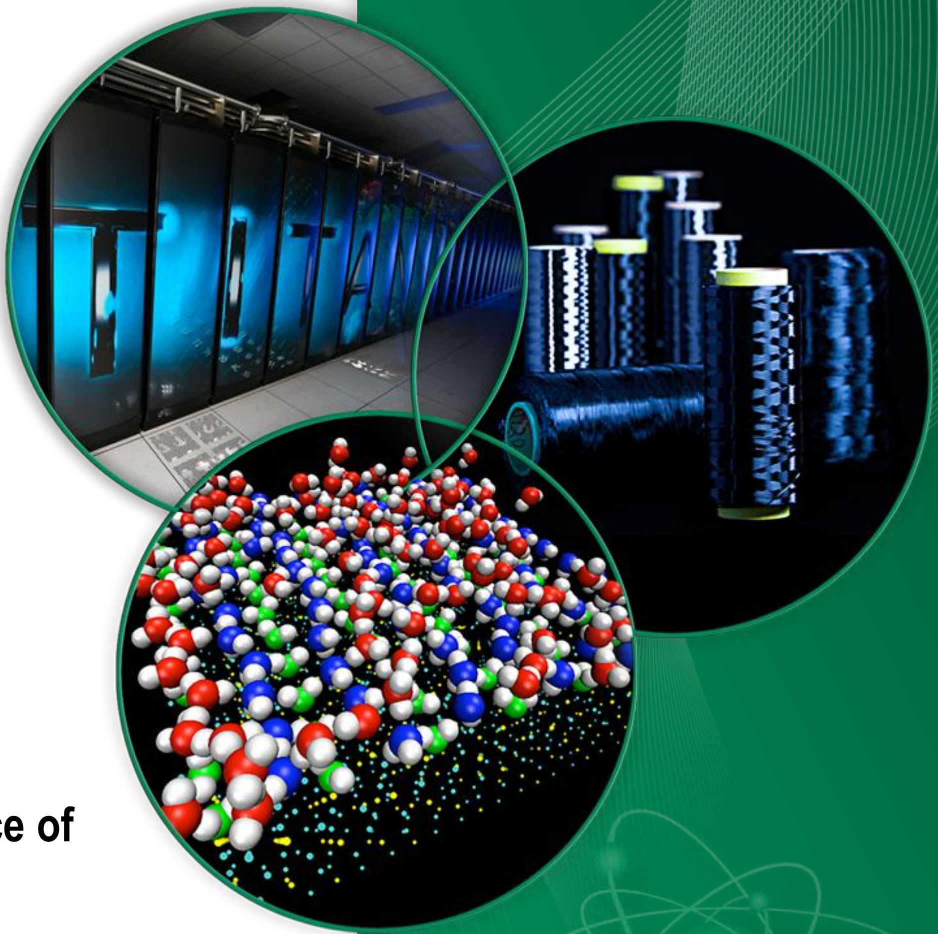
Reactor Types of Interest (Applicability)



Larry Ott
Reactor and Nuclear Systems Division
Oak Ridge National Laboratory

Second Meeting on Increased Accident Tolerance of Fuels for LWRs
NEA Headquarters
Issy-les0Moulineaux, France
29-29 October 2013

Outline

- Background
  - World-Wide
  - USA
- PWRs and BWRs (they are different !)
- LWR accident response

Light Water Reactor (LWR) Nuclear Power Plant (NPP) Development

Gen II

Large commercial NPPs currently in operation throughout U.S. and World-Wide

Gen III

Advanced LWRs

ABWR (GE)
System 80+ (CE)
VVER-428 (RF)
CPR-1000 (China)
APR-1400 (S.Korea)

Gen III+

- Evolutionary Designs
- Improved Economics
- Advanced Safety Features
- Some Passive Design
- Advanced Containment Design
- Simplified System Designs

ESBWR (GE)
AP 1000 (W)
ACR 700 (AECL)
EPR (AREVA)
PBMR (South Africa)
VVER-AES-91&-92 (RF)
VVER-1200 (RF)

LWR NPPs – World-Wide - Operating and Under-Construction (with expected operation within 5 years)

Gen II

Operating : 426 units
Under-Construction : 8 units

Gen III

Operating : 12 units
Under-Construction : 27 units

ABWR (GE)
System 80+ (CE)
VVER-428 (RF)
CPR-1000 (China)
APR-1400 (S.Korea)

Gen III+

ESBWR (GE)
AP 1000 (W)
ACR 700 (AECL)
EPR (AREVA)
PBMR (South Africa)
VVER-AES-91&-92 (RF)
VVER-1200 (RF)

Operating : None
Under-Construction : 26 units

LWR NPPs – USA
Operating and Under-Construction

Gen II

Operating : 99 units
PWRs : 65 units
BWRs : 34 units
Under-Construction : 1 unit

Gen III

Operating : None
Under-Construction : None

ABWR (GE)
System 80+ (CE)
VVER-428 (RF)
CPR-1000 (China)
APR-1400 (S.Korea)

Gen III+

ESBWR (GE)
AP 1000 (W)
ACR 700 (AECL)
EPR (AREVA)
PBMR (South Africa)
VVER-AES-91&-92 (RF)
VVER-1200 (RF)

Operating : None
Under-Construction : 4 units
Vogtle 3 and 4
Summer 2 and 3

LWR NPPs – USA – Operating – all Gen II

PWRs :

B&W		6
CE		12
	2-loop	5
W	3-loop	13
	4-loop	29
Total		65

Containments :

dry-ambient pressure	49
dry-subatmospheric	7
wet-ice condenser	9

BWRs :
(all GE)

Containments -

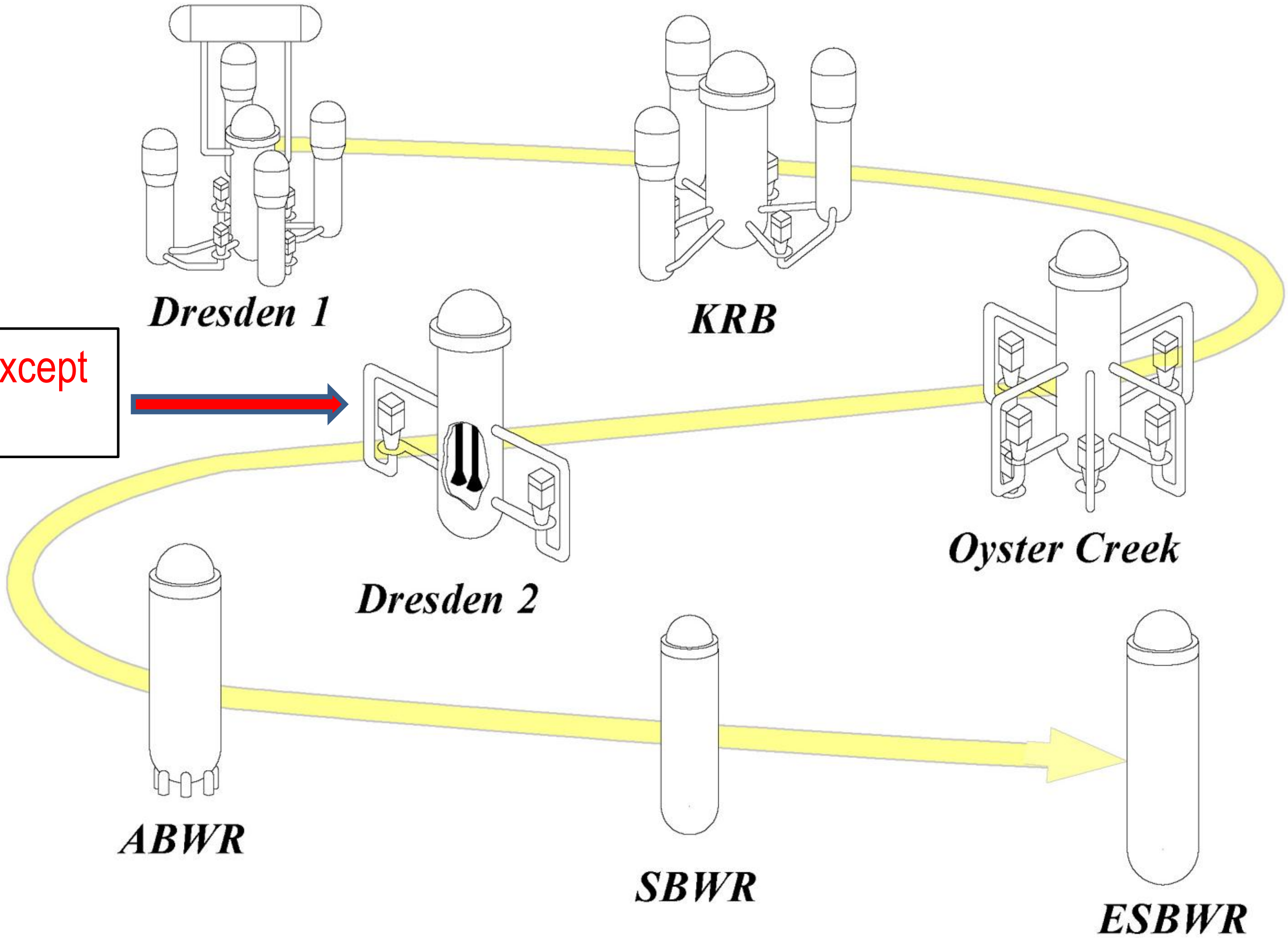
Mark I	22
Mark II	8
Mark III	4
Total	34

Reactor Type :

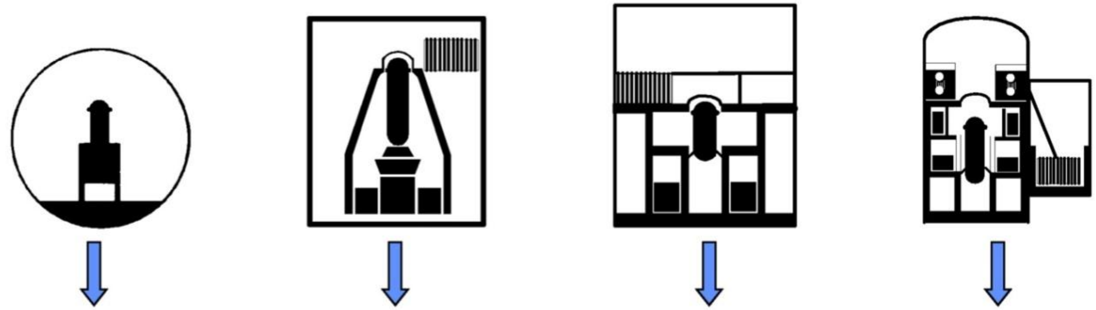
BWR2	2
BWR3	6
BWR4	18
BWR5	4
BWR6	4

BWR Evolution

All U.S. BWRs except Oyster Creek

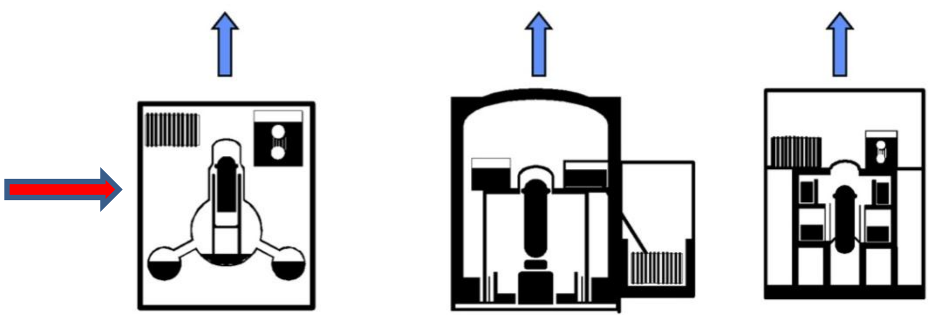


BWR Containment Comparison

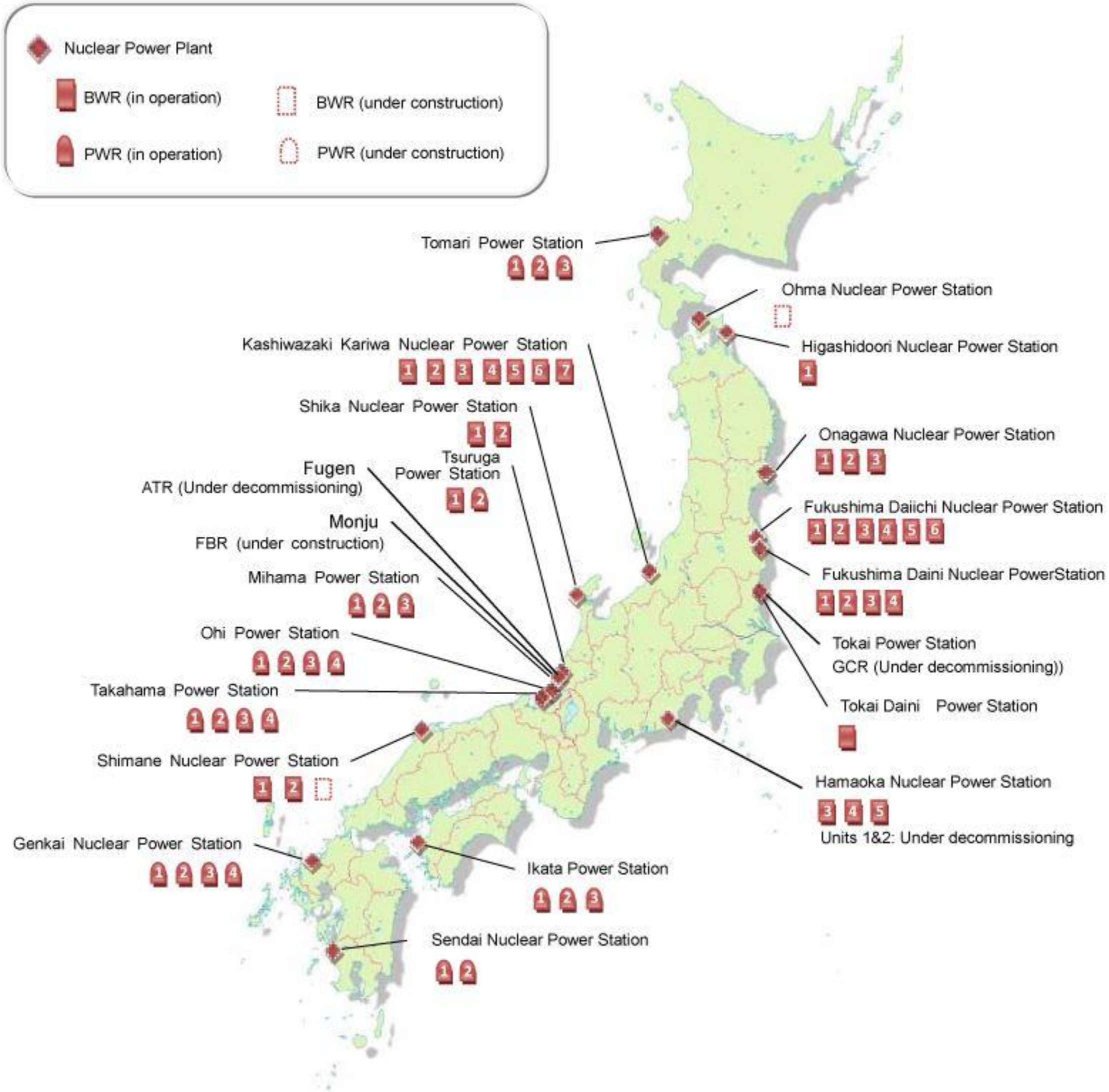


Characteristic	Dry	Mark I	Mark II	Mark III	ABWR	SBWR	ESBWR
Pressure Suppression	No	Yes	Yes	Yes	Yes	Yes	Yes
Drywell and wetwell volume (ft^3 X 10^6)	2.5	0.4	0.5	1.6	0.5	0.3	0.43
Design Pressure (psig)	50	62	45	15	45	55	40
LOCA Pressure (psig)	50	44	42	9	39	42	30

22 U.S. BWRs



NPPs in Japan



PRE-Fukushima Accident :

- PWRs: 24 Units
- BWRs: 30 Units
  - with 2 under const.

Core Damage Frequency for Current-Generation Reactors (U.S.)

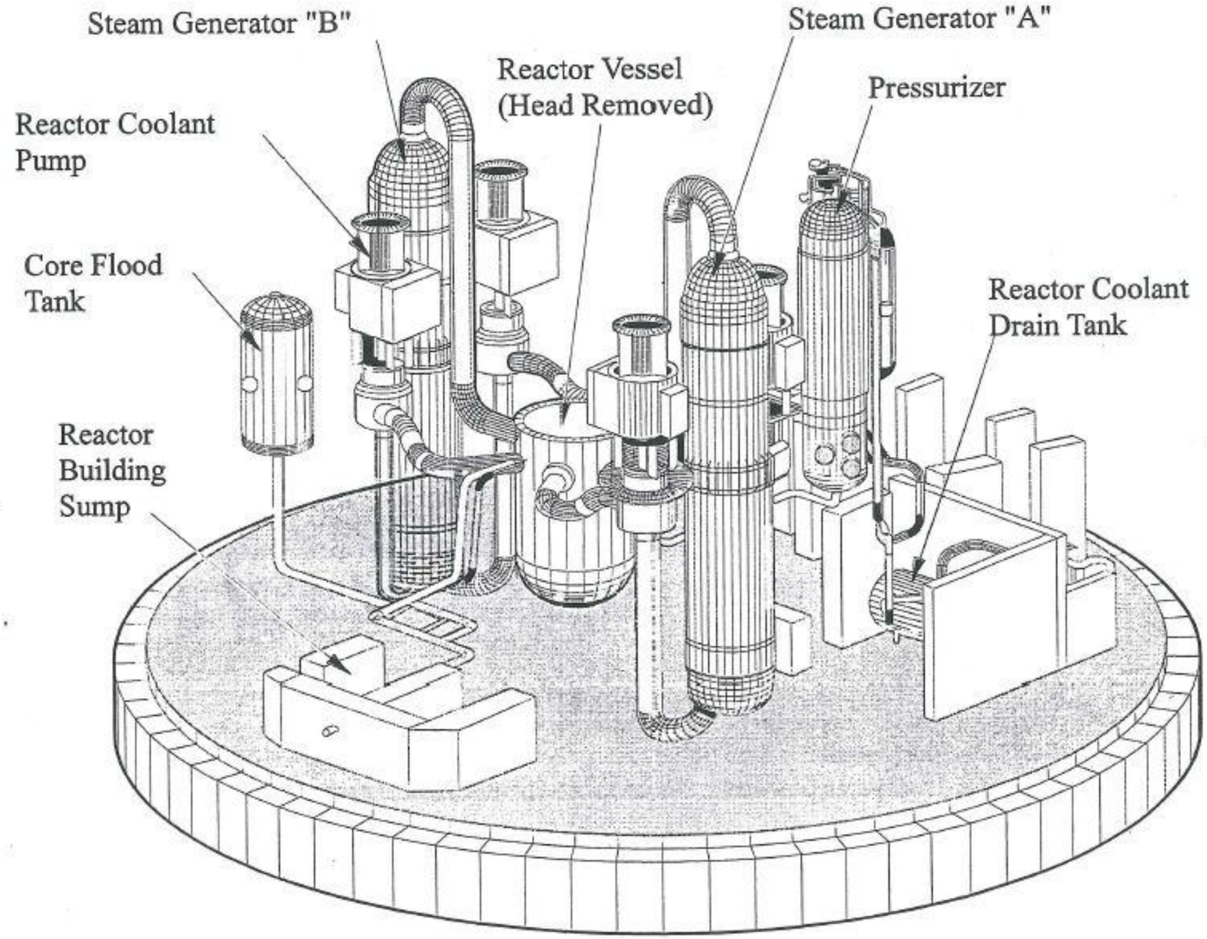
• Grand Gulf	4.0×10^{-6}	Gen II (NUREG-1150)
• Peach Bottom	4.5×10^{-6}	
• Sequoyah	5.7×10^{-5}	
• Surry	4.0×10^{-5}	
• Zion	3.4×10^{-4}	
• US-APWR	1.2×10^{-6}	Gen III and Gen III+ Design Control Documents, Chapter 19, NRC website
• AP1000	2.4×10^{-7}	
• US-EPR	5.3×10^{-7}	
• ABWR	1.6×10^{-7}	
• ESBWR	3.2×10^{-8}	GEH

Focus Probably Should Be On Current Generation NPPs

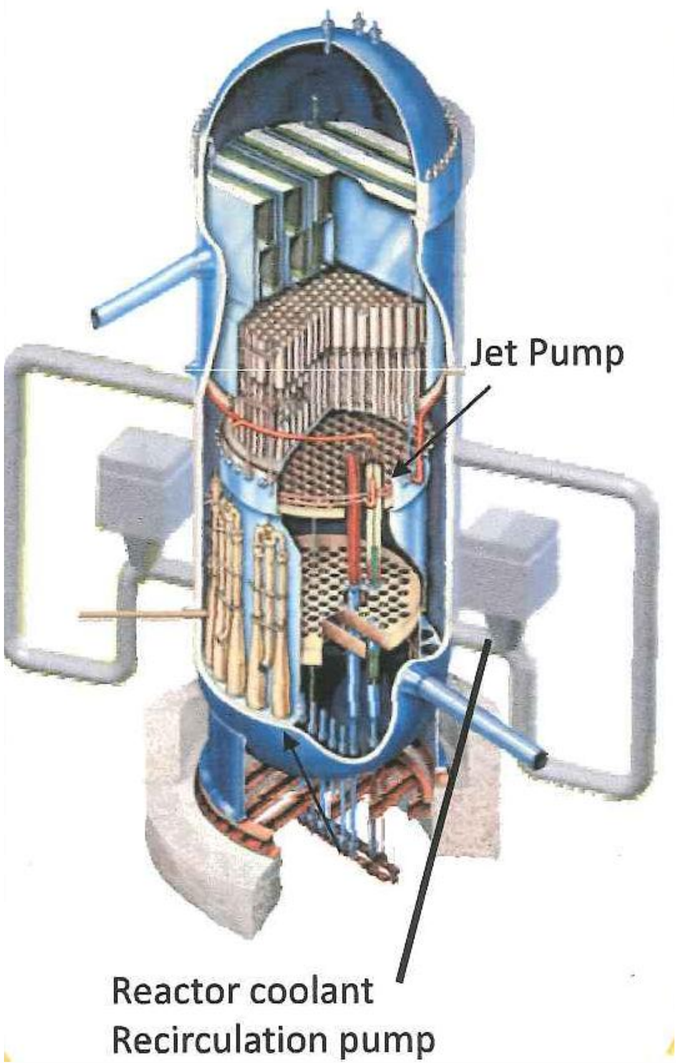
- Operation into the 2030's and maybe the 2040's
- Higher CDFs
- Consider both PWRs and BWRs
- SA Benchmarks :
  - PWRs : TMI-2
  - BWRs : Fukushima Dai-ichi
    - Which unit ??

PWRs and BWRs are different !!

For SA Modeling, ALL Components in a PWR MUST be Considered and Modeled



After SCRAM, the BWR is Essentially an Isolated Vessel

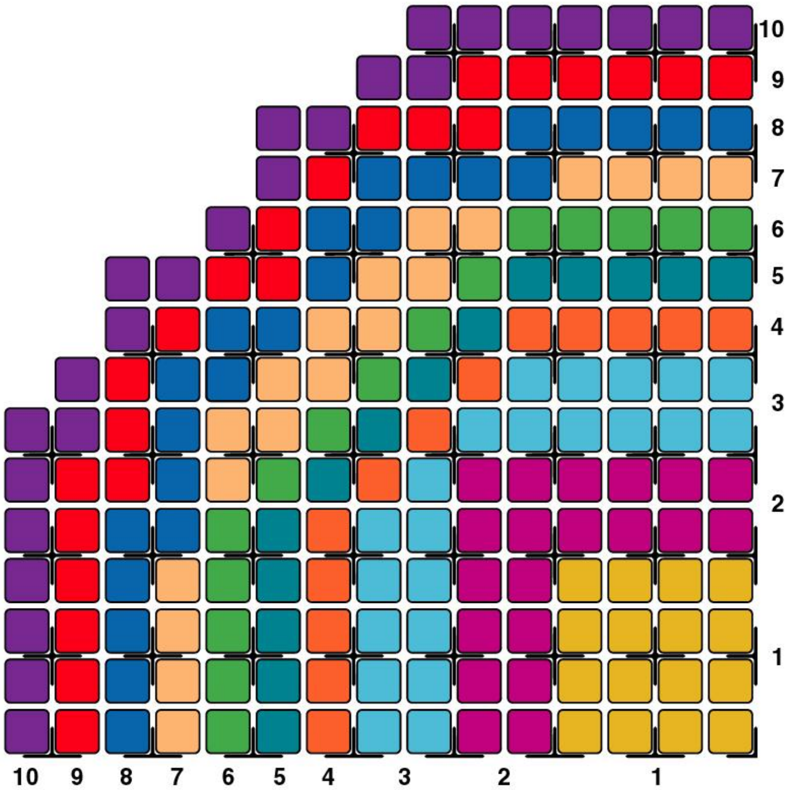


- MSIVs close
- Recir. Pumps trip

Predicted BWR Severe Accident Response Is Different from That Expected of a PWR in Several Aspects

- Much more zirconium metal
  - For example: consider comparable 1000MWe PWRs and BWRs
    - BWR has 3X the zirconium in the PWR core
    - BWR has ~1.5X the fuel in the PWR core
- Isolated reactor vessel
- Different radionuclide release paths
- Reduction in power factor in the outer core region (next slide)

Definition of Radial Zones for Browns Ferry Unit 1 Cycle 6 Core

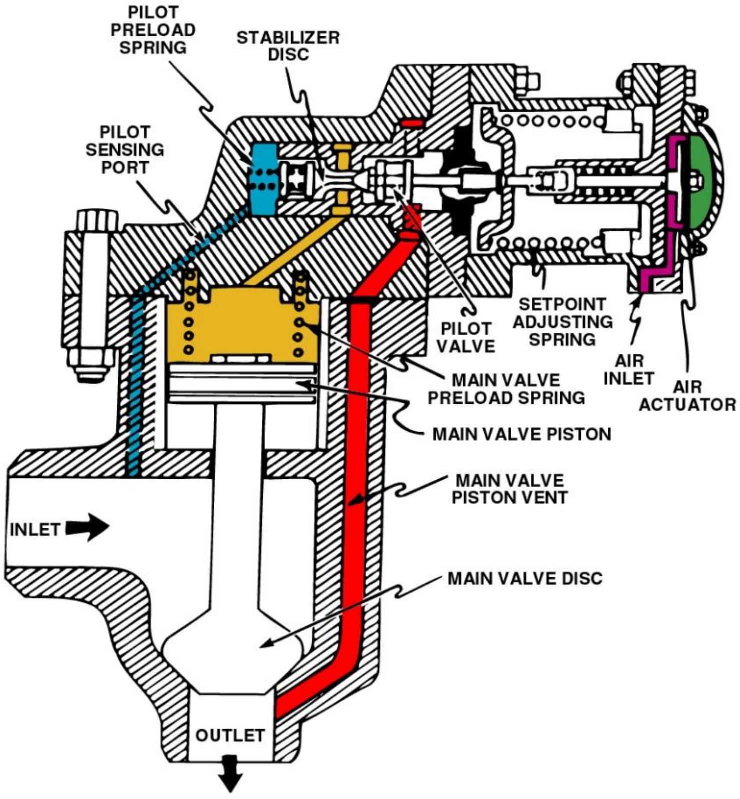


RADIAL ZONE	NO. OF BUNDLES IN 1/4 CORE	POWER RANGE	MEAN PEAKING FACTOR	VOLUME FRACTION
1	16	0.97 – 1.27	1.115	0.08377
2	20	1.06 – 1.32	1.160	0.10471
3	22	1.10 – 1.34	1.233	0.11518
4	13	1.23 – 1.36	1.309	0.06806
5	14	1.10 – 1.33	1.194	0.07330
6	15	1.15 – 1.37	1.258	0.07853
7	19	1.10 – 1.23	1.155	0.09948
8	24	0.92 – 1.13	1.000	0.12565
9	23	0.55 – 0.85	0.670	0.12042
10	25	0.23 – 0.47	0.354	0.13089
	191			

Note

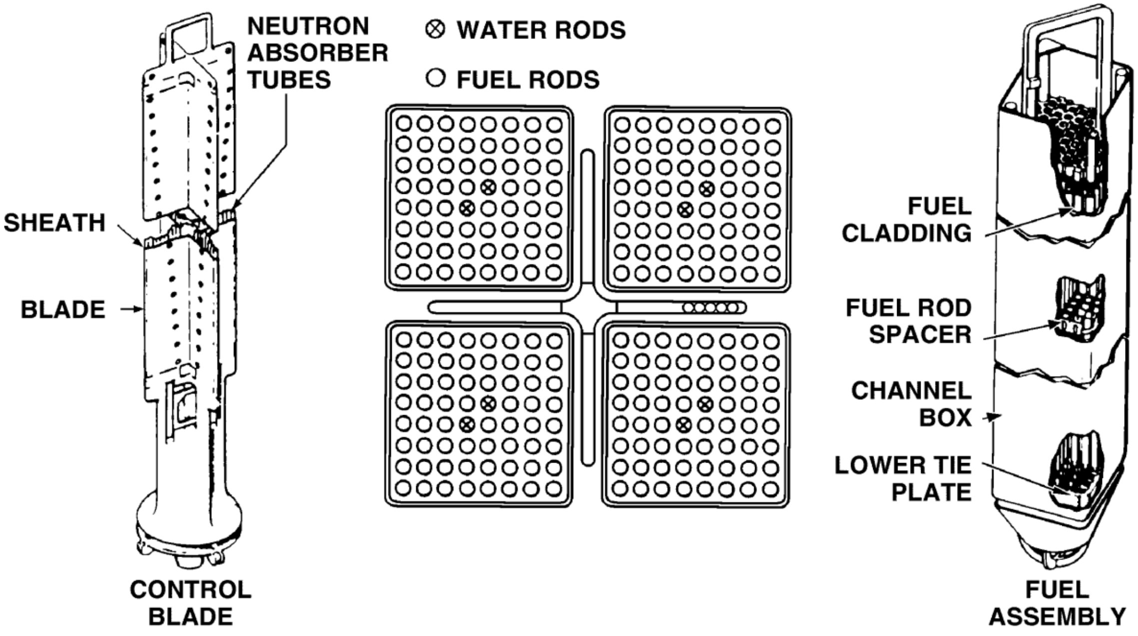
Predicted BWR Severe Accident Response Is Different from That Expected of a PWR in Several Aspects (continued)

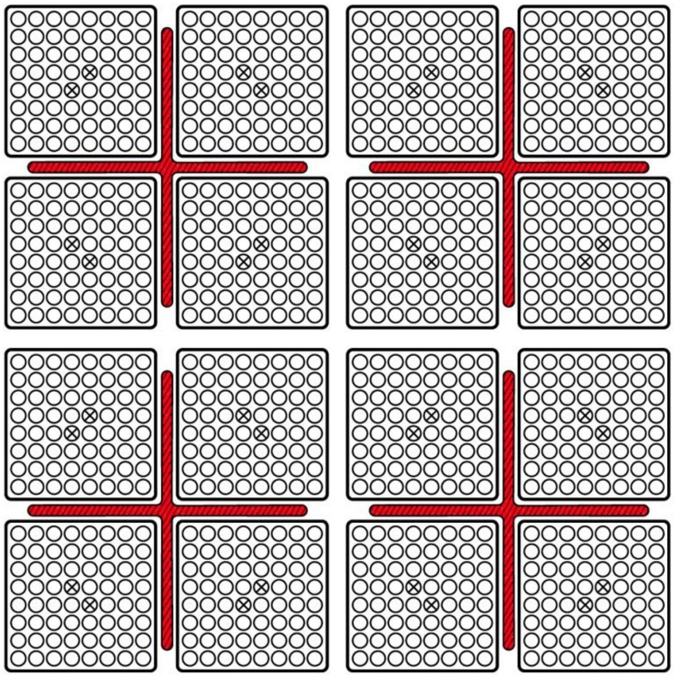
- Effects of safety relief valve actuations
  - 11 SRVs at BF
  - Each SRV rated at ~6.5% of rated steam generation
  - 50-75 psi pressure drop
  - Significant effect on water level and core flow



Predicted BWR Severe Accident Response Is Different from That Expected of a PWR in Several Aspects (continued)

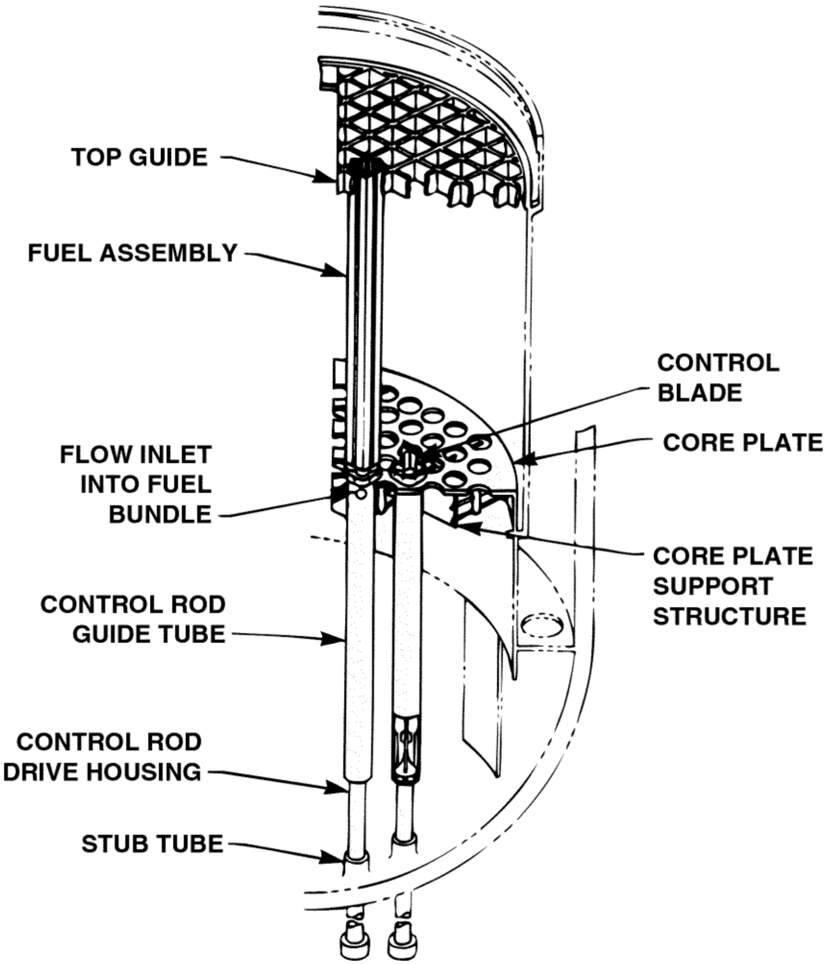
- Progressive relocation of core structures





Predicted BWR Severe Accident Response Is Different from That Expected of a PWR in Several Aspects (continued)

- Importance of core plate boundary
  - Does NOT support the core
- Steel structures in vessel
  - In-core (185 CBs at BF)
    - ~17 mt SS
    - ~1mt B_4C
  - Lower plenum at BF
    - ~90 mt SS
- Large amount of water in vessel lower plenum (enough to quench 3 molten cores)



Questions ??

Potential Hydrogen Generation in BWR Core

NPP	Rated Power (MWt)	Zr (kg)	SS (kg)	Potent.H_2 Generation from Zr (kg)	Potent.H_2 Generation from SS (kg)
1F1	1380	34270	9013	1515	325
1F2-5	2381	46949	12729	2075	459
Browns Ferry	3433	65455	17189	2893	620

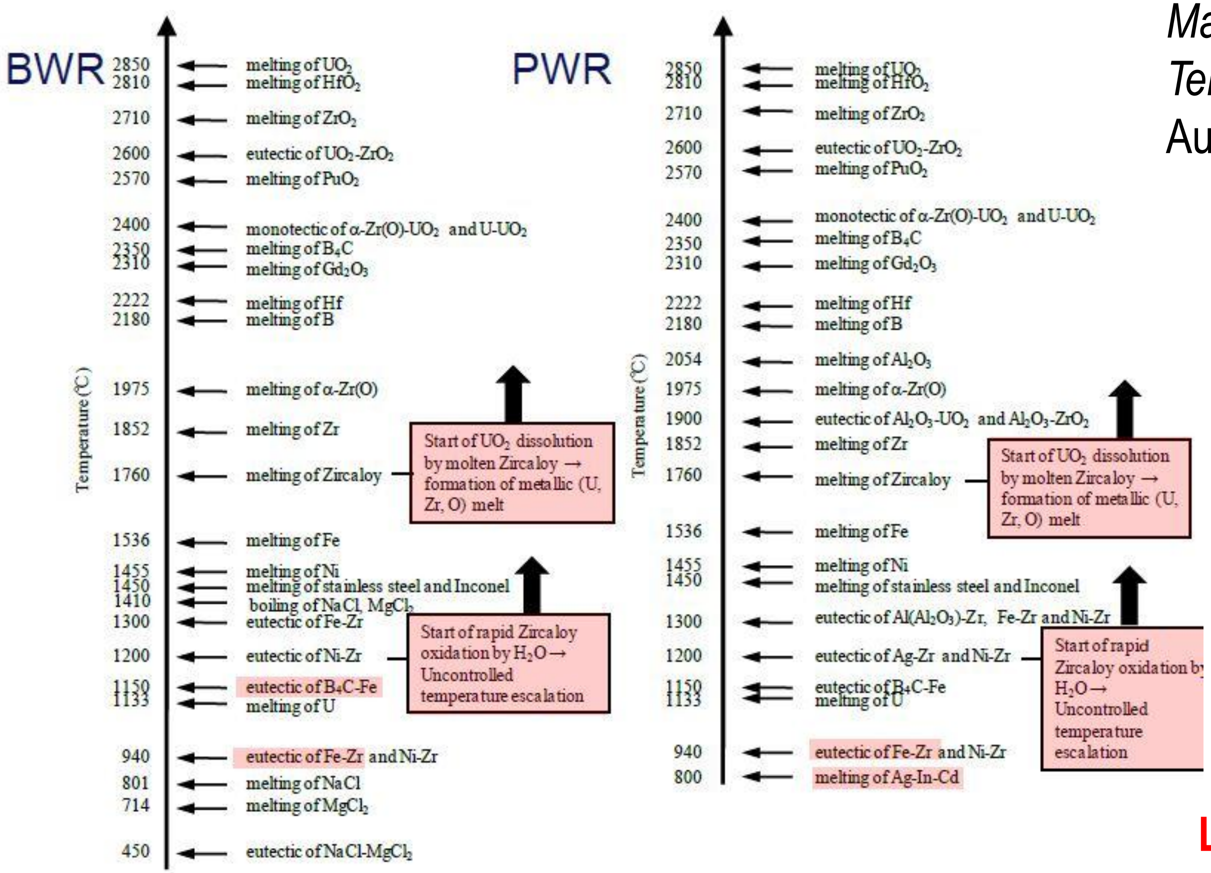
Notes:

•In PWRs and BWRs of comparable powers (e.g. 1000 MWe), there is 3 times the amount of Zr in a BWR core versus a PWR core (and 1.5 times the amount of UO_2)

•For the Fukushima hydrogen explosions, it is estimated that only 100-300 kg H_2 was needed per explosion

Core Degradation Process : Temperature Scale

P. Hofmann, S. Hagen, G. Schanz, and A. Skokan, *Reactor Core Materials Interactions at Very High Temperatures*, **Nuc.Tech. 87(1)** August 1989



Eutectic of Fe-Zr (940°C)
Eutectic of Fe-B_4C (1150°C)
Melting of Ag-In-Cd (800°C)

≦

Rapid Zircaloy oxidation by H_2O (~1200°C)
Melting of Zircaloy (1760°C)

Liquefaction starts with the formation of eutectic mixtures:
<u>Confirmed in all integrated SA experiments</u>

Experimental Bases (PWRs) for Current SA Codes

Test/Accident	Description	Phenomena Tested
PWR		
Loss Of Fluid Test (LOFT)		
FP-2	Large scale fuel bundle severe damage test with reflood	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, quench behavior
Power Burst Facility Severe Fuel Damage (PBF SFD)		
SFD ST	Heatup of PWR fuel assembly. Top of fuel assembly uncovered due to coolant boiloff.	Boiloff rate, temperature and fuel rod damage, H_2 production
SFD 1-1	Small scale bundle heatup with unirradiated fuel. Steam flow through assembly.	Temperature and fuel rod damage, H_2 production
SFD 1-4	Small scale bundle heatup with irradiated fuel rods included helium quench phase	Fuel heatup (temperatures) and damage, H_2 generation
SNL ACRR		
MP series	Small scale simulation of the heatup of PWR in-core debris bed, formation of melt pool and crust	Debris bed melting, formation of ceramic crust, melt pool growth, reformation of crust
ST series	Small scale fission-product release experiments from used irradiated Zircaloy clad fuel	Fission product release from irradiated fuel
Full-Length, High-Temperature (FLHT)		
FLHT-2	Heatup of full-length PWR fuel assembly. Coolant boiloff of	Boiloff rate, fuel heatup (temperatures) and damage, and H_2 generation
FLHT-4	Heatup of full-length PWR fuel assembly. Coolant boiloff.	Boiloff rate, fuel heatup (temperatures) and damage, H2 generation, noble gas release.
FLHT-5	Heatup of full-length PWR fuel assembly. Gradual boiloff of coolant. Most severe of the FLHT tests.	Boiloff rate, fuel heatup (temperatures) and damage, H_2 generation, noble gas release
CORA		
CORA-2	Small (23 rods) fuel assembly with electrical heater rods, INCONEL spacers, reference test, 1987.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation
CORA-3	Small fuel assembly with electrical heater rods, INCONEL spacers, reference test, high temperature, 1987.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, CORA-2/high temperature
CORA-5	Small fuel assembly with electrical heater rods, Ag-In-Cd absorber, 1988.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, CORA-2/Ag-In-Cd absorber.
CORA-7	Large (52 rods) fuel assembly with electrical heater rods. Flow of steam and Ar through assembly, slow cooling, 1990.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, CORA-5/bundle size
CORA-9	Small fuel assembly with electrical heater rods, Ag-In-Cd absorber, 10 bar system pressure, 1989.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, CORA-5/system pressure
CORA-10	Small fuel assembly with electrical heater rods, Ag-In-Cd absorber, low steam flow rate (2 g/s),1992.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, CORA-5/steam flow rate (2 g/s versus standard of 12 g/s).
CORA-12	Small fuel assembly with electrical heater rods, Ag-In-Cd absorber, rapid quenching, 1988.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, CORA-5/quenching
CORA-13	Small electrically heated fuel assembly. Flow of steam and Ar followed by rapid reflood (quenching) of hot assembly, 1990	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, quench behavior, OECD standard problem, CORA-12/quench at higher temperature.
CORA-15	Small fuel assembly with electrical heater rods, Ag-In-Cd absorber, rods with high internal pressure, 1989.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, influence of clad ballooning and bursting.
CORA-29	Small fuel assembly with electrical heater rods, Ag-In-Cd absorber, pre-oxidized cladding, 1991.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, CORA-5/pre-oxidation.
CORA-30	Small fuel assembly with electrical heater rods, Ag-In-Cd absorber, slow heatup (0.2 K/s), 1991.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, CORA-5/heatup rate (0.2 K/s versus standard of 1 K/s)..
PHEBUS		
B9+	Fuel assembly heatup and damage with steam flow followed by He to represent extreme steam starvation.	Fuel heatup (temperatures), damage, and liquefaction, bundle collapse, eutectic behavior, cladding oxidation, H_2 generation
FPT-0	Fuel assembly heatup with steam flow	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation
FPT-1	Integral severe fuel damage tests: fuel bundle, steam generator deposition, containment aerosol/chemistry	Fuel heatup (temperatures), damage, and liquefaction, bundle collapse, eutectic behavior, cladding oxidation, H_2 generation, fission product release, speciation and volatility, Ag aerosol transport and deposition, containment chemistry and deposition, and iodine partitioning
FPT-2	Integral severe fuel damage tests: fuel bundle, steam generator deposition, containment aerosol/chemistry - test includes steam starved period	Fuel heatup (temperatures), damage, and liquefaction, bundle collapse, eutectic behavior, cladding oxidation, H_2 generation, FP release, speciation and volatility, transport and deposition, containment chemistry and deposition, and iodine partitioning
FPT-3	Integral severe fuel damage tests: fuel bundle, steam generator deposition, containment aerosol/chemistry - test includes B_4C control rod	Fuel heatup (temperatures), damage, and liquefaction, bundle collapse, eutectic behavior, cladding oxidation, H_2 generation, FP release, speciation and volatility, B_4C control rod oxidation transport and deposition, containment chemistry and deposition, and iodine partitioning
FPT-4	Melt progression in debris bed geometry with irradiated fuel	Late phase melt progression and low volatility FP release.
QUENCH		
QUENCH -01 through Quench-15	Small (20-30 rods) fuel assembly with electrical heater rods, Ag-In-Cd absorber (one test with B_4C control rod), one test with E110 cladding, two tests with advanced western cladding, remaining tests with Zircaloy-4 cladding, 1998-2009, FZK.	Fuel heatup (temperatures) and damage, cladding oxidation, H_2 generation, quenching.
TMI-2 accident	Full scale PWR accident.	System pressure, RCS piping heatup and final state of reactor core. Indirect measurement of H_2 production.

- >40 Experiments
- Includes
  - large scale tests (LOFT, TMI)
  - debris beds (MP)
  - fission product release series (PHEBUS)
  - Most tests focus on in-core degradation (notably CORA and QUENCH)
  - In-pile with irradated fuel rods (LOFT, TMI, PBF, FLHT, PHEBUS)
  - Out-of-pile tests (CORA and QUENCH)

Experimental Bases (BWRs) for Current SA Codes

Test/Accident	Description	Phenomena Tested
BWR		
Annular Core Research Reactor Damage Fuel Tests (ACRR DF)		
DF-4	Small bundle test that included fuel, channel box and SS control blade with B_4C, 1986, SNL.	Fuel heatup (temperatures), fuel damage, cladding oxidation, H_2 generation, B_4C-SS eutectic interaction, fuel liquefaction, fuel rod collapse
CORA		
CORA-16	Small (18 rods) electrically-heated fuel assembly, with channel walls and B_4C/SS control blade. Flow of steam and Ar, slow cool-down, 1988, FZK.	Fuel heatup (temperatures), fuel damage, cladding oxidation, H_2 generation
CORA-17	Small electrically-heated fuel assembly, with channel walls and B_4C/SS control blade. Flow of steam and Ar, followed by rapid reflood of hot assembly, 1989, FZK.	Fuel heatup (temperatures), fuel damage, cladding oxidation, H_2 generation, CORA-16/quenching
CORA-18	Large (48 rods) electrically-heated fuel assembly, with channel walls and B_4C/SS control blade. Flow of steam and Ar, slow cool-down, 1990, FZK.	Fuel heatup (temperatures), fuel damage, cladding oxidation, H_2 generation, CORA-16/bundle size.
CORA-28	Small electrically-heated fuel assembly, with channel walls and B_4C/SS control blade. Flow of steam and Ar, slow cooldown, preoxidized cladding, 1992, FZK.	Fuel heatup (temperatures), fuel damage, cladding oxidation, H_2 generation, CORA-16/preoxidized cladding
CORA-31	Small electrically-heated fuel assembly, with channel walls and B_4C/SS control blade. Flow of steam and Ar, slow cooldown, slow initial heatup (~0.3 K/s), 1991, FZK.	Fuel heatup (temperatures), fuel damage, cladding oxidation, H_2 generation, CORA-16/heatup rate (0.3 K/s versus 1 K/s)
CORA-33	Small electrically-heated fuel assembly, with channel walls and B_4C/SS control blade. Dry core conditions, slow cooldown, 1992, FZK.	Fuel heatup (temperatures), fuel damage, cladding oxidation, H_2 generation, CORA-31/steam-starved conditions
Full-Length, High-Temperature (FLHT)		
FLHT-6	Heatup of full-length BWR fuel assembly. Gradual boiloff of coolant. Most severe of the FLHT tests, PNL/NRU.	Boiloff rate, fuel heatup (temperatures) and damage, H_2 generation, noble gas release. NOTE: this test was never executed, assembly was fabricated and inserted in the NRU but was canceled by Canadian PM orders.
XR		
XR1-1 and XR1-2	Small fuel assembly, with channel walls and B_4C/SS control blade. 1994, SNL.	Full scale section of a BWR core with all core-plate region component structures (grids, tie plate, nose piece, fuel support piece, and core-plate). Response of lower core structures (~1 m) to prototypic relocating liquid materials from upper core.
XR2-1	Large fuel assemblies (four represented with a total of 71 rods), with channel walls and B_4C/SS control blade. 1996, SNL.	Full scale section of a BWR core with all core-plate region component structures (grids, tie plate, nose piece, fuel support piece, and core-plate). Response of lower core structures (~1 m) to prototypic relocating liquid materials from upper core.

- 9 experiments
- Includes
  - Most tests focus on in-core degradation (DF-4, CORA)
  - One in-pile test (DF-4)
  - No tests with irradiated fuel
  - Out-of-pile tests (CORA and XR)
  - XR focus is on lower 1 m of core (including coreplate)

Mechanism of Debris Relocation to Lower Plenum Not Established for BWR Severe Accident Sequences

- Similar to Three Mile Island?
  - Debris bed forms in core region
  - Sudden entry of large mass of molten material to lower plenum
- Or controlled by BWR Geometry?
  - Most debris passes through core plate
  - Relocating material quenched
  - Debris bed forms in lower plenum

⇒ Sandia XR2-1 “BWR Metallic Melt Relocation Experiment” [NUREG/CR-6527, August 1997] demonstrated BWR-unique early core plate passage of debris

OCDE/GD(96)14 or NEA/CSNI/R(95)21

COPIE REGISTRY
Ne pas prendre S.V.P.
Please do not take

General Distribution
OCDE/GD(96)14

AEN NEA

NEA/CSNI/R(95)21

IN-VESSEL CORE DEGRADATION
CODE VALIDATION MATRIX

OCDE OECD

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
OECD NUCLEAR ENERGY AGENCY
Le Seine Saint-Germain – 12, boulevard des Îles
F-92130 Issy-les-Moulineaux (France)
Tel. (33-1) 45 24 82 00 Fax (33-1) 45 24 11 10

IN-VESSEL CORE DEGRADATION
CODE VALIDATION MATRIX

T J Haste (AEA Technology, Winfrith), B Adroguer (CEA, Cadarache),
R O Gauntt (Sandia National Laboratories), J A Martinez (Nuclear Safety Board, Spain),
L J Ott (Oak Ridge National Laboratories), J Sugimoto (Japan Atomic Energy Research Institute)
and K Trambauer (GRS, Garching)

3. IDENTIFICATION OF PHENOMENA

3.1 Fission and Decay Heat

3.2 Fluid State

3.3 Initial Core Damage

3.4 Oxidation and Hydrogen Generation

3.5 Fission Product Release

3.6 Core Degradation and Melt Progression

For each facility, four sub-sections:

4.1.x.1 objectives
4.1.x.2 facility description
4.1.x.3 test description
4.1.x.4 processes quantified

Integral Experiments - Early Phase

4.1.1 NIELS
4.1.2 CORA
4.1.3 PHEBUS-SFD
4.1.4 PBF-SFD
4.1.5 NRU-FLHT
4.1.6 ACRR-ST
4.1.7 ACRR-DF
4.1.8 LOFT-LP-FP

Integral Experiments - Late Phase

4.1.9 PHEBUS-FP
4.1.10 ACRR-MP
4.1.11 SANDIA-XR

Whole Reactor

4.1.12 TMI-2

Miscellaneous

4.1.13 SCARABEE
4.1.14 ACRR-DC