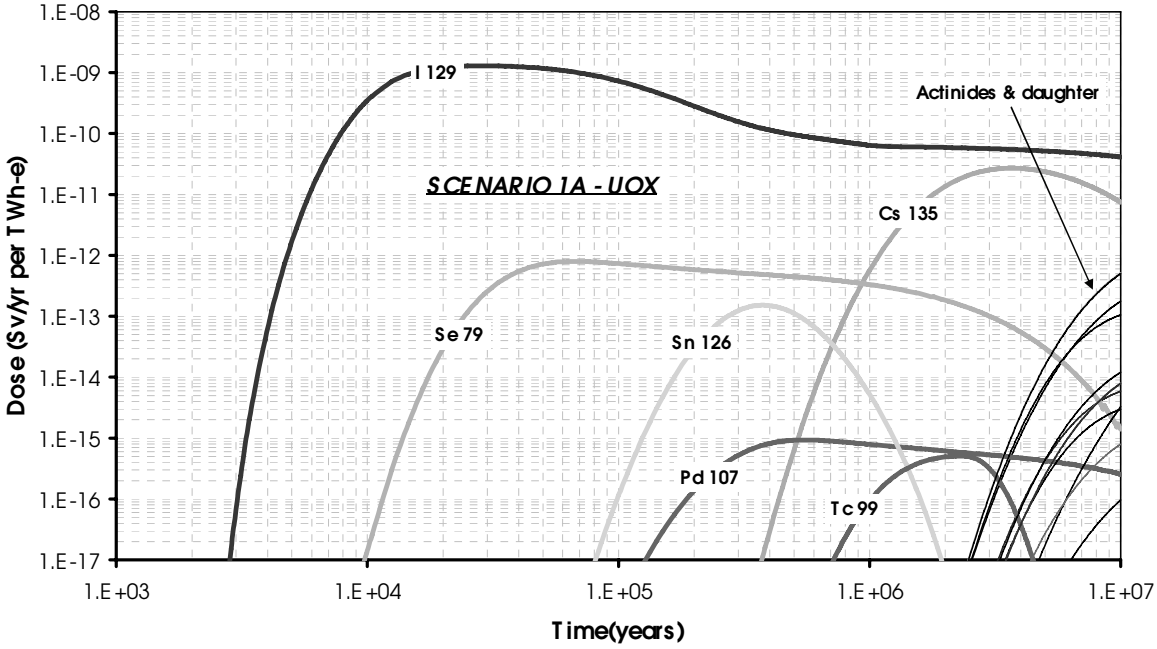


Figure I.6 shows the doses per radionuclide in scheme 1b due to the MOX fuel. The relevant radionuclides are the same than in scheme 1a, but the calculated doses are different. Difference in peak dose in Figure I.5 and Figure I.6 is just a factor 2. Although the inventory of fission products (per TWhe) in the MOX fuel is roughly one order of magnitude smaller than in the UOX, the matrix alteration rate of MOX fuel is 5 times greater than for UOX fuel (due to greater alpha activity).

Figure I.7 shows the doses per radionuclide in scheme 1b due to the vitrified waste. Since the inventory of ^{129}I is only 0.1% of the inventory in scheme 1a, doses due to ^{129}I are much smaller than in the cases of spent fuel. Doses due to the remainder fission products are greater than in scheme 1a because in the calculations the glass dissolves completely in 72 000 years while the UO_2 (and even the MOX) matrix lasts much more.

Figure I.6. Dose per radionuclide due to MOX fuel in scheme 1b

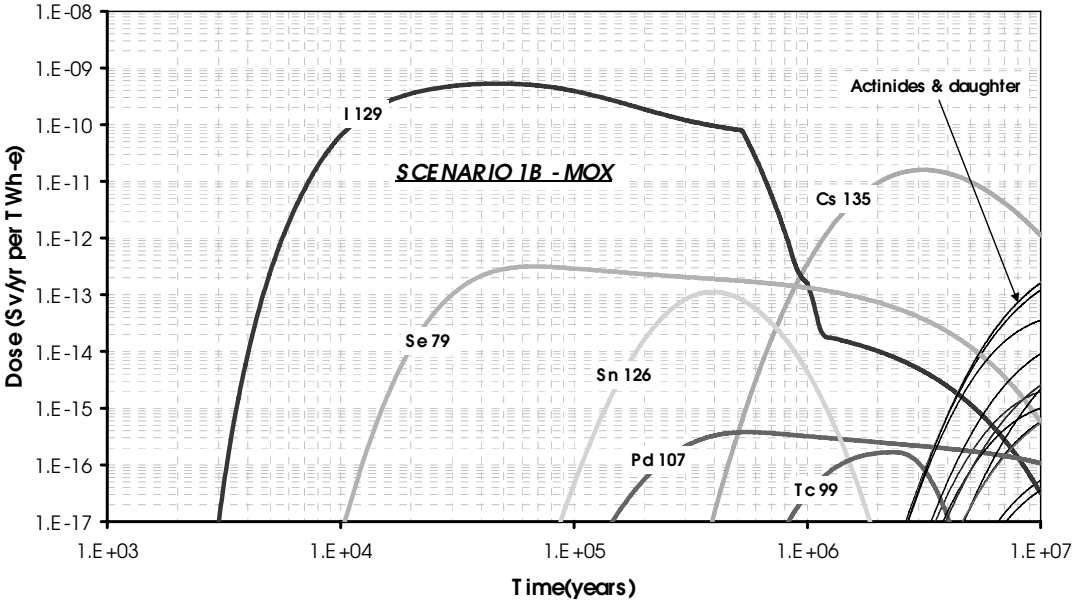
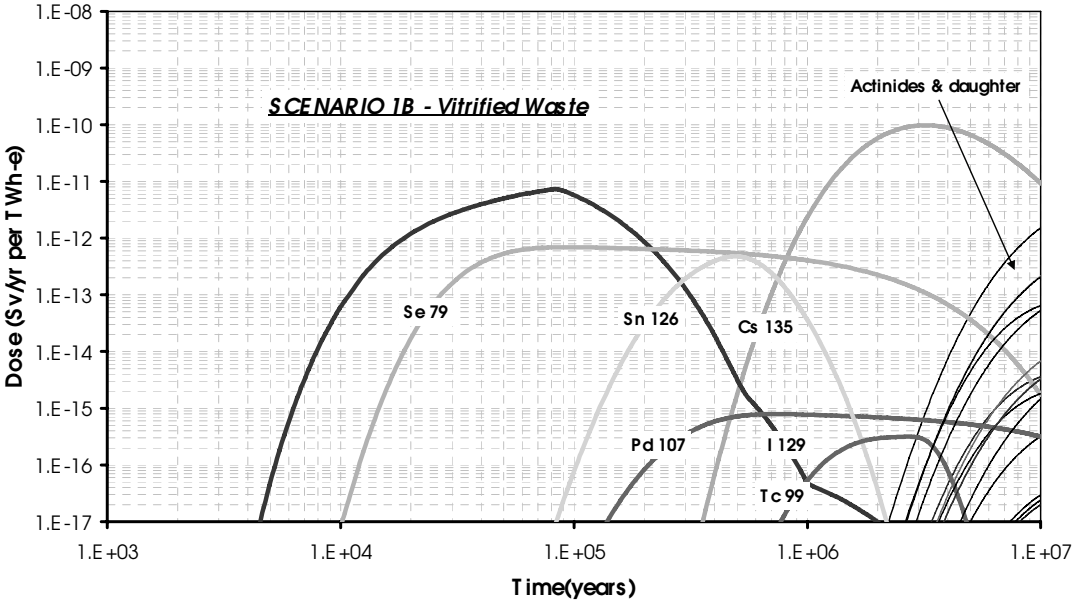


Figure I.7. Dose per radionuclide due to vitrified waste in scheme 1b



The doses per radionuclide in scheme 2a and 3cV1 are similar to those shown in Figure I.7 for vitrified waste in scheme 1b. This is a logical result because the glass matrix alteration model is the same in the 3 cases (1b-vitrified waste, 2a and 3cV1) and there are only small differences in the inventory of fission products (a factor 2 at most).

3.3.2 *Activity fluxes*

Figure I.8. Activity fluxes to the biosphere in scheme 1a

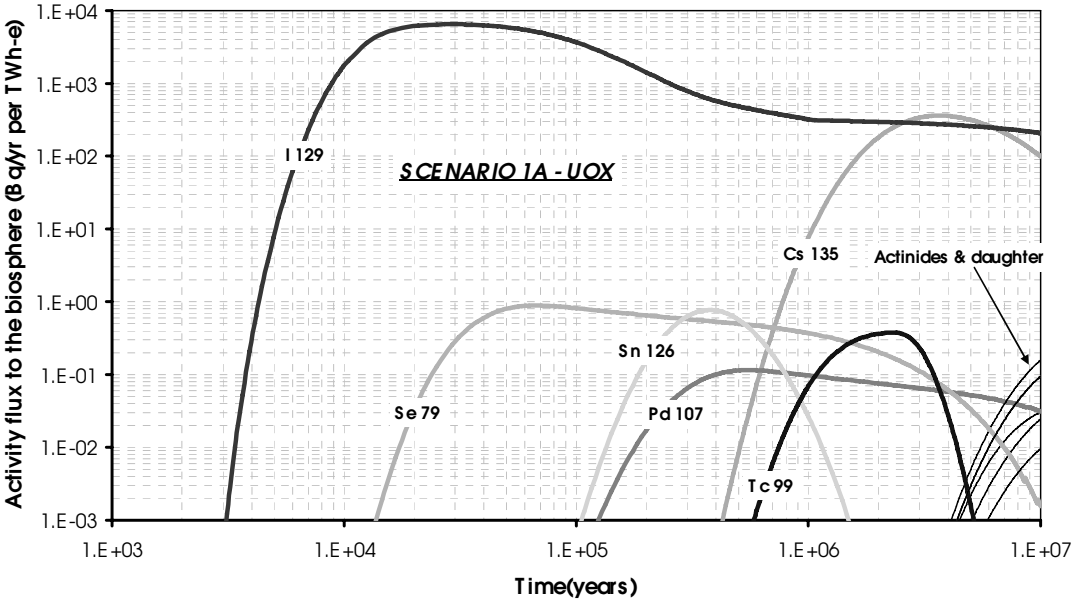


Figure I.8 shows the activity fluxes to the biosphere in scheme 1a (UOX). ^{129}I controls the fluxes and the contribution of ^{135}Cs becomes significant after 10^6 years. Fluxes of actinides and daughters remain below 1 Bq/yr up to 10 million years, because these radionuclides are strongly sorbed in the bentonite barrier and the geosphere.

Figure I.9 shows the activity fluxes to the biosphere in scheme 1b due to the MOX fuel. The relevant radionuclides are the same than in scheme 1a, but fluxes are different.

Figure I.10 shows the activity fluxes to the biosphere in scheme 1b due to the vitrified waste. Since the inventory of ^{129}I is only 0.1% of the inventory in scheme 1a, ^{129}I flux is much smaller than in the case of spent fuel. Fluxes of the remainder fission products are greater than in scheme 1a because in the calculations the glass dissolves completely in 72 000 years while the UO_2 (and even the MOX) matrix lasts much more.

Activity fluxes to the biosphere in scheme 2a and 3cV1 are similar to those shown in Figure I.10 for vitrified waste in scheme 1b. This is a logical result because the glass matrix alteration model is the same in the 3 cases (1b-vitrified waste, 2a and 3cV1) and there are only small differences in the inventory of fission products (a factor 2 at most).

Figure I.9. Activity fluxes to the biosphere due to MOX fuel in scheme 1b

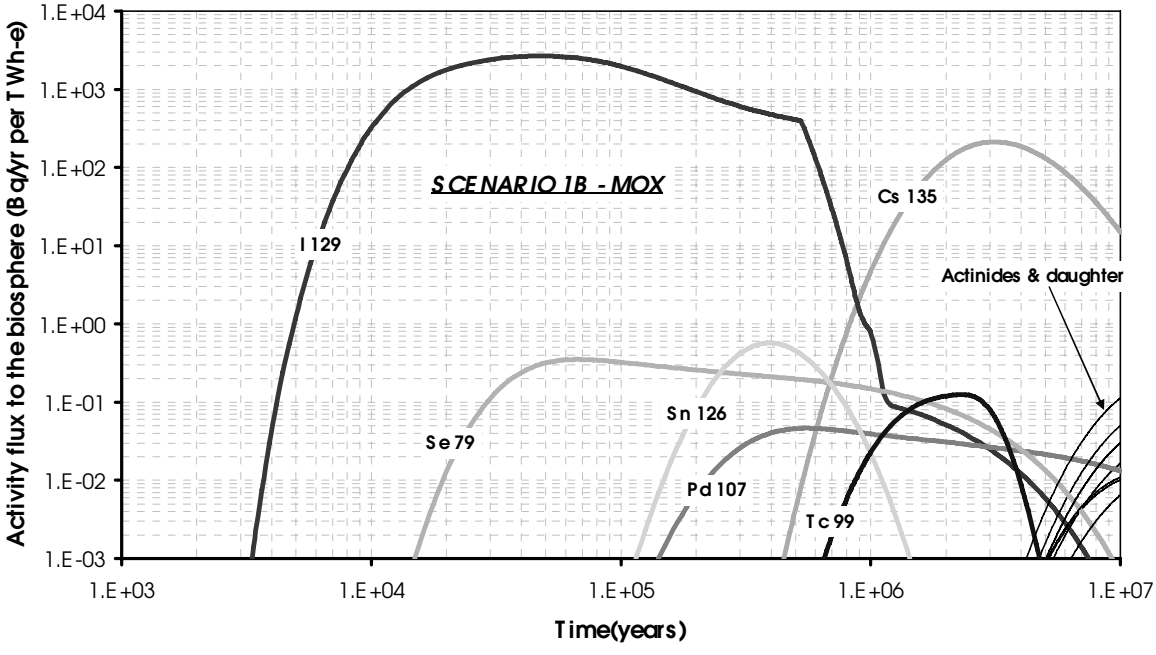
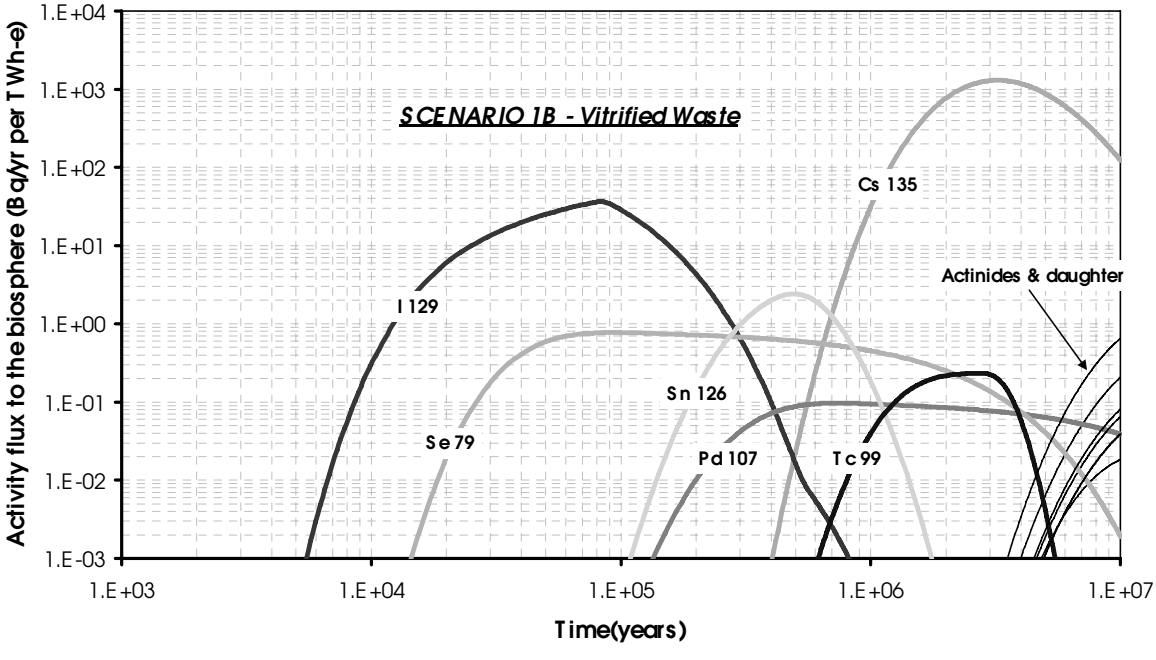
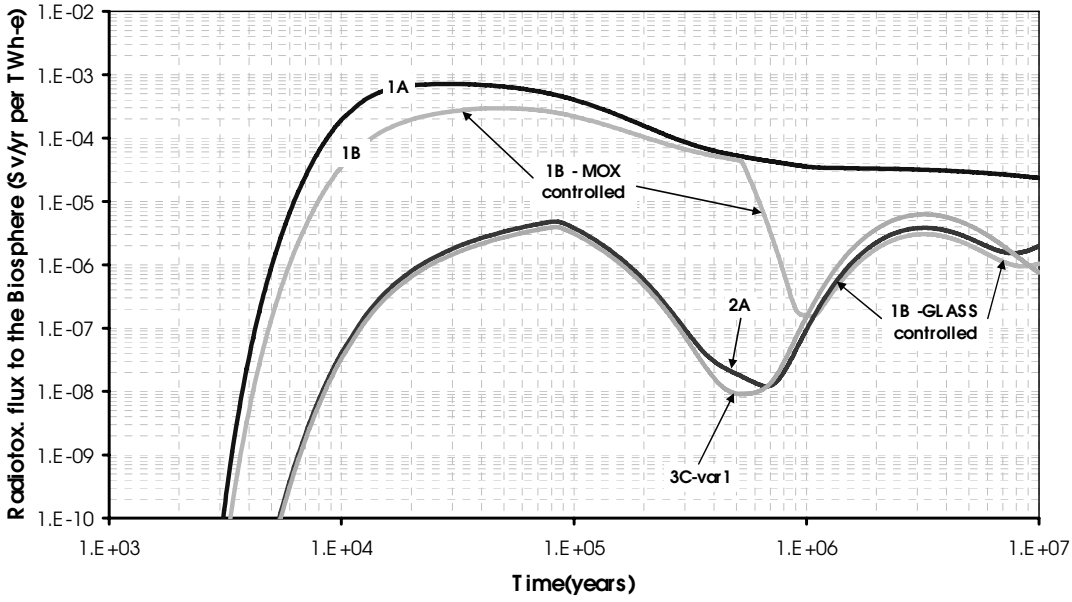


Figure I.10. Activity fluxes to the biosphere due to vitrified waste in scheme 1b



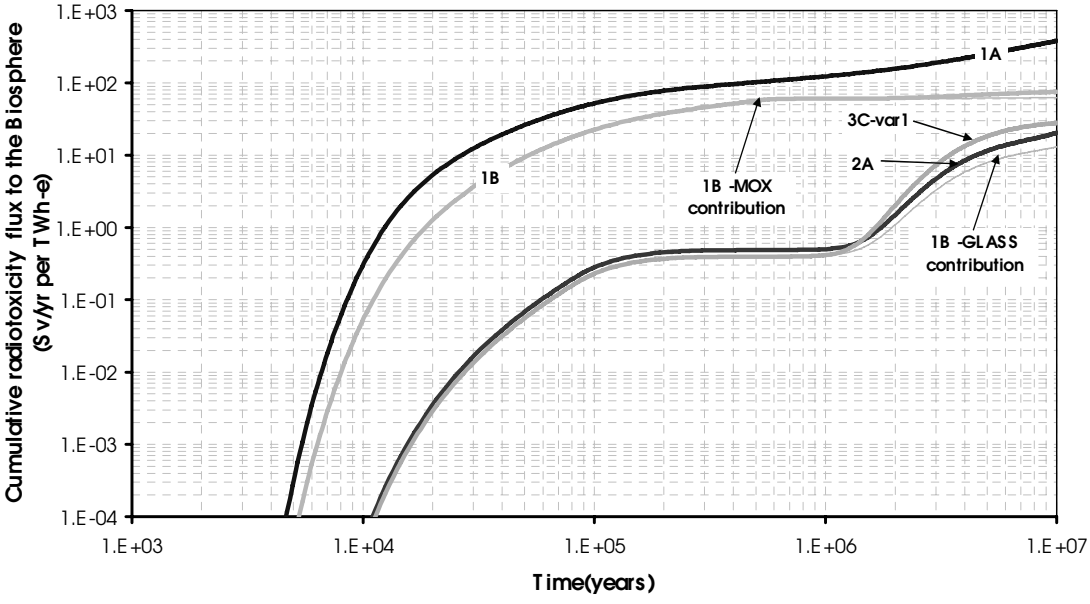
3.3.4 *Radiotoxicity fluxes to the biosphere*

Figure I.11. Radiotoxicity fluxes to the biosphere in the different schemes



3.3.4 *Cumulative radiotoxicity fluxes to the biosphere*

Figure I.12. Cumulative fluxes to the biosphere in the different schemes



Appendix J

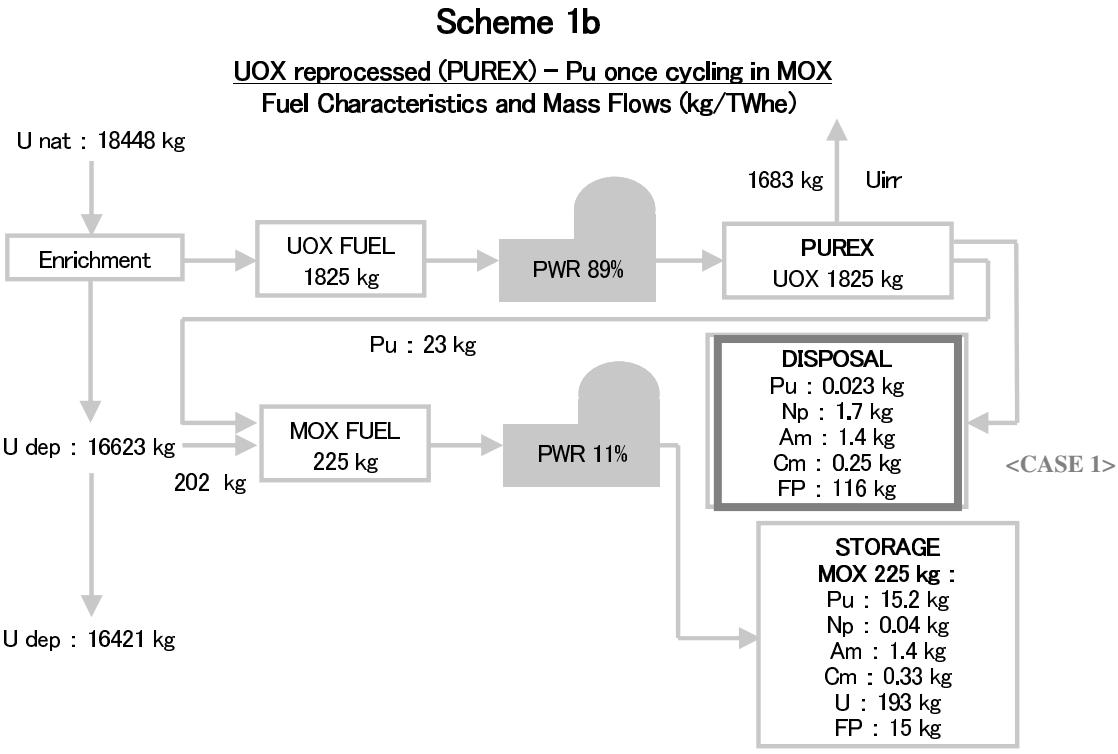
PERFORMANCE ASSESSMENT FOR DIFFERENT TYPES OF VITRIFIED HIGH-LEVEL RADIOACTIVE WASTE USING JNC H12 METHOD[1]

1. Introduction

This paper shows the results of radionuclide transport calculation for the following four types of vitrified high-level radioactive waste (HLW) proposed by NEA expert group:

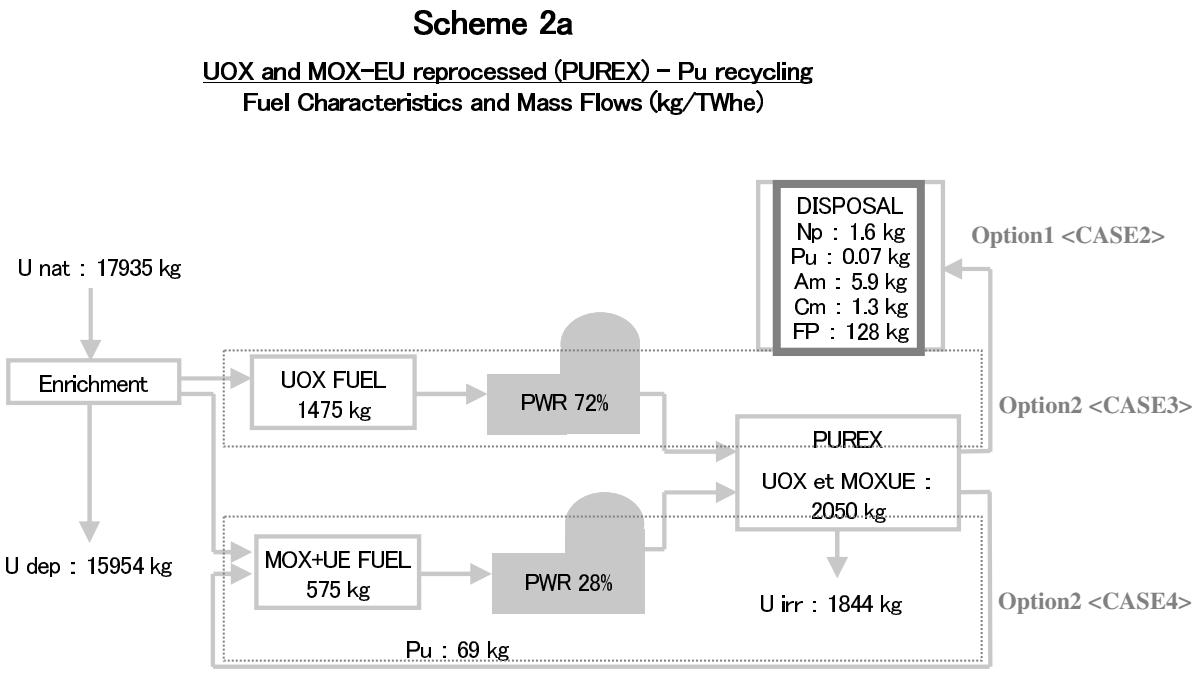
- Scheme 1b
  UOX vitrified HLW (Figure J.1.) ----------<CASE1>
- Scheme 2a: OPTION1
  UOX + MOX-UE vitrified HLW (Figure J.2) ---------<CASE2>
- Scheme 2a: OPTION2
  UOX vitrified HLW (Figure J.2) ----------<CASE3>
  MOX-UE vitrified HLW (Figure J.2) ----------<CASE4>

Figure J.1. Scheme 1b <CASE1>



1. This work was contributed by H. Makino, T. Ebina, M. Yui and T. Ikegami.

Figure J.2. Scheme 2a <CASE 2 – CASE4>



In this study, just inventories of radionuclides depending on the types of proposed HLW has been changed from H12 safety assessment, the other repository conditions (EBS, geosphere and biosphere), models and parameters have been set as the same as the H12 (Reference Case: see Chapter 2).

2. Overview of H12 safety assessment

2.1 Objectives, concept and Reference Case

The objectives of the safety assessment in H12[2] are:

- To establish a reliable safety assessment methodology that is applicable to a variety of geological environments in Japan and a range of repository designs.
- To assess the feasibility of safe geological disposal in Japan.
- To provide a technical basis for site selection and the development of safety criteria.

Here, it should be noted that the H12 has not focused on any actual site and any site has not been decided in Japan yet.

Radionuclides considered in H12 safety assessment are shown in Table J.1.

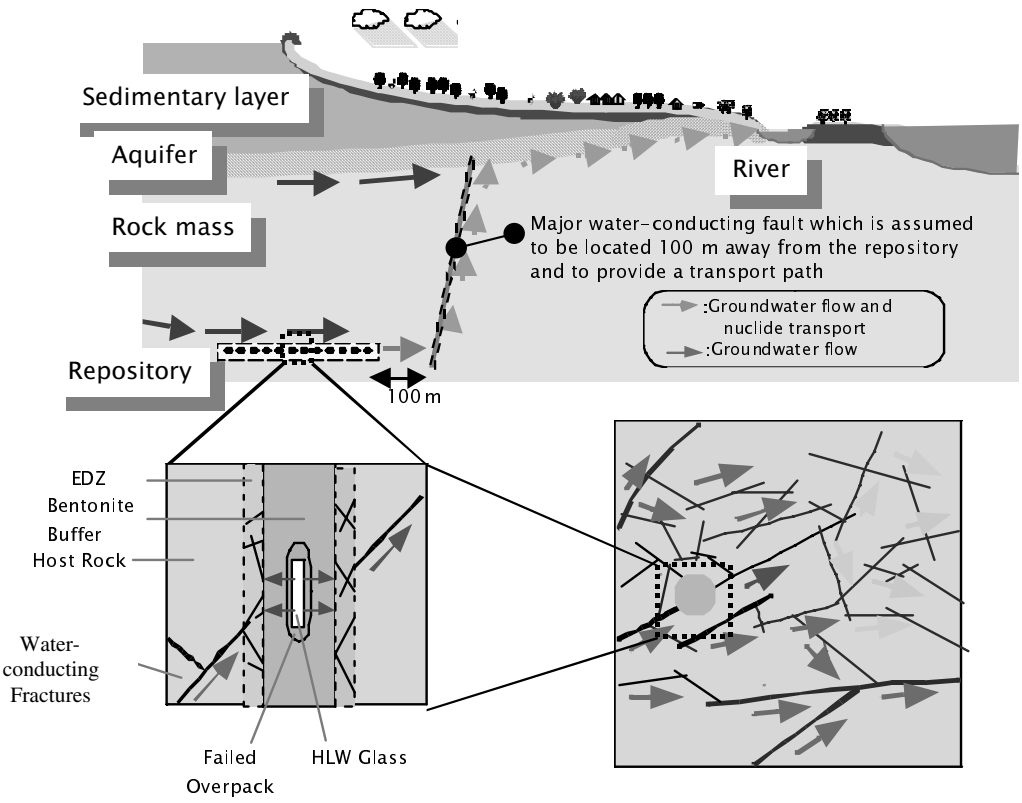
2. H12: Project to establish the scientific and technical basis for HLW disposal in Japan Supporting Report 3 Safety assessment of the geological disposal system, JNC TN1410 2000-004. JNC 2000.

The *Reference Case* is based on the reference disposal system (Table J.2) and Base Scenario[3], together with particular parameters and sets of model assumptions, and provides a point of comparison for the alternative cases in H12. An overview of features relevant to radionuclide release and transport in the Reference Case is given in Figure J.3.

Figure J.4 illustrates the chain of models used directly in the safety assessment or to produce key data, and the associated flow of information between models. This model chain is used for the Reference Case calculations.

To obtain releases from the whole repository, release rate to biosphere (calculated per canister) is multiplied by the number of waste packages in the repository. And, calculations are extended to times beyond the dose maximum.

Figure J.3. Nuclide transport pathway considered in the Reference Case



3. Base Scenario assumes:
   - The geological environment remains stable and the present-day geological conditions remain unchanged indefinitely into future.
   - Each barrier component performs as designed.
   - The current surface environment conditions continue throughout the timescale considered in the safety assessment.
   - Natural geological and climatic phenomena, initial defects and future human activities are regarded as "external events and processes", which are excluded from the Base Scenario.

Figure J.4. Model chain and associated flow of information in the H12 safety assessment

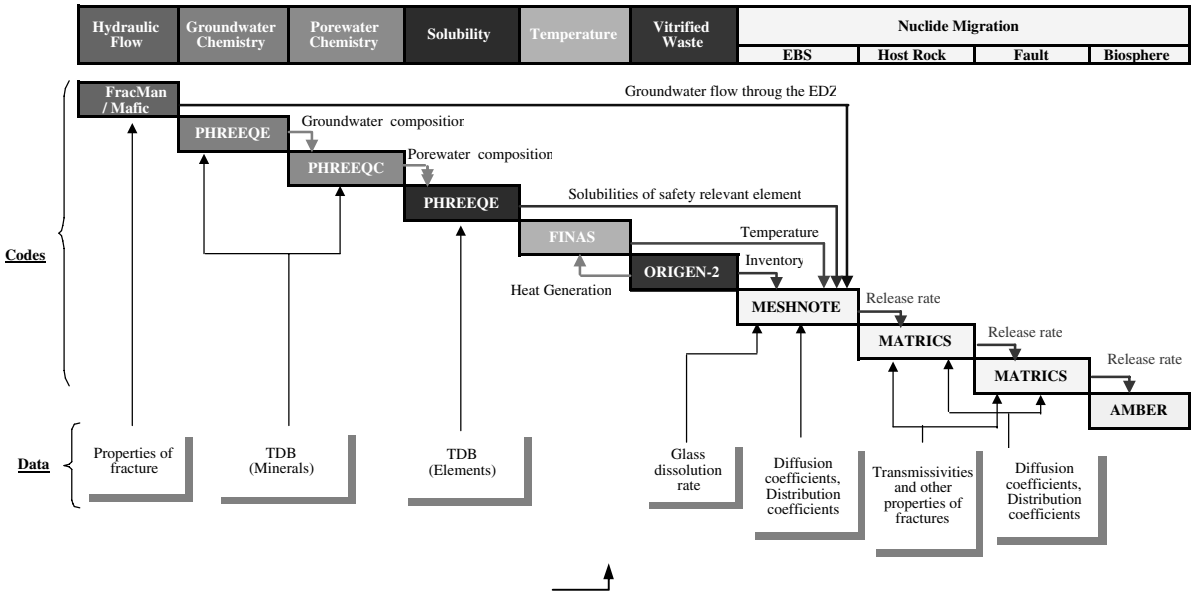


Table J.1. Safety-relevant radionuclides considered in the H12 safety assessment

Activation /Fission Products	^{151}Sm, ^{135}Cs, ^{126}Sn, ^{107}Pd, ^{99}Tc, ^{94}Nb, ^{93}Zr->^{93m}Nb, ^{79}Se
4N Series	^{240}Pu->^{236}U->^{232}Th
4N+1 Series	^{245}Cm->^{241}Pu->^{241}Am->^{237}Np->^{233}U->^{229}Th
4N+2 Series	^{246}Cm->^{242}Pu->^{238}U->^{234}U->^{230}Th->^{226}Ra->^{210}Pb ^{242m}Am->^{238}Pu
4N+3 Series	^{243}Am->^{239}Pu->^{235}U->^{231}Pa->^{227}Ac

Table J.2. System components considered in the H12 safety assessment

Geological environment /Surface environment	System components considered in safety assessment	Reference set of system components (Considered in Reference Case)
Topography	- Mountains - Hills - Plains (plateau, lowlands)	Plains (lowlands)
Groundwater	- Fresh groundwater - Saline groundwater	Fresh groundwater
Rock type	- Crystalline rock: felsic, mafic - Sedimentary rock (soft rock): sandstone, mudstone/tuff - Sedimentary rock (hard rock): sandstone, mudstone/tuff	Granite: Crystalline rock (felsic)
Geosphere-biosphere interface (GBI)	- Surface water (river, lake, etc.) - Well - Marine	Surface water (River)
Design	**System components considered in safety assessment**	**Reference System (Considered in Reference Case)**
Vitrified waste		Model vitrified waste inventory based on JNFL specifications (assuming 40 000 containers)
Canister	- Carbon steel - Composite canister: titanium-carbon steel, copper-carbon steel	Carbon steel: 0.19 m thick (with a minimum design lifetime of 1 000 years)
Buffer material	- Mixture of bentonite and silica sand	Bentonite: 70wt% + silica sand 30wt%, Dry density: 1.6 Mgm^{-3}, thickness 0.7 m
Repository depth	- Up to approx. 1 000 m for hard rock - Up to approx. 500 m for soft rock	1 000 m for hard rock
Temperature at disposal site	The temperature at the disposal site will reach about 55°C after 1 000 years in the case of hard rock, and about 40-50°C after 1 000 years in the case of soft rock.	After 1 000 years, constant at 60°C in the EBS and 45°C in the geosphere
Others	- Tunnel supports: Concrete supports, steel supports - Drift excavation technology: Blasting, Mechanical drilling (tunnel boring machine) - Backfill material: Mixture of bentonite and crushed rock with controlled particle size - Plugging material: Compacted clay, cement, rock, metal - Grouting material: Clay, cement	It is assumed that tunnels do not require supports in a hard rock environment

2.2 Description of the calculation

The Reference Case model assumptions for nuclide transport in H12 are:

<EBS>

- The canister fails one thousand years after closure of the repository.
- The glass waste form will be in contact with water immediately after canister failure. Nuclides in the glass are released congruently as the glass dissolves.
- Aqueous concentrations of nuclides are limited by elemental solubility, shared among the isotopes contained in the glass. Dissolution and precipitation of nuclides occur much faster than their (diffusive) transport and achieve local instantaneous equilibrium.
- Nuclides are transported through the buffer material by diffusion and are retarded by linear, reversible and instantaneous sorption onto the bentonite. The diffusion resistance of the canister corrosion products and their sorption capacity are ignored.
- Diffusion in the buffer material is modeled using a cylindrical radial co-ordinate system.
- Nuclides released from the buffer are mixed instantly and completely with groundwater flowing through the excavation disturbed zone (EDZ) of the repository tunnels. Sorption of nuclides in the EDZ is not considered and thus the total nuclide flux enters fractures intersecting the EDZ.

<Host rock>

- Nuclides are transported as solutes through channels within a network of fractures in the rock mass (granite).
- Hydrodynamic dispersion due to the complex network structure of the fractures is taken into account through a macroscopic dispersion length.
- Nuclides transported in channels can diffuse into the adjacent rock matrix and be retarded by sorption on matrix pore surfaces. Sorption is assumed to be linear, reversible and instantaneous. Sorption on fracture surfaces is not considered.
- Diffusion into fracture filling materials and unconnected fractures is not included.
- Transport within channels in the host rock is represented using a set of model pathways (the multi-pathway model), with each model pathway representing a set of channels of similar transmissivity.

<Major water-conducting fault (MWCF)>

- A repository would be located 100 m from a MWCF.
- Nuclides are transported upwards along the MWCF, traveling the shortest distance to the biosphere.
- All nuclides that have been transported through the undisturbed host rock enter a MWCF and flow upwards into an aquifer.
- Nuclides are transported through the MWCF as solutes by advection and dispersion.

- Nuclides in the fault can diffuse into the adjacent rock matrix and are retarded by sorption on matrix pore surfaces. Sorption is assumed to be linear, reversible and instantaneous. Sorption on surfaces of the fault is not considered.
- Diffusion and sorption onto fault gouge are not included.
- Transport in the MWCF is represented using a model pathway (the single-pathway model).

<Biosphere>

- Nuclides that have been transported through the host rock and the MWCF enter an aquifer near the ground surface and from there to the biosphere via the geosphere-biosphere interface (GBI). A river in a plain (lowlands) is selected as the Reference Case GBI.
- Nuclides that enter the aquifer are transported to the river without retardation due to sorption in the aquifer and then enter the river water directly, bypassing the river sediments.
- Subsequent distribution between compartments is modeled taking into account the transport processes, and various exposure modes and pathways are modeled.

3. Results of transport calculations for the proposed four types of HLW

Radionuclide transport for the four cases (CASE1-CASE4) just considering difference in inventory among the proposed four types of HLW have been calculated (other conditions are the same as H12 Reference Case), and the results are discussed using the following indicators:

- Release rate to biosphere.
- Radiotoxicity (converted from release rate using the ingestion dose coefficients for adults from ICRP72).
- Cumulative radiotoxicity.
- Dose (converted from using flux to dose conversion factors derived in H12 reference case).

Treatment of parameter values in this study that have been proposed by the NEA expert group as "general data" are summarised below:

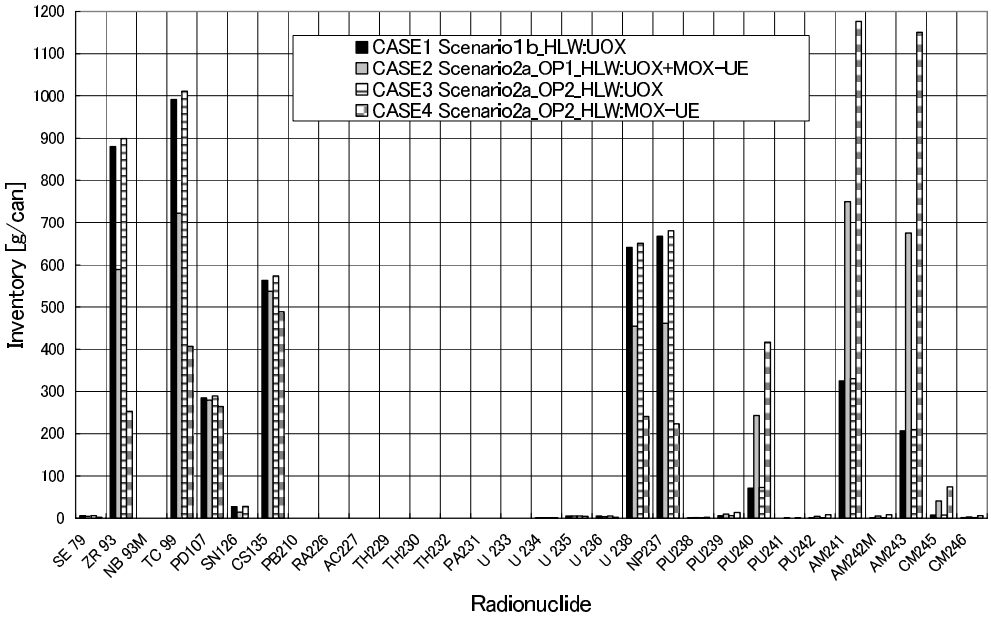
- Inventory for radionuclides shown in Table J.1 have been picked up from the proposed values. However, ^{94}Nb, ^{151}Sm and stable isotopes have not been considered due to lack of data in the proposed values.
- Zero remaining fraction of iodine (^{129}I) in HLW has been assumed as the same as the H12.
- 1 000 years of canister lifetime has been assumed as the same as the H12.
- About 70 000 years of glass matrix lifetime has been assumed as the same as the H12.

3.1 Differences in inventories for the four cases

Difference among high inventory radionuclides for the four cases (Figure J.5) shows:

- ^{135}Cs and ^{107}Pd have relatively small variation.
- ^{79}Se, ^{93}Zr, ^{99}Tc, ^{238}U, ^{237}Np, ^{241}Am and ^{243}Am have variation around or over twice.

Figure J.5. Differences in inventories of high inventory radionuclides for the four cases at disposal



3.2 Release rate to biosphere

Release rates to biosphere and radionuclides having high release rate obtained from the four cases are similar.

Figure J.6 shows release rate of each radionuclide in CASE1 as a representative.

Table J.3 shows radionuclides having high release rate in CASE1 to CASE4, respectively.

Figure J.6. Release rate of each radionuclide (CASE1)

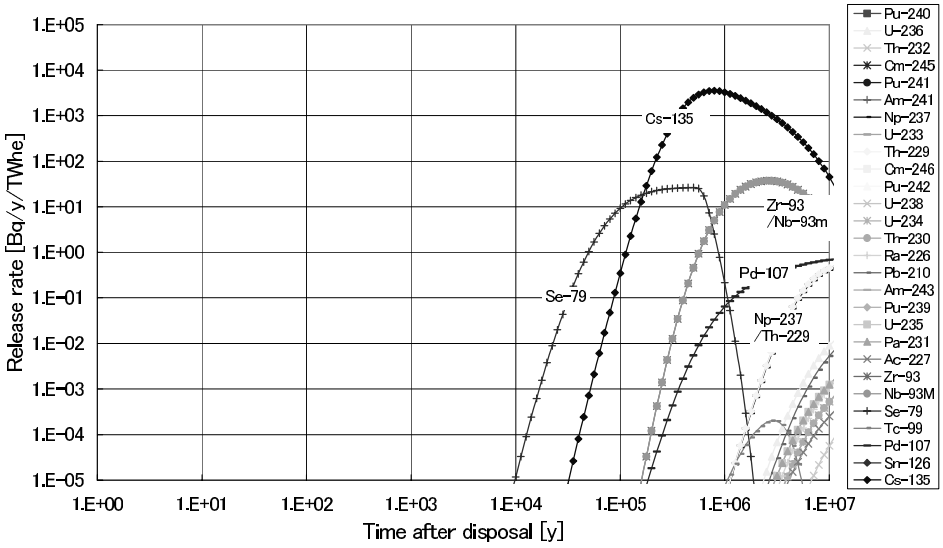


Table J.3. Radionuclides with high release rate

	- 10^5 y	10^5-10^7y
CASE1	1st: ^{79}Se	1st: ^{135}Cs 2nd: ^{93}Zr/^{93m}Nb
CASE2	1st: ^{79}Se	1st : ^{135}Cs 2nd: ^{93}Zr/^{93m}Nb
CASE3	1st: ^{79}Se	1st: ^{135}Cs 2nd: ^{93}Zr/^{93m}Nb
CASE4	1st: ^{79}Se	1st:^{135}Cs 2nd: ^{93}Zr/^{93m}Nb

3.3 Radiotoxicity

Total radiotoxicities of the four cases have very similar values within twice (Figure J.7).

Figure J.8 shows radiotoxicity for each radionuclide in CASE1 as a representative.

Table J.4 shows main contributor for the total radiotoxicity in CASE1 to CASE4, respectively.

Figure J.7. Comparison of total radiotoxicity

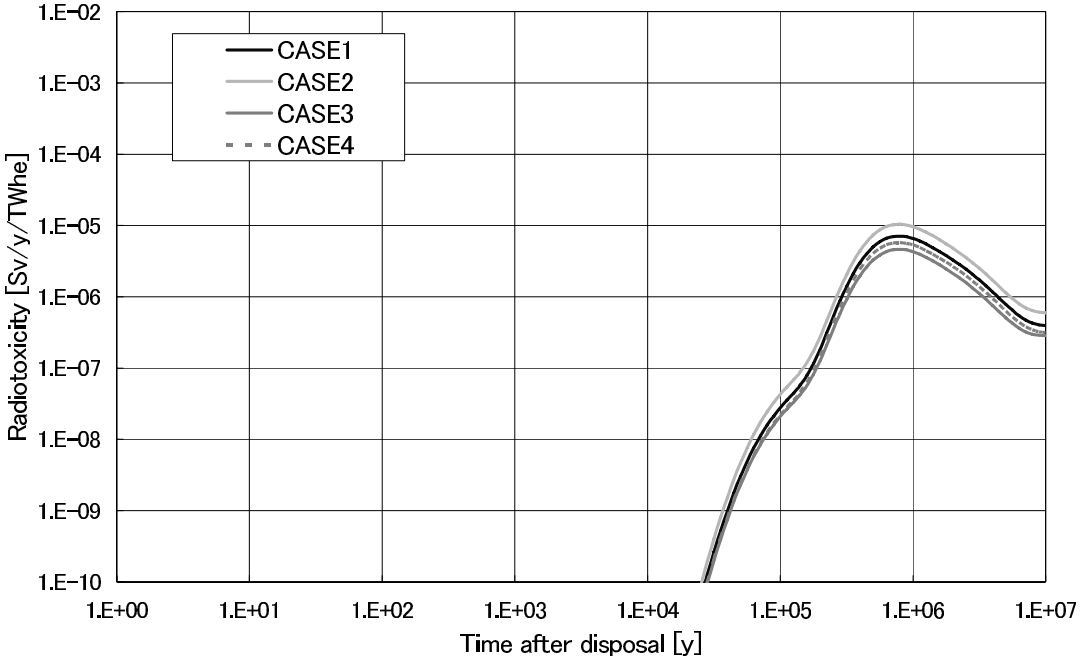


Figure J.8. Radiotoxicity for each radionuclide (CASE1)

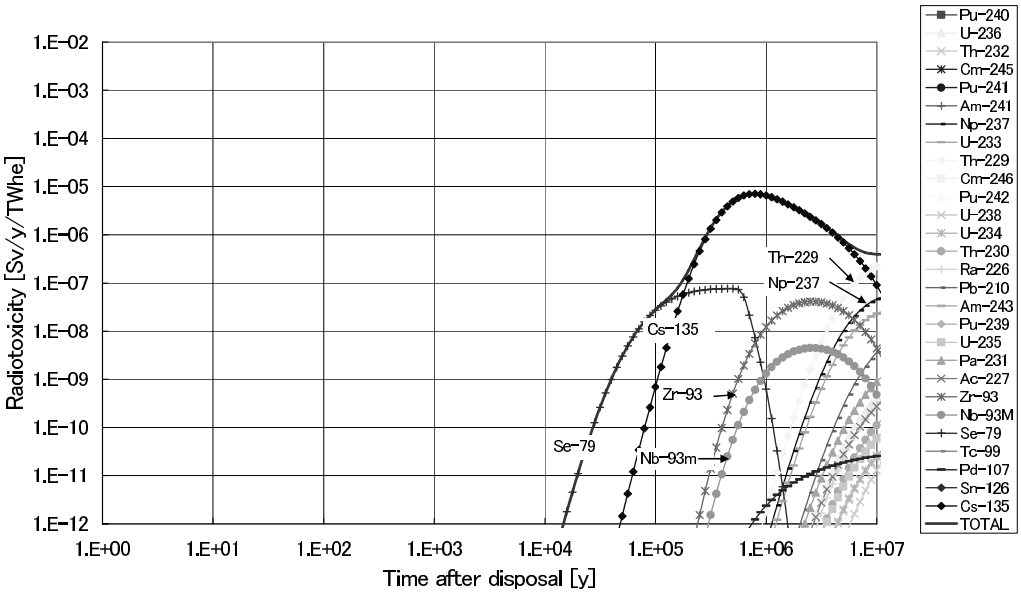


Table J.4. Main contributors for total radiotoxicity

	-10^5y	**10^5-10^7y**
CASE1	1st: ^{79}Se	1st: ^{135}**Cs** -> ^{229}Th
CASE2	1st: ^{79}Se	1st: ^{135}**Cs** -> ^{229}Th
CASE3	1st: ^{79}Se	1st: ^{135}**Cs** -> ^{229}Th
CASE4	1st: ^{79}Se	1st: ^{135}**Cs** -> ^{229}Th

Underlined radionuclide is a main contributor for maximum of total radiotoxicity.

3.4 Cumulative radiotoxicity

Table J.5 shows that cumulative "total radiotoxicity" have no significant difference between the four cases (maximum about factor of 2) and have peak during 10^6y-10^7y (if consider until 10^{10}y, two peaks can be found out; second peak is during 10^8y-10^9y).

Table J.5. Cumulative total radiotoxicity for time interval after disposal

Time interval [y] <after disposal>	Cumulative total radiotoxicity [Sv/Twhe]			
	CASE1	CASE2	CASE3	CASE4
～1.0E+3	<1E-100	<1E-100	<1E-100	<1E-100
1.0E+3～1.0E+4	5.3E-11	8.2E-11	4.2E-11	4.0E-11
1.0E+4～1.0E+5	8.4E-04	1.3E-03	6.7E-04	6.4E-04
1.0E+5～1.0E+6	4.1E+00	6.0E+00	3.3E+00	2.7E+00
1.0E+6～1.0E+7	1.1E+01	1.6E+01	9.1E+00	7.4E+00
1.0E+7～1.0E+8	*8.3E+00*	*1.2E+01*	*6.6E+00*	*6.0E+00*
1.0E+8～1.0E+9	*8.2E+01*	*1.4E+02*	*6.5E+01*	*6.8E+01*
1.0E+9～1.0E+10	*3.3E+01*	*4.6E+01*	*2.6E+01*	*1.7E+01*

3.5 Dose

Total doses of the four cases have very similar values, difference less than factor of 2, (Figure J.9).

Figure J.9 shows dose for each radionuclide in CASE1 as a representative.

Table J.6 shows main contributor for the total dose in CASE1 to CASE4, respectively.

Figure J.9. Comparison of total dose

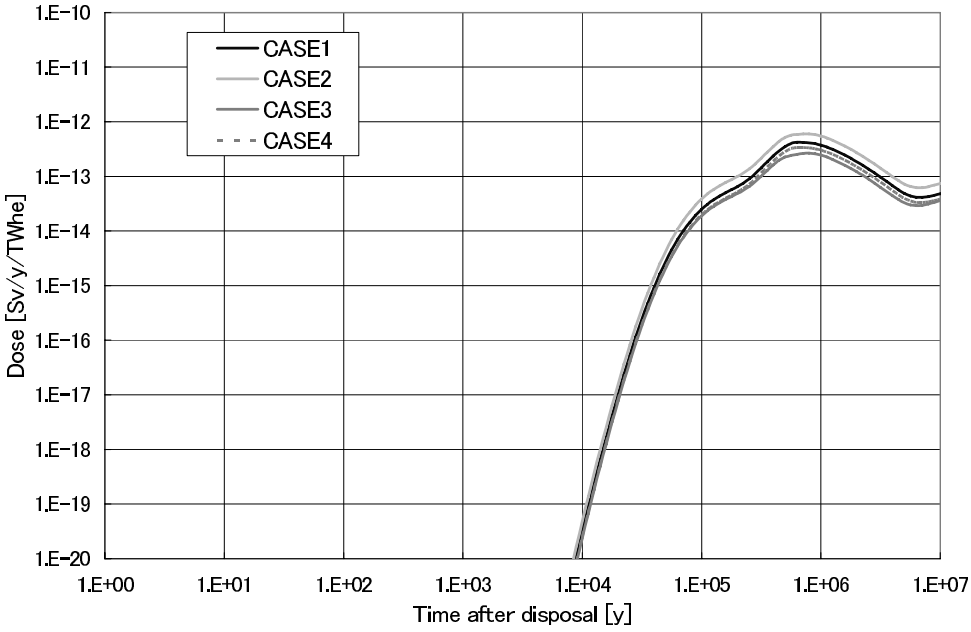


Figure J.10. Dose for each radionuclide (CASE1)

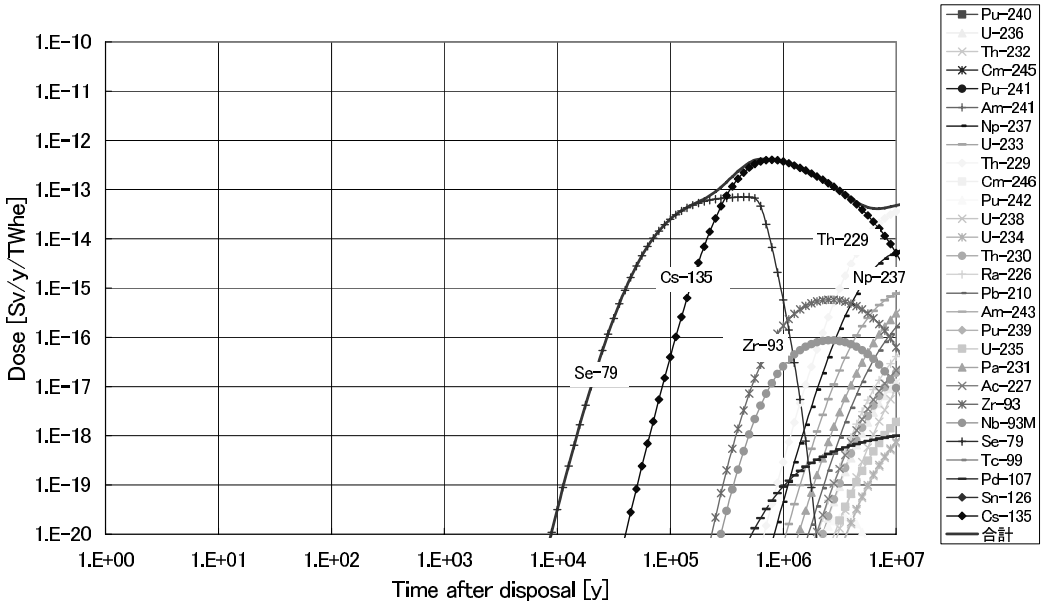


Table J.6. Main contributors for total dose

	- 10^5y	**10^5-10^7y**
CASE1	1st: ^{79}Se	1st: ^{135}**Cs** -> ^{229}Th
CASE2	1st: ^{79}Se	1st: ^{135}**Cs** -> ^{229}Th
CASE3	1st: ^{79}Se	1st: ^{135}**Cs** -> ^{229}Th
CASE4	1st: ^{79}Se	1st: ^{135}**Cs** -> ^{229}Th

Underlined radionuclide is a main contributor for maximum of total dose.

4. Discussion and conclusion

Though important differences in inventory, release rate, radiotoxicity and dose for each nuclide can be found between the four cases, main indicators, total radiotoxicity and total dose, differ less than by factor 2.

Main contributors of the total radiotoxicity and the total dose are ^{79}Se, ^{135}Cs and ^{229}Th. Maximum of the total radiotoxicity and the total dose are dominated by ^{135}Cs for all cases.

Table J.7 shows "ratios of inventory [g/TWhe] against CASE1" and "ratios of release rate [Bq/y/TWhe] against CASE1" for each case. They indicate that differences in release rate for the four cases could be explained roughly by the following four patterns:

- For ^{137}Cs (soluble) and ^{93}Zr (not reached its solubility limit), these two ratios are very similar and proportional impact could be concluded.
- For ^{107}Pd, ^{79}Se, ^{99}Tc, and ^{237}Np that reach solubility limit, these two ratios are different. Instead, new ratios for coefficient of can/TWhe are very similar with the ratios of release rate [Bq/y/TWhe]. This would be caused by the transaction from (4) (different cases have same release rate in Bq/y/can unit due to solubility limit in spite of difference in inventories) to (6) in the following transaction flow undergone in this study:

  (1) Inventory [g/TWhe].

  (2) Conversion from g/TWhe to g/can using coefficient of can/TWhe.

  (3) Inventory [g/can].

  (4) Release rate [Bq/y/can].

  (5) Conversion from Bq/y/can to Bq/y/TWhe using coefficient of can/TWhe.

  (6) Release rate [Bq/TWhe].

- While, ^{238}U reached solubility limit but the ratio of release rate is not consistent with the ratio of inventory and also the ratio for coefficient of can/TWhe. In this case, contribution of isotopes (e.g., ^{236}U) that have remarkable inventory itself and also have a parent having remarkable inventory (^{240}Pu) should be considered.
- For ^{229}Th, the ratio of release rate is the same as the ratio of ^{237}Np due to radioactive equilibrium.

Table J.7. Comparison of differences in inventory and release rate

	Ratio of inventory in unit of g/TWhe against CASE1			Ratio of release rate*1 in unit of Bq/y/TWhe against CASE1		
	CASE2/CASE1	CASE3/CASE1	CASE4/CASE1	CASE2/CASE1	CASE3/CASE1	CASE4/CASE1
Cs-135	1.466	0.808	0.658	1.465	0.808	0.655
Pd-107	1.511	0.808	0.703	1.540	0.795	0.756
Se-79	1.117	0.812	0.305	1.534	0.795	0.748
Zr-93	1.029	0.811	0.218	1.031	0.813	0.218
Tc-99	1.120	0.809	0.311	1.494	0.796	0.687
U-238	1.092	0.807	0.285	1.010	0.795	0.255
Np-237	1.063	0.809	0.254	1.567	0.796	0.783
Th-229	0.575	1.019	0.099	1.567	0.796	0.782
Am-241	3.551	0.808	2.743	*3	*3	*3
Am-243	5.032	0.807	4.224	*3	*3	*3
can/TWhe*2	1.541	0.795	0.757			

*1 : These ratios are common with Radiotoxicity and Dose
*2 : Ratio of can/TWhe coefficinet against CASE1
*3 : Release rates are cut off due to very low value (under 1E-30 [Mq/y/TWhe])

The total radiotoxicity and the total dose in unit of Sv/TWhe could be an indicator to compare difference among HLWs in the view point of radionuclide transport, but could not be extended to discussion in the view points of difference in system performance and/or safety margin against safety standard due to lack of an important information "total amount of HLW".

Importance of ^{135}Cs as main contributor would call attention about importance of reduction of FP's inventories from HLW rather than those of actinides.

Differences in coefficients of can/TWhe as well as differences in inventories would contribute to differences in release rates for especially radionuclides limited by solubility.

Finally, it could be concluded that all indicators are not significantly different among the four types of vitrified high-level radioactive waste proposed by NEA expert group.

Appendix K

HLW DISPOSAL – SALT CONCEPT

1. Introduction

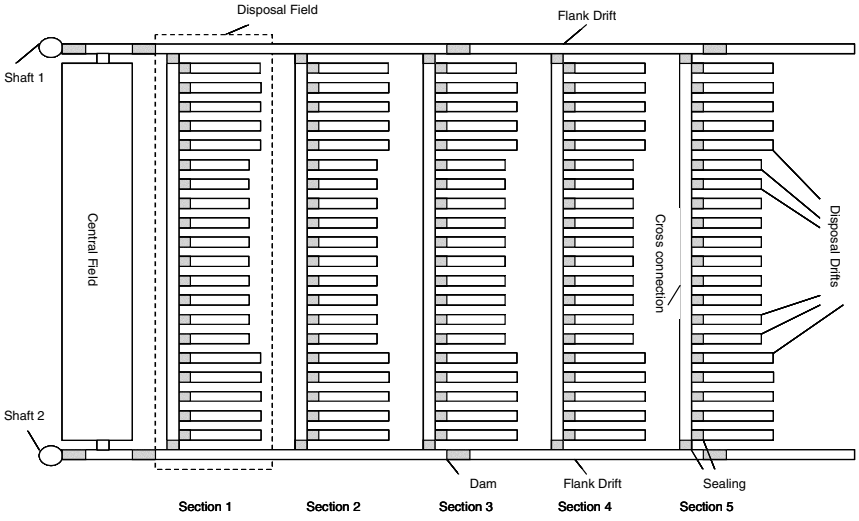
This appendix is the contribution of the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbh to the NEA/NDC working group on "Impact of Advanced Nuclear Fuel Cycle Options on Waste Management Policies". The central part of this appendix is the long term performance assessment calculation of a repository defined in the working group. Because the evolution of a repository in rock salt of the dome type differs in some important aspects from the behaviour of other systems a description of the disposal facility in rock salt and the behaviour of the rock salt facility are given.

During discussions in the working group some waste types were considered which gave rise to concerns with regard to disposability. To give the reader information which kind of problems could appear and which type of waste could cause such problems a paragraph is included to this appendix.

2. Description of the disposal facility in rock salt

The disposal facility is assumed to be located in rock salt of dome type at a depth of 870 m. The thickness of the overburden is close to 300 m. The waste is disposed of in horizontal drifts in a single story (single layer) arrangement. The disposal facility consists of the central field, two flank drifts and five cross connections, with 100 disposal drifts (Figure K.1). One cross connection and the adjacent 20 disposal drifts and the two segments of the flank drifts form one section. Access to the facility is provided by two shafts.

Figure K.1. Sketch of the disposal facility



The volumes and dimensions of the near field assemblies such as flank drifts, disposal drifts, shafts and central field can be found in [7].

2.1 Scheme description

For the consequence calculation the same brine intrusion scheme as in [8] was considered.

After termination of the disposal in the respective disposal section, the section will be backfilled and sealed to maintain it in a brine-free state. It is assumed that a layer of anhydrite is located in proximity to the final repository and that in response to mechanical stresses in the anhydrite, hydraulic connections between the aquifer and the repository are established and ground water may flow into the repository. The amount of the water inflow is controlled by the resistance of the flow paths. It is further assumed that the sealing of the shaft will fail.

In addition, the existence of brine pockets in the vicinity of the disposal drifts cannot be excluded. Mechanical stresses could trigger the release of brine from such pockets.

The brine intrusion scheme belongs to the altered evolution schemes as its probability of occurrence is considerably lower than one. It was necessary to switch to the disturbed evolution scheme because the normal evolution scheme does not give any radionuclide release. The normal (disposal) evolution scheme is not appropriate for an evaluation of fuel cycle schemes.

In the scheme, two competing processes will occur. On one hand, depending on the inflow location and the arrangement of the flow barriers, the repository will be filled with brine. On the other hand, convergence of the salt will cause a reduction of void spaces. Some drifts which are located close to the inflow or may have none or less effective barriers may get filled with brine. Drifts far away from the inflow location with many, good working barriers and increased salt creep may, in contrast, stay in a dry state. The convergence, which is the decrease of void volume per unit time, depends on the initial creep rate, the spatial temperature distribution, the pressure and the void volumes of the building components.

After brine intrusion into the disposal drifts corrosion of the disposal casks will start. This corrosion will lead to a failure of the cask and mobilisation of the waste. If the solubility limits of the respective nuclides in the fluid are exceeded, precipitation will occur. Nuclides go into solution if nuclide concentrations in the solution fall below solubility limits. Once a drift is completely filled the brine will be squeezed out. This squeezing is a consequence of the convergence of the void volumes.

We have to separate the behaviour of the disposal facility into two phases. During the inflow phase the competition between brine inflow and convergence of void spaces occurs. During the long term phase the squeezing out of the eventually contaminated brines is acting.

With progressing convergence, the contaminated brine will be pressed out from the repository into the aquifer and from there via the geosphere into the biosphere. During the transport the phenomena precipitation, dissolution, radioactive decay, adsorption and desorption will determine the nuclide inventory of the transported brine.

The transport in the geosphere is modeled by an aquifer with stationary water flow. Flow rates of the aquifer and dimensions are listed in [8]. The contaminated brine coming from the anhydride vein and the shaft is diluted in this permanent flow and transported to an access point. The biosphere is not

explicitly simulated. The dose is calculated by means of dose conversion factors. These dose conversion factors are listed in Table K.3 in the Chapter 9.

3. Calculation approach

The transport calculations in both the near-field and the far-field are performed using the MARNIE code. A description of the programme can be found in [8].

The calculation model used for the simulation is to a large extent the same as in [7]. In contrast to [7] a single story (single layer) disposal facility was assumed. For this performance assessment the most upper story of [7] is selected. The dimensions of the galleries and shafts as well as the filling and backfilling strategies can be found in [7].

Results of previous performance assessments (e.g. [8]) showed that the group of parameters related to the convergence of the salt is the most sensitive. The temperature field of the rock salt is one of these parameters.

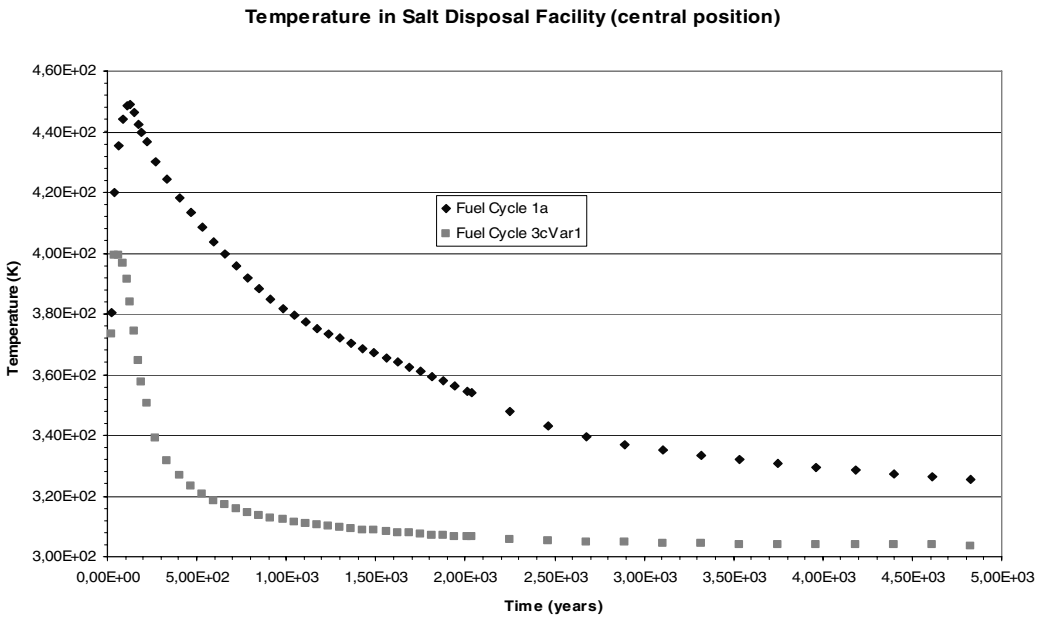
Because of different waste characteristics especially with regard to the thermal heat load of the waste to be disposed of the temperature field used in [7] could not adapted. Instead, temperature field calculations had to be carried out for each fuel cycle scheme. A well – founded estimation of the temperature is essential because the convergence depends to the power of 4 on the temperature.

The maximum temperature for a disposal in salt is expected to be limited by future regulations to 200°C at the contact surface between disposal cask and rock salt. Within the estimation of the temperature field a bulk temperature was calculated. The maximum bulk temperature has been fixed at 180°C. Coming from this maximum bulk temperature the upper boundary of the heat load for the disposal drifts has been determined. The maximum heat load for a short disposal drift has been estimated to be in the range of 130 to 135 kW while the maximum heat load for a long disposal drift has to be in the range of 140 to 145 kW.

Figure K.2 shows the evolution of the temperature (in Kelvin) at a central point of the disposal field for waste from the fuel cycle schemes 1a and 3cV1. The temperature for disposal of FC1a-waste reaches the maximum at 129 years with 450K (177°C) and is decreasing to 340K after 2 680 years. The temperature for disposal of FC3cV1-waste reaches the maximum at 64 years with 400K (127°C) and is decreasing rapidly to 340K after 593 years. The ambient temperature of the host rock is assumed to be 303K (30°C). The fast decrease of the temperature of the FC3cV1-waste is due to the fact that the heat load of this waste is decreasing drastically. At 50 years after unloading of the reactor the heat load is 518 W per TWhe and after 400 years the heat load is 1.7 W per TWhe.

The two completely different shapes of the temperature curves have a significant influence on the behaviour of the disposal facility in rock salt. For the fuel cycle 1a the salt creep will be drastically increased due to increasing temperature and stay for a very long time at a relatively high level. For fuel cycle 3cV1 the salt creep also will be increased but not to such a high extend than for fuel cycle 1a. After a relatively short time of increased salt creep the creep will decrease and remain for a long time at the level of unaffected salt.

Figure K.2. Evolution of temperature relevant for salt creep calculation



Because of similar heat load characteristics the shapes of the temperature curves of fuel cycles 1b and 2a are somehow similar to the temperature curve of fuel cycle 1a. So the four disposal facilities to be calculated can be grouped into two categories: “hot” disposal facilities (FC1a, FC1b and FC2a) and “cold” disposal facilities (FC3cV1).

The above mentioned different behaviour of the disposal facilities due to different temperature curves is characterised by the fact that the squeezing out of the contaminated brine in the “hot” disposal facilities stopped (due to salt convection in the respective disposal drift) after about 200 years. The phase for squeezing out the brine for the “cold” disposal facilities is lasting until about 50 000 years at a much lower squeezing rate.

3.1 Inventory of disposal facility

It is assumed that the used fuel elements are stored in POLLUX casks. These casks are described in [9]. For the vitrified waste disposal in the cask named POLLUX-HAW [5] is assumed. The POLLUX cask is assumed to be loaded with eight used fuel elements, the POLLUX-HAW cask with a waste equivalent corresponding to 6.9 tons of heavy metal initially loaded to the reactor.

It was not part of the exercise described here to investigate whether the waste coming from this 6.9 tons of heavy metal (tHM) could be taken up by the waste products loaded into the cask and if the cask itself is appropriate for the waste coming from the fast reactor (3cV1). The packing of the casks for the waste coming from 3cV1 is as dense as possible. There is only about 1 m left for the spacing between the casks. This is due to the constraints of the (minimum) heat load per disposal drift. These two points demonstrate that the disposal model used for this study is very close to the limits of design for waste coming from fuel cycle 3cV1. A necessary optimisation process (which was not possible in this work) would perhaps come to a smaller distance of the disposal drifts.

The fuel cycles 1a, 2a and 3cV1 produce only one type of high active waste. These are vitrified waste for 2a and 3cV1 and used fuel elements for fuel cycle 1a. Fuel cycle 1b generates used fuel elements as well as vitrified waste. The number of used fuel elements and vitrified waste packages for fuel cycle 1b has been estimated in the previous part of this report. The ratio between these two high level waste types is a characteristic of this fuel cycle. Due to programme and disposal model constraints the estimated ratio of waste types could not be established in one rock salt disposal facility. To overcome these constraints two separate disposal facilities for the respective waste have been introduced.

The mass load, heat load, number of casks per short and long drifts, the number of casks, mass of heavy metal and produced energy per disposal facility are listed in Table K.1. It is assumed that a cask for used fuel elements takes up eight fuel elements with a mass of 538.5 kgHM per element and the cask for vitrified waste the amount of waste coming from 6.9 t heavy metal initially loaded to the reactor.

Table K.1. Loading of the disposal facility

	Unit	Fuel cycle				
		1a	1b UOX	1b MOX	2a	3c
Ton of heavy metal per TWhe	tHM/TWhe	2,05	1,80	0,225	2,05	0,849
TWhe per cask	TWhe	2,10	3,78	19,00	3,366	8,12
Thermal load per TWhe	kW/TWhe	2,11	1,19	0,841	2,00	0,518
Thermal load of cask	kW	4,43	4,498	16,00	6,73	4,20
Number of casks long drift		33	32	9	21	34
Number of casks short drift		30	30	8	20	31
Total number of casks		3 150	3 100	850	2 050	3 250
Total mass of heavy metal in disposal facility *	tHM	13 570	21 390	5 865	14 150	22 425
Total produced energy of disposed waste	TWhe	6 615	11 718	16 275	6 900	26 930

* See text.

Table K.2 lists the total amount of waste given to the respective disposal facility. The yellow colour is standing for “hot” and the blue colour for “cold” disposal facilities. Radionuclides dominating the total dose are coloured orange (rows). This will be discussed later.

As described above two types of disposal facilities are identified (cold and hot). The temperature dependence of the behaviour of the disposal facility in the long term phase is so strong that only two calculation runs have to be performed. These two cases are a disposal facility for waste coming from fuel cycle 1a and fuel cycle 3cV1. A look at Table K.2 shows that the inventory of Cs for fuel cycle 2a is higher than for fuel cycle 1a. This can be a source for a different shape of the dose curve. It has been decided to introduce fuel cycle 2a as additional case for the performance assessment calculation for the disposal facility in salt. Fuel cycle 1b UOX and 1b MOX are not calculated.

Table K.2. Loaded radionuclides or radionuclide groups in the disposal facility (in mol)

Fuel cycle	1a	1b UOX	1b MOX	2a	3c
Th_Chain	5,78E+05	1,21E+04	7,68E+03	3,42E+04	6,13E+03
Np_Chain	1,69E+05	1,30E+05	9,01E+03	1,45E+05	1,35E+03
U_Ra_Chain	5,19E+07	8,22E+04	2,33E+06	5,42E+04	6,85E+04
Ac_Chain	8,17E+05	2,73E+04	7,89E+04	7,97E+04	1,34E+04
^{79}Se	1,41E+03	2,21E+03	6,44E+02	1,45E+03	4,37E+03
^{93}Zr	1,83E+05	2,87E+05	5,96E+03	1,74E+05	3,74E+05
^{99}Tc	1,94E+02	3,04E+05	9,04E+03	2,01E+05	5,15E+05
^{107}Pd	5,14E+04	8,08E+04	5,36E+03	7,19E+04	2,53E+05
^{126}Sn	4,18E+03	6,58E+03	3,20E+02	3,14E+03	1,93E+04
^{129}I	3,24E+04	5,10E+01	1,86E+03	3,60E+01	1,13E+02
^{135}Cs	8,06E+04	1,27E+05	6,27E+03	1,09E+05	6,88E+05
^{147}Sm	5,41E+03	8,53E+03	2,73E+02	6,02E+03	2,31E+04

4. Results of the calculations and discussion

The results of the performance assessment calculations for disposal facilities in salt for fuel cycles 1a, 2a and 3cV1 are shown in Figures K.3 to K.5.

The calculated dose rates are extremely low. The maximum is about eight orders of magnitude below the regulatory limit recently applicable in Germany (3 x 10^{-4} Sv). The differences between the three curves are smaller than two orders of magnitude.

Figure K.3. Calculated dose for waste from fuel cycle 1a

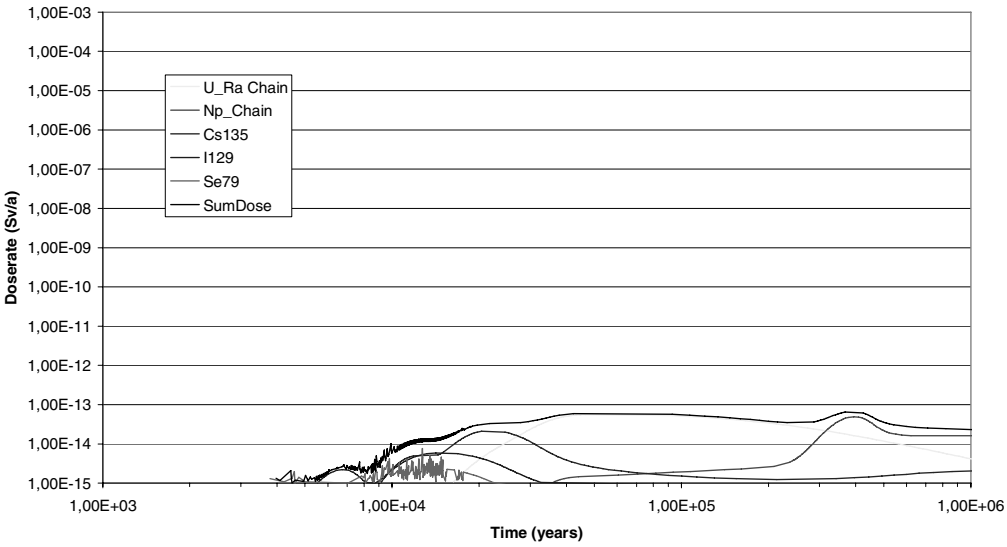


Figure K.4. Calculated dose for waste from fuel cycle 2a

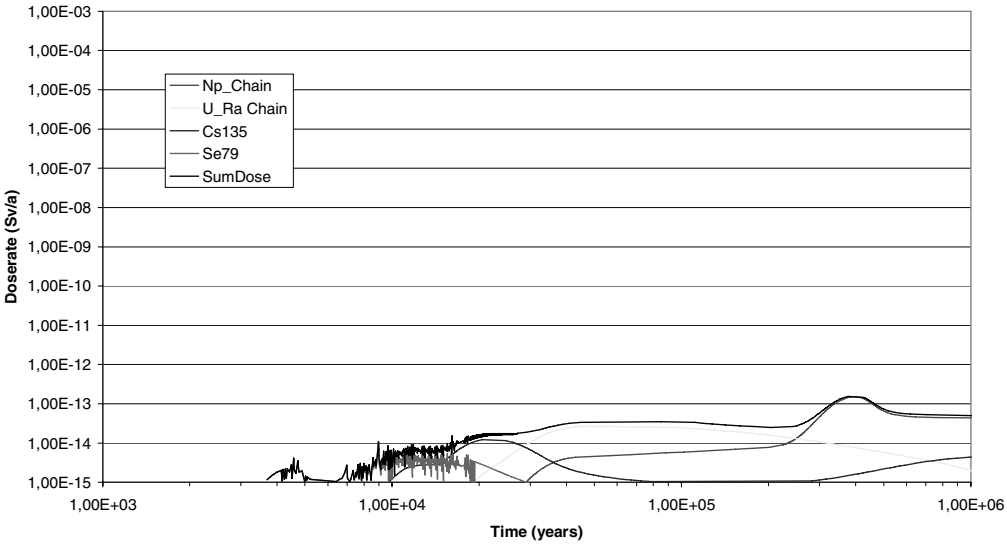
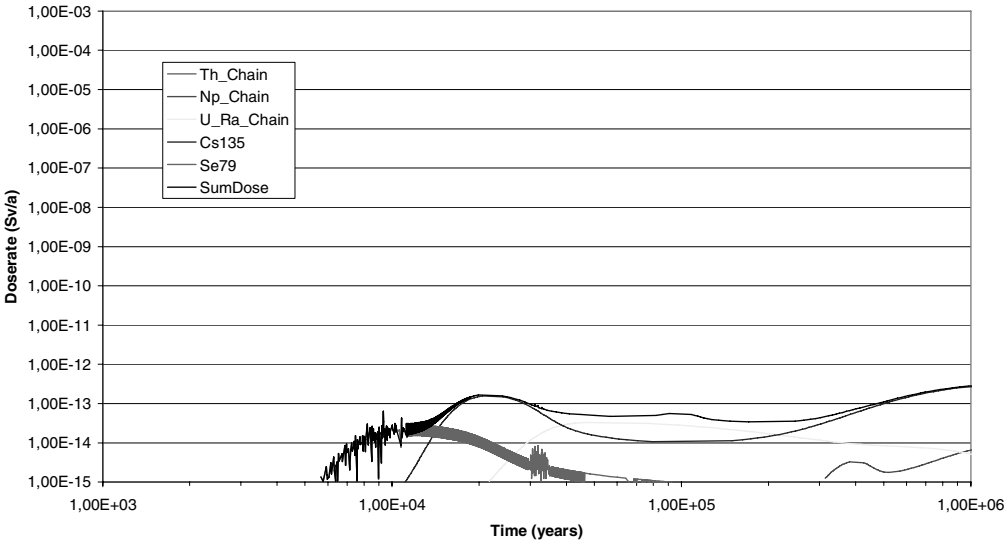


Figure K.5. Calculated dose for waste from fuel cycle 3cV1



5. Discussion

This discussion section is intended to clarify which role the presented performance assessment results might play with regard to the intent of the project, i.e. the evaluation and comparison of fuel cycles. First, for each cycle it needs to be investigated whether disposal of the resulting waste is feasible and can be implemented in a safe manner. The PA studies presented here are meant to explore this question. In addition, it might be desirable to identify advantages and drawbacks of the considered

fuel cycles. The question arises, whether the radiological dose calculation in the PA exercise described above might serve for such a comparison.

A central point for the performance assessment in salt is that the scheme under investigation is not the normal evolution but one alternative evolution scheme. It should be kept in mind that the normal evolution scheme leads to zero release. Some conditions for this alternative scheme have been chosen arbitrarily. One of these conditions is for example the point at which the anhydrite vein is connected to the disposal field. A change of this connection point can easily result in a difference in the calculated dose of some orders of magnitude.

Parameter uncertainties have been investigated in some previous studies (e.g. [8]). The parameters with the highest sensitivity to the result are related to the description of the salt convergence (e.g. initial convergence rate, temperature, pressure). Small uncertainties of these parameters can easily result in a difference in the calculated dose of some orders of magnitude.

Considering spread of dose due to possible uncertainties and the missing optimisation for fuel cycle 3cV1 disposal facility it is to conclude that the two orders of magnitude difference between the calculated dose curves is too small to give a preference to one fuel cycle. It can be concluded that the calculated doses can serve as indicators about whether safe disposal can be implemented for waste resulting from each cycle but can not serve as criteria to compare the cycles.

6. Relative importance of radionuclides

In many cases it is asked which radionuclides dominate the performance assessment calculation. This question is important for the identification of possible targets for P&T – strategies for a successful reduction of the total dose coming from disposal.

The dose curves for the performance assessment in salt show that some radionuclides which are relevant in other host rocks do not give a significant contribution to the total dose in this case (e.g. ^{93}Zr, ^{99}Tc, ^{107}Pd, ^{126}Sn, ^{147}Sm, Ac-Chain, Th-Chain). These radionuclides or radionuclide groups must be evaluated as less important with regard to disposal in rock salt.

Due to the increasing uncertainty with time, the performance assessment in salt the Np-peak (after 200 000 years) must be evaluated as less important. The Uranium-Radium decay chain is dominating the total dose of the performance assessment for salt over a very long time for all three calculated cases. The Uranium-Radium decay chain must be evaluated as important. The ^{238}U is a good candidate to be considered in P&T-strategies. But for a successful reduction of dose after disposal the ^{238}U amount in the disposal facility must decrease below the solubility limit. At the moment during reprocessing about 1% of the ^{238}U present in spent fuel elements comes to the waste. This residue in the vitrified waste must be reduced by several orders of magnitude before being below the target. Actually it is planned to reduce the residue to 0.1%, which is not sufficient for this requirement. It is expected that within the near future it will not be possible to decrease the amount below the solubility limit.

The last class of radionuclides that are considered as candidates for an effective dose reduction are ^{79}Se, ^{129}I and ^{135}Cs. In Figures K.3 to K.5 ^{79}Se and ^{135}Cs dominate the dose at an early time. This is the basis for this classification.

In the dose curve of fuel cycle 1a the dose coming from ^{129}I gives a significant contribution to the total dose at an early time. For the fuel cycles 2a and 3cV1 the ^{129}I does not appear in the dose curve

because during the fuel cycles 2a and 3cV1 the iodine is expected to be to a high extent spilled into the sea. Only 0.1% is assumed to come to the vitrified waste. If the dilution into the sea of ^{129}I will be stopped (due to future regulations) in future the dose of ^{129}I would be in the range of 10 to 20% of the calculated dose of ^{135}Cs. However, the overall picture of the calculated total dose doesn't change if iodine reprocessing assumption is applied or not.

7. Variations in the strategy of fuel cycle 3cV1

Another option for the waste treatment which has been discussed is to separate the heat producing fission products Sr and Cs (element wise) from the waste, store them for about 300 years and dispose them of afterwards. This is seen as a possibility to decrease the heat load of the waste and increase the capacity of the disposal facility.

Contributing to this discussion some aspects of the point of view of rock salt disposal have to be mentioned. First ^{135}Cs is the radionuclide which contributes to a very high degree to the total dose. This means that if this radionuclide is coming back to disposal high attention has to be paid to appropriate disposal conditions and a careful disposal strategy. A good strategy would be to mix this ^{135}Cs with high heat producing waste, which guarantees a fast salt creep and void volume reduction. This means that Cs has to be separated from the hot waste, stored for a long time and somehow mixed (or stored together) with hot waste afterwards.

The separation of heat producing radionuclides could be one strategy for decreasing the needed disposal volume. A second strategy could be a prolonged interim storage of the waste coming from the fuel cycle 3cV1. It is expected to influence the behaviour of the rock salt disposal facility in the same way as the separation of heat producing radionuclides and does not need a radionuclide separation.

Compared to other rock types heat producers in a rock salt disposal facility are categorised in a different way. For a maximum salt creep it is necessary to be as close as possible to the temperature limit. In a possible optimisation procedure a rock salt disposal facility would be optimised towards a high temperature. If the heat load of possible future waste streams (for a rock salt disposal facility) is drastically reduced the number of casks per disposal volume has to be decreased to reach the necessary heat generation rate per volume. In this case it is expected if the head load decreases by a factor close to 100 the number of casks have to be decrease by a factor of 100 or even more.

8. Possible problems of waste types from advanced fuel cycles

Advanced nuclear fuel cycles may produce nuclear waste types with different characteristics with regard to their physical and chemical form. The physical and chemical waste form must be investigated carefully to guaranty feasibility of disposal. One of the possible problems with regard to the physical waste form could be an increased radiation level of the waste which can cause rock salt destruction.

Rock salt destruction is a complicated issue and has up to now not completely been understood. The next abstract summarises the knowledge about the rock salt destruction mechanism.

9. Mechanism of rock salt destruction

The following explanation to the effect of radiolytic destruction of rock salt is an extraction (translation) from a GRS report [1]. It summarises the research activities until 1997 and makes an assessment of the possible consequences of an irradiation of rock salt in a disposal facility. Research activities from 1997 to 2002 have been reviewed in [2].

Salt rock is subject to radiolysis if it is exposed to radiation. The host rock in the vicinity of the waste is exposed to γ–rays and neutrons emitted by the waste. The absorption of neutrons leads to the activation of nuclides, mainly chlorine-35. The decay of these activated nuclides results in over 90% of high-energy γ–radiation; which are subsequently thermalised.

The absorption of thermal γ–radiation in rock salt leads to radiation damage of the crystal lattice via a complex mechanism. At high γ–irradiation doses (> 10^6 Gy) chlorine gas (Cl_2) and sodium in metallic form (colloidal Na) are generated. For the disposal of HAW (vitrified waste from reprocessing) this radiolysis process is a local effect limited to an area of approximately 50 cm around the disposal borehole. The generation of chemically very reactive irradiation products and the destruction of rock salt may have a significant influence on the safety of the disposal of high active, heat generating waste. In case of a fast spontaneous back reaction of the molecular irradiation products a temperature increase followed by thermo-mechanical consequences seems possible.

Because of the long timeframes it is impossible to investigate radiation damage formation in the laboratory under conditions that fully represent disposal specific boundary conditions. Owing to the complex radiolysis mechanism a simple extrapolation of experimental results is misleading. As a consequence, qualitative and quantitative predictions of the radiation damage processes of rock salt in a repository must be based on model calculations.

Important parameters for the model calculations are the γ–dose rate and the temperature of the rock salt, both of which have an influence on the generation rate of the (sodium) colloids. For example, at any given dose rate colloid generation is only possible in a certain temperature range. Also, the colloid generation rate is lower at higher dose rates than at lower ones. This means that temperature and radiation information (total dose, dose rates) must be considered together in order to assess the damaging potential of heavily radiating radioactive waste.

Model calculations taking into account these parameters and their temporal and spatial changes in a repository have been carried out in order to estimate the effect of rock salt irradiation by (conventional) vitrified waste coming from reprocessing of LWR fuel [1]. Radiation doses are in the order of magnitude of 10^8 Gy and temperatures are in the range of up to 200°C. The waste is assumed to be disposed of in boreholes which are arranged in a rectangular field with a distance of 50 m between two adjacent boreholes. The GRS report [1] concludes that in this disposal situation severe consequences due to rock salt irradiation are not to be expected. This conclusion is only valid with above mentioned parameters.

The radiation damage formation in rock salt that is caused by gamma- and neutron radiation is still a matter of research. It has reached a point at which the principal mechanisms have been understood and at which it is possible to calculate to a certain extent a degree of damage. It is far away from the point at which clear acceptance criteria can be formulated.

10. Results of the performance assessment for the rock salt disposal facility

The normal evolution scheme results in zero release. For the purpose of this study an alternative evolution scheme has been considered. Performance assessment calculations for a rock salt disposal facility have been done for waste coming from fuel cycles 1a, 2a and 3cV1. The calculated doses for these alternative evolution schemes are considerably small. This means that the dose is more than seven orders of magnitude below the regulatory limit. The difference between the dose curves is smaller than two orders of magnitude. Because uncertainties are much higher than the difference of the calculated dose curves no preference could be given to one of the fuel cycles.

A more important aspect of the disposal of waste of different fuel cycle is the evaluation of the feasibility of disposal. For fuel cycles 1a, 1b (UOX and MOX) and 2a the feasibility of disposal is given. The feasibility of disposal for fuel cycle 3cV1 is not investigated to the full extent but is likely to be given. New waste types of advanced fuel cycles have to be investigated carefully before a feasibility of disposal could be attested. The mechanism of rock salt destruction could be problematic. Rock salt destruction is not only related to the radiation level of the waste as single value but it depends on a combination of increased temperature and radiation level.

The analysis of the calculated dose rates results in a preference list of important radionuclides as targets for successful P&T strategies. This list assigns the highest priority to the fission products ^{79}Se, ^{129}I and ^{135}Cs.

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[9] EC (2000), *Spent fuel disposal Performance Assessment (SPA project)*, European Commission, Community Research, Nuclear Science and Technology, Project Report, EUR 19132 EN.

Parameter Tables

Table K.3. Dose conversion factors used in the PA for salt

Radionuclide	Conversion factor (Sv/y)/(Bq/m^3)
^{248}Cm	1,40E-06
^{244}Pu	2,50E-06
^{240}Pu	2,20E-06
^{236}U	2,20E-07
^{232}Th	1,60E-05
^{232}U	6,20E-06
^{245}Cm	3,00E-06
^{241}Am	2,70E-06
^{237}Np	6,20E-06
^{233}U	3,00E-07
^{229}Th	5,40E-06
^{246}Cm	3,10E-06
^{242}Pu	2,00E-06
^{238}U	3,10E-07
^{242m}Am	2,60E-06
^{238}Pu	2,00E-06
^{234}U	2,40E-07
^{230}Th	2,80E-06
^{226}Ra	1,50E-05
^{247}Cm	3,10E-06
^{243}Am	3,50E-06
^{239}Pu	2,20E-06
^{235}U	9,50E-07
^{231}Pa	1,40E-05
^{79}Se	2,30E-07
^{99}Tc	4,90E-09
^{129}I	3,70E-07
^{135}Cs	8,60E-08

Table K.4. Solubility limits (PA Salt)

Element	Solubility limit
	mol/L
Cm	1,00E-04
Am	1,00E-04
Pu	1,00E-06
Np	1,00E-05
U	1,00E-04
Th	1,00E-06
Ra	1,00E-05
Pa	1,00E-06
Se	1,00E-04
Tc	1,00E-04
I	1,00E-00
Cs	1,00E-00

Table K.5. K_d value and retardation coefficient

Element	K_d	R
	m^3/kg	
Cm	1	4 801
Am	1	4 801
Pu	1	4 801
Np	3,00E-02	145
U	2,00E-03	10,6
Th	3,00E-01	1 441
Ra	9,00E-04	5,32
Pa	1,00E+00	4 801
Se	3,00E-04	2,44
Tc	7,00E-03	34,6
I	5,00E-04	3,4
Cs	1,00E-03	5,8

Retardation coefficient:

$$R = 1 + \frac{(1-n)}{n} \rho \, K_d$$

with

n = 0,2 porosity
ρ = 1 200 kg/m^3 rock salt density
K_d= K_d parameter.

Table K.6. Estimation of volume of disposal facility in rock salt

Part of disposal facility	Height (m)	Width (m)	Length (m)	Volume (m^3)	Quantum of parts	Volume total (m^3)
Shaft	–	6,65	870,0	30 217,00	2	60 434
Central field				108 867,00	1	108 867
Flank drift	3,8	7,20	886,0	24 241,00	2	48 482
Cross connection	3,8	7,20	322,5	8 823,60	5	44 118
Disposal drift (long)	3,3	4,70	229,5	3 559,55	10*5	177 977
Disposal drift (short)	3,3	4,70	219,5	3 404,45	10*5	170 222
Sum						610 100

Appendix L

TECHNICAL PARAMETERS AND UNIT COSTS

1. Waste generation parameters

1.1 Front-end fuel cycle

Processes		Waste type	Unit	Nominal value	Source
Conversion		LILW-SL	m^3/tU		
Enrichment		LILW-SL	m^3/kSWU	0.04	[1]
Fuel fabrication	UOX	LILW-SL	m^3/tHM	0.6	same as MOX
		LILW-LL	m^3/tHM	–	
	MOX	LILW-SL	m^3/tHM	0.6	[2]
		LILW-LL	m^3/tHM	0.6	[2]
	Am target	LILW-SL	m^3/tHM	0.6	same as MOX
		LILW-LL	m^3/tHM	0.6	same as MOX
	Other fuels	LILW-SL	m^3/tHM	0.6	same as MOX
		LILW-LL	m^3/tHM	0.6	same as MOX

1.2 Reactor operation

Reactor types	Waste type	Unit	Nominal value	Source
PWR	LILW-SL	m^3/TWhe	12.8	IAEA
	LILW-LL	m^3/TWhe	0.3	IAEA
CANDU	LILW-SL	m^3/TWhe	16.0	IAEA
	LILW-LL	m^3/TWhe	2.6	IAEA
FR	LILW-SL	m^3/TWhe	6.4	IAEA
	LILW-LL	m^3/TWhe	0.3	IAEA
LMFR, GCFR	LILW-SL	m^3/TWhe	6.4	same as FR
	LILW-LL	m^3/TWhe	0.3	same as FR
ADS	LILW-SL	m^3/TWhe	6.4	same as FR
	LILW-LL	m^3/TWhe	0.3	same as FR

1.3 Reprocessing

	Waste type	Unit	Nominal value	Source
PUREX	LILW-SL	m^3/tHM	1.21	Briefing note, Nirex (2004)
	LILW-LL	m^3/tHM	0.8	Briefing note, Nirex (2004)
	HLW	m^3/tHM	0.128	NSC WG on Flowsheet Study
UREX	LILW-SL	m^3/tHM	1.21	same as PUREX
	LILW-LL	m^3/tHM	0.8	same as PUREX
	HLW	m^3/tHM	0.128	same as PUREX
Advanced PUREX	LILW-SL	m^3/tHM	1.21	same as PUREX
	LILW-LL	m^3/tHM	0.8	same as PUREX
	HLW	m^3/tHM	0.128	same as PUREX
OREOX	LILW-SL	m^3/tHM	2.11	Korean DUPIC Study (Appendix D)
	LILW-LL	m^3/tHM	0.125	
	HLW	m^3/tHM	0.128	
Pyrochemical (Metal fuel)	LILW-SL	m^3/tHM	0	same as Pyrochemical (nitride fuel)
	LILW-LL	m^3/tHM	2.25 (= 2.69-0.44[1)])	same as Pyrochemical (nitride fuel)
	HLW	m^3/tHM	0.72	NSC WG on Flowsheet Study
Pyrochemical (Carbide fuel)	LILW-SL	m^3/tHM	0	CEA (Appendix E)
	LILW-LL	m^3/tHM	2.9	
	HLW	m^3/tHM	0.21	
Pyrochemical (Nitride fuel)	LILW-SL	m^3/tHM	0	JAERI
	LILW-LL	m^3/tHM	2.69	
	HLW	m^3/tHM	0.15	

1) Cladding metal included in HLW.

2. Unit costs

2.1 Front-end fuel cycle

		Unit	Lower bound	Nominal value	Upper bound	Source
Natural uranium[1)]		$/kgU	20	50	80	[3]
Conversion		$/kgU	3	5	8	[3]
Enrichment		$/SWU	80	100	120	[3]
Fuel fabrication	UOX	$/kgHM	200	250	300	[3]
	MOX	$/kgHM	1 000	1 250	1 500	[3]
	MOX(+Np)	$/kgHM	1 000	1 250	1 500	same as MOX
	DUPIC[2)]	$/kgHM	472	675	870	Korean DUPIC Study
	MOX-EU	$/kgHM	1 100	1 375	1 650	10% higher than MOX
	MOX-EU (+Am)	$/kgHM	1 200	1 500	1 800	20% higher than MOX
	Am target	$/kgHM	1 000	1 875	3 000	LB: same as MOX NV: 50% higher than MOX UB: 2 times higher than MOX
	FR-MOX	$/kgHM	1 000	1 500	2 000	[3], but LB is same as MOX
	FR-metal	$/kgHM	1 400	2 600	5 000	[4]
	ADS-fuel	$/kgHM	5 000	11 000	15 000	[4]
	LMFR-fuel	$/kgHM	1 100	1 650	2 200	10% higher than FR MOX
	GCFR-fuel	$/kgHM	1 100	1 650	2 200	same as LMFR-Fuel

1) It is assumed that depleted uranium has the same price as natural uranium.
2) The value includes the cost of DUPIC fuel transport to the reactor.

2.2 Reactor investment and operation and maintenance

			Unit	Lower bound	Nominal value	Upper bound	Source
Investment cost	PWR		$/kWe	1 200	1 600	1 900	[5]
	CANDU		$/kWe	1 200	1 600	1 900	same as PWR
	FR		$/kWe	1 200	1 900	2 300	LB: same as PWR as a target NV, UB: 20% higher than PWR
	ADS	Reactor	$/kWe	1 200	1 900	2 300	same as FR
		Accelerator	$/Wbeam	5	15	20	[4]
	GCFR		$/kWe	1 200	1 900	2 300	same as FR
	LMFR		$/kWe	1 200	1 900	2 300	same as FR
Load factor	PWR		%	85	90	95	5% higher than in [4] considering recent developments
	CANDU		%	85	92	95	LB, UB: same as PWR NV: 2% higher than PWR
	FR		%	80	85	95	LB, NV: 5% lower than PWR UB: same as PWR
	ADS		%	75	80	85	LB, NV: 5% lower than FR UB: 10% lower than FR
	GCFR		%	80	85	95	same as FR
	LMFR		%	80	85	95	same as FR
Fixed charge rate for investment			%/year	6	9	12	adapted from [5]
Fixed charge rate for D&D			%/year	8	8	8	[4]
Annual Rx O&M cost rate			%/year	3	4	5	adapted from [5]

2.3 Spent fuel transport and storage

			Unit	Lower bound	Nominal value	Upper bound	Source
Transport	UOX S/F		$/kgHM	40	50	60	[4]
	MOX S/F		$/kgHM	60	90	240	LB: same as UB of UOX S/F NV: 50% higher than LB UB: 4 times higher than UOX S/F
	ADS S/F		$/kgHM	600	900	2 400	10 times higher than MOX
	Other fuel S/F		$/kgHM	60	90	240	same as MOX
Interim storage[1)]	UOX S/F	fixed	$/kgHM	40	50	60	[4]
		per y	$/kgHM	5	5	5	[4]
	MOX S/F	Fixed	$/kgHM	60	90	240	LB: same as UB of UOX S/F NV: 50% higher than LB UB: 4 times higher than UOX S/F
		per y	$/kgHM	5	7.5	20	”
	ADS S/F	Fixed	$/kgHM	600	900	2 400	10 times higher than MOX
		per y	$/kgHM	50	75	200	”
	Other S/F	Fixed	$/kgHM	60	90	240	same as MOX
		per y	$/kgHM	5	7.5	20	”
DUPIC S/F transport and interim storage			$/kgHM	40	160	200	Korean DUPIC study

1) It is assumed that interim storage cost consists of fixed cost plus cost varying depending on the period of storage. Therefore, the unit cost for interim storage is calculated as “fixed cost + varying cost/year x period of storage”.

2.4 Reprocessing

	Unit	Lower bound	Nominal value	Upper bound	Source
UOX PUREX	$/kgHM	700	800	900	[4]
MOX PUREX	$/kgHM	700	800	1 000	[4]
FR-MOX PUREX	$/kgHM	1 000	2 000	2 500	[4]
UOX UREX	$/kgHM	600	800	1 200	LB: 15% lower than UOX PUREX considering tech. developments NV: same as UOX PUREX UB: 20% higher than UOX PUREX considering uncertainties
UOX advanced PUREX	$/kgHM	700	1 000	1 300	LB: same as UOX PUREX NV: 25% higher than UOX PUREX considering additional processes UB: 30% higher than UOX PUREX considering additional processes and uncertainties
MOX advanced PUREX	$/kgHM	700	1 000	1 500	LB, NV: same as UOX Advanced PUREX UB: 15% higher than UOX Advanced considering multiple recycling
FR MOX advanced PUREX	$/kgHM	1 000	2 200	3 000	LB: same as FR MOX PUREX NV: 10% higher than FR MOX PUREX considering additional processes UB: 20% higher than FR MOX PUREX considering additional processes and uncertainties
FR-metal fuel PYRO	$/kgHM	1 000	2 000	2 500	[4]
ADS-fuel PYRO	$/kgHM	5 000	7 000	12 000	[4] but UB is a little lower considering technical progress
GCFR-fuel PYRO	$/kgHM	1 000	2 400	3 000	LB: same as FR PYRO NV, UB: 20% higher than FR PYRO considering additional processes
LMFR-fuel advanced PUREX	$/kgHM	1 000	2 200	3 000	same as FR Advanced PUREX

2.5 Dry storage, packaging and long-term storage

		Unit	Lower bound	Nominal value	Upper bound	Source
Dry storage (100 y)	UOX S/F	$/kgHM	100	150	250	own estimate
	MOX S/F	$/kgHM	200	300	500	2 times higher than UOX S/F
	MOX(Pu+Np) S/F	$/kgHM	200	300	500	same as MOX S/F
	DUPIC S/F	$/kgHM	70	100	150	a little lower than UOX S/F
	Irradiated Am Target	$/kgHM	200	300	500	same as MOX S/F
	UOX PUREX HLW	$/m^3	80 000	120 000	200 000	own estimate
	Other HLW	$/m^3	80 000	120 000	200 000	same as UOX PUREX HLW
Packaging	UOX S/F	$/kgHM	100	200	350	[6]
	MOX S/F	$/kgHM	200	400	700	2 times higher than UOX S/F
	MOX(Pu+Np) S/F	$/kgHM	200	400	700	same MOX S/F
	DUPIC S/F	$/kgHM	80	150	250	a little lower than UOX S/F
	Irradiated Am Target	$/kgHM	200	400	700	same MOX S/F
	UOX PUREX HLW	$/m^3	100 000	200 000	400 000	[6]
	Other HLW	$/m^3	100 000	200 000	400 000	same as UOX PUREX HLW
Long-term storage[1)]	Depleted Uranium	$/kgU	2.6	3.6	4.6	[4]
	Reprocessed Uranium	$/kgU	2.6	3.6	40	same as DepU, but UB much higher considering possible radioactivity
	Am	$/kgHM	28 000	42 000	112 000	50 times higher than MOX S/F interim storage
	Cm	$/kgHM	56 000	84 000	224 000	100 times higher than MOX S/F interim storage

1) This study does not consider what to be followed after the long-term storage.

2.6 Waste disposal

	Unit	Lower bound	Nominal value	Upper bound	Source
LILW (short lived) near-surface disposal	$/m^3	1 200	2 000	3 000	[7]
LILW (long lived) cavern-based and geological disposal	$/m^3	4 000	6 000	8 000	[7]
Disposal galleries[1)] (underground cost)	$/m^3	600	1 200	2 000	[6]
Unit volume of disposal galleries to be excavated for heat generating waste	m^3/kW	10	20	30	own estimate

1) It is assumed that geological disposal cost (underground cost) is proportional to the volume of disposal galleries to be excavated. It is also assumed that the volume of disposal galleries to be excavated is proportional to the decay heat at the time of disposal.

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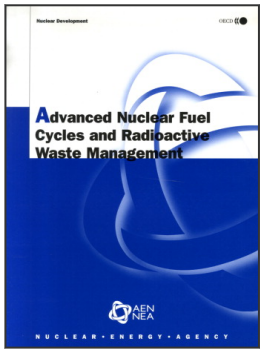

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