NAWG 2013 - Nagoya

Key NA input to build a safety case for direct disposal of SF in Japan

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Introduction

- Natural analogues (NAs) have been previously used to support the safety case for direct disposal of spent fuel (SF)
- Focus of such work was set by key barriers of specific national disposal concepts, e.g.:
  - Swedish / Finnish KBS-3 concept ➔ NAs of Cu corrosion and the longevity of the surrounding bentonite
  - Yucca Mountain site ➔ NAs of corrosion and uraninite leaching under unsaturated conditions
  - Opalinus Clay in Switzerland ➔ NAs to demonstrate the diffusive barrier provided by the host rock
- **What are the NA needs for Japan, where consideration of direct disposal of SF is only now starting?**

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Boundary conditions in Japan

- Inventory of SF: UO_2 from LWRs, MOX, damaged fuel / corium from Fukushima Dai-ichi (FDI)
- Possibly co-disposal with HLW / TRU: repository depth >c300m
- Possible volunteering approach to siting (or siting near FDI???)
- In any site, tectonic activity may be significant, potentially high geothermal gradient, risk of hydrothermal water flow from distant (10s of km) volcanic activity
- **Need to quickly develop state-of-the-art safety case → take over as much as possible from international knowledge base**

International knowledge base

Safety cases for direct disposal of SF in other national programmes have limited applicability to Japan due to:

- Completely different host rocks (e.g. Salt)
- Different tectonic setting (e.g. Shield rocks of Scandinavia & Canada)
- Different ambient conditions at disposal depth (e.g. Temperature, redox, ...)
- Different types of fuel (e.g. CANDU)
- Different regulations (e.g. Cut-off times)

→ nevertheless, common problem areas are identified that need to be handled in the Japanese concept / safety case

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Key issues

- Instant release fraction (IRF), especially I-129: tends to dominate doses for most scenarios
- Grow-in of high toxicity daughters that can dominate doses if OP failure at long times / rapid transport pathways to biosphere
- Radiolysis high and can degrade matrix if no redox buffer
- Common requirement for long EBS performance, therefore very high fabrication quality needs to be demonstrated
- Higher thermal output, therefore need for careful heat management
- Assure negligible criticality risk
- No international experience for disposal of corium
- For programme flexibility, extended period of ease of recovery of potential resource desirable.

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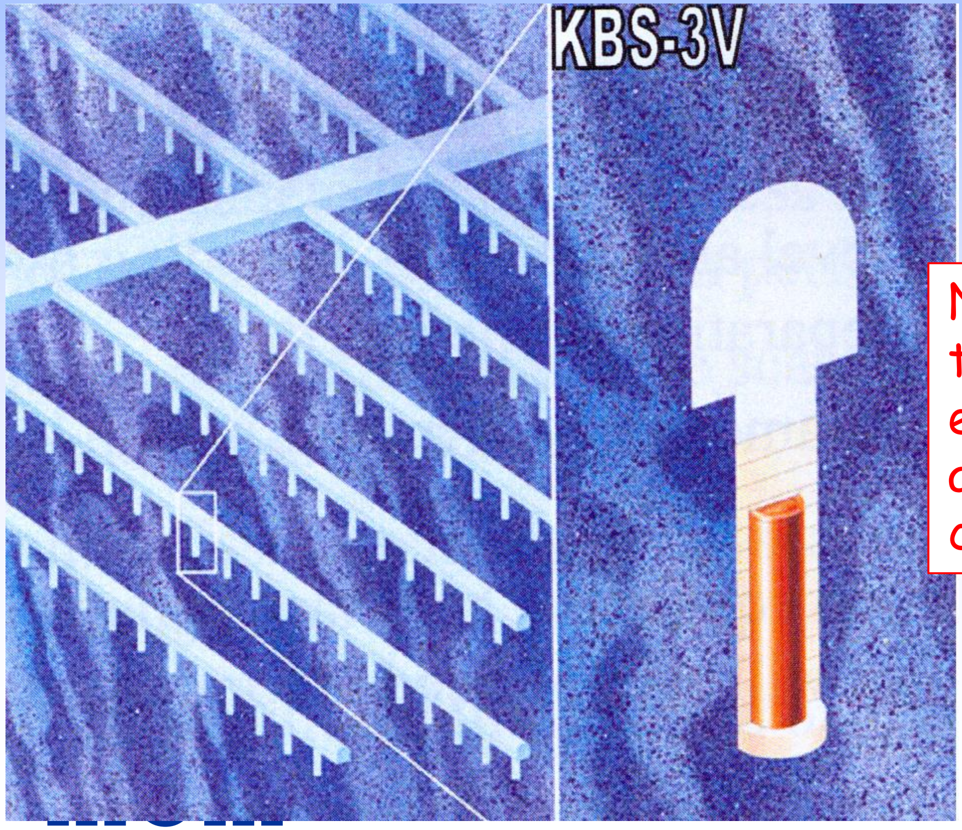
Concept & layout

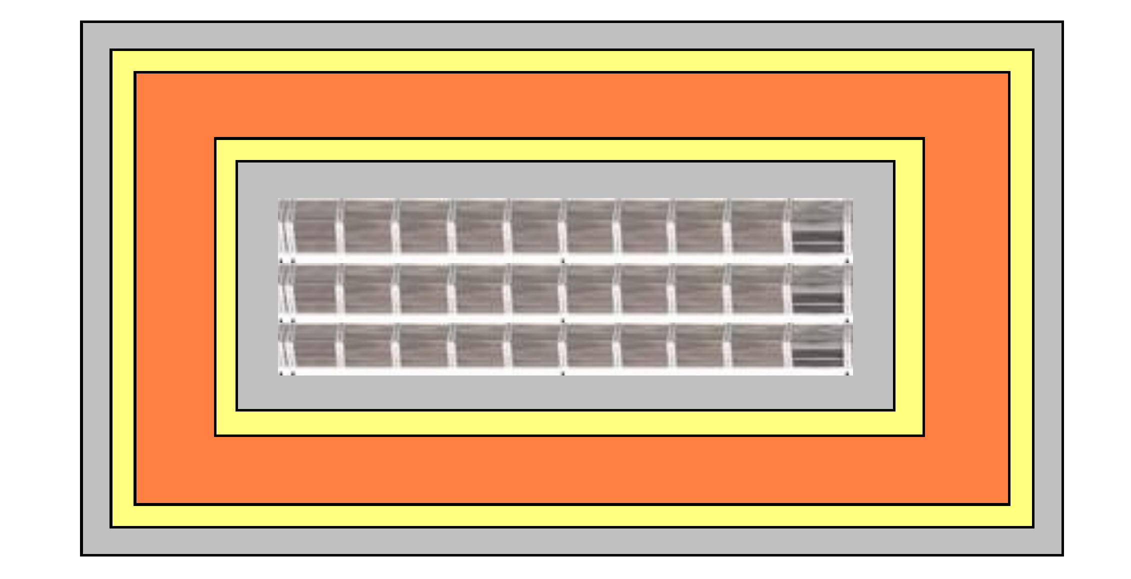
- Larger package size and requirement for high quality EBS makes borehole disposal options tricky except in high performance geosphere (multi-package vertical in salt, horizontal in clay)



NB this option demonstrated to be impractical in Sweden: even less likely to be appropriate for Japanese conditions

Concept & layout

- In-tunnel disposal feasible if EBS quality can be assured (probably requires some form of PEM, as Japanese conditions likely to be very humid)



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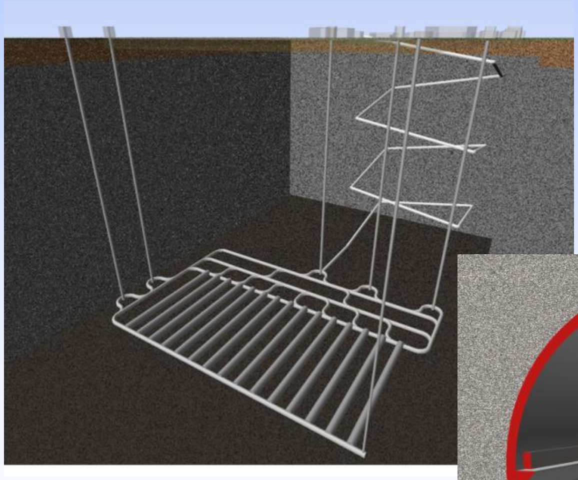
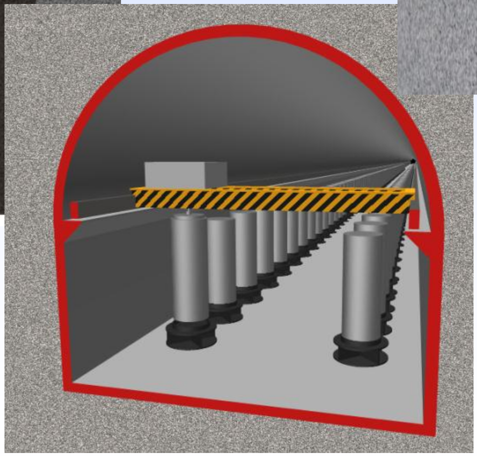
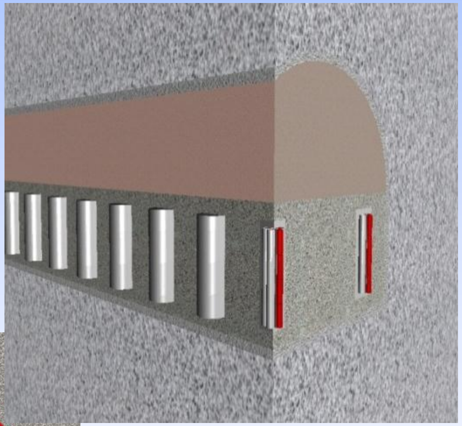
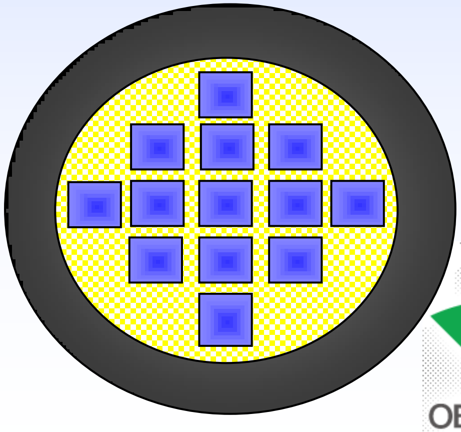
Concept & layout

- Cavern disposal designs could be advantageous if rock suitable for their construction. These may be especially useful for managing FDI damaged fuel and allowing flexibility to respond to uncertainties in future nuclear programme

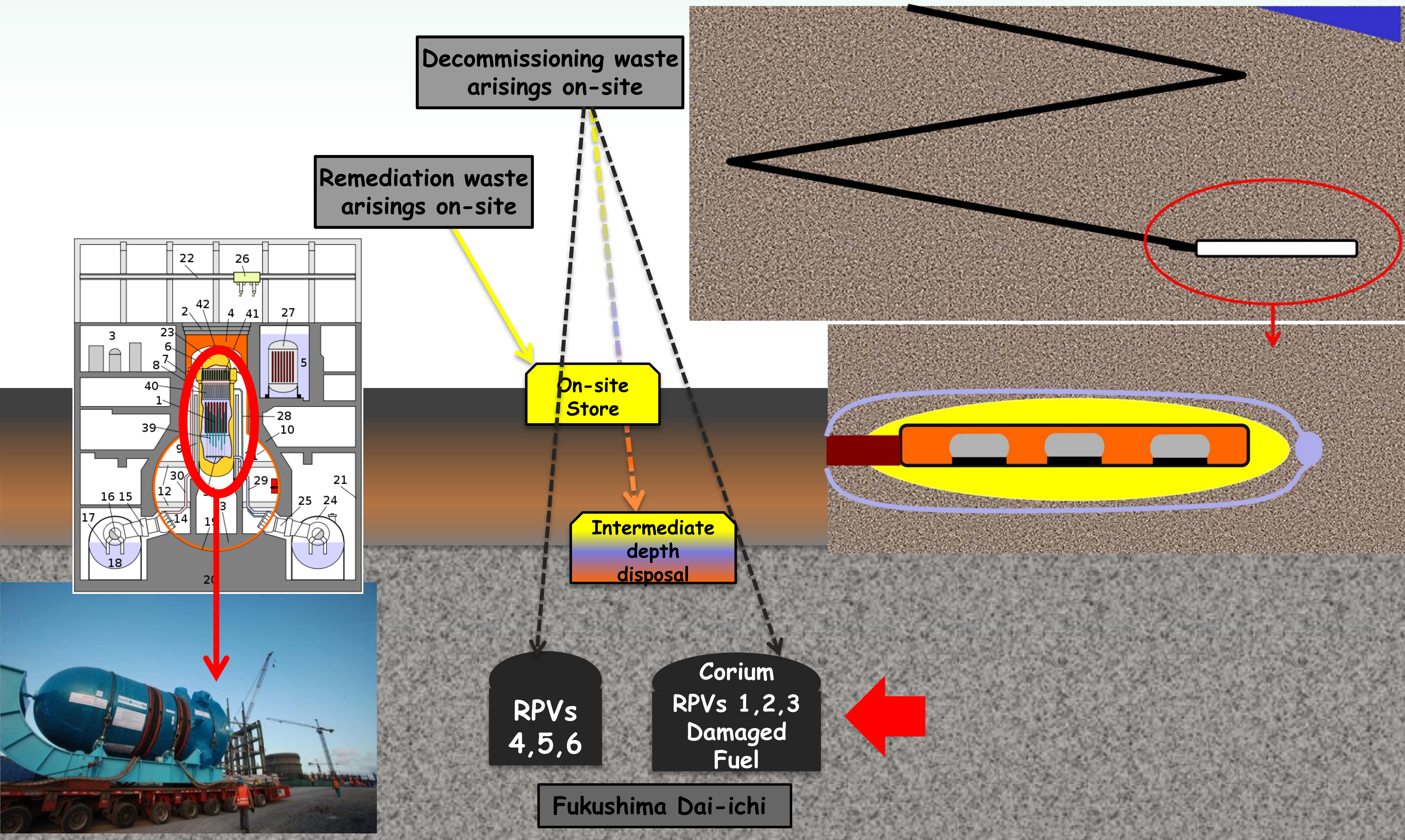








Special considerations for FDI



Overpack

Need is for assured long distribution of failure times:

- Copper options require high quality fabrication and also suitable geochemical / tectonic settings

- Steel can be easily QAd (very thick OPs could be sealed using screwed or bolted lids): **can required performance be demonstrated?**

- Ti is also a potential candidate, maybe with steel insert for redox control. High quality fabrication again needed.

- Modern materials (ceramics, cermets, etc.) could also offer potential, especially when combined with a steel insert (corrosion may be negligible, so good mechanical failure model needed)

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Buffer / backfill

- Bentonite has many advantages, but may be difficult to QA and assure resistant to thermal transient (e.g. supporting fundamental studies, use of PEM or cavern). Performance for SF could be improved by:
  - Including an "I-getter"
  - Including material to reduce criticality risk (boron glass, depleted uranium, NB could also be included inside SF overpack)
- Alternative materials might be considered in some cases, e.g.
  - Zeolites / vermiculite (maybe easier to QA, higher T stability, Iodine retention)
  - Specialist cements / concrete (less high pH concern than glass, although TD database limited at high pH)

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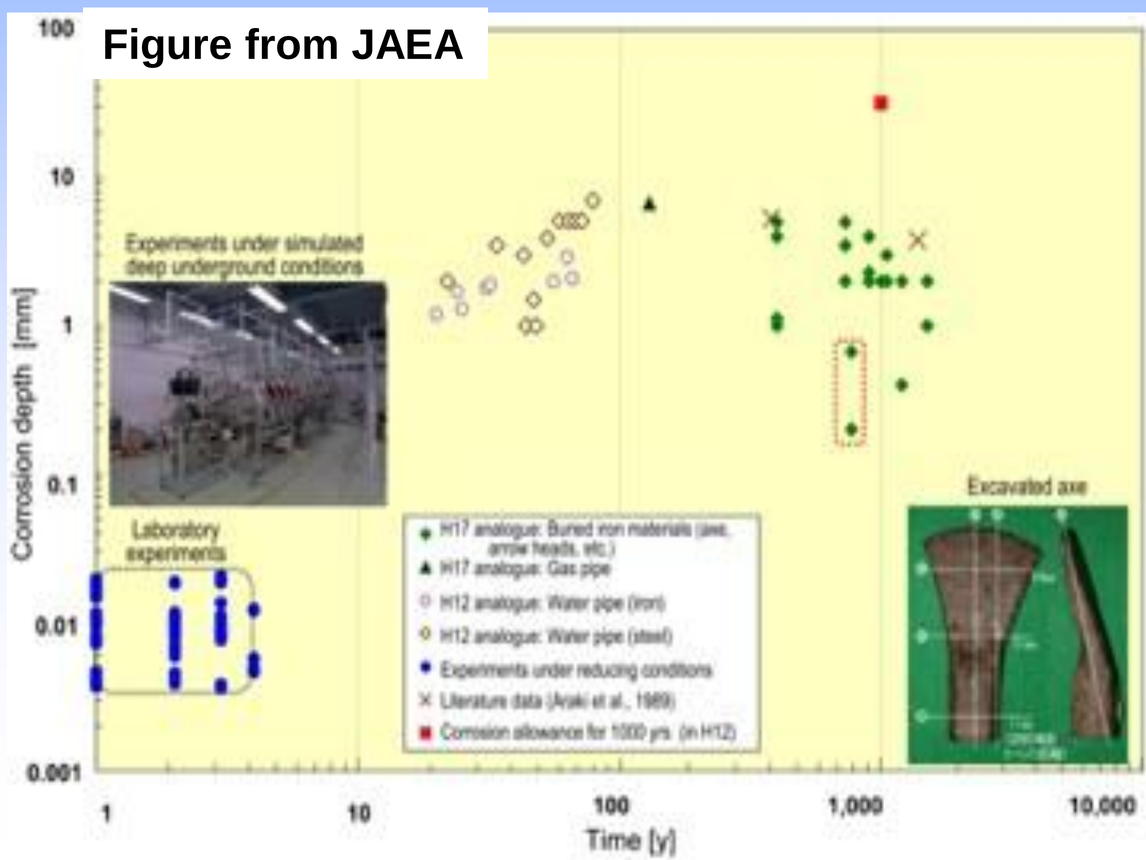
NA priorities

Analogue evidence is needed to support EBS options that can provide the key roles of:

- Assuring long enough complete containment for major reduction in total radiotoxicity of the inventory (10 – 100 ka)
- Spreading release of "instant release fraction" RN, especially I-129, but also Cl-36, Cs-135 (ideally over c100 ka)
- Assuring redox buffering of radiolytic oxidant to justify slow fuel matrix leaching and low solubilities of key RN
- Corium matrix stability
- Assessing rigour of spent fuel safety case for cementitious system

Overpack longevity

- Long-term lab tests under reducing conditions and NAs indicate extremely low corrosion as oxide layer builds up
- **Can NAs support models of local variations in corrosion rate to support assessment of distribution of failure times?**



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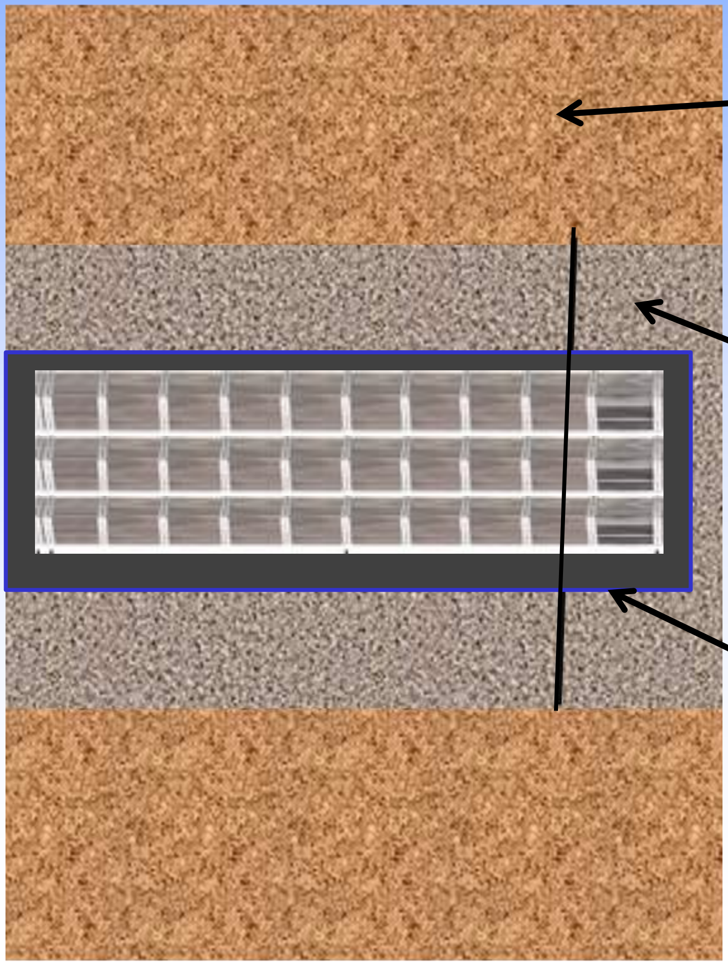
OP failure

Even very conservative assumptions lead to lifetimes > 10 ka: more realistically 100 ka can be expected with a distribution of failures over a similar period



Compacted bentonite maintains diffusive environment

Corrosion of about 90 % of OP before mechanical failure (**have corrosion products a barrier role?**)

Remnant steel still available: can it be assured to buffer redox?

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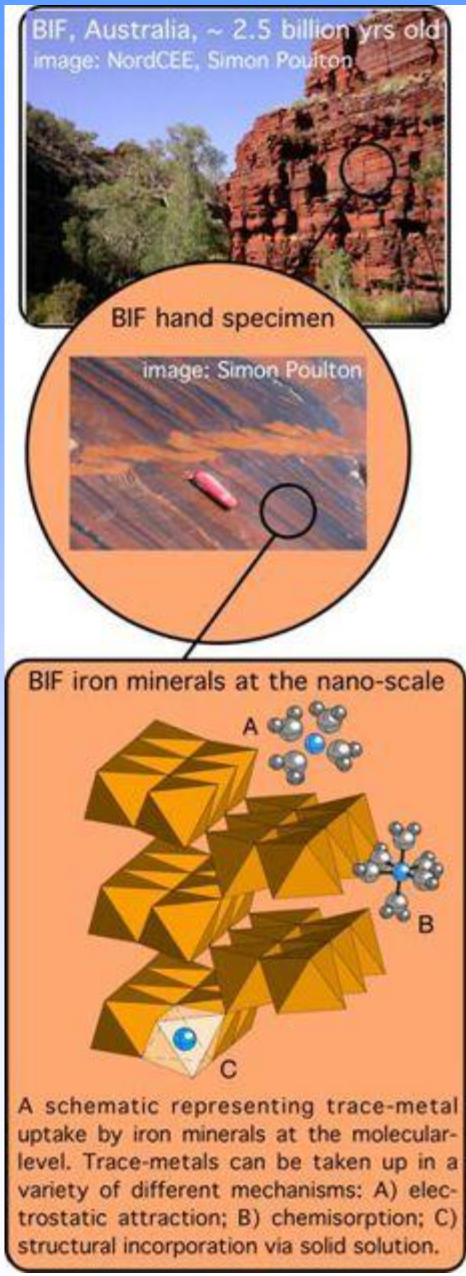

Immobilisation in iron oxides

- Many examples of ore deposits stable over millions of years
- Extreme example of Banded Iron Formations that have preserved trace element signatures in their structures for 2.5 Ga – more than half the age of the earth
- **Could these be used as NAs to justify more realistic models of the barrier role of failed overpack?**

http://www.see.leeds.ac.uk/admissions-and-study/research-degrees/essi/peacock-shaw-krom/

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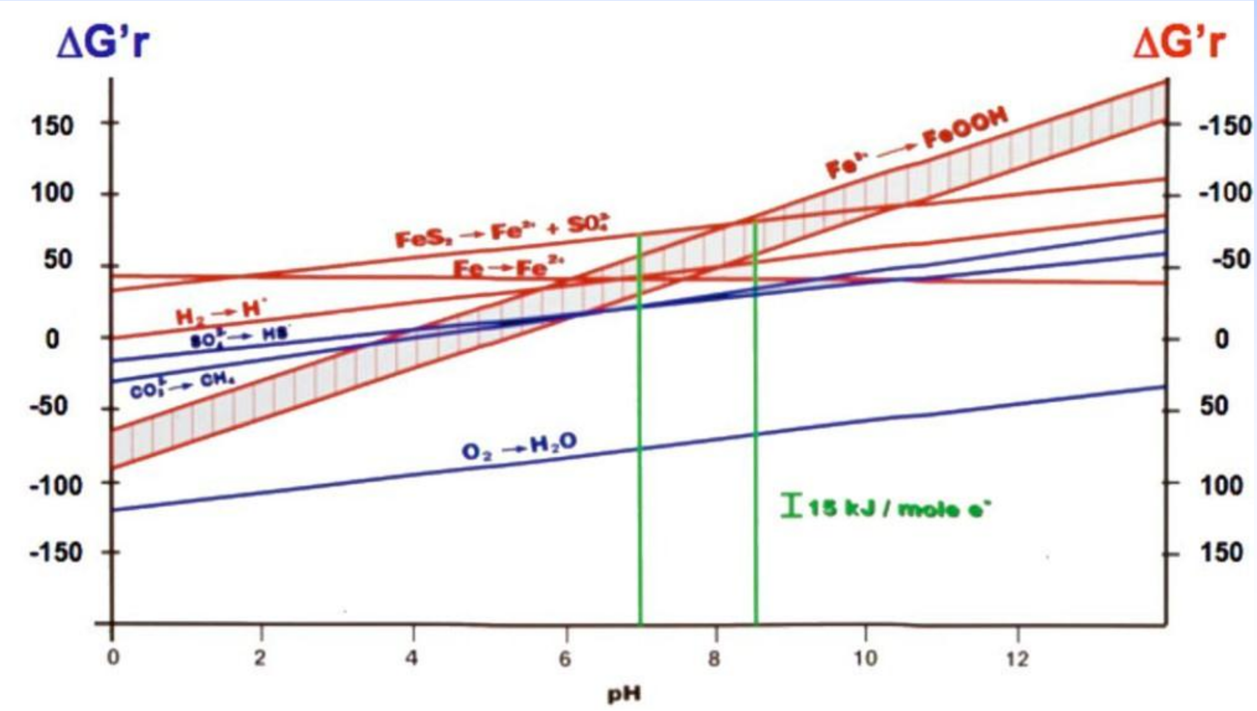
Redox buffering

- The arguments supporting very long life of a steel overpack may, however, weaken arguments that remaining steel / corrosion products will buffer radiolytic oxidants
- **For low temperature systems, microbial catalysis may play a key role: are there relevant long-term analogues?**



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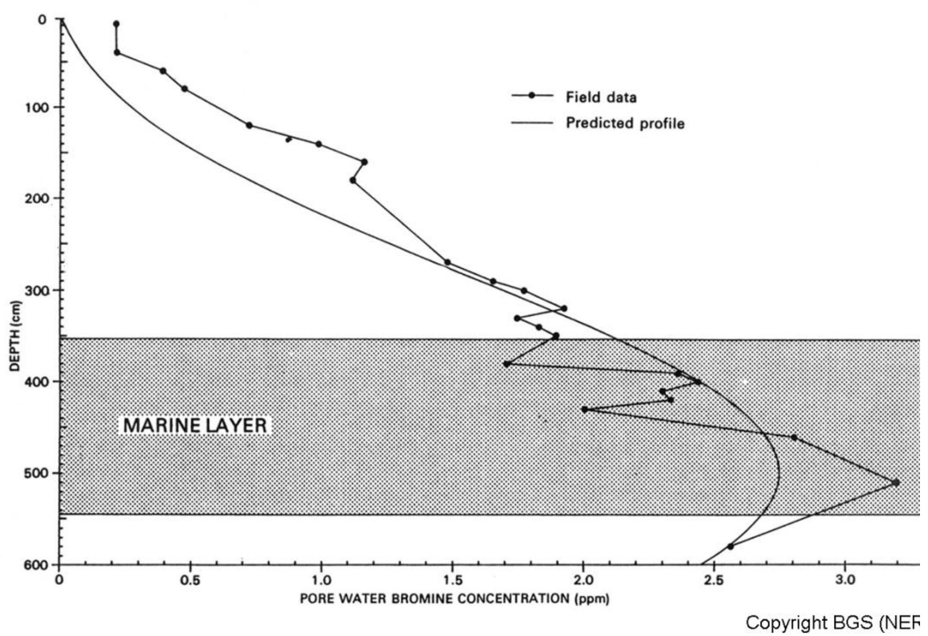
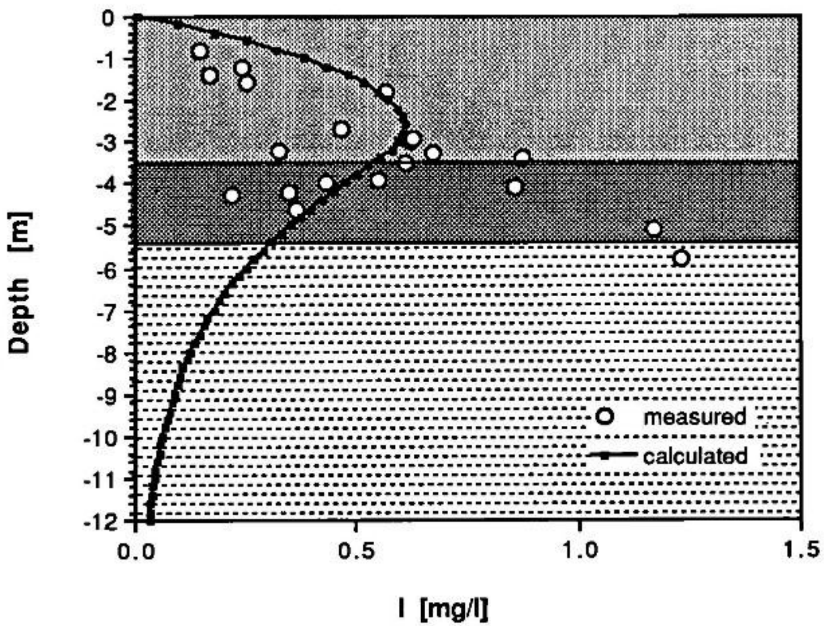
I-getters

- Retention of I (and other IRF RN) will reduce doses by temporal dilution – already shown in principle in the L. Lomond NA
- **Are there other NA locations that would strengthen such arguments (e.g. marine sediment overlain by zeolites; marine sediment overlain by bentonite under high pH conditions?)**





Corium stability

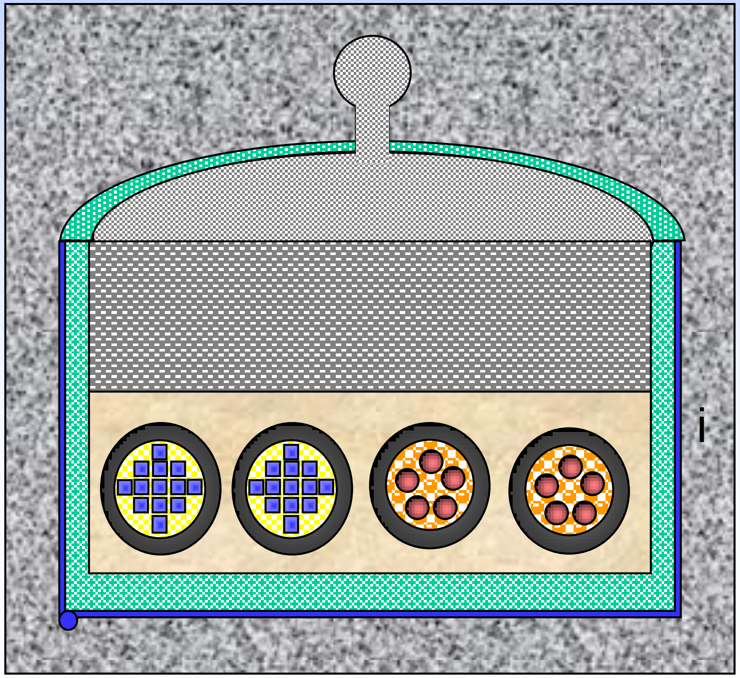
- Corium is a complex, heterogeneous material – within the reactor pressure vessel or especially if quenched by reaction with concrete after "melt through"
- **Are any equivalent materials found in nature – e.g. after igneous intrusion into an ore body? If so, where would the best examples be found?**

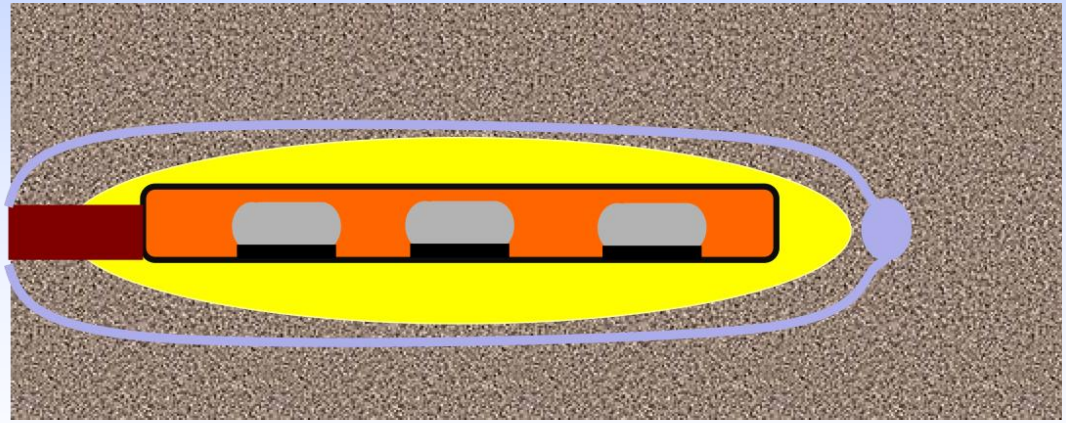
Fig. 8. Pool surface during the interaction

Fig. 10. One of the volcanoes found on the corium surface.

Hyperalkaline conditions

- Disposal concepts may include large quantities of concrete, especially in the case of weaker / more permeable rock
- **High pH conditions generally slow steel corrosion, but may increase uncertainties on SF leaching. Are there relevant analogues (e.g. U ore body contacted by hyperalkaline fluids)?**





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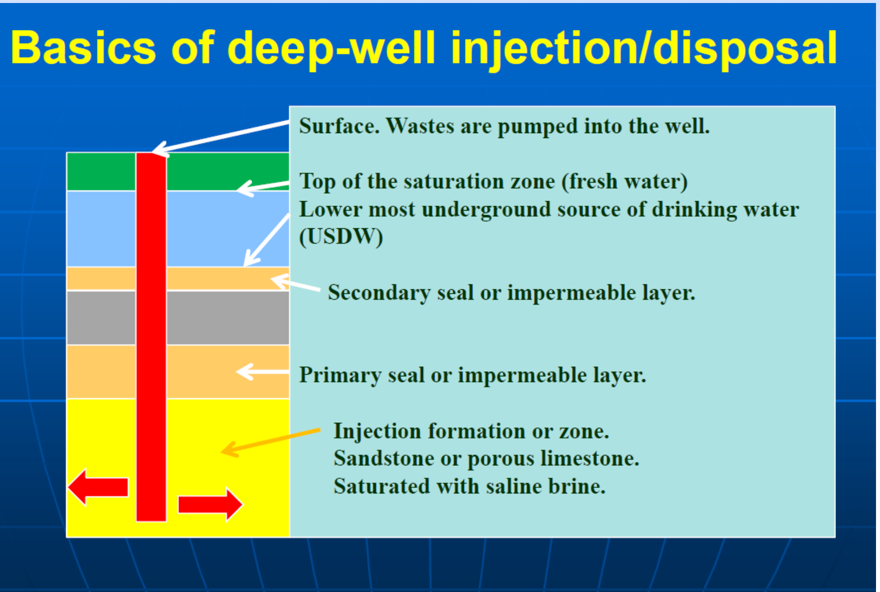
Colloids

- Poorly-reversible sorption of RN onto / uptake into colloids may increase their mobility. This may be especially relevant for concepts without assured colloid filtration (e.g. SF / corium with a cement-dominated EBS)

- **Are colloids stable at high pH? What are suitable NAs for testing models of colloid transport of RN (e.g. liquid radioactive waste injection sites)? Can NAs indicate if high pH buffers (zeolites) act as colloid filters?**



	HLW Process waste	Intermediate waste	Low-level waste
pH	1 to 3	~ 13	~8
TBP	50 mg/L	30 mg/L	None
^{90}Sr	2.49 Ci/L	0.51 mCi/L	2.97 µCi/L
^{137}Cs	0.30 Ci/L	0.41 Ci/L	4.05 µCi/L
^{239}Pu	100 to 500 µg/L	10 to 30 µg/L	< 1 µg/L

Conclusions

- Direct disposal of spent fuel in Japan may present some unique challenges for development of a robust safety case
- NAs can help development of appropriate disposal concepts, test the models and databases used to quantify their performance and provide supporting arguments to strengthen the associated safety case
- **Possibly as important as technical support, NAs can play an important role in building public acceptance. Here analogue systems should focus on issues of concern to the general public – e.g. demonstration that even movement of an active fault would not cause major loss of performance (possible NA – ore body intercepted by fault, especially if output can be provided in a user-friendly format (video / animations))**

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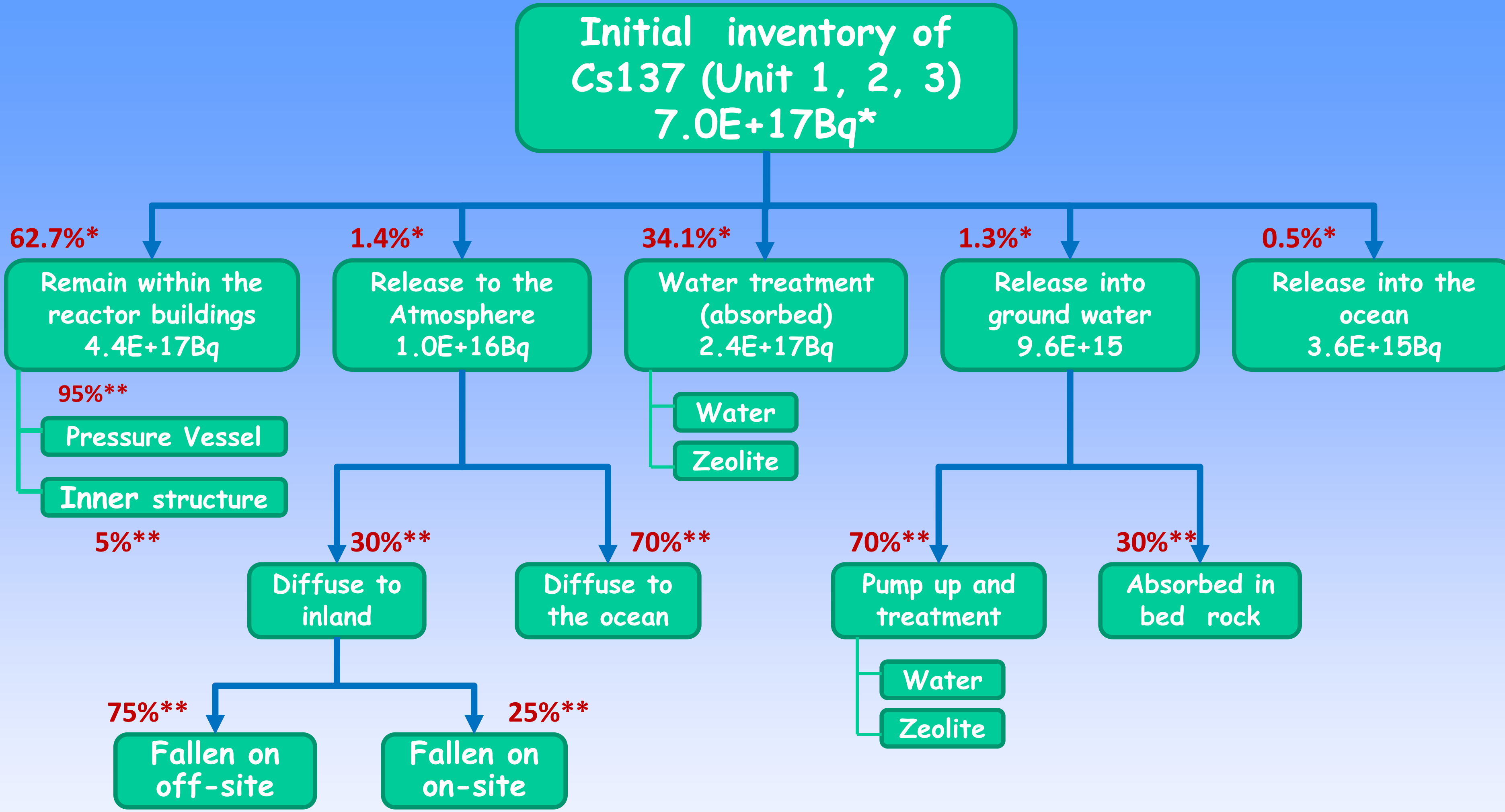


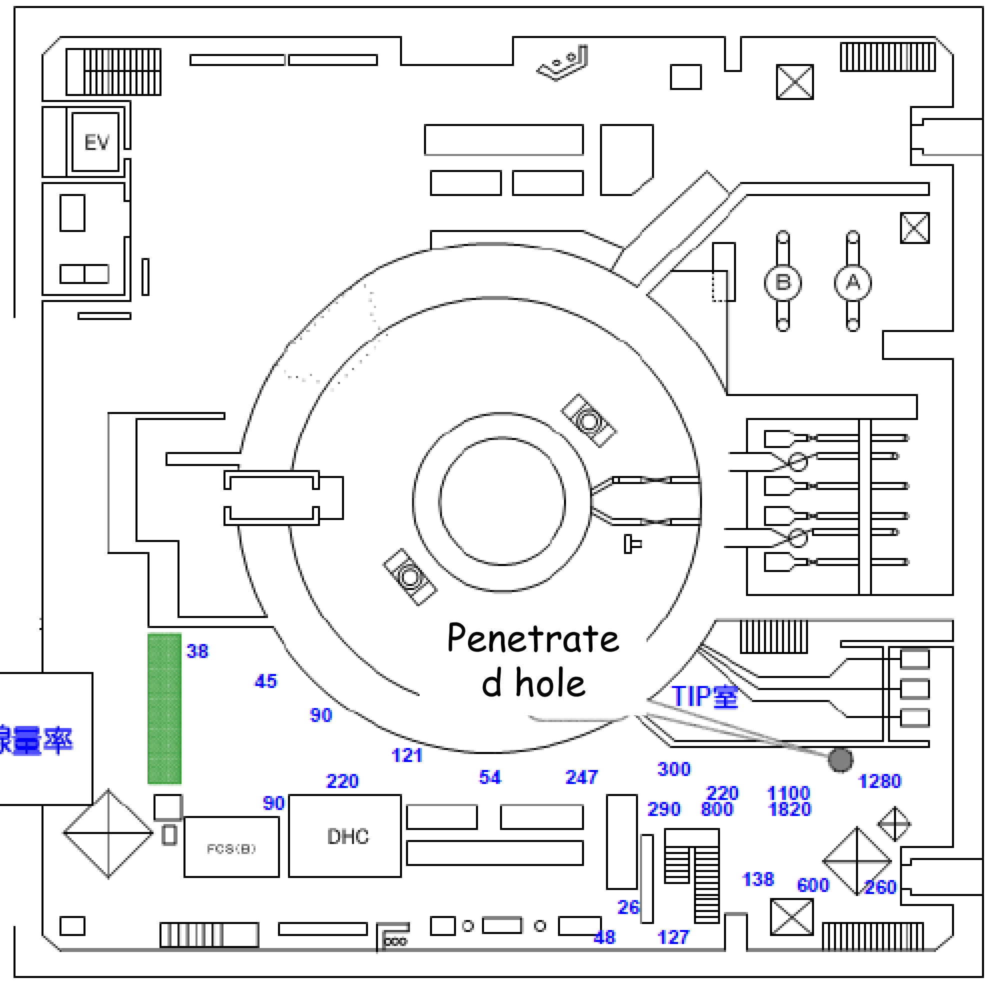
* JAEA Estimation
** Just assumption

Result of survey on Unit-1 NPP ground floor level

（参考）過去の線量率データ

１号機　原子炉建屋　１階

Past survey on Oct. 2011
Ground floor of Unit-1



【凡例】
H23.10月までに測定した雰囲気線量率（mSv/h）

仮設遮へい壁

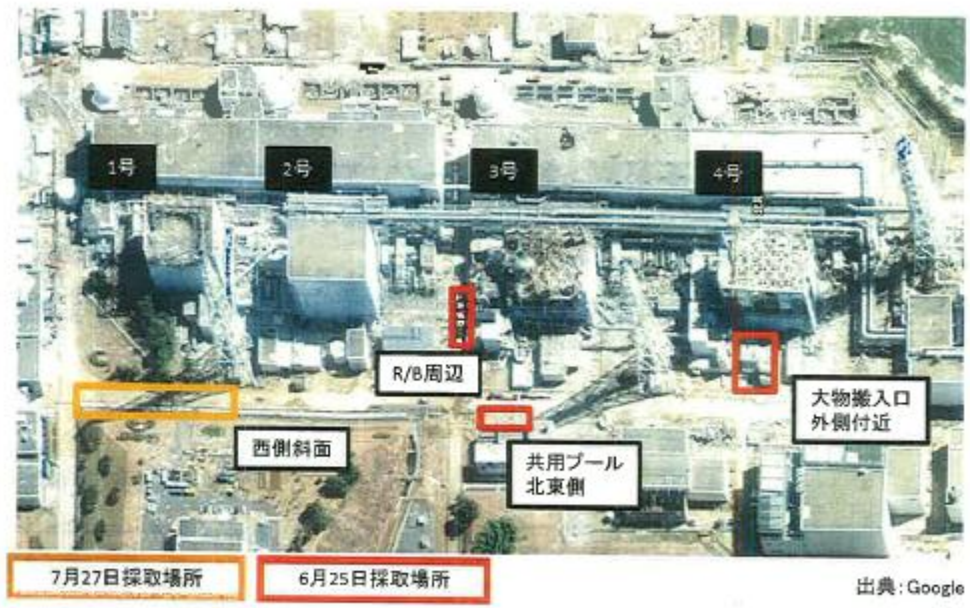
Temporary Shielding wall

Sampling of various debris (concrete, iron, metal, gravel and tree)

1. 性状調査 ガレキ等の核種分析に向けた試料採取

Sampling points
ガレキ・伐採木の試料採取場所

Sampling points around reactor building
原子炉建屋周辺のガレキ試料採取場所



◆ 採取した試料は、JAEA原子力科学研究所に輸送して核種分析を実施予定

分析予定核種

- γ核種:Co-60、Nb-94、Cs-137、Eu-152、Eu-154
- β核種:H-3、C-14、Cl-36、Ca-41、Ni-59、Ni-63、Se-79、Sr-90、 Tc-99 I-129、 Pu-241
- α核種:U-233、234、235、236、238、Np-237、Pu-238、239、240、242、Am-241、242m、243Cm-244、245、246

Fukushima Dai-ichi waste partitioning

Rubble removed from top of the R/B

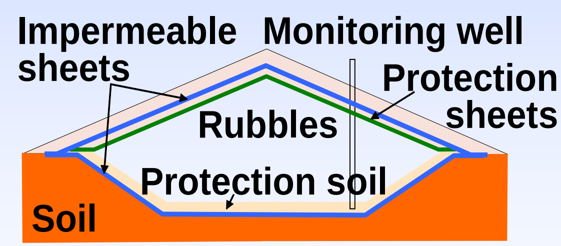
Top of the R/B (Unit 3)

Top of the R/B (Unit 4)

< 0.1mSv/h*1 → **Open air storage**

0.1mSv/h ～ 10mSv/h → **Temporary storage facility**

10mSv/h ～ 1Sv/h → **Temporary storage facility**



1Sv/h < →

Temporary storage area with shielding ability
Container
Container storage in building

*1 Dose rate at the surface
*2 R/B : Reactor Building

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