Application of MELCOR at GRS Regarding Spent Fuel Pool Analyses and Assessment of SAMG Procedures

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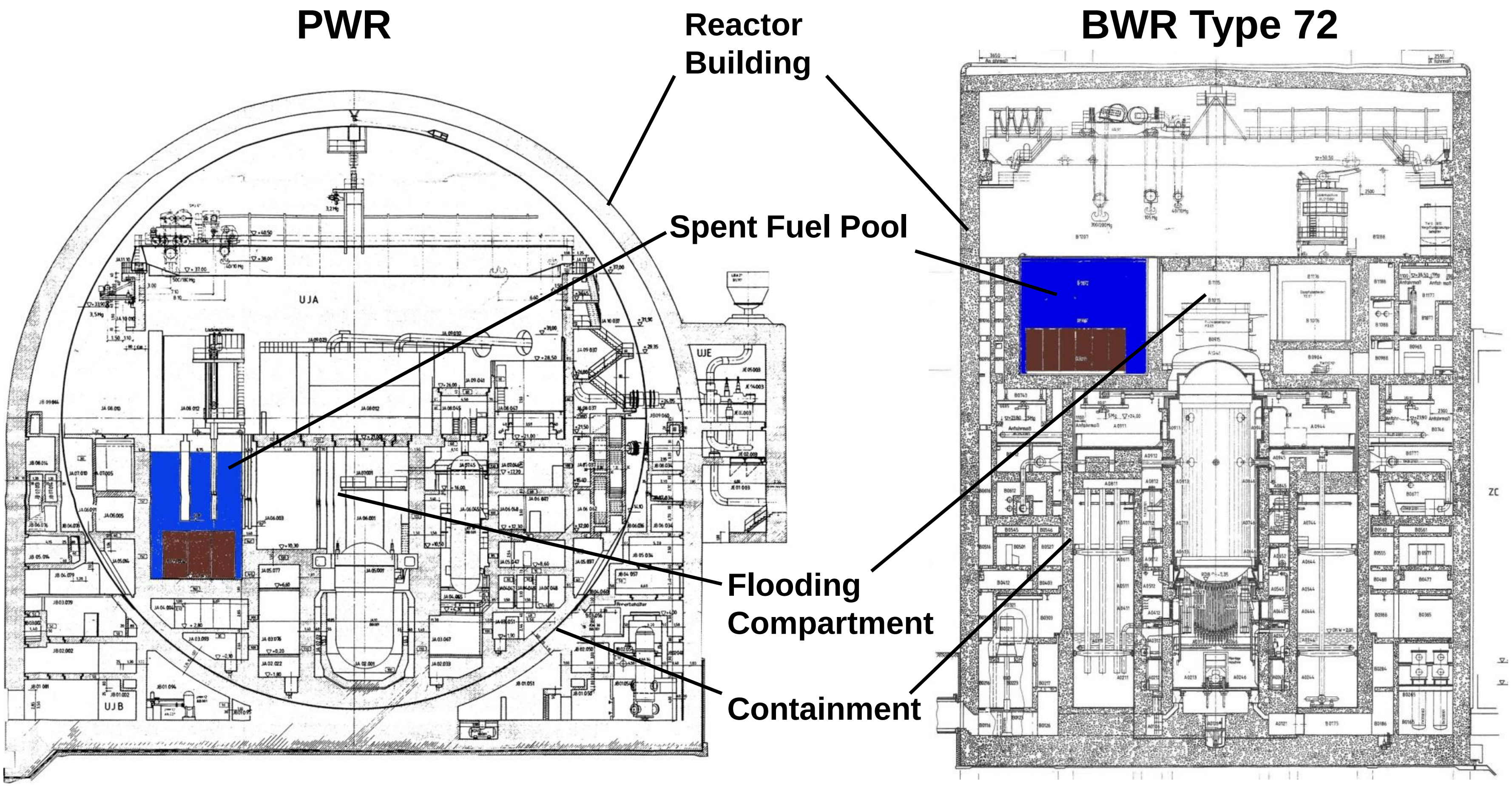
Introduction

- GRS is using the MELCOR Code since 1992. The application of MELCOR started with Code Version 1.8.3 and currently we are using MELCOR 1.8.6.
- The main application of MELCOR at GRS are:
  - Assessment of accident management measures realized at German NPPs, like
    - the assessment of a **P**assive **A**utocatalytic **R**ecombiner (PAR) system, and
    - secondary and primary side bleed & feed.
  - Support of PSA Level 2 Studies performed at GRS:
    - PSA Level 2 BWR Type 72,
    - PSA Level 2 PWR 1300 MW,
    - PSA Level 2 BWR Type 69, and
    - PSA Level 2 PHWR.
- Two current projects which are mainly focused on the application of MELCOR are regarding
  - severe accident analyses of spent fuel pools (recently terminated), and
  - severe accident analyses for the assessment of SAM measures at German PWR (running).

Severe Accident Analyses for Spent Fuel Pools of PWR and BWR – General Boundary Conditions

- General boundary conditions of the MELCOR analyses:
  - the modeling includes spent fuel pool, reactor circuit, containment/reactor building (closed) and relevant compartments of adjacent buildings,
  - passive autocatalytic recombiners (PARs) and filtered containment venting are considered as realized in the plants, other SAM measures have not be considered,
  - different loadings of the pools:
    - **standard loading** during normal power operation (shortly after finishing in-service inspection ⇨ highest decay heat for that operating mode),
    - **partial loading** during in-service inspection (one third of the core has been moved into SFP; connection with filled flooding compartment and reactor pressure vessel),
    - **full loading:** inclusion of the whole core from RPV into SFP; pool connected to filled flooding compartment and RPV, and
    - **full loading:** inclusion of the whole core from RPV into SFP; pool separated from flooding compartment (worst case regarding the timing of severe accident sequence).
  - Postulated initiating events:
    - Long lasting failure of SFP heat removal is assumed for standard loading and
    - “Station Black-out” is assumed for both partial loading and full loading.
- Conceptual differences between reactor types PWR and BWR.

Severe Accident Analyses for Spent Fuel Pools PWR and BWR – German PWR and BWR Plants



- SFP located inside containment
- PARs above SFP region

- SFP located outside containment
- no PARs at SFP region

Severe Accident Analyses for PWR Spent Fuel Pools – Characteristics of Modeling (1)

- Embedded masses of structures (racks, tools, fuel assemblies, FA adapters etc.):

	Standard Loading	Partial Loading, Flooding Compartment filled	Full Loading, Flooding Compartment filled	Full Loading, Pool separated
Total mass of embedded structures	595.725 t	653.080 t	766.211 t	766.211 t

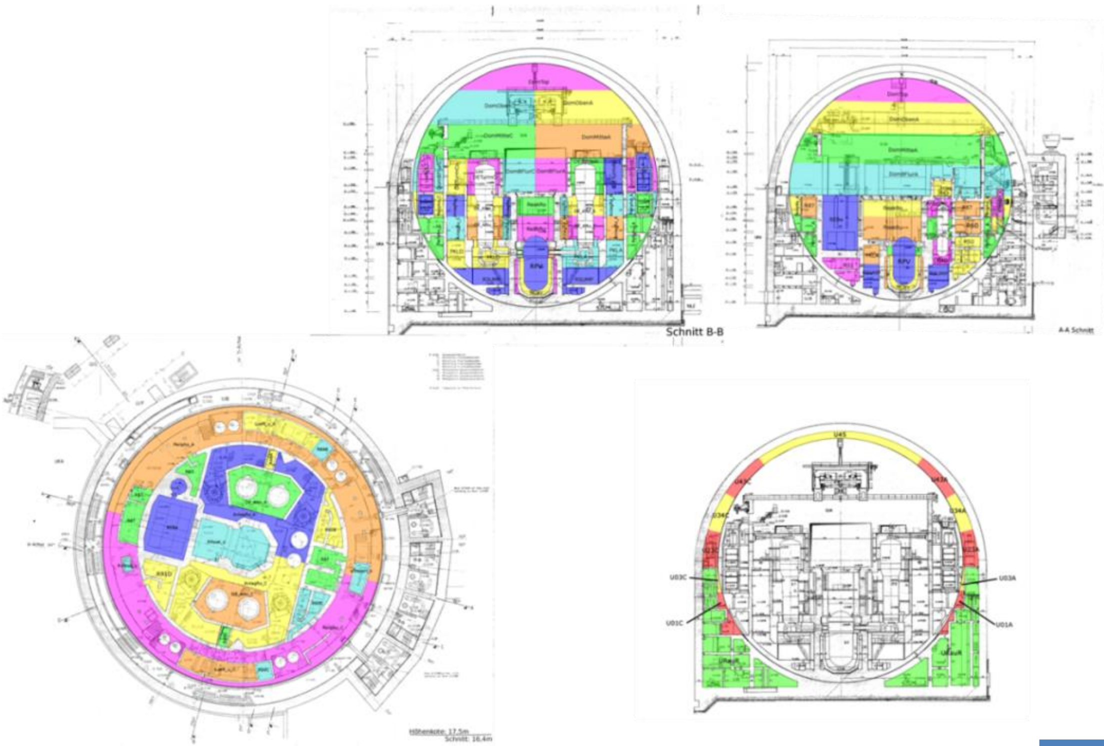
- Number of fuel assemblies (FA) and decay heat (DH) at initiation of the event:

	Standard Loading	Partial Loading, Flooding Compartment filled	Full Loading, Flooding Compartment filled	Full Loading, Pool separated
Number of FA from core	-	65 (≈1/3 of Core)	193	193
Number of "old" FA	475	475	475	475
DH from core [MW]	-	4.11	12.35	12.35
DH "old" FA [MW]	2.43	1.25	1.25	1.25
Total DH [MW]	2.43	5.36	13.60	13.60

- Inventories of radionuclides provided by the utility of the reference plant are considered for both FA from core and "old" FA embedded inside SFP.

Severe Accident Analyses for PWR Spent Fuel Pools – Characteristics of Modeling (2)

- Nodalisation of the containment.



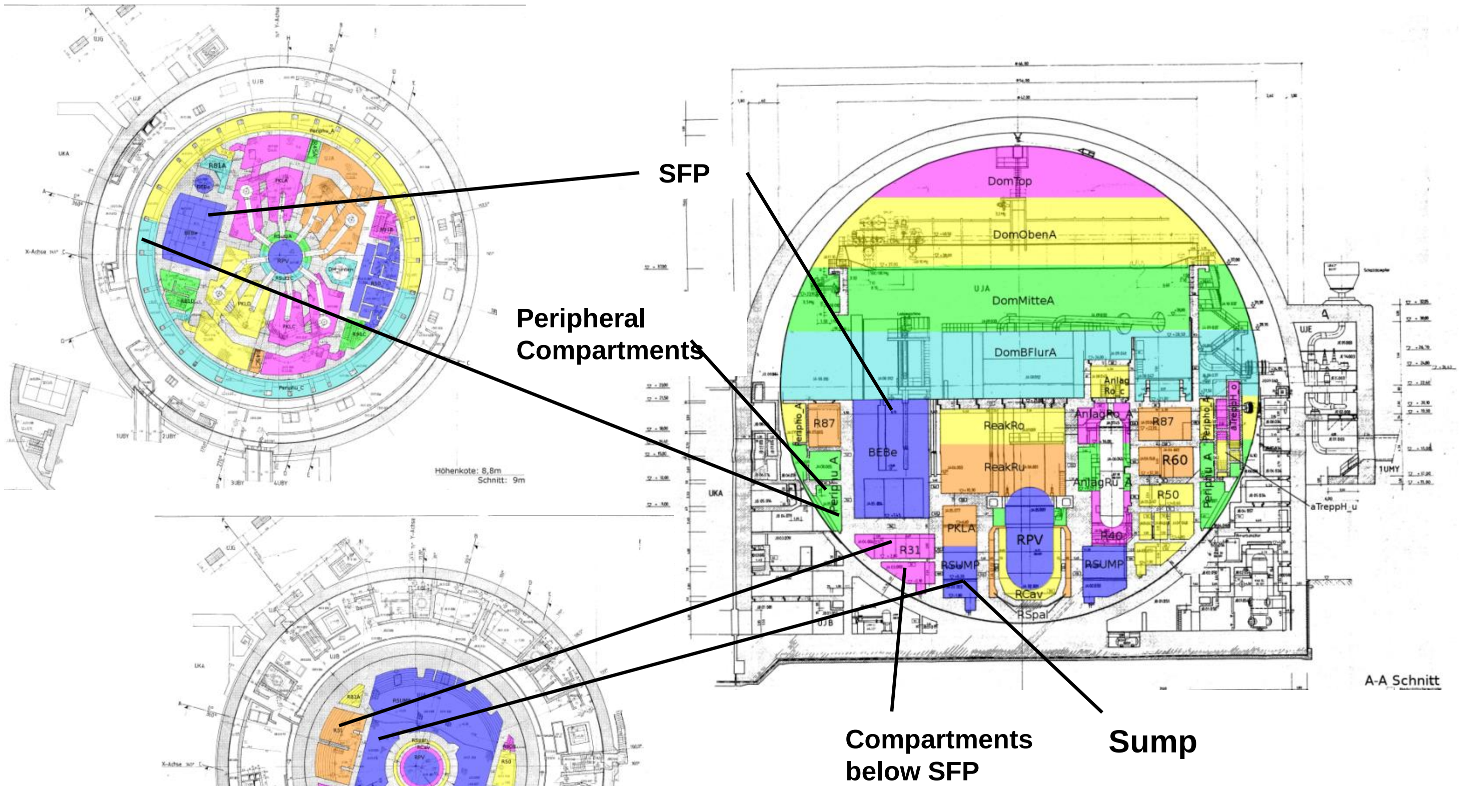
Sections of reactor building for a German PWR

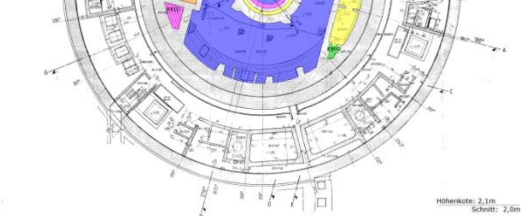
- Detailed modelling of the containment and reactor building annulus.
- Consideration of 58 passive autocatalytic recombiners (PARs) distributed on 37 control volumes.
- Depletion of hydrogen and carbon monoxide is calculated by the PAR model.

Region / Components:	CV	HS	FL
Containment	77	228	263
RB Annulus	12	43	26
Burst Elements (Doors/BM)	82/56		
PARs	58 PARs inside 37 CVs		

Severe Accident Analyses for PWR Spent Fuel Pools – Characteristics of Modeling (3)

- Cavities for calculating MCCI:

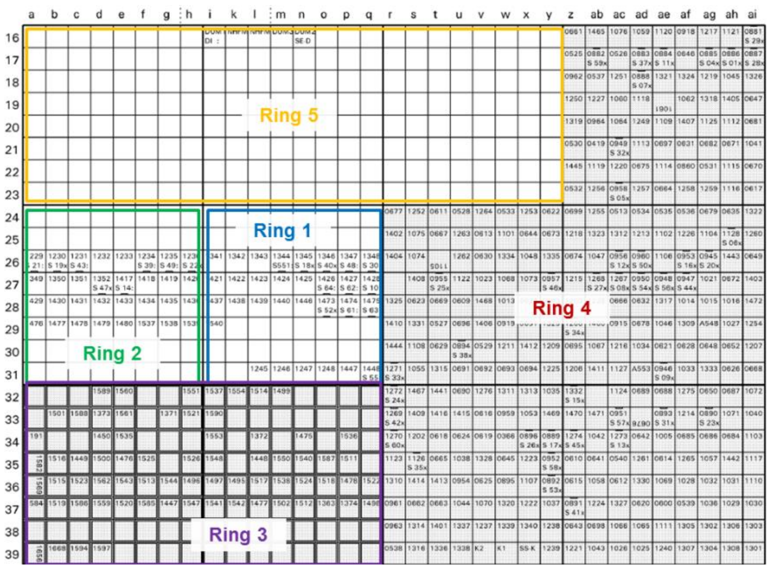




- Cavity 0: Lower region SFP
- Cavity 1: Compartments below SFP
- Cavity 2: Compartments beside SFP
- Cavity 3: Containment sump

Severe Accident Analyses for PWR Spent Fuel Pools – Characteristics of Modeling (4)

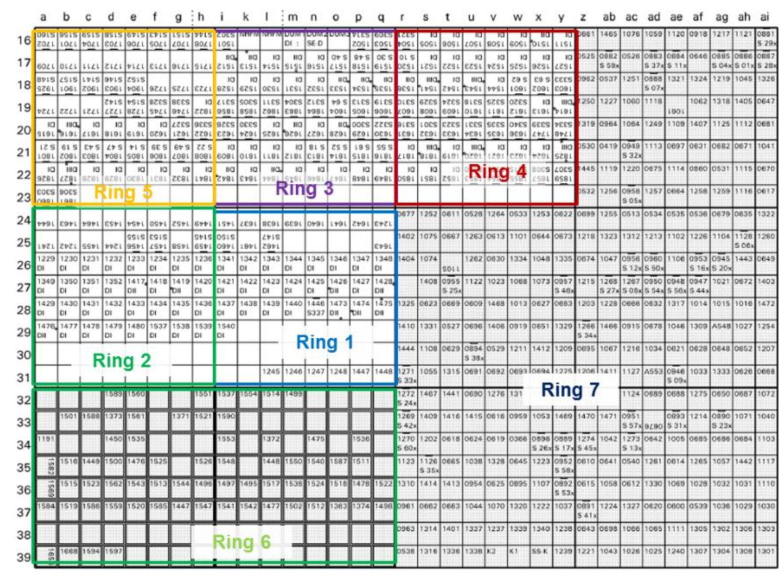
- Loading and assigning to radial rings:



Fuel assembly arrangement (standard loading)

- For partial loading the arrangement of full loading is used with only one-third of the total numbers of FA from core inside radial rings 1 to 5.

- 16 racks (8 x 8) and 3 racks (1 x 8)
- 5 radial rings for the standard loading (5th ring models empty racks and gaps between the racks),
- 8 radial rings for partial and full loading (8th ring models the gaps between the racks)



Fuel assembly arrangement (full loading)

Severe Accident Analyses for PWR Spent Fuel Pools – Calculated Progression of the PWR Sequences

Analysis with four cavities ⇨ Relocation into sump

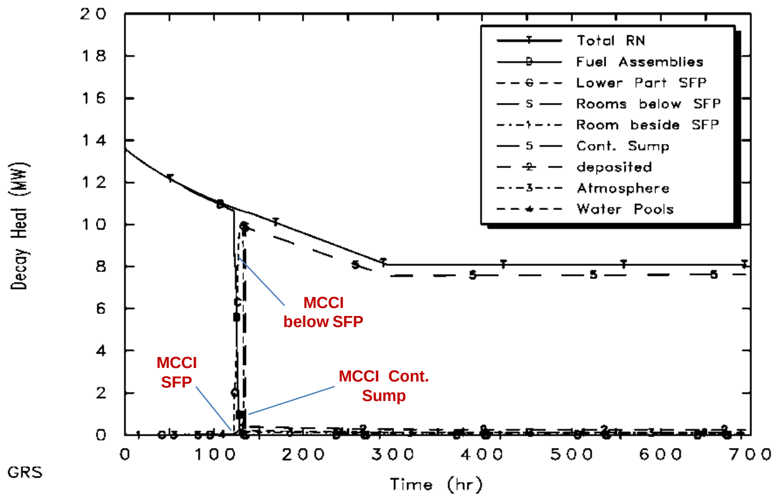
	Standard Loading	Partial Loading	Full Loading, Flooding Compartment filled	Full Loading, Pool separated***
Initiating event	00:00:00	00:00:00	00:00:00	00:00:00
Water Level at top edge of racks	342:08:20	105:41:40	112:58:20	50:28:30
Failure Fuel Assemblies	-	538:34:30	165:42:00	65:29:44
Water Level at lower support plate of racks	677:38:20	239:04:00*	174:02:54	99:03:20
Start of significant relocation	-	538:50:30	166:01:40*	82:26:40*
Water completely evaporated	-	284:51:40*	180:31:40	108:50:00
Failure of steel liner	-	538:53:40	185:22.31	109:25:00
Start of MCCI	-	-	366:15:50	121:20:03
Failure of concrete of the bottom of SFP	-	-	413:36:40	132:06:14
Relocation in compartments below SFP	-	-	413:36:40	132:06:14
Relocation in to sump	-	-	-**	134:23:20
Start first venting	-	-	477:48:20	-
Stop first venting	-	-	483:38:20	-
End of calculation	694:26:40	694:26:40	497:28:20	694:26:40

Relocation very late; small contribution of oxidation under air atmosphere; no MCCI

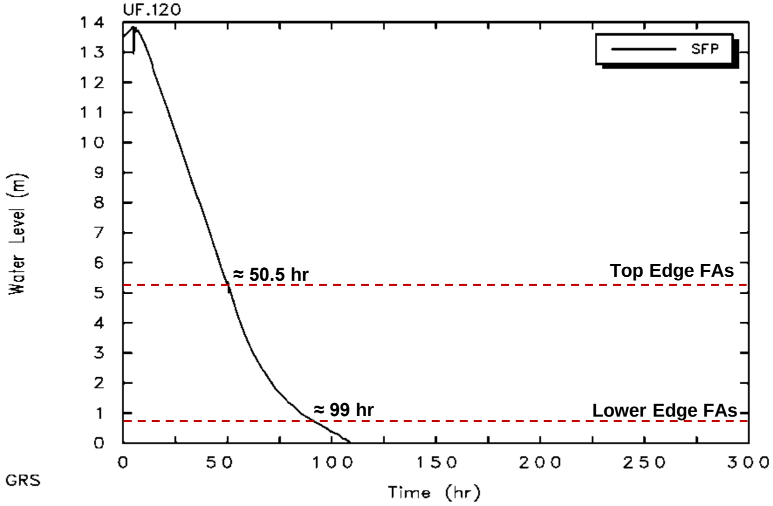
* Calculated point in time cannot be depicted in the right chronological order
** No 4th cavity (sump) has been used
*** Calculation with four cavities

Analysis with only three cavities ⇨ No relocation into sump; no termination of MCCI ⇨ filtered containment venting

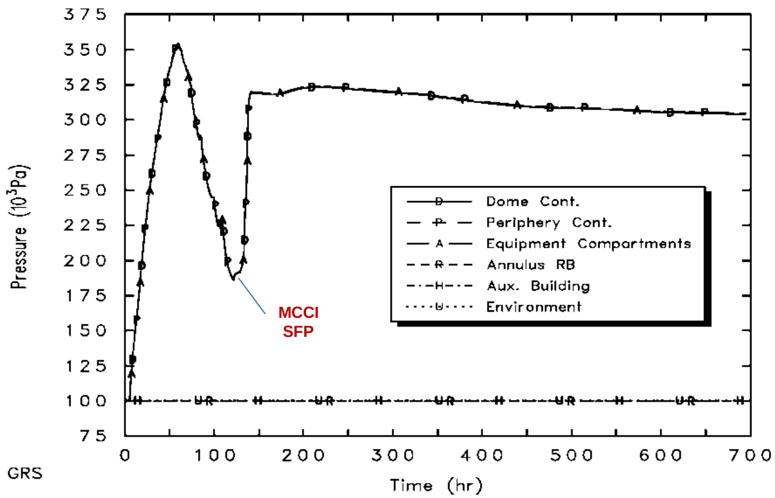
Severe Accident Analyses for PWR Spent Fuel Pools –
Results for a PWR Spent Fuel Pool; SFP Fully Loaded and Separated (1)



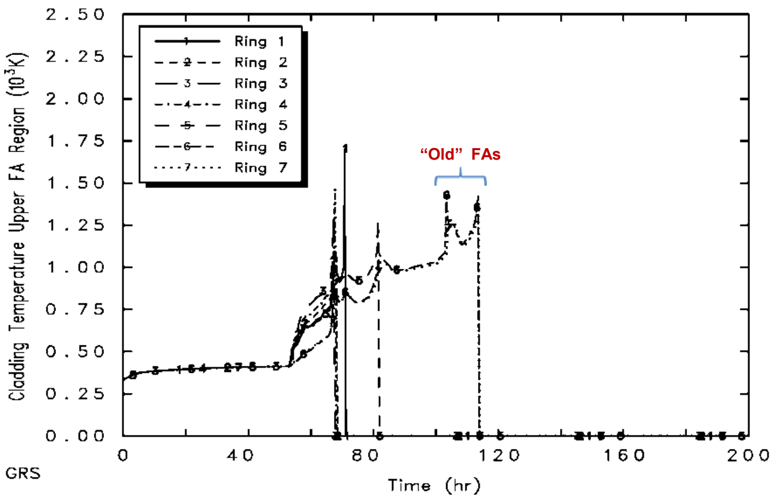
Distribution of Decay Heat



Water Level SFP

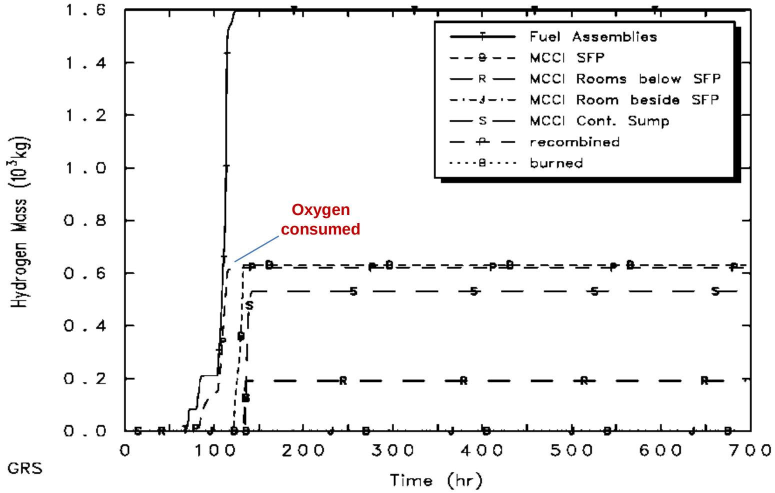


Pressure Containment, Aux. Building, Environment

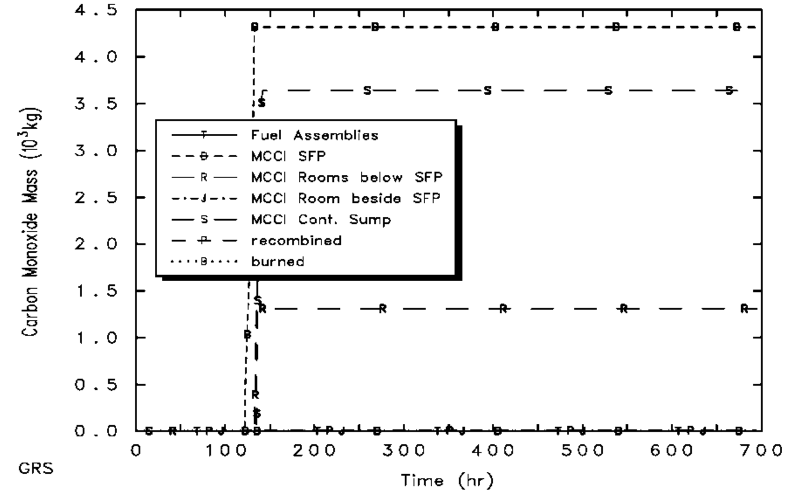


Cladding Temperature upper Level of FAs

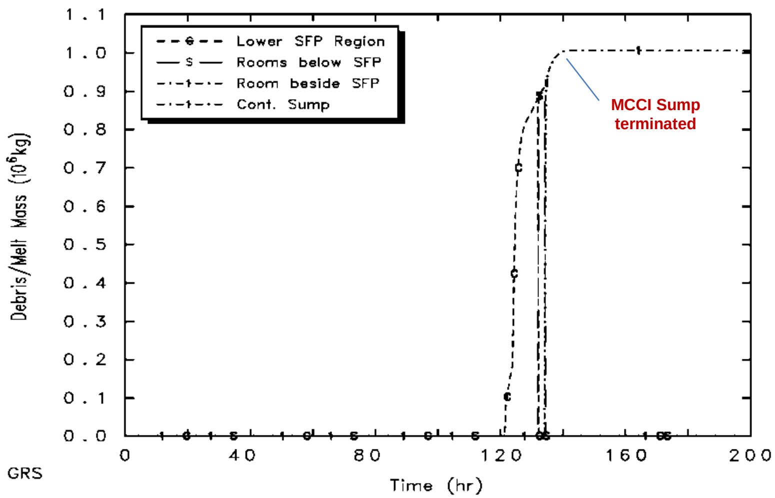
Severe Accident Analyses for PWR Spent Fuel Pools – Results for a PWR Spent Fuel Pool; SFP Fully Loaded and Separated (2)



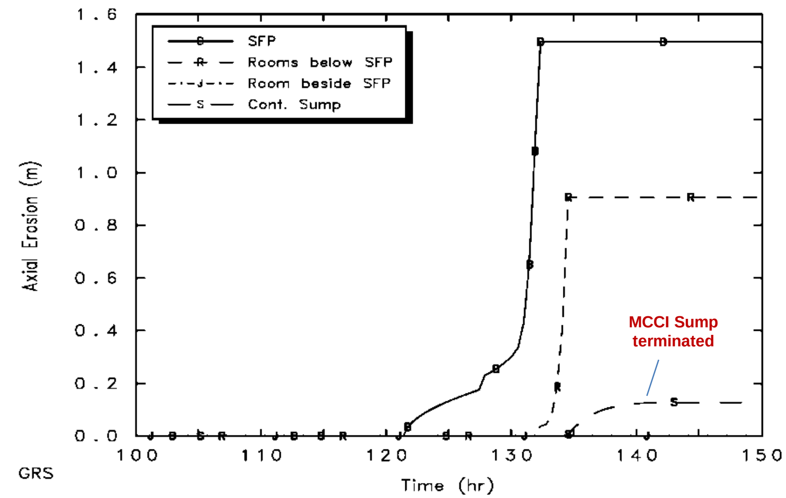
Generation and Reduction of Hydrogen



Generation and Reduction of Carbon Monoxide

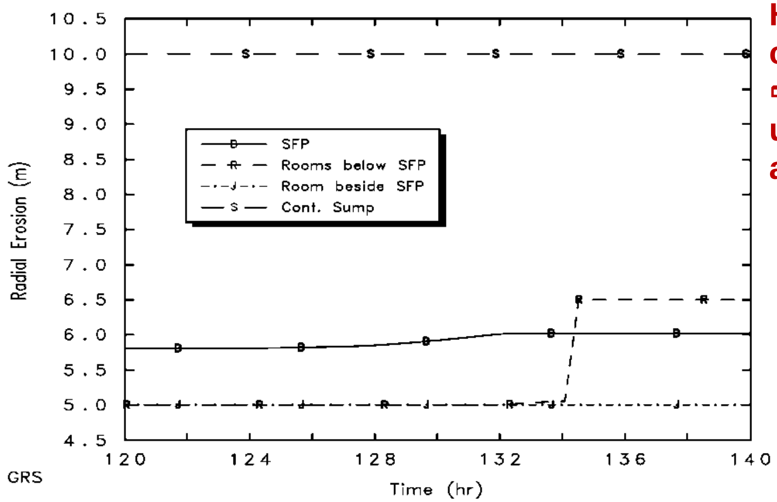


Relocated Mass of Debris/Melt



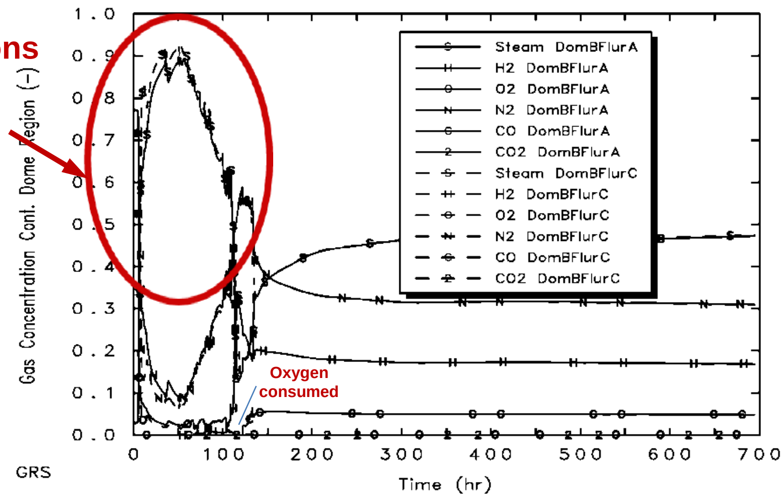
Axial Erosion

Severe Accident Analyses for PWR Spent Fuel Pools – Results for a PWR Spent Fuel Pool; SFP Fully Loaded and Separated (3)



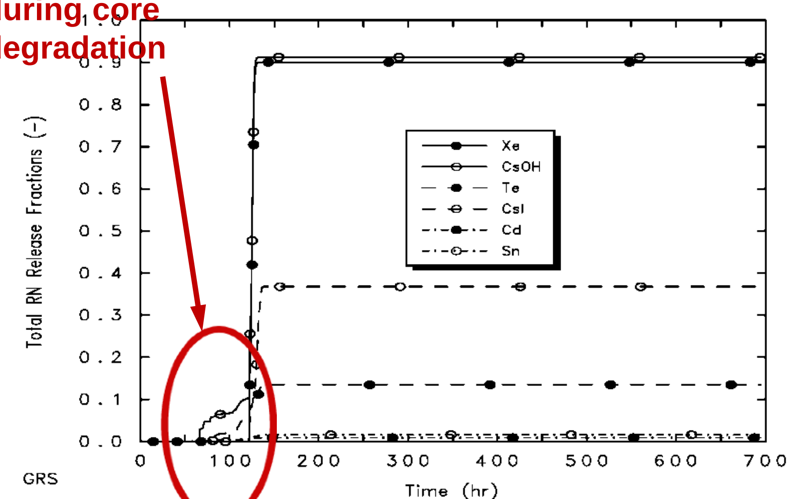
Radial Erosion

High Steam concentrations ⇨ Oxidation under steam atmosphere

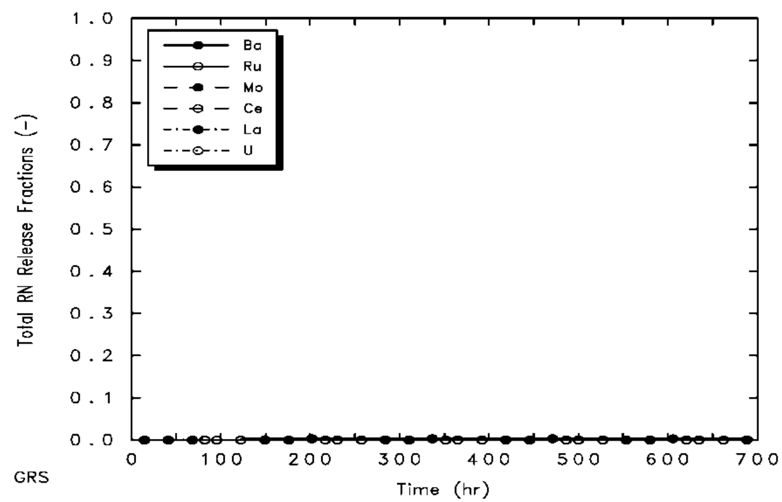


Gas Concentrations Containment Dome

Small releases during core degradation

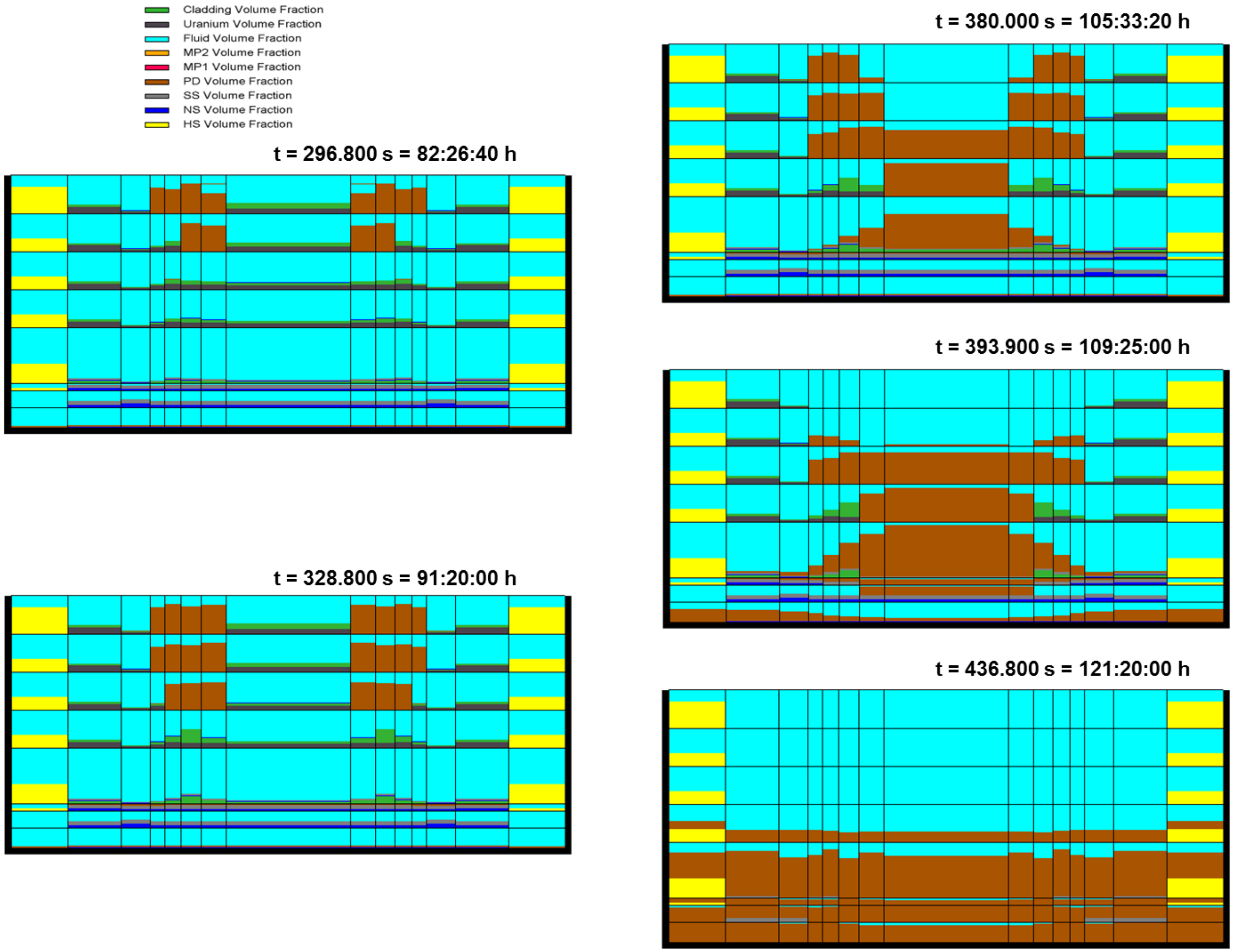


Total Release of Noble Gasses and High Volatile Classes



Total Release of Less Volatile Classes

Severe Accident Analyses for PWR Spent Fuel Pools – Results for a PWR Spent Fuel Pool; SFP Fully Loaded and Separated (4)



- Lower part of SFP (lower 8 axial meshes)
- First relocation to SFP bottom at 82:26 hours,
- Failure of steel liner at 109:25 hours
- Start of transfer into cavity at 121:20 hours (about 788 tons)

Transfer into Cavity 0

Severe Accident Analyses for Spent Fuel Pools PWR and BWR – Lessons Learned from Deterministic SA Analyses of SFPs (1)

PWR and BWR:

- Very unlikely events especially SBO with fully loaded and separated SFP only possible during repair work inside RPV.
- Leaks in the lower part of the SFP are considered as practically eliminated for German NPPs due to the design of the pool (only pipe connections in the upper part of SFP, at least 6 m above the top edge of racks; design against earthquakes between VI and VIII on the EMS/MSK scale).
- Evaporation extends over several days ⇨ steam concentrations inside SFP region and containment/reactor building remain at large values ⇨ impact of air oxidation small for transients.
- Only for low decay heat inside SFP, where uncovering of the fuel assemblies is terminated before their heat-up, air oxidation could occur after steam concentration has been depleted.

PWR:

- Heat transfer by thermal radiation on the containment (calculated by a control function model) ⇨ stronger heat-up of the containment above SFP can be expected ⇨ an endangerment of its integrity for German PWR could be excluded by a structural mechanics assessment.
- It might be helpful to initiate filtered containment venting earlier in case of SA inside SFP in order to reduce the thermal load of the containment.
- For PWR it is very likely that SA sequences inside SFP run into filtered containment venting.

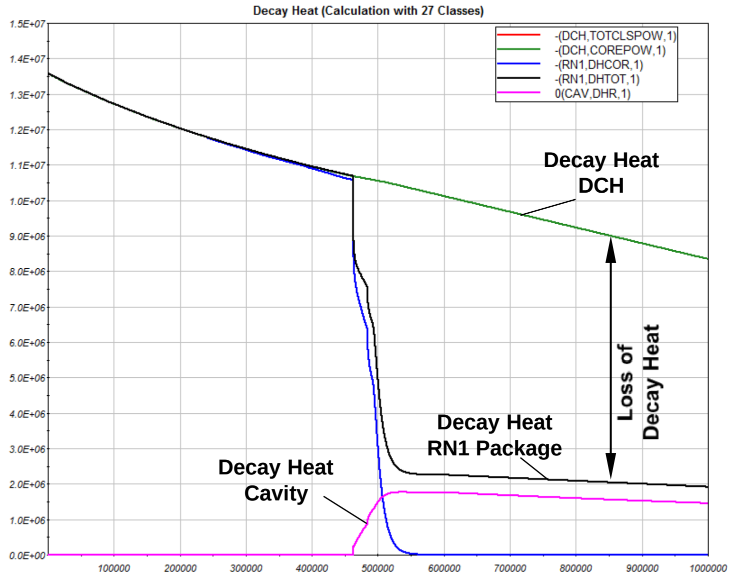
Severe Accident Analyses for Spent Fuel Pools PWR and BWR – Lessons Learned from Deterministic SA Analyses of SFPs (2)

BWR:

- All sequences progress into an unfiltered release into environment due to open ventilation ducts and/or open connection to the turbine hall by failed pressure flaps.
- During the damage of fuel assemblies and the oxidation of the fuel rod cladding and canisters deflagrations occur above the SFP region; only a moderate pressure increase can be expected due to open connection to the environment.
- A thermal loading of the concrete structures of the reactor building above the SFP might be possible.

Severe Accident Analyses for Spent Fuel Pools PWR and BWR – Lessons Learned from Deterministic SA Analyses of SFPs (3)

- Application of MELCOR:
  - Defining of new RN classes has been tried out in order to consider the different radionuclides inventory of both FAs from core and very “old” FAs. Worked well up to start of transfer to Cavity Package ⇨ wrong results regarding decay heat (loss of decay heat).



  - Only minor releases of radionuclides calculated during core damage phase. Major releases calculated during melt down of the debris bed. Application of release models onto low fuel temperatures during core damage.

Severe Accident Analyses for Spent Fuel Pools – Probabilistic Results (1)

- Development of a first approach for consideration of spent fuel pools in a Level 2 PSA under consideration of the results of the deterministic SA analyses.
- Two sections of the Level 2 event tree for SFP:
  - 1st Phase: Uncovering of the fuel assemblies – start of significant relocation of debris/melt onto SFP bottom
    - Phenomena similar to the in-vessel SA scenario of the reactor in addition with SFP specific issues:
      - Cladding oxidation under air atmosphere
      - Thermal loading of structures of SFP
      - Thermal loading of containment structures above SFP
      - Re-criticality issues, and
      - Loading of reactor building and release into environment (BWR)
  - 2nd Phase: Relocation of significant amount of corium onto SFP bottom – termination of release into environment
    - Phenomena similar to the ex-vessel SA scenario of the reactor in addition with SFP specific issues:
      - Relocation of corium into underlying compartments,
      - Melt attack to containment structure from inside beside the SFP (German PWR),
      - Melt attack to the containment from outside (German BWR),
      - Potential flooding of corium by sump water (German PWR), and
      - flooding of corium by condensed water inside reactor building (German BWR).
- 15 branches of Level 2 event tree has been considered for PWR.

Severe Accident Analyses for Spent Fuel Pools (cont'd) – Probabilistic Results (2)

- Release categories and their calculated contributions:
  - **PWR:**

Release Categories	Frequency for transition from CDS → RC
FKA-BE (open containment lock)	1.00 %
FKB1-BE (no containment isolation)	0.10 %
FKB2-BE (early failure of containment)	0.11 %
FKE-BE (late failure of containment)	5.37 %
FKF-BE (Unfiltered Venting, near the ground)	0.09 %
FKH-BE (Filtered Venting, near the ground)	0.84 %
FKI-BE (Filtered Venting, stack)	92.49 %
FKJ-BE (only design leakage containment)	0,00 %

  - **BWR:**
    - Only one release category can be considered (unfiltered release through ventilation ducts and/or via turbine hall)
    - Differentiation can be made regarding potential deposition of radionuclides in the reactor building, ventilation system, and turbine hall.

Severe Accident Analyses for Assessment of SAMG Procedures – SAM Concept German PWRs (1)

- In the plants are available: Operational Manual, Emergency Operating Manual (preventive and mitigative EOPs), and the new “Handbook for Mitigative Measures” (SAMG)
- Criteria for the entrance into the PWR SAMG:

Operational Mode	Criteria
RPV closed	Temperature fuel assembly outlet > 650 °C **or** Dose rate containment > 30 Gy/h
RPV opened	Temperature reactor circuit > 95 °C **or** Water level RPV < Mid-loop level for at least 30 minutes
Spent Fuel Pool	Water level below 5.2 m **or** Water temperature inside SFP > 120 °C

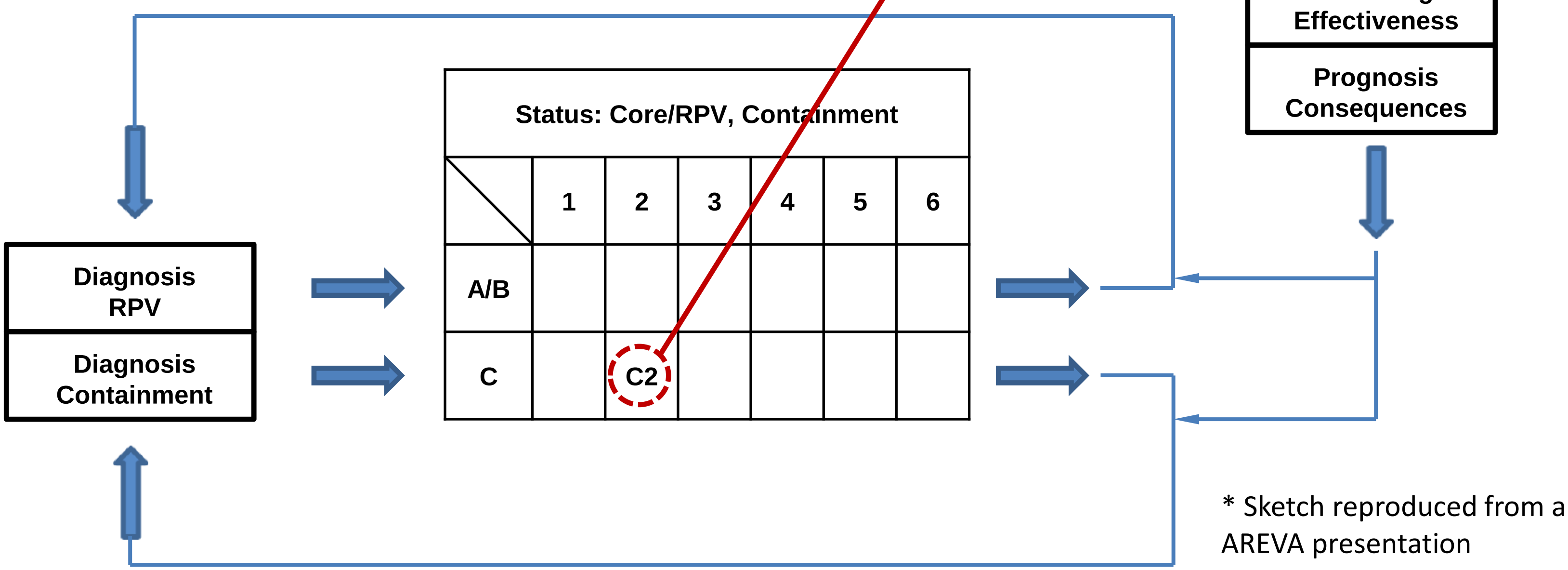
Severe Accident Analyses for Assessment of SAMG Procedures – SAM Concept German PWRs (2)

Status Core/RPV (periodic checking):

A/B: Core damaged, RPV intact
C: Core damaged, RPV failed
S1: Shutdown mode; most of decay heat inside RPV
S2: Failure Cooling SFP; most of decay heat inside SFP

Status Containment (periodic checking):

1. Cont. intact
2. Integrity Cont. endangered
3. Bypass to secondary side
4. Bypass to RB annulus
5. Bypass to auxiliary building
6. Leak at containment



* Sketch reproduced from an AREVA presentation

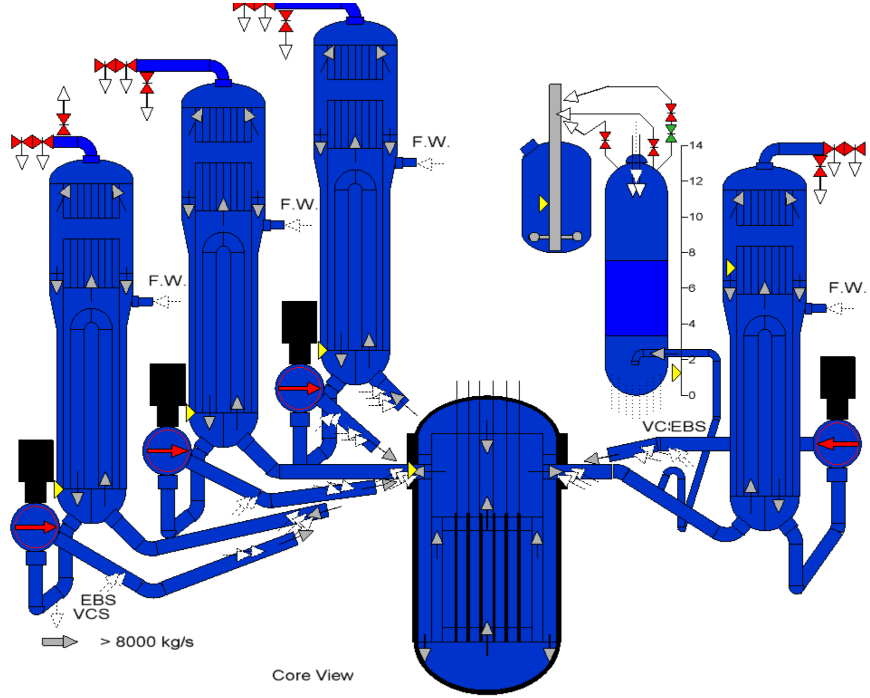
Severe Accident Analyses for Assessment of SAMG Procedures – Severe Accident Analyses PWR (1); General Information

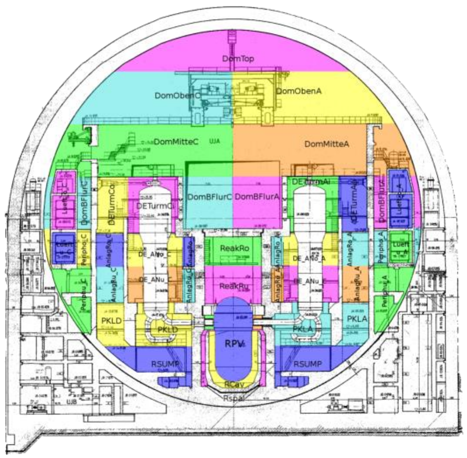
- A project on behalf of the Federal Ministry BMUB is running at GRS regarding the assessment of the improvement of existing SAM and the new SAMG for PWR by deterministic analyses using MELCOR.
  - Analyses of two events “Station Blackout (SBO)” and “Small break LOCA with multiple failures” (significant contribution to core damage states or release categories of PSA Level 2). Postulated boundary conditions:
    - SBO: Secondary side Bleed & Feed available, passive injection of feedwater from feedwater tank, mobile pump available, primary Bleed & Feed not necessary,
    - SB LOCA: 20 cm^2 leak at hot leg, feedwater system and emergency feedwater system failed, failure of switching to sump suction mode, failure of HP sump suction.
  - Calculation of the SBO event with both the status of the EOPs up to Fukushima (base cases) and the improved EOPs (e.g. increased capacity of batteries, mobile generators, etc.), comparable assessment of the analyses in order to show the benefit.
  - Severe accident analyses of the SB LOCA under consideration of selected procedures of the SAMG concept developed by AREVA and implemented in the PWR plants.

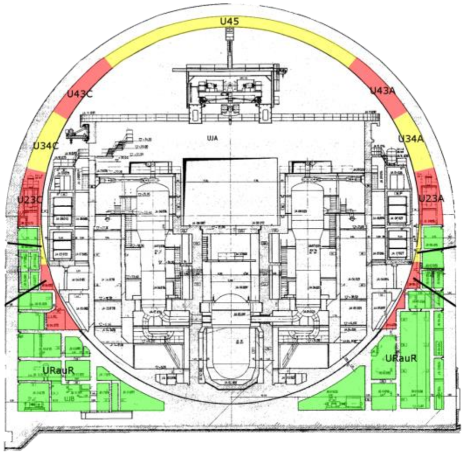
⇨ Quantification and assessment of the benefit due to the improvement of SAM strategy of PWR.

Severe Accident Analyses for Assessment of SAMG Procedures – Severe Accident Analyses PWR (2); Plant Nodalisation

- 2-Loop-Modelling (represents one single and one triple Loop)
- RPV: 6 CVs, 17 FLs, 40 HSs, 5 radial rings, 15 axial meshes
- Each Loop: 6 CVs and heat structures
- Surge Line: 1 CV and heat structures
- Pressurizer: 3 CVs, 3 HSs, 1 PORV and 2 safety valves
- Relief tank: 1 CV, heat structures and bust disks







- Detailed modelling of the containment and reactor building annulus.
- 58 PARs distributed on 37 control volumes.
- Depletion of hydrogen and carbon monoxide is calculated by the PAR model.

Severe Accident Analyses for Assessment of SAMG Procedures – Severe Accident Analyses PWR (3); Preliminary Results Base Cases

SBO

Event	Time [hh:mm:ss]
Station Blackout	00:00:00
Speed MCP < 93%	00:00:04
SCRAM	00:00:04
1st Opening PRZ RV	00:00:06
Pressurization FW Tank	00:30:16
Start Mobile Pump	00:34:57
4 of 4 SG Water Level < 4,0 m	00:50:31
Water Level SG1 < 0.1 m	01:18:55
Water Level SG3 < 0.1 m	01:22:12
Initiation of Secondary Side Bleed	**01:26:13**
p_{HL} < 111 bar	01:31:52
PRZ Water Level < 2.28 m	01:32:10
ECCS Signal	01:32:10
Begin Acc. Injection HL and CL	01:41:52
p_{HL} < 10 bar	02:07:32
Begin Injection Mobile Pump	**01:27:00**
Begin Injection FW Tank	**01:27:40**
End of Injection FW Tank	**08:46:40**
End of Injection Mobile Pump	**17:34:22**
Start Core Uncovery	42:47:14
Start Cladding Failure	43:51:31
Full Uncovery of Active Core Region	44:13:13
Start Failure Lower Support Plate	45:13:16
End of Acc. Injection HL	45:16:34
RPV Failure	50:23:04
Start of MCCI	50:23:04
Start 1st Filtered Depressurization Containment	92:52:44
End of 1st filtered Depressurization Containment	114:24:36
Start 2nd filtered Depressurization Containment	187:02:36
End of 2nd filtered Depressurization Containment	200:25:18
End of Simulation	277:46:40

SB LOCA

Event	Time [hh:mm:ss]
"20 cm^2" Leak	00:00:00
$\Delta p_{SB/Atm}$ > 30 mbar	00:00:23
p_{HL} < 132 bar	00:00:50
"100 K/h" Cooldown	00:01:12
PRZ Waterlevel < 2,28 m	00:01:23
ECCS Signal	00:01:23
HP Pump on (1x)	**00:01:53**
HP Pump on (3x)	**00:01:53**
p_{HL} < 111 bar	00:02:03
Emerg. Ventilation Syst. Annulus on	00:06:23
Isolation of Acc. Cold Legs	00:09:43
3 of 4 SG < 4 m	01:20:44
FT1 u. FT3 empty	03:14:32
Switching to Sump Suction Failed	**03:14:32**
Start Injection Acc. (HL)	**03:28:20**
End of Injection Acc. (HL)	**04:08:18**
Water Level RPV < MIN3	04:52:20
Start of Core Uncovery	06:24:28
T_{Core} > 400°C	06:45:09
Start Cladding Failure	07:01:32
Full Uncovery of Active Core Region	07:43:39
Failure Lower Support Plate	08:51:26
Uncovery Lower Plenum of RPV	08:54:33
RPV Failure	**22:37:15**
Start of MCCI	22:37:15
End of Simulation	27:46:40

Severe Accident Analyses for Assessment of SAMG Procedures – Severe Accident Analyses PWR (4); Selected SAMG Measures

- **SBO**:
  - Connecting of two mobile diesel generators 10 hours after SBO initiation:
    - **Mobile EDG1**: Recovery of electrical supply for instrumentation and extra borating system ⇨ Injection of 4 x 2 kg/s available.
    - **Mobile EDG2**: Recovery of electrical supply for one bunkered train of the ECCS system ⇨ RHR of reactor circuit and SFP (≈ 20 MW, alternating operation) available.
- **SB-LOCA**:
  - Different plant states under examination (A/B1, A/B6, C1, C2, and C6). Measures under consideration:
    - **A/B1** and **A/B6**: Injection into RPV to avoid RPV failure (by volume control system, accumulators and/or from SFP), Maximizing heat removal from reactor building by ventilation system, filtered containment venting, and recovery of 1 of 4 redundancy of the RHR system for long-term cooling.
    - **C1 – C3**: Injection into RPV to terminate MCCI inside the reactor cavity (by volume control system, accumulators and/or from SFP), Maximizing heat removal from reactor building by ventilation system, filtered containment venting (order of measures is dependent on the plant state).

Conclusions

- Recent and current activities at GRS with application of the MELCOR code has been presented.
- Analyses for severe accidents inside spent fuel pools of German PWR and BWR have been performed using the MELCOR 1.8.6 code in the frame of a R&D project. Four different loadings of the SFPs have been considered. The lessons learned are:
  - The postulated sequences are very unlikely events.
  - Leaks in the lower part of the SFP are considered as practically eliminated for German NPPs.
  - In case of transients plenty of time is available for preventive SA measures.
  - Oxidation under air atmosphere are less important for severe accidents starting from loss of SFP cooling.
  - For German PWR it is very likely that SA sequences inside SFP run into filtered containment venting.
  - For the BWR all scenarios will progress into an unfiltered release of radionuclides into environment.
  - A first approach of a Level 2 event tree has been developed for the SFPs of PWR and BWR.
- An assessment of the improvement of the SAM concept for German PWR by MELCOR severe accident analyses is currently underway at GRS.
- The status of the project has been presented and first results for the base cases have been discussed.

Thank you for your attention!
Questions?

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