Disposal of Spent Advanced Reactor Fuel: Where Do We Begin?

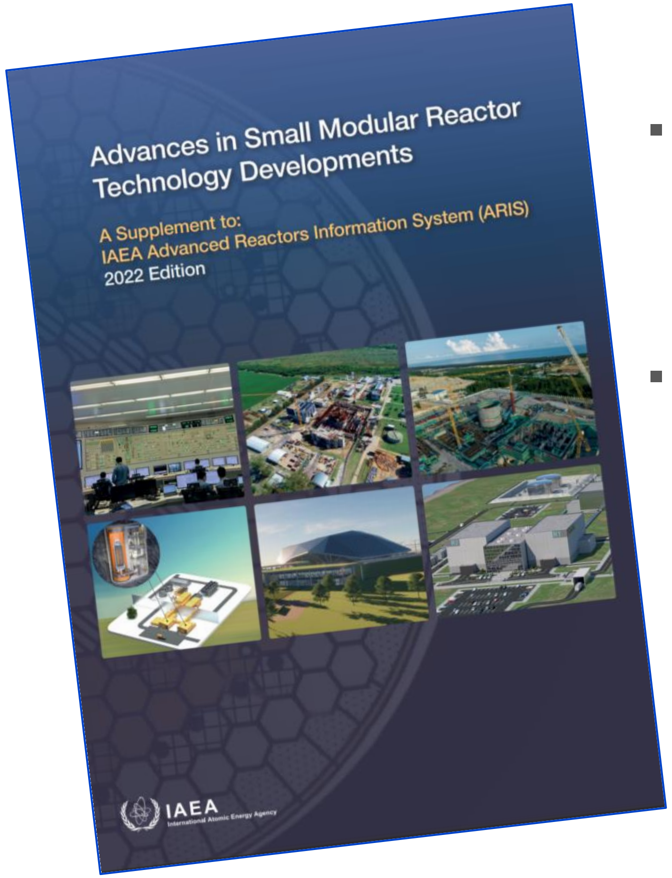
NUCLEAR

Rise of new and innovative technologies, future nuclear systems, and impact on the development of DGRs

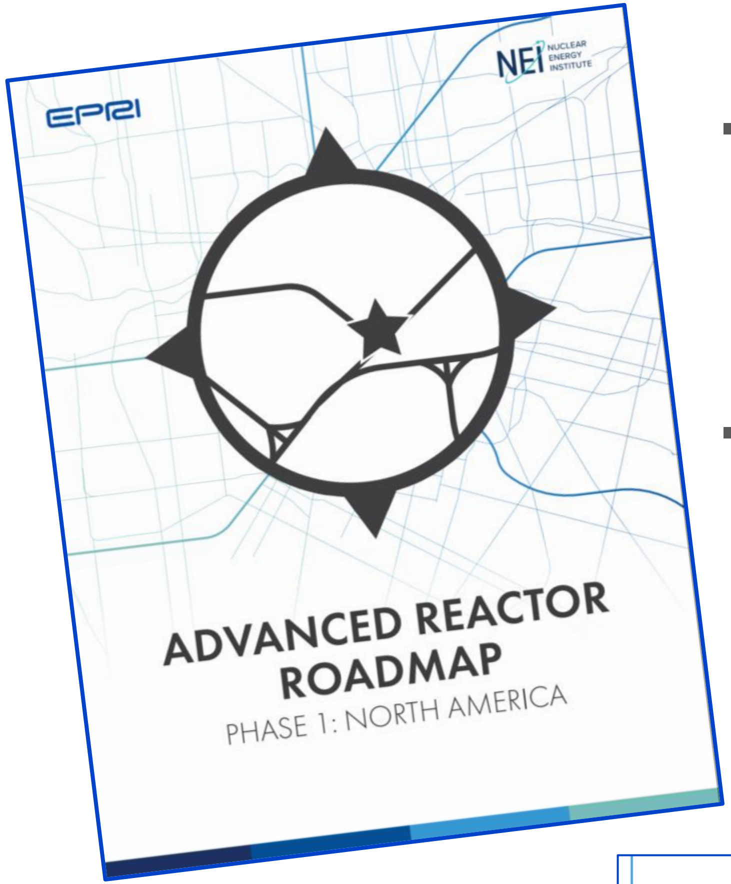
Laura McManniman
Senior Technical Leader
Used Fuel and High-Level Waste Program

NEA 7th International Conference on Geological Repositories (ICGR-7)
29 May 2024

Global Trends in Advanced Reactors



- Captures designs under different stages of design and deployment
- 83 reactor designs from 18 different countries listed



- Outlines critical strategies and support needed for successful widespread deployment of Advanced Reactors
- Presents a series of prioritized actions that will evolve over time as strategies are refocused

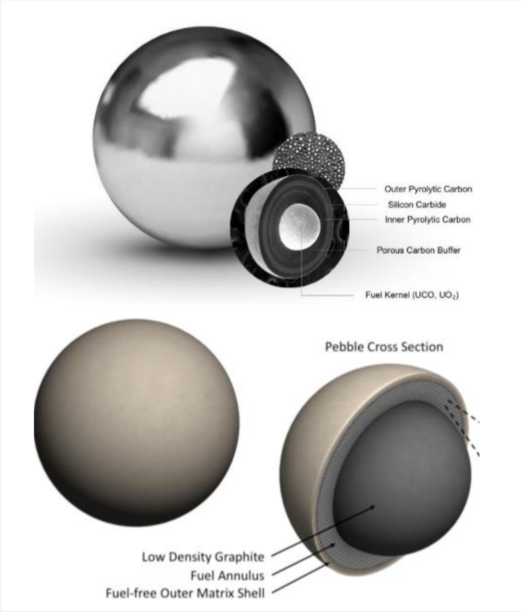
Lots going on in reactor development space

Advanced Reactor Fuel Options

Ceramic/Oxide

Mature technology with understood, qualified, and licensed back-end solutions

TRISO

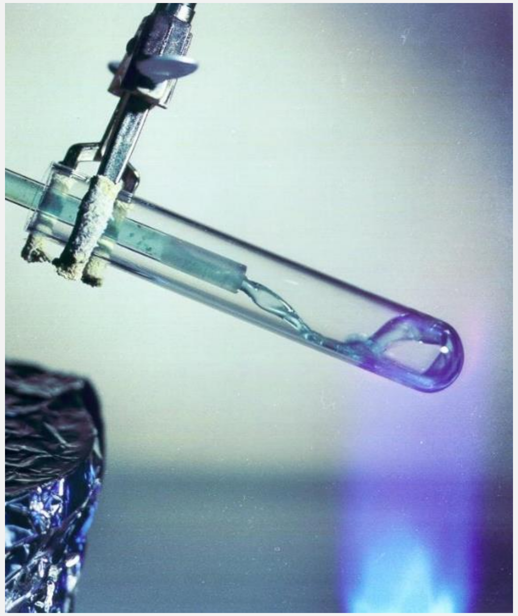


Evolving technology with **historical backing** being deployed in diverse designs

Metallic

Lab-experience provides **basis for development** of newer options

Dissolved



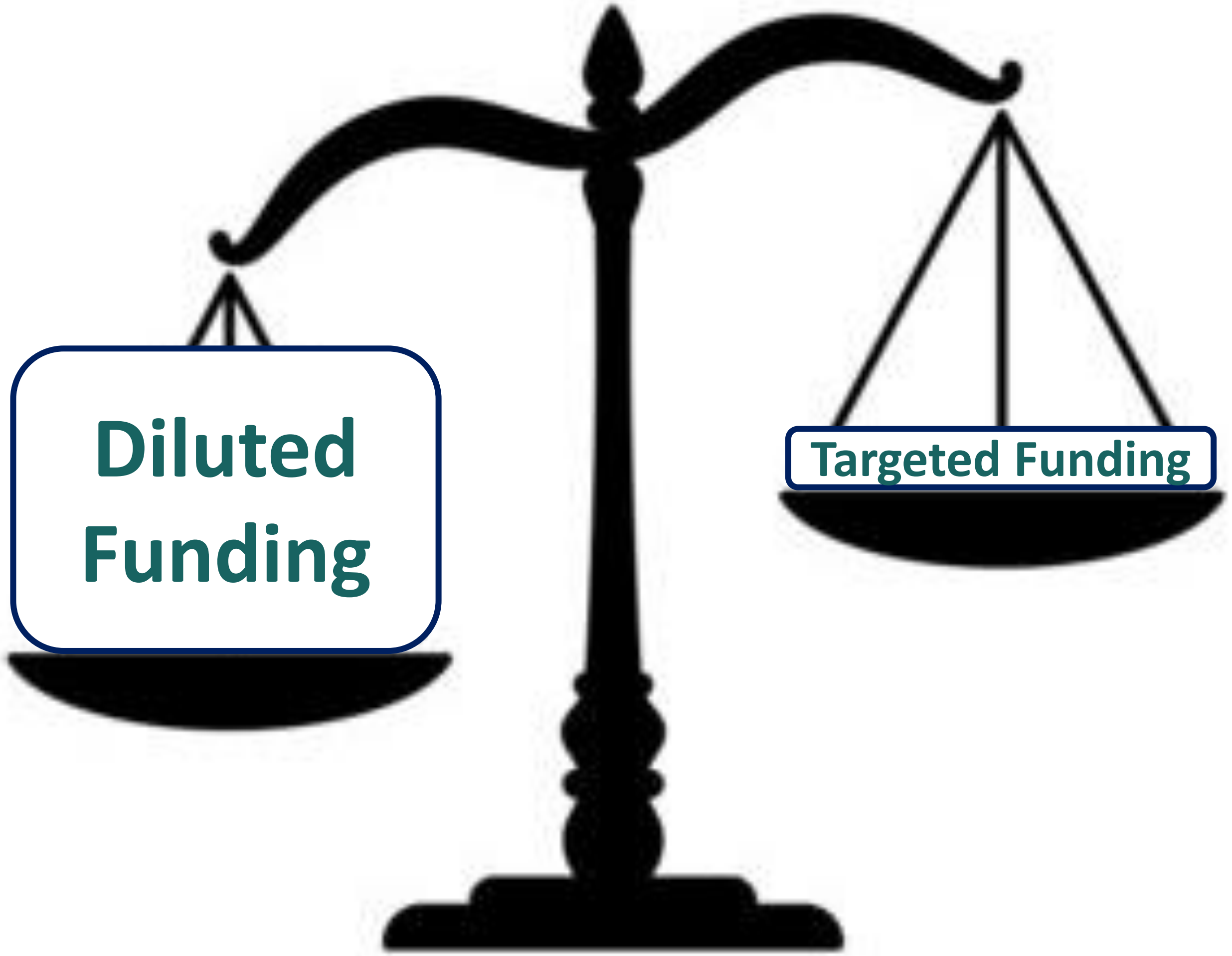
Emerging concepts being supported with ongoing coupled effects tests

Multiple specific designs exist within each fuel technology category

Images of TRISO are copyright of X-Energy and Kairos, respectively

Global Trends in Advanced Reactors

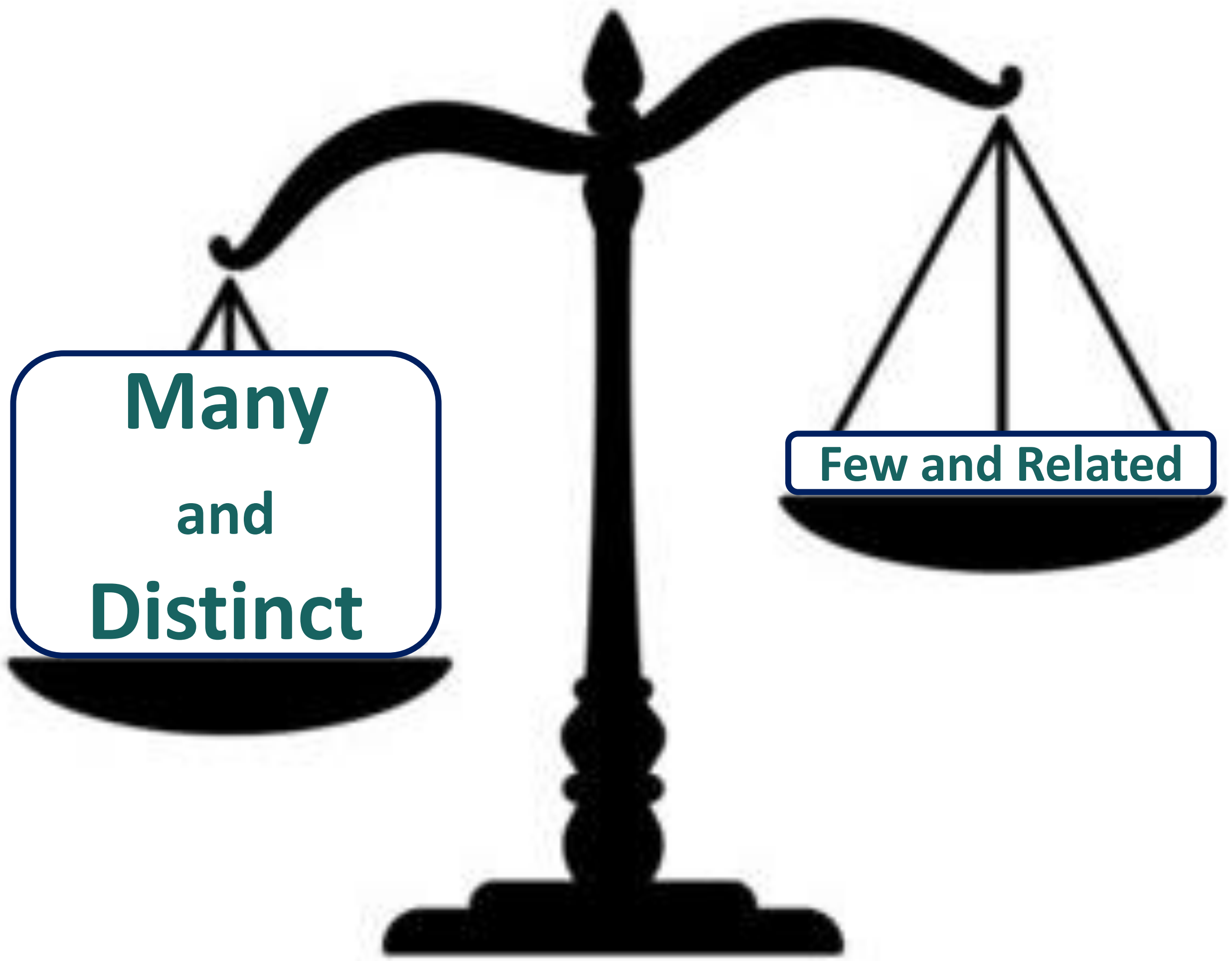
Investment by Nation



EPRI

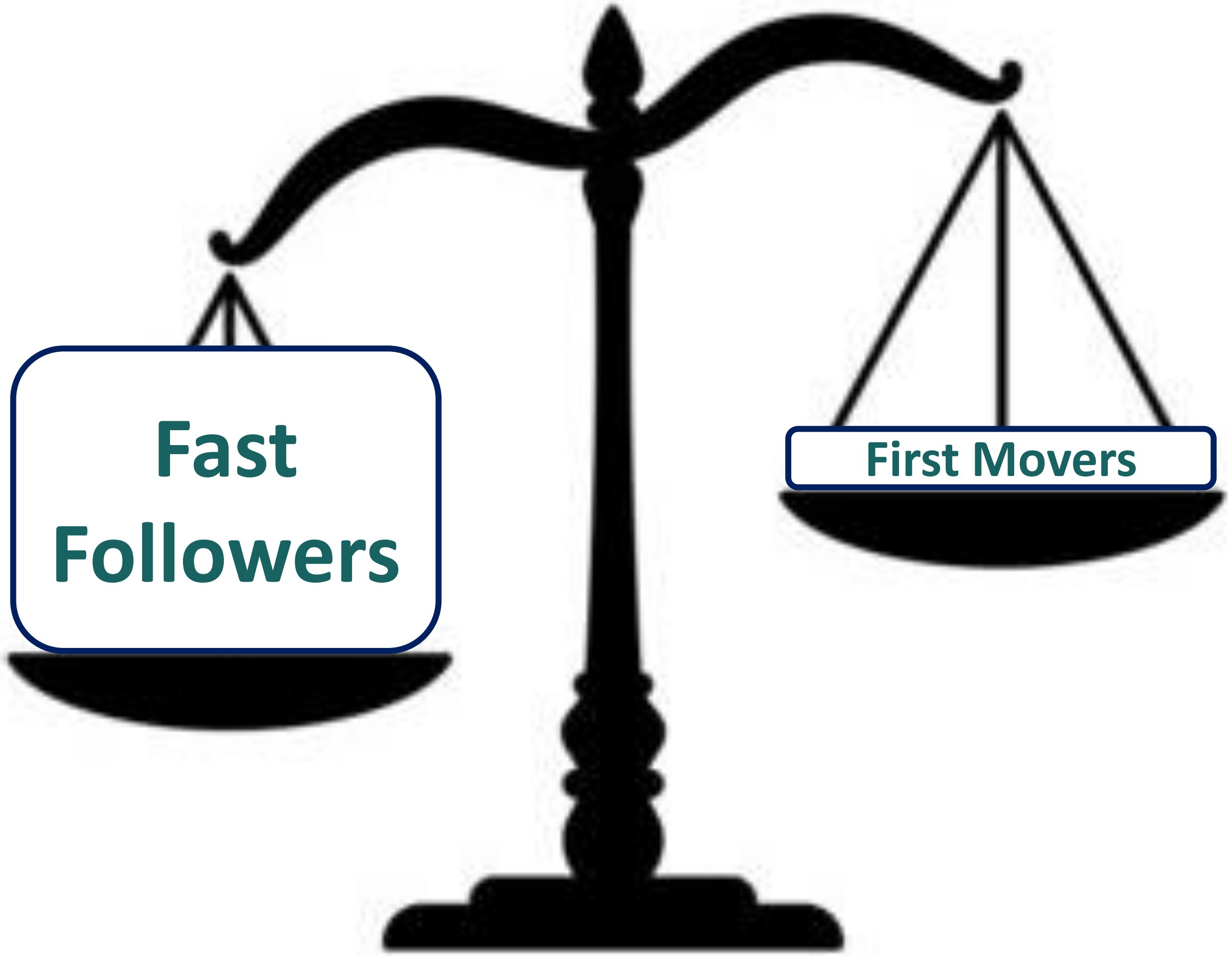
Global Trends in Advanced Reactors

Design Concepts



Global Trends in Advanced Reactors

Interest in New Projects



So how do you design a disposal research program?

What do we need to know?

Disposal

- Generic waste acceptance criteria for AR fuel and waste, along with associated conditioning and packaging requirements prior to disposal
- Identification of compatibility of novel fuel and waste with existing oxide disposal concepts / DGRs

Transportation

- Criticality benchmarks to optimize package design and loading
- Identification of technical gaps for metallic and liquid fuels to inform R&D investments

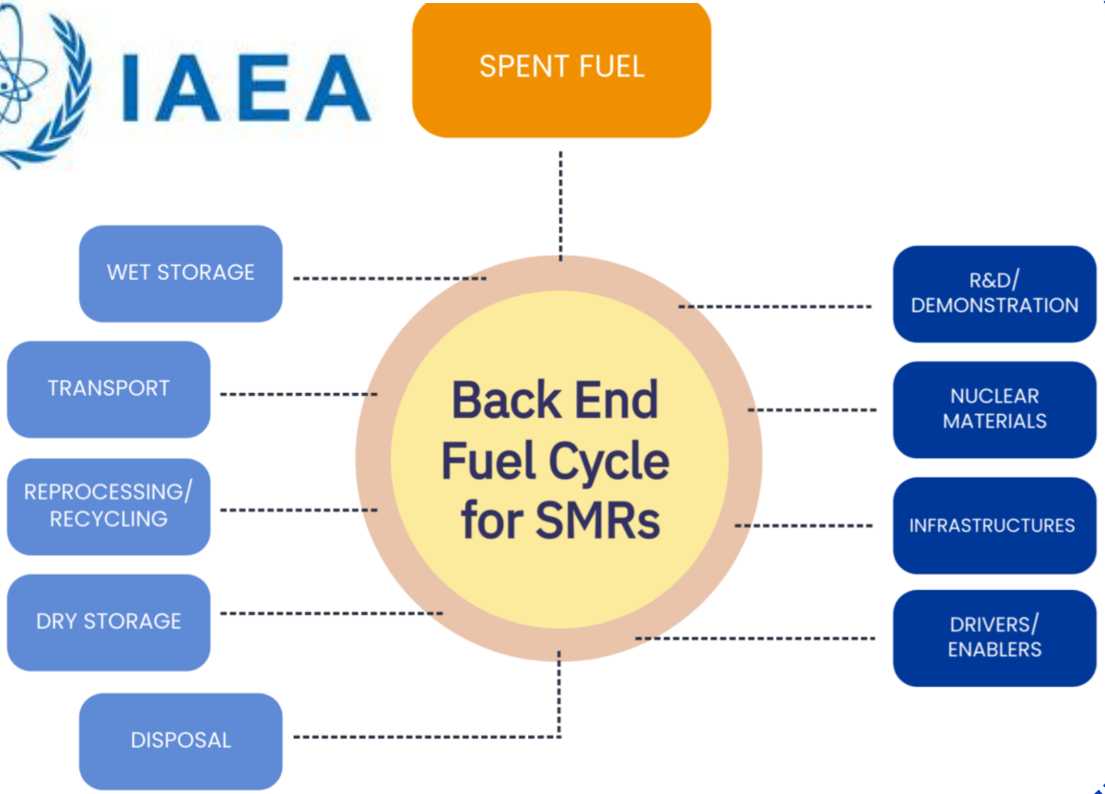
Recycling

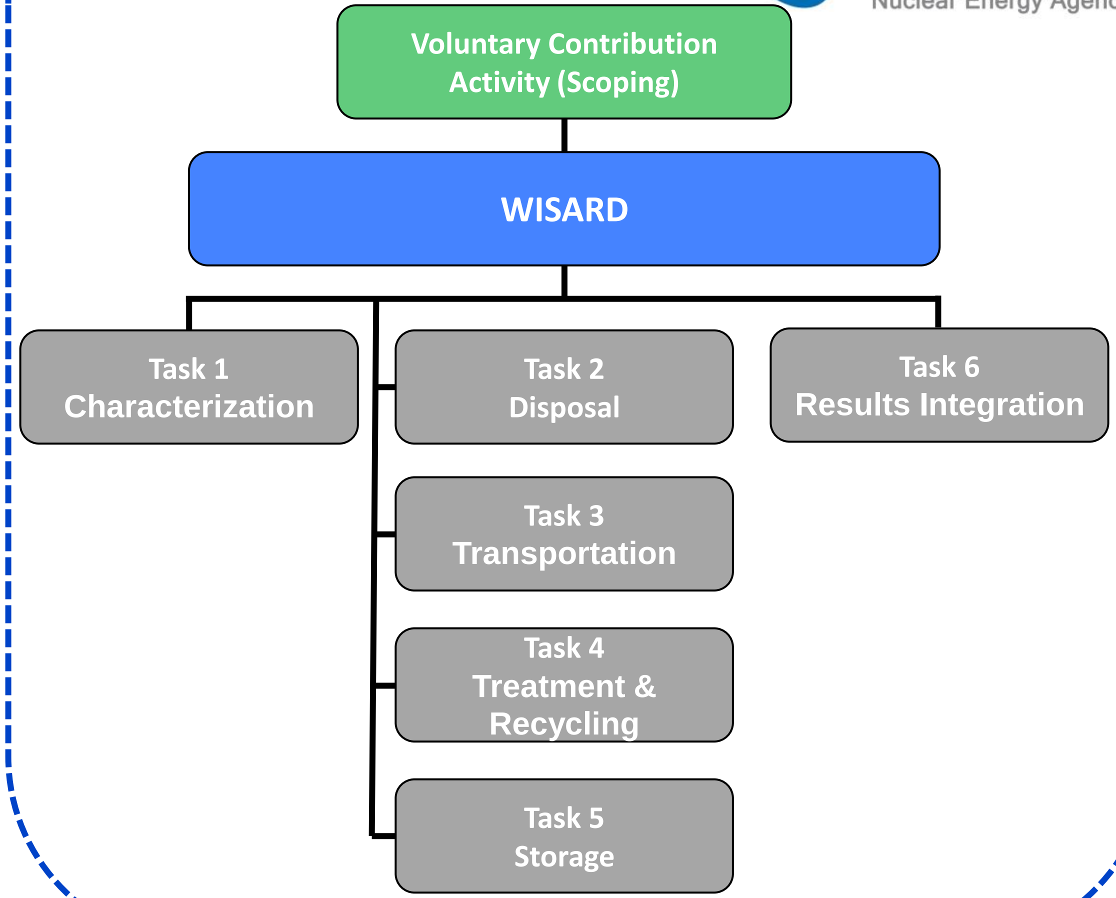
- Knowledge of waste streams from contemporary recycling processes
- Waste forms produced from novel recycling techniques

Storage

- Validate simulations for discharge decay heat and activity of contemporary AR fuel forms
- Comparison of opportunities to adapt existing infrastructure vs opportunities to introduce new innovative concept for footprint minimization

Global Collaboration





To Conclude…

- Lots of advanced reactor designs are under consideration in differing stages of development
- Results in lots of possible back-end fuel cycle permutations
  - Lots of possible waste package characteristics to go to DGR
- Can leverage existing knowledge to support development of options for some fuel types, other are more limited
- Collaboration is one way we can leverage our research dollars to maximize results
  - Many efforts ongoing to provide a means to do so

TOGETHER…SHAPING THE FUTURE OF ENERGY®

www.epri.com

Eef Weetjens

Dedicated R&D Toward Optimal Solutions in Radioactive Waste Management

sck cen
Belgian Nuclear Research Centre

Challenges in RWM

Old & new

- Dealing with **legacy waste**
  - Conventional
    - Spent UOX fuel, vitrified high-level waste
  - Unconventional
    - Bituminised waste, decommissioning waste, waste from experimental reactors,…
- Anticipating **new waste streams**
  - AR/SMR fuels
  - AR/SMR reprocessing waste
  - Coolant waste

Optimal WM solutions

Evaluation of options informed by dedicated R&D

Nuclear fuel cycle

- Fuel type & burn-up
- Reprocessing method
  - None, aqueous or pyroprocessing
- Advanced separation options
  - Fissile/fertile material
  - Other elements (circular economy)

waste?

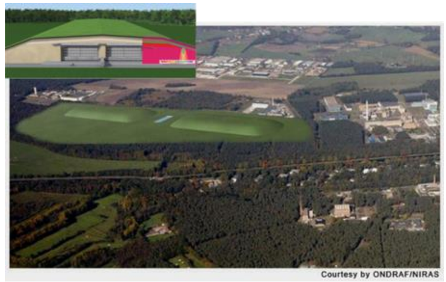
disposability?

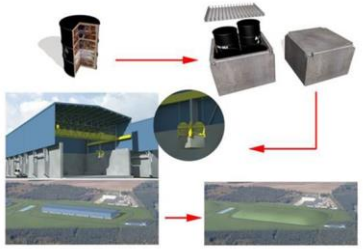
Geological disposal

- Conditioning methods
  - Vitrification
  - Cementation
  - Alkali-activated materials
  - Ceramic matrices
- Disposal methods
  - Host formation & depth
  - Engineered barrier system
  - GDF lay-out

Management of Belgian radioactive waste

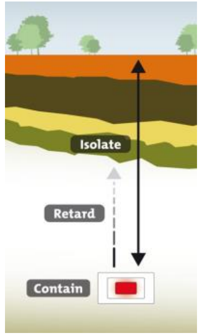
- Short-lived low and intermediate level waste (category A)

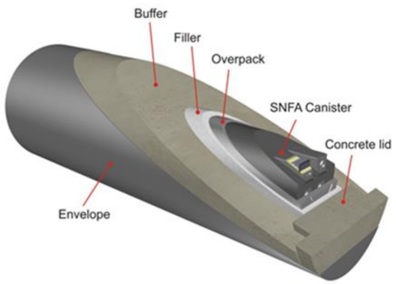


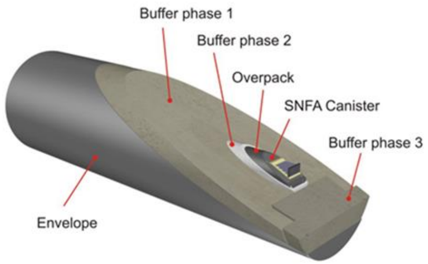


Surface disposal site in Dessel
Operations to be started 2027

- Long-lived and/or high-level waste (category B&C)







First decision in 2022 but no site yet
Reference = deep geological disposal in poorly indurated clay First disposal of heat-emitting waste foreseen >2100

- Radioactive radium-bearing waste (RRA)

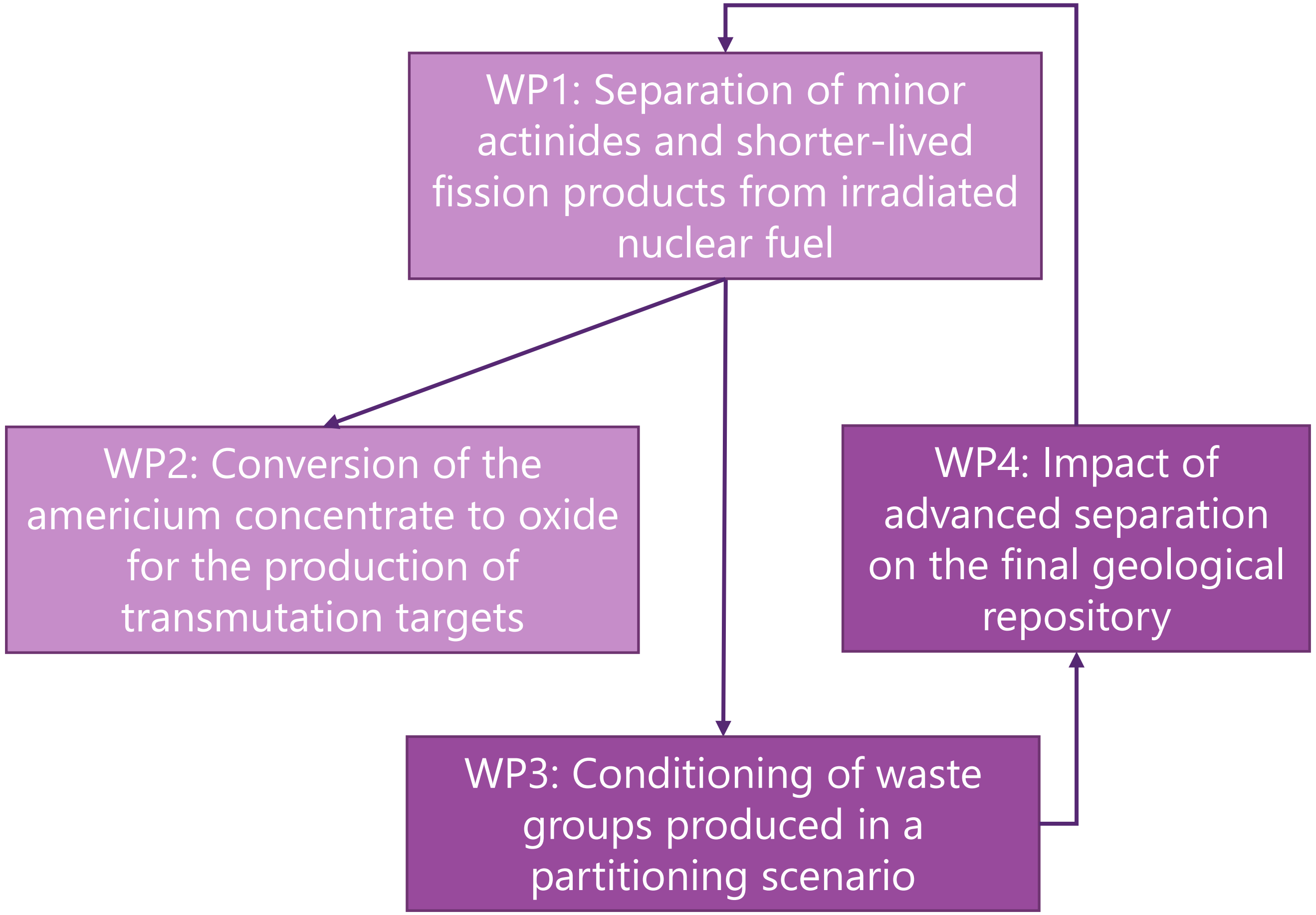
Reference = disposal at shallow depth
Legislative and regulatory framework under development

The ASOF project

Advanced Separation for the Optimal management of spent Fuel

- To build expertise and advance fundamental R&D in order to make more informed decisions concerning the “advanced separation” option for Belgian nuclear spent fuel

- Focus on separation of
  - Minor actinides
  - Cs and Sr



Immobilisation in a ceramic matrix

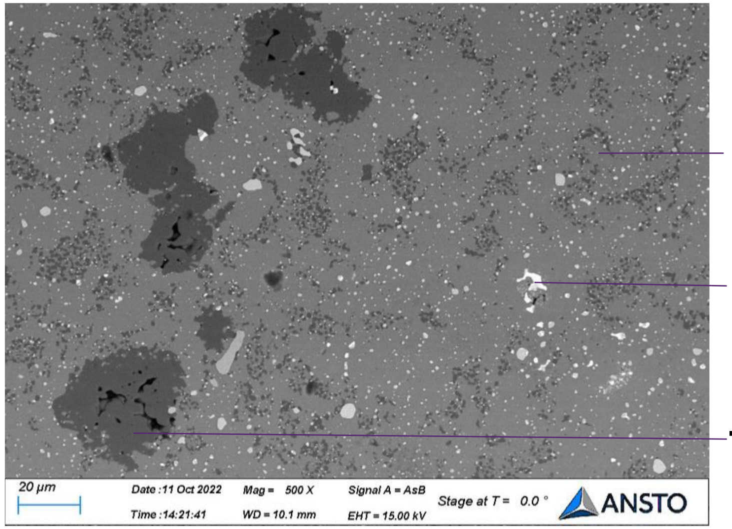
- **Synroc**: proven capability to immobilize complex waste streams and high leaching durability

Phase	Nominal composition	Wt.%[a]	Key radionuclides in lattice
Hollandite	$Ba(Al,Ti)_2Ti_6O_{16}$	30	Cs, Rb
Zirconolite	$CaZrTi_2O_7$	30	RE, An
Perovskite	$CaTiO_3$	20	Sr, RE, An
Ti oxides and Ca-Al-Titanates (e.g. loveringite)	TiO_2, Ti_nO_{2n-1}, Ca-Al-Ti (CAT) phases [8]	15	
Alloy phases		5	Tc, Pd, Rh, Ru etc.

RE = rare earths; An = actinides. [a]Wt.% of phase in Synroc-C with 20 wt.% HLW.

- Objectives:
  - demonstrate feasibility to immobilise the ASOF waste streams by adaptation of the formulation
  - demonstrate the high durability in cementitious environment (high pH conditions)

- Production and testing of
  - Synroc batches for rest waste stream with waste loadings of 18 and 30.5 wt.% (ANSTO)
  - Synroc batches for Cs/Sr waste stream with waste loadings of 5 to 9 wt. % (UoS)



Zirconolite

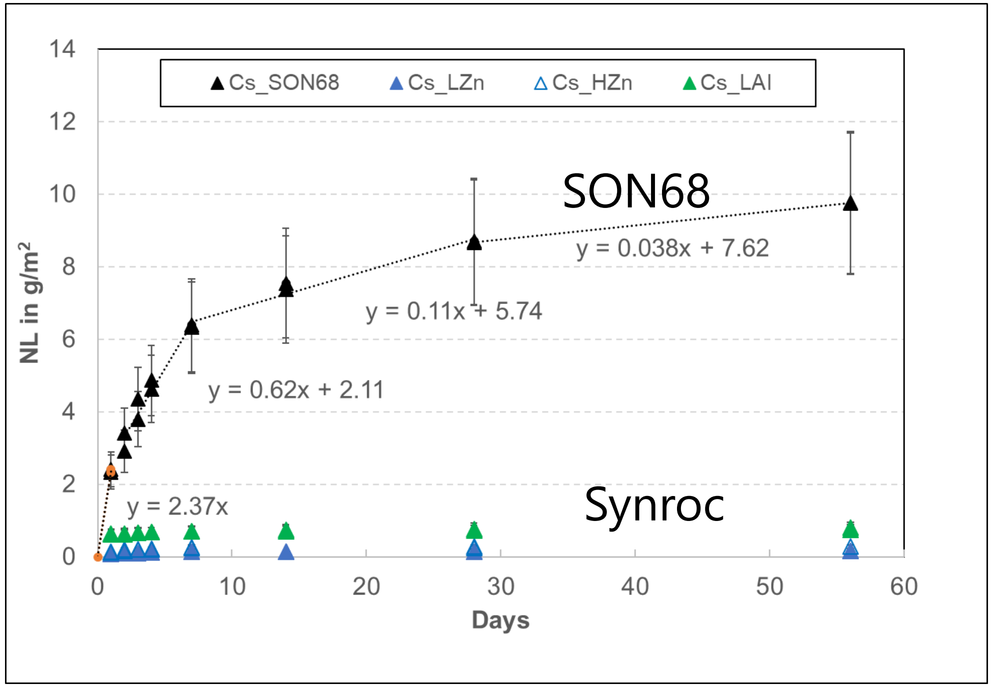
Metal alloy

Ti-oxide

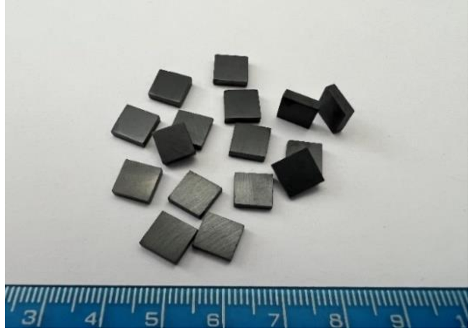
Durability testing in high pH conditions (cementitious conditions)

- Static leaching tests using powdered sample (75 – 150 µm), at 90 °C in KOH solution at pH 12.5 in anoxic conditions, with test durations: 1, 2, 3, 4, 7, 14, 28 and 56 days
- Tests with reference waste glass SON68 to compare stability vitrified waste ←→ Synroc

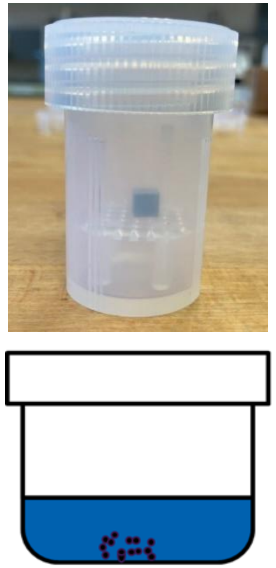
Normalized Cs release for Synroc and SON68



Powder 75 – 150 µm



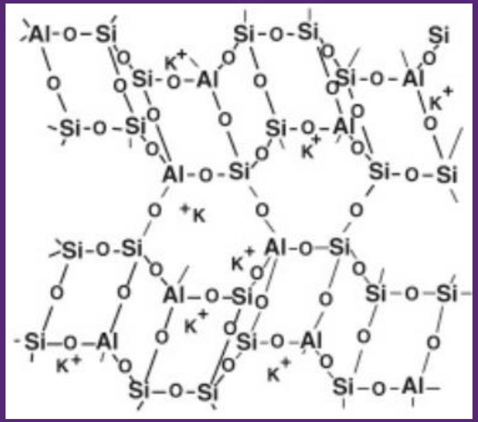
Monoliths 8 x 8 x 2 mm



- The release rate was up to (>)100 times lower for Synroc than SON68, depending on the element
- The exact formulation of the Synroc and the different waste loading had a relatively small effect on the durability

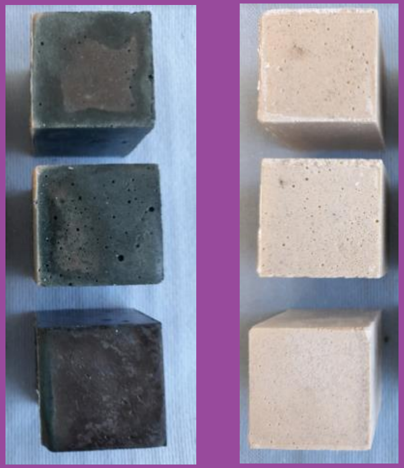
Immobilisation in alkali-activated matrices (geopolymers)

Alkali-activated materials



- Prepared from aluminosilicate precursors (slag/metakaolin)
- Activated by a strong basic solution
- More chemically durable than equivalent OPC materials

Development of reference materials



- Materials with good mechanical strengths (F_c>10 MPa) were developed
- A large range of water-to-binder ratios can be employed

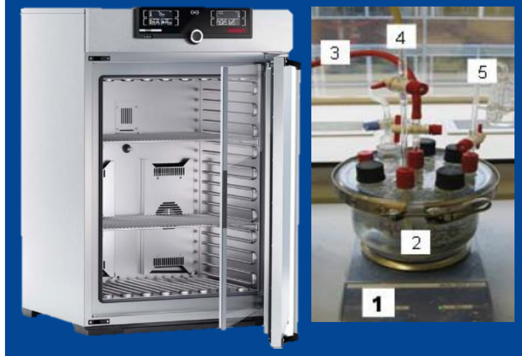
➔ Applicable to multiple types of waste

Waste solution to immobilize

Element	Concentration (Mol/L)
Cs	3.85E-03
Sr	1.84E-03
Ba	2.85E-03
Rb	8.76E-04
Na	6.06E-04

- Acidic solution (0.03M HNO_3)
- Non-active simulant is used
- Actual waste stream would generate heat

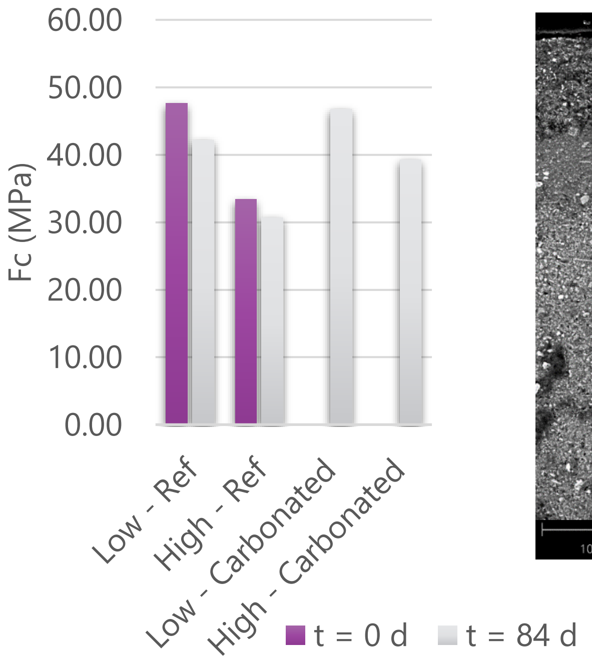
Evaluation of durability

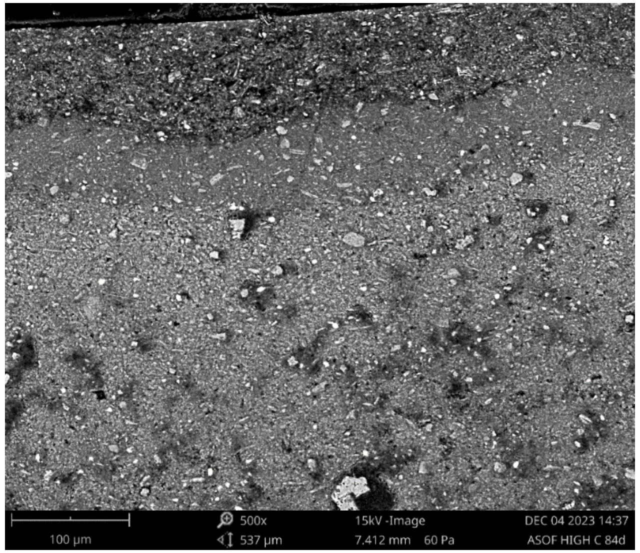


- Carbonation: accelerated in 1% CO_2 atmosphere
- Leaching in:
  - Deionized water
  - NH_4NO_3 (accelerated)
- Post-mortem characterization

Durability testing of alkali-activated waste forms

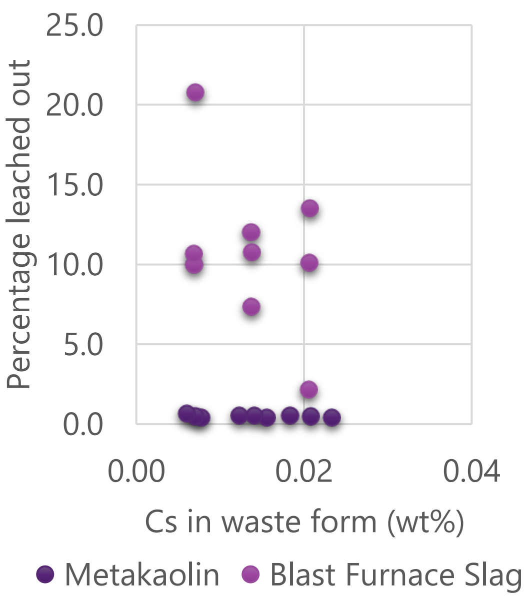
Carbonation

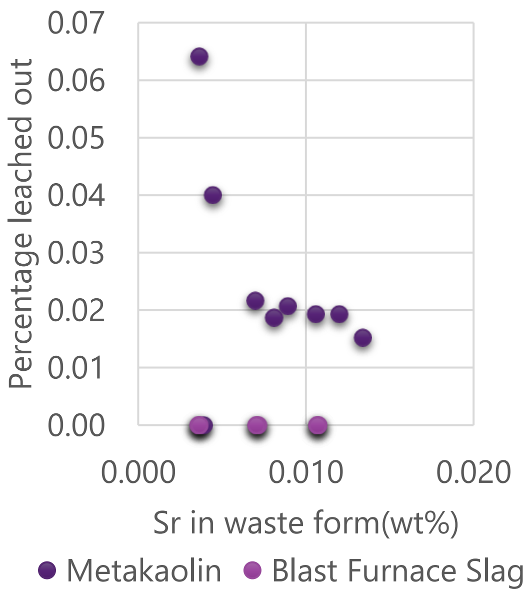




- Mechanical strength is not affected by carbonation
- Limited alteration of the microstructure, mainly due to precipitation of sodium carbonate

Leaching

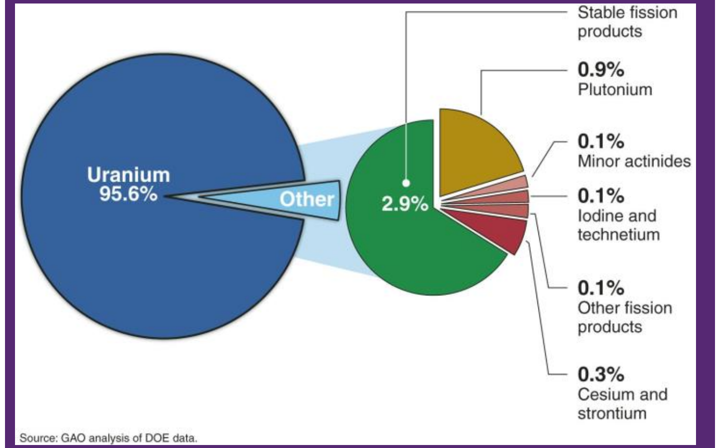




- Leaching rates in deionized water are low
- Leaching of Cs/Sr depends on the precursor used in the waste form
- Accelerated leaching tests in NH_4NO_3 stay within reasonable limits

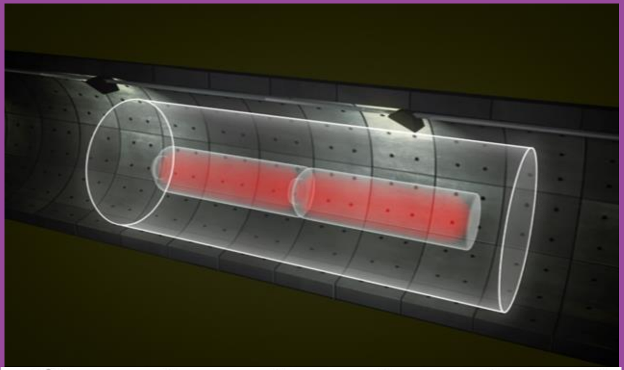
Impact assessment: waste form metrics

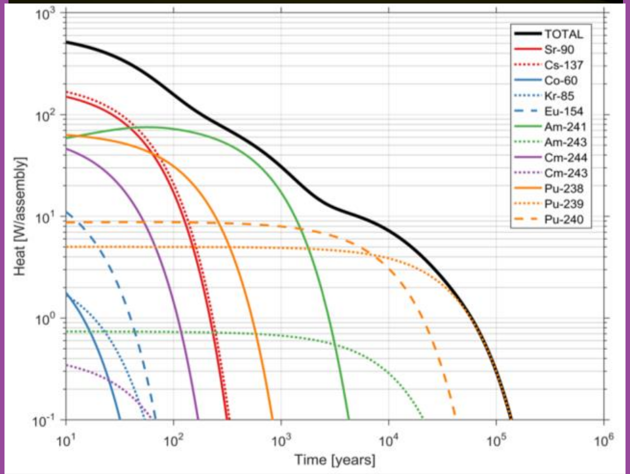
Mass/volume



Derivation of radiological & chemical inventories per waste canister taking into account separation efficiency & matrix waste loadings

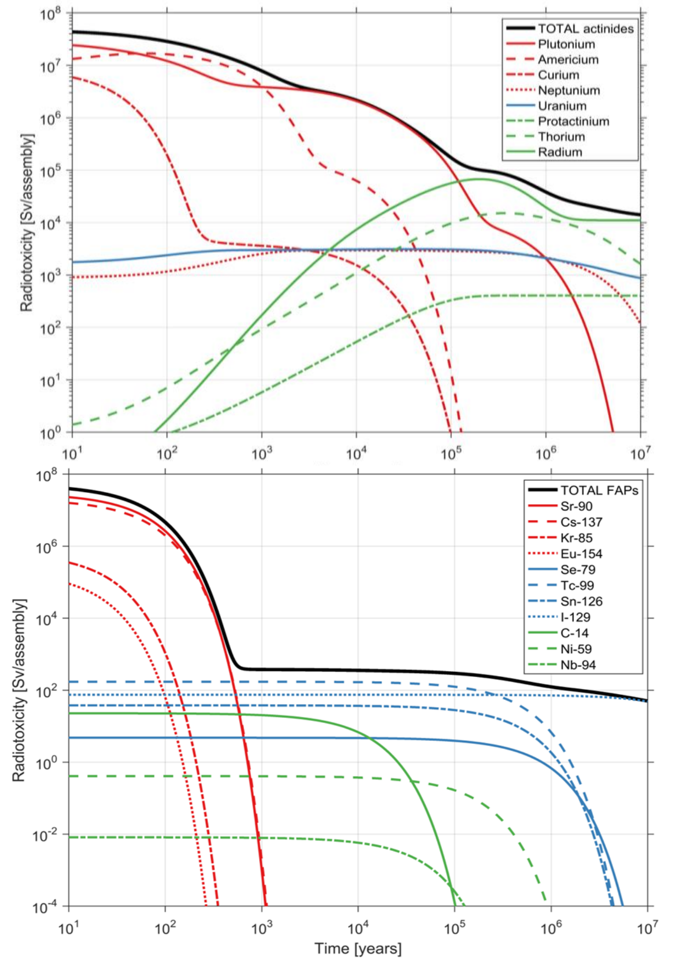
Decay heat



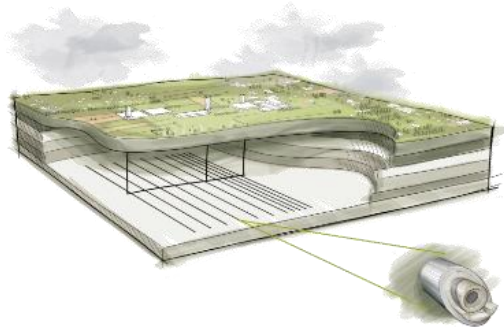


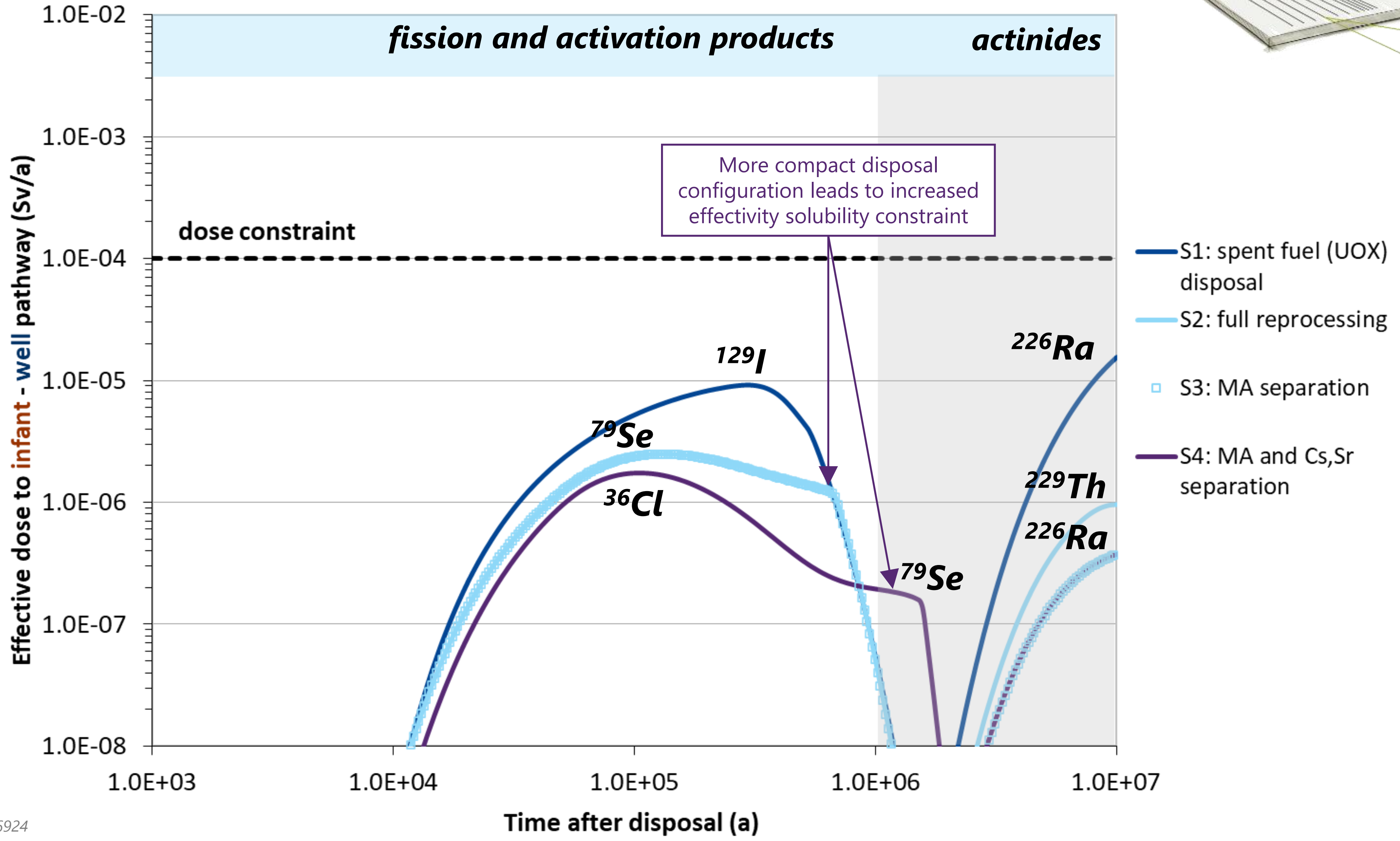
Derivation of decay heat generation per canister

Radiotoxicity (act × IDF)



Radiological impact



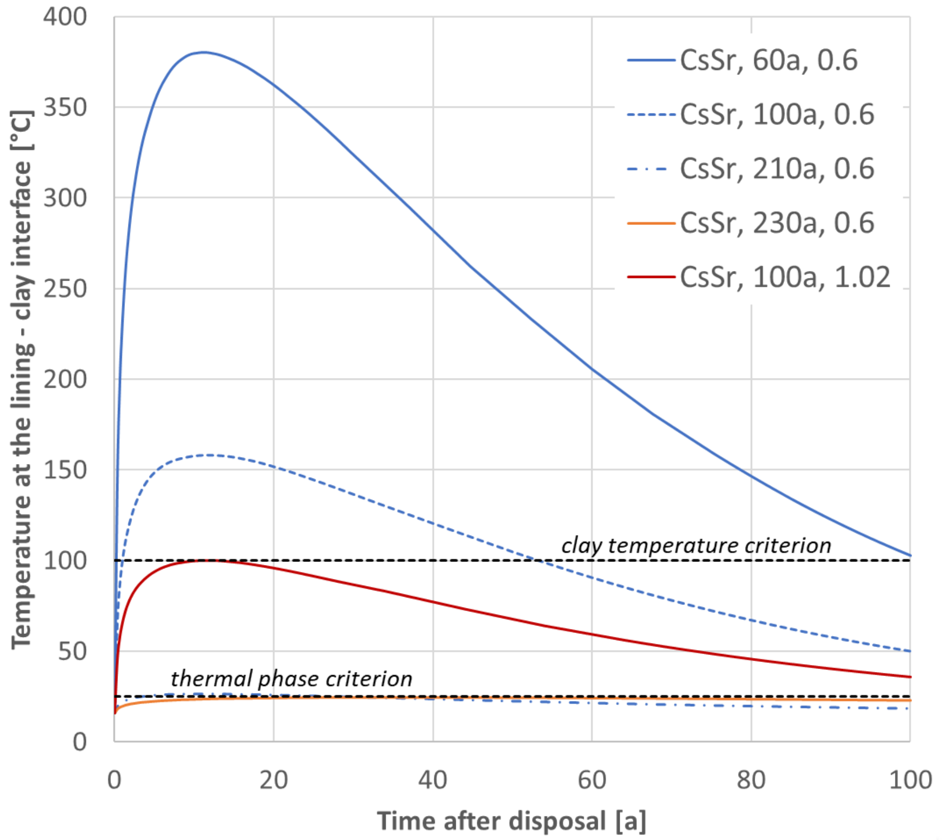


Disposal of Cs/Sr waste (P&C) in poorly indurated clay

- Cs/Sr waste is not acceptable in surface disposal facility
  - Heat generation
  - Contains ^{135}Cs ($t_{1/2}=2.30\times10^{6}$ a)
  - Disposal capacity
    - 500a decay storage to reach "DL" of ^{137}Cs
    - 600a decay storage to reach "DL" of ^{90}Sr

- Geological disposal - monolith-B
  - >200 a cooling time necessary

- Geological disposal - "hot" monolith?
  - Possible with ~100 a cooling, but low waste loading



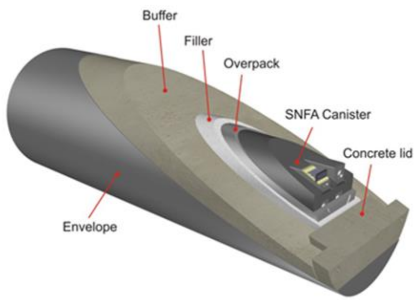
Supercontainer:* *2 canisters

W/m × 6.3

Monolith-B:* *8 canisters

GDF size (footprint)

4 UOX assemblies / Supercontainer



- Direct disposal of Belgian UOX fuel: ~**14.6 km** of disposal galleries needed

Scenarios 2/3 : full reprocessing / separation of MA			
waste loading (wt.%)	*18.5*	*22.5*	*33*
gallery length SC-1 (HLW) (m)	8951	7359	5018
gallery length CB-2 (UC-C) (m)	1478		
total gallery length (m)	**10 429**	8837	6496

Scenario 4 : advanced separation of MA and Cs/Sr			
	glass	**synroc**	**geopolymer**
waste loading (wt.%)	*18.5*	*28.5*	*20*
gallery length CB-2 (RP) (m)	1419	582	2256
waste loading (wt.%)	*0.75*	*0.47*	*1.42*
gallery length CB-2 (Cs/Sr) (m)	2486	2508	2380
gallery length CB-2 (UC-C) (m)	1478		
total gallery length (m)	**5383**	**4568**	**6113**

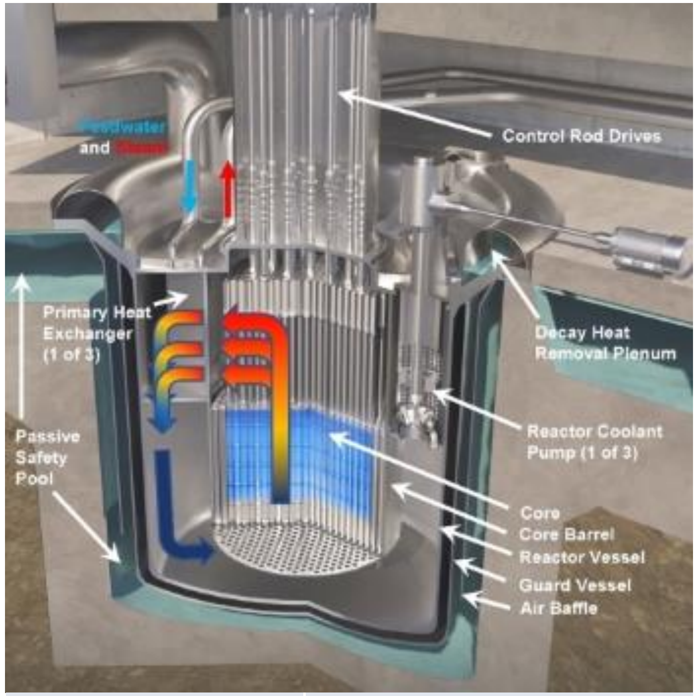
- Full reprocessing would require **10.4 km** of disposal galleries (a decrease of ~30%)
- Effect of additional separation of MA on canister waste loading for vitrified HLW?
- In case of MA *and* Cs/Sr separation, the rest product can be more compactly disposed of in monoliths instead of supercontainers: **5.4 km** of disposal galleries

SMR/AR research at SCK CEN

- Besides Myrrha (ADS demonstrator), current focus on **LFR-SMR**
  - November 2023: MoU signed by Westinghouse Electric Company, Ansaldo Nucleare, ENEA, RATEN and SCK CEN to accelerate the commercialization of lead-cooled fast reactors based on the Westinghouse LFR design

- Waste management is not a priority R&D topic at the moment
- Still good practice to initiate early evaluations on
  - waste stream identification and characterization
    - LFR-MOX fuel, Pb-coolant, reactor vessel, reflectors, control rods
  - feasibility of fuel reprocessing
  - disposability of waste streams
    - "disposability" : evaluation of compatibility with existing disposal concepts. In Belgium: concrete based disposal packages in mined galleries in a poorly indurated clay host formation
    - ⚠ Low heat tolerance of clay host rock

Figure taken from Westinghouse Nuclear website



Thermal power (MWth)	950
Outlet temperature (°C)	530 (phase 1); 650 (phase 2)
Spectrum (thermal/fast)	Fast
Fuel type	UO_2 pellets or MOX, interchangeably, as near-term fuel; Uranium nitride (UN) pellets as longer-term advanced fuel
Fuel (LEU/HALEU/HEU)	HALEU for UO_2 pellets, otherwise plutonium-uranium oxide for MOX

Info taken from NEA SMR dashboard

Acknowlegdements

- My co-authors
  - BRUGGEMAN CHRISTOPHE (ASOF Project Leader)
  - LEMMENS KAREL, FERRAND KARINE, QUOC TRI PHUNG, FREDERICKX LANDER, GOVAERTS JOAN

- With the help of the Belgian Energy Transition Fund under the SPF Economy

SCK CEN
Belgian Nuclear Research Centre

Foundation of Public Utility

Registered Office: Avenue Herrmann-Debrouxlaan 40 – BE-1160 BRUSSELS
Operational Office: Boeretang 200 – BE-2400 MOL

Tokyo Tech

Optimizing the Fuel Cycle and Advancing Deep Geological Repositories with Digital Twins and Innovative Technologies

M. Nakase, T. Okamura and T. Nishihara
Institute of Innovative Research, Laboratory for Zero Carbon Energy,
Tokyo Institute of Technology

27-31 May, 2024, Busan, South Korea

ICGR-7, Empowering Progress in DGRs

Rise of new and innovative technologies, future nuclear systems, and impact on the development of DGRs

Contents

- Innovative Nuclear Energy System Lab. (introduction)
- Optimization of the NFC
- Efficient calculation of thermal design of repository
- Conclusion

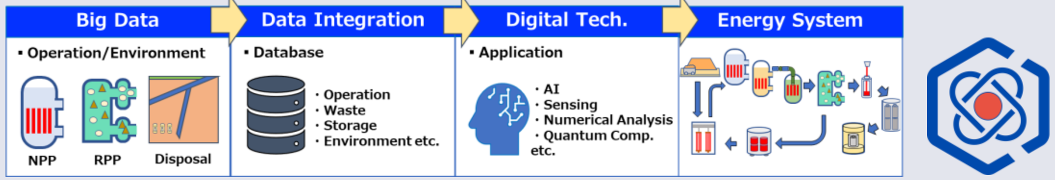
Nakase Lab in Tokyo Tech

Innovative Nuclear Energy System Lab.

Science × Engineering × Digital

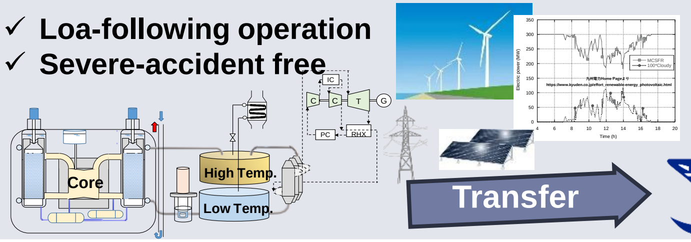
Innovative Nuclear Energy System

■ Digital Transformation (DX) on Nuclear

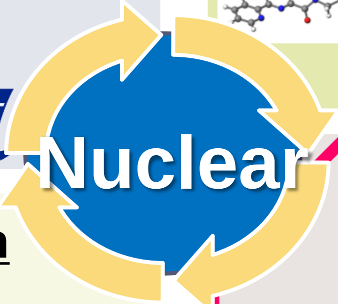


■ Molten Salt Fast Reactor

- ✓ Loa-following operation
- ✓ Severe-accident free



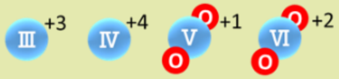
Small Molten Chloride Fast Modular Reactor for maritime



Separation Science

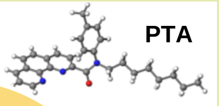
■ Actinide Science

- Valence control/extraction separation
- Solution & Complex Chemistry

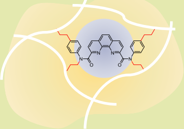


- Organic synthesis, characterization

Solvent extraction

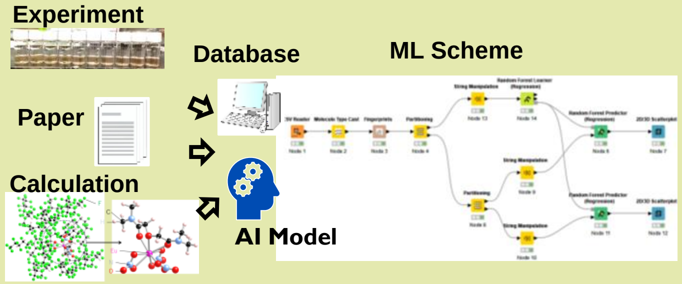


Gel liquid extraction



Coordination space control

■ Chemoinformatics



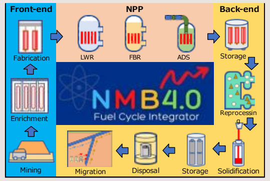
Reverse design, new extractants, solvent exploration

Integration of Nuclear Fuel Cycle

■ Nuclear Material Balance Code 4.0

#1 Users in the world (Over 120 users/30 organization)

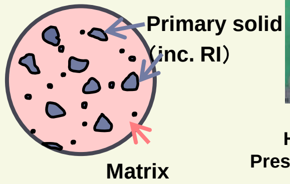
- ✓ Quantitative study of future scenarios for nuclear energy utilization
- ✓ Nuclear energy utilization policy and R&D strategy planning
- ✓ Introduction effects and strategies for introducing innovative nuclear reactors

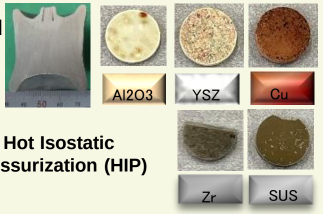


Fukushima Reconstruction & Revitalization

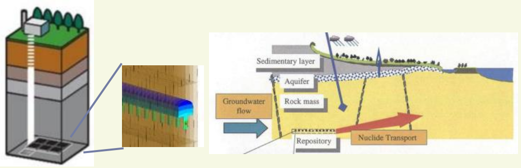
■ Fuel Debris/Waste Management

Integral approach of solidification, disposal & safety assessment





Hot Isostatic Pressurization (HIP)



Rational waste management strategies by utilizing Chemical & Digital technology!

Hybrid Solidification

- Adaptable to a wide variety of wastes by a single concept
- Matrix characteristics allow for safety assessment
- Waste: ALPS precipitation, slurry, silver adsorbents, etc.

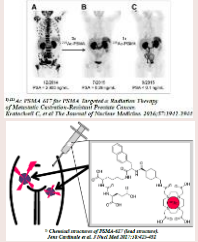
Composite solidification (Artificial rock、SYNROCK)

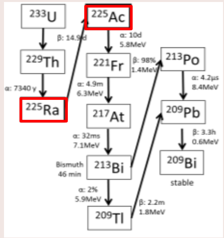
- Fuel debris, more robust solidified than vitrification, natural analog

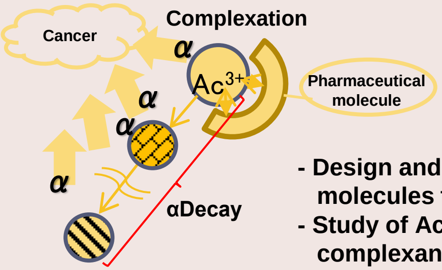
- Chemical analysis
- First Principle Calculations
- Radiation Effects
- Leachability
- Long-term stability

Medical

■ Targeted α radiation therapy





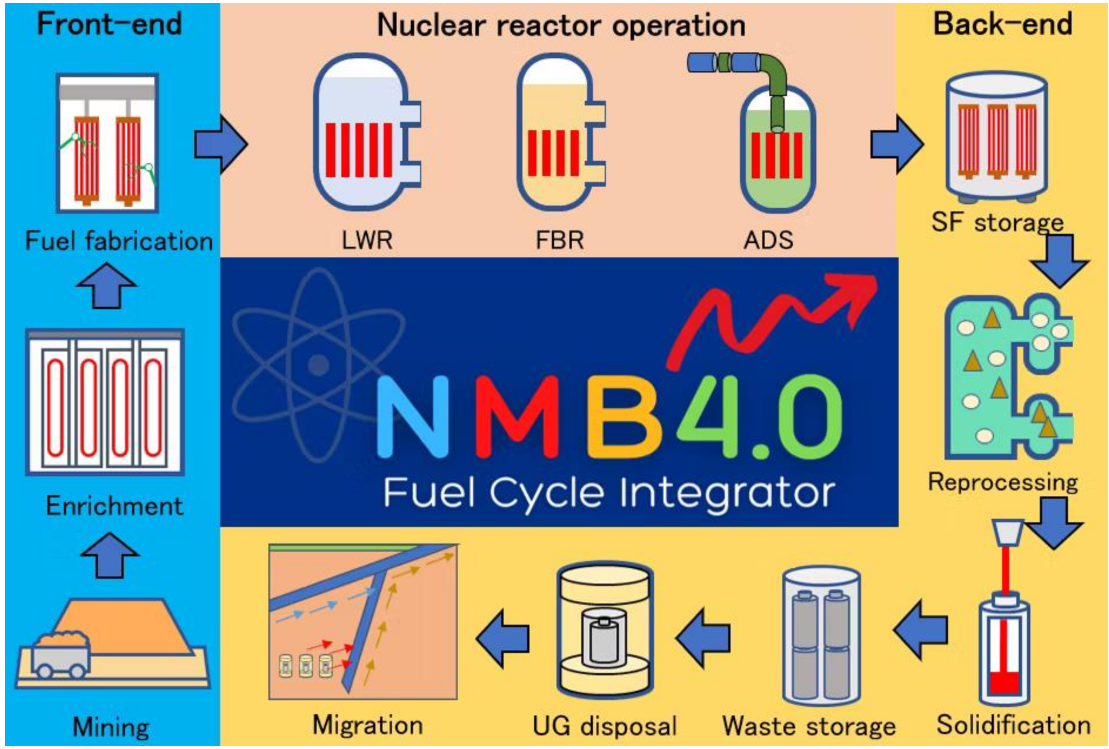


- Design and develop chelator molecules for RI nuclides.
- Study of Ac^{3+} and Ra^{2+} complexant structures

Optimization of Nuclear Fuel Cycle

Necessity of optimization of NFC

- Nuclear energy is essential to achieve a Zero-Carbon society by 2050.
- In Japan, completing a closed NFC to enhance energy security is necessary, and the final disposal is also one of the keys.
- Higher-burnup and spent MOX fuels require MA partitioning for final disposal



Tokyo Tech and JAEA developed an integrated nuclear fuel cycle simulator, "NMB4.0"

Optimized NFC

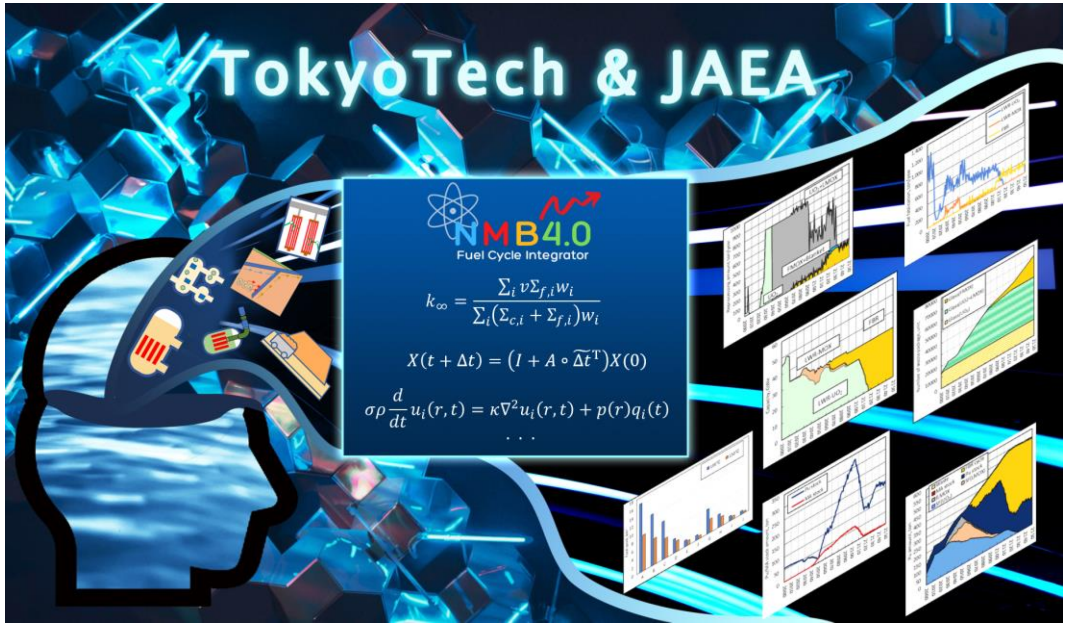
- ✓ Reduction of the volume of HLW and footprint of the geological repository
- ✓ Long-term risk reduction in geological disposal
- ✓ Meet all the criteria of each step of the NFC
- ✓ Recycling the usable material
- ✓ Evaluation of the introduction effect of innovative reactors
- ✓ Extension of the operation of NPPs or replacement?
- ✓ Required performance of the second reprocessing plant

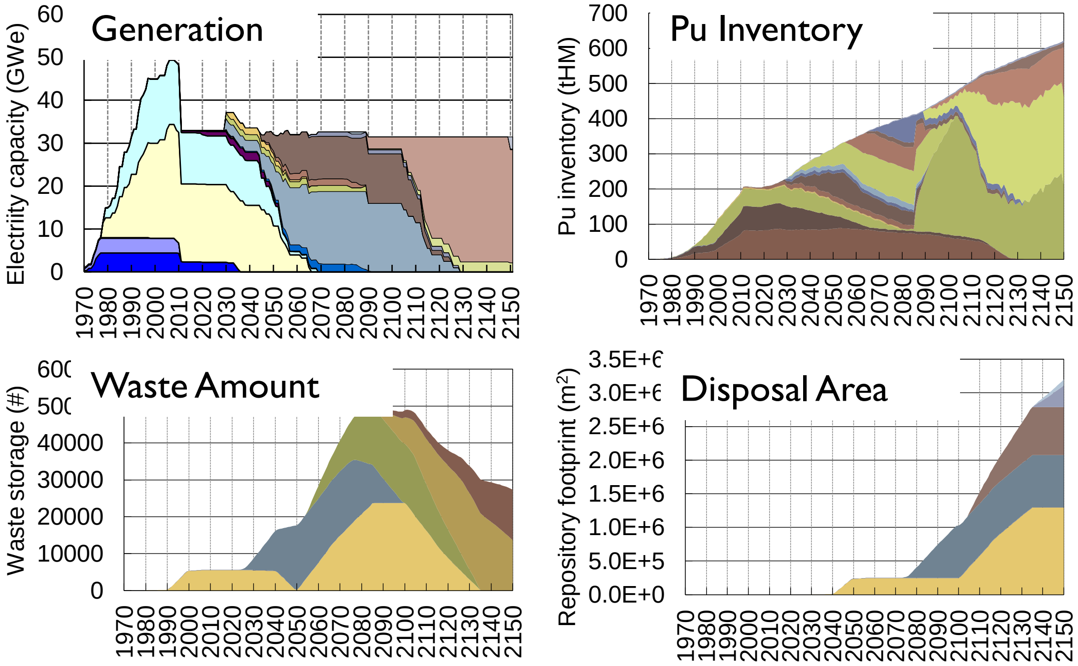
⋮

Fuel Cycle Integrator NMB4.0

Opensource & No.1 scale of user community

Users: 30 organizations 120 users / Policy making & National project



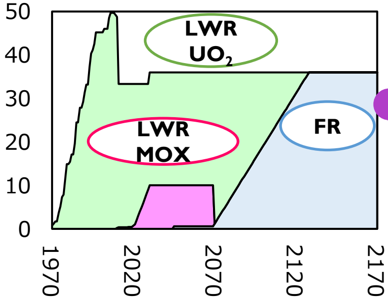


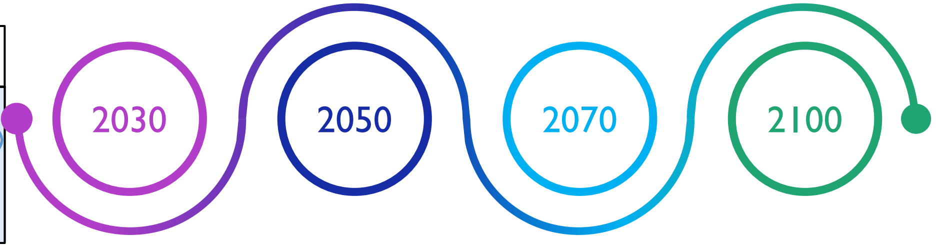
Analysis of the material flow in a virtual space and dynamic simulation are possible.

- Investment/R&D strategies, waste reduction technologies ...
- Detailed information about the facilities(capacity, operation etc), material quantity is needed

How is the optimization of NFC possible?

■ Set the scenario





- NPPs restart
- Rokkaho RPP
- FBR Demonstration
- Replacement to FBR
- LWR/FBR Coexisting
- 2nd RPP start
- Transition to FBR

■ Trace the waste info

Waste(n)
- ･Owner
- ･Waste type
- ･Solidified date
- ･Spent fuels
- ･Inventories
- ⋮

- ･Spent fuel I
- ⋮
- ･Spent fuel n

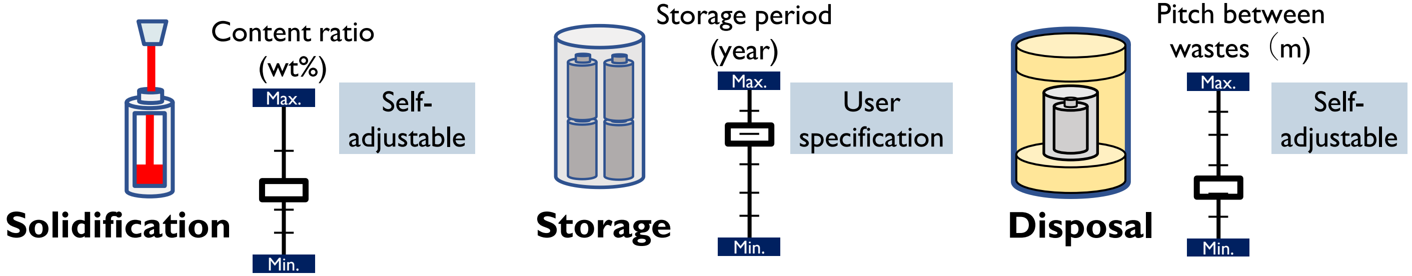
NMB4.0 Fuel Cycle Integrator

- ･Owner
- ･Fuel type
- ･Operation period
- ･Burn up
- ･Fabrication
- ･Inventories …

■ Adjustment of the design parameters

Solidification + Storage + Disposal

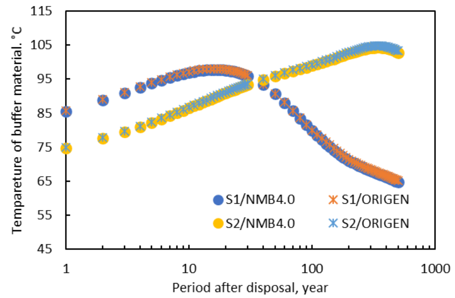
Combination of solidification, storage and disposal is specified for each of waste.

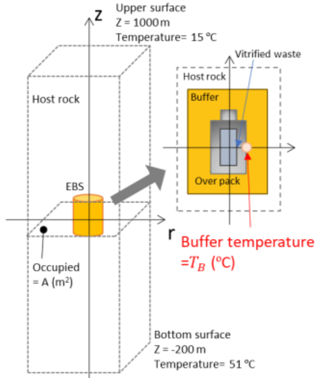


NMB automatically adjusts solidification, storage, and disposal conditions followed by generated waste from each reactor
⇒ Evaluation of the scale of repository and safety assessment.

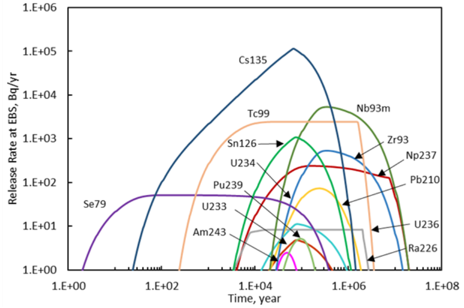
■ Scale of repository & Safety assessment

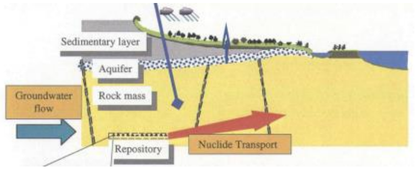
□ Thermal analysis of repository



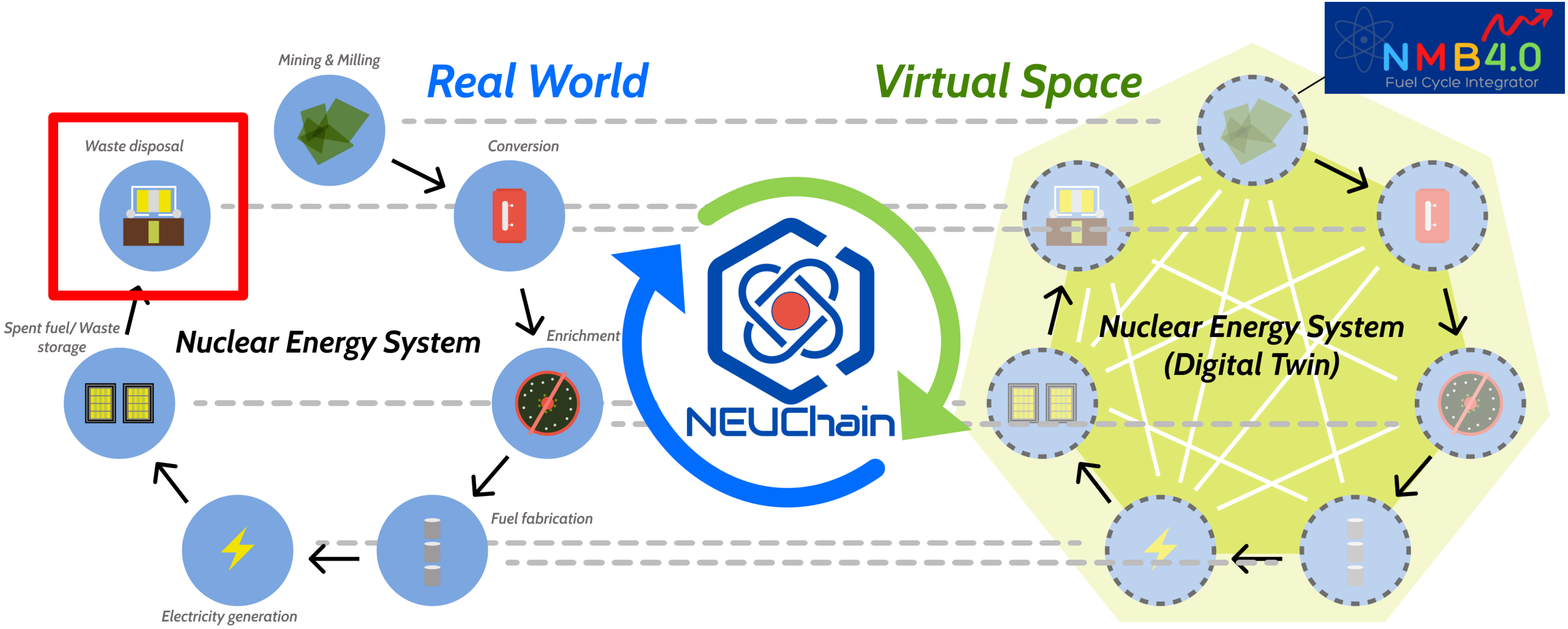


□ Migration and safety analysis





The next-gen Digital Twin Solution of Nuclear Energy Systems



The digital twin, which combines real-world and simulated data from NMB with NEUChain technology as an interface, enables further detailed discussions on the scenario, security, and knowledge management.

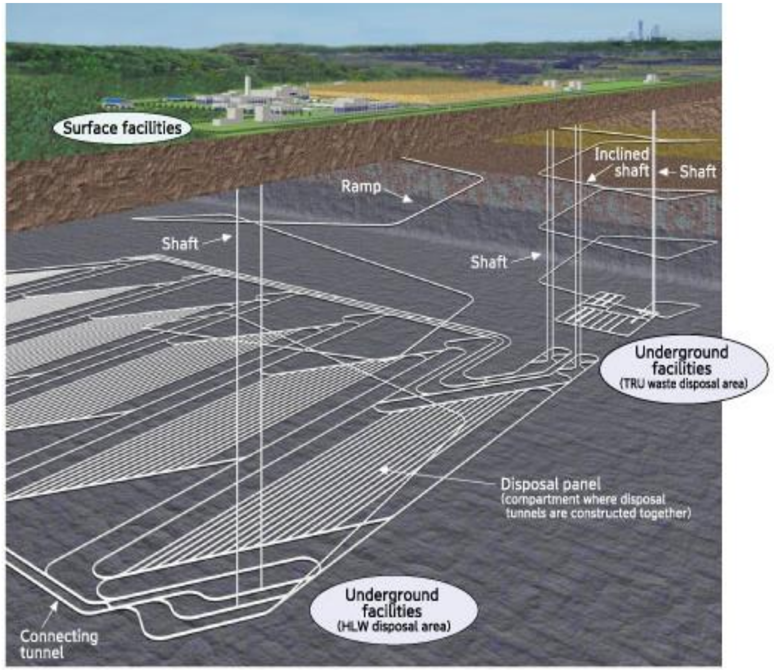
Geological disposal – Japanese case

Geological disposal area (Assumption)

- Vitrified waste (*40,000 unit); $6km^2$
- TRU (*19,000m^3) ; $1.5km^2$
- Total foot print; 10 km^2

Simple estimation (calculation) based on single reference vitrified glass

Deeper than 300m below the surface



Source : Nuclear Waste Management Organization of Japan

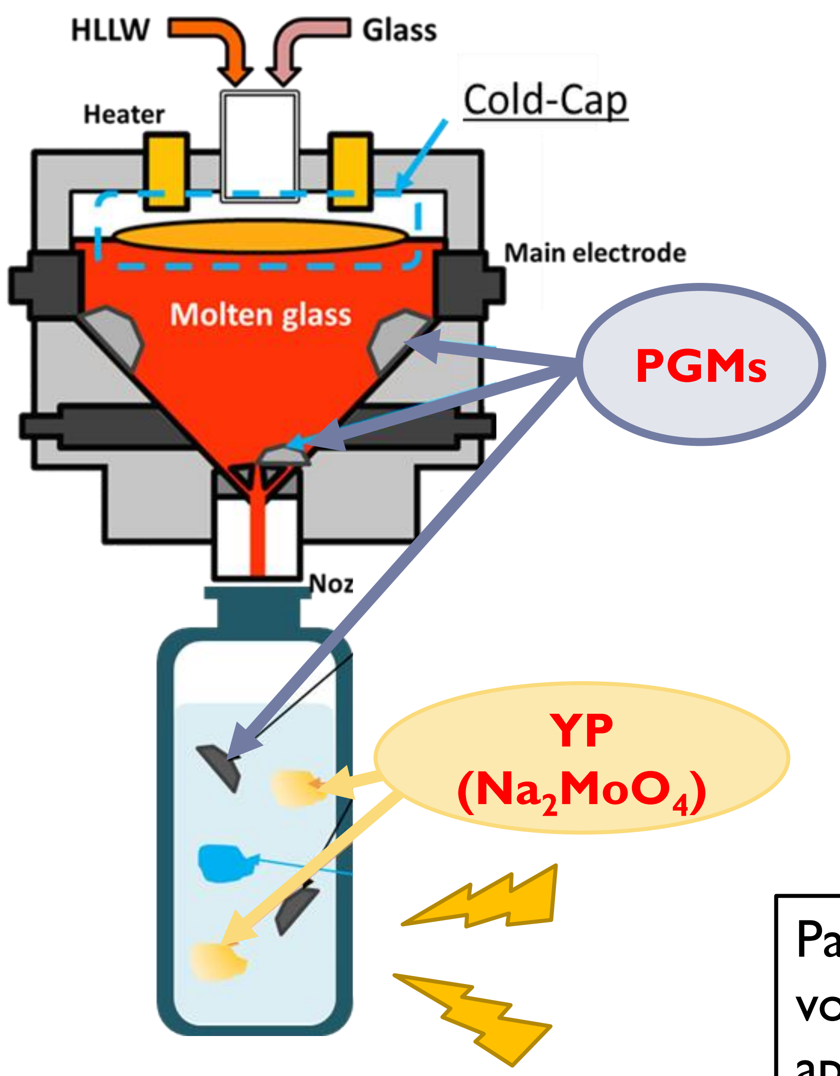
✓ The repository site may not always be uniform, especially in Japan. Then, a more detailed and flexible design concept may be needed.

- Small and decentralized emplacement

✓ An efficient design study considering the inventories based on numerical simulation is practical.

- Full-scale calculation at reduced cost, considering the entire NFC

Criteria for vitrified wastes



✓ Stable operation of glass melter (Japanese type)
✓ Warrant the quality of vitrified waste
(Mo; ≦ 1.5 wt%, PGMs; ≦ 1.25 wt%*1)

➡ Mo・PGMs（Ru・Rh・Pd） removal

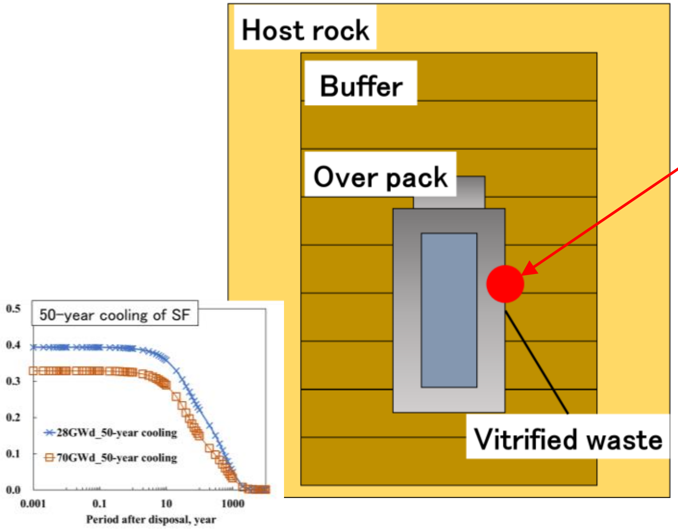
✓ Cooling requirement on storage facility
(Ave. ≦2.30 kW/unit, Max = 2.80kW/unit*2)

➡ Cs, Sr, MA (Am) removal

*1. Y. Inagaki, et,al., J Nucl Sci Tech, 46, 677-689, 2009
*2. Nuclear Regulation Authority

Partitioning can reduce the amount of HLW but may increase the volume of low-level waste instead, and the removed elements must be appropriately treated.

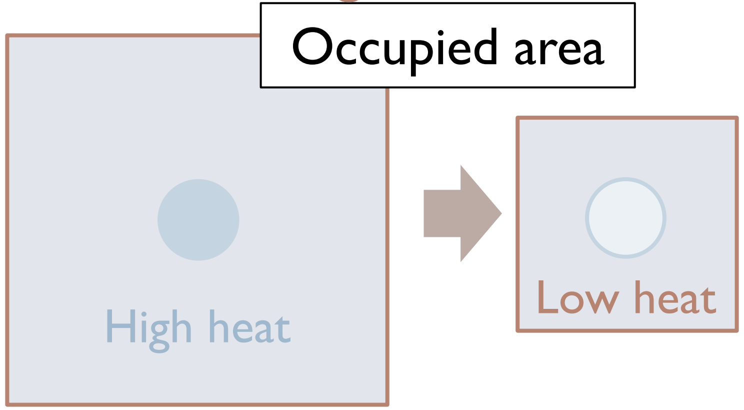
Volume and number of the vitrified wastes



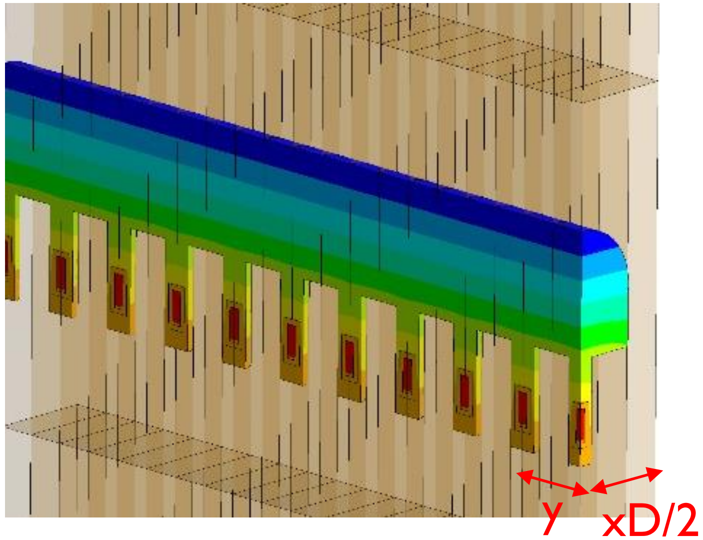
- Upper-temperature limit of buffer bentonite (≦100℃ to prevent Illitization of bentonite)

High heat generation ⇒ Larger Occupied area

Lower heat generation ⇒ Smaller Occupied area



The chemical property also governs the waste loading limit, including Na content, to enable the melted glass to pass through the nozzle.



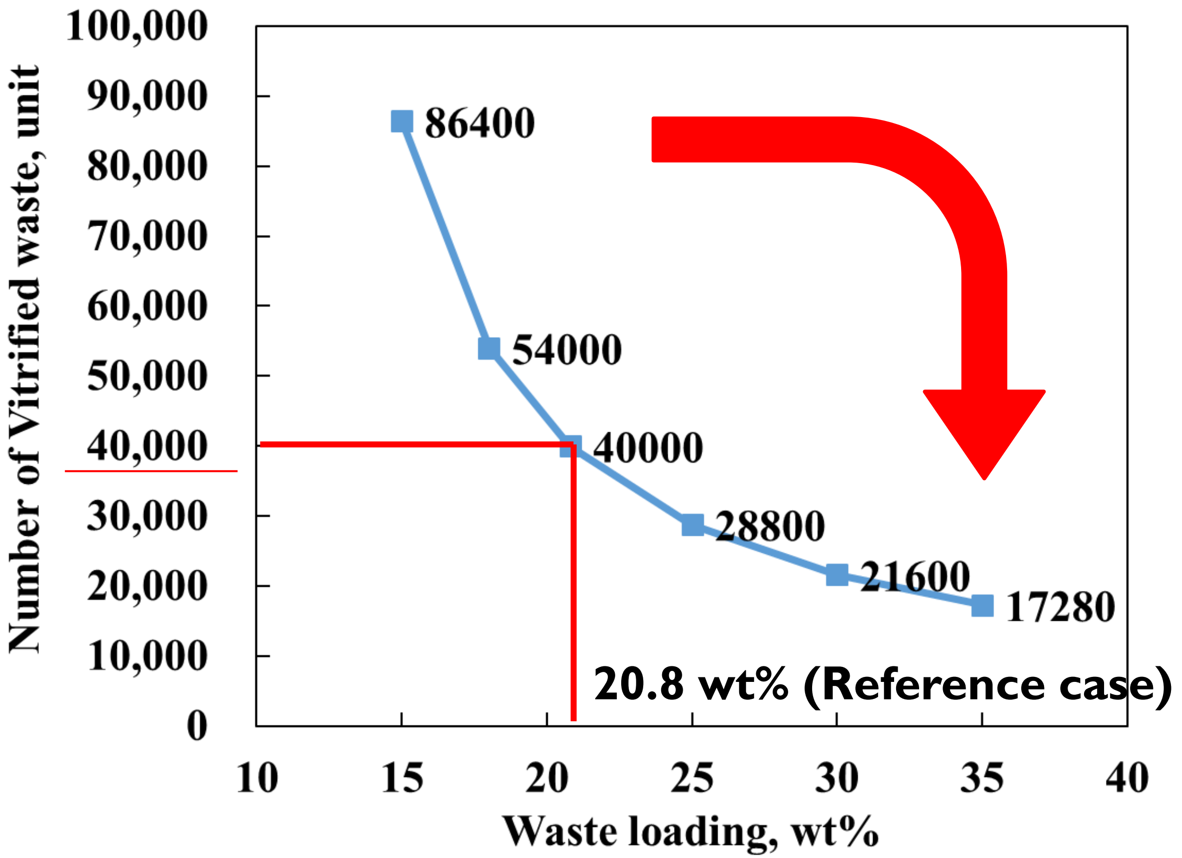
Cs, Sr, Am removal or longer storage period?

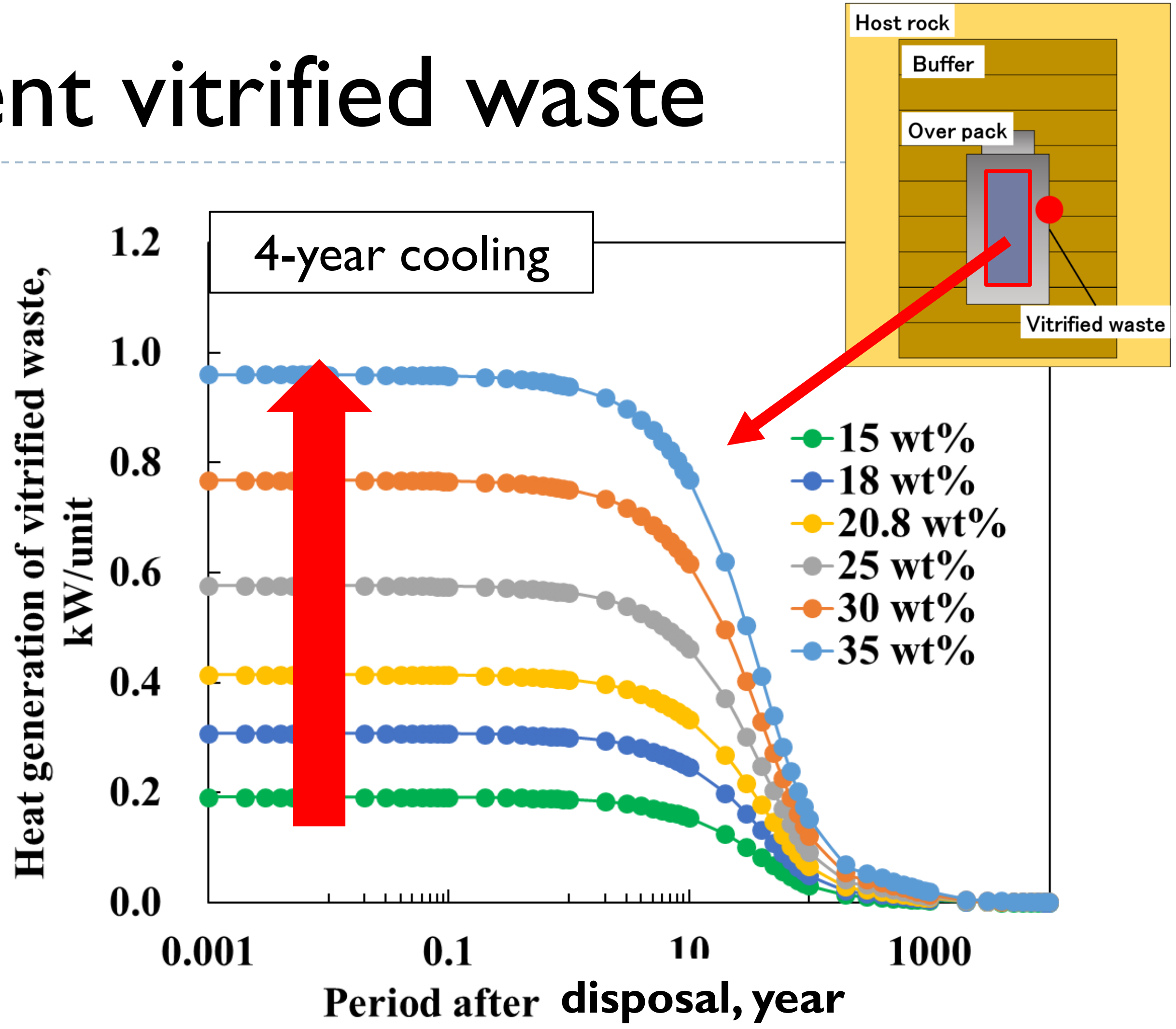
Key design parameters; Pitch(y) and tunnel distance (xD)

→ **$S\ [m^2] = y\ [m] \times xD\ [m]$** should be minimized.

*Actual wastes don't always emit uniform heat

Effect of high content vitrified waste





- ✓ The number of vitrified waste decreased with an increase in waste loading.
- ✓ The heat generation rate arises with an increase in waste loading.

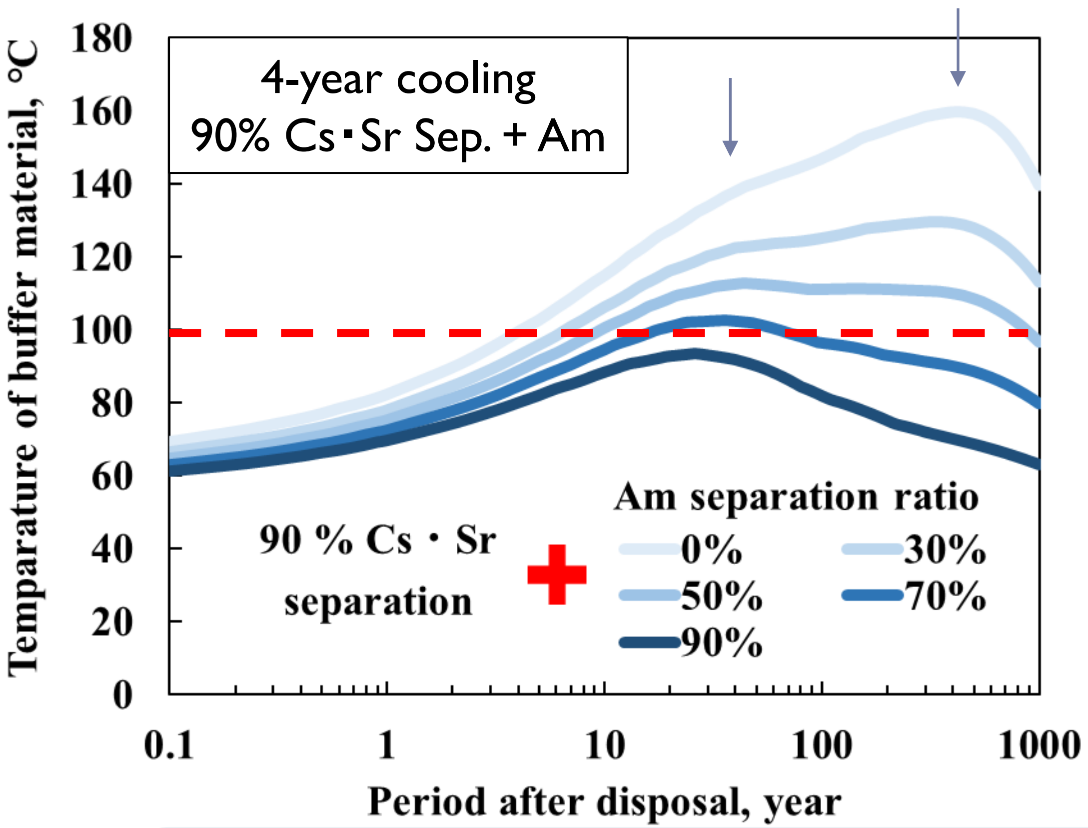
Possibility of expanding the disposal area depending on the waste loading.

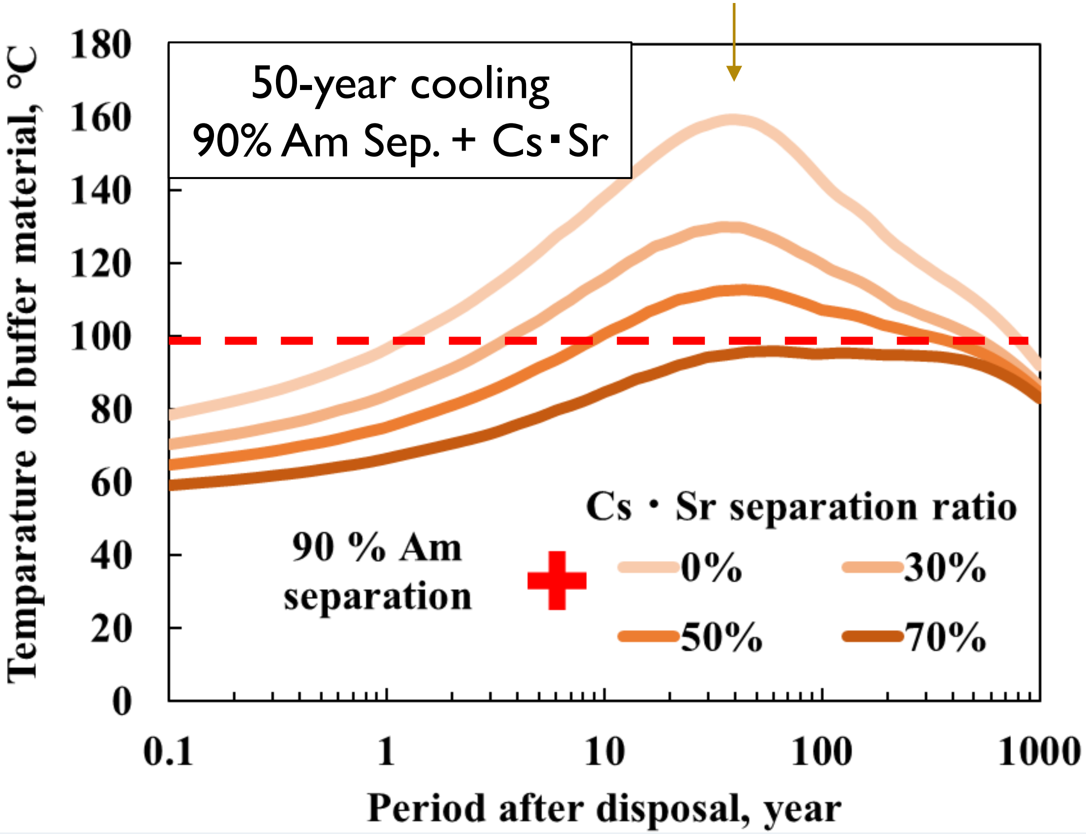
Effect of Cs･Sr & Am separation

Horizontal

Required

- ☐ 35wt%, Mo･PGMs 70% separation, 13.9 m²





- ✓ Combination of Cs･Sr＋Am separation is effective(timing of the temp limit is different).
- ✓ Waste contents, emplacement method, and repository layout govern max buffer temperature.
- ✓ Rough separation of heat-emitting nuclei and increase in waste loading is effective.
- ✓ PGM limit consideration is needed when waste loading is increased.

Impact of partitioning on reducing footprint

CP; Cooling Period WL;Waste Loading

<table>
<tr><th>CP of SF, year</th><th>Mo・PGM, %</th><th>Cs・Sr, %</th><th>Am, %</th><th>WL, wt%</th><th>Emplacement method</th><th>CAERA, kg/m^2</th><th>Reduction Effect</th></tr>
<tr><td>4</td><td>0</td><td>0</td><td>0</td><td>20.8</td><td>Vertical</td><td>0.97</td><td>0%</td></tr>
<tr><td>4</td><td rowspan="11">70</td><td>90</td><td>0</td><td rowspan="11">35</td><td rowspan="2">Vertical</td><td>2.25</td><td>-56.8%</td></tr>
<tr><td>50</td><td>0</td><td>90</td><td>2.25</td><td>-56.8%</td></tr>
<tr><td rowspan="5">4</td><td rowspan="5">90</td><td>0</td><td rowspan="9">Horizontal</td><td>3.26</td><td>-70.2%</td></tr>
<tr><td>30</td><td>2.94</td><td>-67.0%</td></tr>
<tr><td>50</td><td>4.17</td><td>-76.7%</td></tr>
<tr><td>70</td><td>5.63</td><td>-82.7%</td></tr>
<tr><td>90</td><td>7.20</td><td>-86.5%</td></tr>
<tr><td rowspan="4">50</td><td>0</td><td rowspan="4">90</td><td>2.94</td><td>-67.0%</td></tr>
<tr><td>30</td><td>4.17</td><td>-76.7%</td></tr>
<tr><td>50</td><td>5.63</td><td>-82.7%</td></tr>
<tr><td>70</td><td>7.20</td><td>-86.5%</td></tr>
</table>

Reference

10~15% UP

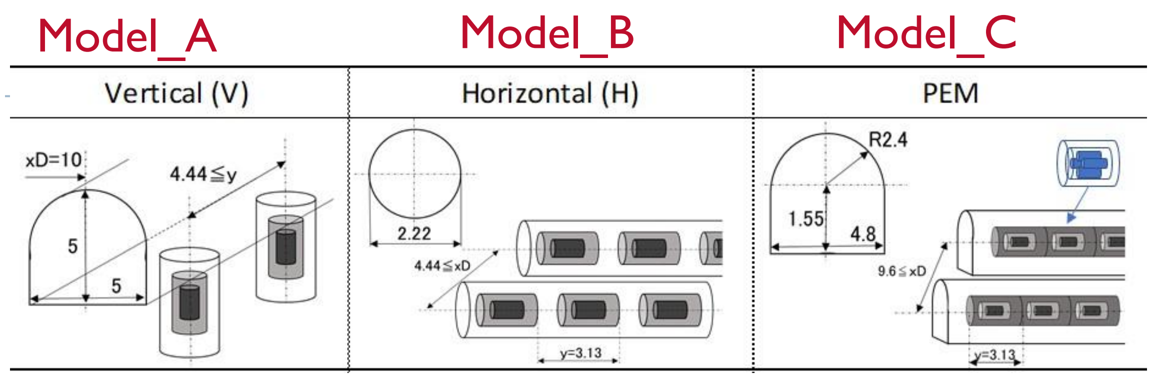
MAX

MAX

- Many iterative calculations were required to find the minimum size of the repository.
- A more detailed thermal design study is needed to implement the disposal.

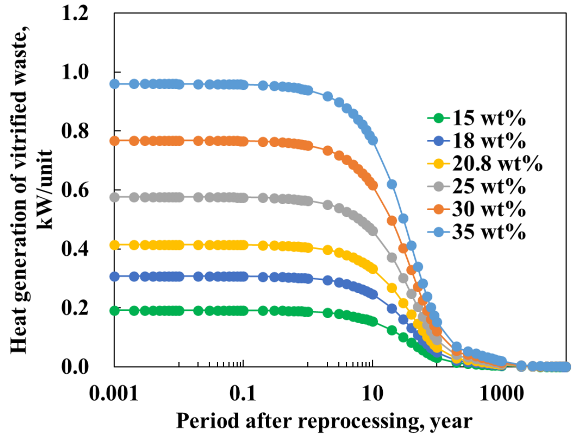
Efficient calculation of thermal design of repository

Calculation Flow

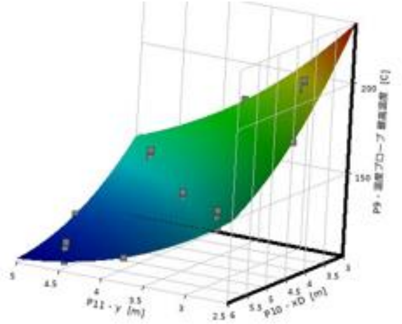


1. Prepare the engineering data
2. Select the emplacement type and create the geometry and the strategy of meshing
3. Set the heat source of vitrified waste

Calculation by ORIGEN



4. Set the geometric parameters
5. Create the response surface



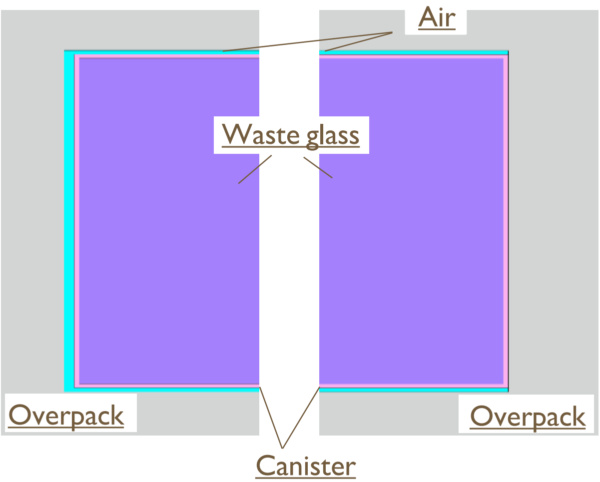
6. Criteria; max buffer temperature < 100℃, Target; smallest S [m^2]

➡ Find the candidate layout by response surface combined with Machine Learning

Skip the iterative thermal conductive calculation

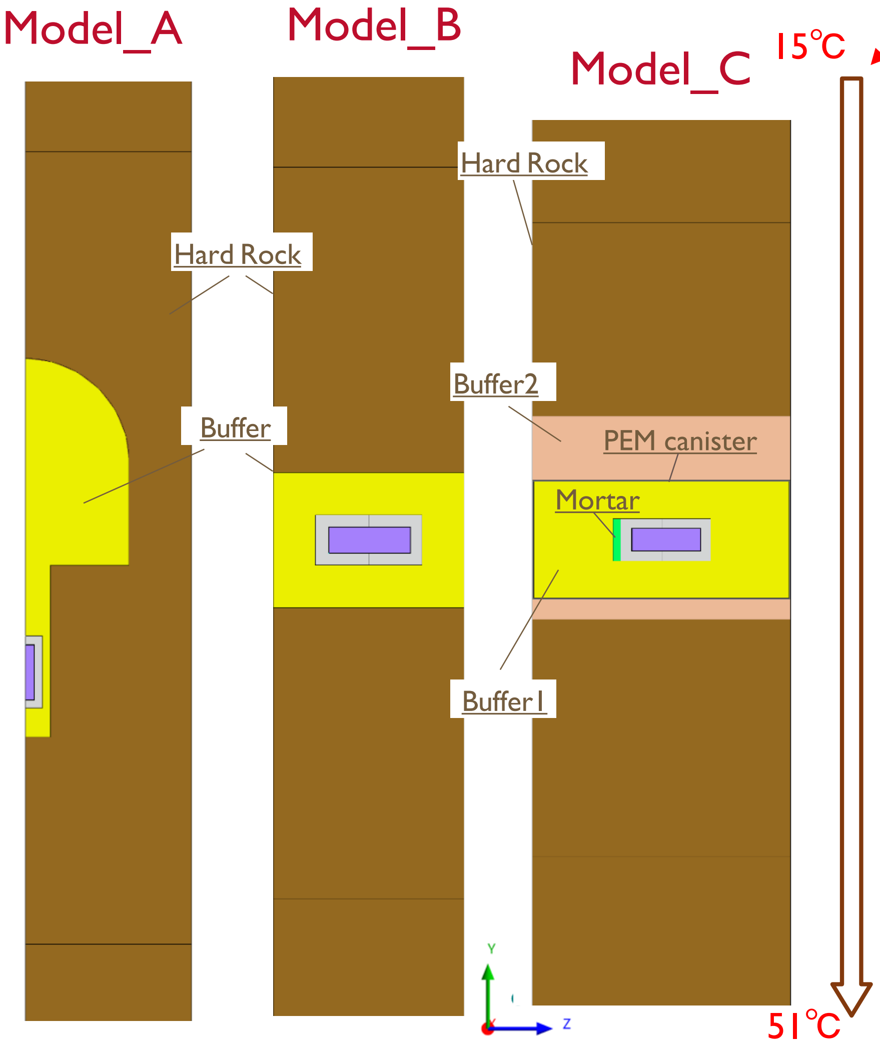
7. Confirmation calculation to check whether all the criteria were met with the suggested geometries → The smallest repository area is obtained!

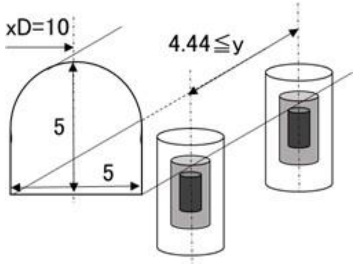
Geometry, mesh and material properties



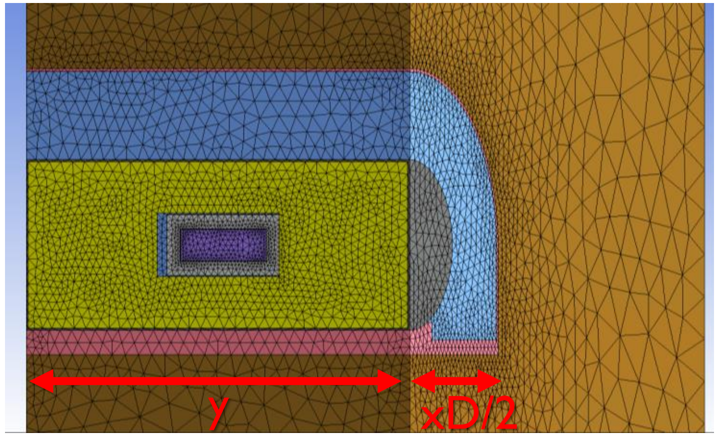
	Density [kg/m³]	Thermal conductivity [W/mK]	specific heat [J/kgK]
Hard Rock	2670	2.8	1000
Buffer	1710	0.78	590
Air	1.21	0.0256	1010
Overpack	7860	51.6	473
Canister	7920	16	499
Waste glass	2740	1.05	884
PEM container	7920	16	499
Mortar	2000	0.55	880

Rock temperature increases by 3 °C every 100 m





Valuables: depth, size of the waste, pitch and distance (engineering limit exists)

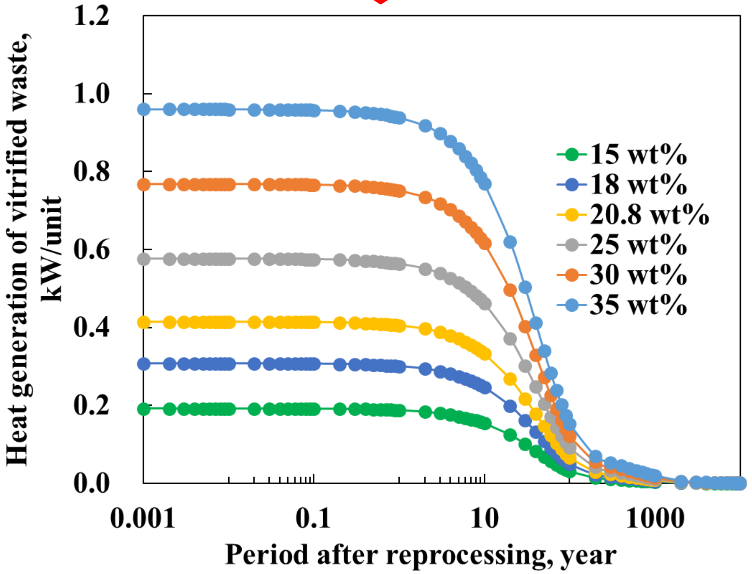


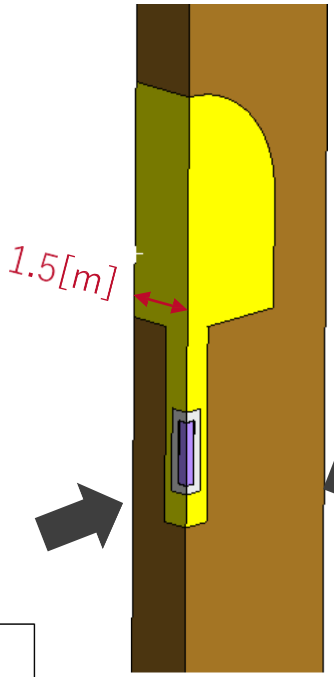
The mesh is automatically renewed in good quality!

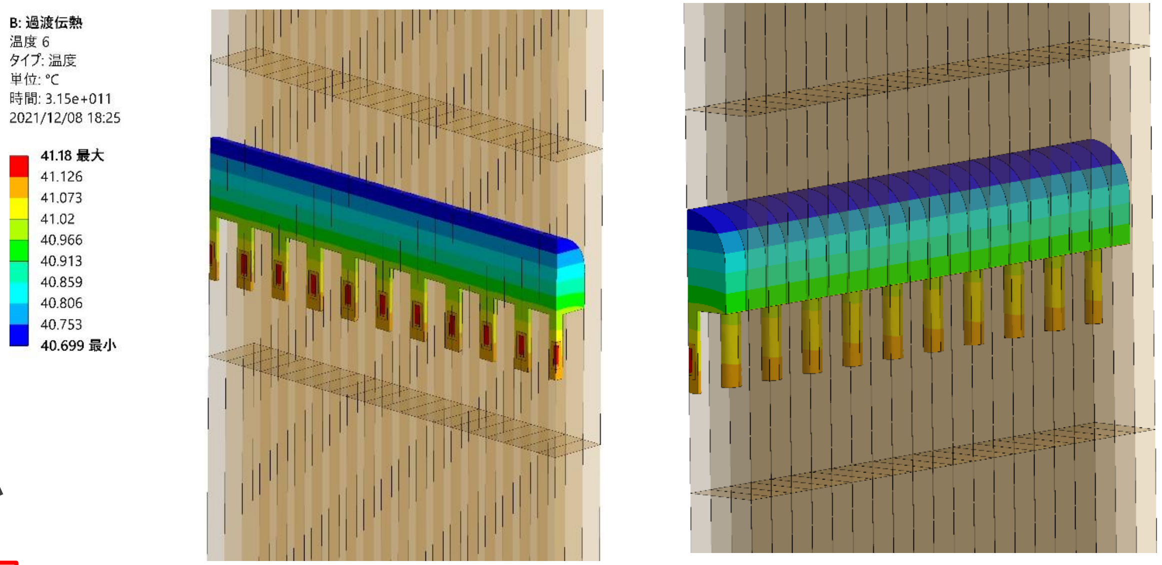
Thermal calculation

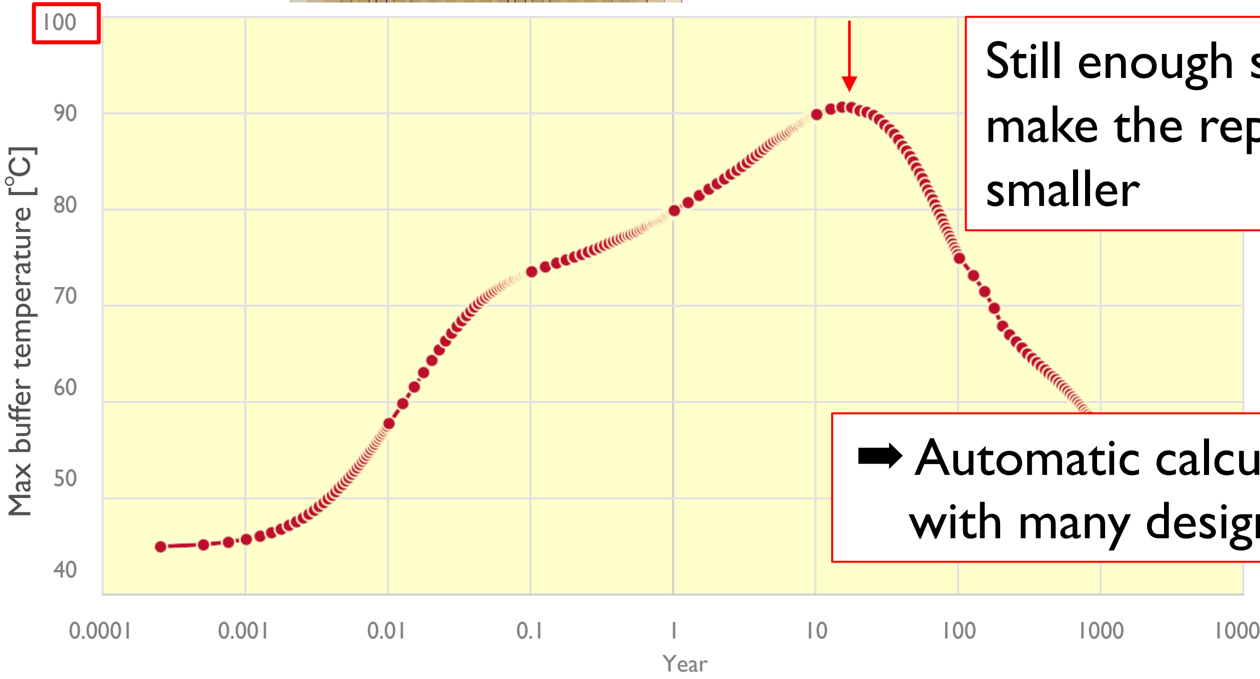
Vertical emplacement

Reactor type, Plant operation, Cooling period, Reprocessing, Waste loading, Storage period







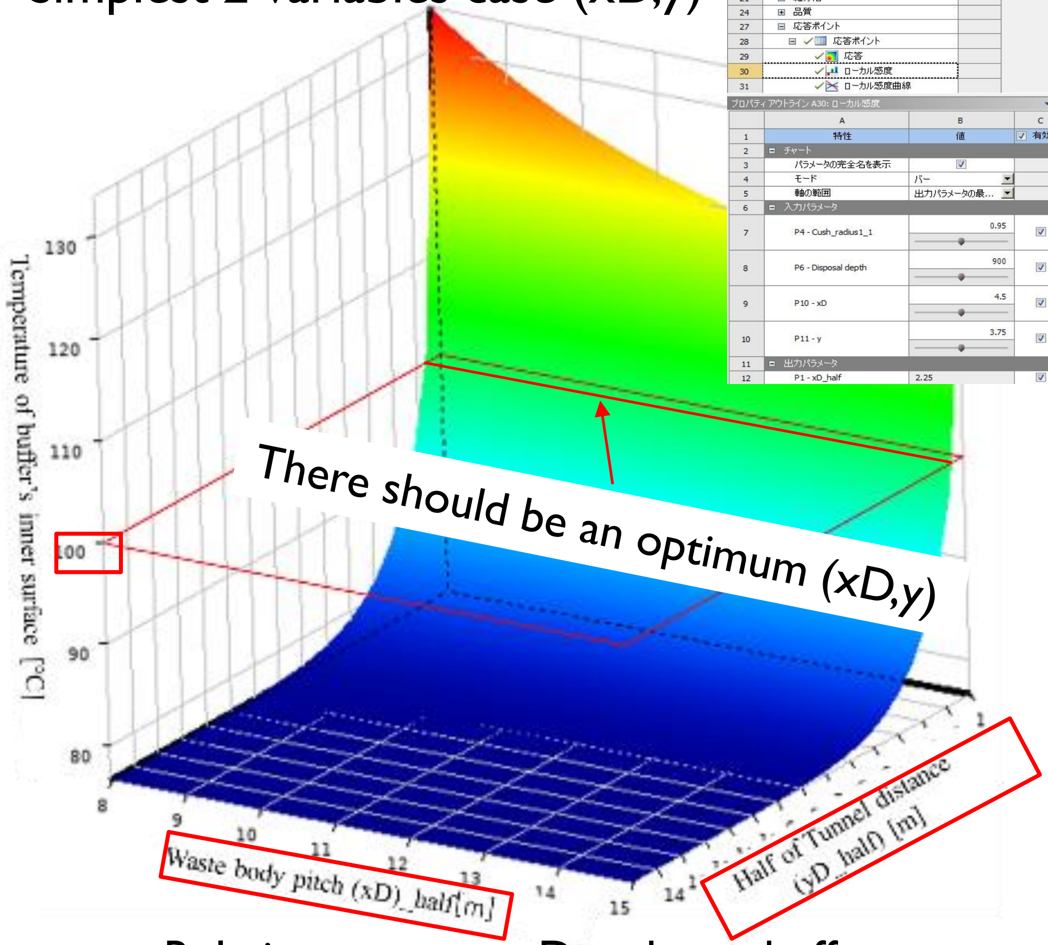


Still enough space to make the repository smaller

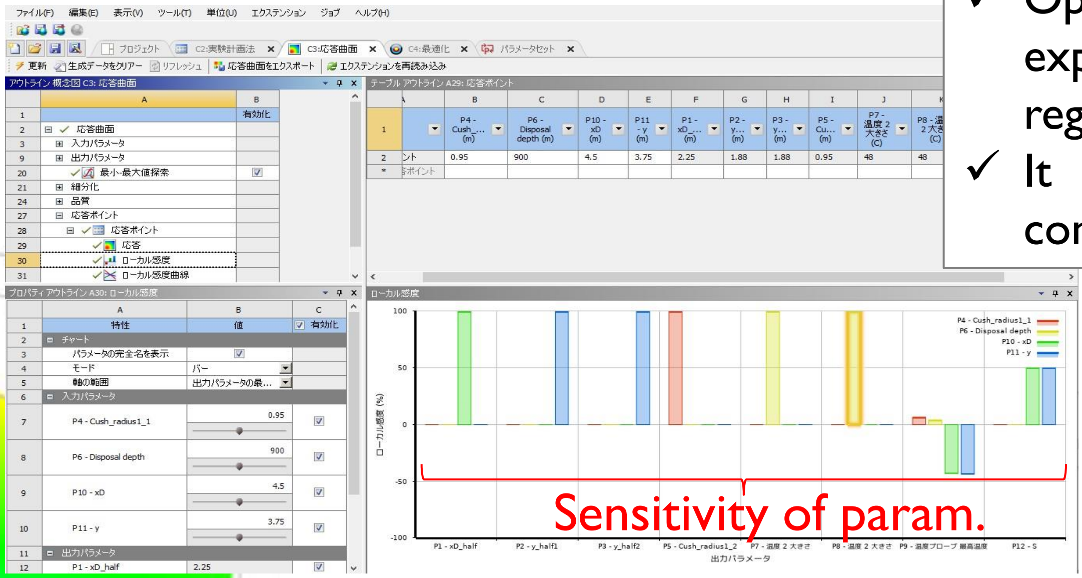
➡ Automatic calculation with many design points!

Response surface

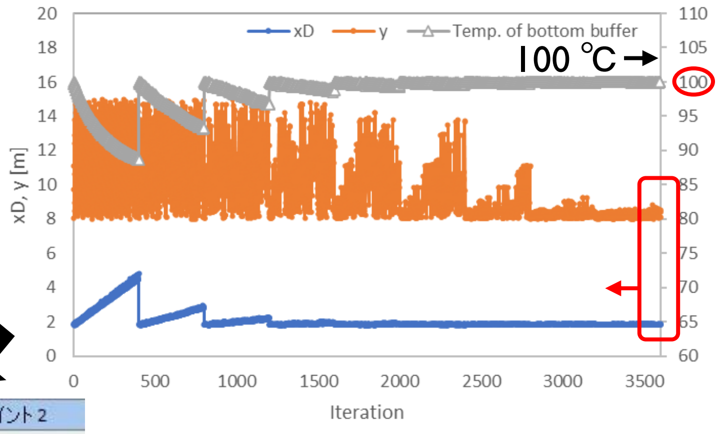
Simplest 2 variables case (xD,y)



Relation among y, xD and max buffer temp.
(There are more influential factors)



- ✓ Optimum geometry (xD, y) exploration by response surface with regression or machine learning.
- ✓ It is fast because the thermal conductivity calculation is omitted.



Suggested geometries

9		候補ポイント 1	候補ポイント 2
10	P4 - Cush_radius1_1 (m)	0.503	0.505
11	P6 - Disposal depth (m)	803	806
12	P10 - xD (m)	5.54	5.66
13	P11 - y (m)	4.79	4.72
14	P9 - 温度プローブ 最高温度 (C)	★★★ 100	★★★ 99.9
15	P12 - S (m^2)	✕✕ 26.6	✕✕ 26.7

Confirmation calculation

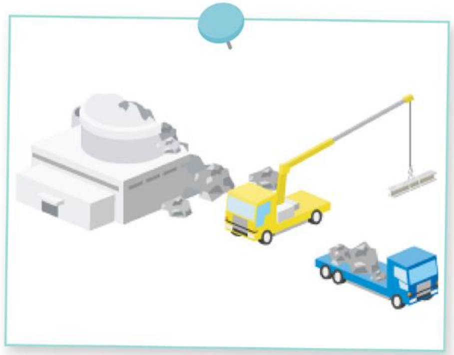
Optimum design

Further study
- ☐ Consider the diverse vitrified waste (heat-emission is different)
- ☐ Large (whole)-scale thermal calculation: models to find the reasonable waste layout
- ☐ Combination with NMB and NEUChain; Digital twins

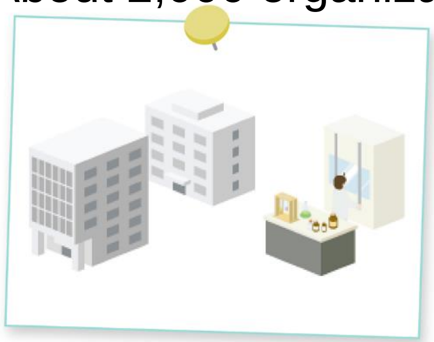
Expand the digital twin to other wastes

× Low-level, Experimental, Medical RI wastes

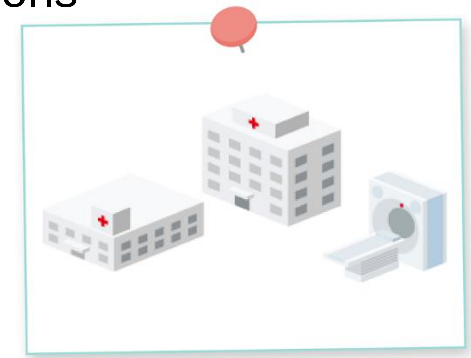
About 2,000 organizations



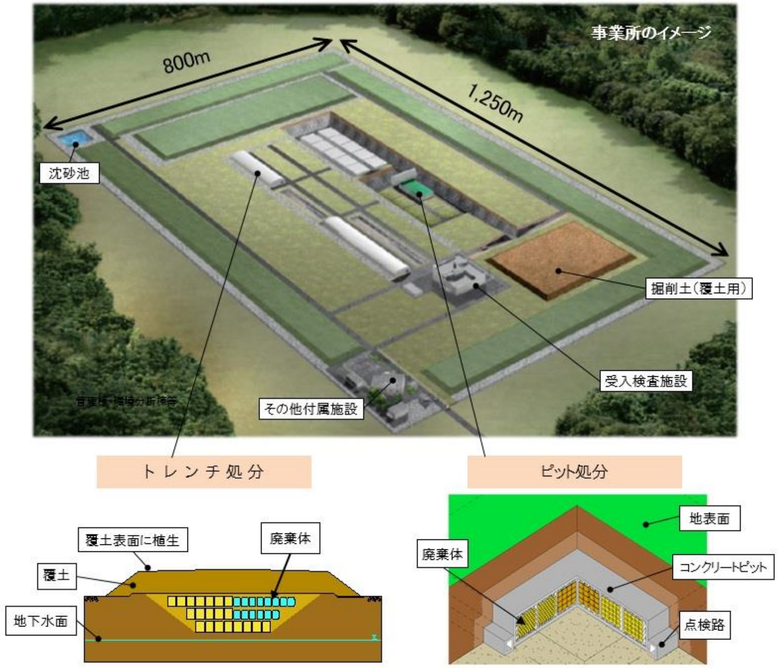
Decommisoning: Research Reactors (Except commercial)



Experiment: Univ., Research Institute

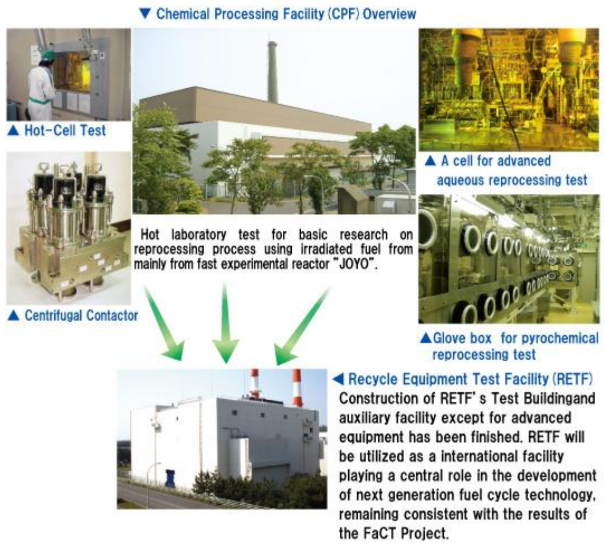


Medical: Hospital



Pictures: https://www.jaea.go.jp/04/maisetsu/

JAEA has to handle disposal of radioactive waste from various sources across Japan. The scale of the planned facility is about 750,000 waste bodies (220,000 for pit disposal and 530,000 for trench disposal) in terms of 200-liter drums.



STRAD Project: Treatment of radioactive liquid waste @ JAEA

- ✓ Promote the R&D for treatment of a wide variety of legacy radioactive liquid waste stored in the Chemical Processing Facility "CPF"
- ✓ NEUChain participates in the development of data management system

Digital twin for 1F decommission

Tokyo Tech | Open Innovation Platform | × NEUChain TEPCO

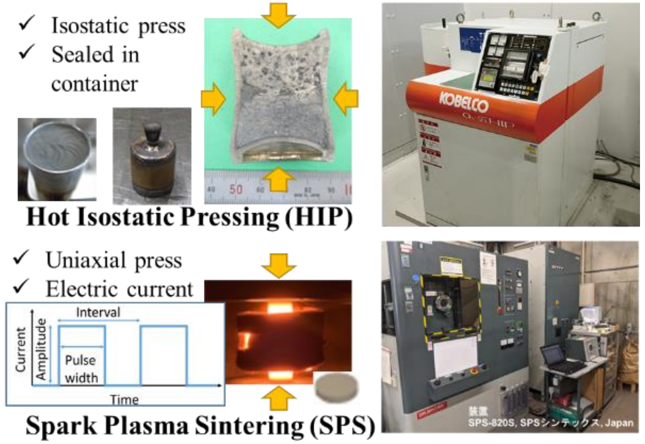
TEPCO Collaborative Research Centre for the Creation of Frontier Technologies for Decommissioning

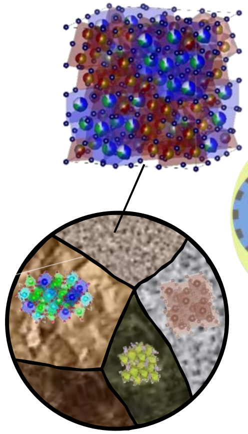
Prof. Takeshita

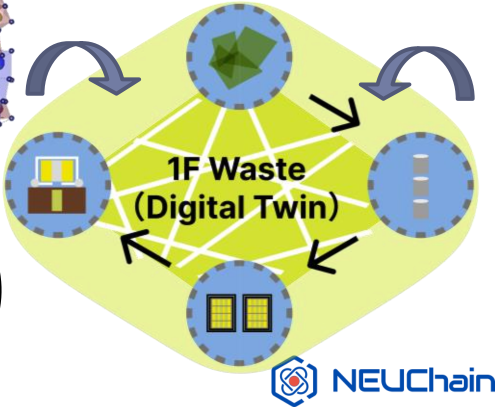
✓ TEPCO's technology and field experience + Tokyo Tech's research capabilities
✓ Working together to pursue solutions to TEPCO's "desire" for decommissioning projects

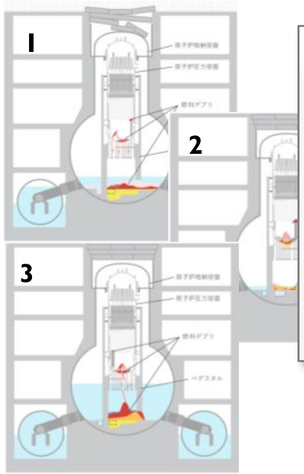
Research topics

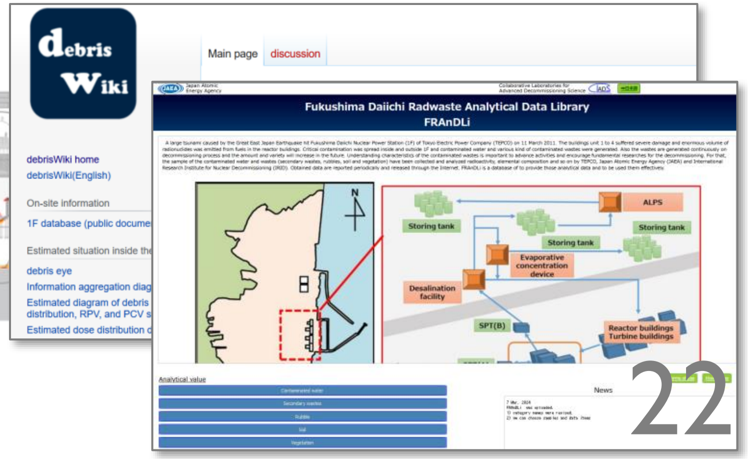
- Mechanisms of fuel damage progression
- Fuel debris re-criticality prevention
- On-line alpha emission and analysis
- Disposal of fuel debris→SYNROC by SPS
- Data management & System integration
- Volume reduction and solidification of radioactive solid waste
- Reactor building leakage investigation technology
- High-performance air purification systems
- Extreme environment sensing











Conclusion

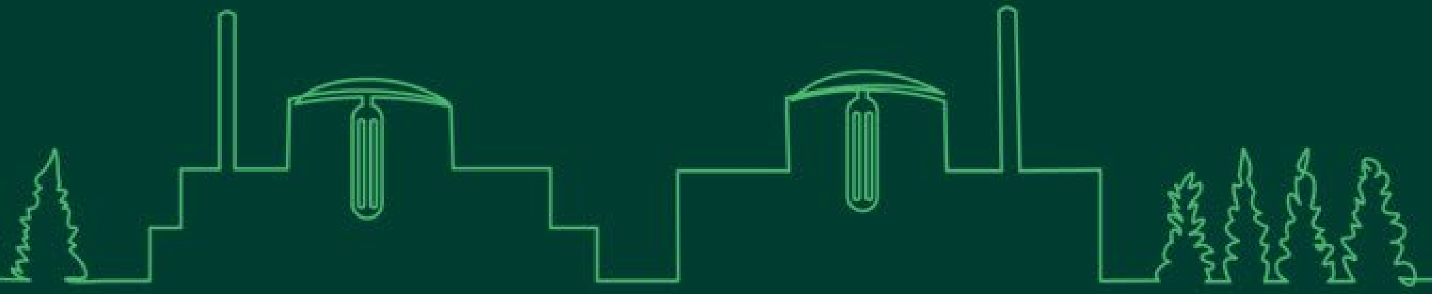
- Rationalization and optimization of NFC with introducing new technology are important. Digital twins approach combined with NMB and NEUChain platform is one of the option.
  - Efficient calculation of the thermal design of a repository can enhance the effect of digital twins. There are some chances to use machine learning to take the best of digital twins.
- We should consider how innovative technology should be installed while connecting the existing NFC. Waste should be carefully considered from the beginning.
- Strategic study based on the mass balance calculation (dynamic NFC simulation) is needed to explain why the technology must be developed and installed to the system.
  - Evaluation from the disposal viewpoint may be the most important for the public people.
- Japan has to complete 1F decommissioning projects including 1F waste disposal. We should consider the current NFC concept, but it may be a chance to add some new idea.
- It is essential to attract younger researchers in the disposal field. To do so, combining the digital technology is one of a good solution

Acknowledgement

This research was supported by METI projects from Radioactive Waste Management Funding and Research Center and TEPCO Collaborative Research Centre for the Creation of Frontier Technologies for Decommissioning in Tokyo Tech.

30 years of operating experiences from Loviisa NPP's LILW final repository

7th International Conference on Geological Repositories

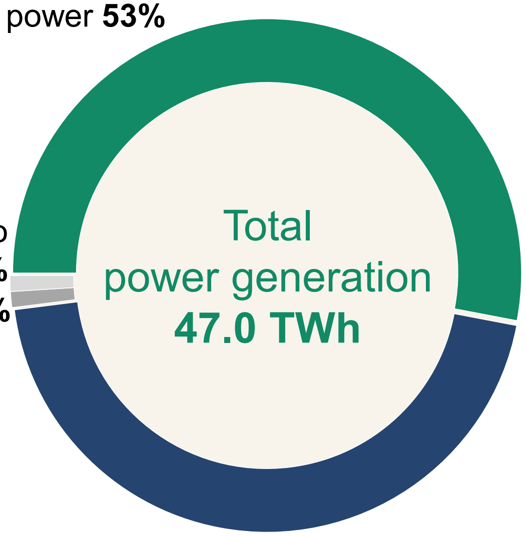


Antti Ketolainen, Fortum

29 May 2024, Busan, Korea

Fortum Nordic based clean energy company - strong in nuclear

Fortum's power generation in 2023



Key figures 2023

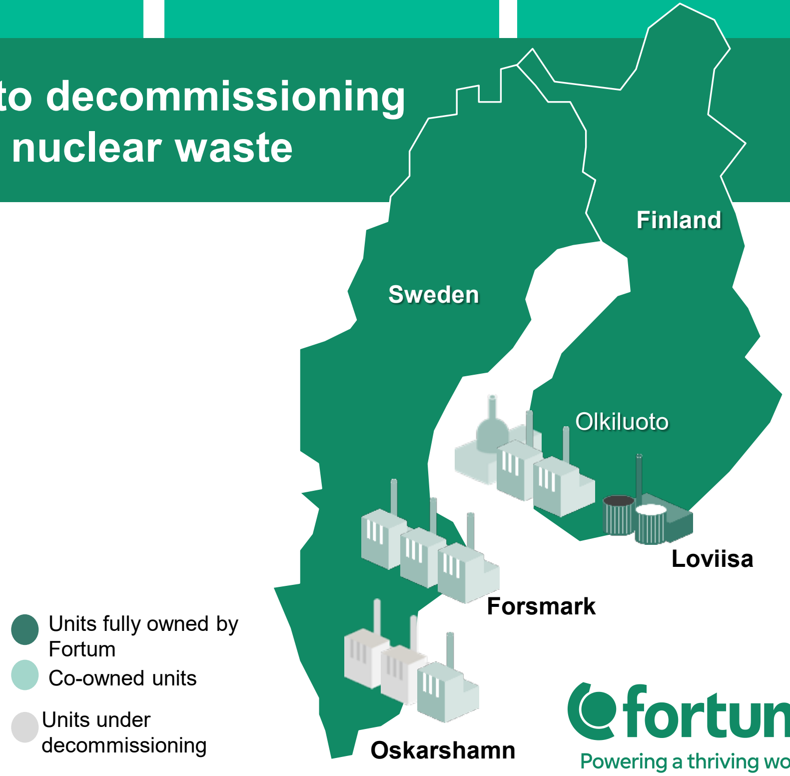
Nuclear generation	**24.8 TWh**
Total nuclear capacity	**3.2 GW**
Share of Fortum's total power generation	**53%**
Nuclear professionals	**~750**

We have 40+ years' track record of safe nuclear operations and we are forerunners in responsible waste management

- Fully-owned nuclear power plant in Loviisa, Finland
- Co-owned nuclear power plants in Finland and Sweden
- Nuclear services provider with innovative products and services
- New Nuclear Feasibility Study in Finland and Sweden
- In-house engineering and project competences

Expertise from new build to decommissioning and final disposal of nuclear waste

Unit	Mwe (net)	Fortum Share %
Loviisa 1	507	100
Loviisa 2	507	100
Olkiluoto 1	890	26.6
Olkiluoto 2	890	26.6
Olkiluoto 3	1600	25
Forsmark 1	988	23.4
Forsmark 2	1120	23.4
Forsmark 3	1172	20.1
Oskarshamn 3	1400	43.4
Oskarshamn 1	decom	43.4
Oskarshamn 2	decom	43.4



fortum
Powering a thriving world

Fortum's nuclear services
- covering the entire nuclear power plant lifecycle

Strong in-house nuclear engineering	Nuclear operator experience based on proven solutions	Projects delivered to a global customer base	Proactive and strong co-operation in international nuclear forums

Newbuild, licensing and commissioning

- Licensing and safety design capabilities
- Engineering services for newbuild
- Plant design
- Small modular reactor (SMRs) consulting

Operating and maintenance

- Operational support
- Maintenance and outage optimisation
- Engineering for upgrade and plant modernisation projects, e.g. automation and process renewal

Plant safety and process simulations

- Deterministic Safety Analysis
- Safety guidelines and analysis
- Probabilistic risk assessment
- Radiation safety analyses

Plant modernisation, lifetime management

- Dynamic simulation to define technical requirements for new equipment with Apros®
- Process and instrumentation and control design verification and testing
- Virtual commissioning

Waste management, decommissioning

- NURES® radioactive liquid purification
- Nuclear waste treatment, storage and disposal
- Expertise in final disposal of radioactive waste
- Extensive nuclear decommissioning services

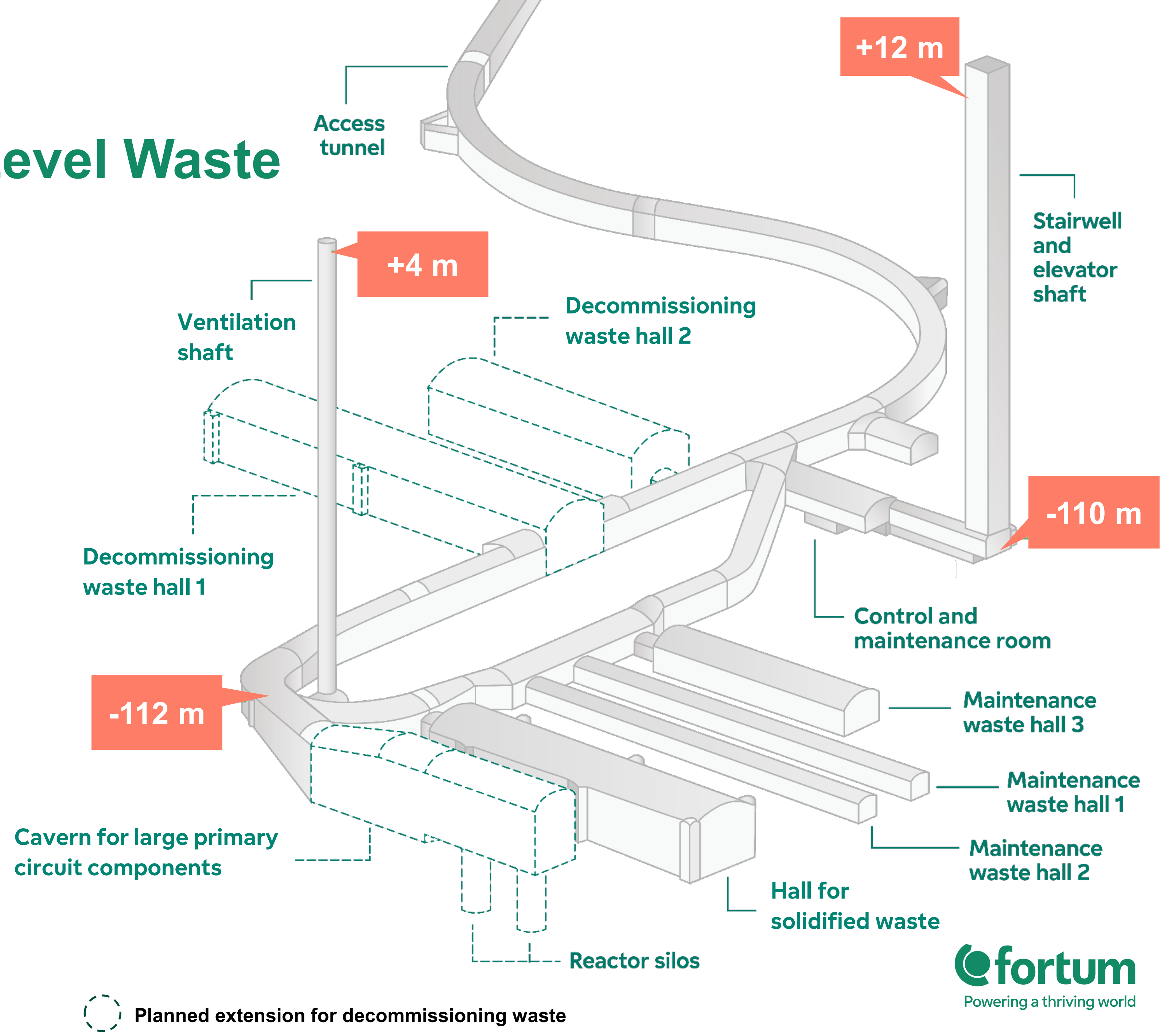
fortum
Powering a thriving world

Final Repository for Low and Intermediate Level Waste

Excavated volume
100 000 m³

Commissioned in **1997**

Access tunnel
1,1 km



fortum
Powering a thriving world

Loviisa LILW repository – location and organisation

- Fortum as NPP and waste disposal facility operator at the same site; enables holistic optimisation of logistics and processes
- NPP's waste management organisation is also responsible for operation of the LILW disposal facility; optimisation of operations

Integration with NPP has positive impact from public acceptance perspective

Safe and sustainable waste management with minimized O&M costs

Seamless and efficient licensing

More efficient waste acceptance criteria determination

fortum
Powering a thriving world

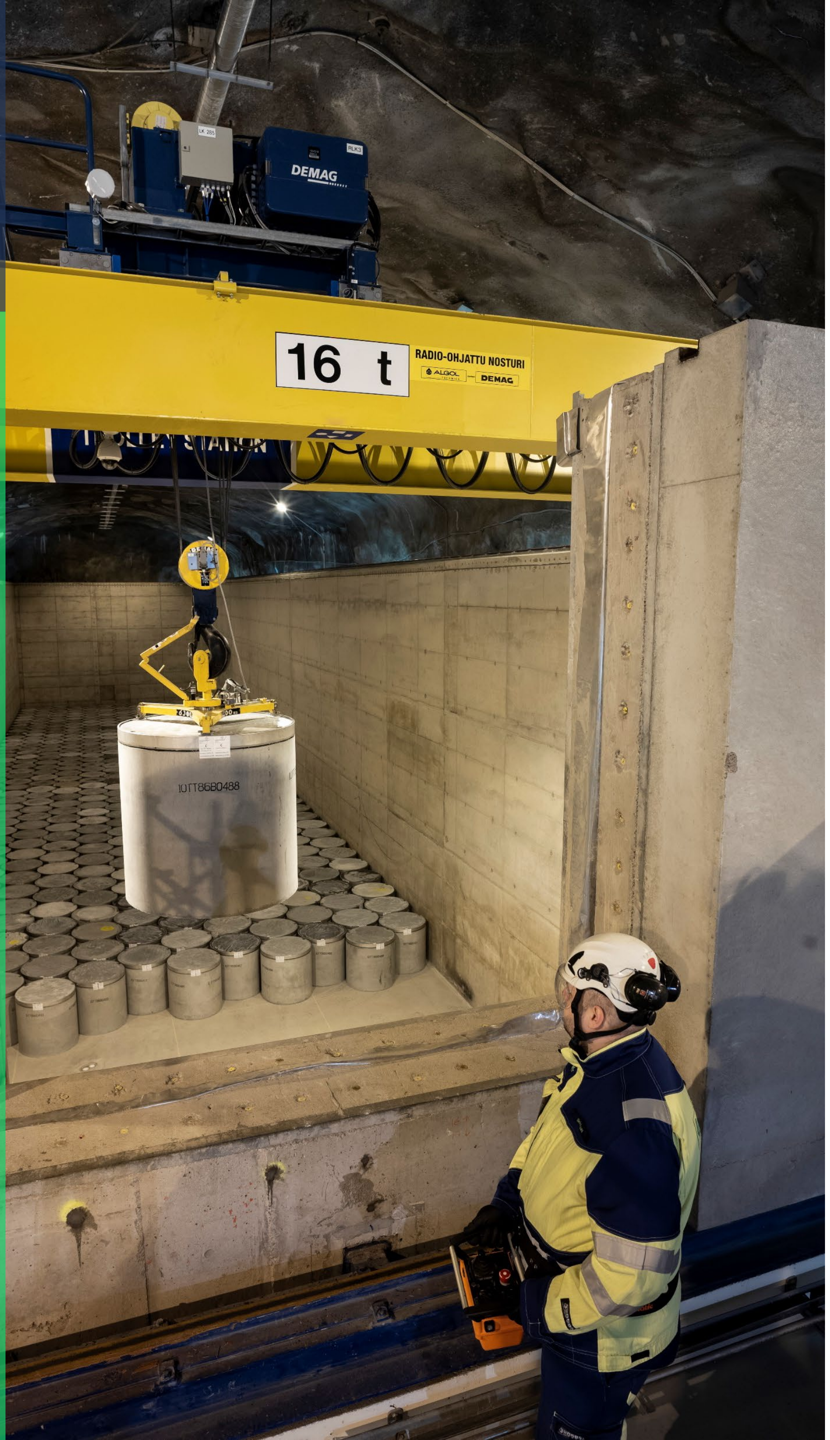
Loviisa LILW repository – concept and justification

- Radiological composition of waste considered in repository design
- The repository contains both engineered and natural barriers against nuclide migration to environment.
- Different premises for different wastes depending on material, radioactivity and contamination level and waste fraction/component size. Separate spaces for low and intermediate level waste.
- Applied and optimized safety case methodology

Spatial arrangements and engineering barriers optimisation based on graded approach

Minimized occupational and public exposure

Optimised and transparent safety justification to support smooth licensing



Loviisa LILW repository – Operation and aging management

- Similarly, as with operation of nuclear power plant, hierarchy and structured process to develop operating and maintenance instructions ensures efficient commission and operation phase
- Aging management, incl., e.g. maintenance instructions, of equipment and machinery follows the approach applied in the NPP, driven by the hierarchy of systems and equipment based on their plant availability, safety and operational criticality.
- Comprehensive condition monitoring programs developed and implemented for repository premises and waste packages (rock caverns, concrete structures, shotcrete, reinforcement bolts and LLW packages)

Clear and unambiguous operating instructions and practices enhance sustainable and cost-efficient waste management

Aging management of facility's SSCs optimized based on equipment criticality classification

Monitoring of aging management and long-term safety of repository ensured through systematic condition monitoring programs

fortum
Powering a thriving world

Fortum's other selected waste disposal related project references

- Expertise and experience from Loviisa LILW has been deployed also for other waste disposal facilities, incl.
  - TVO NPP, Finland - Safety Case - Low and Intermediate Level Waste Repository
  - TVO NPP, Finland - Landfill Safety Case
  - SKB, Sweden - SFR Safety Case informal review
  - KORAD, Korea - consultation on periodic safety review for LILW disposal
  - Posiva ONKALO DGR, Final disposal project – broad and long-term technical support to Posiva (e.g. Spent nuclear fuel database, operational and post-closure safety assessments, safety case study, operating instructions etc.)

Thank you!

Antti Ketolainen
Director, Decommissioning and Waste
Nuclear Services
Fortum Power and Heat Oy
antti.ketolainen@fortum.com
https://www.linkedin.com/in/anttiketolainen/

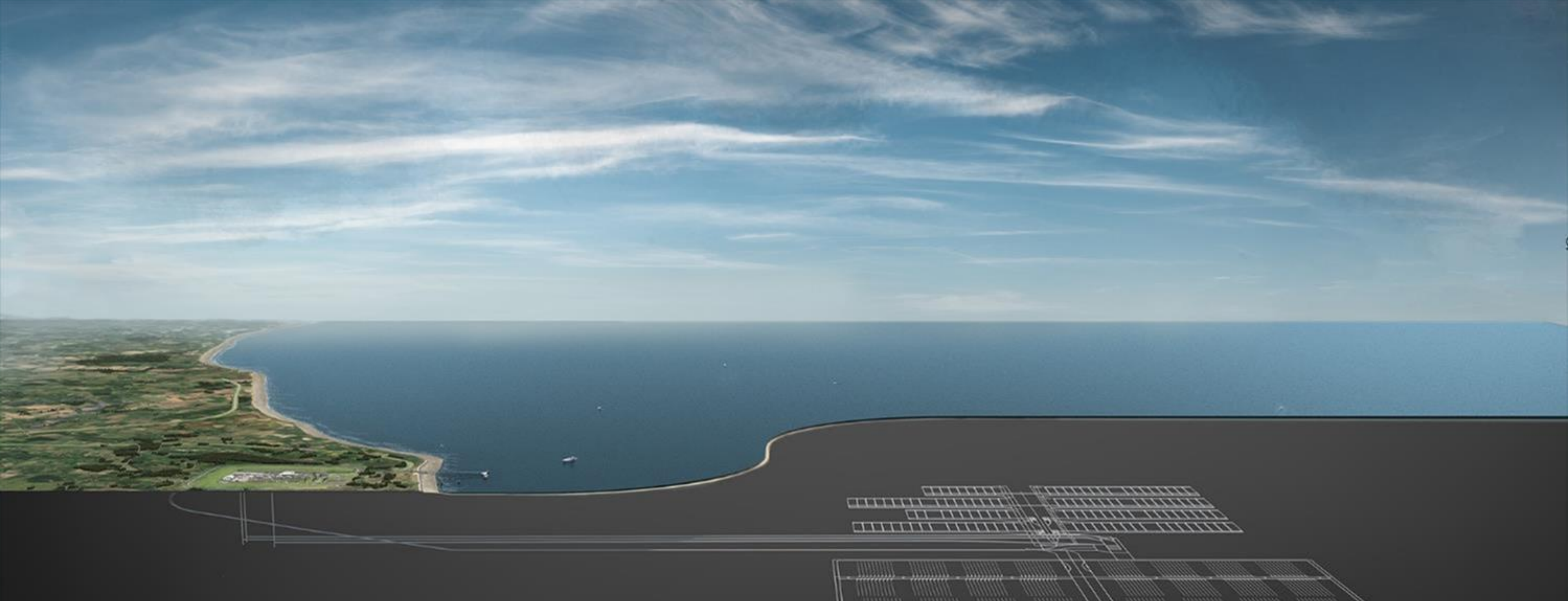
Nuclear Waste Services

ICGR-7 Session 4C

Rise of new and innovative technologies, future nuclear systems, and impact on the development of DGRs

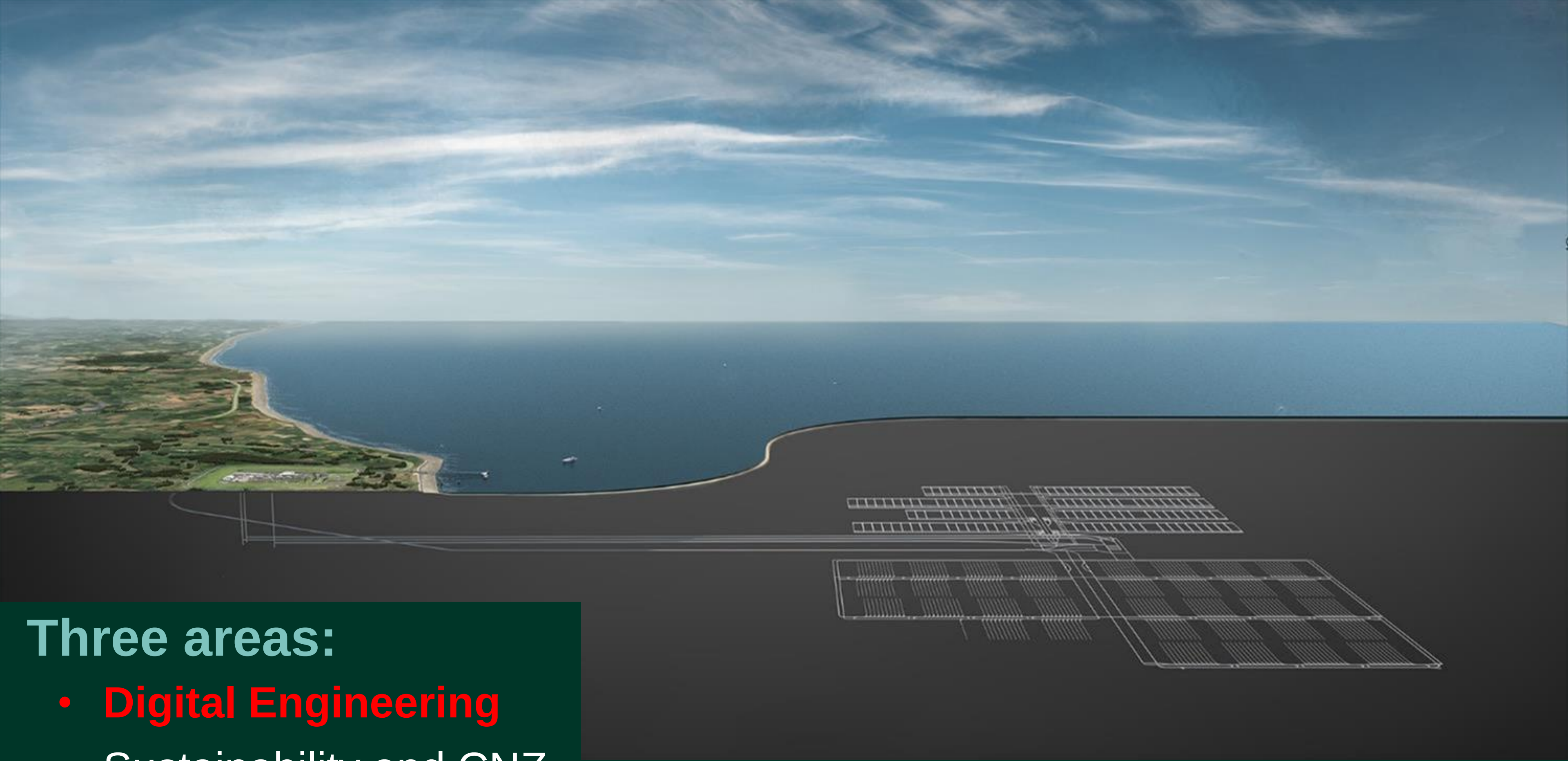
John Corderoy

Chief Technical Officer, NWS



Three areas:

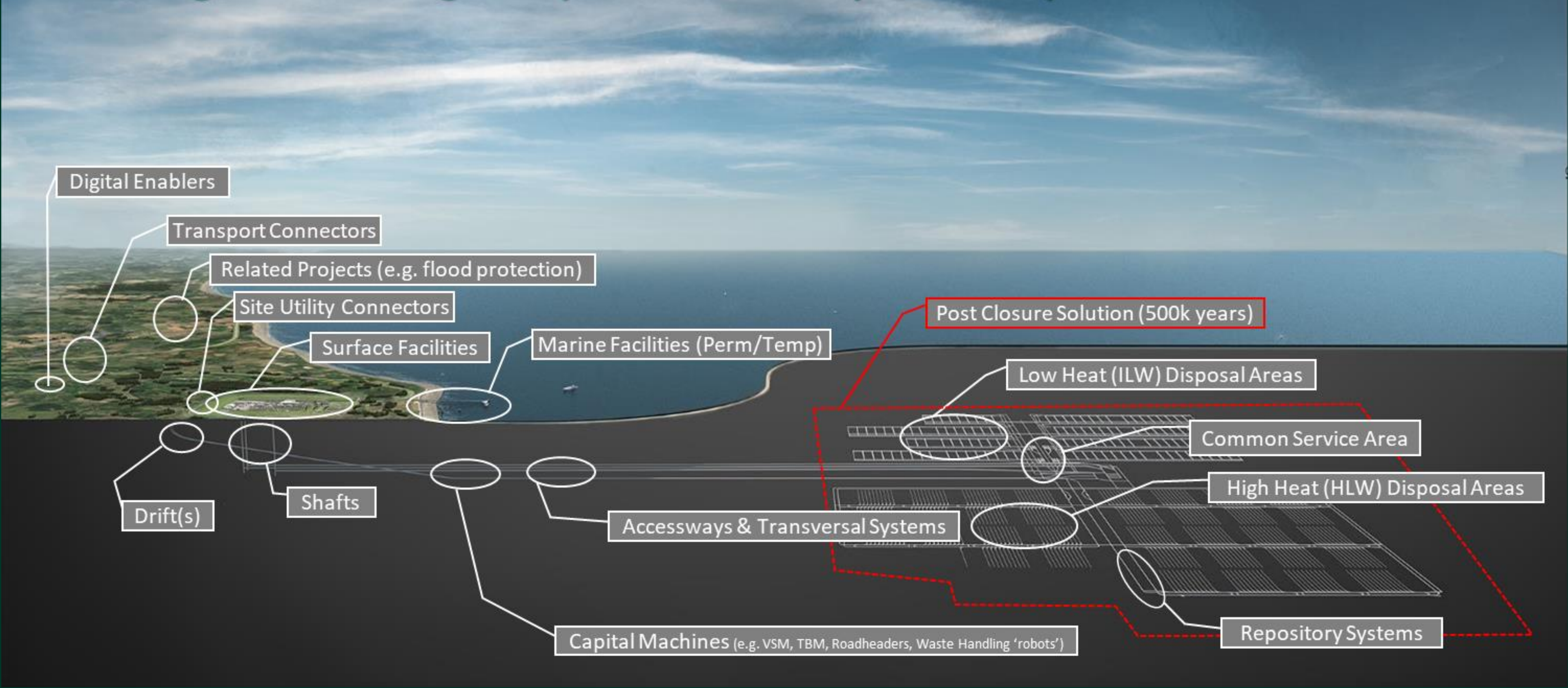
- Digital Engineering
- Sustainability and CNZ
- Artificial Intelligence



Three areas:

- **Digital Engineering**
- Sustainability and CNZ
- Artificial Intelligence

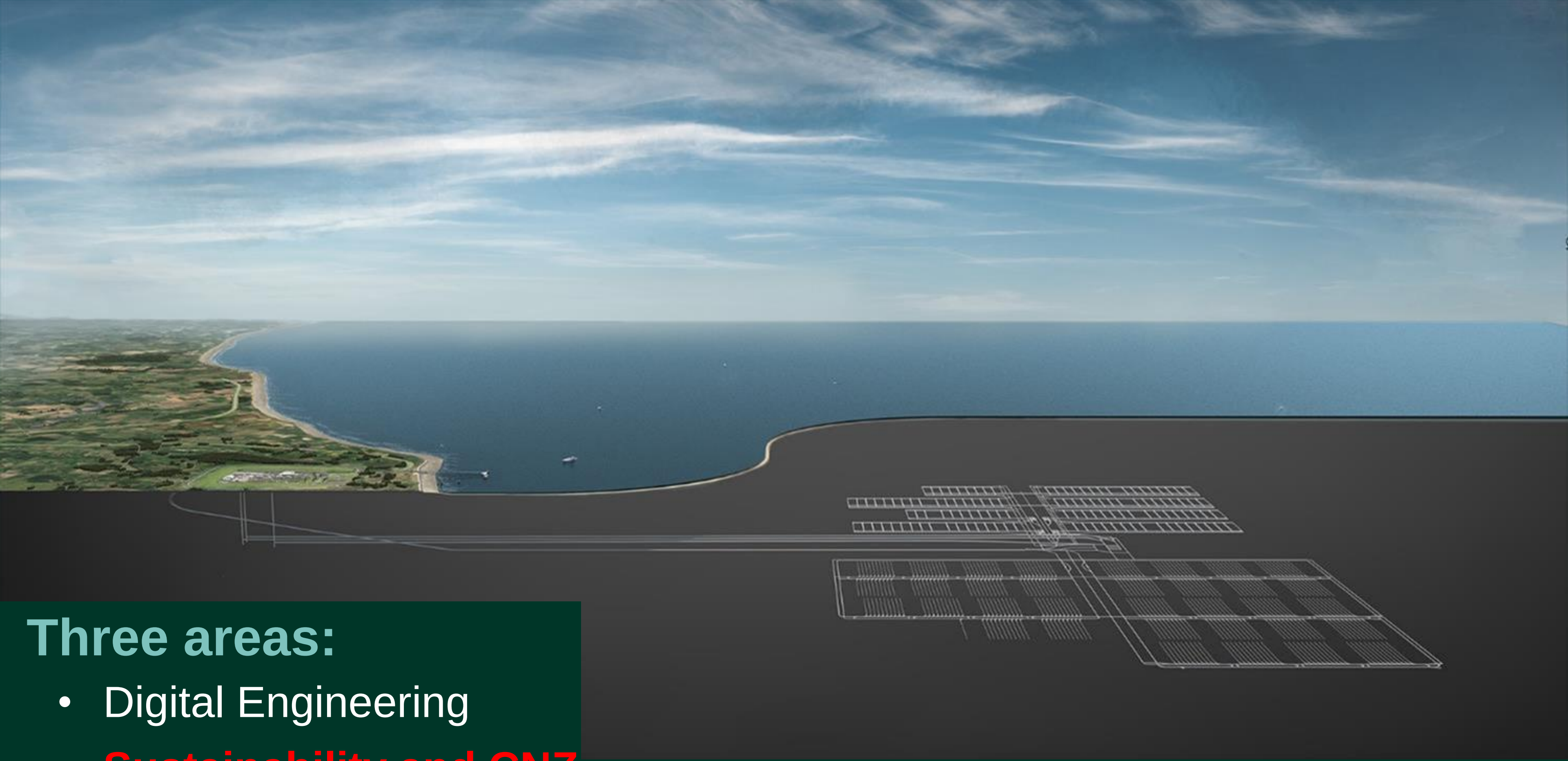
GDF High Level Design Scope – GDF is a "system of systems"



Nuclear Waste Services

Nuclear Waste
Services

Nuclear Waste Services



Three areas:

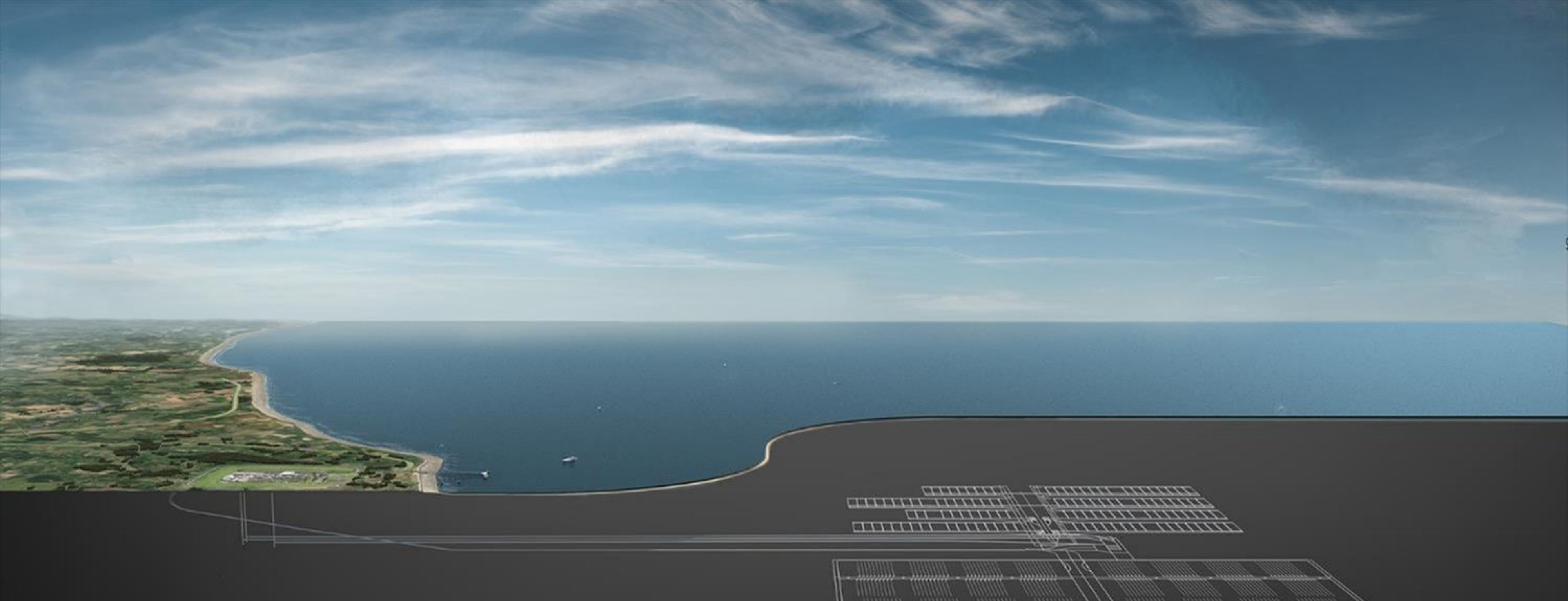
- Digital Engineering
- **Sustainability and CNZ**
- Artificial Intelligence


1 NO POVERTY
2 ZERO HUNGER
3 GOOD HEALTH AND WELL-BEING
4 QUALITY EDUCATION
5 GENDER EQUALITY
6 CLEAN WATER AND SANITATION
7 AFFORDABLE AND CLEAN ENERGY
8 DECENT WORK AND ECONOMIC GROWTH
9 INDUSTRY, INNOVATION AND INFRASTRUCTURE
10 REDUCED INEQUALITIES
11 SUSTAINABLE CITIES AND COMMUNITIES
12 RESPONSIBLE CONSUMPTION AND PRODUCTION
13 CLIMATE ACTION
14 LIFE BELOW WATER
15 LIFE ON LAND
16 PEACE, JUSTICE AND STRONG INSTITUTIONS
17 PARTNERSHIPS FOR THE GOALS


1 NO POVERTY
2 ZERO HUNGER
3 GOOD HEALTH AND WELL-BEING
4 QUALITY EDUCATION
5 GENDER EQUALITY
6 CLEAN WATER AND SANITATION
7 AFFORDABLE AND CLEAN ENERGY
8 DECENT WORK AND ECONOMIC GROWTH
9 INDUSTRY, INNOVATION AND INFRASTRUCTURE
10 REDUCED INEQUALITIES
11 SUSTAINABLE CITIES AND COMMUNITIES
12 RESPONSIBLE CONSUMPTION AND PRODUCTION
13 CLIMATE ACTION
14 LIFE BELOW WATER
15 LIFE ON LAND
16 PEACE, JUSTICE AND STRONG INSTITUTIONS
17 PARTNERSHIPS FOR THE GOALS

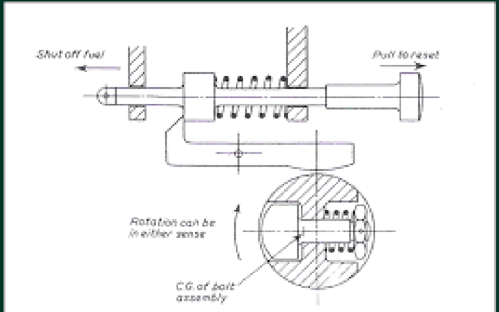

Heliogen
WWW.HELIOGEN.COM


ROAD
CLOSED

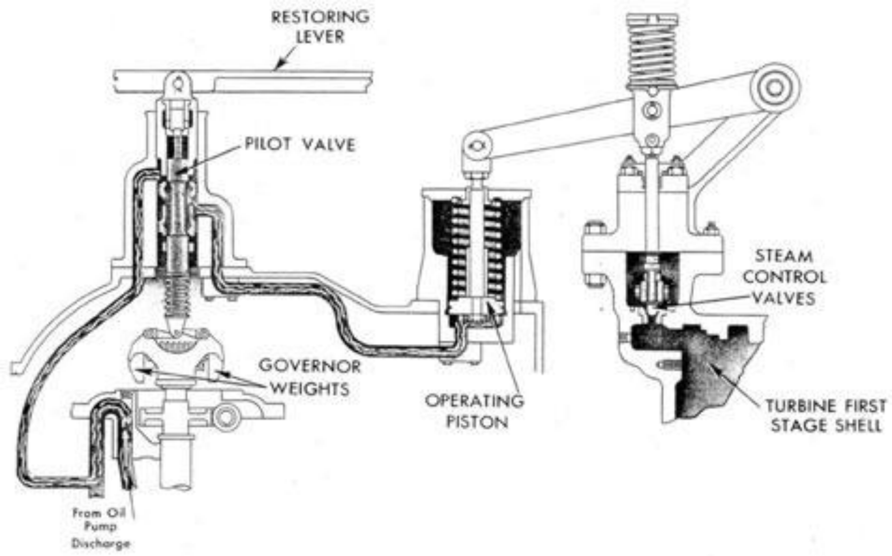


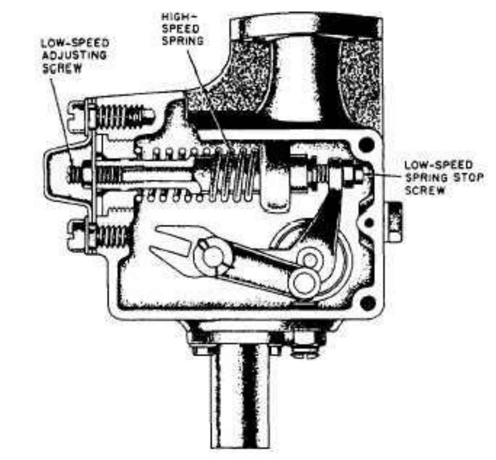
Three areas:

- Digital Engineering
- Sustainability and CNZ
- **Artificial Intelligence**

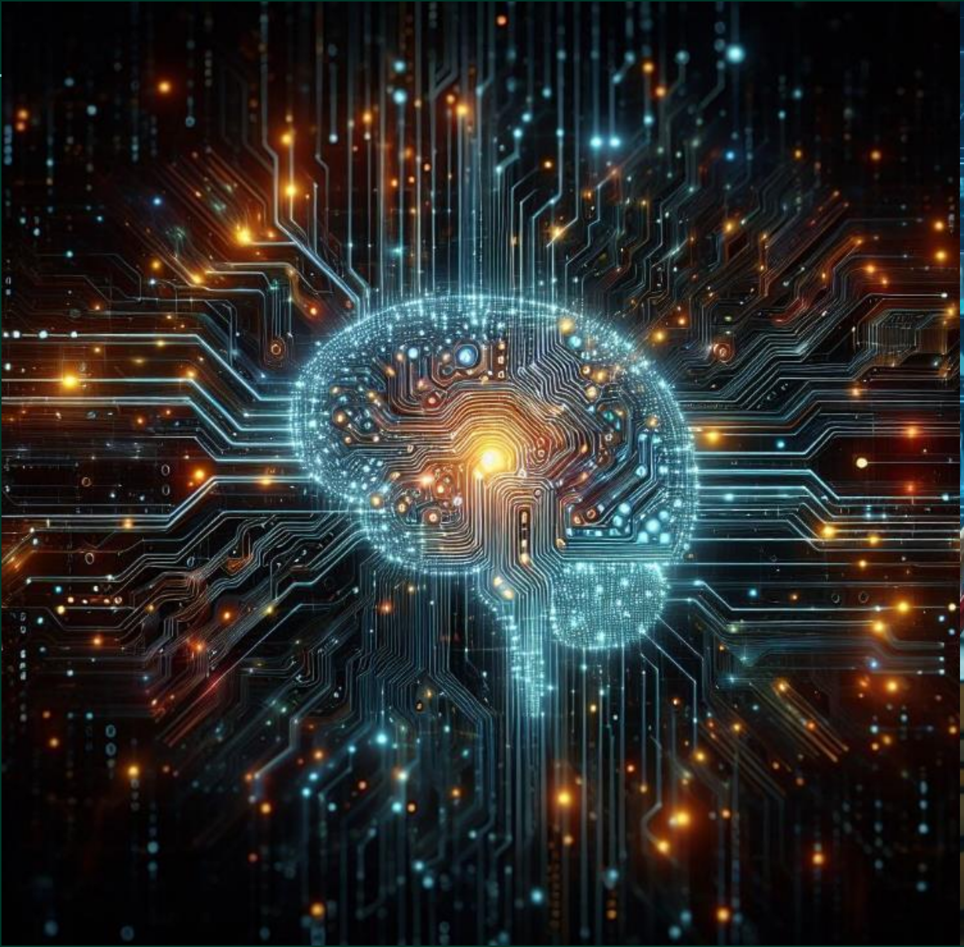


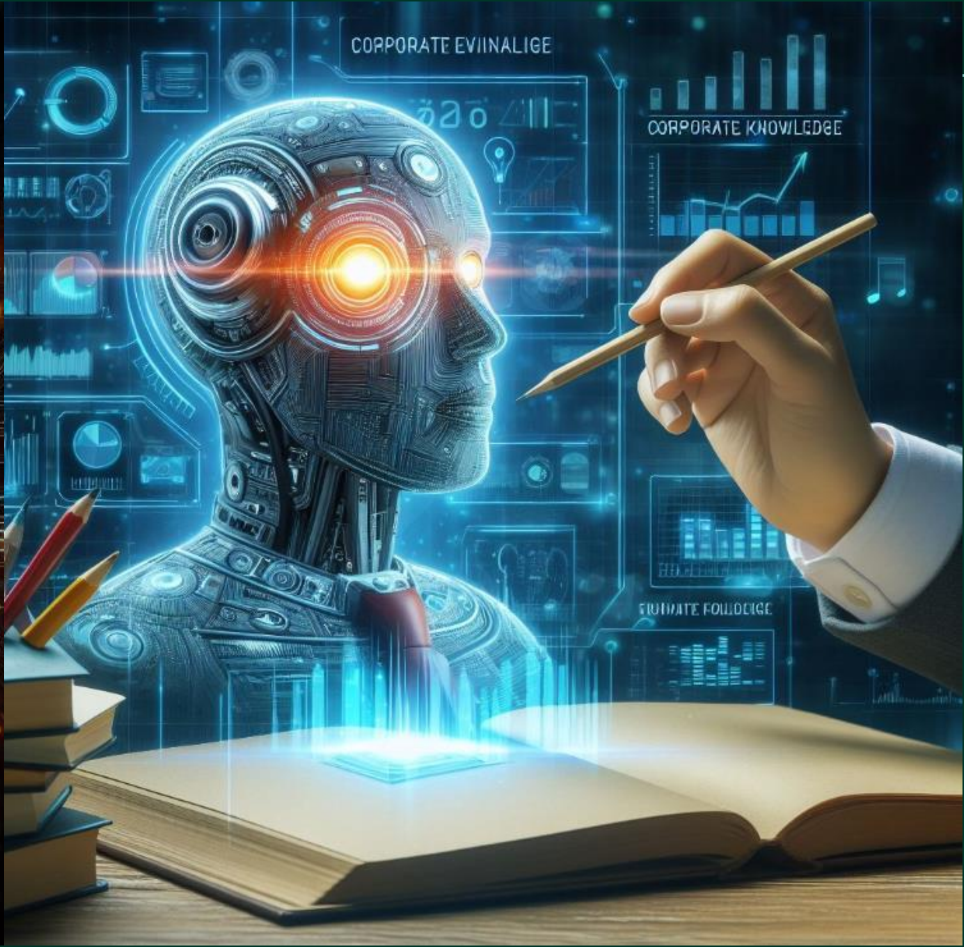
Overspeed trip.





Nuclear Waste Services




CORPORATE EVIINALIGE
CORPORATE KNOWLEDGE

Nuclear Waste Services