IAEA SAFETY ASSESSMENT TRAINING AND EDUCATION PROGRAMME

Design Extension Conditions and Severe Accidents in Light Water Reactors

Joint ICTP-IAEA Nuclear Safety Institute Workshop
ICTP, October 2015

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Objective of this presentation is to provide an compact overview of severe accident progression and to identify supporting research programmes.

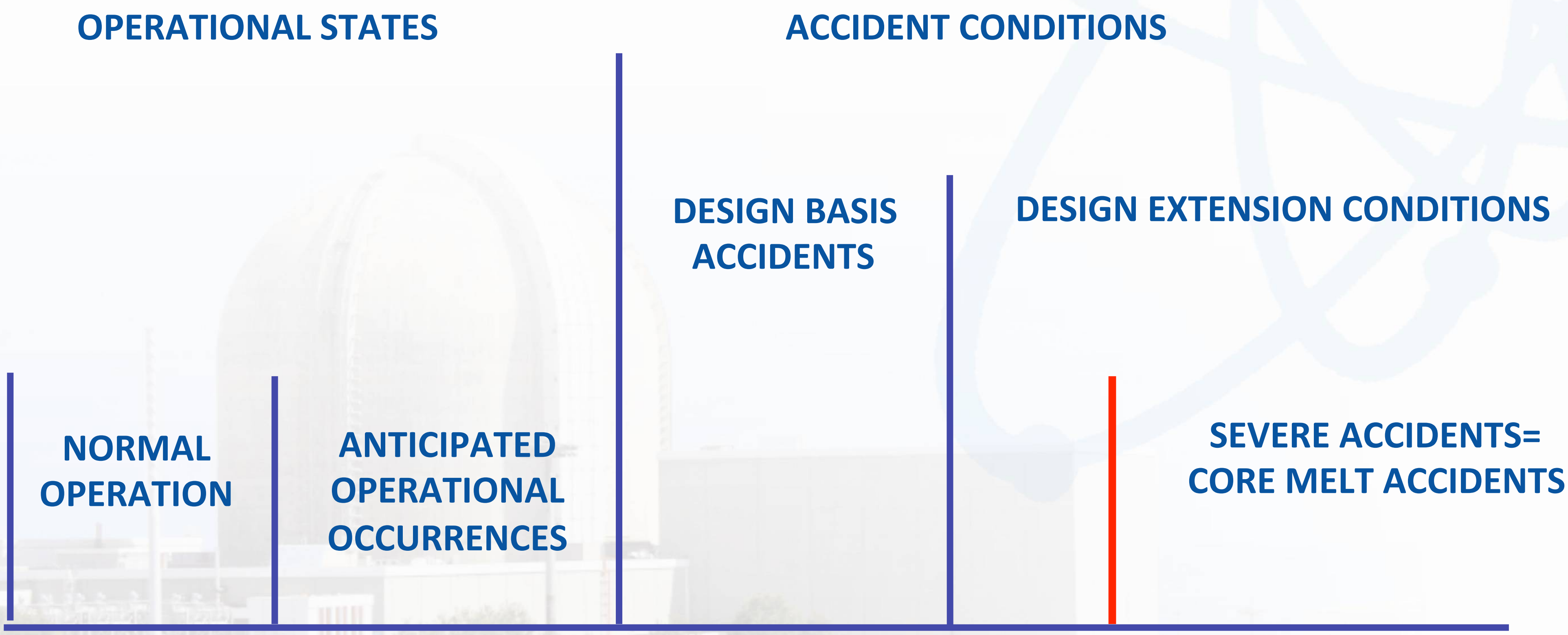
This presentation is based on lectures on severe accidents given in the framework of IAEA SAET programme

Presentation content

- **Objectives and introduction**
- **Phases of severe accident progression**
  - Initial fuel damage
  - Fuel melting and relocation to lower elevations of the RV
  - Failure of the reactor vessel
  - Ex-vessel phase
- **Other severe accident phenomena challenging containment integrity**
  - Creep rupture of reactor coolant system pressure boundary during in-vessel core degradation
  - Hydrogen combustion in containment
  - Steam explosion
- **Design extension condition analyses**

DESIGN EXTENSION CONDITIONS ARE TO BE CONSIDERED IN THE DESIGN OF NEW REACTORS

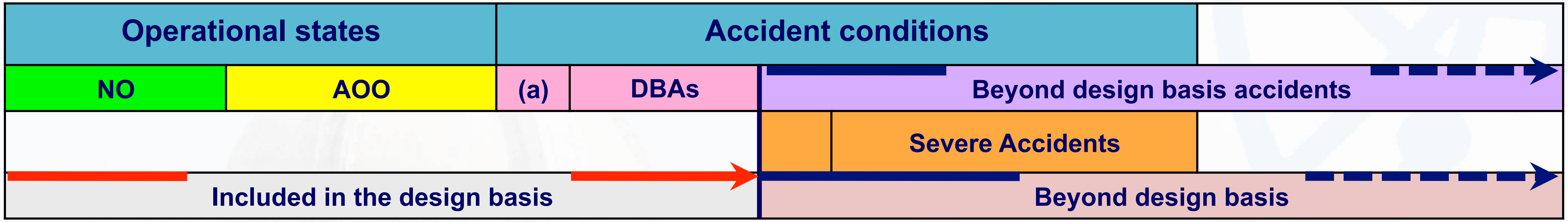
NPP states to be considered in plant design and safety analyses



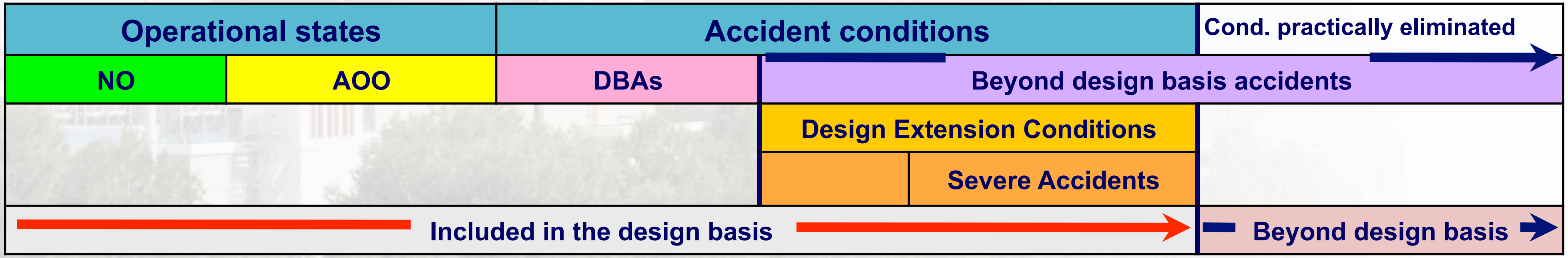
SSR-2/1

IAEA Safety Requirements for Design, SSR 2/1 versus NS-R-1, plant states

NS-R-1, 2000



SSR-2/1, 2012



NPP STATES CONSIDERED IN THE PLANT DESIGN AND ANALAYSES

- **Design basis accident** is an accident causing accident conditions for which a facility is designed in accordance with **established design criteria and conservative methodology**, and for which releases of radioactive material are kept within acceptable limits
- Design extension conditions are accident conditions that are not considered for design basis accidents, but that are **considered in the design process of the facility in accordance with best estimate methodology**, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could **include severe accident conditions**.
- **Accident conditions** mean deviations from normal operation that are less frequent and more severe than anticipated operational occurrences, and which **include design basis accidents and design extension conditions**
- **Severe accident** means accident conditions more severe than a design basis accident and that include **significant core degradation**; for light water reactor it is **synonymous for core melt accidents**

IAEA view on design extension conditions

- All plant states shall be **either considered in the design, or practically eliminated**
- Design extension conditions including **severe accidents are part of the design basis**
- Complex sequences including **multiple failures** shall be considered in the design
- **Safety objectives and acceptance criteria shall be established for all plant states**, including design extension conditions
- **Dedicated measures** shall be implemented to mitigate **design extension conditions including severe accidents**
- **Independence between design provisions at different levels** of defence shall be maintained to the extent possible

Overview of LWR Severe Accident Progression

This part of the presentation will:

Identify major features of severe accident behavior and phenomena that drive radiological release to the environment

- ✓ **CORE DAMAGE PROCESS**
- ✓ **SOURCES OF CHALLENGES TO CONTAINMENT INTEGRITY**

Severe accident research

TMI-2 initiated international research programs in reactor response to 'beyond design basis' (severe) accidents that continues today.

- Initially developed a general understanding of possible consequences of events previously considered incredible
- Later focused on developing data to validate computer models

Accident (SA) is most likely to be initiated by:

- ✓ Loss of primary coolant (LOCA)
- ✓ Increase the power output in the core or that restrict heat removal from the core without any loss in primary coolant
- ✓ External events
- ✓ In addition, major engineered safety features are assumed to fail over a longer period of time, causing core uncovery.
- ✓ Unlike a design-basis accident (DBA) a SA is characterized by the start of cladding failure and corresponds to a temperature threshold of 1100-1200°C at core exit (CET).

KEY PHENOMENA IN SEVERE ACCIDENTS - OVERVIEW

THERMO - MECHANICS

CORE DEGRADATION

MELT RELOCATION AND SLUMP

CORE DEBRIS IN LOWER HEAD

RPV FAILURE

← LATE PHASE MELT PROGRESSION →

← NATURAL CIRCULATION →

← FUEL COOLANT INTERACTIONS →

← IN VESSEL MELT RETENTION →

FAILURE MODE

← LATE IN-VESSEL FP RELEASAE →

← FP RETENSION, RESUSPENTION AND REVAPORIZATION →

RADIO - CHEMISTRY

FISSION PRODUCTS RLEASE

FISSION PRODUCT TRANSPORT

FISSION PRODUCT RELEASE INTO CONTAINMENT

Typically there are three phases recognized in severe accident in vessel progression

- **Phase 1: Initial Fuel Damage**
  - Fuel rod heating to ~1400^{0}COxidation of fuel cladding (acceleration in heatup)
  - Control rod melting (Ag-In-Cd alloy melting temperature ~800^{0}C)
- **Phase 2: Core Melting & Relocation**
  - Clad failure and material interactions cause partial liquefaction of fuel and formation of particulate debris
  - Melt / debris relocates downward
  - Debris accumulates on lower core support structures and in the lower head
- **Phase 3: Reactor Vessel Lower Head Fails**
  - Discharge of core debris into containment
  - Core debris interactions with containment structures

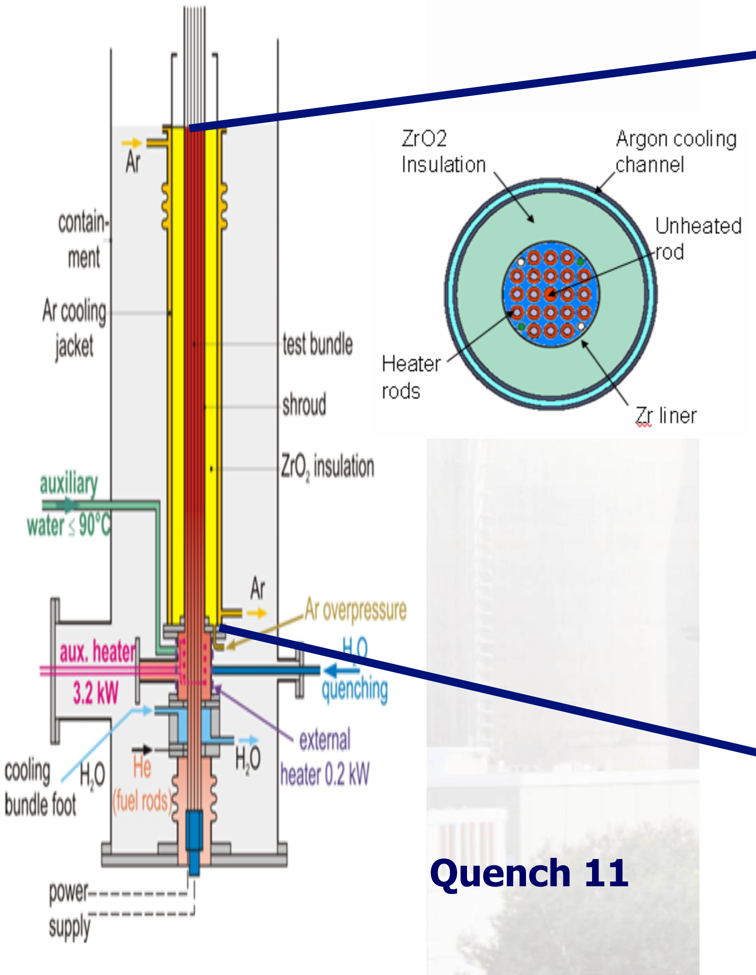
Severe Accident Phase I

Initial Fuel Damage

Phase I: Initial Fuel Damage

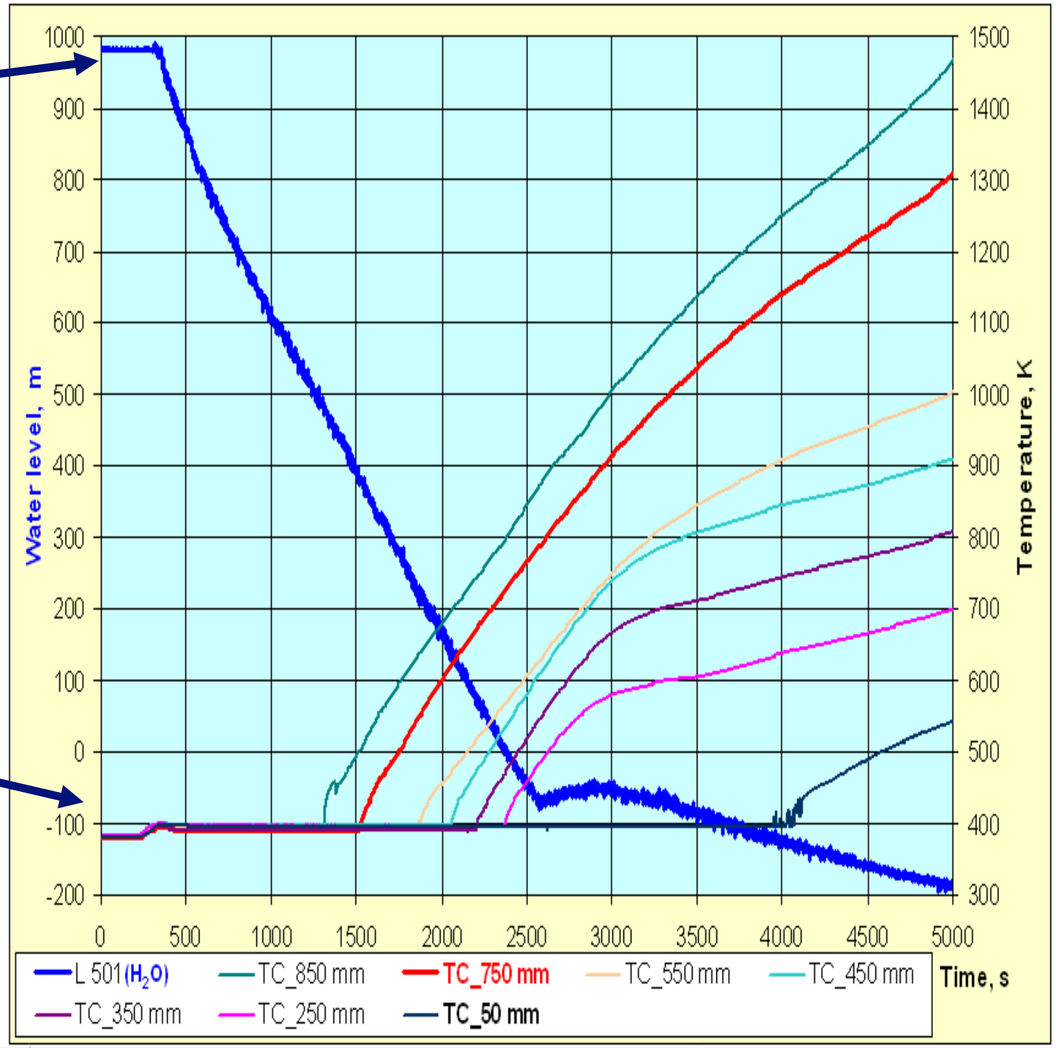
- Starts by uncovering the upper portion of the core
- Due to the decay heat the fuel/clad temperatures initially increase at a rate of approx. 0.1 to 0.4 ^{0}C/sec.
- When surface temperatures exceed approximately 1000^{0}C, an exothermic reaction between Zr and steam adds more energy to the fuel that decay heat, increasing the rate at which fuel/clad temperatures rise to approximately 1^{0}C/sec.
- Above 1400^{0}C the kinetics of the Zr-steam increases extremely driving clad surface heat-up rates above 5^{0}C/sec and temperatures above melting point of Zircaloy
- The oxidation of the metallic Zr leads to production of hydrogen

Accident Progression - Phase 1: Core uncovery initiates core heatup



Boil-off (QU-11)

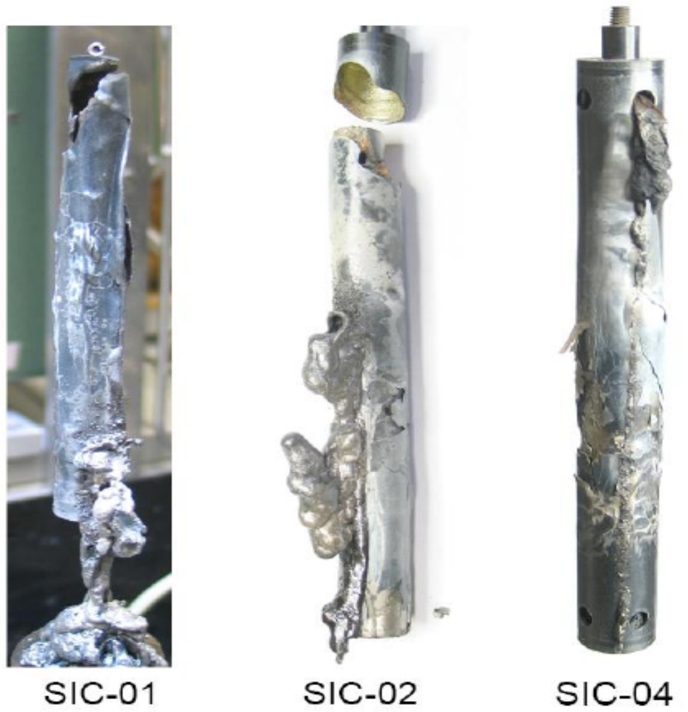
Quench 11



J. Stuckert, FZK/IMF-III 12 th QUENCH Workshop. October 24-26. 2006 4

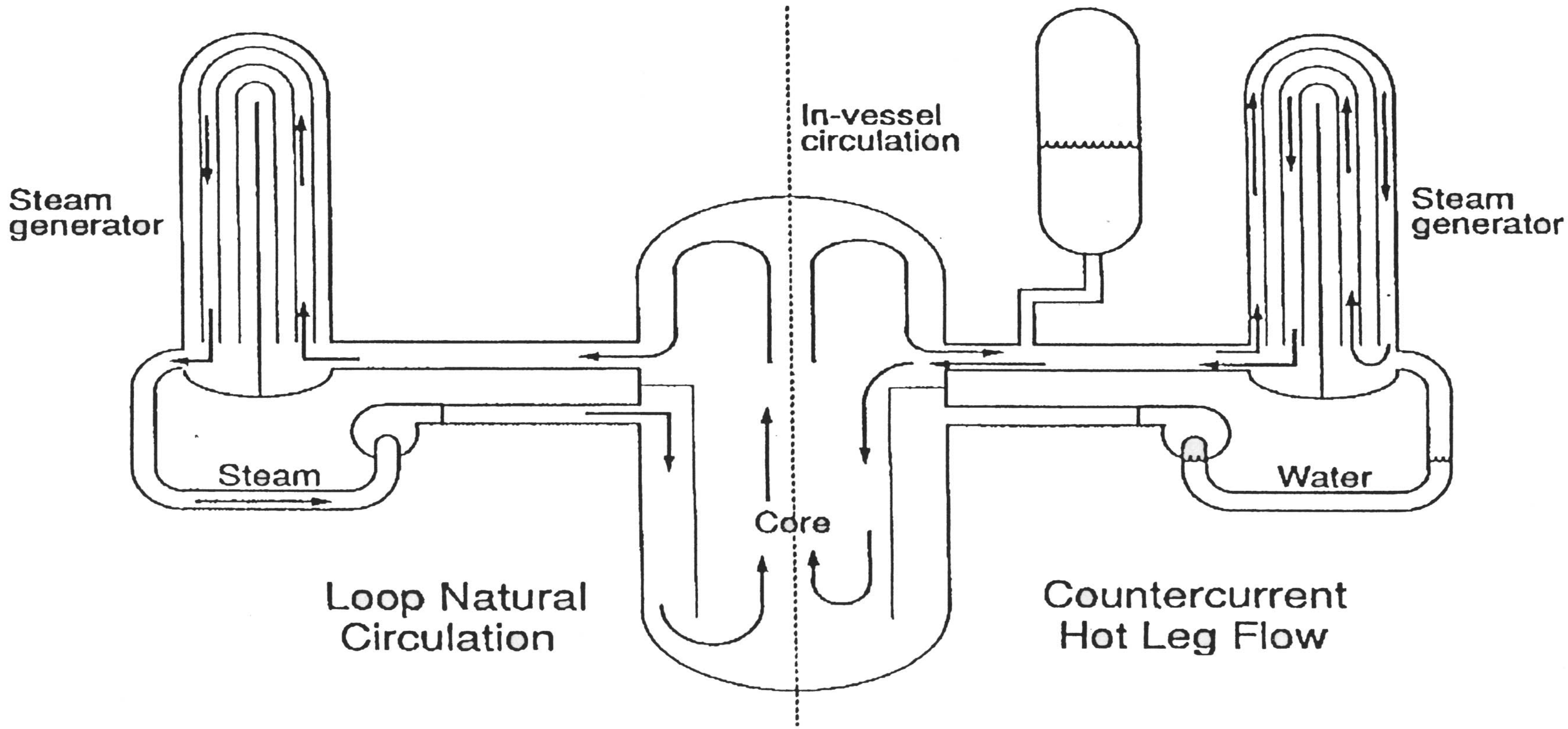
Accident Progression - Phase 1: Zircaloy oxidation dominates early response

- Major features: Initiation of clad oxidation & control rod melting
  - Oxidation: Reaction of exposed metallic surfaces (Zirconium clad) to steam
    - ✓ "Run-away" exothermic oxidation at temperatures greater than ~1200^0**C** Oxidation heat generation rate limited primarily by
      - Availability of steam
      - Availability of unoxidized material
    - ✓ Oxidation terminated by
      - Relocation of metallic Zircaloy to colder region
      - Complete oxidation of Zircaloy
  - Control rod melting
    - ✓ Ag-In-Cd alloy melting temperature ~ 800^0**C**



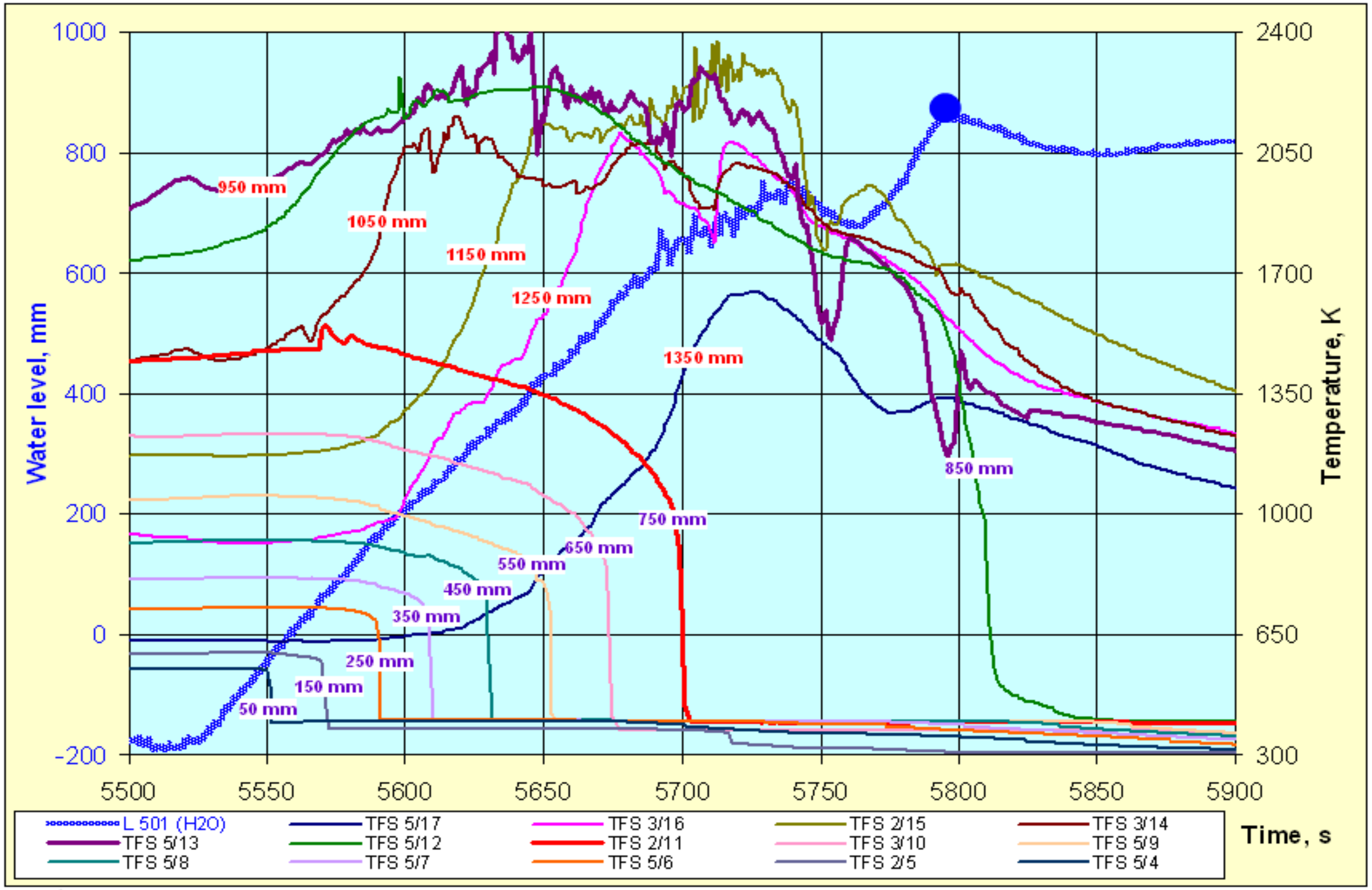
SIC-Tests des FzK, M. Steinbrück, CSARP 2007

Natural circulation helps remove heat from the core but accelerates failure of piping and other RCS components



Heating can be accelerated by reflood (1/2)

Quench-11



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12 th QUENCH Workshop. October 24-26. 2006

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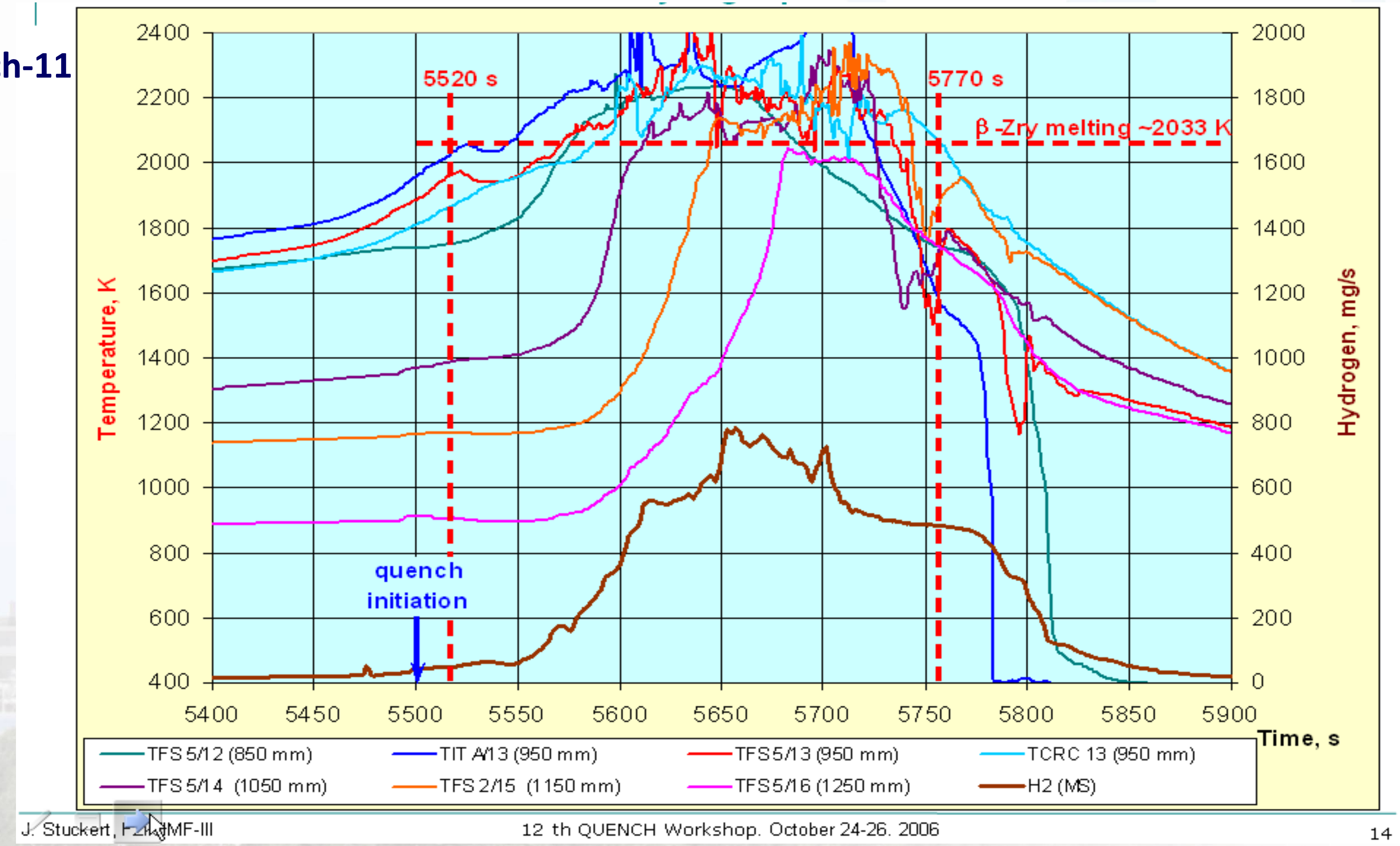
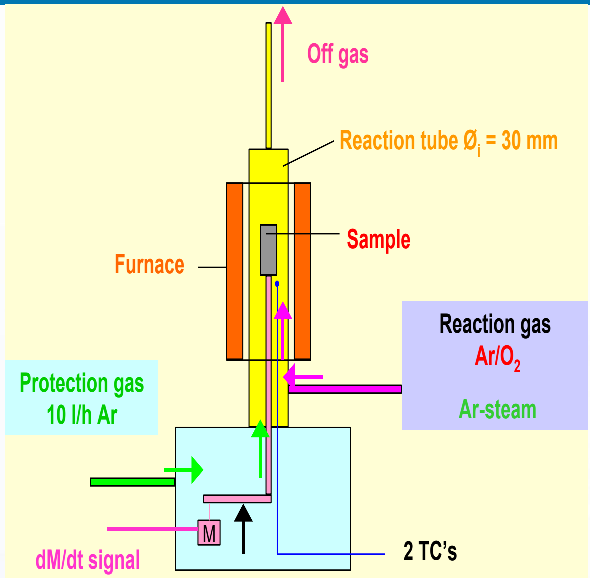
Heating can be accelerated by reflood (2/2)

Quench-11



Separate effects facilities at KIT

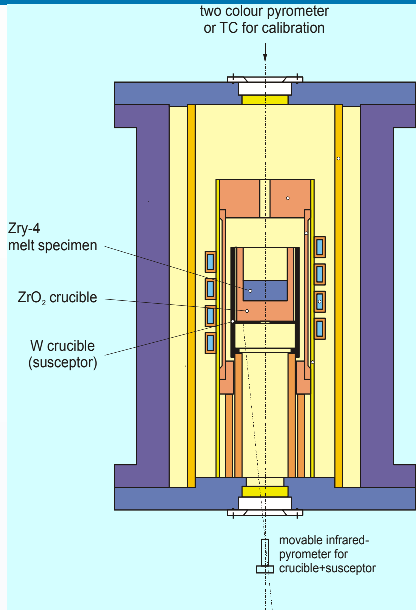


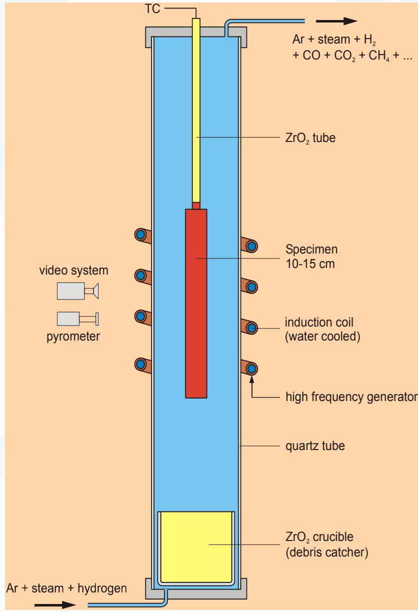
TG

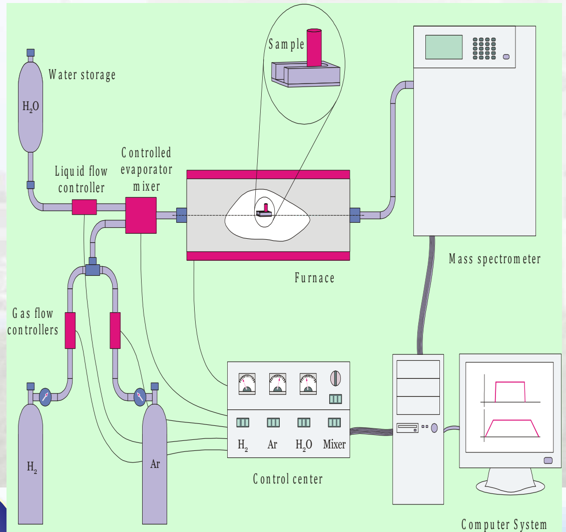
1300 °C

oxidising, reducing atmosphere ($H2O$<0.2 bar)

Specimens: << - 2 cm







BOX Facility

1700 °C

oxidising, reducing atmosphere (incl. steam)

specimens: 1-2 cm

LAVA Furnace

2300 °C

inert, reducing atmosphere

specimens: 1-2 cm

QUENCH Rig

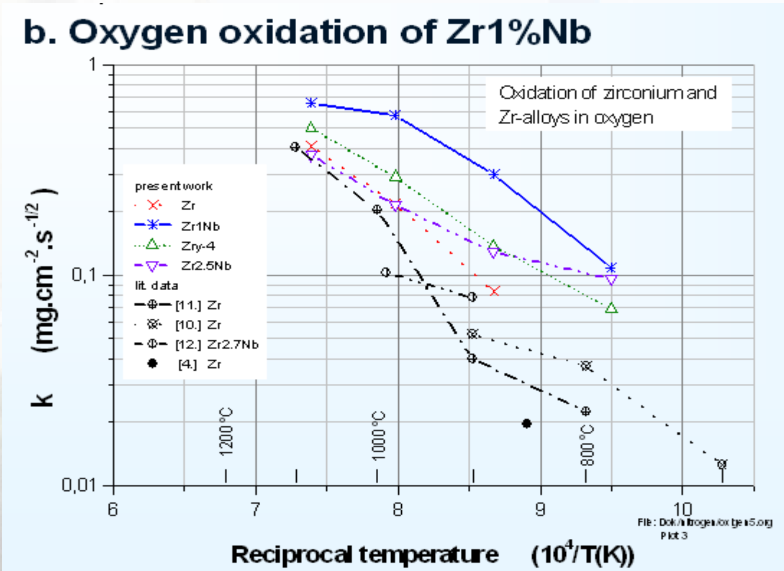
2000 °C

oxidising, reducing atmosphere (incl. steam)

specimens: 15 cm

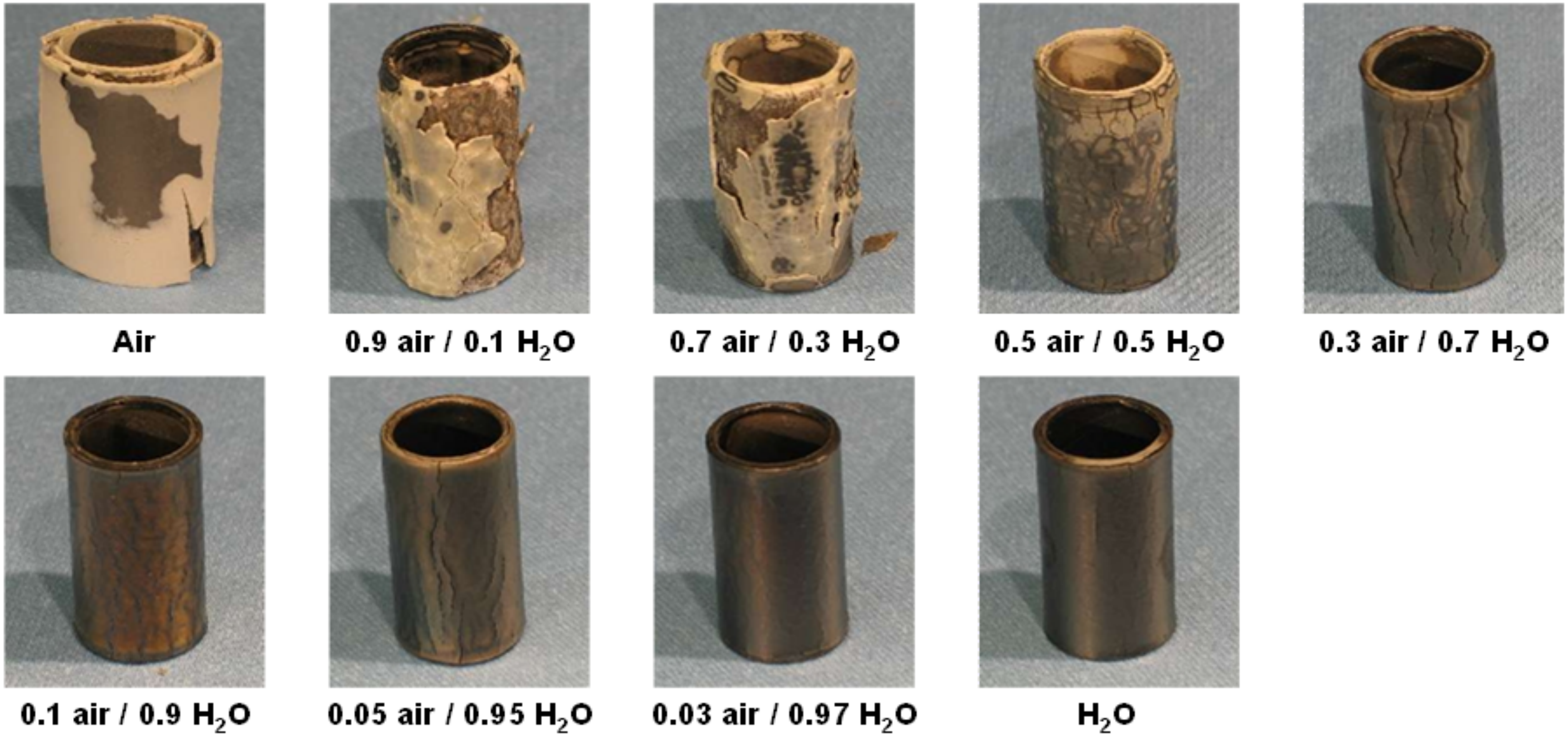
Comparison of Zr and Zr-Nb Oxidation

Zr1%Nb in air	Zr1%Nb in O_2	Zircaloy-4 in O_2	Zr1%Nb in N_2	Zircaloy-4 in N_2
880 °C, 212 s	880 °C, 227 s	880 °C, 160 s	980 °C, 8 h	1080 °C, 4 h
ECR = 7,32%	ECR = 8,28%	ECR = 2,31%	ECR = 1,21%	ECR = 1,55%



Oxidation in steam/air mixtures

Oxidation of Zircaloy-4 in mixed air-steam atmospheres



Post-test appearance of specimens oxidised 1 h at 1200 °C

Effects of Phase 1 Features on Accident Progression

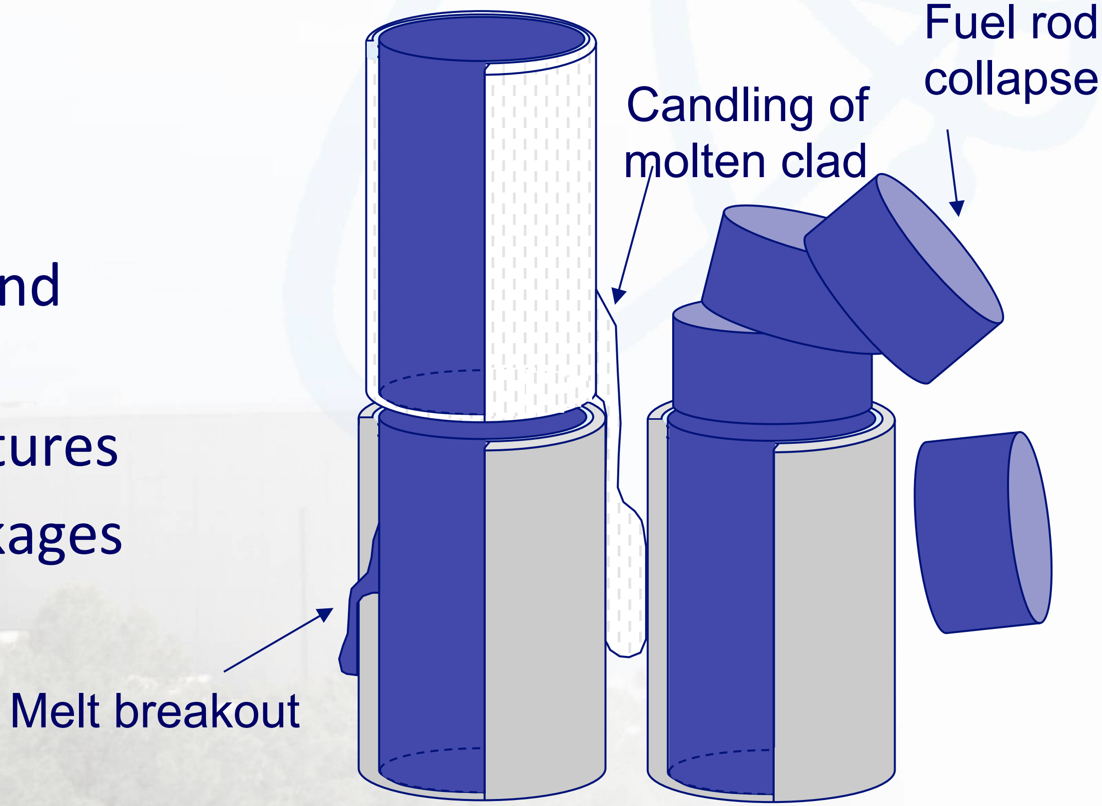
- Heat of reaction causes significant increase in fuel assembly heat up rate
- Melting of cladding initiated
- Potential melting a downward "candling" of molten control rod & clad material
  - Refreezes at lower elevation, reducing coolant flow area
- Large amounts of hydrogen generated > Major source of hydrogen to containment

$$Zr + 2H_2O \rightarrow 2H_2 + ZrO_2$$

Fuel melting and relocation to lower elevations of the RV

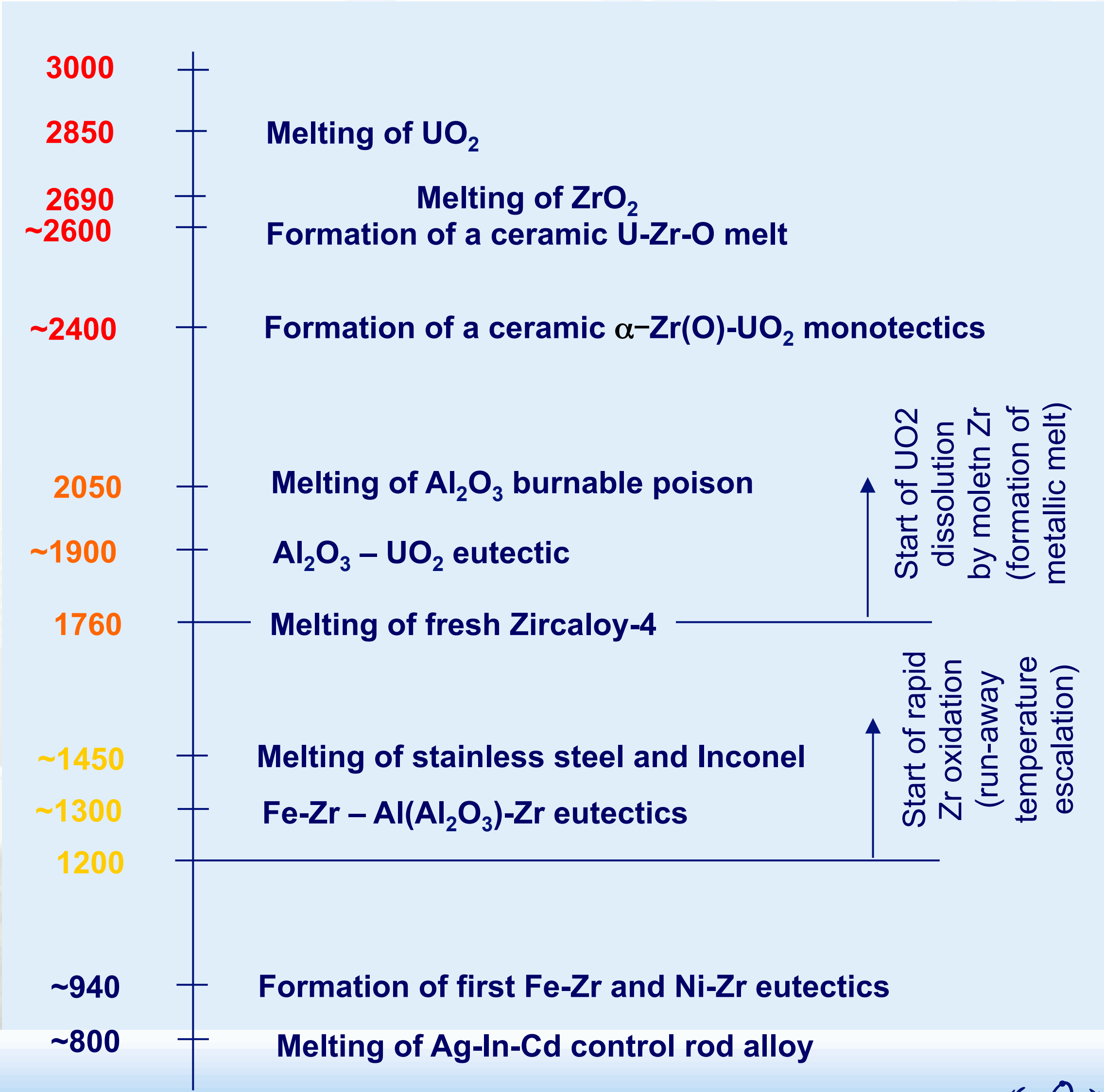
Accident Progression Phase 2

- Major feature: Fuel melting and relocation to lower elevations of the RV
  - Major changes in core geometry
  - Separation of metallic and ceramic materials
  - Wide range of temperatures
  - Formation of local blockages



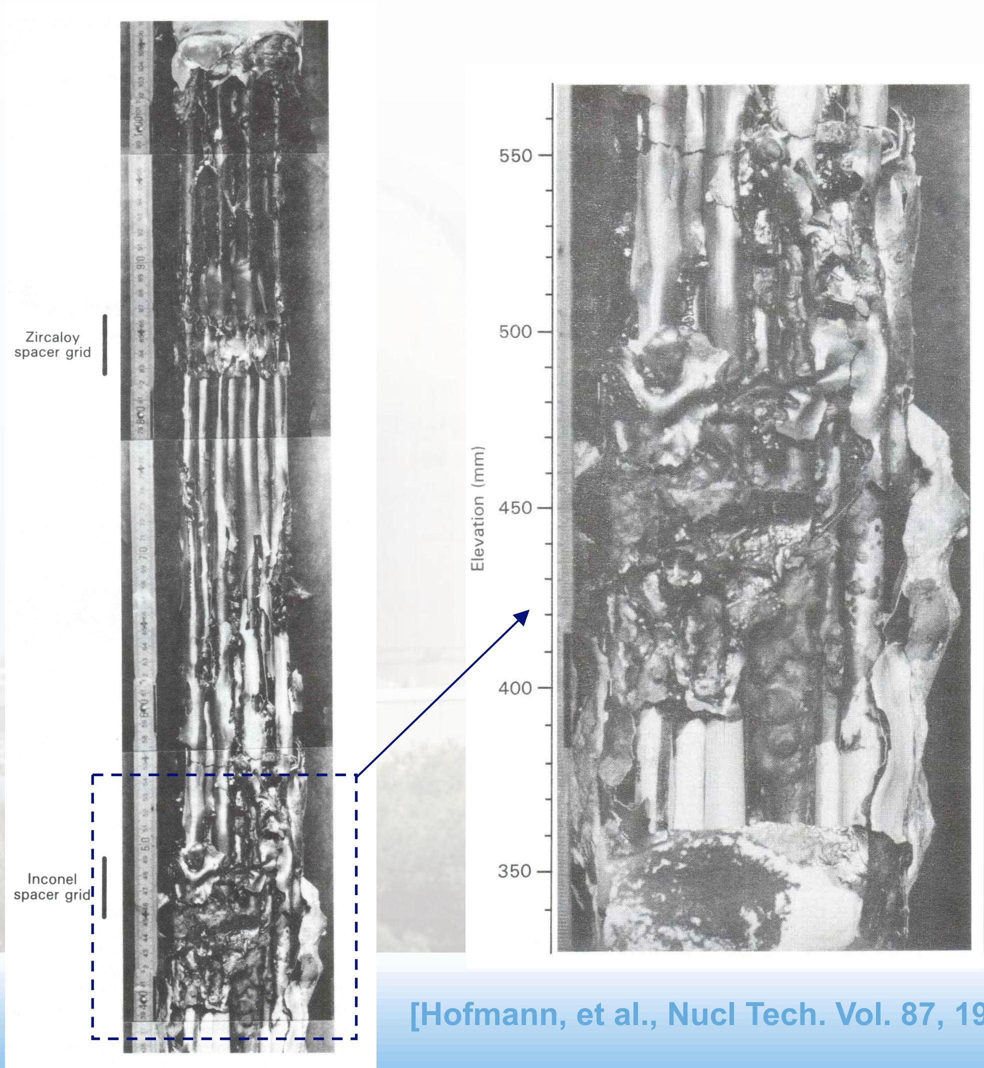
Effects of Increasing Temperature on Core Materials

- Core 'melting' and relocation affected by eutectic interactions among various core materials



- Chemical interactions between core materials can result in liquefaction of the core at temperatures well below melting points of fuel and cladding
  - Extended irradiation may also result in fuel slumping well below its melting point
- Loss of the original core geometry can occur over a wide range of temperatures
  - Ceramic fuels will retain integrity well after failure of metallic structures and materials

Core Material Response to High Temperatures

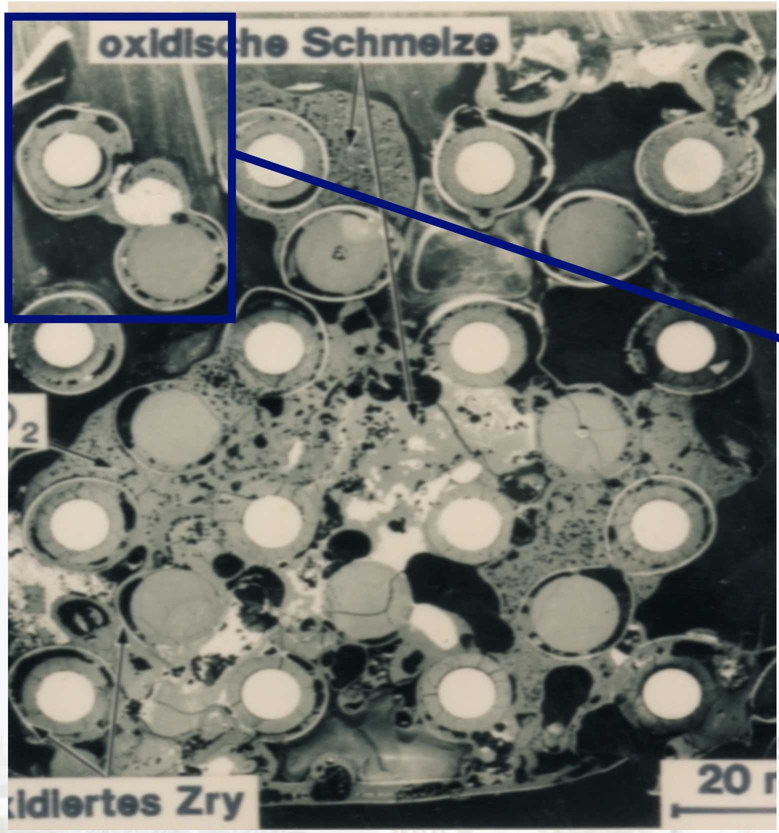


[Hofmann, et al., Nucl Tech. Vol. 87, 1989]

- In-pile fuel bundle degradation experiments provide the basis for severe accident simulation codes
  - ACRR (Sandia – USA)
  - PBF, LOFT (Idaho – USA)
  - CORA (KfK, Germany)
  - FLHT (PNL, USA)
  - PHEBUS (France)

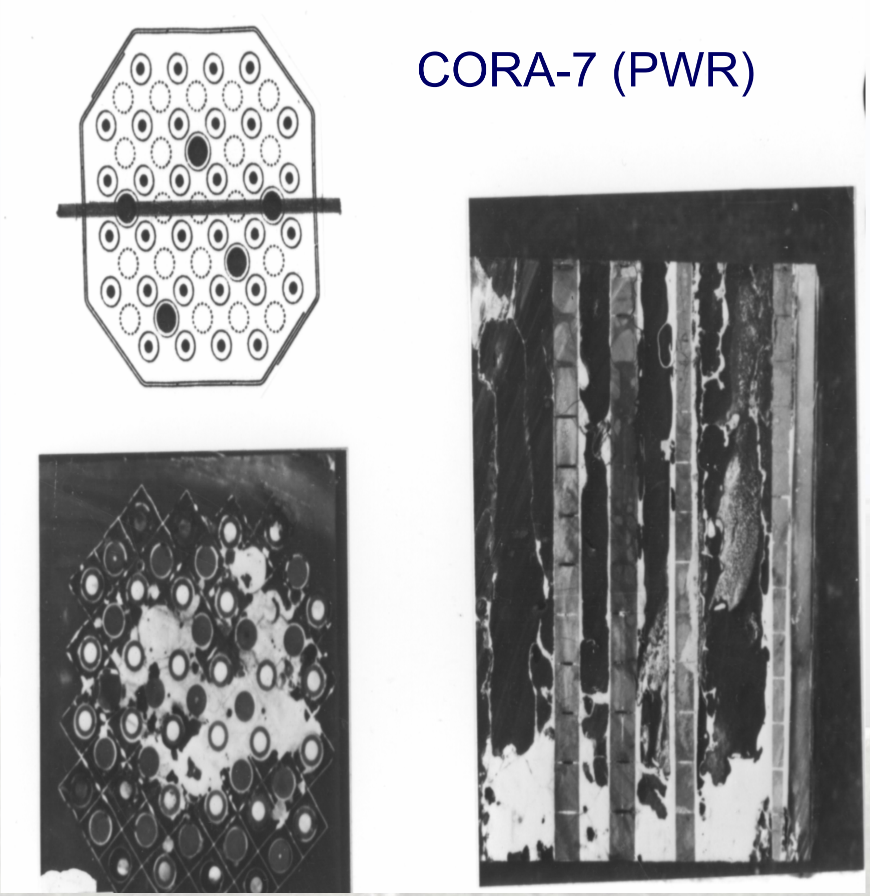
- Useful literature reviews:
  - Hobbins, et al., Nucl. Tech., 95, Sept. 1991.
  - Hofmann, J. Nucl Mat, 270, 1999.

Molten Zr can liquefy UO_2 - CORA-5

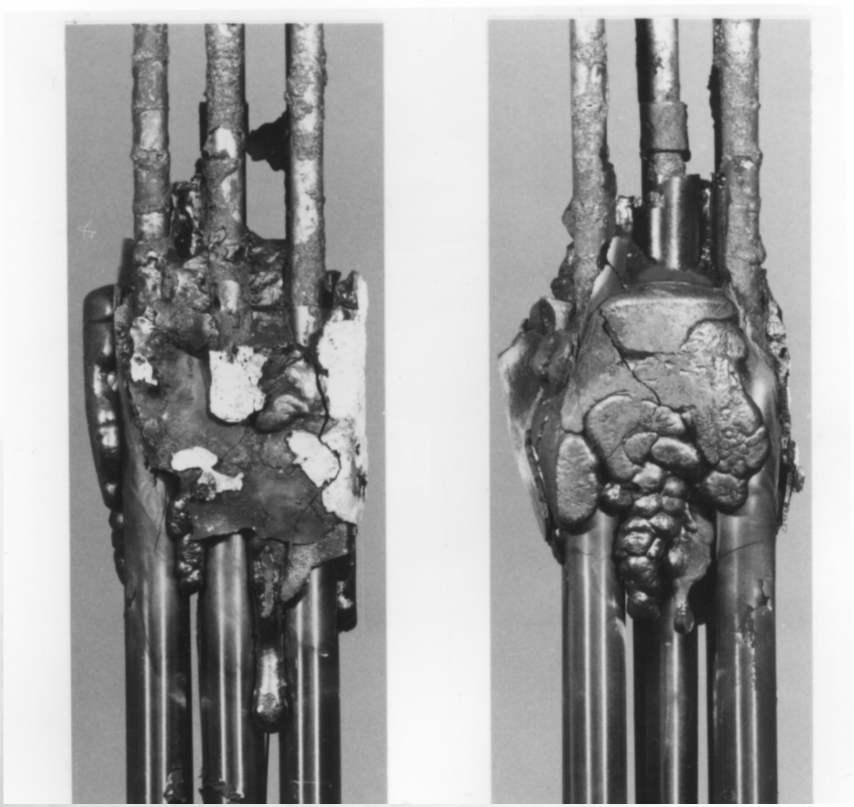


Metallic U-O-Zr blockages can form at temperatures above 2000 K

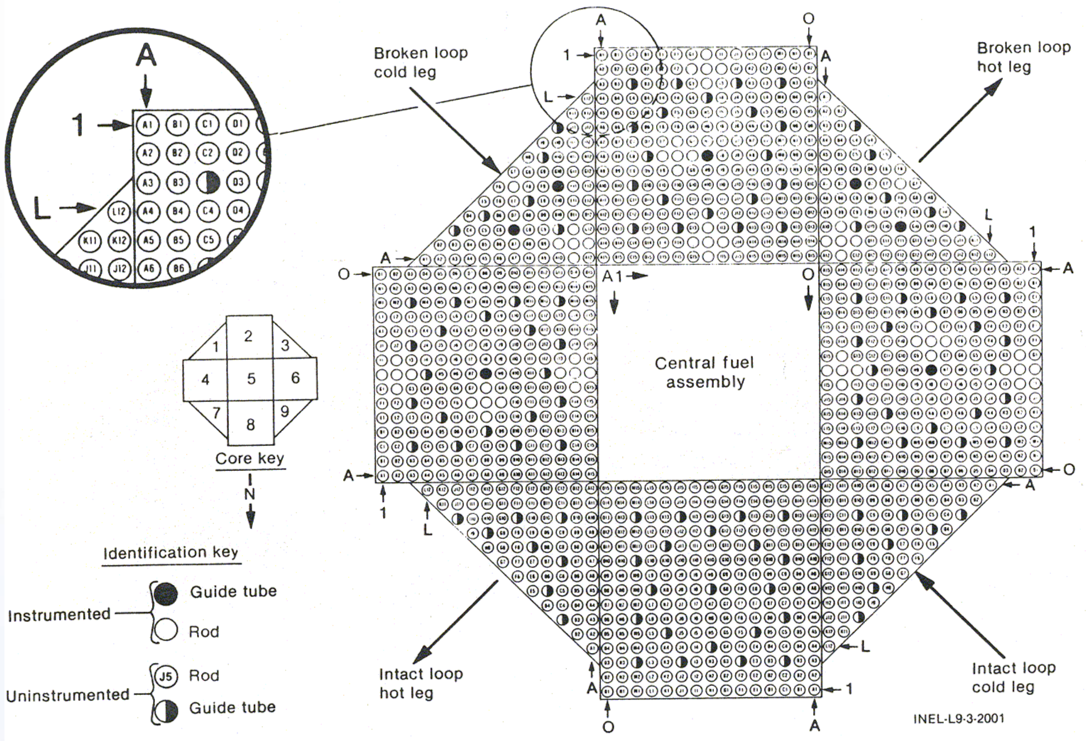
CORA-7 (PWR)



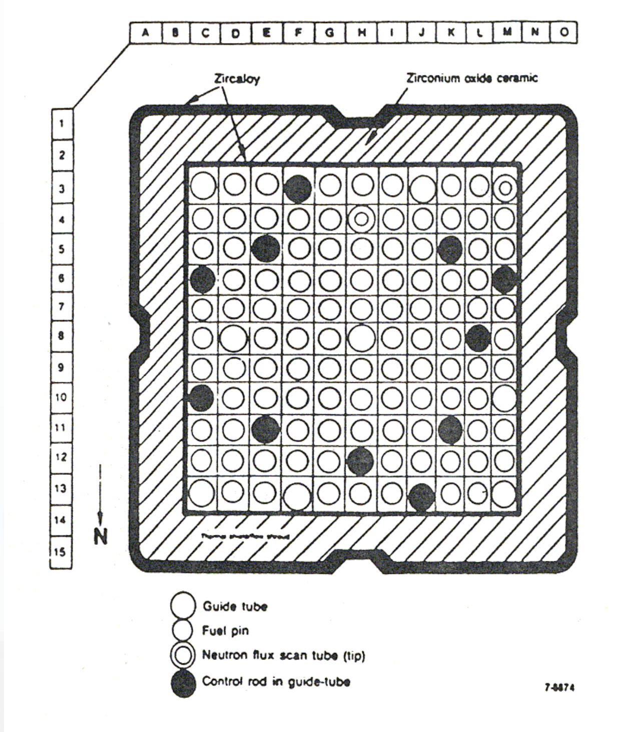
ESBU-2



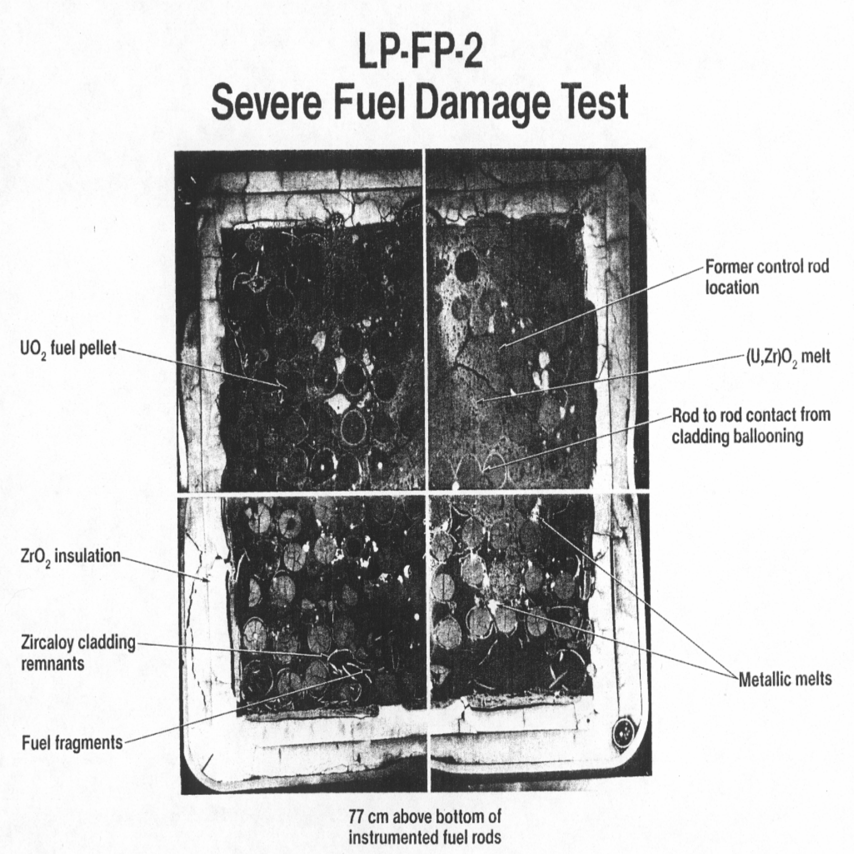
LOFT LP-FP Experiments (Idaho NL, USA)

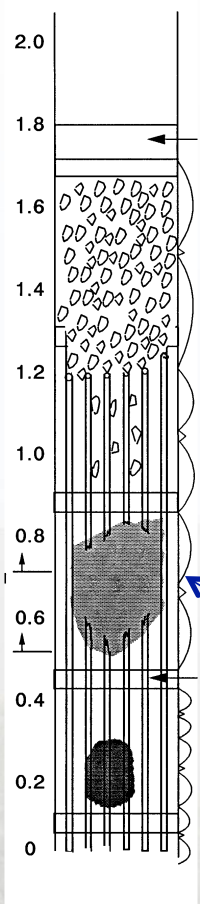


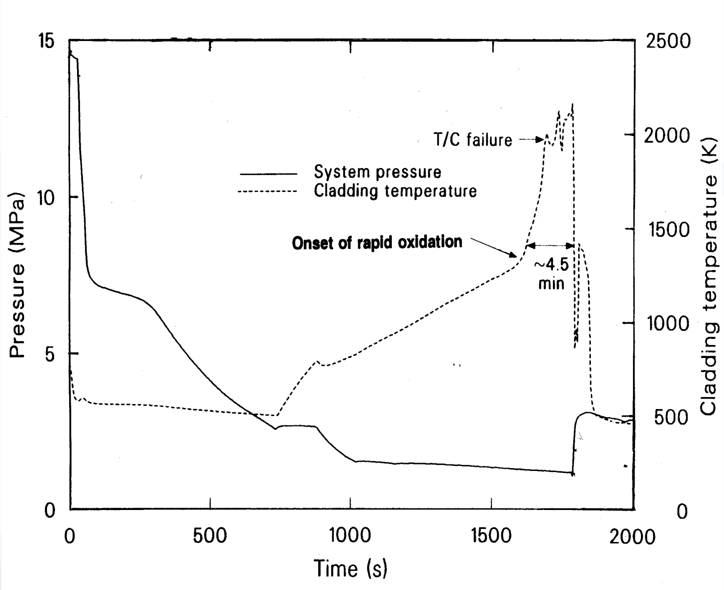
The LOFT core configuration showing the fuel rod and guide tube locations, as well as the locations of the cold and hot legs.



- **Medium-length in-reactor series, 2 tests of large radial scale, with ~120 rods including PWR control rods; OECD international project**
- **Degradation through to large melt pool formation, fission product release**
- **LP-FP-2 widely analysed; unique scale, f.p. release, late phase phenomena**



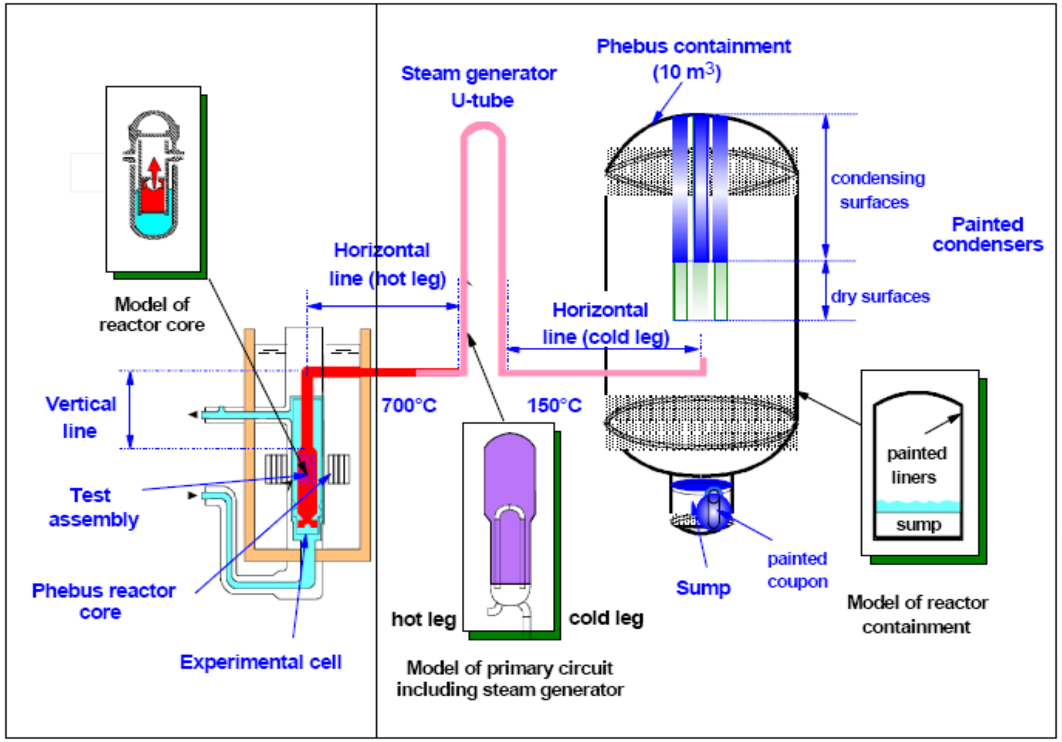




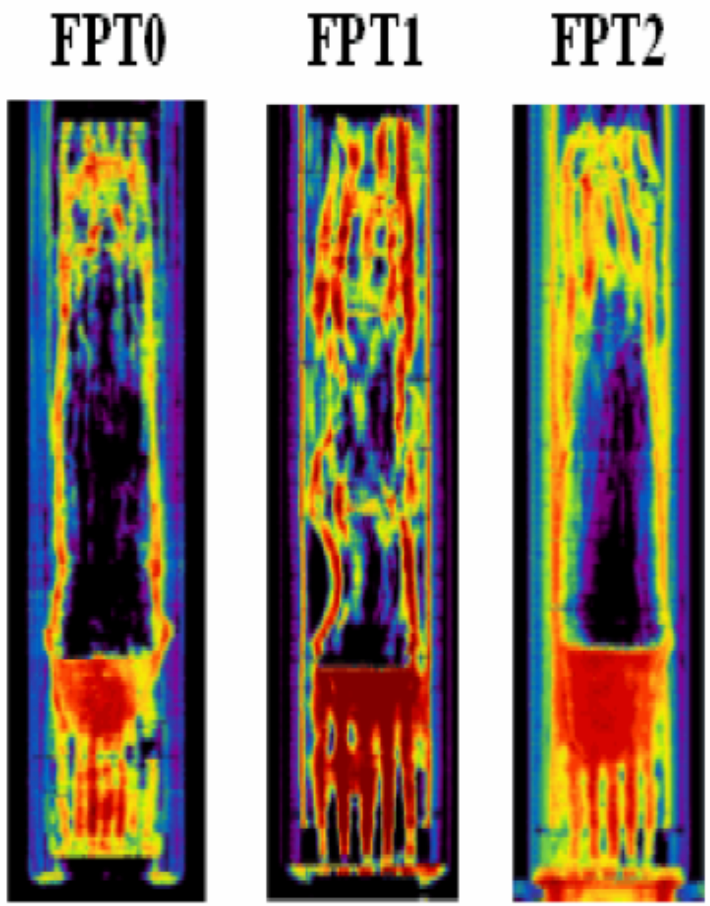
Ceramic blockage
(U-Zr)-O_2

Metallic blockage
(U-Zr) - Ag

Phebus Facility (IRSN



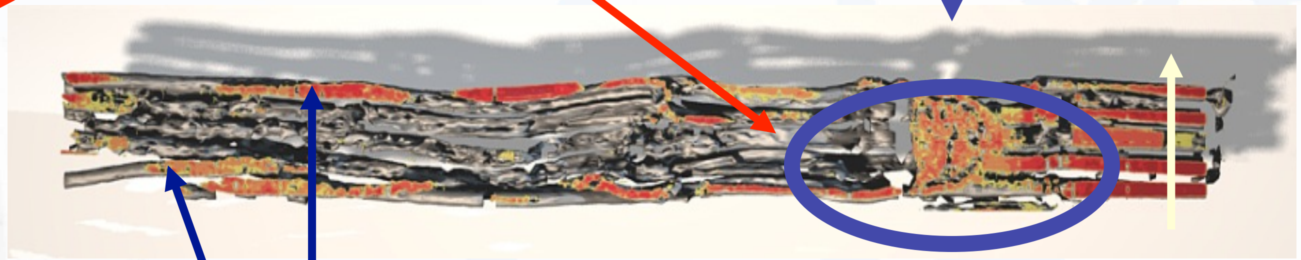
Real fuel elements w/o and w FP used since 1993 to assess core melting and FP release and behaviour



Fuel rod bundle degradation (PHEBUS FPT1)

Cavity

Molten pool formation

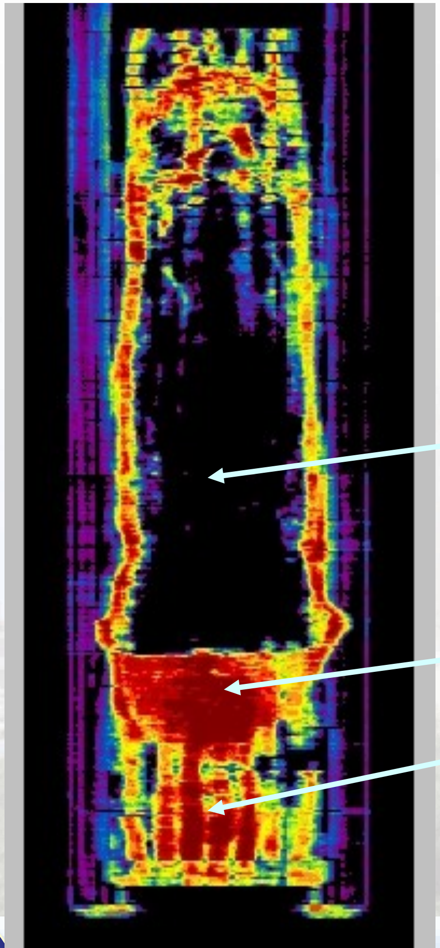


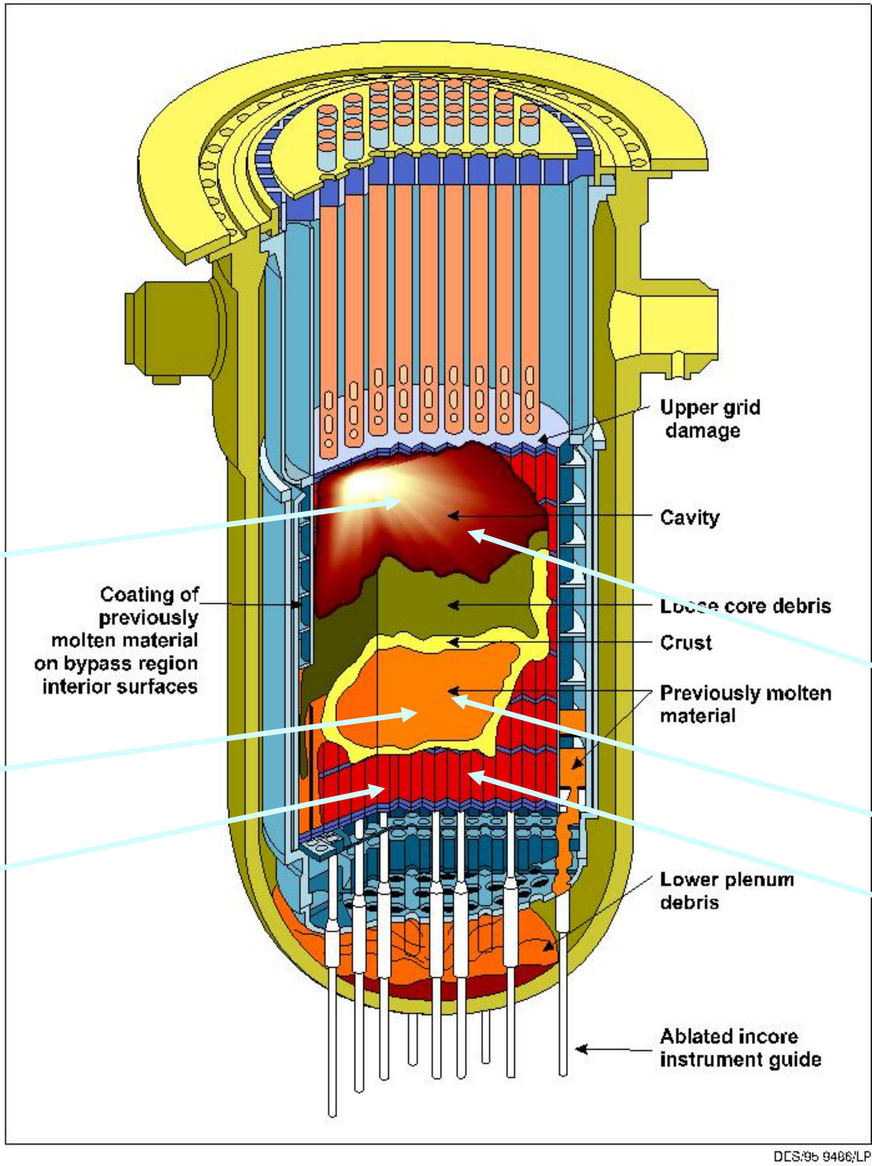
Standing degraded fuel rods at upper elevation

Quite intact fuel rods at lower elevation

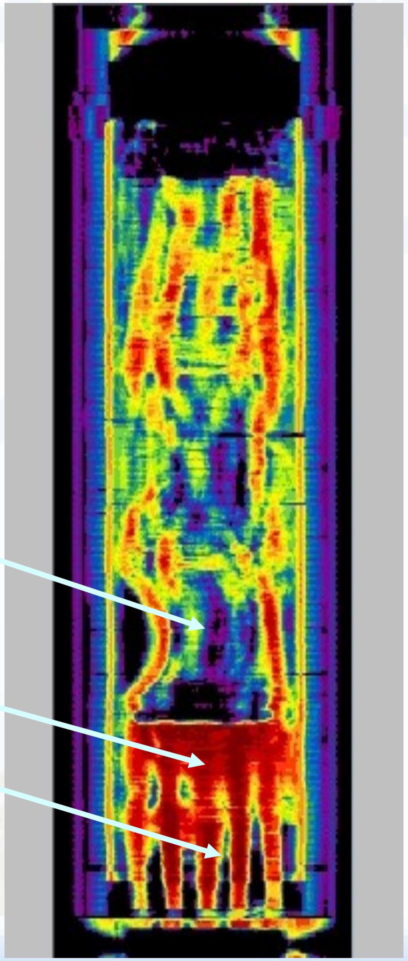
Phébus and TMI: Cavity, molten pool, FE stubs

PHEBUS FPT0 test





PHEBUS FPT1 test



General overview of SA progression based on experiments and analyses

- T_{peak} > 1500K
  - Onset (likely) of rapid oxidation and temperature excursion
  - Failure of Inconel/SS spacer grids, typical BWR/PWR control elements due to material interactions
- T_{peak} > 2000K
  - Formation of U-Zr-O rivulets and droplets (steam-limited, rapid initial heatup
- T_{peak} > 2600-2800K
  - Liquefaction and slumping of [U-Zr]-O_2

General overview of SA progression with addition of water to the core

- T_{peak} < 1500K (When flooding initiated)
  - Rapid cooling
- 1500 K < T_{peak} < 2000 K (When flooding initiated)
  - Onset of rapid oxidation and temperature excursion
  - Limited change in fuel rod geometry
- 2000 K < T_{peak} < 2200 K (When flooding initiated)
  - Accelerated oxidation and temperature excursion
  - Oxidation of U-Zr-O droplets/rivulets
  - Possible accelerated liquefaction and slumping of $(U\text{-}Zr)\text{-}O_2$

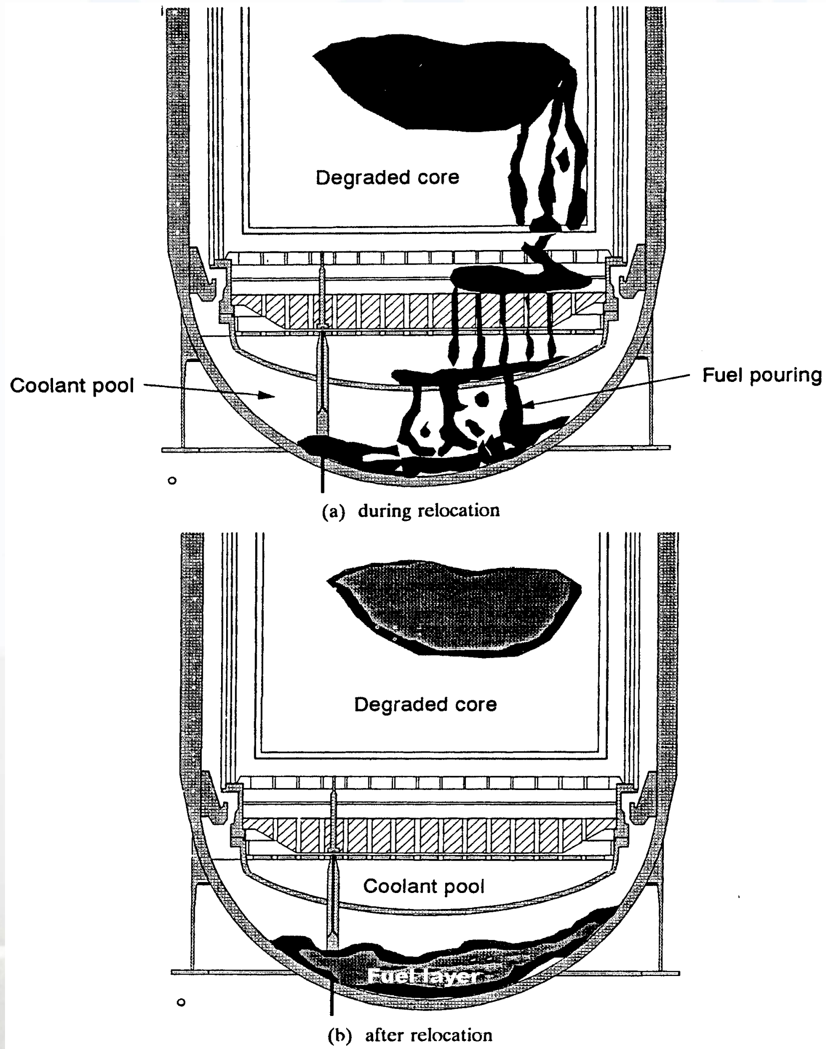
Accident Progression Phase 2 – summary

- Major changes in core geometry
- Separation of metallic and ceramic materials
- Wide range of temperatures
- Formation of local blockages
- Experiments and analysis demonstrate
    - ✓ Coolability of LWR bundles for temperatures well above 1500 K
- Negative impact of Zircaloy oxidation and melting during water addition
    - ✓ Accelerated heating and increased likelihood for fuel melting

Melt relocation into the lower plenum

Major uncertainties include:

- Configuration of relocating debris/melt
- Temperature of relocating material
- Crust formation and heat transfer mechanisms on lower head surface



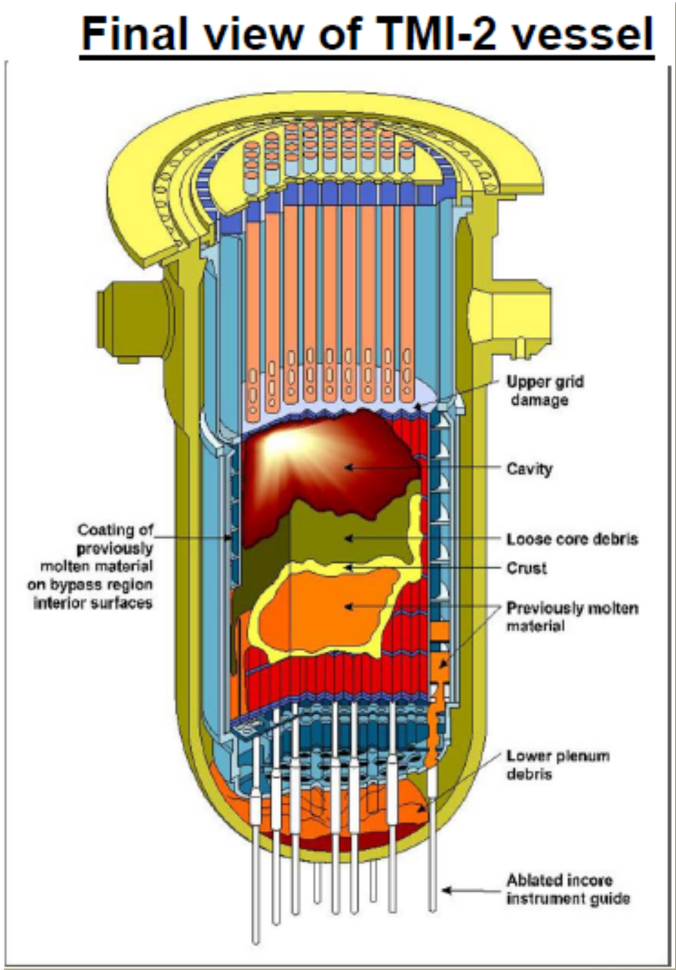
(a) during relocation
(b) after relocation

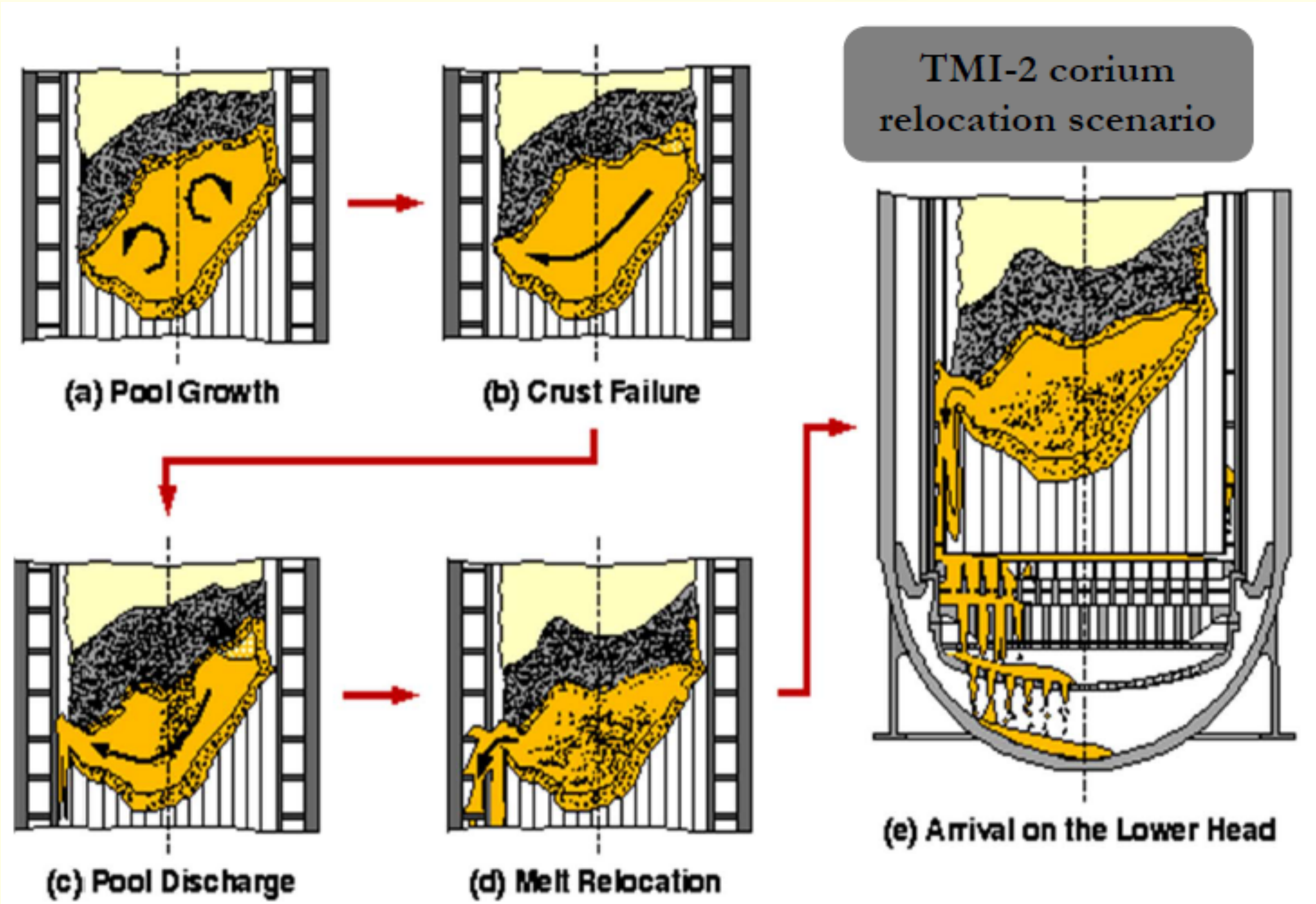
Failure of the reactor vessel

Accident Progression - Phase 3

- **Major features: Molten Debris Attacks Lower Head**
  - Debris coolability in lower head remains a major area of research
  - Lower head penetrations important for some reactor vessels
  - TMI-2 lower head did not fail in spite of molten pour of a considerable mass of material
    - ✓ Molten material submerged in pool of water
    - ✓ Crust formation against inner surface of lower head wall provided an insulating layer that limited heat transfer

TMI-2 corium relocation scenario



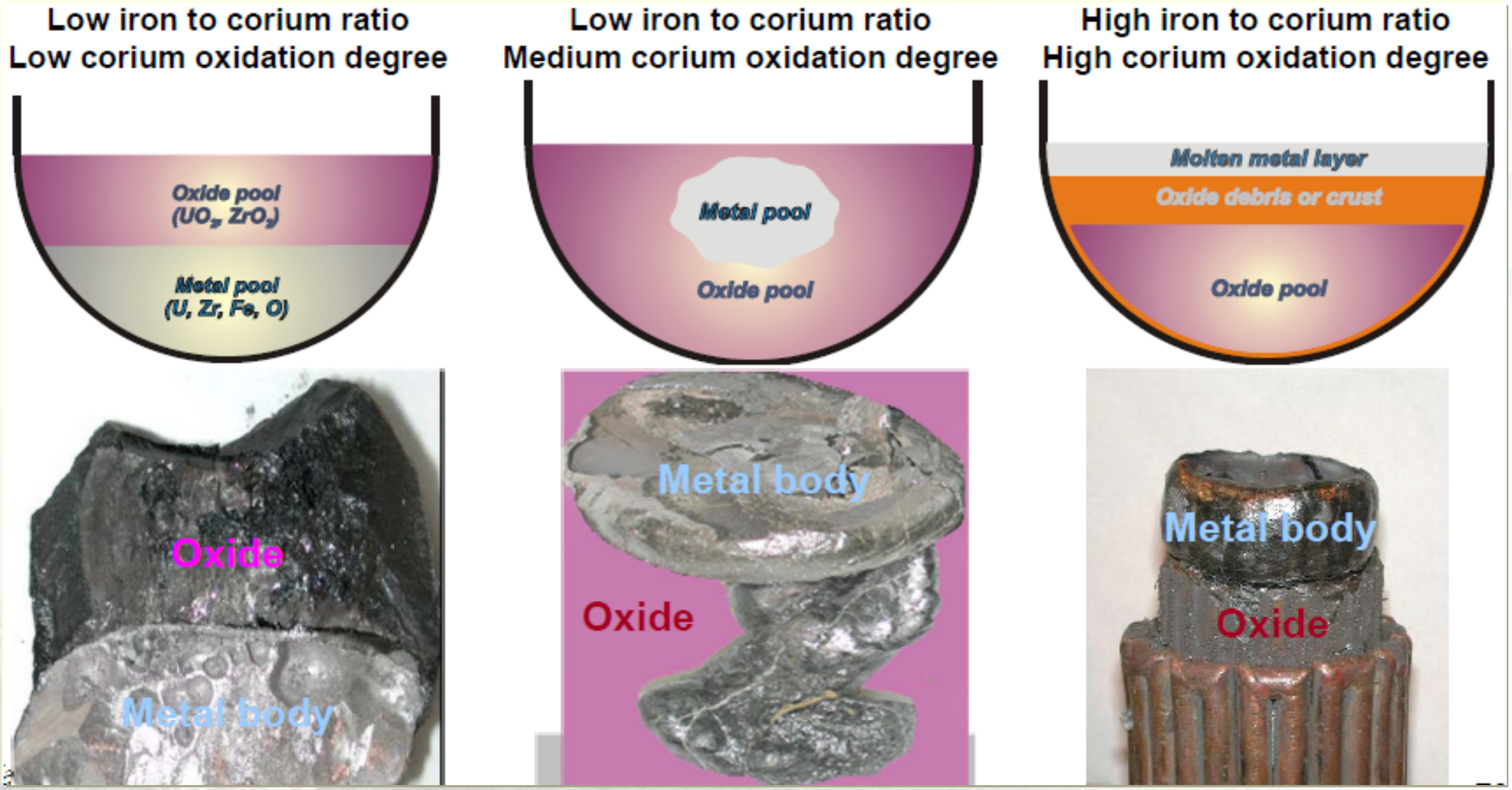


Major Lower Head Failure Research Projects

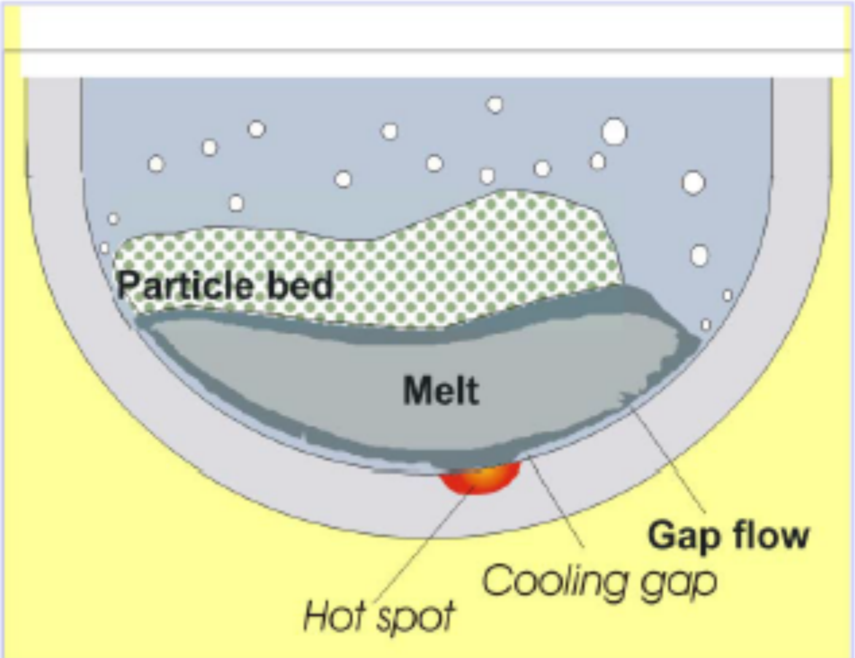
In-vessel Melt Quenching	Heat Transfer from a Molten Pool	Gap Cooling Mechanism	RPV Failure Mechanisms
FARO (JRC – Ispra, EC) ALPHA (JAERI, Japan) QUENCH (KfK, Germany)	RASPLAV (RRC-KI, Russia) COPO2 (Finland) ACOPO (UCSB, USA)	ALPHA (JAERI, Japan) EPRI/ FAI (USA) RPV Programme (TUM Siemens, Germany)	LHF (SNL, USA) FOREVER (KTH, Sweden) CORVIS (PSI, Switzerland)

Results of MASCA project:

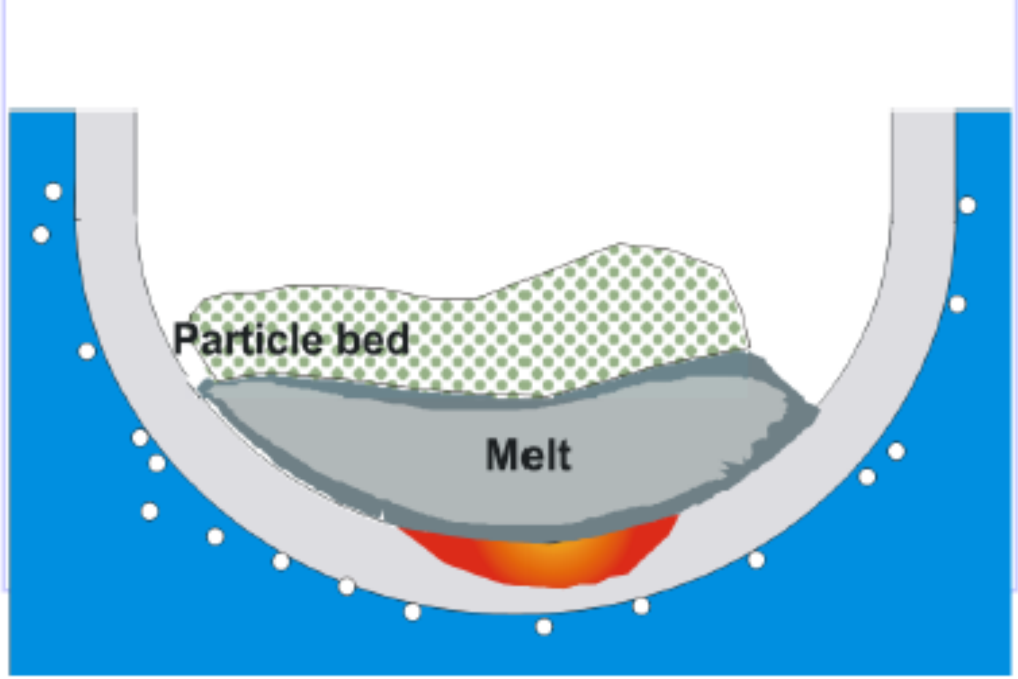
configuration of corium depending on content of molten steel



Options for molten corium stabilization in the reactor vessel



Option 1: Internal cooling

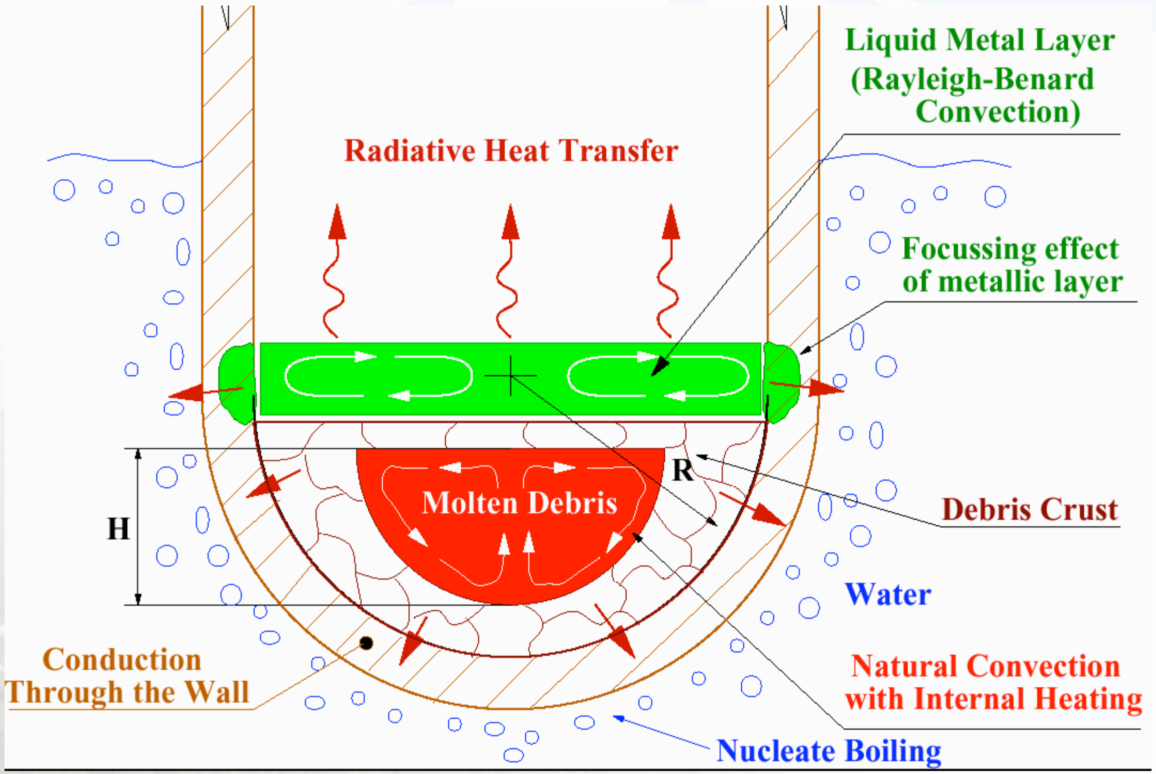


Option 2: External cooling

The In-Vessel Melt Retention (IVMR) Strategy

The reactor vessel lower head containing the melt pool is cooled from outside, which keeps the outer surface of the vessel wall cool enough to prevent vessel failure.

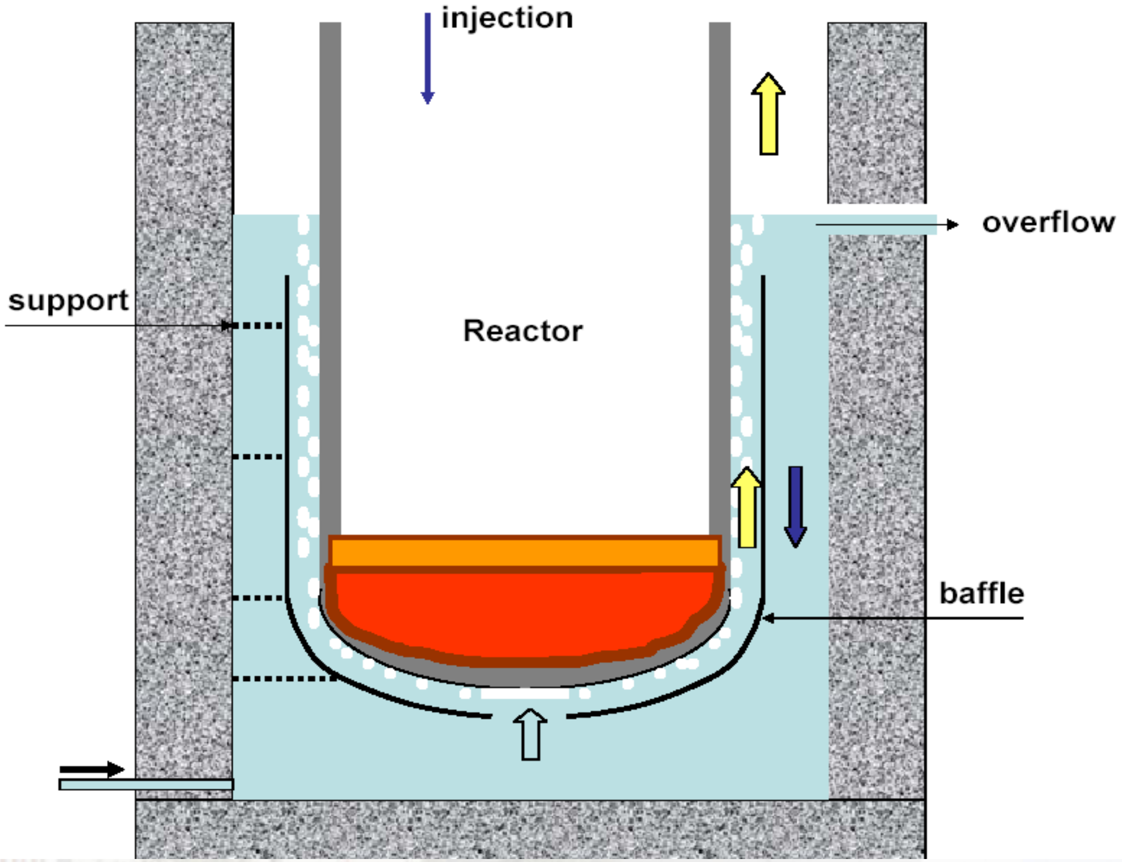
- Thermal-hydraulics of oxidic corium pool
  - Approximately 50% of residual power is transfered upwards (to the metal layer)
  - When water is absent (i.e. no reflooding of primary circuit), heat transfer at the top of metal layer is limited by radiation (i.e.: low heat transfer)
  - Redistribution of power to the lateral surface of the metal layer
  - **Heat flux concentration = focusing effect !!**



Advantages of in-vessel retention strategy

- No concern with direct containment heating
- Ex-vessel steam explosion is eliminated
- Ex-vessel melt/debris attack on the basemat and its coolability is of no concern
- The threat for the late failure of the containment due to the generation of non-condensible gases from MCCI is eliminated
- The source term (the accumulation of fission products in the containment) will be much less, since there will not be any ex-vessel source term

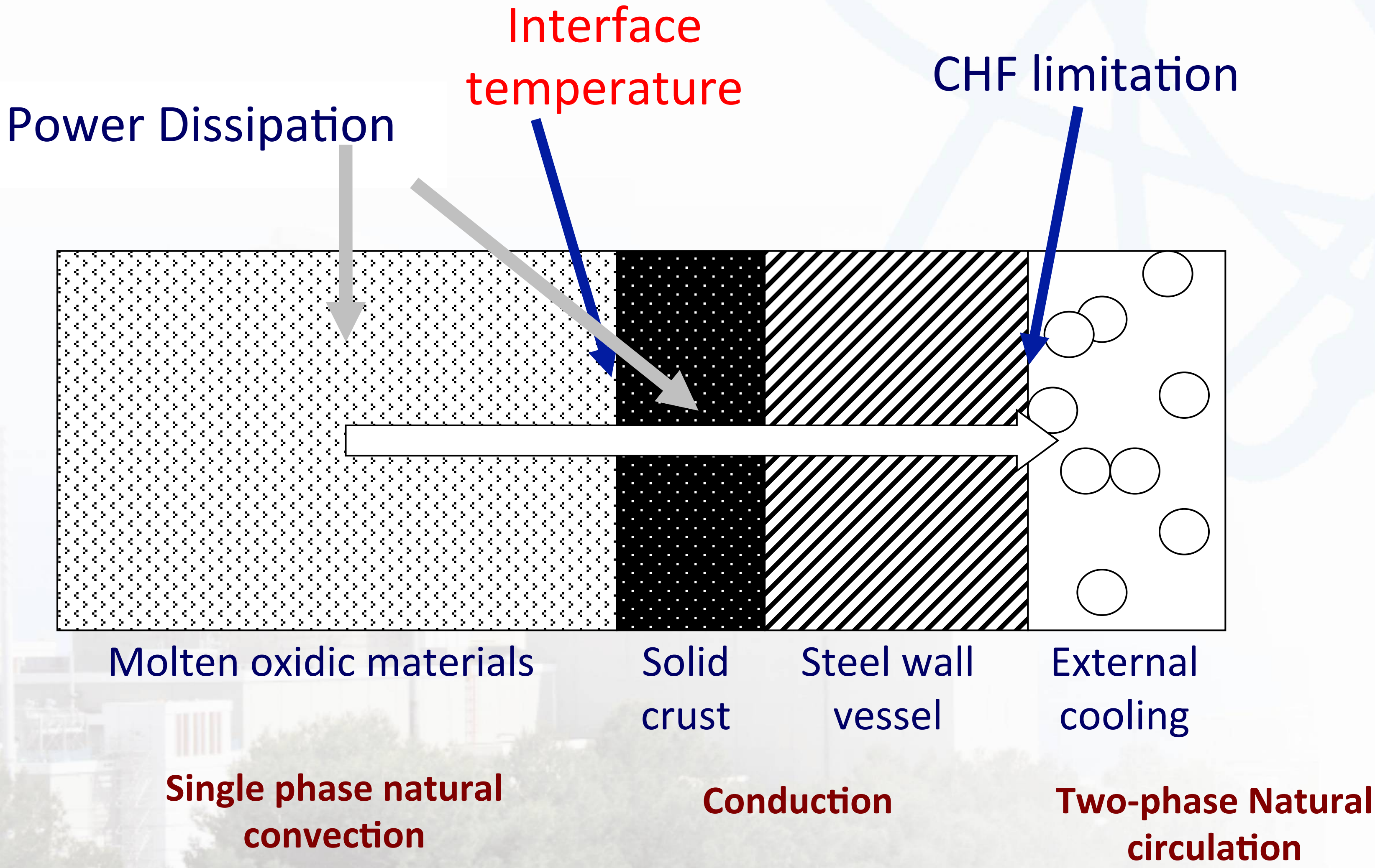
Schematic of baffle design for IVR cooling in VVER-1000



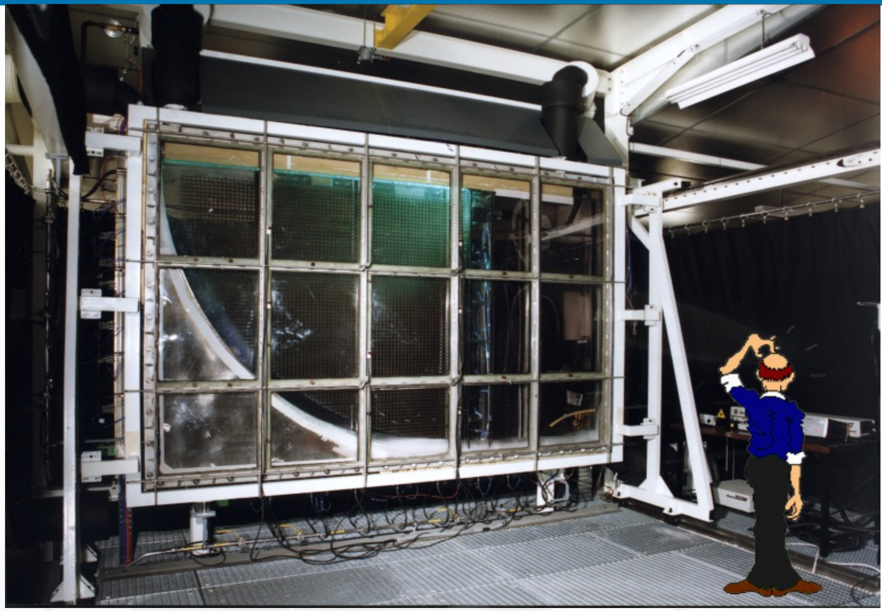
Maximum heat flux from the metal layer is at the level of 0.6-1.0 MW/m^2 (estimated conservatively for physically reasonable sequences), and the CHF is 1.8-2.0 MW/m^2

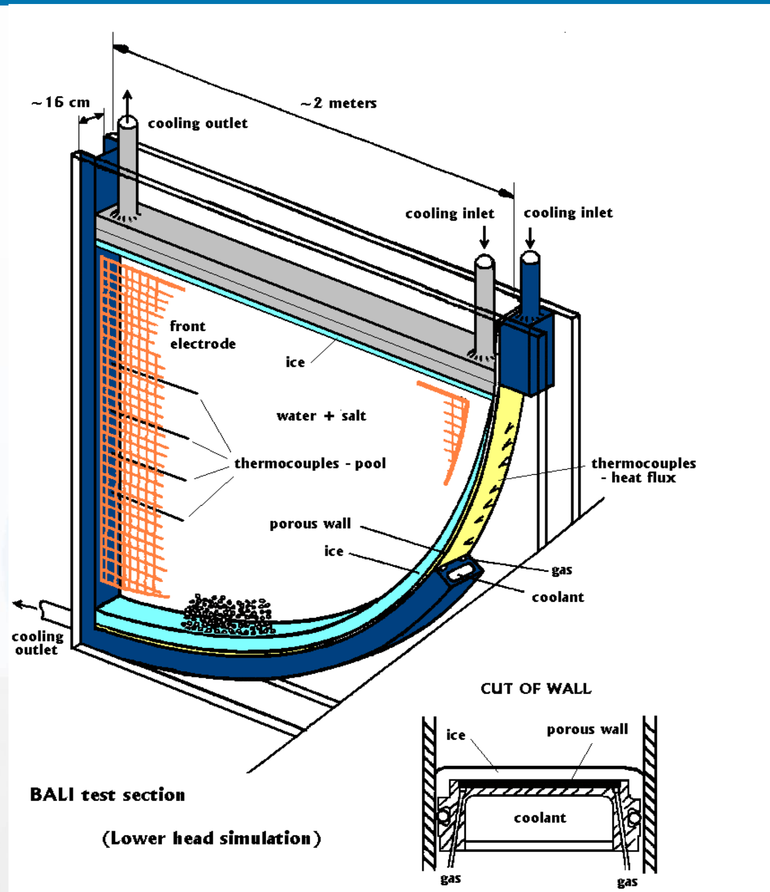
The above result points to a comfortable margin-to-failure which guarantees the IVR success in VVER-1000/320 plant.

Heat transfer from molten corium



Heat Transfer in Corium Pool: BALI test section (CEA) (Simulant fluid: water)

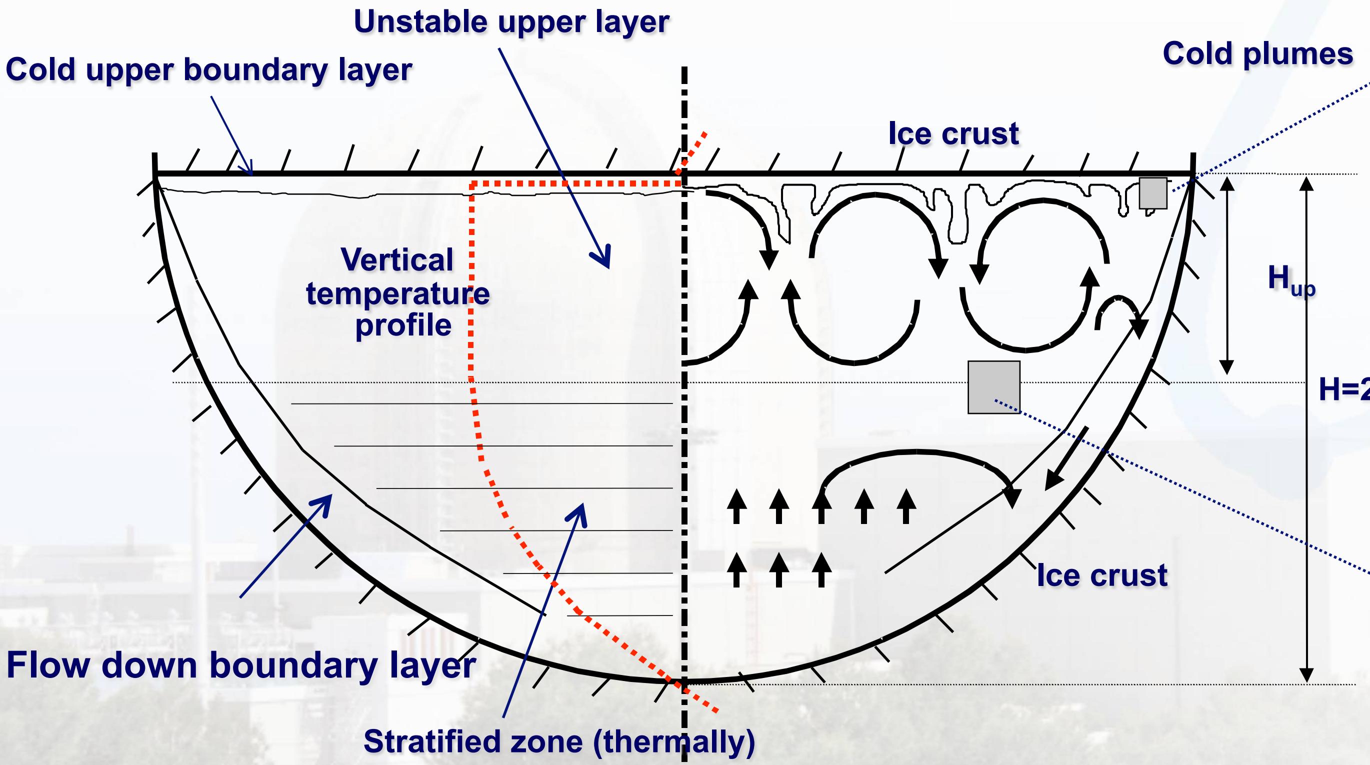


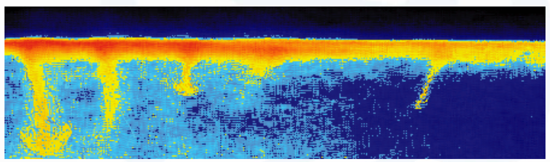


Scaling principles: $\Pr = \frac{\upsilon}{\alpha} \quad e = \frac{H}{R} \quad Ra_i = \frac{g\beta Q H^5}{\lambda \upsilon \alpha}$

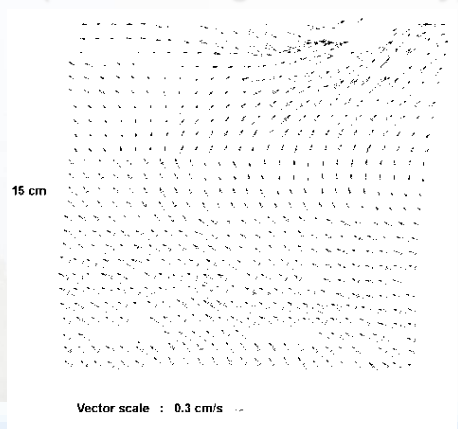
Flow configuration; natural convection in a corium pool surrounded by a solid crust

Solid crust=imposed temperature at the pool boundary (Tliquidus)

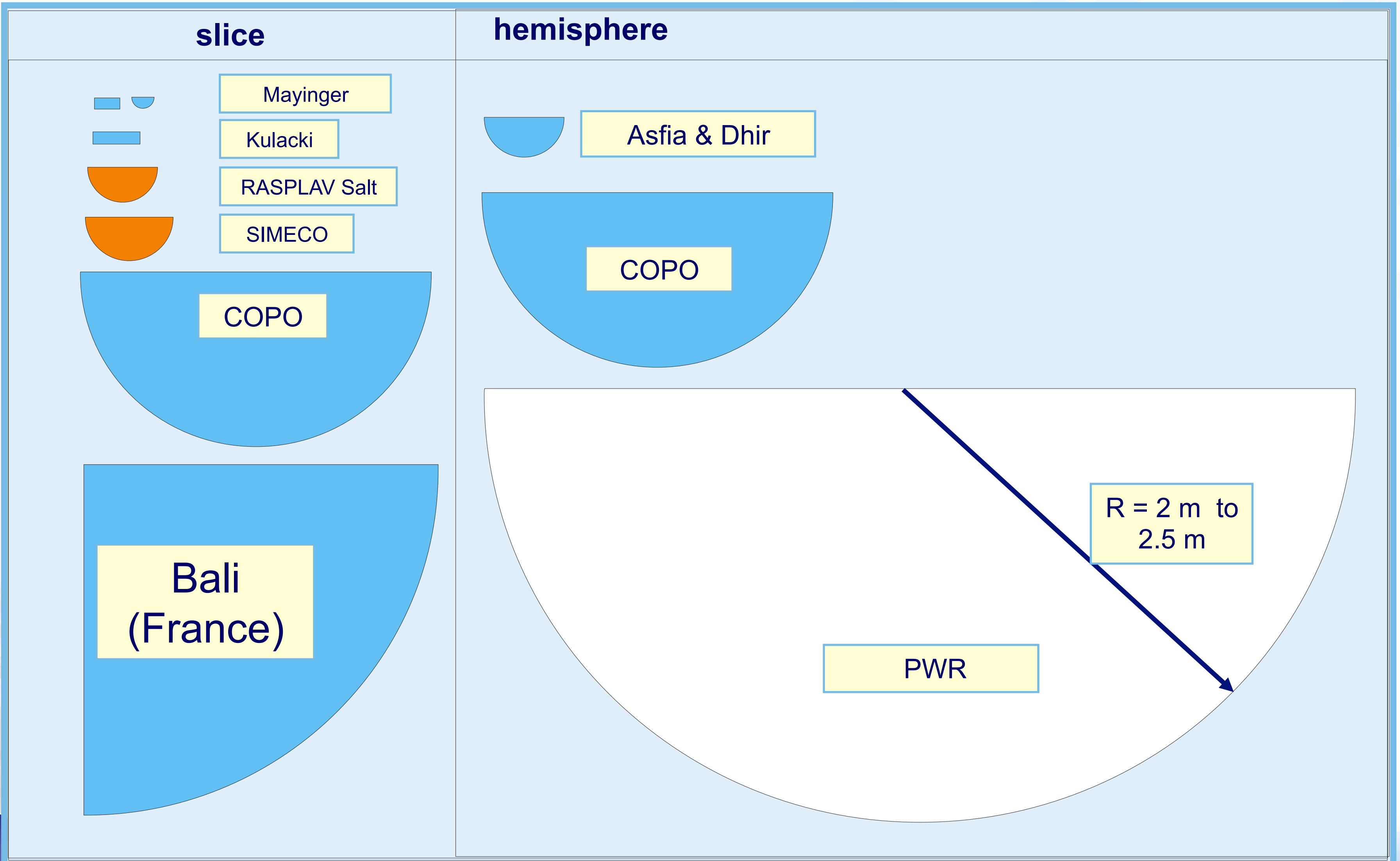




Numerical picture derived from LIF video film (Indexed colors)



Various experiments with water (1970-2012)



Ex-vessel Period of Accident Progression

Severe accident progression and phases of severity- ex-vessel phenomena

- **direct containment heating** due to high pressure expulsion of the corium from the reactor vessel
- **hydrogen combustion (deflagration/detonation),** globally or locally (heat produced by hydrogen combustion is about 120 MJ per kg of hydrogen)
- **core-concrete interactions** with additional production of hydrogen and carbon monoxide
- **foundation melt-through** due to core-concrete interactions
- long term **containment pressurization** and/or increase of temperature due to production of steam, hydrogen, carbon monoxide and carbon dioxide

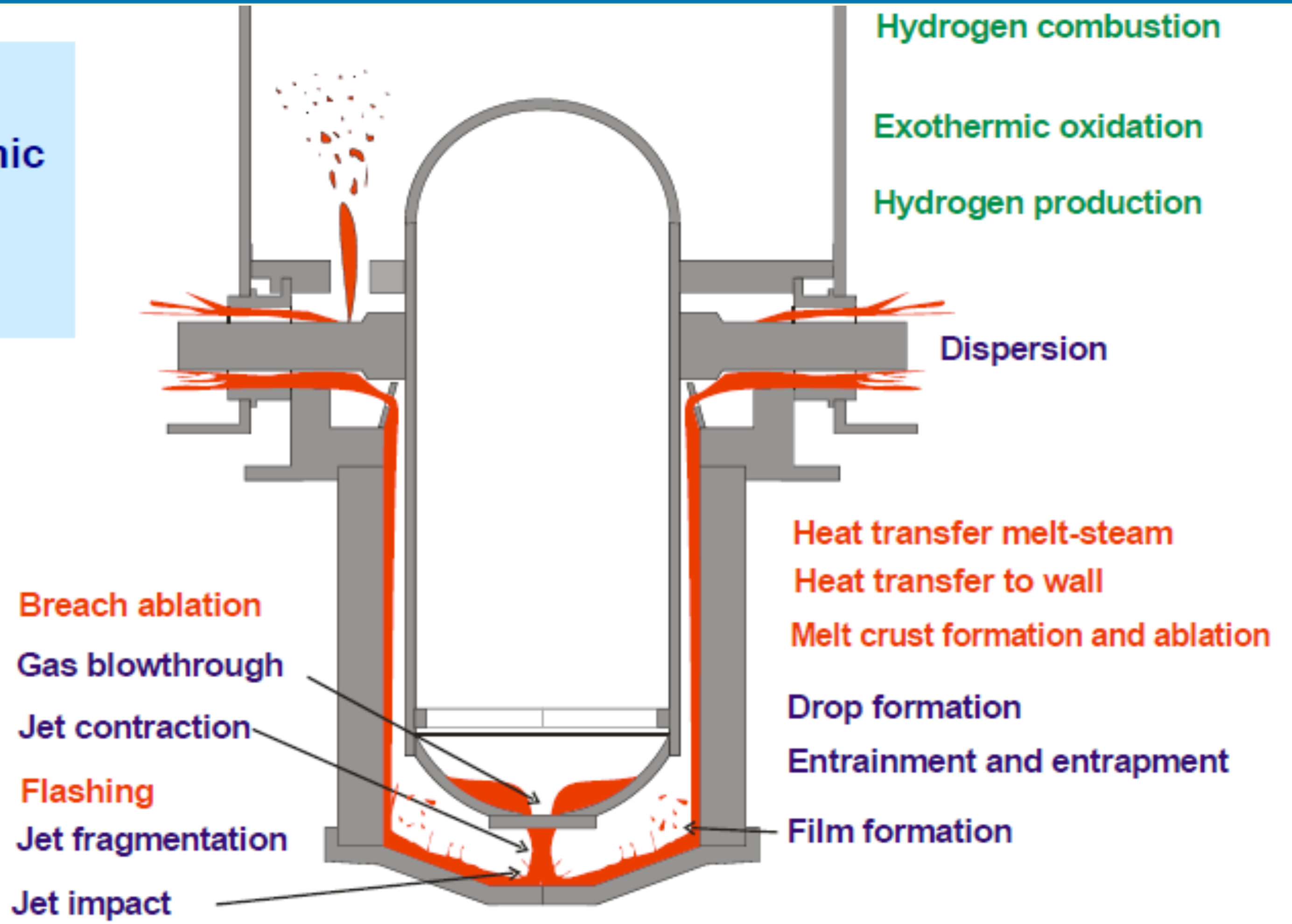
Phenomena during Direct Containment Heating (DCH)

Processes:

Fluid dynamic

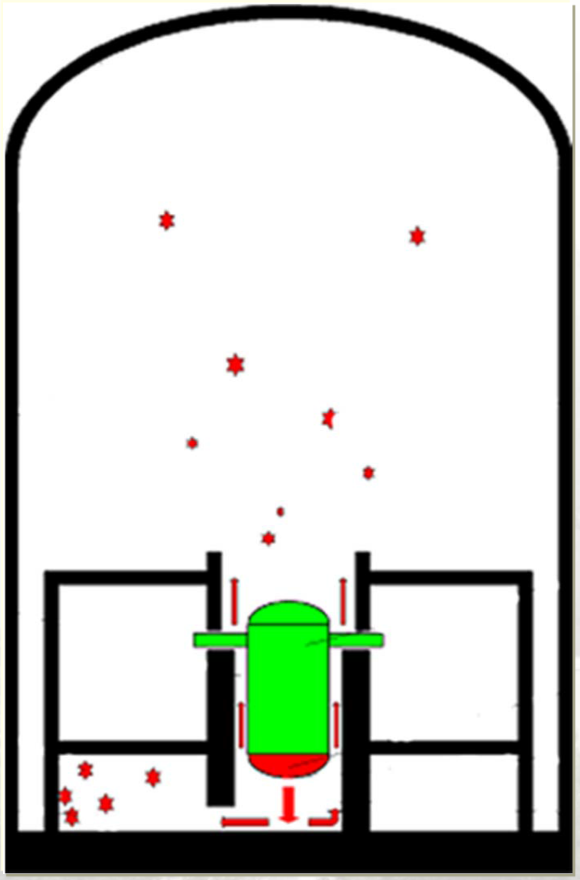
Thermal

Chemical



Key issue: How much of the melt interacts efficiently with the steam and atmosphere

Direct Containment Heating (DCH)



Containment pressurization due to

- RCS steam blowdown
- Debris / gas heat transfer
- Metal - steam exothermic reactions
- Hydrogen combustion

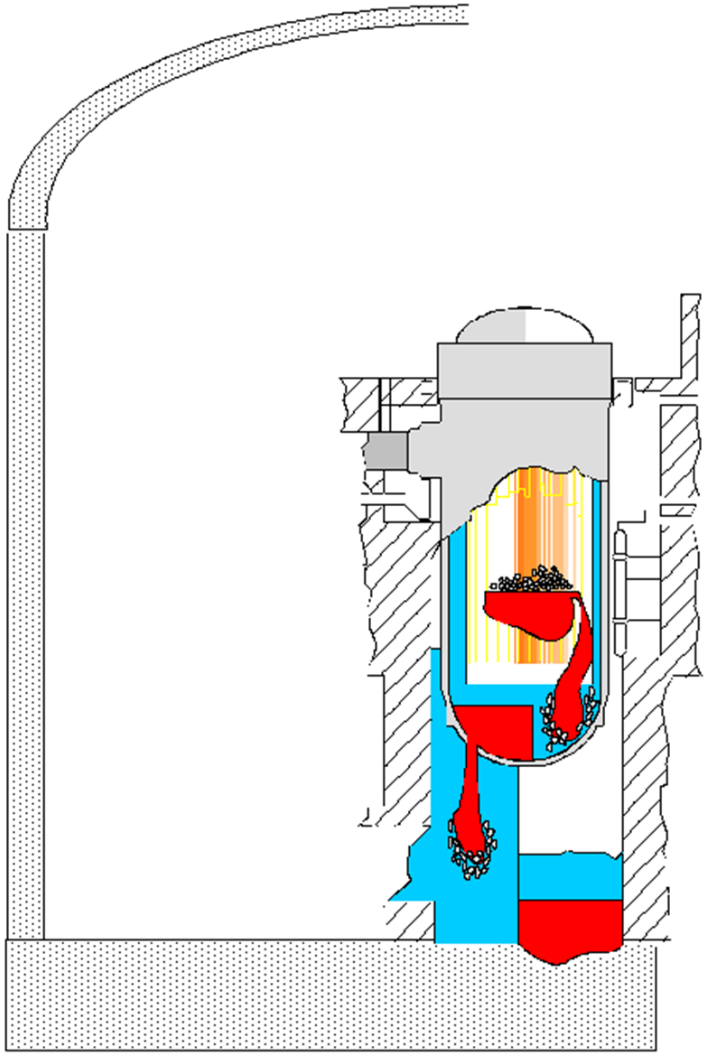
Other hazards

- Melt liner contact
- Compromise of safety equipment
- Intensified aerosol release (source term)

Possible positive effect

•Dispersed corium is in a long-term coolable configuration

Risks of Late Containment Failure



Main configurations:

- **Wet cavity**

– Formation of a debris bed

– Coolable ?

– Formation of a pool interacting with concrete?

- **Dry cavity:**

– Corium-Concrete Interaction

– Spreading to reduce heat fluxes

Risk of containment breach due to Molten Core Concrete Interaction

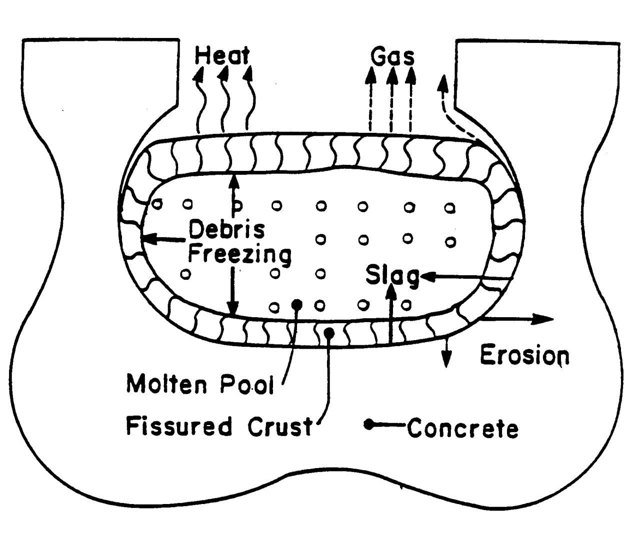
•In-vessel corium retention

•Ex-vessel core catchers

Molten Core-Concrete Interactions (MCCI)

- **Exothermic chemical reactions between core debris and concrete**
  - Large quantities of gas generated by concrete decomposition
  - Physical and chemical interactions between concrete decomposition gases and core debris release non-volatile fission products
  - Vertical and horizontal erosion of concrete basemat destroys containment foundation

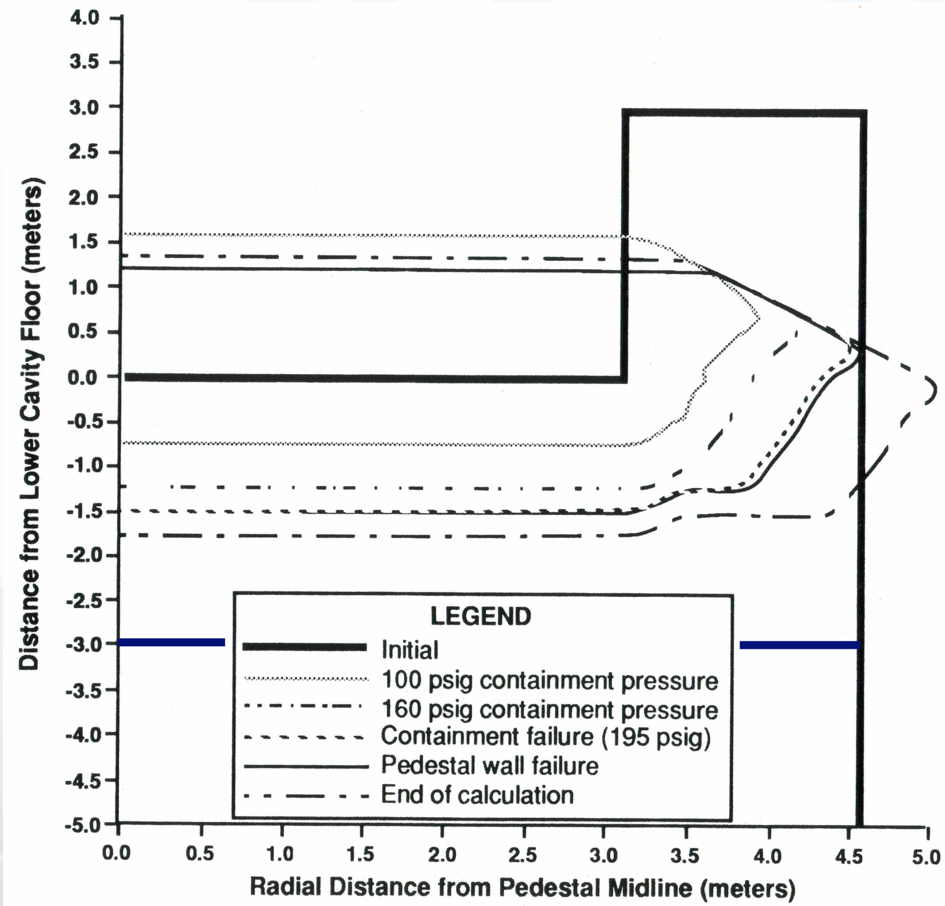
Property	Basalt (Siliceous) Concrete	Limestone Concrete
Solidus Temp (C)	1350	1420
Liquidus Temp (C)	1650	1670
Ablation Temp (C)	1450	1500



* Major components lost by decomposition: SiO_2, CaO, MgO

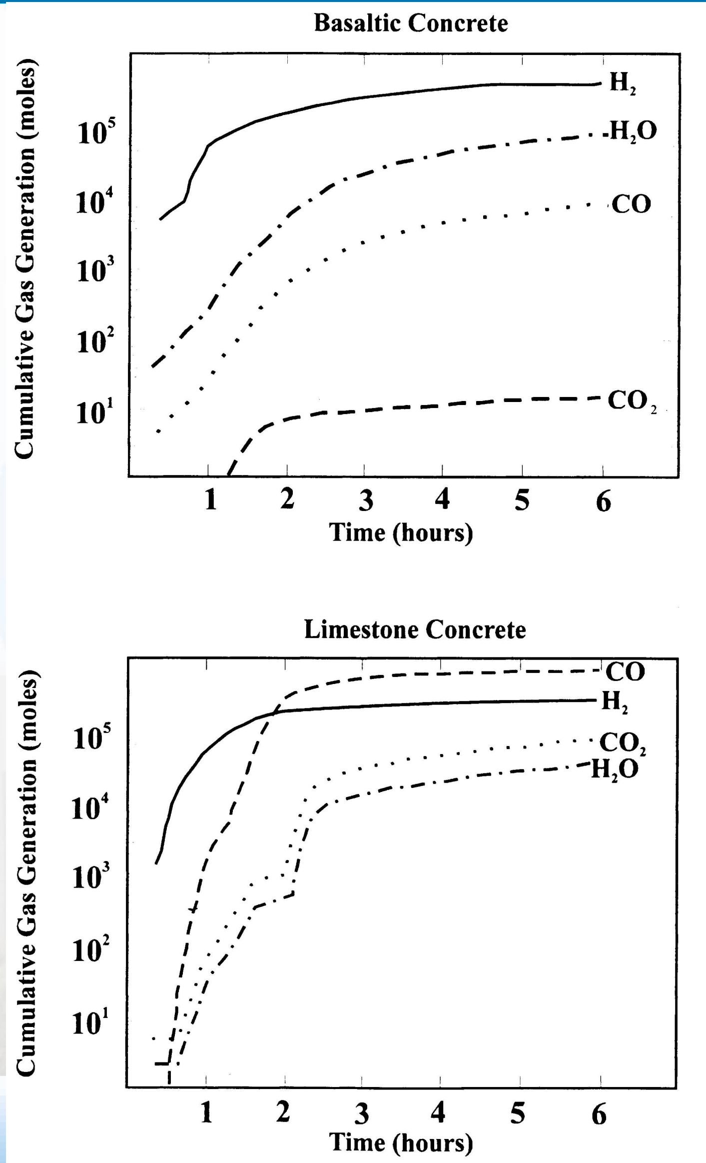
Effects of MCCI on Accident Progression

- Containment Structure Penetration
- High local atmosphere temperatures
  - Potential for local heating of containment pressure boundary
- Non-condensible gas generations
  - Significant contributor to containment pressure late in an accident sequence



Gas Generation from MCCI

- Quantity of gases released during MCCI depends on initial concrete composition
  - Resulting partial pressure of water vapor higher in Basaltic concretes
  - CO as contributing flammable gas more significant in Limestone concrete

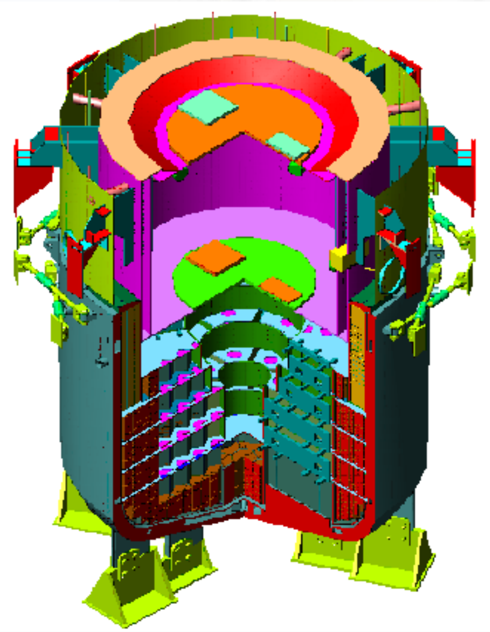


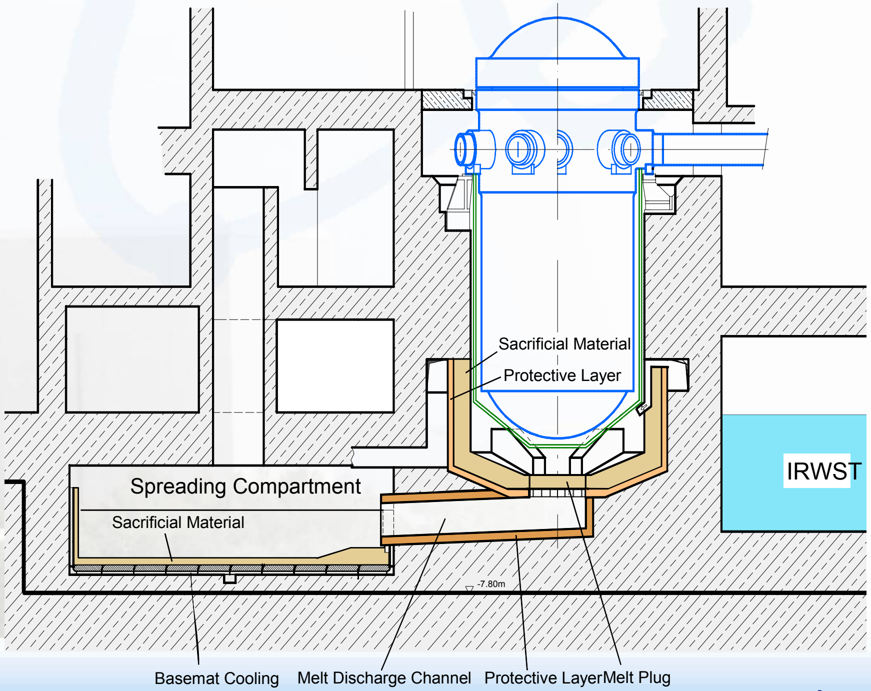
Major MCCI Research Programs

Test Program	Institution	Type of Concrete	Melt Composition
BETA	KfK/FRG	●Siliceous ●Limestone/ Common Sand	Iron/Alumina and Steel/Oxide + Zr
TURC	SNL/USA	●Limestone/ Common Sand	$UO_2 – ZrO_2$ + Zr Stainless steel UO_2-ZrO_2
SURC	SNL/USA	●Limestone/ Common Sand ●Siliceous	$UO_2 – ZrO_2$ + Zr Steel + Zr
ACE	ANL/USA	●Limestone/ Common Sand ●Siliceous	UO_2, ZrO_2 etc. + Steel, Zr
MACE	ANL/USA	●Limestone/ Common Sand ●Siliceous	$UO_2 – ZrO_2$ + Zr

The ex-vessel melt retention strategy

- **Control corium through spreading, use of sacrificial materials and cooling.**





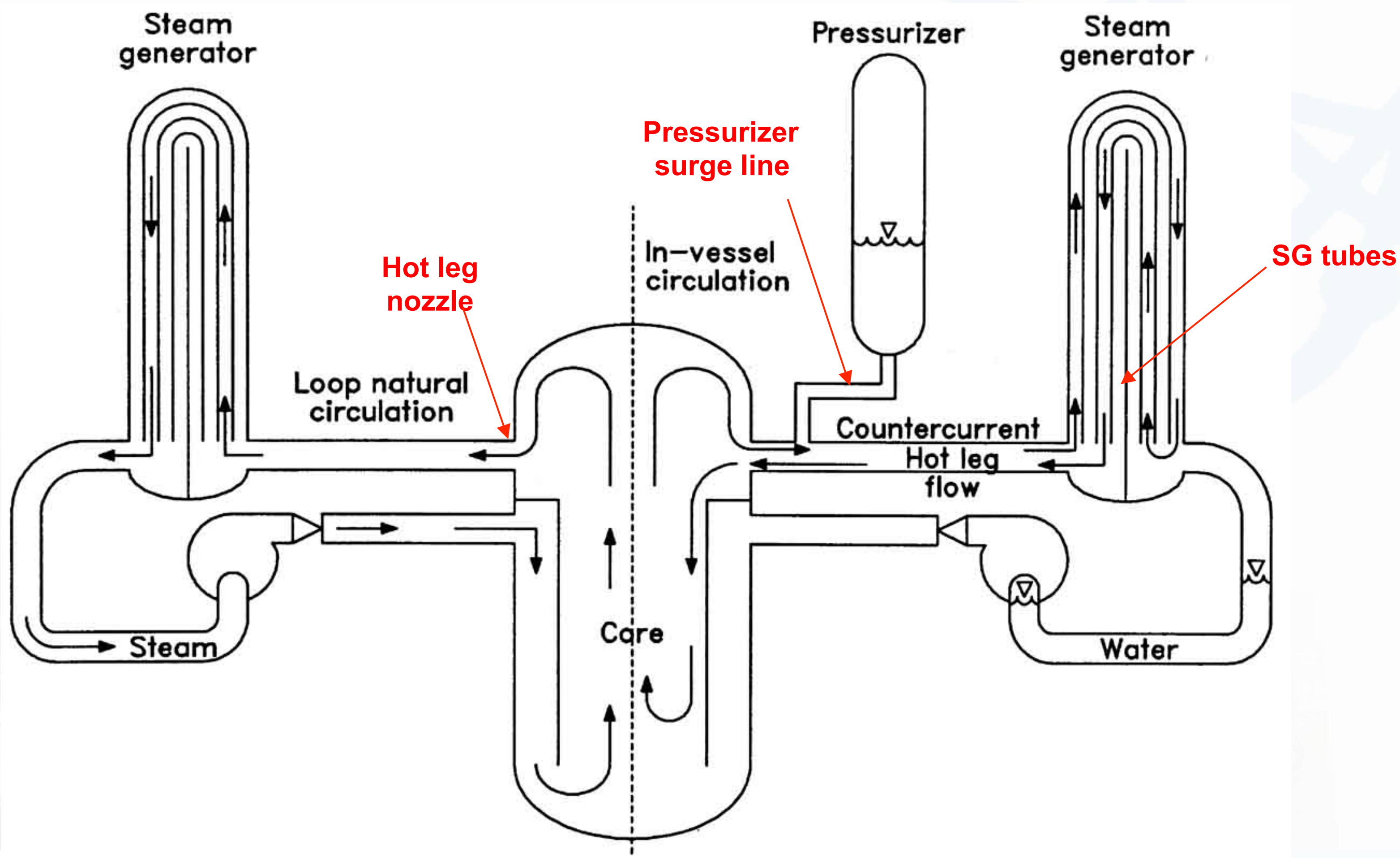
Other Severe Accident Phenomena

- Creep rupture of reactor coolant system pressure boundary during in-vessel core degradation
- Hydrogen combustion in containment
- Steam explosion

Induced Rupture of the Reactor Coolant System During Core Degradation

- Hot gases released from top of core during early phases of fuel damage
- Natural circulation flow patterns created
  - Hot gases cooled by transferring heat to colder surfaces
- Excess heating of pressure boundary can lead to creep rupture
  - Locations of concern: hot leg nozzles, pressurizer surge line, steam generator tubes

Natural Circulation Flow Patterns During In-vessel Core Degradation



Hot leg natural circulation may result in piping failure and depressurization (1)

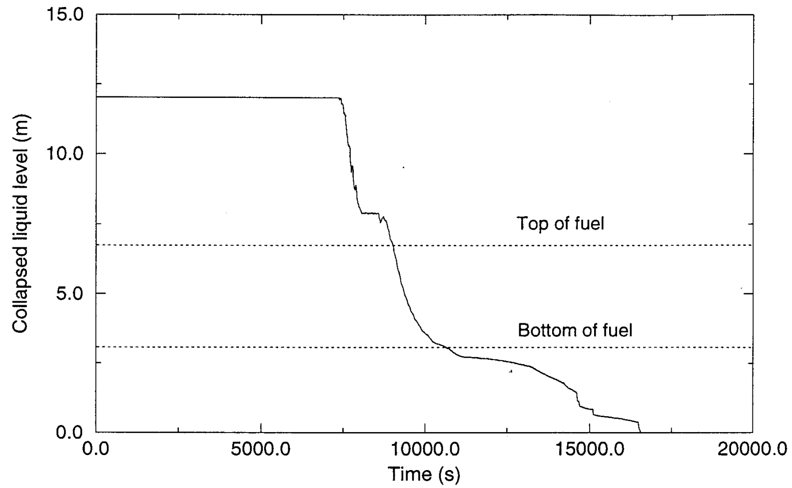
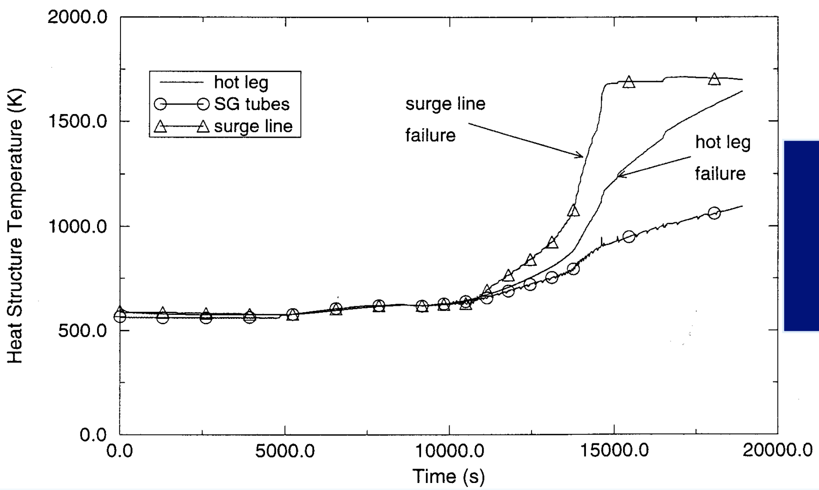
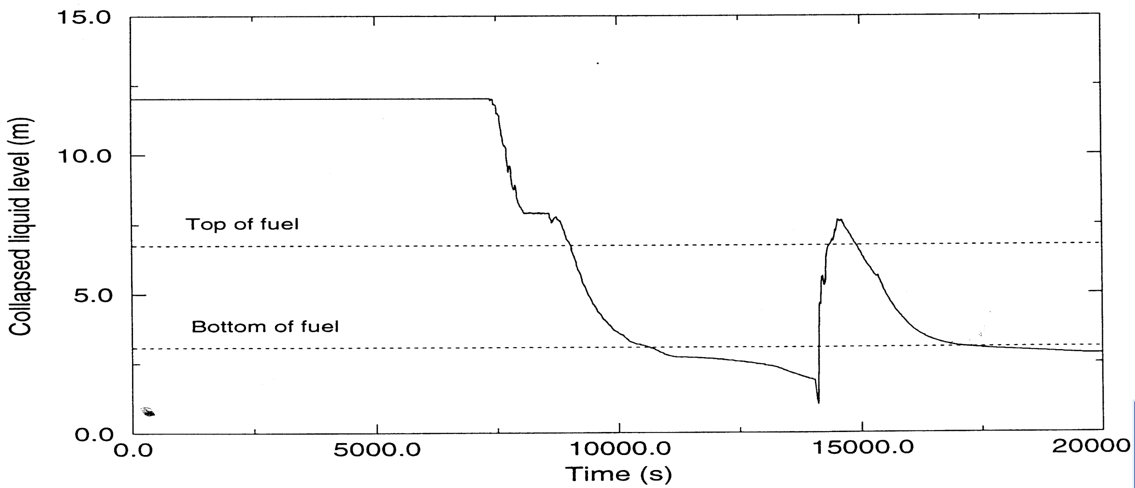


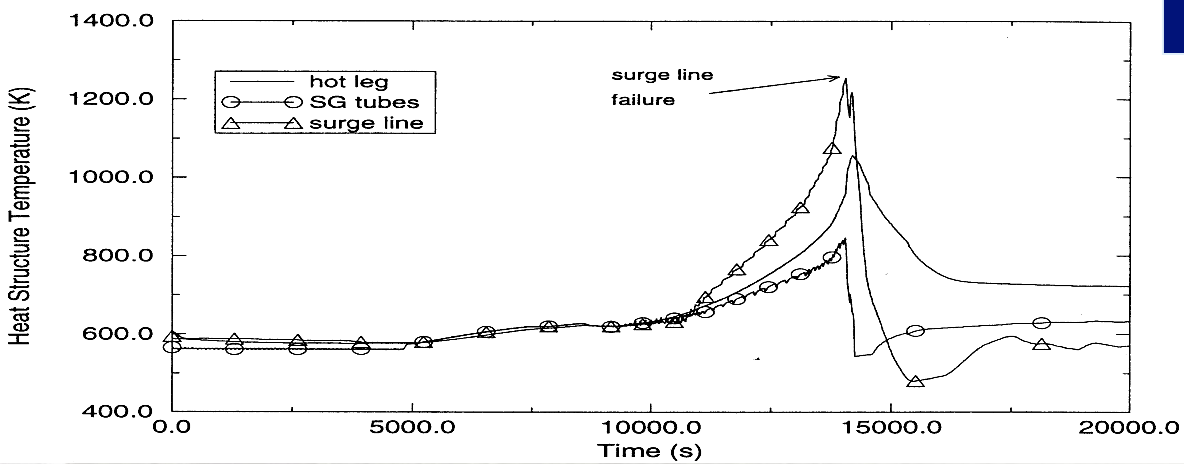
Figure 7. Collapsed liquid level in the reactor vessel for Case 1.



Piping failure and depressurization ignored

Hot leg natural circulation may result in piping failure and depressurization (2)



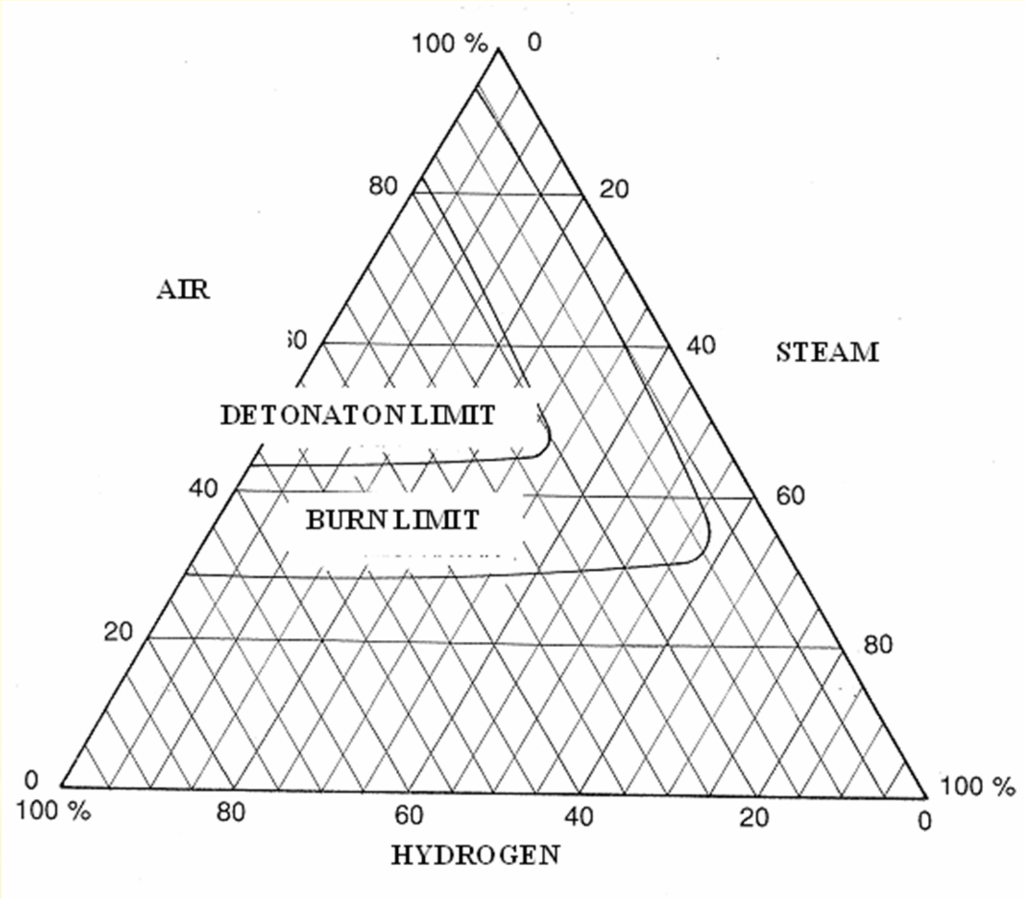


Piping failure and depressurization considered

Hydrogen Combustion in Containment

- Hydrogen released to containment from RCS
  - Transients: Pressurizer relief line (via quench tank)
  - LOCA: pipe break
- Hydrogen mixes with containment atmosphere
  - Distribution and local concentrations depend on flow field in containment
    - ✓ Pressure-drive flow among neighboring compartments
    - ✓ Natural convection
    - ✓ Ventilation system
- Combustion possible when local conditions exceed flammability criteria

Hydrogen flammability limit – Shapiro diagram



Hydrogen combustion: $2H_2 + O_2 \rightarrow 2H_2O$ + 120 MJ/kg

In-vessel hydrogen generation

— Cladding

$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$

— Structural materials (steel)

$2Cr + 3H_2O \rightarrow Cr_2O_3 + 3H_2$

— Fuel

$UO_2 + xH_2O \rightarrow UO_{2+x} + xH_2$

with $0 < x < 0.1$

Ex-vessel hydrogen generation

3 ex-vessel phenomena with potential for H2 generation:

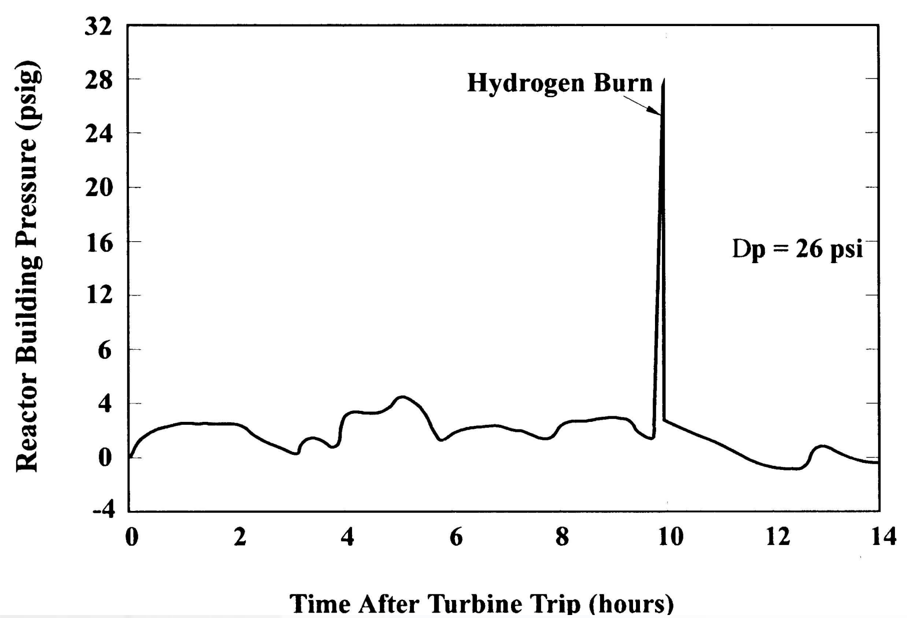
— Melt dispersal in containment after RPV failure (DCH)

— Molten fuel-coolant interactions (MFCI - steam explosions)

— Molten core-concrete interaction (MCCI - basement erosion)

Effect of Hydrogen Burns on Accident Progression

- Combination of high "base" pressure and hydrogen burn can lead to short-lived pressure loads that challenge containment capacity
- In a PWR containment, this usually requires flammable mixture in a very large volume.

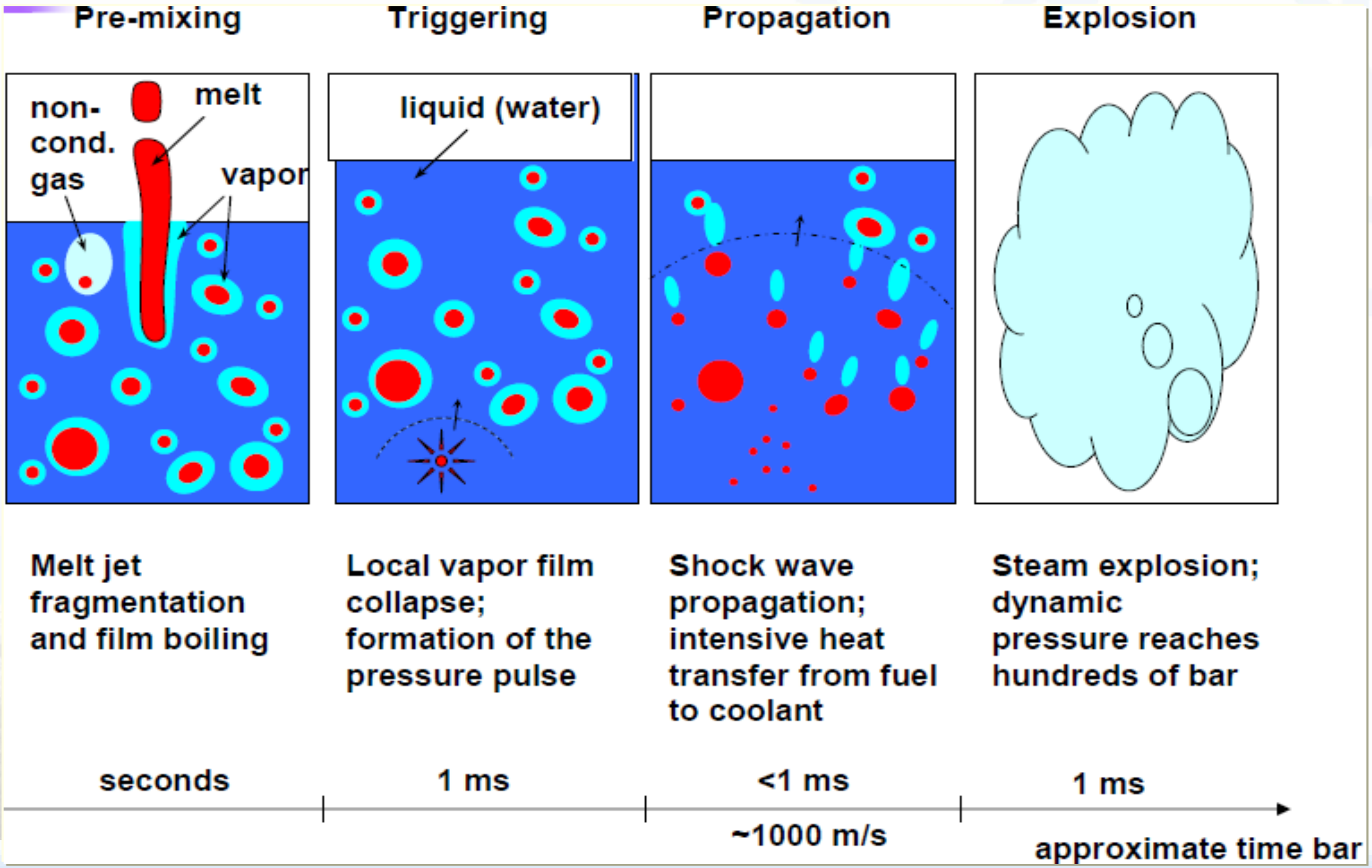


Hydrogen burn during the TMI-2 accident

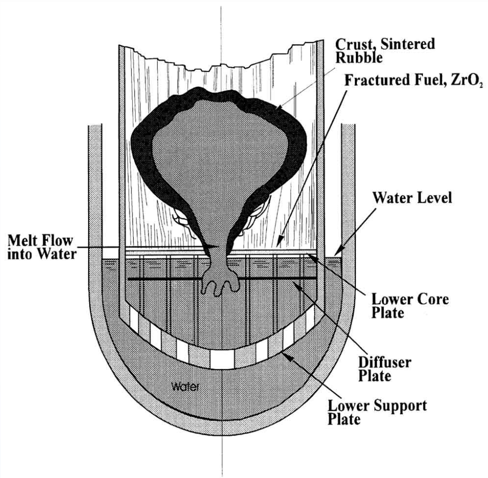
Steam Explosion

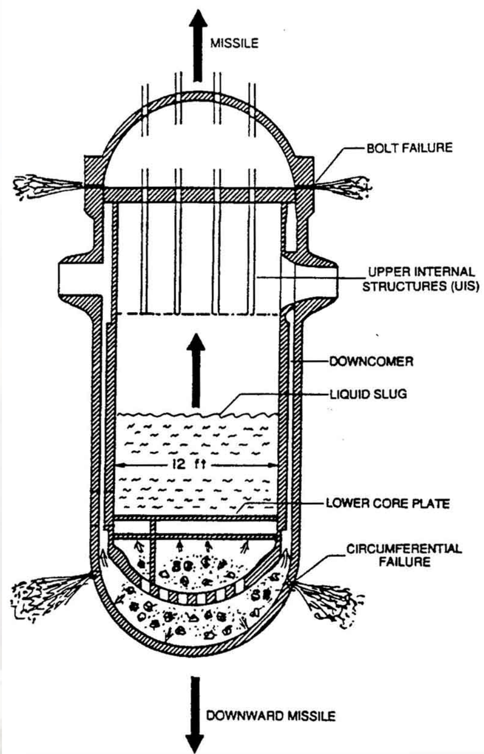
- A dynamic process that can occur when a large quantity of molten core debris relocates into a pool of water
  - In-vessel: Pour of molten material into RV lower head (Phase 2)
  - Ex-vessel: Low-pressure pour of melt into reactor cavity (Phase 3)
- A steam explosion requires four sequential phases of melt-coolant interaction to occur:
  - Course mixing of melt and water
  - Collapse of vapor film at heat transfer interface causing an accelerated energy release ("trigger")
  - Propagation of the pressure pulse through the mixture to form a shock wave
  - Outward expansion of the shock wave (damage mechanism)

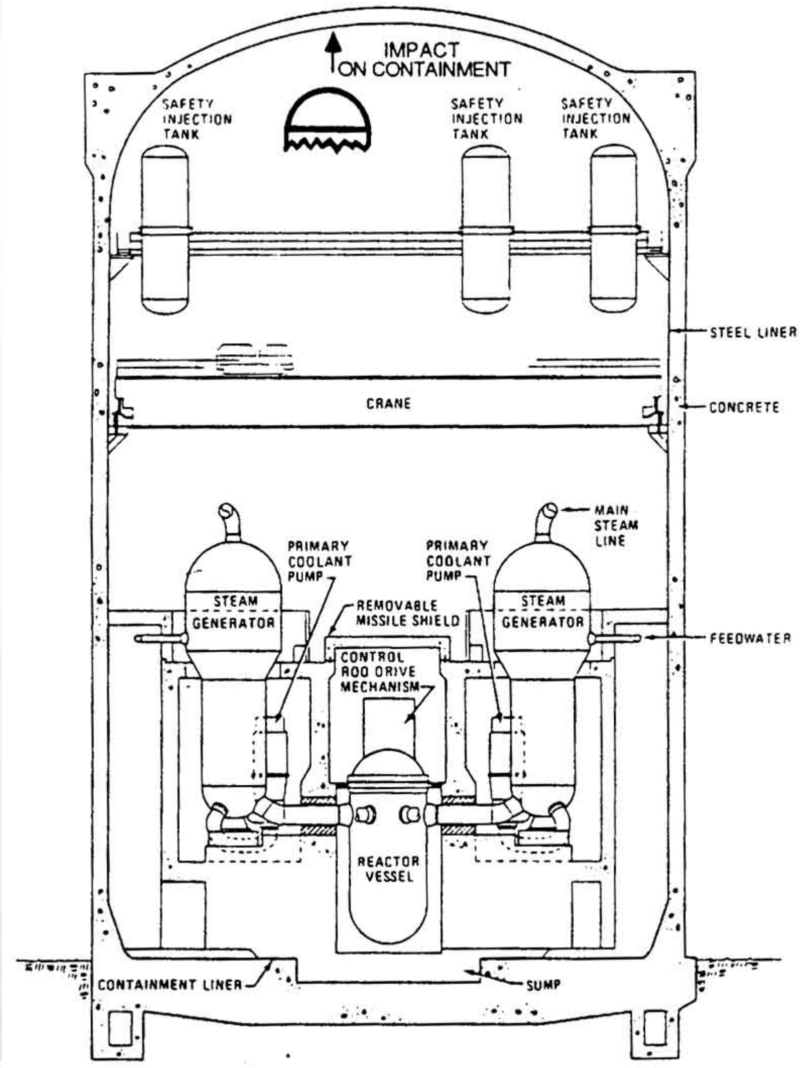
Fuel-coolant interaction phenomena and time scale



In-vessel Steam Explosion

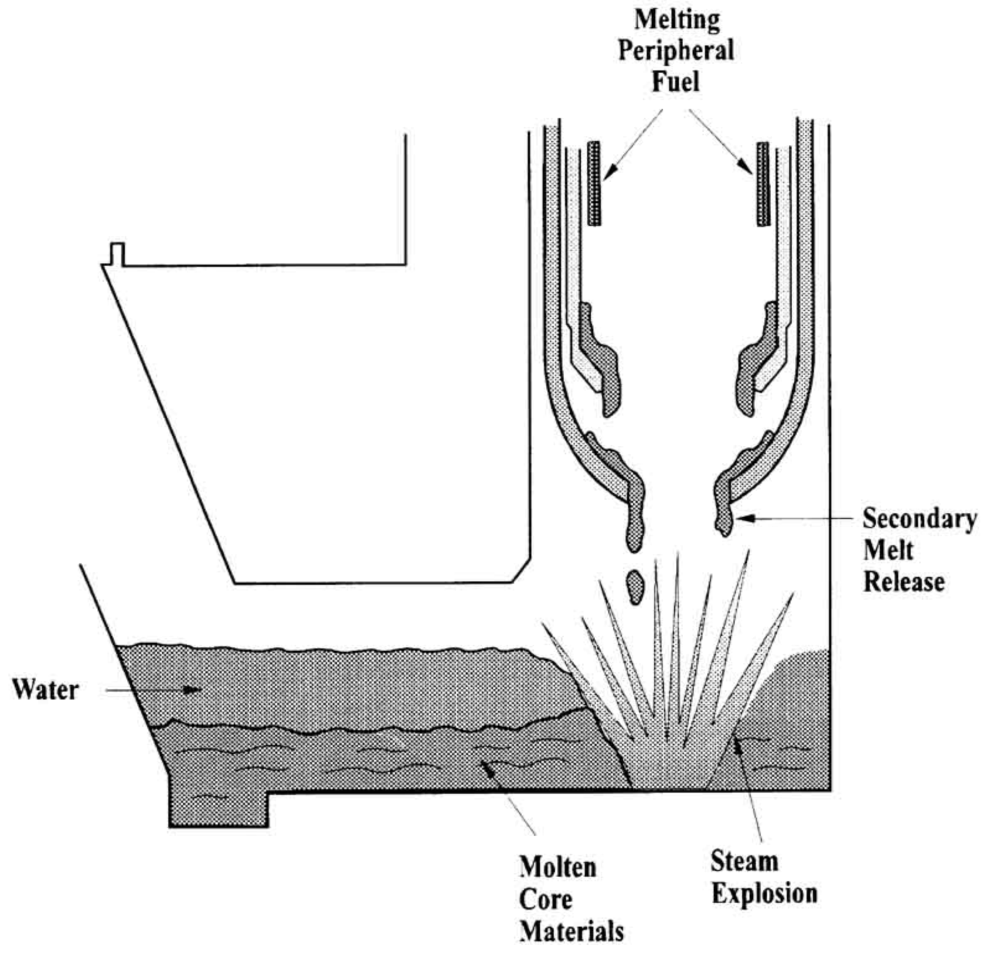






Ex-vessel Steam Explosion

- Pour of molten debris from reactor vessel into reactor cavity (full of water)
- Containment failure mechanism not clear for PWRs
  - Explosion not confined (no obvious missile)
  - Cavity walls strong



Summary Severe Accident Phenomena

- Severe accident phenomena span a wide range of technical disciplines
  - Thermal-hydraulics - Heat transfer
  - Fuel behavior - Material science
  - Reaction chemistry - Structural analysis
- Uncertainties remain in many areas, but sufficient knowledge is available to perform a credible assessment of accident progression for most sequences.

Summary of challenges to barriers resulting from severe accidents

- **Slow RPV melt through;** with a possibility of ex-vessel steam explosion, generation of missiles, dynamic loads of the containment and ex-vessel containment phenomena
- **Hydrogen combustion;** (deflagration/detonation) leading to fast loading with possible early containment failure
- **Containment pressurization;** due to generation of steam or non-condensable gases from decomposition of the containment concrete and combustion of combustible gases
- **Core-concrete interaction;** possible loss of containment integrity due to basemat melt-through
- **Containment by-pass;** e.g. steam generator (SG) tube rupture or damage or interface systems and direct release of reactor coolant and FPs to outside containment

Analysis of design extension conditions

IAEA SSR-2/1 on analysis of design extension conditions (art. 5.27)

- An **analysis of design extension conditions for the plant shall be performed.**
- The main technical objective of considering the design extension conditions is to provide assurance that the design of the plant is such as to prevent accident conditions not considered design basis accident conditions, or to mitigate their consequences, as far as is reasonably practicable.
- This might require additional safety features for design extension conditions, or extension of the capability of safety systems to maintain the integrity of the containment. These additional safety features for design extension conditions, or this extension of the capability of safety systems, shall be such as to ensure the capability for managing accident conditions in which there is a significant amount of radioactive material in the containment (including radioactive material resulting from severe degradation of the reactor core).
- The plant shall be designed so that it can be brought into a controlled state and the containment function can be maintained, with the result that significant radioactive releases would be practically eliminated.
- The effectiveness of provisions to ensure the functionality of the containment could be analysed on the basis of the **best estimate approach**.

Various applications of severe accident analysis

- Analytical **support for design** of plant systems
- Demonstration of acceptability of the design in **licensing**
- Analytical support for **development of accident management programmes**
- Determination of **source terms for emergency planning**
- Support for **Level 2 PSA**
- Resolution of **severe accident issues**
- Development of **training programmes**

Objectives of severe accident analysis for design

- Verification of **compliance with the acceptance criteria**, in particular with the radioactive release targets
- Evaluation of **ability of design (in particular containment) to withstand severe accidents** and to identify particular vulnerabilities
- Demonstration of **capability of equipment** including instrumentation **to survive severe accident conditions** and be used in accident management
- Assessment of **doses to the control room operators** and in all other locations where operator activities may be required
- Determination of the **source term** - an input for off-site emergency planning
- Identification of **accident management measures** that could be carried out to mitigate the effects, but specific supporting calculations are needed

Specific tasks for analysis supporting accident management

- Selection of key symptoms
- Selection of mitigation strategies
- Determination of expected positive effects and possible negative effects of the strategy
- Specification of set-points to initiate and to exit a strategy
- Confirmation of choice of symptoms for long-term processes
- Prioritisation and optimisation of strategies
- Evaluation of effectiveness of systems to perform intended functions
- Specifications of environmental conditions for operation of instrumentation and NPP systems
- Recommendations for equipment or instrumentation upgrades
- Computational aid development (simplified diagrams for assessment

SR No. 56 (2008) Approaches and tools for severe accident analysis for NPPs - Contents

Characteristics of computer codes for severe accident analysis

- **Wide range of processes to be covered (thermal-hydraulics, chemistry, metallurgy, FP transport)**
- **Phenomena to be modelled**
  - **Core degradation and fuel melting, vessel melt through**
  - **In-vessel and ex-vessel cooling of core melt**
  - **In-vessel melt retention**
  - **Fuel-coolant interaction, steam explosions**
  - **Distribution of heat inside the RCS**
  - **High-pressure melt ejection/direct containment heating**
  - **Hydrogen generation, distribution and combustion**
  - **Failure or by-pass of the containment**
  - **Release and transport of fission products**
  - **Core-concrete interaction, basemat melt through**
- **Knowledge of phenomena and validation of codes limited (large uncertainties in calculations to be considered)**

Level of understanding of phenomena for in-vessel analysis

- **Well understood phenomena**
  - Majority of phenomena in early phase of core degradation (boil-off, recriticality, reflooding before significant oxidation, cladding balooning, dissolution of fuel and other materials, …)
- **Low level of knowledge of phenomena**
  - Hydrogen production during flooding of degraded core
  - Recriticality of degraded core
  - Steam flow through the degraded core
  - Formation of debris
  - Formation of molten pool, formation of crust, its stability, break-through
  - Molten core relocation

Level of understanding of phenomena for ex-vessel analysis

- **Well understood phenomena**
  - **Both local and global containment pressurization**
- **Low level of knowledge of phenomena**
  - **Long lasting processes, including late phase of in-vessel phenomena as a boundary condition**
  - **Natural convention in the containment**
  - **Heat exchange with structures**
  - **Temperature stratification (typically underpredicted by integral codes)**
  - **Hydrogen distribution**
  - **Material interactions, mainly molten corium concrete interaction**

Examples of computer codes used for severe accident analysis

- Computer codes for analytical support of AM
  - Prevention (EOP domain)
    - Standard best-estimate TH codes
      - RELAP5/MOD3.2
      - ATHLET
      - CATHARE V1
  - Mitigation (SAM domain)
    - Integral fast running codes for analysis of both in-vessel and ex-vessel phenomena
      - MAAP 4
      - MELCOR 1.8.
      - ASTEC
    - Detailed codes for analysis of in-vessel phenomena
      - SCDAP/RELAP5/MOD3.2
      - ATHLET-CD
      - ICARE/CATHARE V1
    - Detailed codes for analysis of ex-vessel phenomena
      - CONTAIN
      - COCOSYS
      - GOTHIC
    - Special codes for analysis of individual phenomena (melt spreading, hydrogen detonation, ...)

Phenomenon	ASTEC V0.3	MAAP 4.0.3	MELCOR 1.8.4	ATHLET-CD	ICARE/ CATHARE V1	SCDAP/ RELAP5 3.2
Fission and decay						
· Recriticality	-	-	-	✓	-	-
· Boron dilution effects	-	-	-	✓	-	-
· Absorber/fuel separation	✓	✓	✓	✓	✓	✓
Thermal-hydraulic						
· 2-phase	-	✓	✓	✓	✓	✓
Thermal non-equilibrium	-	-	-	✓	✓	✓
Momentum equation	-	-	✓	✓	✓	✓
Flexible nodalisation RCS	✓	-	✓	✓	✓	✓
Core reflood	-	-	-	✓	✓	✓
Non-condensibles	✓	✓	✓	✓	✓	✓
Impact of core degradation on flow paths	✓	✓	-	✓	✓	✓
Impact of blockage formation	✓	✓	-	✓	✓	✓
Core bypass	✓	User input	-	✓	✓	✓
Reflux condenser mode	-	✓	✓	✓	✓	✓
Natural gas convection within RPV	✓	✓	-	✓	✓	✓
Natural gas convection within RCS	-	✓	-	✓	✓	✓

Table A2-1 : Main features of severe accident codes (1)

Phenomenon	ASTEC V0.3	MAAP 4.0.3	MELCOR 1.8.4	ATHLET-CD	ICARE/ CATHARE V1	SCDAP/ RELAP5 3.2
Core heat transfer						
Radiation radial	✓	✓	✓	✓	✓	✓
Radiation axial	✓	?	✓	✓	✓	-
Radiation from molten pool	✓	✓	✓	-	✓	-
Initial core damage						
Fuel/cladding contact	✓	?	-	✓	✓	? ✓
Ballooning	✓	✓	-	✓	✓	? ✓
Oxide flowering	-	-	-	-	✓	? -
Oxide shattering	-	User input	-	-	-	? ✓
Irradiated fuel effects	-	-	-	-	-	? ✓
Non-fuel dissolution	-	✓	✓	-	✓	? ✓
Fuel dissolution	✓	✓	✓	✓	✓	✓
Oxide shell failure	-	P	P	P	✓	✓
Absorber models	AIC and B_4C	AIC or B_4C	AIC and B_4C	AIC or B_4C	AIC and B_4C	AIC or B_4C
Spacer grid	-	-	-	-	✓	✓
Canister wall	-	✓	-	✓	-	✓
Lower core support plate model	✓	✓	-	-	✓	-
Upper plenum structure model						✓

Table A2-1 : Main features of severe accident codes (2)

Phenomenon	ASTEC V0.3	MAAP 4.0.3	MELCOR 1.8.4	ATHLET-CD	ICARE/ CATHARE V1	SCDAP/ RELAP5 3.2
Oxidation and hydrogen						
Zirconium	✓	✓	✓	✓	✓	✓
During quenching	No model	No model	No model	✓	No model	✓
Double-sided oxidation	✓	User input	-	✓	✓	✓
U-Zr-O	✓	✓	-	-	✓	✓
During fuel/coolant interaction	No model	No model	No model	No model	-	No model
Particulate debris	-	✓	✓	-	✓	✓
Stainless steel	-	✓	✓	-	✓	✓
B_4C	-	-	✓	-	✓	✓
Impact of air	-	-	✓	-	✓	-
Relocation and pool formation						
Relocation velocity	-	User input	User input	User input	✓	✓
Heat transfer to cladding	✓	User input	User input	✓	✓	✓
Formation particulate debris	✓				✓	✓
Coolability model for particulate debris	-	✓	✓	-	✓	No model
Formation of metallic blockages	✓	-	✓	✓	✓	✓
Radial spreading	-	User input	?	-	✓	✓
Molten pool behaviour in core						
Stratification (metallic/oxidic)	-	-	-	-	-	✓
Heat transfer (transient / steady	-/✓	-/✓	-	-	✓	✓✓
Interaction with supporting	✓	✓	✓	-	✓	✓
Melting of structures above the	-	✓	✓	-	✓	✓
Failure criteria for	-	✓	-	-	-	✓
Relocation of non-molten	✓	-	-	-	-	✓

Table A2-1 : Main features of severe accident codes (3)

Phenomenon	ASTEC V0.2	MAAP 4.0.3	MELCOR 1.8.4	ATHLET-CD	CATHARE/ ICARE-2	SCDAP/ RELAP5 3.2
Fuel coolant interaction						
Melt fragmentation	✓	✓	User input	-	-	User input
Melt dispersal	-	-	-	-	-	-
RCS pressurisation	✓	✓	✓	✓	✓	✓
Steam explosion	-	-	-	-	-	-
Lower head behaviour						
State of the metallic and oxidic melt	Mixed	User input	Mixed	-	Mixed	Mixed
Debris coolability model	-	✓	✓	-	✓	✓
Pool coolability model	-	Optional	-	-	✓	
Detailed lower head failure model	-	✓	✓	-	✓	
Fission product release from fuel						
High volatile fission products	✓	✓	✓	✓	✓	✓
Medium and low volatile	✓	✓	?	✓	-	?
Release from molten pool	-	User input	-	-	-	-
FP transport in RCS & conn. lines						
Deposition in main coolant lines	✓	✓	✓	✓	-	
Revolatilization in main coolant lines	✓	✓	✓	✓	-	
Deposition in connecting lines	-	-	✓	✓	-	
Revolatilization in connecting lines	-	-	✓	✓	-	
Pool scrubbing	-	✓	✓	-	-	
Deposition in dry steam generator	✓	✓	✓	✓	-	
Chemistry						
Iodine	-	-	-	-	-	-
B_4C	-	-	✓	-	✓	✓

Table A2-1 : Main features of severe accident codes (4)

Difficulties in performing deterministic safety analysis of design extension conditions/severe accidents

- **Worldwide, there is no widespread agreement on the best approach to severe accident analysis**. The approach varies from predominantly probabilistic approach used in USA to the concept of address severe accidents with deterministic criteria typical for Europe.
- Although it is well established including IAEA Standards that analysis of severe accidents is performed with **best estimate approach (to the extent possible), [GSR-4] requires that the analysis still shall be conservative**. This can be ensured considering sufficiently large margins (significantly larger than in case of design basis accidents) in interpretation of the results in terms of predicted timing and severity of phenomena.

Difficulties in performing deterministic safety analysis of design extension conditions/severe accidents

- Another issue is connected with assumptions regarding **operability of plant systems in case of severe accidents.** Consideration of all plant systems even beyond their normal operating range is usually recommended and acceptable for development of severe accident management guidelines, but is very complicated to rely on survivability of systems in demonstrating acceptability of the plant design.
- In addition, majority of systems would not be available due to complete lack of normal and emergency power supply. **It is therefore advisable to demonstrate acceptability of the design using only systems dedicated to severe accident mitigation.** It is also in accordance with the requirement on independence of provisions at different levels of defence.

PIEs leading to BDBA or severe accidents (IAEA)

- The severe accidents result from sequences in which the safety systems have malfunctioned and some of the barriers to the release of radioactive material have failed or have been bypassed. **These sequences should be selected by adding additional failures or incorrect operator responses to the DBA sequences (to include safety system failure).**
- The most rigorous way of identifying severe accident sequences is to use the results of the Level 1 PSA. However, it might also be possible to identify representative or bounding sequences from an understanding of the physical phenomena involved in severe accident sequences, the margin existing in the design, and the amount of system redundancy remaining in the DBAs.
- Examples of severe accident initiators include the following:
  - Complete loss of the residual heat removal from the reactor core
  - LOCA with a complete loss of the emergency core cooling
  - Complete loss of electrical power for an extended period

Advisable assumptions for deterministic analysis of severe accidents for licensing of new reactors

- Best estimate computer codes to be used to the extent possible
- Best estimate of plant parameters and performance of the systems may apply
- Conservative assumptions may be relaxed, e.g. SFC does not typically apply
- Operator actions from MCR should not be credited before 30 minutes from the accident initiation (1 hour outside MCR)
- Demonstration of capability to perform required actions and survivability should be provided
- Use of systems whose failure led to the given severe accident should not be credited
- Preferably, analysis should consider only use of the systems dedicated for mitigation of severe accidents
- Large uncertainties in predictions (timing and severity) should be taken into account in interpretation of results

Adopted deterministic acceptance criteria for severe accidents

- Molten corium shall be coolable inside the RPV (in-vessel corium retention strategy)
- Reactor coolant system pressure should be reduced below 2 MPa at the time of molten corium relocation to the reactor bottom head
- Containment design pressure and temperature shall not be exceeded (500 kPa, 150 ^{O}C)
- Global hydrogen deflagration shall be avoided (average hydrogen concentration below 10 %)
- Survivability of the equipment important for the containment performance, including penetrations, isolation devices, hatches shall be ensured
- Radiological (EUR) criteria
  - **Atmospheric release of caesium-137 below 30 TBq**
  - **No Emergency Protection Action beyond 800 m**
  - **No Delayed Action beyond 3 km**
  - **No Long Term Action beyond 800 m**
  - **Limited economic impact**

CONCLUDING SUMMARY

Threats considered

Threats to early failure of containment integrity – to be practically eliminated

- High pressure melt ejection
- In-vessel steam explosion
- Hydrogen detonation or large scale combustion
- Direct containment heating
- Ex-vessel steam explosion

Threats to late failure of containment integrity – to be mitigated

- Molten Corium Concrete Interactions (MCCI) with potential containment melting through
- Hydrogen production and potentially local combustion
- Containment overpressurization (filtered-venting, if necessary)
- Significant fission product release through the containment leakages
- Containment bypass

Measures for ensuring containment integrity in case of severe accidents

- Fast RCS depressurization in the case of the core damage
- In-vessel corium retention by flooding the reactor cavity in combination with injection into RPV, using a baffle for streamline coolant flow around the vessel
- Installation of passive autocatalytic recombiners for severe accidents
- Containment filtered venting (not necessary, just as an additional protection)
- Ventilation of the surrounding structure of the primary containment, operable in severe accident conditions
- Instrumentation provided to allow the necessary actions to be carried out and the response monitored
- Ensuring habitability of the main control room
- Implementation of the plant specific Severe Accident Management Guidelines (SAMG)

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Common Acronyms

- ADS Automatic Depressurization System
- ADV Atmospheric Dump Valve
- AFW Auxiliary Feedwater System
- AM Accident Management
- APET Accident Progression Event Tree
- ATWS Anticipated Transient Without SCRAM
- B&W Babcock & Wilcox
- BWR Boiling Water Reactor
- CCFP Conditional Containment Failure Probability
- CCI Core Concrete Interaction
- CD Core Damage
- CDF Core Damage Frequency
- CE Combustion Engineering
- CET Containment Event Tree
- CFF Containment Failure Frequency
- CHR Containment Heat Removal
- CRD(HS) Control Rod Drive (Hydraulic System)
- CS Containment Spray
- DBA Design Basis Accident
- DCH Direct Containment Heating
- DW Drywell (BWR)
- ECCS Emergency Core Cooling System
- ERVC External Reactor Vessel Cooling
- FSAR Final Safety Analysis Report
- FCI Fuel-Coolant Interaction
- FEM Finite Element Method
- HPCS High Pressure Core Spray
- HPI High Pressure Injection
- HPME High Pressure Melt Ejection
- IPE Individual Plant Examination
- ISLOCA Interfacing System Loss of Coolant Accident
- IVR In-Vessel Recovery or Retention
  [different subjects, same acronym]

Common Acronyms (cont'd)

- LERF Large Early Release Frequency
- LHF Lower Head Failure
- LOCA Loss of Coolant Accident
- LPI Low Pressure Injection
- LPCS Low-pressure Core Spray
- LWR Light Water Reactor
- MAAP Modular Accident Analysis Program
- MACCS MELCOR Accident Consequence Code System
- MCCI Molten Core Concrete Interaction
- MSIV Main Steam Isolation Valve
- OTSG Once-Through Steam Generator
- PCS Power Conversion System
- PDF Probability Density Function
- PDS Plant Damage State
- PORV Power (or Pilot) Operated Relief Valves
- PWR Pressurized Water Reactor
- QHO Quantitative Health Objective
- RCP Reactor Coolant Pump
- RCS Reactor Coolant system
- ROAAM Risk Oriented Accident Analysis Methodology
- RPS Reactor Protection System
- RPV Reactor Pressure Vessel
- RST Revised Source Term
- RWST Refueling Water Storage Tank
- SAMG Severe Accident Management Guidelines
- SBLOCA Small Break LOCA
- SBO Station Blackout
- SERG Steam Explosion Review Group
- SG Steam Generator
- SGTR Steam Generator Tube Rupture
- SRV Safety Relief Valve
- STCP Source Term Code Package
- SV Safety Valve
- TAF Top of Active Fuel (in reactor core)
- TMI-2 Three Mile Island Unit 2
- UHI Upper Head Injection
- VB (Reactor Pressure) Vessel Breach
- WW Wetwell (BWR)

…Thank you for your attention