European MELCOR User Group Meeting 2012 (EMUG 2012)

Cologne , Germany, April 16 - 17, 2017

MELCOR Modeling and Simulation of Fukushima Daiichi Unit 3 Severe Accident

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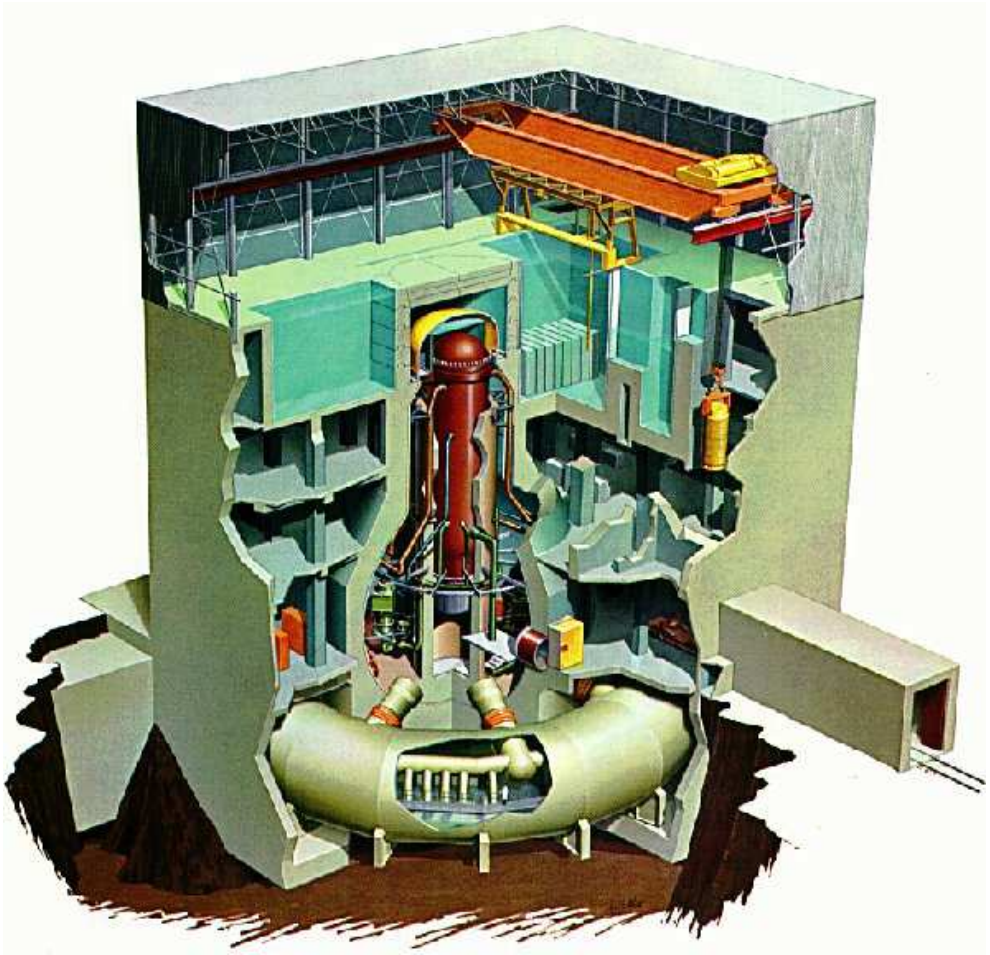
Outline

- Plant data of Fukushima-Daiichi-3 BWR
- Summary of main events during the accident
- Modeling of reactor system and containment
- Initial conditions in the simulation
- Preliminary results

Plant data of Fukushma-Daiichi-3

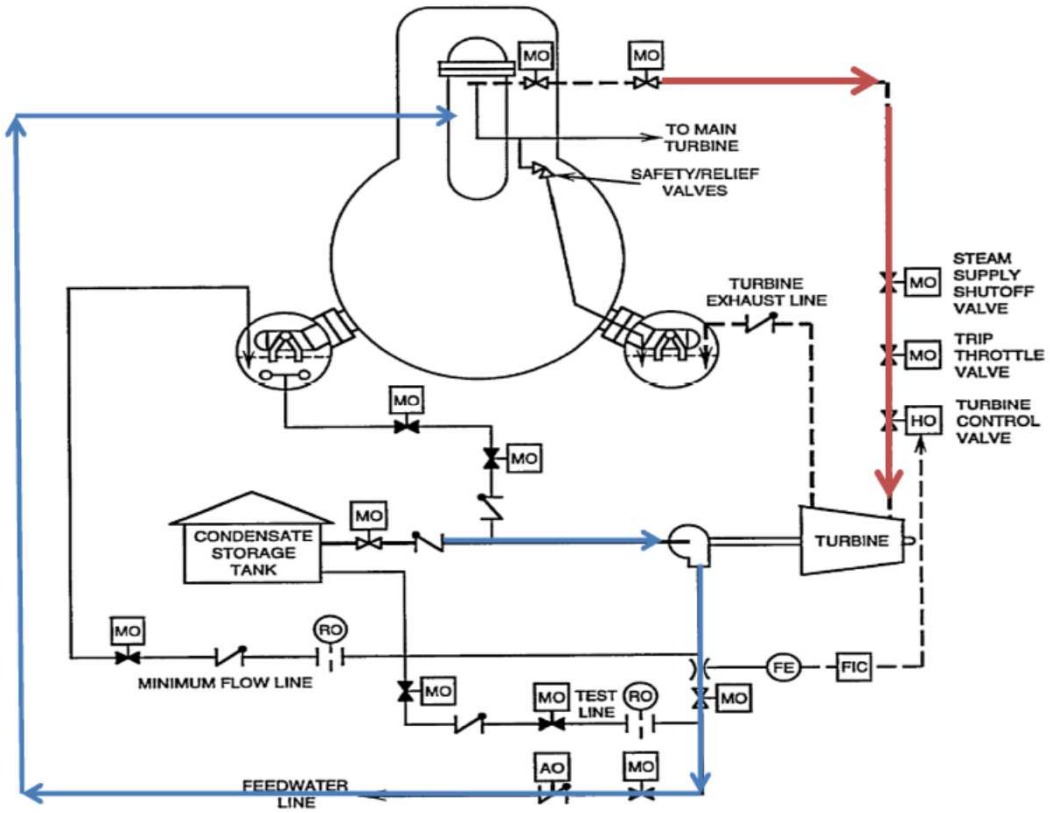
- Fukushima Daiichi Unit 3 (1F-3) is a GE type-4 BWR with MARK-I containment.
- 1F-3 was rated as the power capacity of 2381 MWth.
- It has four external recirculation loops as well as four steam lines.
- The operating pressure of 1F-3 is 7.03 MPa.
- The core has 548 fuel assemblies and 137 control rods with B_4C as absorber material.
- Each assembly has a fuel box to form a independent fluid channel.
- The reactor vessel is about 21 m high and has an inside diameter of 5.6 m.

Mark-I Containment of 1F-3



RCIC & HPCI of 1F-3

- Units-2 & 3: Passive ECCS (Steam-driven turbine → pump)



RCIC & HPCI

- Reactor Core Isolation Condenser system to cool down the reactor
- High Pressure Coolant Injection system to cool down the reactor
- Need DC power (battery) to control valves
- The storage tank will boil off w/o feed water (AC power needed)

RCIC nominal flow rate : 97 t/h	Stop at **high** level of water (L8) manually
HPCI nominal flow rate: 965 t/h	Start at **low** level of water (L2) automatically

Explosion of 1F-3

- H2 deflagration / detonation

Time (JST)	Unit
14/3 11:01	3

Summary of main events

	Unit 3	
	Status before the earthquake: in operation	
3/11		
	14:47	Reactor scram (high seismic acceleration) Control rods fully inserted (sub-critical) Turbine trip Loss of the external power supply
	14:48	Emergency diesel generator (emergency DG) turned on Main steam isolation valve (MSIV) closed
	14:52	Safety relief valve (SR valve) repeatedly opened and closed from this point onwards
	15:05	Reactor core isolation cooling system (RCIC) manually turned on
	15:25	RCIC trip (L-8)
	15:38	All AC power supply lost
	15:42	TEPCO judged that an event falling under Article 10 of the NEPA (loss of all AC power supplies) had occurred.
	16:03	RCIC manually turned on
	20:30	RCIC in operation Lighting in Central Operating Room (temporarily secured and in preparation)
	23:35	Water level on the decrease (400 mm at 22:58→350 mm (wide range))
3/12		
	11:36	RCIC trip
	12:35	High pressure coolant injection system (HPCI) turned on (L2)
	12:45	Reactor pressure on the decrease (7.53 Mpa at 12:10→ 5.6 MPa)
	20:15	Reactor pressure on the decrease (0.8 MPa)
3/13		
	2:42	HPCI stopped
	4:15	Reactor water level was judged to have reached the top of active fuel (TAF).
	5:10	Due to stoppage of HPCI, injection by RCIC into the reactor was attempted. As RCIC could not be turned on, the event was judged by TEPCO to fall under Article 15 of the NEPA (loss of reactor cooling function).
	6:00	Water level in the reactor: -3500 mm (wide range)
	7:39	Spraying onto the PCV began. Water level as of 7:45: TAF -3,000 mm. Reactor pressure: 7.31 MPa. DW pressure: 460 kPa. SC pressure: 440 kKPa.
	8:41	The second valve (AO valve) was set to "open" for venting.
	9:08	Operation to reduce pressure in the RPV by relief valve (SRV) It appears that some time after this point the safety relief valve (SRV) was closed and opened, due to issues with maintenance of air pressure for driving SRV and excitation on the electro-magnetic valve on the air supply line.
	About 9:20	Decrease trend of pressure inside PCV detected
	9:25	Injection of fresh water (borated) into the reactor through the Fire Extinguishing Line began.
	11:17	Vent line AO valve found closed (through loss of pressure in the tank) From this point on, it was difficult to keep the AOV open due to issues with maintenance of air pressure for driving AOV and excitation on the electro-magnetic valve on the air supply line, and the operation to open it was repeated multiple times.
	12:30	Operation to open the AO valve on the pressure chamber side.
	13:12	Fresh water injection to the reactor was switched to seawater injection.
	22:15	Diesel-driven fire pump (D/DFP) stopped (before it ran out of fuel)

Assumption of initial state

- The plant is initially running at full power.
- At the time of accident occurrence, the whole core is irradiated for 1.6 years averagely with availability of 75%.
- At full load, steam and feed water flow rate is set at1200 kg/s.
- The bypass flow rate in the core is at a fraction of 7.5% of total flow through the core.
- The main recirculation pump provide a pressure head of 55 m, and feed water temperature is set at 184 ^{o}C.
- The initial water level in the reactor vessel is kept at 13.125 m.

Summary of nodalization

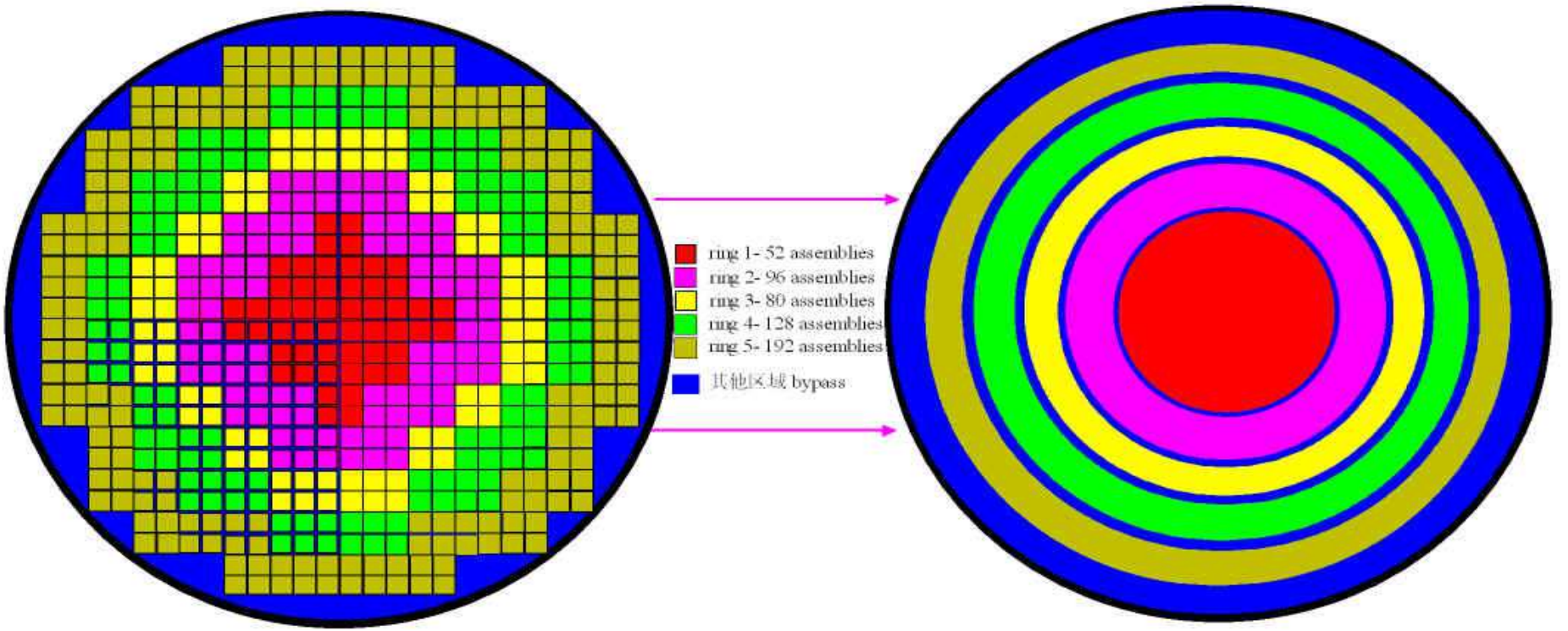
- The MELCOR model for 1F-3 comprises
  - 38 control volumes;
  - 69 flow paths;
  - 100 heat structures, among which 75 represent the core and lower plenum region.

Main parameter of the plant

Parameter/unit	value	Parameter/unit	value
Core thermal power (MW)	2381	Fuel rod pitch(mm)	15.8*
Reactor operation pressure(MPa)	7.03	Average irradiation time (years & availability)	1.6/0.75*
Nominal steam flow rate (kg/s)	1200*	UO2 inventory (ton)	94
Core flow area in moderator tank (m^2)	10.16*	Total Zircalo y mass(incl. cladding、fuel box、grid. Spacer, etc.) (ton)	37.8*
Core bypass flow area (m^2)	4.84*	Control poison mass(kg)	1665*
Feedwater temperature (^oC)	184*	Total mass of stainless steel in core (ton)	48.7*
Core inlet temperature (^oC)	280.1*	RPV internal height (m)	21.1
Core outlet temperature (^oC)	286.5*	RPV internal diameter (m)	5.6
Fraction of bypass flow to total flow (%)	7.5*	RCIC flow rate (ton/h)	97
Initial RPV water level (m)	13.125	HPCI flow rate (ton/h)	965
Number of fuel assemblies	548	SRV open setpoint (MPa)	7.59/7.66/7.73
Core active length (m)	3.68	SRV full discharge flow rate (ton/h)	2900
Number of control blade	137	dry well volume (m^3)	4240
Fuel rod diameter (mm)	12.25	Suppression chamber atmospheric volume (m^3)	3160
Fuel pellet diameter (mm)	10.58*	Suppression chamber pool volume (m^3)	2980
Cladding thickness (mm)	0.73*	Reactor building free volume (m^3)	38000*

*** Values estimated**

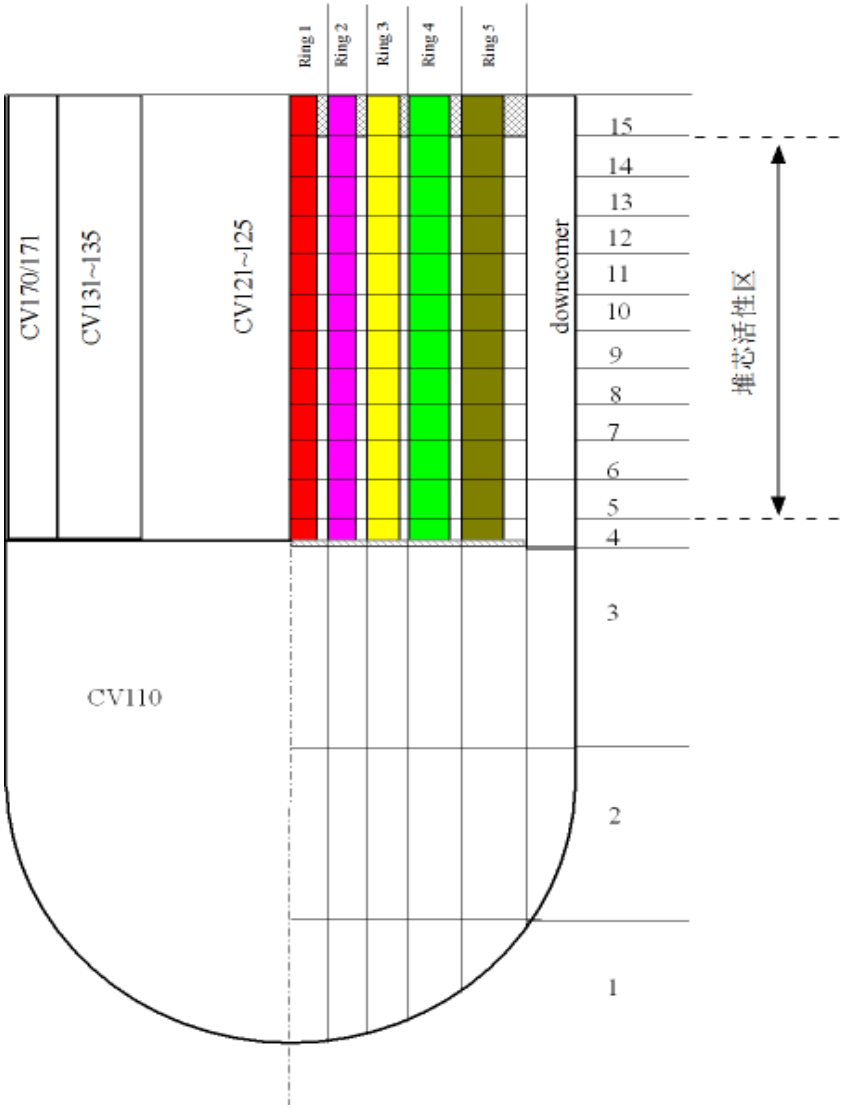
Modeling of the plant



Ring No.	1	2	3	4	5	total
No of fuel assemblies	52	96	80	128	192	548
Flow area in fuel channels,m^2	0. 505	0. 932	0. 777	1. 243	1. 904	5.323
Flow area in bypass,m^2	0.229	0.424	0.354	0.566	3.268	4.841
Control rod number	13	24	20	32	48	137

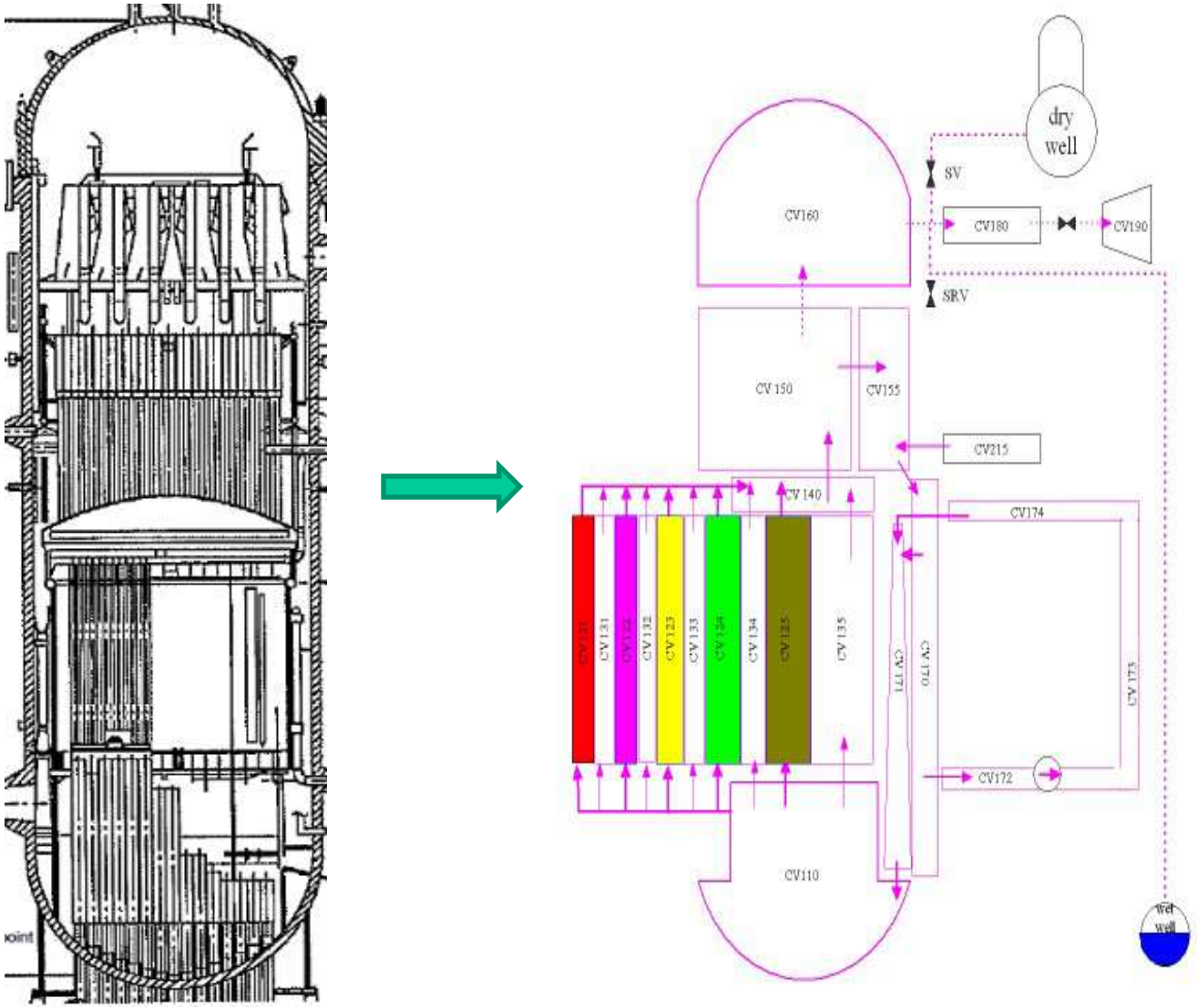
Radial nodalization of reactor core

Modeling of the plant



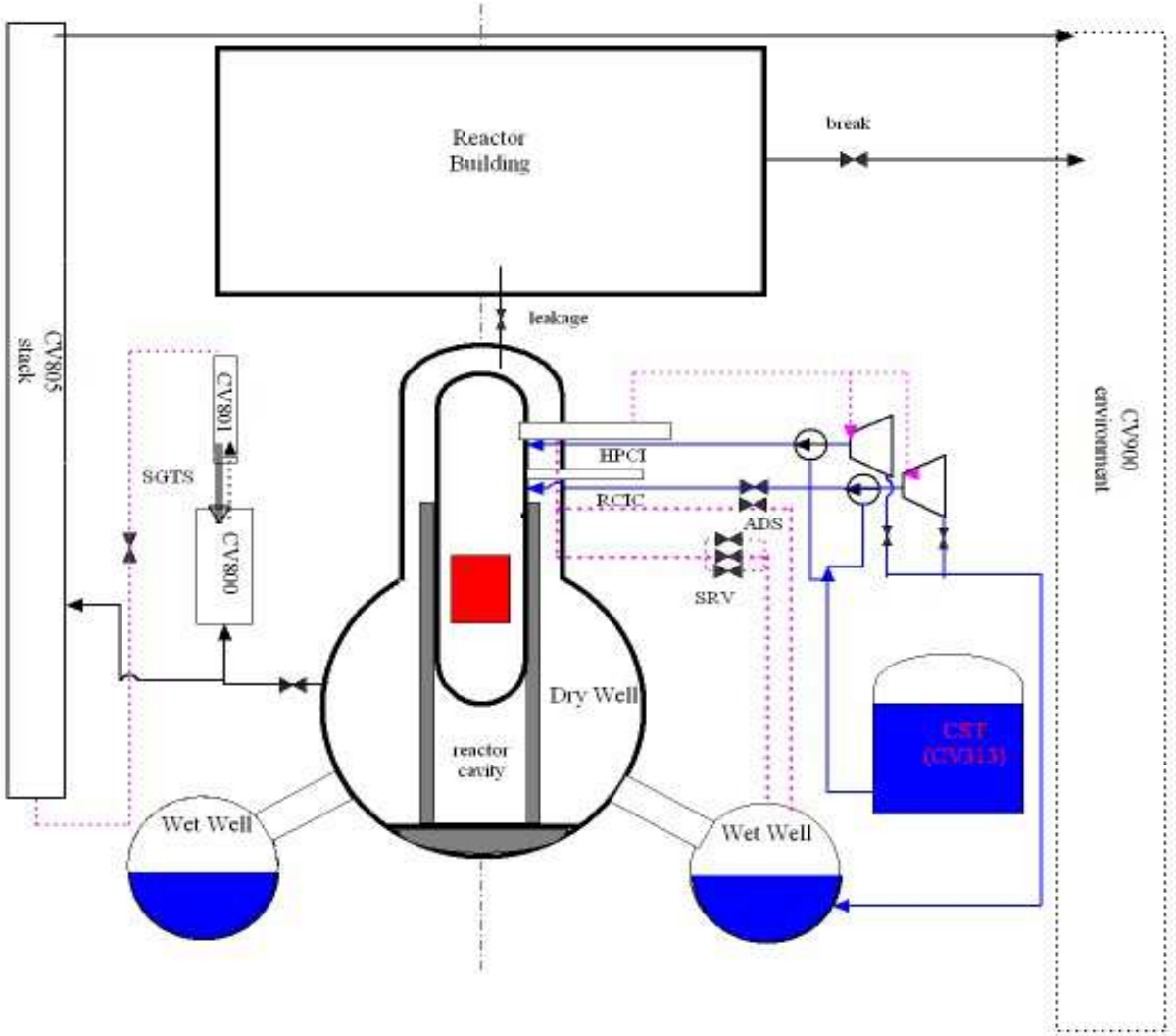
Axial nodalization of reactor core and lower plenum

Modeling of the plant



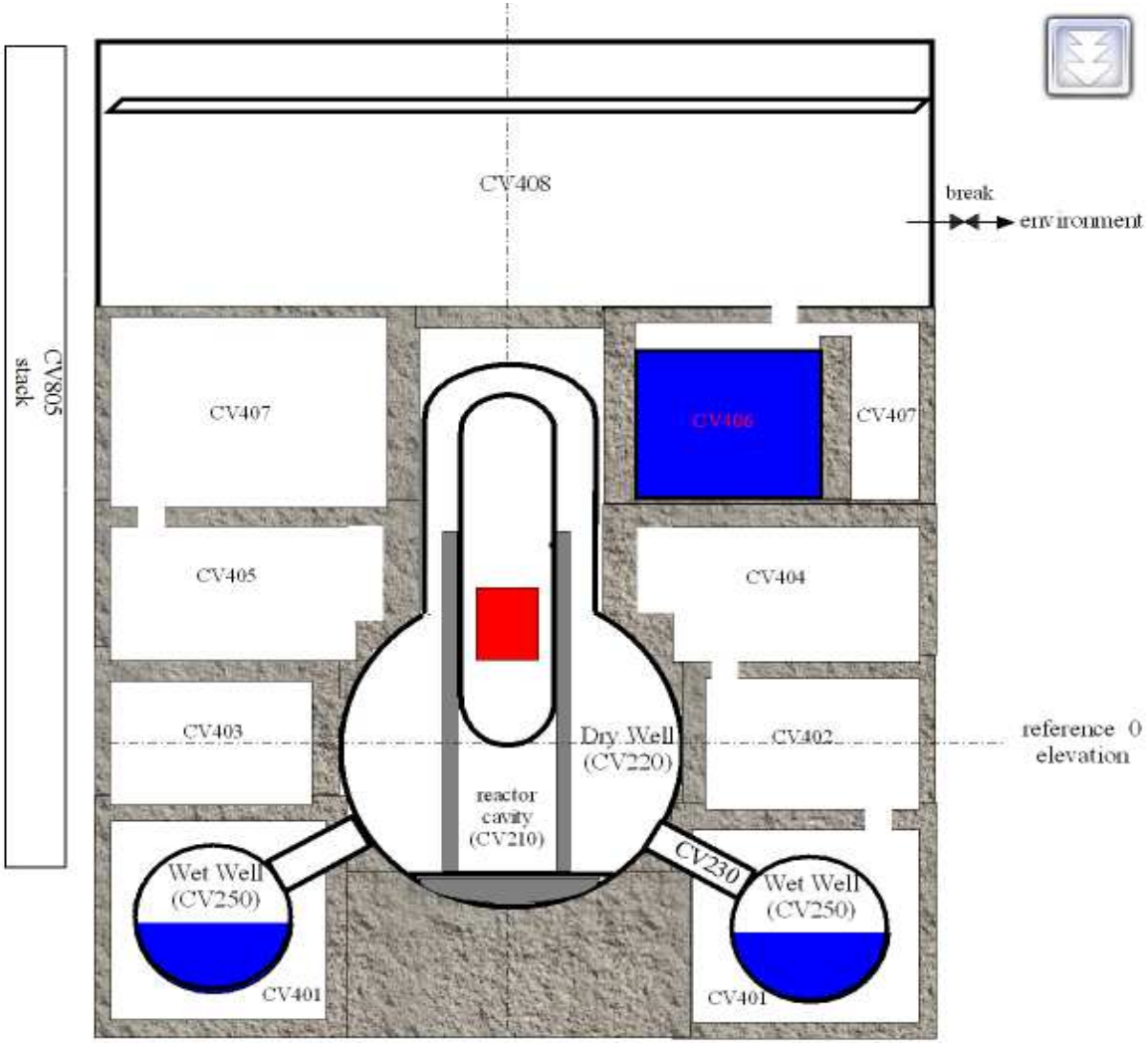
Nodalization of reactor vessel and primary circuit

Modeling of the plant



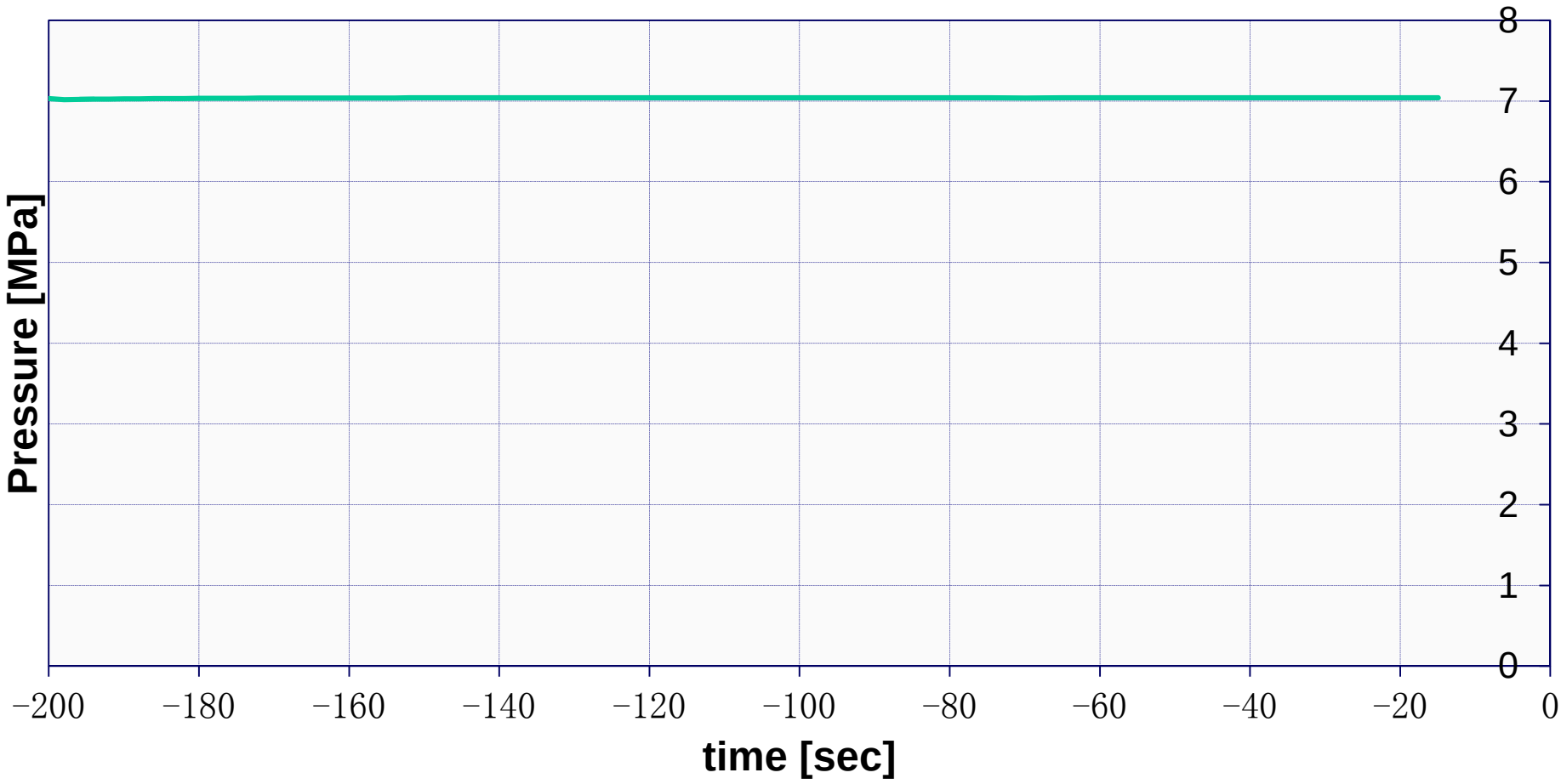
Modeling of safety systems available for action

Modeling of the plant



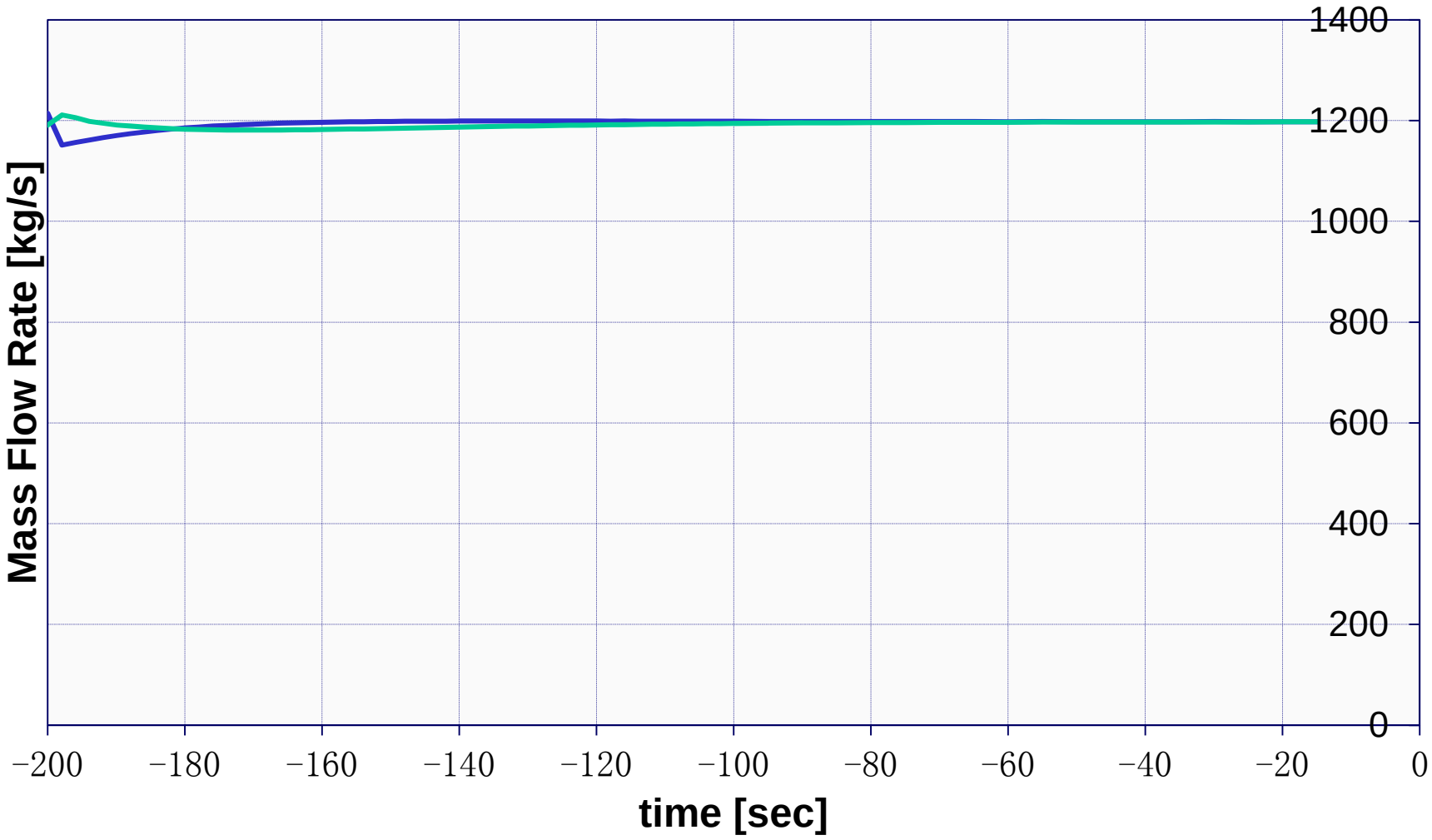
Primary & secondary containment nodalization

Results of steady-state run



Pressure in the reactor vessel dome

Results of steady-state run



Feed water and steam flow rate

SBO scenario for simulation

➢ Assumptions for 1F-3 station blackout scenario

- Reactor scram completed within 10 s, and containment isolated in 2 minutes.
- There is no break along the reactor pressure boundary after earthquake or tsunami.
- RCIC was active to inject cooling water and take steam to drive turbine and pump during 19~39 minutes & 1.3~20.8 hr since earthquake
- HPCI was active to inject cooling water and take steam to drive turbine and pump during 21.8~35.6 hr since earthquake
- The source water of RCIC and HPCI is from the CST, the exhaust vapor was discharged into the wet well
- The leakage ratio of containment is set at its design level, when pressure in the containment exceeded its design pressure (0.48 MPa), the leakage ratio will increase 10 times
- Operator start to depressurize the reactor by opening one set of SRV at 42.4 hr
- Fire pump kept injecting water into vessel between 42.6~55.5 hr and 63.2~68.2 hr
- Containment venting line opened at between 42.6~44.5 hr, 45.7~47.4 hr, 53.4~57.4 hr and 62.6~68.2 hr.
- H2 formation from radiolysis in core or in spent fuel pool is not considered .

Sensitivity study of SBO scenario

- ***Impact of fire pump flowrate on the mitigation of core degradation***
  - **Case1**: flow rate kept constant at 15 t/h during 42.6~55.5 hr and 63.2~68.2 hr.
  - **Case2**: flow rate kept constant at 10 t/h during 42.6~55.5 hr and 63.2~68.2 hr.
  - **Case3**: flow rate kept constant at 5 t/h during 42.6~55.5 hr and 63.2~68.2 hr.

Results of simulations

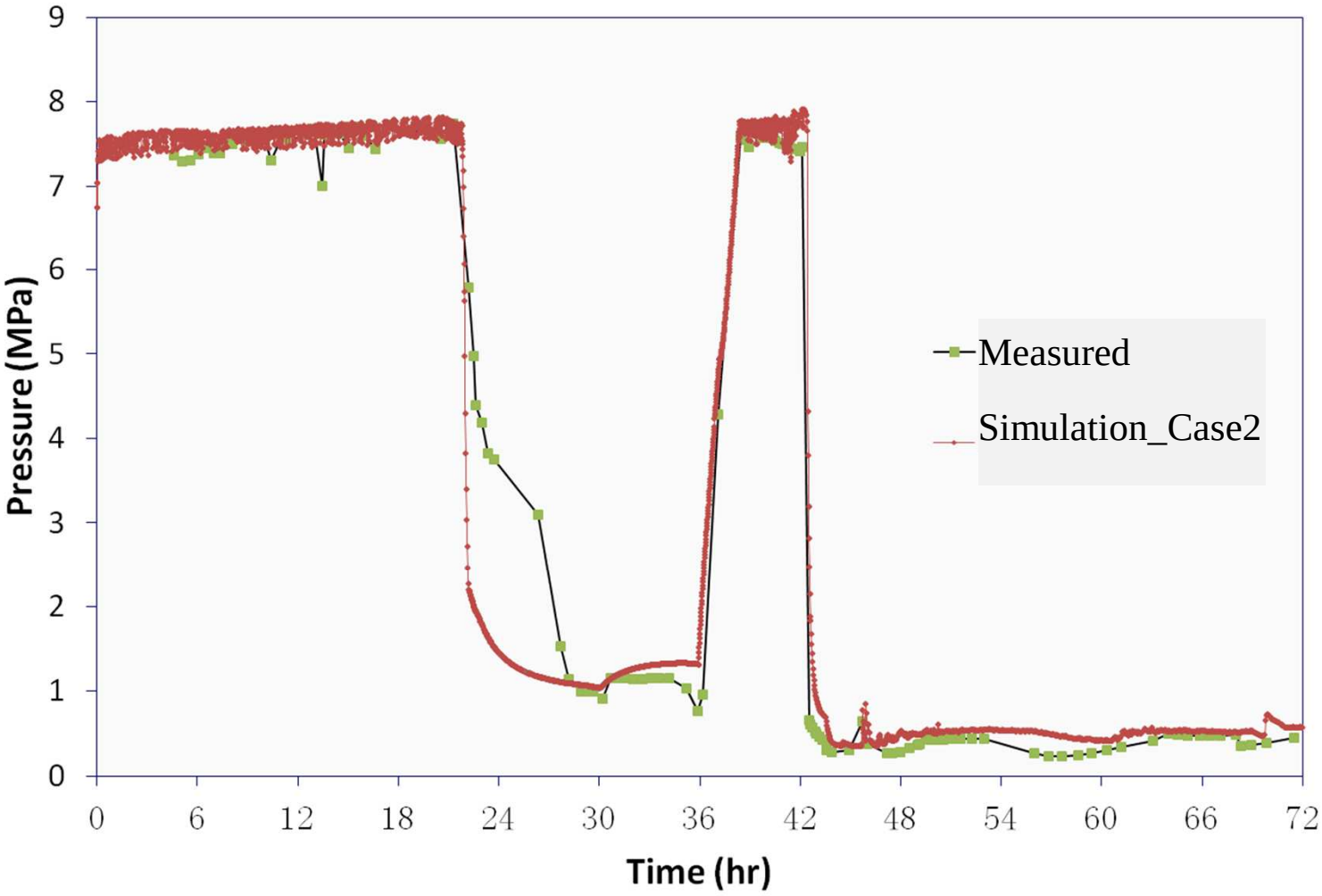
Value / Parameter (unit)	Case1	Case2	Case3
Oxidization fraction of zircaloy (%)	38	50	71
Fraction of degraded fuel slumped into lower head (%)	0	18	100
Failure time of core support plate	N/A	58.7	58.1
Time at which the average h2 concentration in upper reactor building reaches deflagration limit (hr)	N/A	60.4	51.3
Failure time of vessel (hr)	N/A	N/A	71
Total mass of water injected by fire pump (ton)	307	205	103

Results of simulations

LOCAL TIME		Real Time table (s)	Simulated Time table (s)	SIMULATION EVENT
2011年 3月11日	14:46	0.0	0	EARTHQUAKE
	14:47	60	10	REACTOR SCRAM
	14:48	120	120	Main steam isolation valve closed
	NA	NA	150	RHRS started
	14:52	360	280	Safety valve started to repeat of open and close
	15:05	1140	1140	RCIC manually turned on
			2250	Water level reach L8
	15:25	2340	2340	RCIC TRIP(L8)
	15:38	3120	3120	DG power supply lost (SBO initiated)
	16:03	4620	4620	RCIC manually turned on
2011年 3月12日	11:36	75000	75000	RCIC trip
	12:10	77040	77040	Reactor pressure at 7.53 MPa
	12:35	78540	78540	HPCI system turned on
2011年 3月13日	2:42	129360	129360	HPCI exhausted
	4:15	134940	130000	Reactor water level reached top of TAF
	NA	NA	145800	Zr-steam reaction initiate
	NA	NA	146300	Rupture of fuel claddings
	6:00	141240	150500	Water level in reactor -3500 mm
	9:08	152700	152700	Depressurize the RPV by open SRV
	9:25	153480	153480	Fir pump start to inject fresh or sea water into vessel
	22:15	199740	199740	Fire pump stopped work
2011年 3月14日	6:00	227640	227640	Fire pump resume work
	11:01	245700	217400(first deflagration)	H2 explosion/fire pump stopped work again
			259200	Calculation end

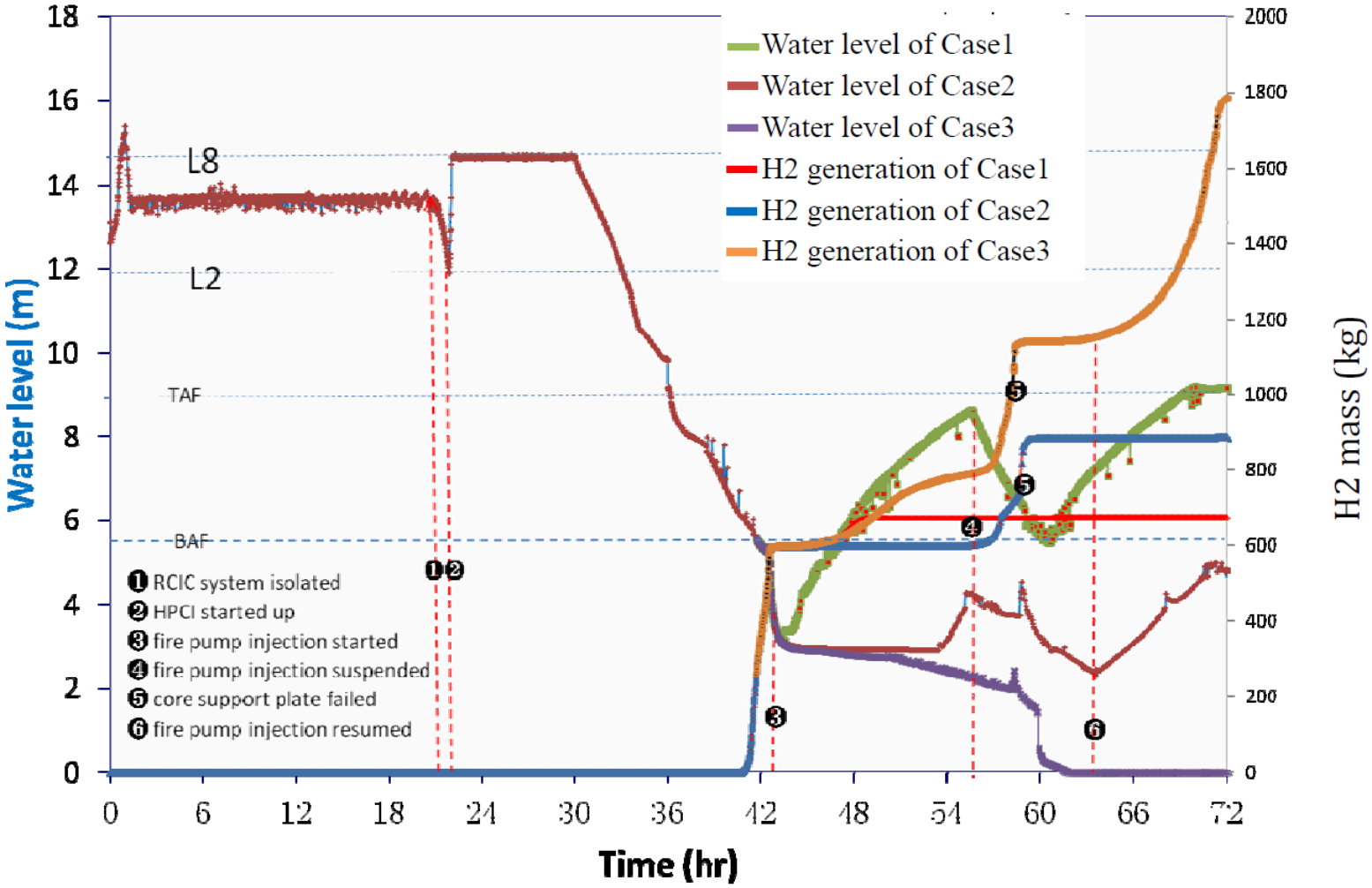
Comparison of simulation with accident progression in 72 hrs (Case2)

Results of simulations



Reactor pressure

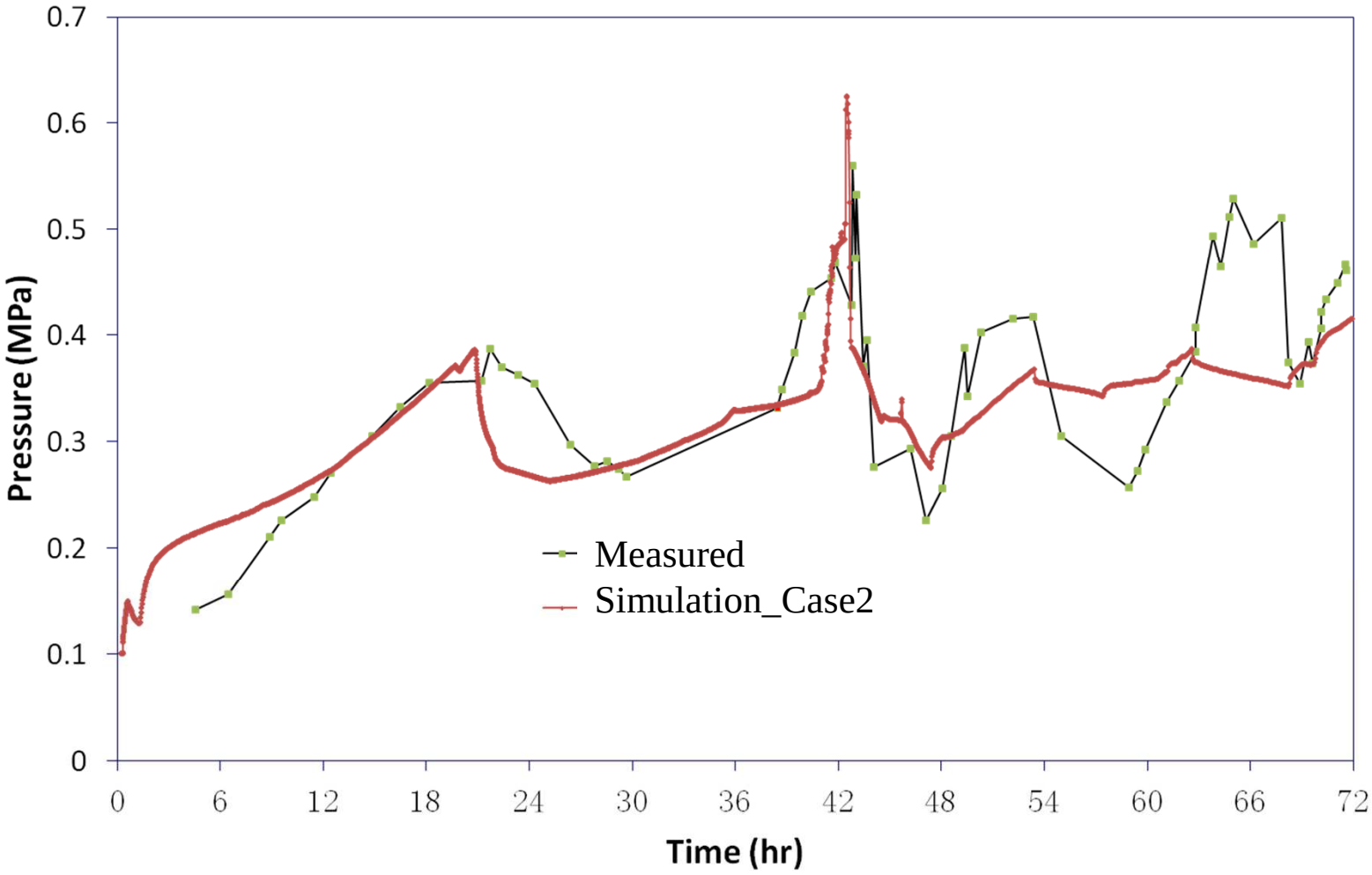
Results of simulations



Fire pump injection: Case1 - 15 t/h, Case2 - 10 t/h, Case3 - 5 t/h (Chen et al. 2012)

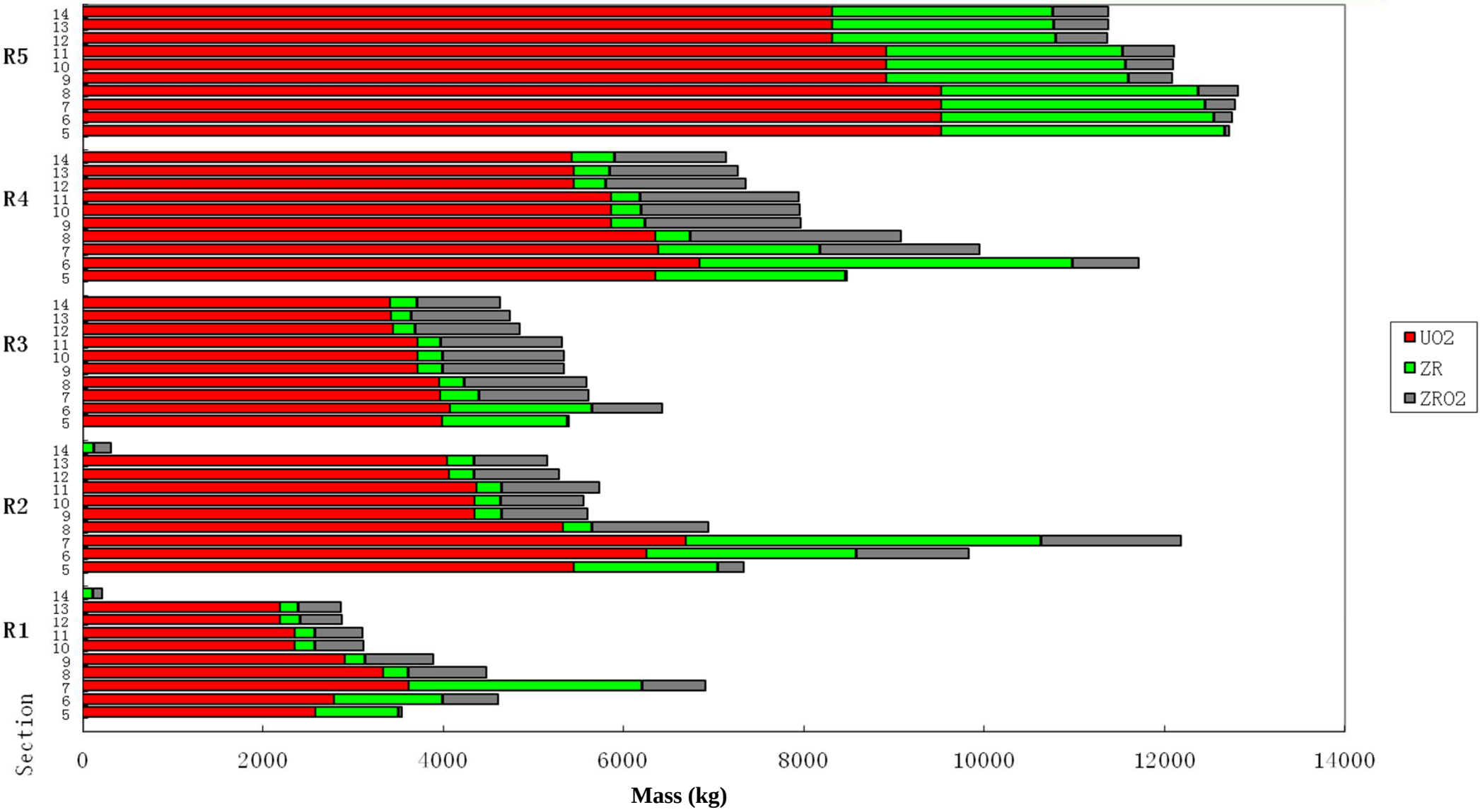
Water level & H2 generation

Results of simulations



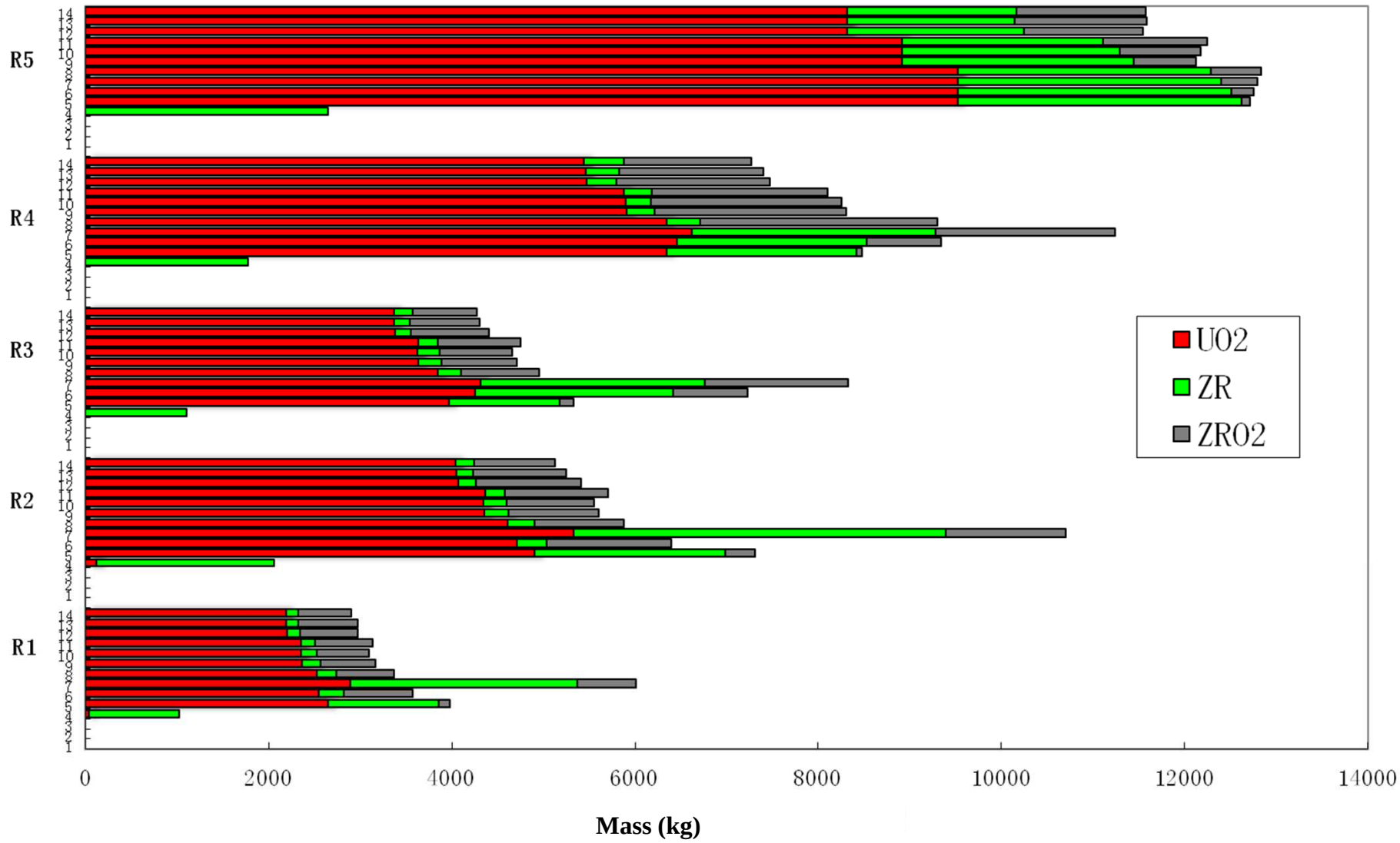
Containment pressure

Results of simulations



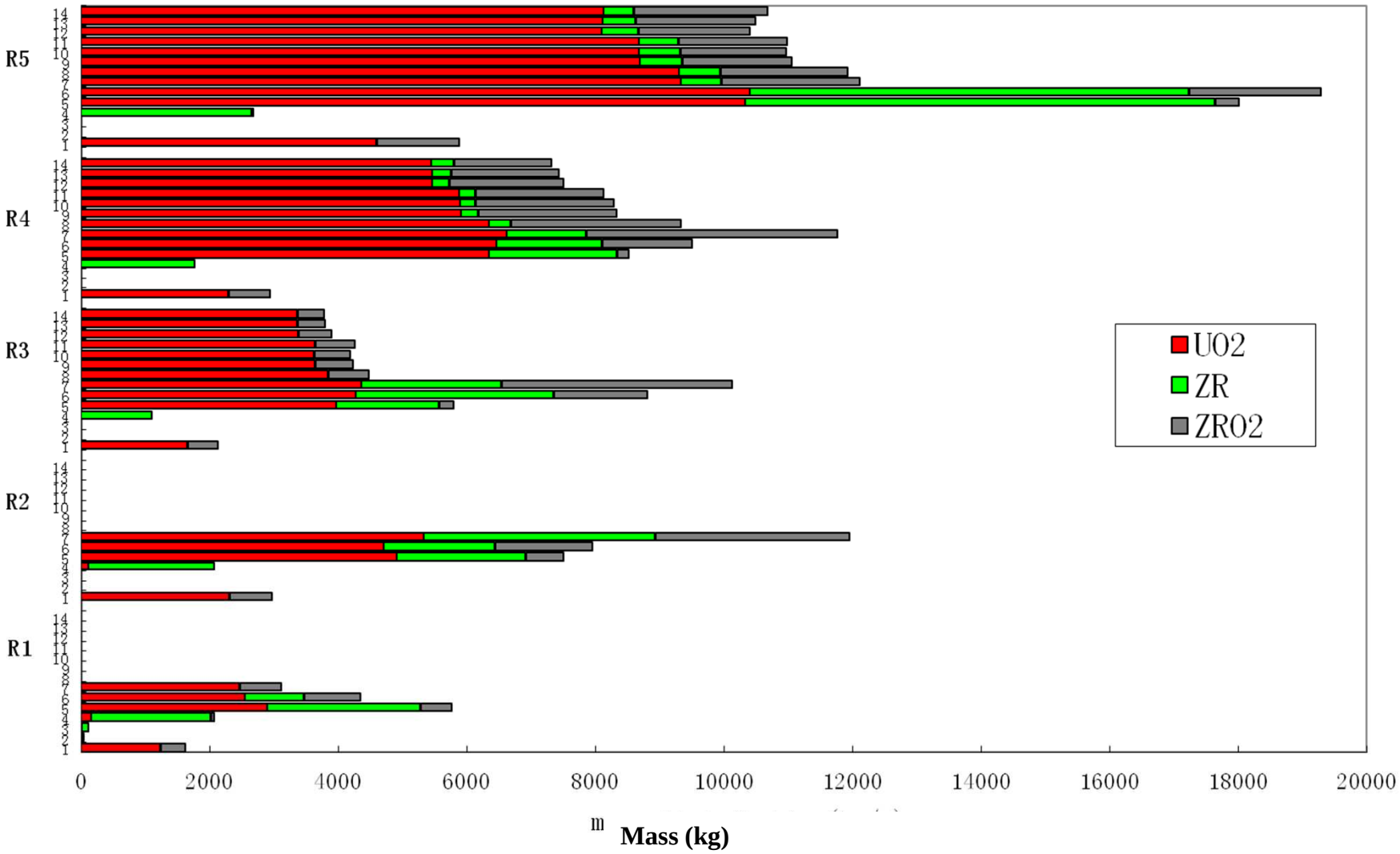
Fuel mass distribution in reactor core at time of fire pump injection

Results of simulations



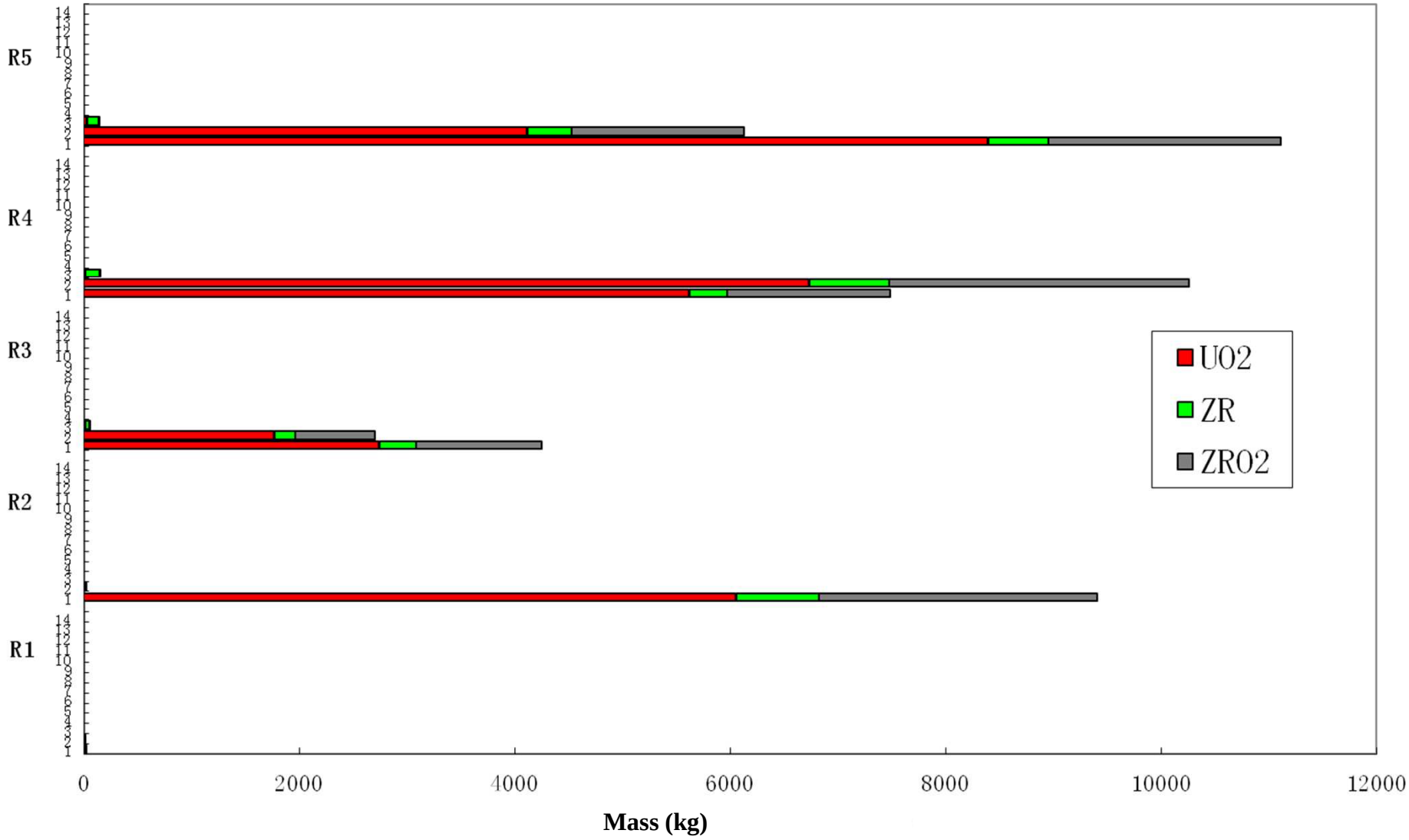
Fuel mass distribution in reactor core at 72 hr (Case1)

Results of simulations



Fuel mass of reactor core at 72 hr (Case2)

Results of simulations



Fuel mass of reactor core at 72 hr (Case3)

Concluding remarks

- A MELCOR input deck is under development to simulate Fukushima-Daiich-3 accident.
- The preliminary results show:
  - The passive safety systems RCIC + HPCI could work for more than 30 hours to efficiently remove the decay heat.
  - By the time of fire pump injection (~42 hours) the core still maintained its initial geometry.
  - Coolability of the entire core was reached if water injection of 15 t/h would be secured, while with 5 t/h water injection, the core was completely degraded, eventually leading to vessel melt-through.
- More verification of plant data will be performed to improve modeling fidelity.