Transient Effects and Severe Accidents (BDBAs) in LWRs

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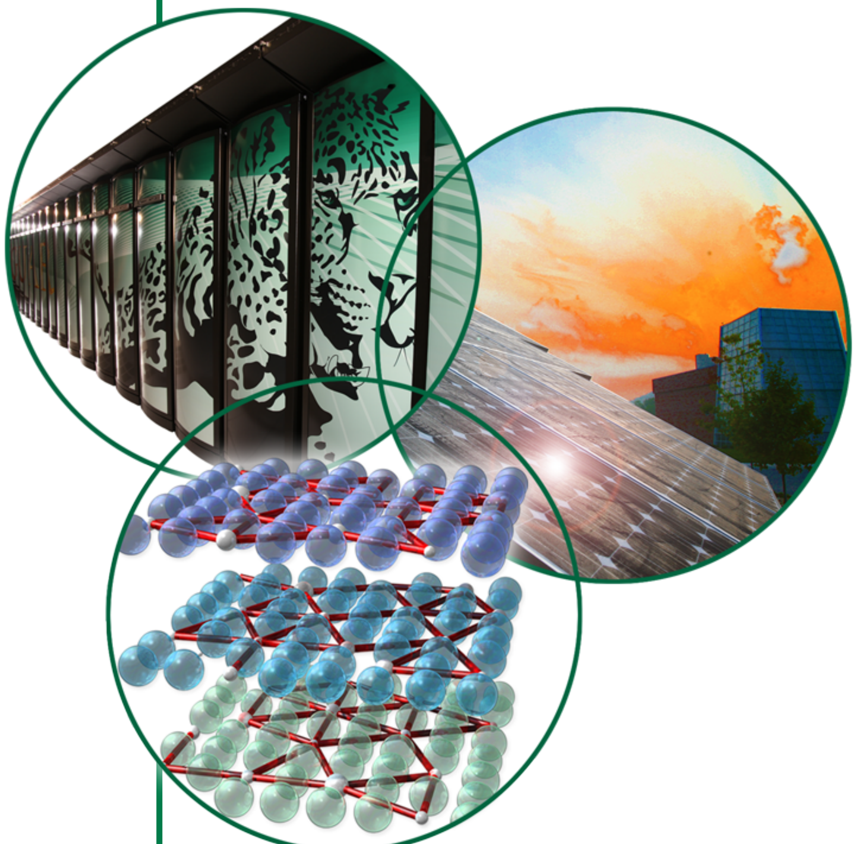

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Current Commercial UO2/Zirconium Alloy Cladding Configuration(s)

- **Methodologies (NRC design guidelines/regulations/etc) to address the "Front End", "Operation", and "Back End" for UO2/ Zirconium Alloy(s) fuel/cladding systems have been encoded in Government Regulations over the last 50+ years.**
- **UO_2 pellets in Zr alloy cladding**
  - **Meets or exceeds current regulations**
  - **Data and model validation: extensive**
- **Existing LWR reactors**
  - **Requires <u>forced transport</u> of heat to <u>ultimate heat sink</u> (not passive)**
  - **Extensive existing regulations**
  - **New fuel design constraints: RPV & core internals material and geometry, pumping power and ECCS system design, Spent Fuel Pool (SPF) design**

Accident Tolerant Fuel Design Goals (to decrease public risk)

Decrease/Prevent

Accident Initiators	Fuel Failure	Release & Consequences

Accident Initiators

- Manufacturing defects
  - Missing pellet surface
- Dimensional changes
  - Swelling
  - Bowing
- Flow blockage
- CRUD & Corrosion
- Loss of cooling
- Reactivity Insertion
- Other

Fuel Failure

- Decay heat
- Cooling
  - DNB, CHF
  - Thermal resistances
  - Geometry
- Structure limits
  - Melting point
  - Eutectic interactions
  - Ductility & strength
    - Oxidation
    - Hydriding
  - Fatigue & creep
- Phenomena
  - Fretting
  - Ballooning
  - PCMI, FCCI, SCC
  - Oxidation
- Leakage/melting/etc

Release & Consequences

- Containment Failure
  - H2 generation
  - DCH
  - FCI
  - MCCI
- RN Mobility
  - Retention
  - Particle size
- Released RN Composition

Operation

- Operational Accidents
  - AOO ($\sim 10^{-2}$ / reactor year)
  - Anticipated Accident / Design Basis Accident (10^{-2} to 10^{-5} / reactor year)
    - LOCA , RIA
  - Beyond Design Basis Accident (10^{-5} to 10^{-7} / reactor year)
    - i.e. Severe Accidents

DBAs: LOCA

- With respect to the LOCA Design Bases Accident in current regulations:
  - Focused on a guillotine break of primary coolant piping of a reactor at full power with subsequent scram with UO_2 fuel and zirconium alloy cladding,
    - Must avoid cladding temperatures > 1200°C in the resulting transient
    - Must keep through wall cladding reaction to <17%
    - Must maintain >1% ductility in the cladding
    - Must maintain a coolable geometry without dispersal of fuel into the coolant
  - Note 1: at full power, the UO2 fuel has a high centerline temperature and a large temperature gradient within the fuel pellet and as a result a large amount of internal energy at the beginning of the accident. In a SA, this is not the case – low radial thermal gradient (decay heat)
  - Note 2: lower fuel centerline temperatures (due to higher thermal conductivity) would impact DBAs and possibly AOOs
- These requirements essentially determine the design, capacity, and response of the ECCS systems
  - The above limits/restrictions could/would change for different fuel/cladding system; would need experiments to determine the new limits

DBAs: RIAs

- With respect to the Reactivity Insertion Accident (RIA) Design Bases Accident in current regulations:
  - Focused on a sudden explusion of a control element (rod in a PWR , blade in a BWR) from the core with the reactor at full power with UO_2 fuel and zirconium alloy cladding,
    - Sets a limit (W/g) on the fuel power generation due to the reactivity insertion (a function of the cladding oxidation and hydride content)
    - Must maintain a coolable geometry without dispersal of fuel into the coolant
  - These "limits" have been determined experimentally (for example in TREAT and CABRI) ; new fuel/cladding systems would need these RI "limits' to be determined experimentally (ergo, domestically TREAT)
- These requirements essentially determine the allowable control rod worth in the core design

Beyond Design Basis Accidents (BDBAs) i.e. Severe Accidents (SAs)

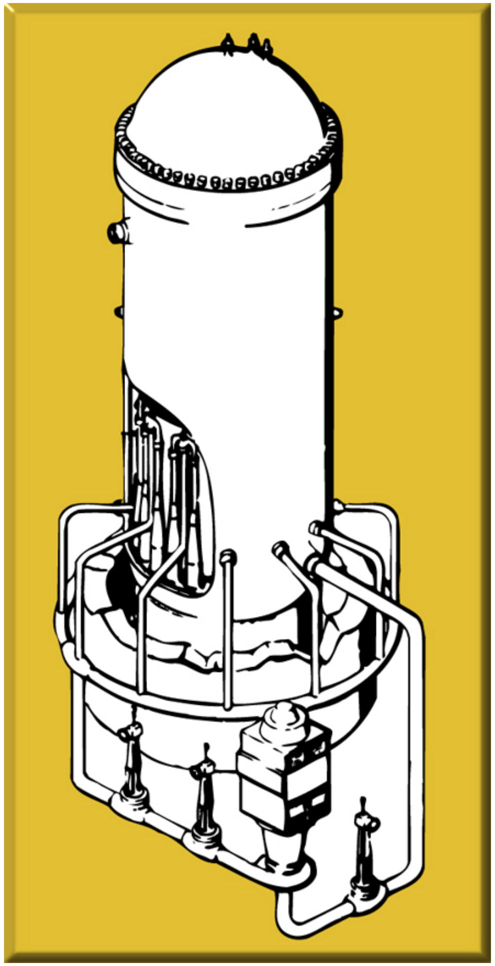
Three Mile Island Units 1 and 2
March 28, 1979

- Reactor scram: 04:00 3/28/79
- "Small break LOCA and loss of coolant"
- Core melt and relocation: ~05:00 – 07:30 3/28/79
- Hydrogen deflagration: 13:00 3/28/79
- Recirculation cooling: Late 3/28/79
- Phased water processing: 1979-1993
- Containment venting 43kCi Kr-85: July 1980
- Containment entry: July 1980
- Reactor head removed and core melt found: July 1984
- Start defuel: October 1985
- Shipping spent fuel: 1988-1990
- Finish defuel: January 1990
- Evaporate ~2.8 M gallons processed water: 1991-1993
- Cost: ~$1 billion

NRC Severe Accident Sequence Analysis (SASA) Programs Initiated in Late 1980

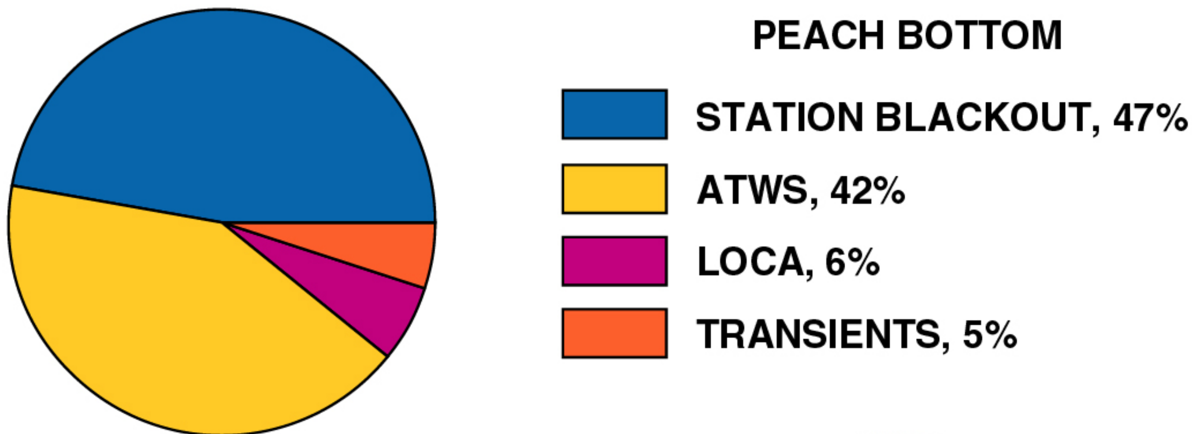
- Response to Three Mile Island
- PWR SA studies
  - SNL
  - INL
  - LANL
- BWR SA studies at ORNL
  - Follow-on to initial NRC SASA Program
  - Conducted at Oak Ridge National Laboratory, 1980-1999
  - Also Evaluated BWR Owners Group Emergency Procedure and Severe Accident Guidelines for NRR

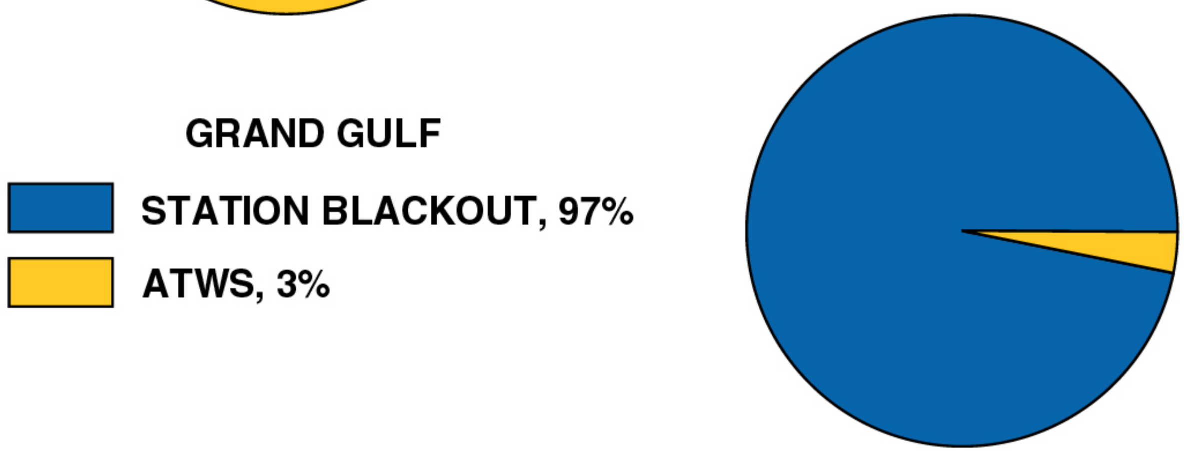
BWR Severe Accident Technology Activities at ORNL (1980-1999)



- Accident progression studies
  - Event sequence
  - Timing
  - Code application and model development
- Analytical support of experiments
  - Pretest planning
  - Posttest analyses
  - Diverse locations
    - ACRR (Sandia)
    - NRU (Chalk River)
    - CORA (Karlsruhe)
  - Code and model development
- Accident management strategies
  - Preventive
  - Mitigative
- Extension to advanced reactor designs

Boiling Water Reactor Contributors to Core Damage Frequency – NUREG-1150





LOCA (here) refers to a large break loss of coolant accident; the above are SA initiators and in all cases significant water/Zr reaction can/will occur

The Most Probable BWR Accident Sequence Involving Loss of Injection Is Station Blackout

BWR 3/4/5's with evolving containment design

			Station Blackout Core Damage Frequencies
Mark-1	**Peach Bottom**		~5E-6
	Short-term	**5%**	
	Long-Term	**42%**	
Mark-2	**Susquehanna***		
	Short-term	**52%**	
	Long-Term	**10%**	
Mark-3	**Grand Gulf**		
	Short-term	**96%**	
	Long-Term	**1%**	
ABWR			~1E-7
ESBWR			~3E-8

*From Plant IPE (NPE 86-003)

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.

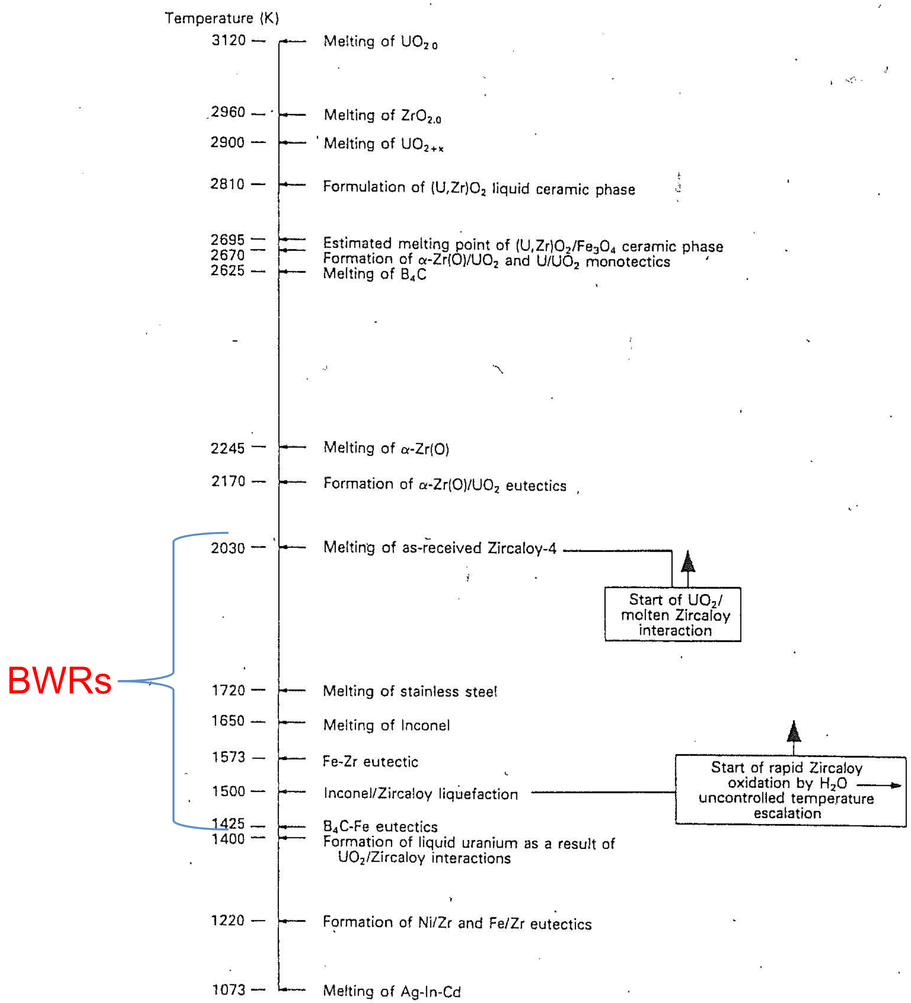
- >40 Experiments
- Includes
  - large scale tests (LOFT, TMI)
  - debris beds (MP)
  - fission product release series (PHEBUS)
  - Most tests focus on in-core degradation (notably CORA and QUENCH)
  - In-pile with irradated fuel rods (LOFT, TMI, PBF, FLHT, PHEBUS)
  - Out-of-pile tests (CORA and QUENCH)

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)

Core Degradation Process: Temperature Scale



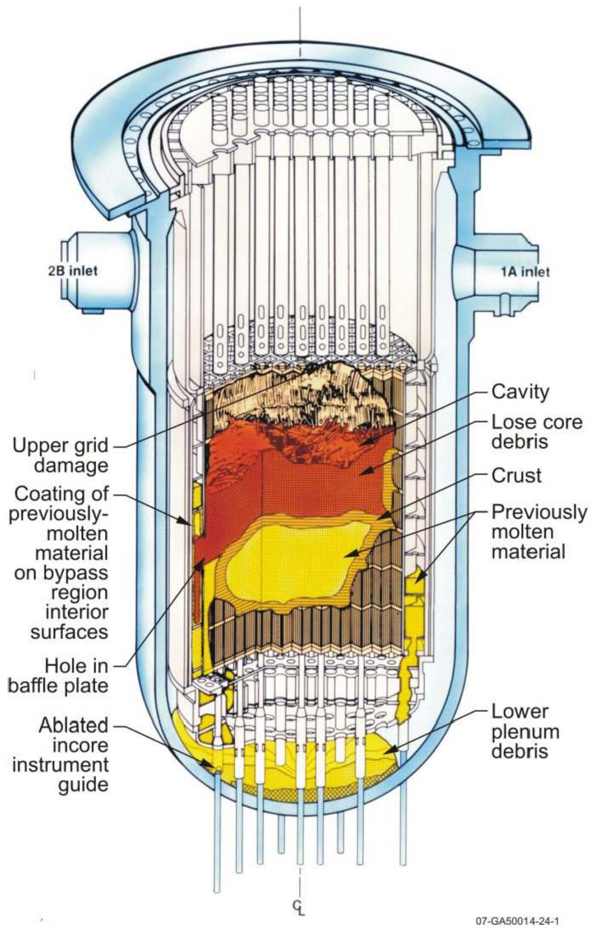
LWR severe-accident-relevant melting and eutectic temperatures.

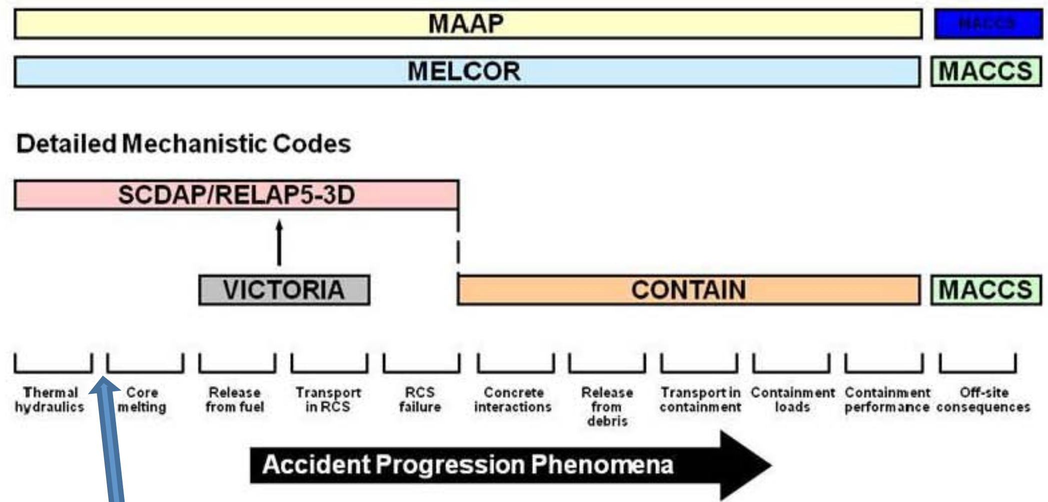
P.Hofmann,S.Hagen,G.Schanz, and A.Skokan, *Reactor Core Materials at Very High Temperatures*, **Nuc.Tech. 87 (1)**, 46, August 1989

Liquefraction starts with the formation of eutectic mixtures

- **Separate-effects materials interaction tests (Hofmann,et al)**
- **Confirmed in all integrated SA experiments**

Severe Accident Phenomena Modeled by U. S. – Developed Codes





Core-heatup, clad oxidation, H_2 generation

H_2 deflagation (if any)

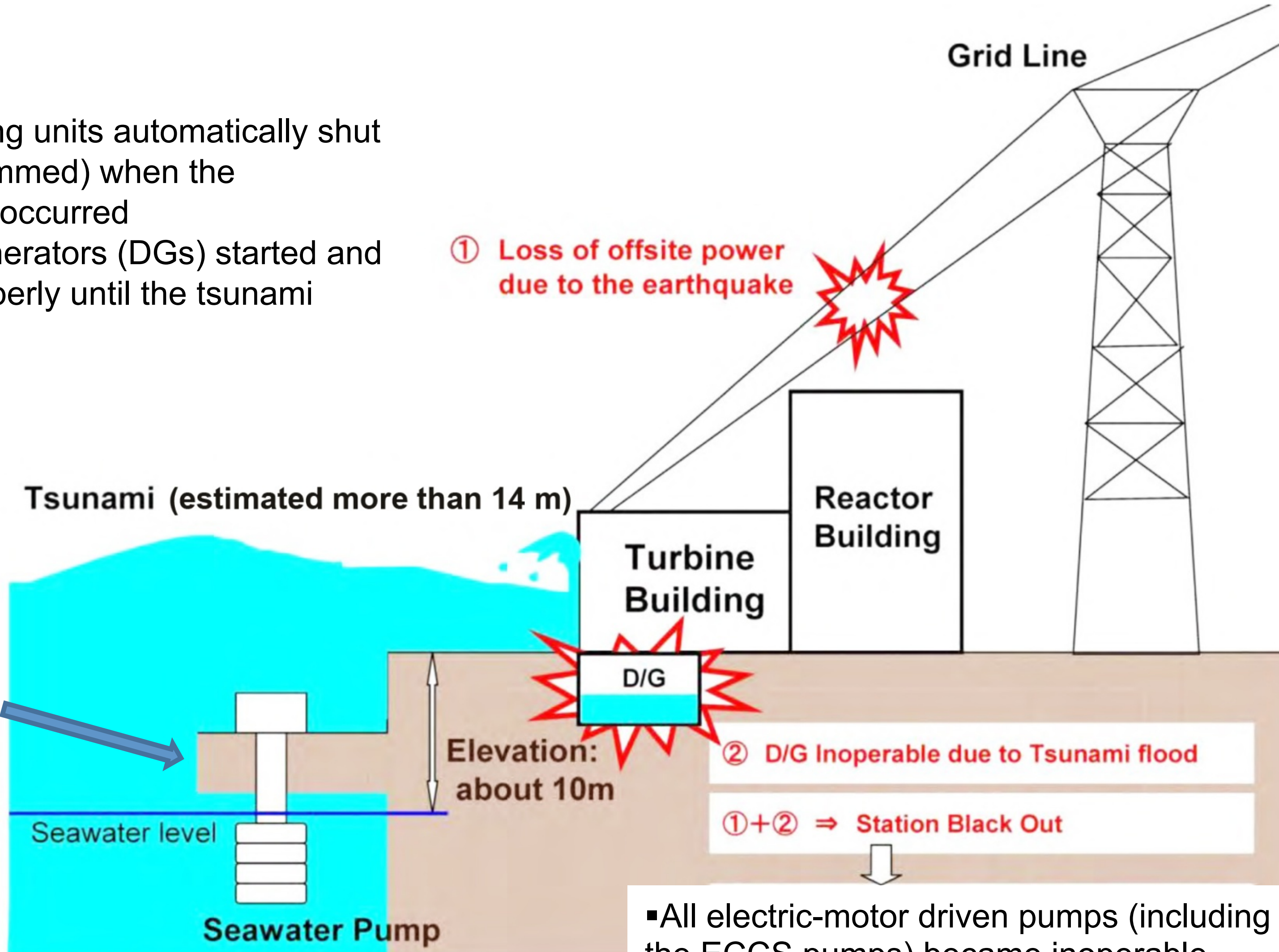
Example SA Case: SBO in a BWR

At Fukushima, the Earthquake + the Tsunami Created SBO Accidents in 1F1, 1F2 and 1F3

Note:

- All operating units automatically shut down (scrammed) when the earthquake occurred
- Diesel Generators (DGs) started and worked properly until the tsunami struck

Tsunami destroyed UHS



- All electric-motor driven pumps (including the ECCS pumps) became inoperable
- The steam-turbine-driven pumps (RCIC and HPCI) were available

OAK RIDGE National Laboratory

Station Blackout Involves Failure of AC Electrical Power

- **Loss of offsite power**
- **Emergency diesel-generators do not start and load**

Short-Term Station Blackout	Long-Term Station Blackout
Immediate Loss of Water Makeup	Loss of Water Makeup Following Battery Exhaustion
1F1	**1F2 and 1F3**

The First Accident Sequence Studied by the ORNL BWR SASA Group was a SBO at the Browns Ferry Unit 1 (BWR4 with a Mark-1 Containment)

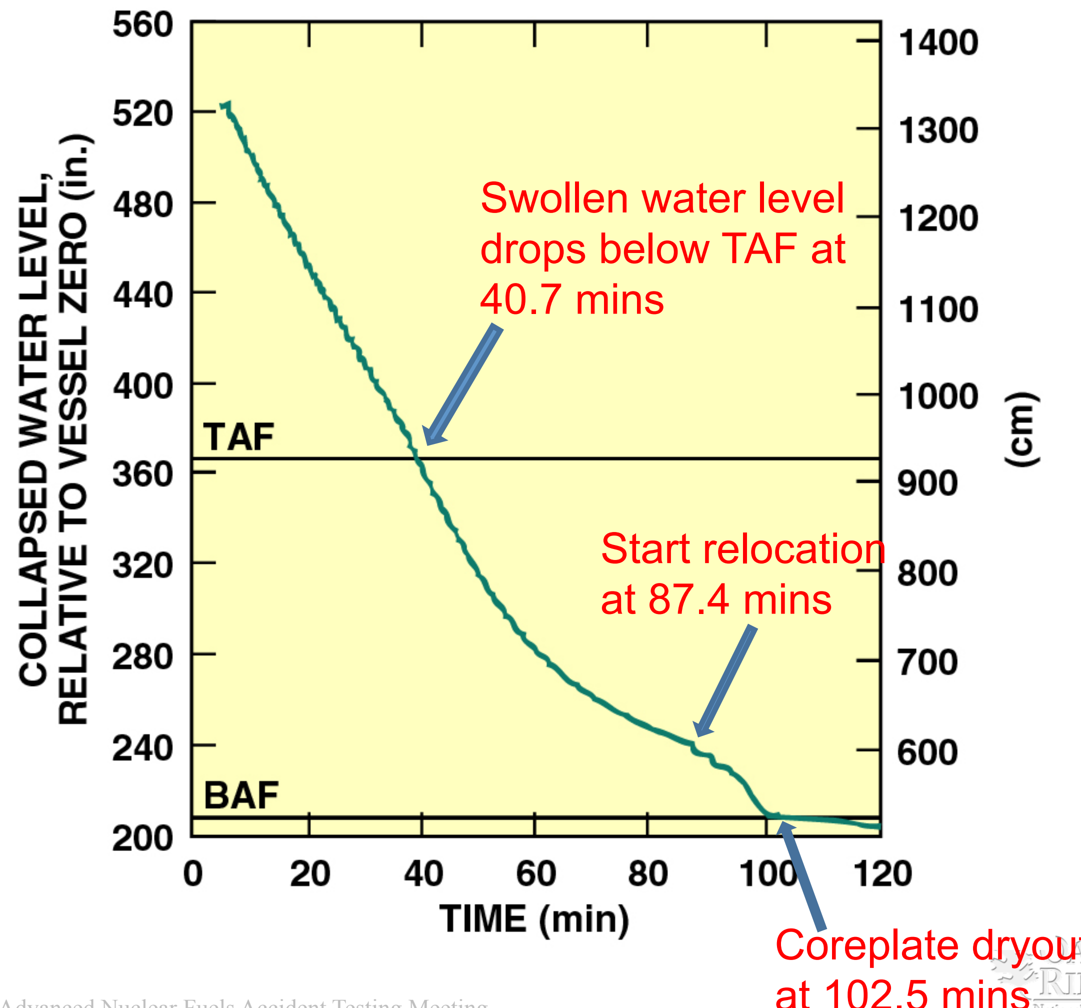
Published
Nov. 1981

Published
Aug. 1982

Short Term SBO for Grand Gulf: Collapsed Water Level Within RPV (No ADS Actuation)

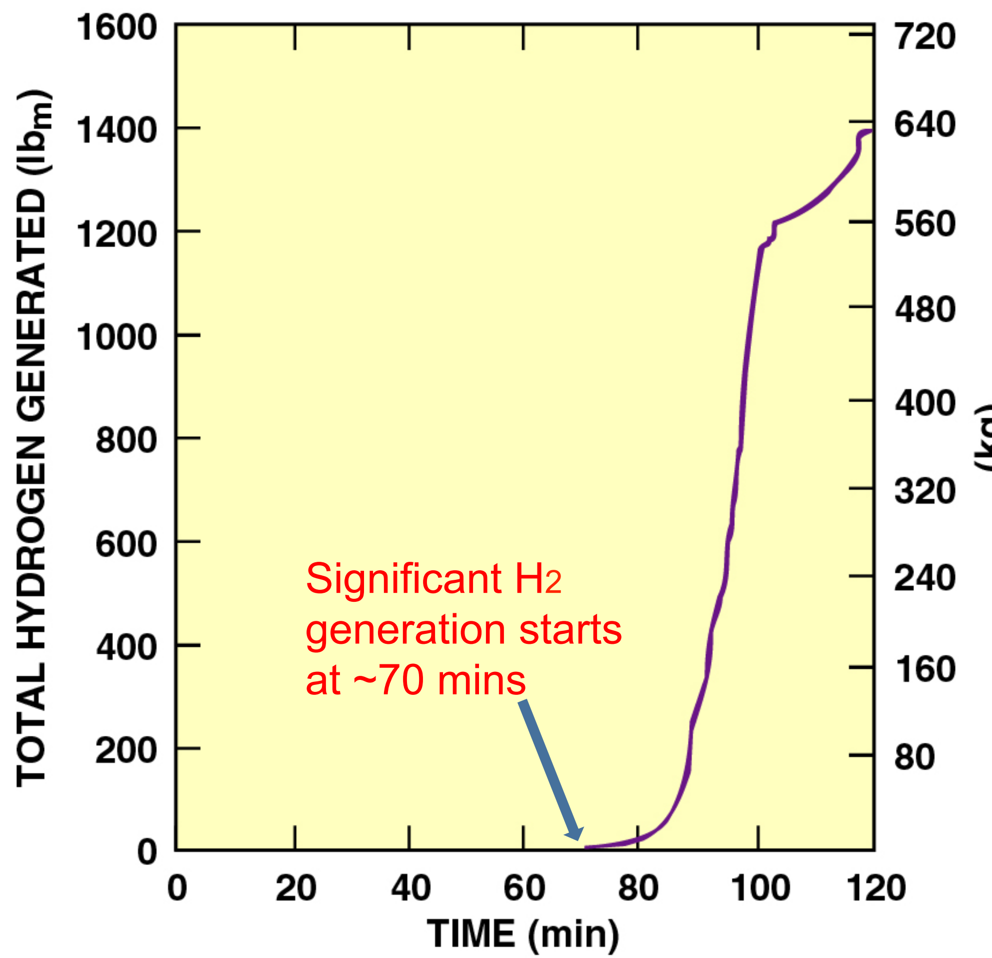
Illustration of ST-SBO timing : progression occurs very quickly

Grand Gulf Short Term Station Blackout without ADS Actuation



The Steam-Rich Situation Attendant to Core Relocation without ADS Produces Large Amounts of Hydrogen

Grand Gulf Short Term Station Blackout without ADS Actuation



Hydrogen Generation Within a BWR Core

1) $Zr + 2H_2O = ZrO_2 + 2H_2$ $\Delta H_{rx} =$ 140000 cal/gm mole Zr

2) $3Fe + 4H_2O = Fe_3O_4 + 4H_2$ $\Delta H_{rx} =$ 51667 cal/ gm mole Fe

3) $B_4C + 9H_2O = 4HBO_2 + CO + 7H_2$ $\Delta H_{rx} =$ 17000 cal/gm mole B_4C

4) from all the degraded core experiments (in BWR geometries):

a) <30% of the Zr reacted in-core

b) <10% of the steel reacted

c) <5% of the B_4C reacted

d) nearly all of the B4C was tied up as eutectic material with the steel components and Zr (~1425 K)

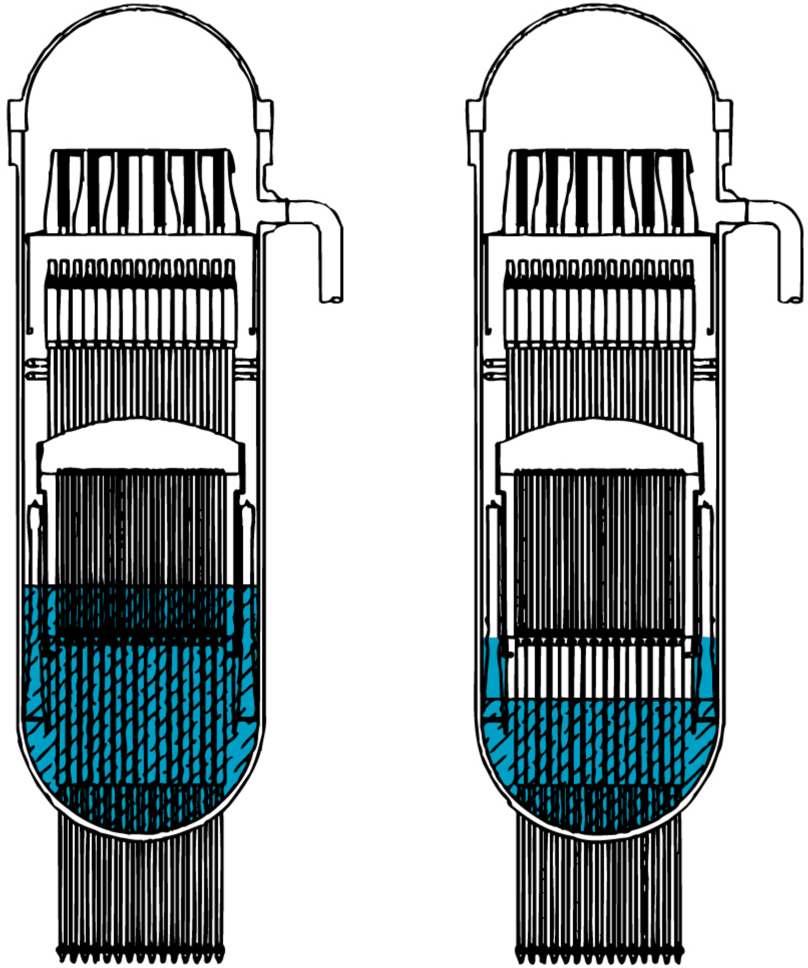
e) steel components also formed eutectics with the Zr (~1573 K)

f) liquified eutectics rapidly relocated lower into the core or onto coreplate before resolidifying

Potential Hydrogen Generation in a BWR Core

NPP	Rated Power (MWt)	Approximate Mass of Core Materials (excluding fuel) (considering only the fuel assemblies and control blades) (kg)			Potential Hydrogen (H_2) Generation (assuming all the material reacts) (kg)		
		Zircalloy	Stainless Steel	B_4C	Zircalloy	Stainless Steel (assuming only Fe reacts)	B_4C
Fukushima Unit 1	1380	34270	9013	531	1515	325	136
Fukushima Units 2-5	2381	46949	12729	750	2075	459	192
Browns Ferry	3433	65455	17189	1013	2893	620	259

BWR Emergency Procedure Guidelines Call for Manual Actuation of ADS Valves at or About One-Third Core Height; Flashing Drops Water Level Below the Core Plate

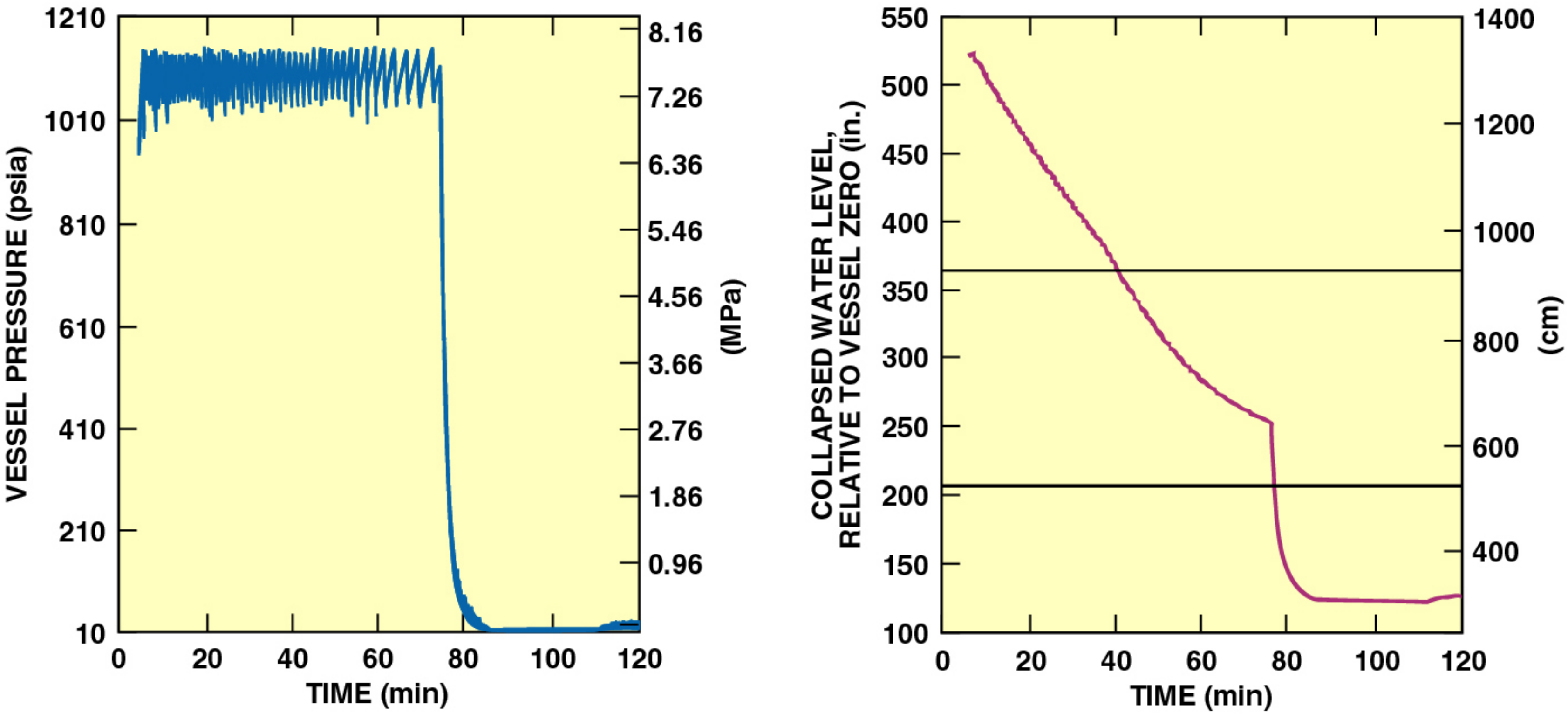


Manual Reactor Vessel Depressurization Provides an Advantage Because

- Steam cooling of uncovered region of core delays onset of core melting
- Core is steam-starved when runaway metal-water reaction temperatures are reached
  - i.e., little or no oxidation and H_2 generation

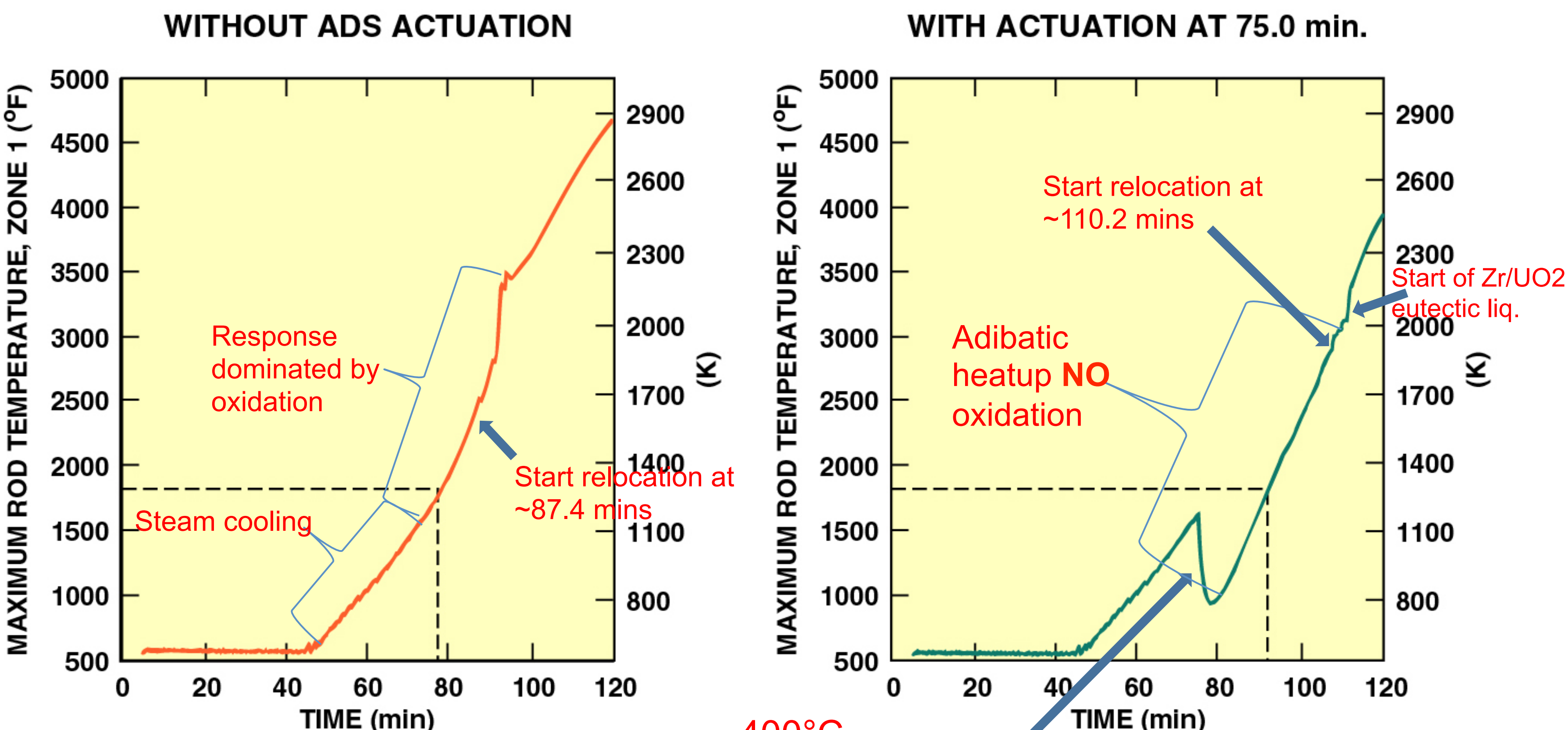
The EPGs Provide for Manual Actuation of ADS at About One-Third Core Height

Grand Gulf
Short Term Station Blackout
ADS Actuation at 75.0 min.



Vessel Depressurization at One-Third Core Height Provides Steam Cooling that Temporarily Reverses Core Heatup

Grand Gulf Short Term Station Blackout

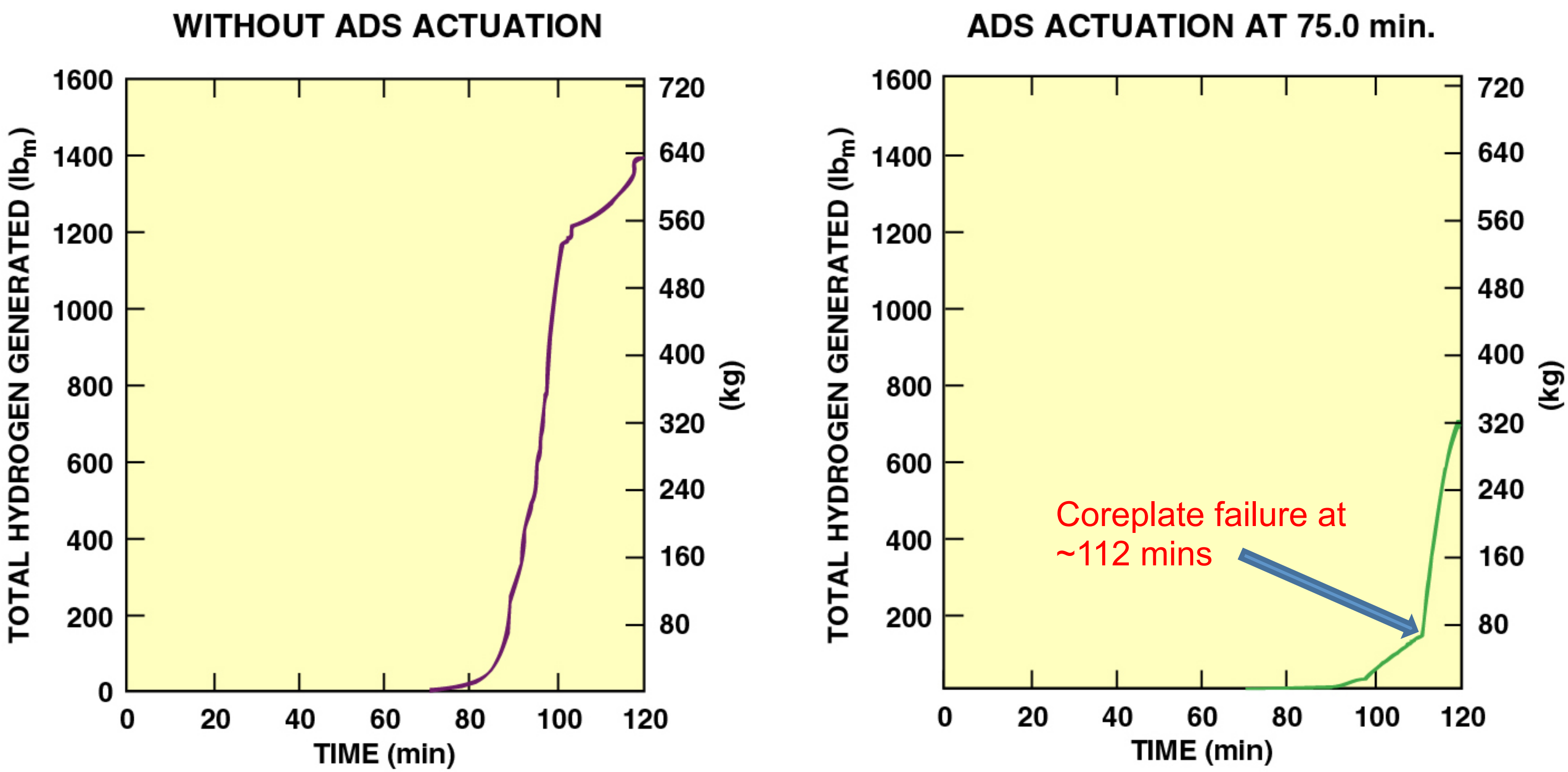


~400°C temperature drop

Margin = 110.2 – 87.4 = ~23 mins

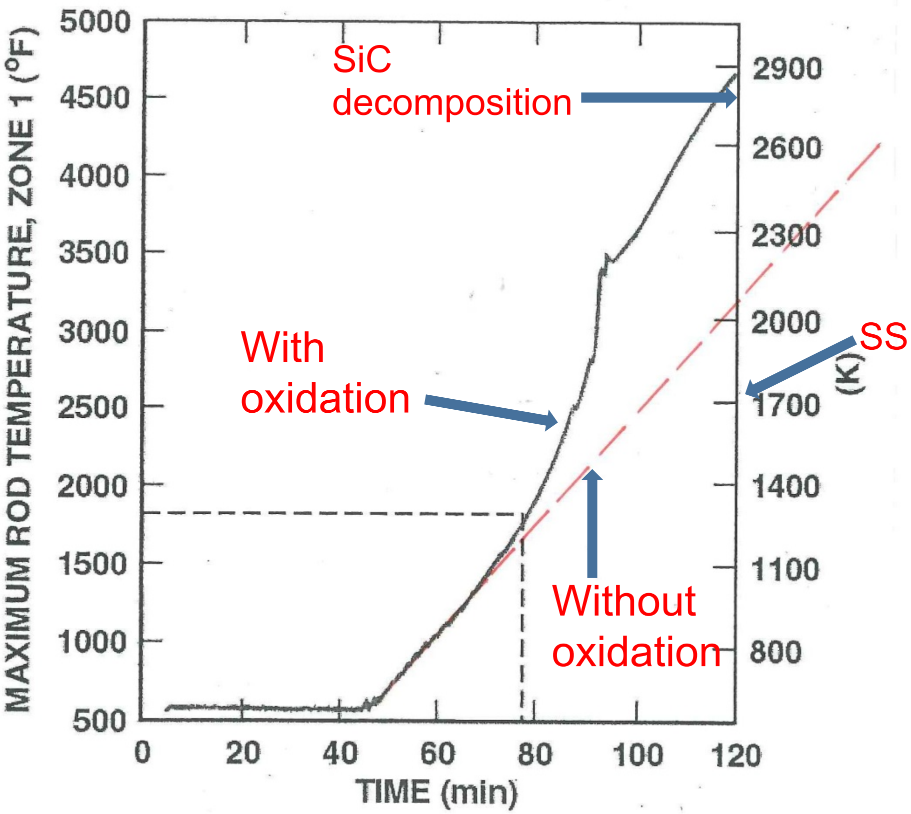
Vessel Depressurization at One-Third Core Height Delays Release of Significant Hydrogen

Grand Gulf Short Term Station Blackout



Noncondensables (primarily H_2) generated in a SA will build-up in containment : primary reason for the existing Hardened Vent Systems (HVS) which allow venting through filters and the SGTS before containment leakage/failure

BWR SBO : Bases for Extrapolation ??



NOTE: the rate of coolant boiloff and subsequent structural heatup is dependent on the core decay heat and the timing when coolant injection is lost (1F1, within 1hr of scram; 1F2, ~68 hrs after scram)

- Timing, conditions at onset of degradation
- Ergo, higher melting core / non-hydrogen generating materials
- Estimate of margins (time)

NOTE

- Without cooling, the core temperatures **WILL** continue to increase
- The SA has **NOT** been stopped
- The decay heat will be transferred to other RPV structures (all stainless steel)
  - Core shroud head and standpipes
  - Plate-dryers
  - Steam piping
  - Core shroud

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for the U.S. Department of Energy

Higher Melting / Lower H_2 Producing Core Components WILL NOT Preclude a SA

- There are no "silver" bullets
  - Without core cooling , the SA will march-on
- Does allow an increase in margin (time) to initiation of core component degradation – although this may be measured in minutes NOT hours
  - If LP coolant injection had been started 2 hrs earlier, may have saved 1F3
  - If H2 generation had been drastically reduced, probably no explosions in 1F1, 1F3 and 1F4
- NEED to consider materials-interaction experiments (reactions [if any] and the kinetics) AND component interactions with steam
  - Could eliminate (or drastically reduce) H_2 generation and the additional chemical energy input
- Besides the fuel/cladding system, MUST consider other components within the core (ergo, a SS control blade with B_4C absorber) and the RPV (SS components)

Summary/Conclusions

- Reactor safety is determined by the system performance, which includes the fuel as well as ECCS and operator actions
- There are a range of accidents that must be considered in evaluation of accident tolerant core materials
- Broad range of accident testing needed to understand fuel/ core materials behavior under accident conditions
  - Currently fuel/cladding basis was determined through a large experimental program
  - Fuel/Clad behavior in high temperature steam environments is one such requirement for LOCA, SBO, and other scenarios
- Criteria, metrics, an evaluation methodology and analysis tools are needed to understand the benefits of new fuel/core materials concepts