Application of MELCOR to Severe Accident Analyses for Spent Fuel Pools of German Nuclear Power Plants

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Introduction (1 of 3)

- GRS is working on a research project financially supported by the German Federal Ministry of Economics and Technology (BMWi) regarding the extension of probabilistic analyses for spent fuel pools (SFP).
- Appropriate methods for consideration of SFP inside Level 2 PSA shall be developed for both PWR and BWR. Main goals are:
  - identification of the impact of severe accidents inside SFP onto plant behavior and
  - quantification of related releases of radionuclides into environment.
- Supporting deterministic analyses of both the accident progression inside the plant and the structural behavior of the pool structures under the loads of severe accident sequences will be performed.
- The accident progression is being analyzed for both PWR and BWR pools by using the integral codes MELCOR and ASTEC.
- Outcome of these analyses will be the thermal-hydraulic behavior inside SFP under accident conditions, the behavior of embedded structures, and the release of radionuclides. ⇨ Basic approach for consideration of SFP within Level 2 PSA, quantification of event trees and possible mitigative accident measures.

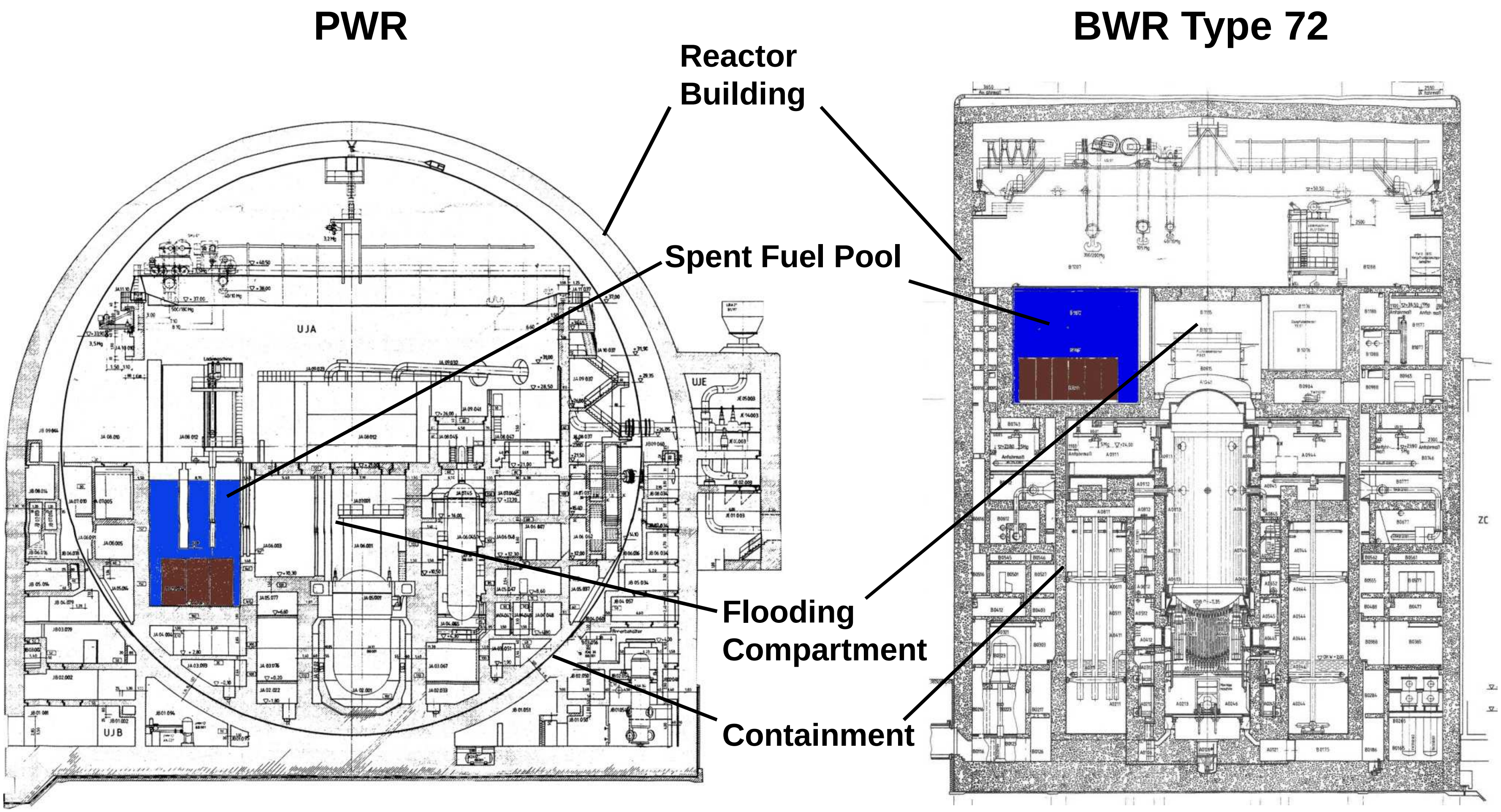
Introduction (2 of 3)

- Range und objectives of the integral severe accident analyses:
  - Description of the phenomenology of severe accident progression inside spent fuel pools of both PWR and BWR from the beginning of evaporation up to the failure of the pool structure.
  - Quantification of:
    - timing of phenomena inside SFP,
    - temperature distributions inside SFP,
    - expected amounts of melt,
    - impact of cladding oxidation under air atmosphere,
    - load of the pool structure due to melt, and
    - radiological risk potential.
  - Provision of data for the foreseen deterministic analyses of the structural behavior of the pool walls by a Finite Element Code ⇨ dedication of failure modes for the pool structures.
  - Support for the development of an event tree of a Level 2 PSA for spent fuel pools.

Introduction (2 of 3)

- General boundary conditions of the MELCOR analyses:
  - consideration of both PWR and BWR spent fuel pools,
  - usage of the spent fuel pool related MELCOR models SFP-PWR and SFP-BWR,
  - beside the SFP, the modeling also includes the containment and adjacent building compartments,
  - passive autocatalytic recombiners (PAR) are considered as realized in the plant,
  - regarding the reactor circuit only the outer structures are modeled,
  - station Black-out is assumed as initial event,
  - different loadings of the pools:
    - partial loading during normal power operation (shortly after finishing in-service inspection ⇨ highest decay heat for that operating mode),
    - typical loading during in-service inspection (connection with filled flooding compartment), and
    - inclusion of the whole core from RPV into SFP; pool separated from flooding compartment (worst case).
  - Containment (PWR) and reactor building (BWR) closed and opened.
- Conceptual differences between reactor types PWR and BWR have to be considered for the analyses.

German PWR and BWR Plants



- **SFP located inside containment**
- **PAR above SFP region**

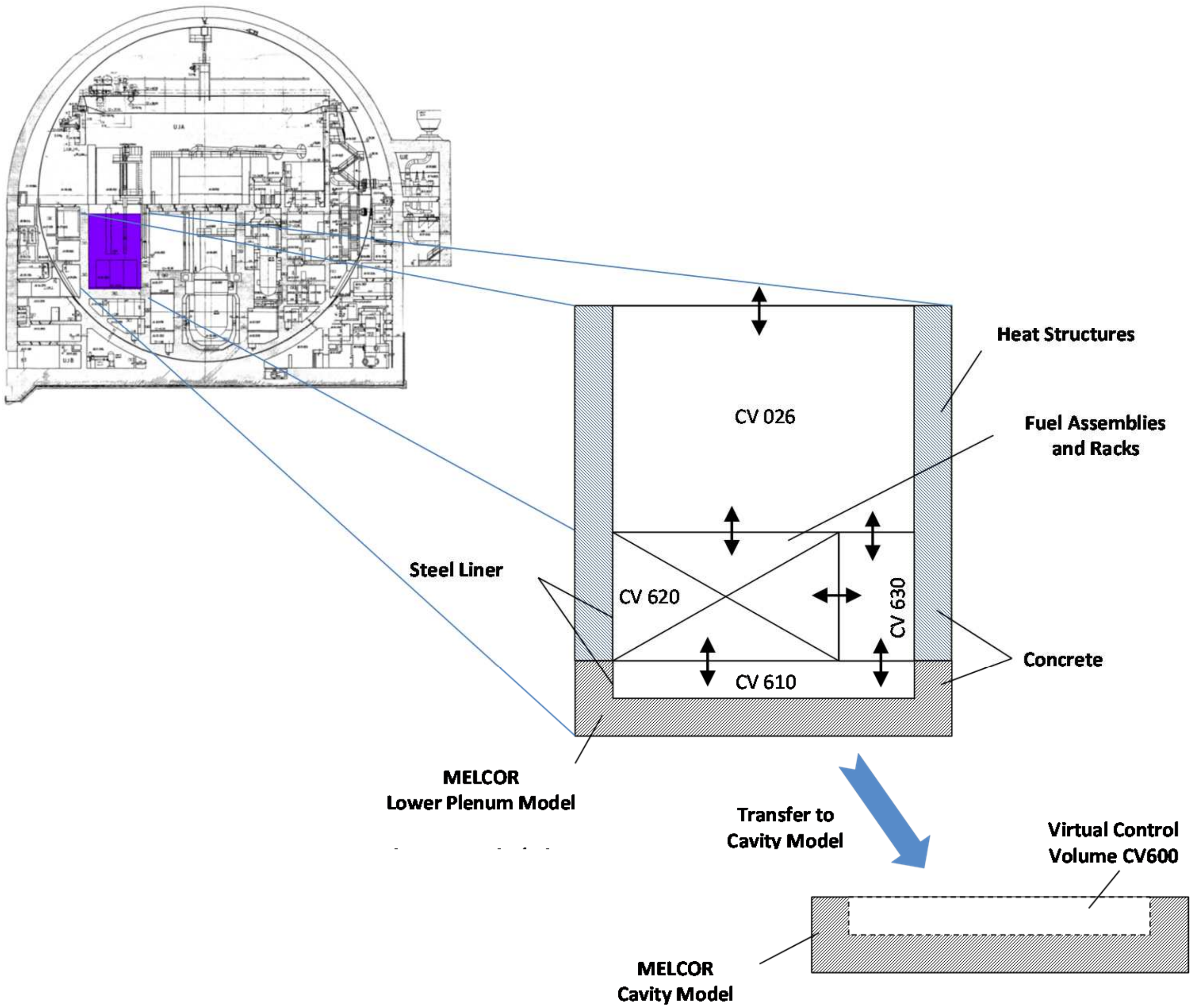
- **SFP located outside containment**
- **No PAR at SFP region**

First Results of MELCOR 1.8.6 for a Generic PWR Spent Fuel Pool

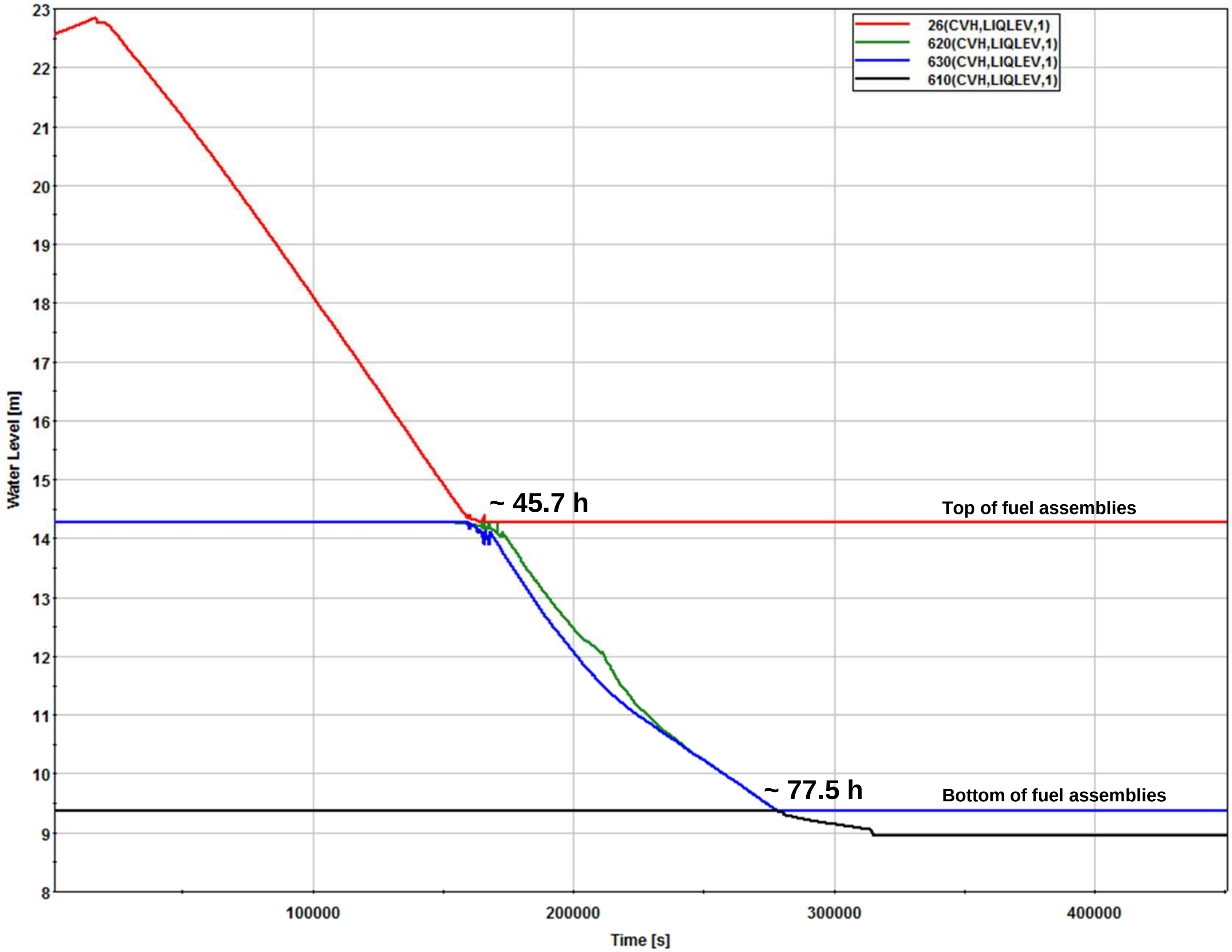
- First results of a MELCOR 1.8.6 analysis of a Station Black-out event are shown. That calculation should demonstrate the applicability of MELCOR on severe accident sequences inside SFP.
- Characteristics of the modeling:
  - typical dimensions of a PWR spent fuel pool (A = 98.2 m^2, height water column = 13.55 m),
  - water volume (≈ 1330 m^3) is being depicted by four control volumes,
  - wall structures of the pool are modeled as heat structures (oxidation and melting possible),
  - bottom area of the pool is modeled by the MELCOR Lower Plenum Model (flat bottom),
  - one core inside the pool (≈ 12.3 MW), pool separated from flooding compartment,
  - inventory of radionuclides like power operation mode, time offset for decay heat of 124 h,
  - 5 radial rings with fuel assemblies, 6^{th} ring as water gap,
  - 3 axial meshes for lower region including supporting plate of the racks, 12 axial meshes for the fuel assemblies, top plate of the racks in the upper axial mesh (COR Package),
  - temperature criterion for the failure of the steel liner at the bottom of the pool, six penetrations, cavity model is switched on with the failure of the liner, and
  - detailed modeling of the containment.

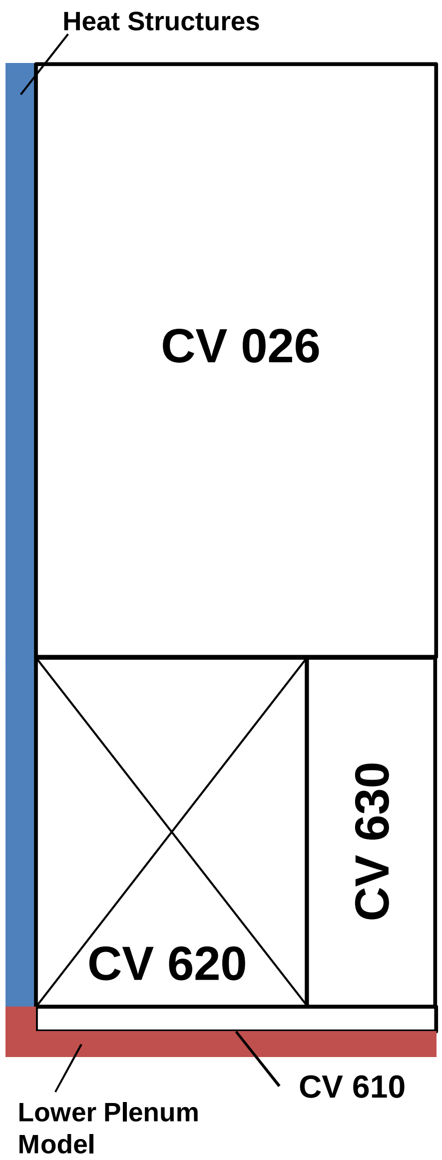
Currently used data doesn`t completely match the real plant conditions!

First Results of MELCOR 1.8.6 for a Generic PWR Spent Fuel Pool – Nodalisation

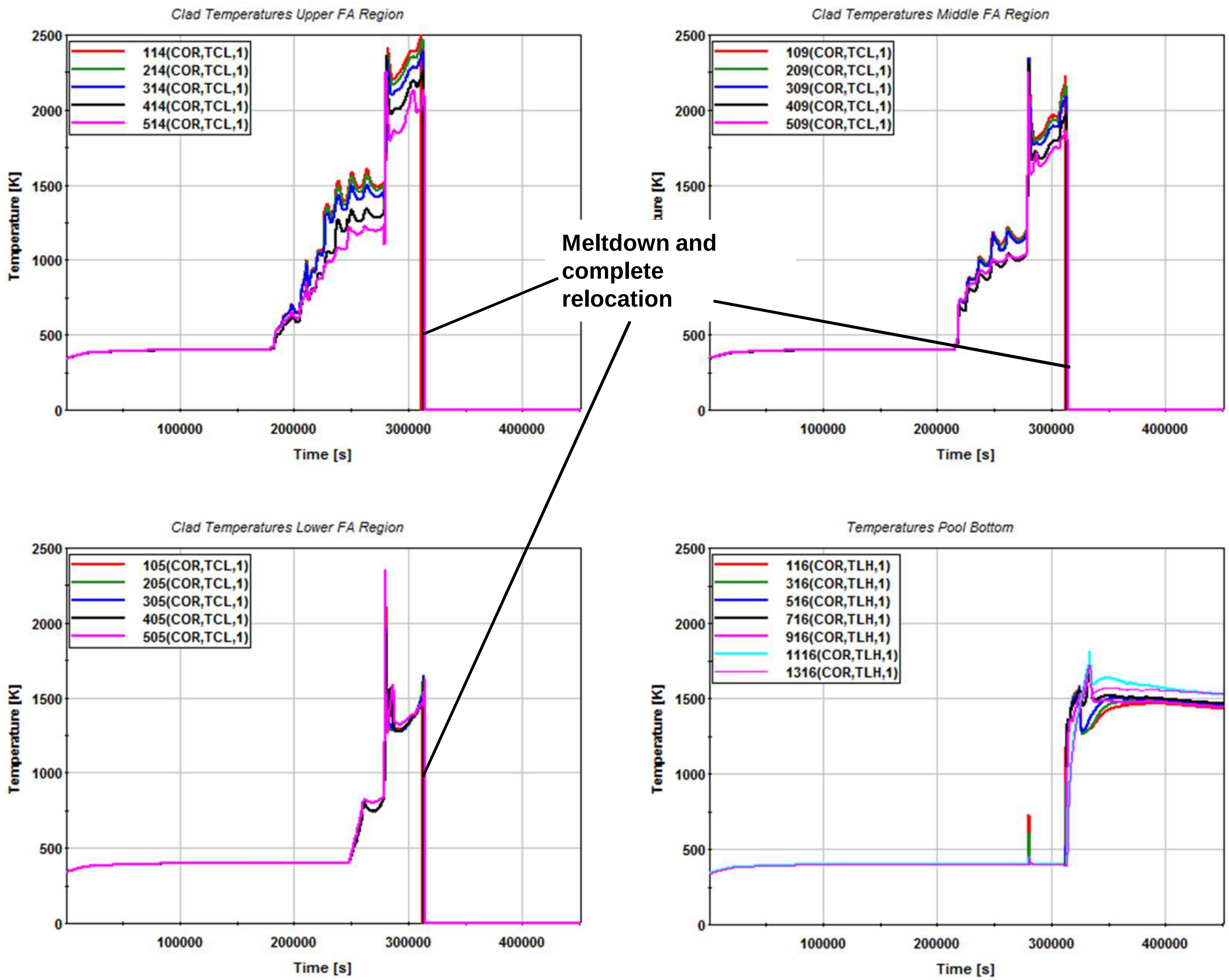


First Results of MELCOR 1.8.6 for a Generic PWR Spent Fuel Pool – Calculated Water Level

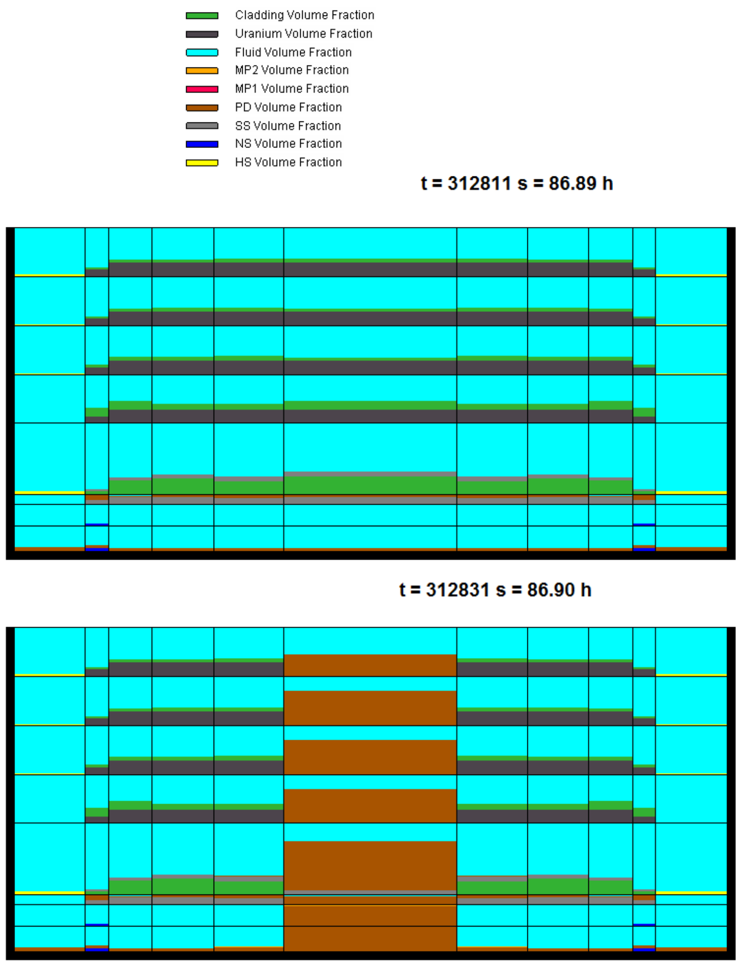


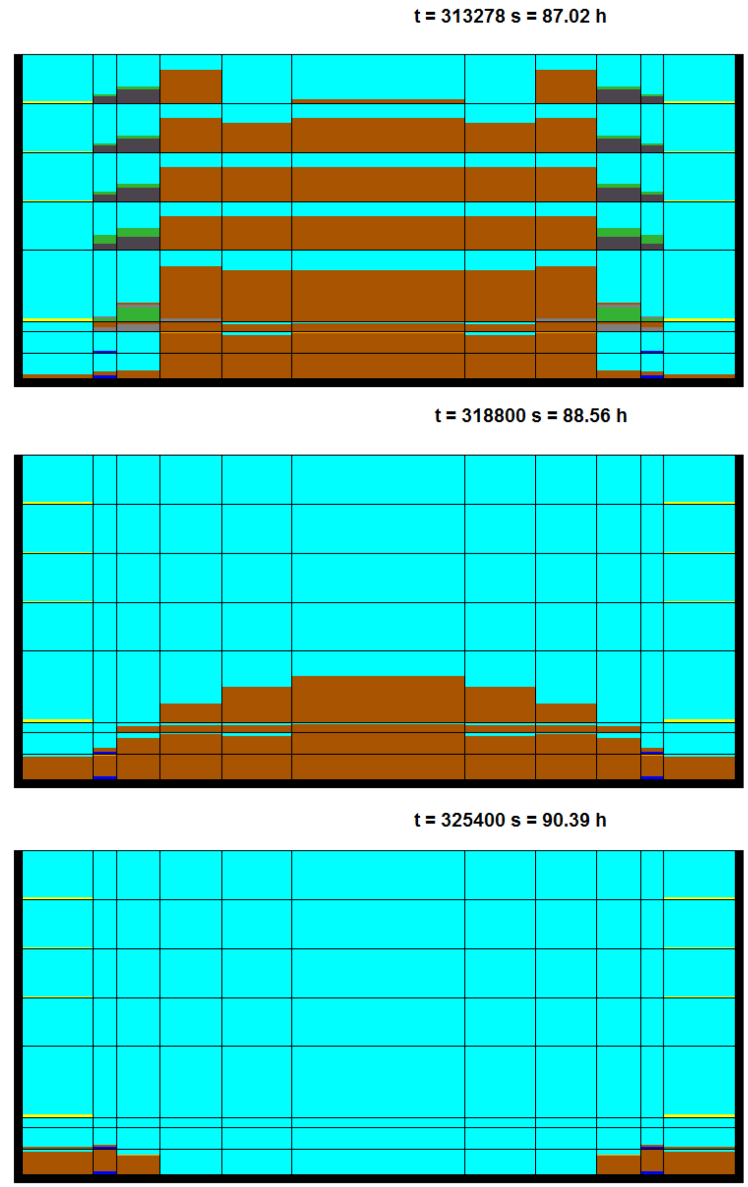


First Results of MELCOR 1.8.6 for a Generic PWR Spent Fuel Pool – Calculated Temperatures



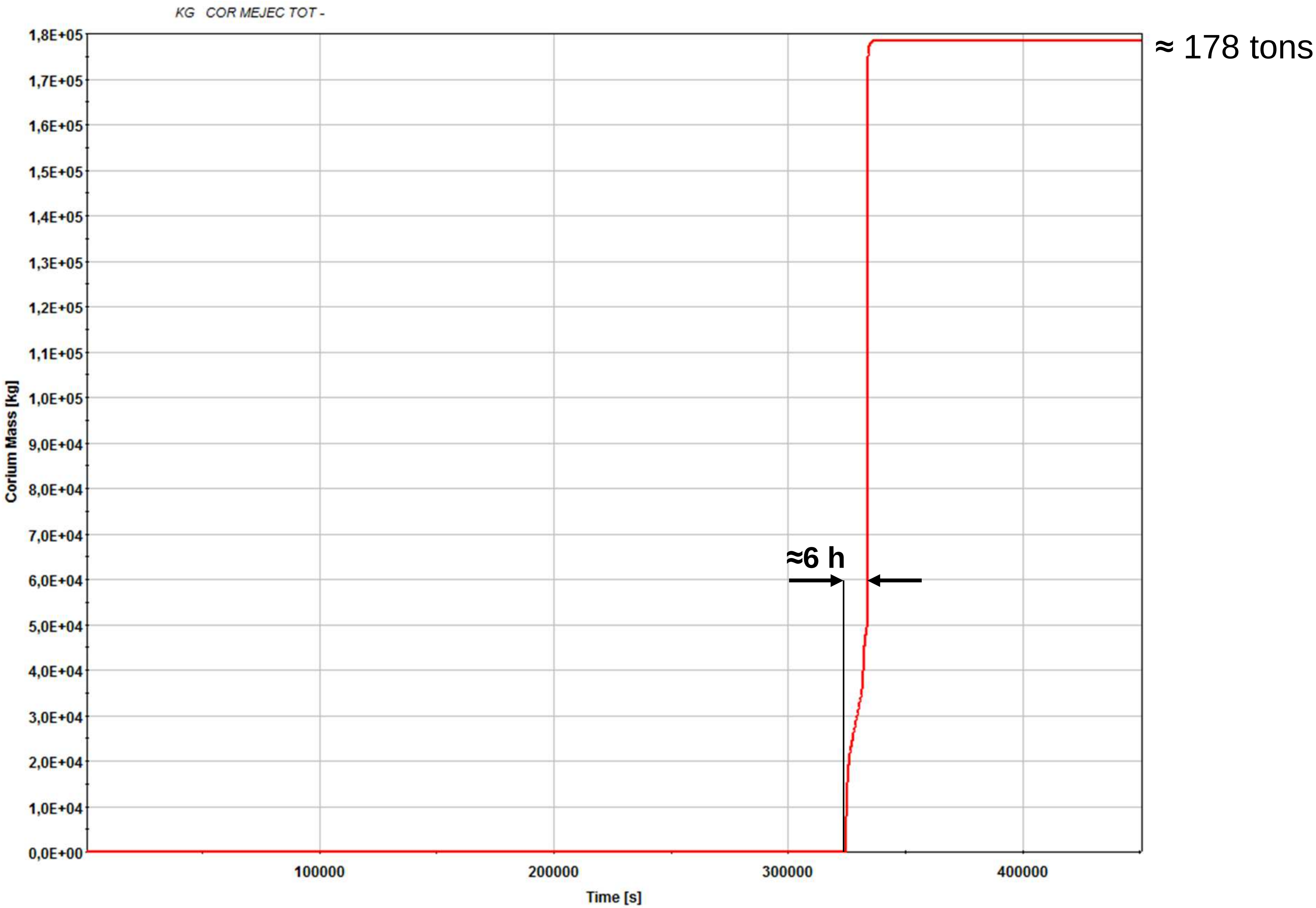
First Results of MELCOR 1.8.6 for a Generic PWR Spent Fuel Pool – Calculated Transfer into Cavity (lower eight meshes)



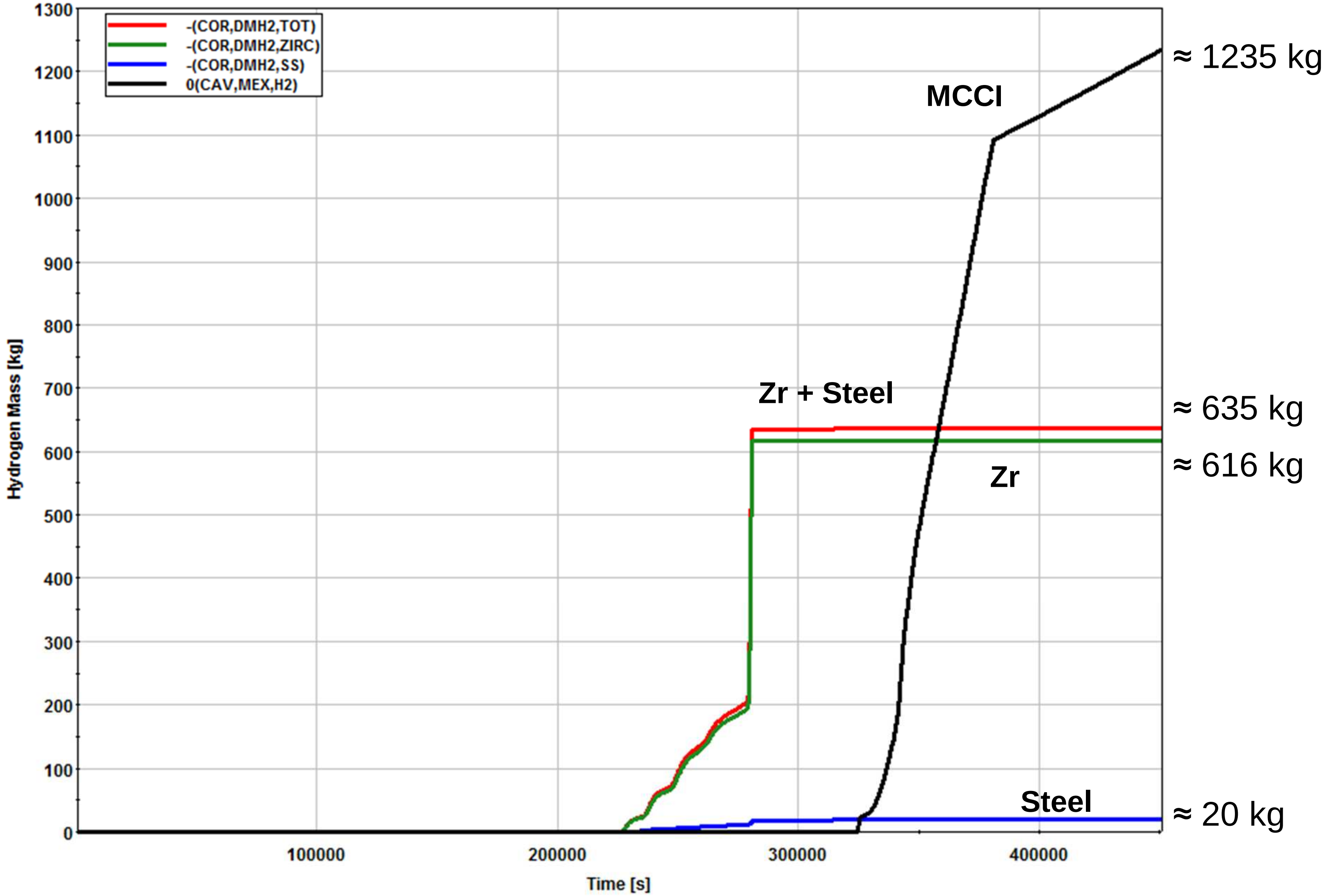


- Failure of penetrations between 86.90 h and 87.02 h
- End of relocation into cavity at 93.0 h

First Results of MELCOR 1.8.6 for a Generic PWR Spent Fuel Pool – Calculated Mass of Corium Ejected



First Results of MELCOR 1.8.6 for a Generic PWR Spent Fuel Pool – Calculated Hydrogen Masses



Lessons Learned and Open Issues

- **Lessons learned:**
  - MELCOR is able to calculate a severe accident sequence inside SFP with long-lasting evaporation (Station Black-out, no leak at the pool).
  - Current results seem to be reasonable from a point of engineering judgment.
  - Usage of the virtual volume connected to the CAV Package seems to work well in order to trigger the MCCI simultaneously to the failure of the steel liner of the pool.
- **Open issues:**
  - Contribution of oxidation under air atmosphere during long-lasting evaporation sequences ⇨ combination of oxidation models under steam and air atmosphere (extent of oxidation under air dependent of the steam fraction?)?
  - Concrete is assumed as an isolation (NINSLH is used) in the lower plenum model ⇨ concrete layer is not considered in the creep model?
  - Failure model of the fuel rods (temperature criterion or lifetime model?).

Future Activities Regarding MELCOR

- Final adjustment of the PWR input deck with a couple of real plant data, e.g.:
  - mass distribution of racks and tools embedded in the pool,
  - consideration of spent fuel assemblies of previous cycles (low decay rate),
  - modification of the inventory of radionuclides.
- Extension of the analyses on the two other operational modes with different loadings of the pools.
- Consideration of additional release paths in case of an open containment (only for operational mode during in-service inspections and if containment cannot be easily closed before reaching damage of fuel assemblies).
- Upgrading of the input deck with a RPV for the operational mode with connection to the flooding compartment (typical loading during in-service inspection) ⇨ fuel assemblies remaining inside RPV has to be modeled by heat structures.
- Consideration of current experimental investigations concerning SFP, like Sandia Fuel Program.
- Application of MELCOR for a spent fuel pool of a German BWR Type 72.
- Usage of MELCOR for severe accident analyses during shutdown modes of both PWR and BWR plants (new project, running since April 2012 at GRS).