Requirements on decay heat and source term calculations for interim dry storage

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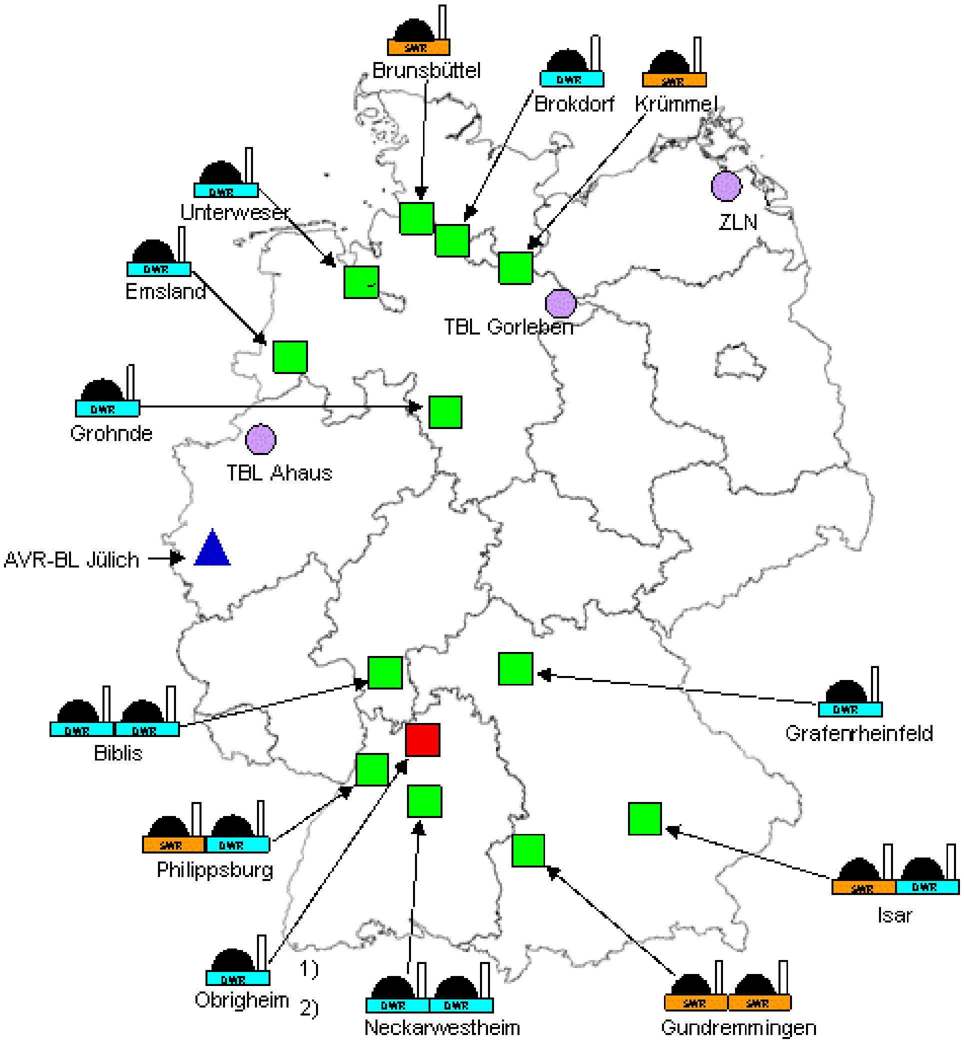
German intermediate HLW storage facilities



On-site Storage Facility (12)

Central Storage Facility (3)

NPP out of Operation (10)

Constraints of spent fuel casks (transport and storage) to be respected for each loading

- Max. total heat load of the cask
- Min. total heat load of the cask
- Max. decay heat of FA per position
- Max. cladding temperature of FA
- Max. tangential stress of the fuel rod
- Max. creep strain of the fuel rod
- Max. temperature of the basket
- Max. burn-up of FA per position
- Min. burn-up
- Max. enrichment of FA
- Max. number of MOX-FA
- Pu-vector of MOX-FA
- Max. average dose rate of the cask ($n + \gamma$)
- Max. dose rate at specific points of the cask ($n + \gamma$)
- Min. cooling time of the FA
- Total activity of FA in the cask
- ....

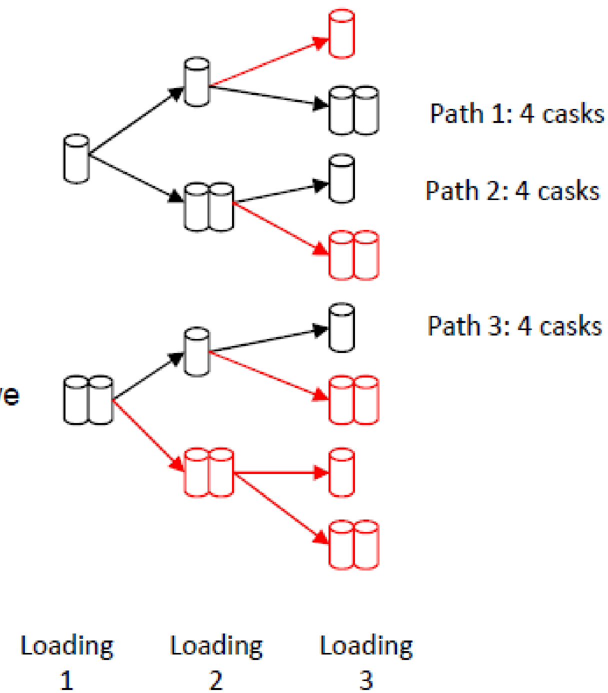
CASKLOAD (Studsvik) optimization code

- 3 loading dates.

- One cask type.

- Load 1 or 2 casks per loading date.

- 4 casks are needed to remove all assemblies from the pool.



- This is the **loading tree.**

- Each brach followed from the start to the end constitutes a **path.**

- Only 3 of the 8 possible paths are possible.

Codes for decay heat & n + ɣ source terms

- Codes for source term calculations must be validated
- Validation methods: comparison “theory-measurement” of nuclide vectors and/or integral measurements
- Licensees usually have to validate codes independently even if they are taken “off-the-shelf”.
- Most popular codes in Germany: SCALE and CASMO/SIMULATE/SNF code package
- For nuclide vector validation PSI PROTEUS Phase II data have been used by PreussenElektra (13 samples)
- Many integral measurements have been done on loaded casks

Quality of nuclide vector predictions (1/2)

- PreussenElektra: comparison with Proteus Phase II data
- Simplified boundary conditions: burnup simulation done in an infinite lattice of assemblies with CASMO5 and match of sample burnup based on Nd149 burnup calibration
- The above analysis has already been done many times and by many groups – but each licensee must do a repetition with exactly the planned production code version and calculation process
- The objective: determination of C/E ratios for relevant nuclides (i.e. n + ɣ radiation outside of dry storage cask)
- Nuclides considered: Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-244, Sr-90, Ru-106, Rh-106, Sb-125, Cs-134, Cs-137, Ba-137m, Ce-144, Pr-144, Pm-147, Eu-154, Eu-155
- The outcome: penalty factors for relevant nuclides or integral penalty factor for radiation field to make conservative prediction of radiation intensity

Quality of nuclide vector predictions (2/2)

- The best cases C/E-1 values are of the order of 10%. Especially for fission products some very large deviations are observed and systematic biases exist

- Oddly, often very large variations exist between samples of similar burnup. They are very difficult to explain.

- From the perspective of reactor physics, i.e. core design, all the above variations appear very large. Reactor simulators like SIMULATE5 can predict critical boron concentrations, control rod worth, fuel assembly power with better precision (usually around 5%).

- Sensitivity analyses concerning influence of operating conditions (i.e. “true” sample burnup) and influence of cross section uncertainties for PROTEUS samples has been performed. In most cases the strength of their influence is about of equal magnitude. They cannot explain variations of sometimes more than 10-20% between samples of similar burnup.

Publications with relevant information

- O. Leray et al., “Nuclear data uncertainty propagation on spent fuel nuclide compositions”, Annals of Nuclear Energy 94 (2016)
- G. Jlas, H. Liljenfeldt, “Decay heat uncertainty for BWR used fuel due to modeling and nuclear data uncertainties”, Nuclear Engineering and Design 319 (2017)
- O. Leray et al., “Uncertainty propagation of fission product yields to nuclide composition and decay heat for a PWR UO2 fuel assembly”, Progress of Nuclear Energy (2017)
- Rochman et al., “Uncertainties for Swiss LWR spent nuclear fuels due to nuclear data”, EPJ Nuclear Sci. Technol. 4, 6 (2018)
- …

Summary & expectations (1/2)

- Source term determination for spent fuel are important for backend costs: i.e. minimum time to defuel reactor building, minimum number of storage containers, minimum size of storage site,…

- A rich knowledge base exists with regard to tracking of XS and operating uncertainties from BOL to EOL. Uncertainties in operating boundary conditions can be reduced by using a validated reactor model for burnup calculations

- From my perspective it seems that a lot of effort recently has gone into “software” side research (i.e. total Monte Carlo methods), not so much into the “hardware” side research (i.e. XS and fission yield measurements)

- Re-evaluation of relevant cross sections (i.e. Cm244 chain) and fission yields (i.e. Ce-144, Pm147) for spent fuel source term determination

- Re-evaluation of experimental uncertainty for nuclide concentration measurements

Summary & expectations (2/2)

- Concerning library acceptance: it would be helpful if a standard set of benchmarks exists which demonstrate the changes in outcome (from the user's perspective) from lib update to lib update and make the value of a new update more transparent

- In the past "problems" with libraries have been addressed by changing some internal "parameters" in the codes to best reproduce validation test cases

- Updating libraries may have unintended consequences if not everything is consistently updated; this may be revealed by running reference benchmarks as mentioned above

- The lib community is very specialized, in the future a more holistic approach might be useful: i.e. stronger collaboration with code developers and end-users

- The lib community suffers from lack of access to "good" integral benchmark data which is often proprietary; NEA might help to facilitate confidentiality agreements