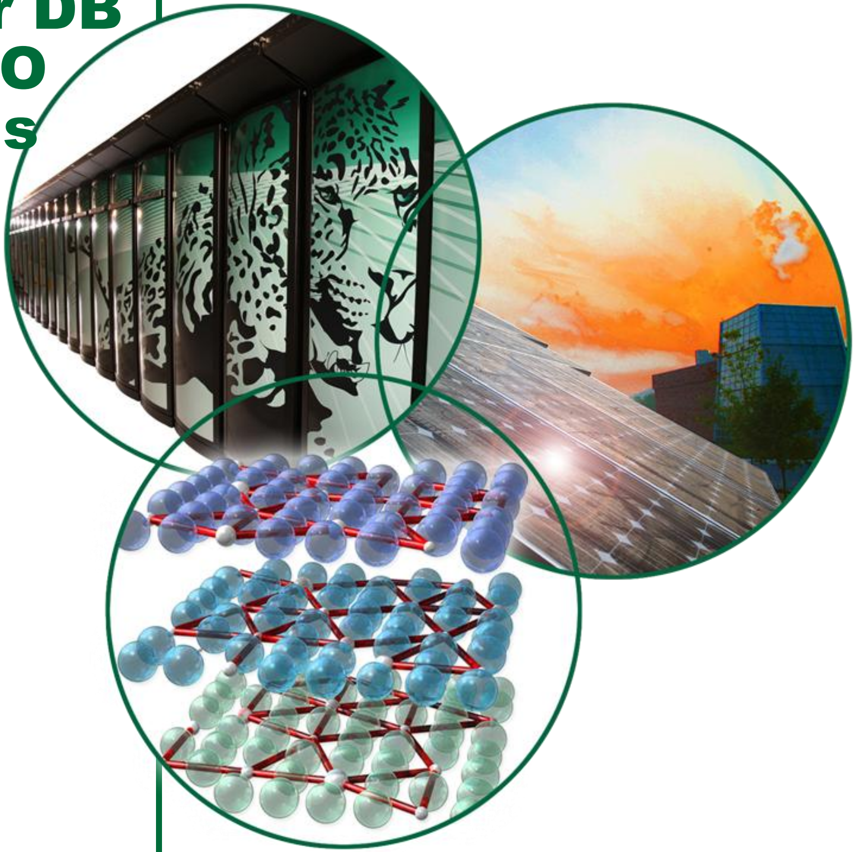
Preliminary Assessment of Impact of Candidate Accident Tolerant Fuels/Cladding on Predicted Reactor Behavior at Normal Operating Conditions and under DB (LOCA and RIA) and BDB (STSBO and LTSBO) Accident Conditions



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U.S. DEPARTMENT OF ENERGY

OAK RIDGE NATIONAL LABORATORY
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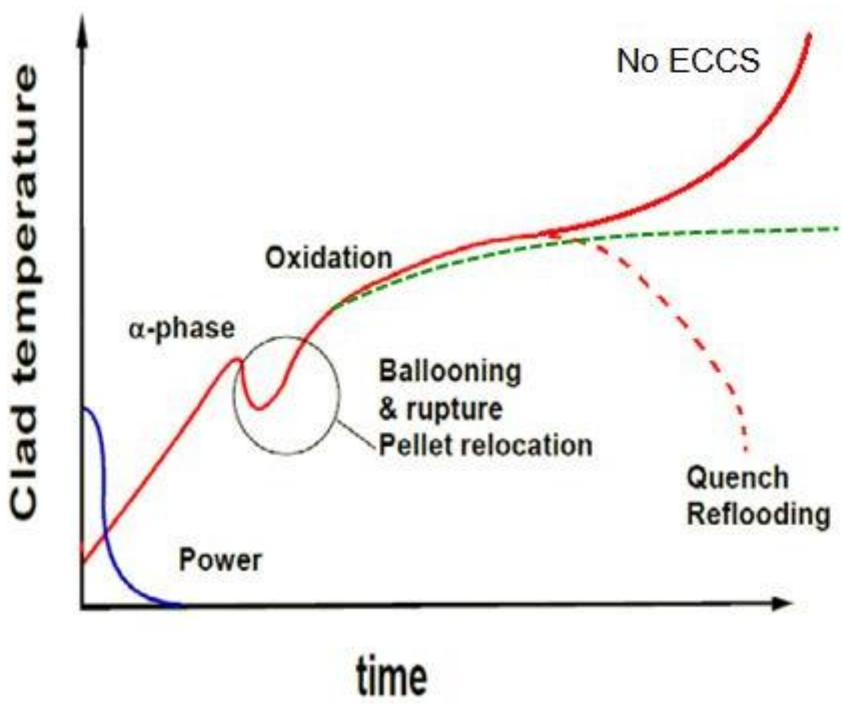
Feedback from Materials Behavior on Accident Progression and Outcome

- Upon uncovering the core, decay heat drives fuel/clad temperature uniformly
- In case of oxide fuel temperatures >1200˚ C cladding oxidation significantly increases fuel temperature and results in rapid hydrogen production



Improvements in safety margins can be achieved by utilization of advanced fuel/cladding materials and configurations that exhibit :

- Slower kinetics of reaction with steam
- Smaller magnitude of enthalpy of oxidation
- Less susceptible to unfavorable interaction with fuel and core components
- Provide additional barriers to fission product retention up to high temperatures

- D.R. Olander, upcoming book on Nuclear Fuels
- E. Kolstad, W. Wiesenack, B. Oberländer, T. Tverberg, "Overview of LOCA Testing Program at Halden," EHPG-Meeting Sandefjord, 2-7 October 2011

Areas For Consideration: <u>all</u> must be considered when developing new fuel/cladding system

Front End
- Fabrication & Inspection
- Transportation & Handling
- Accidents

Current UO_2/Zr system: easy to manufacture, existing infrastructure (cost and qualification are big factors in new systems)

Operation
- Normal Operation
- Operation Accidents
  - Anticipated Operational Occurrences
  - Postulated Accidents (Design Basis)
  - Beyond Design Basis Accidents

Decreasing probability

Preliminary Simulation Focus

Back End
- Storage - Short, Intermed., Long
- Transportation & Handling
- Processing / Fuel Cycle
- Accidents

Cladding integrity issues

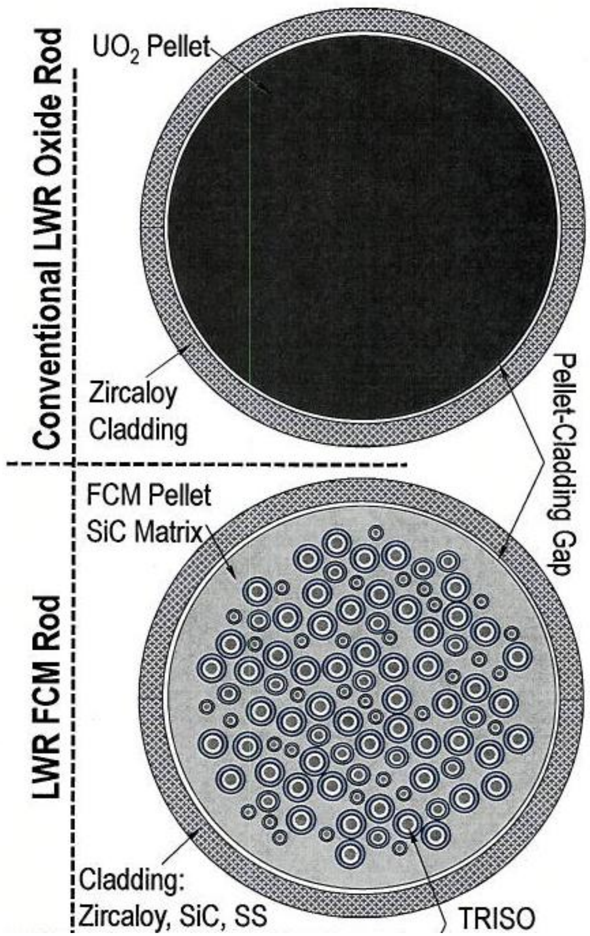
Operation

- **Normal Operation**
- **Operational Accidents**
  - **AOO (~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**

(AOO) Anticipated operational occurrences:
AOO mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit (**for example**, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, or loss of all offsite power)

Normal Operations

Normal Operations

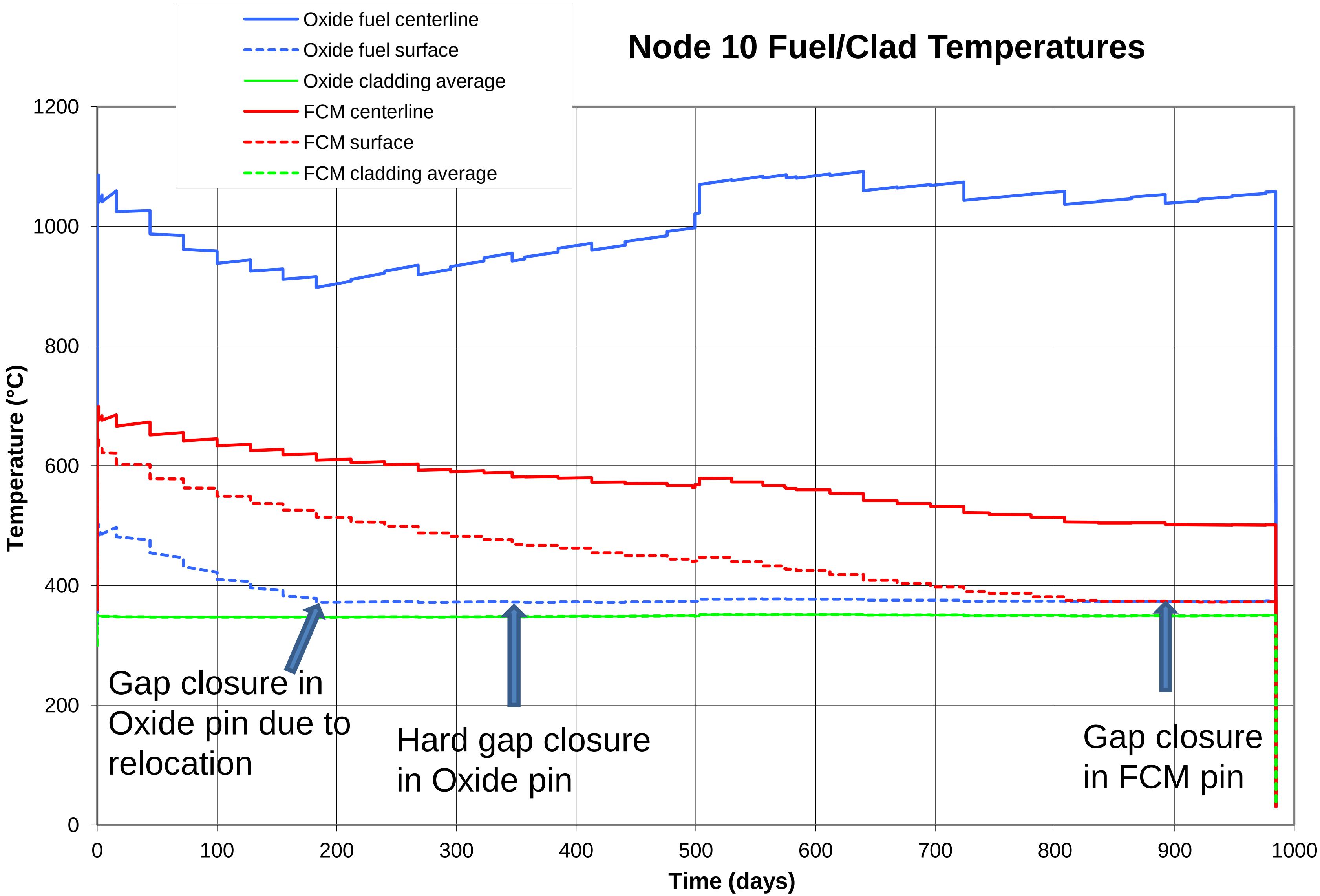


- **UO_2/Zr**
- **FCM/Zr**

•FCM thermal expansion < Oxide
•FCM volumetric swelling during irradiation << oxide
•FCM does not fracture and relocate
•Clad creep is ~ the same
•Essentially no FGR in FCM fuel
•FCM thermal conductivity > Oxide

▪Simulations use identical power operating histories
▪Both simulations use Zr cladding
▪A modified version of FRAPCON-3 used for the analyses

Much Lower Fuel Centerline Temperatures in FCM Fuel

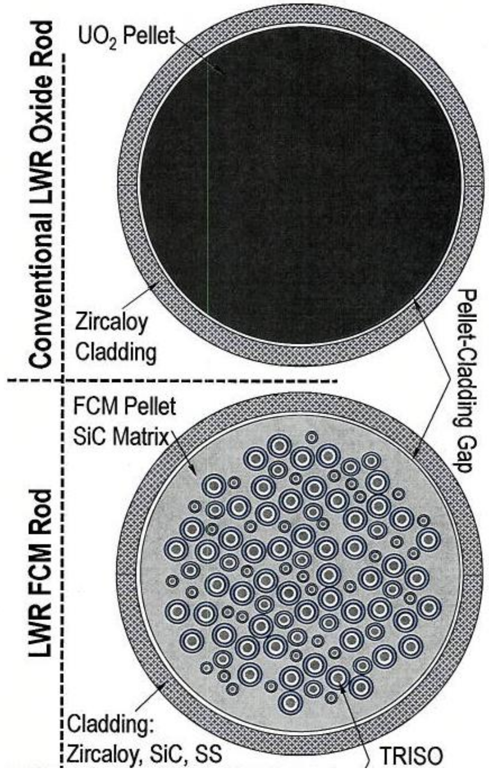


Design Basis Accidents (DBAs)
i.e. RIA or LOCA

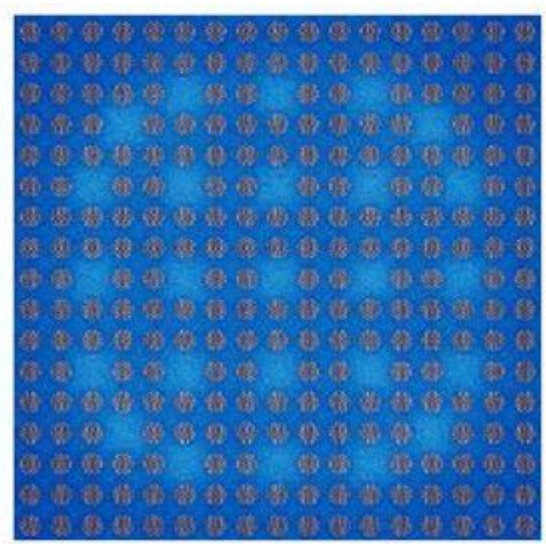
DBAs: RIAs

- With respect to the Reactivity Insertion Accident (RIA) Design Bases Accident in current regulations:
  - Focused on a sudden removal 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, NSRR 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
- RIAs for reactor systems utilizing advanced accident tolerant fuels/cladding currently being studied at BNL (M.Todosow)

Preliminary RIA Simulations (M. Todosow – BNL)



- **AP1000-like 3-D PARCS core model**
  - **SERPENT-PARCS**
- **Based primarily on the AP-1000 FSAR**
- **Some simplifications to design (BOL, only IFBA poison rods, eight fuel types, three reflector types)**
- **Conventional UOX model developed as a benchmark for studies of advanced fuels (e.g. FCM fuel)**

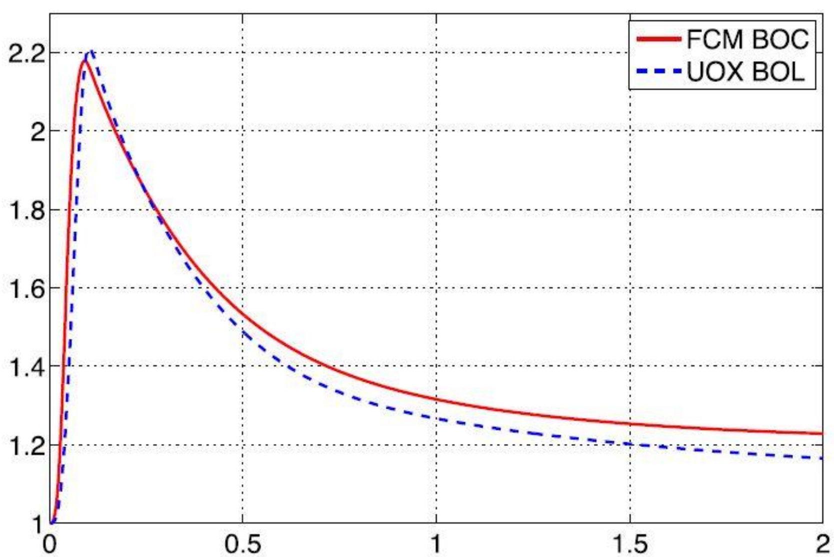


Preliminary RIA Simulations (M. Todosow – BNL)

V	V	V	V	V	V	0	0	0
V	V	V	V	0	0	0	0	0
V	V	V	0	0	0	11	0	12
V	V	0	0	0	6	0	9	0
V	0	0	0	3	0	7	0	5
V	0	0	6	0	8	0	10	0
0	0	11	0	7	0	1	0	4
0	0	0	9	0	10	0	8	0
0	0	2	0	5	0	4	0	2

Rod Bank Ejected

	Configuration
1	Rod bank (OUT)
2	Rod bank (IN)
3	Rod bank (OUT)
4	Rod bank (OUT)
5	Rod bank (Adjusted)
6	Rod bank (Adjusted)
8	Rod bank (OUT)
9	Rod bank (OUT)
10	Rod bank (OUT)
11	Rod bank (OUT)
12	Rod bank (Ejected)
0	Unrodded



FCM ejected worth is \$0.517, UOX ejected worth is \$0.523

- Rod ejection (bank 12) for a BOC FCM and BOL UOX core
- Rod bank configuration similar for FCM and UOX cores
- Bank 5, bank 6 and soluble boron adjusted to give bank 12 similar reactivity worth

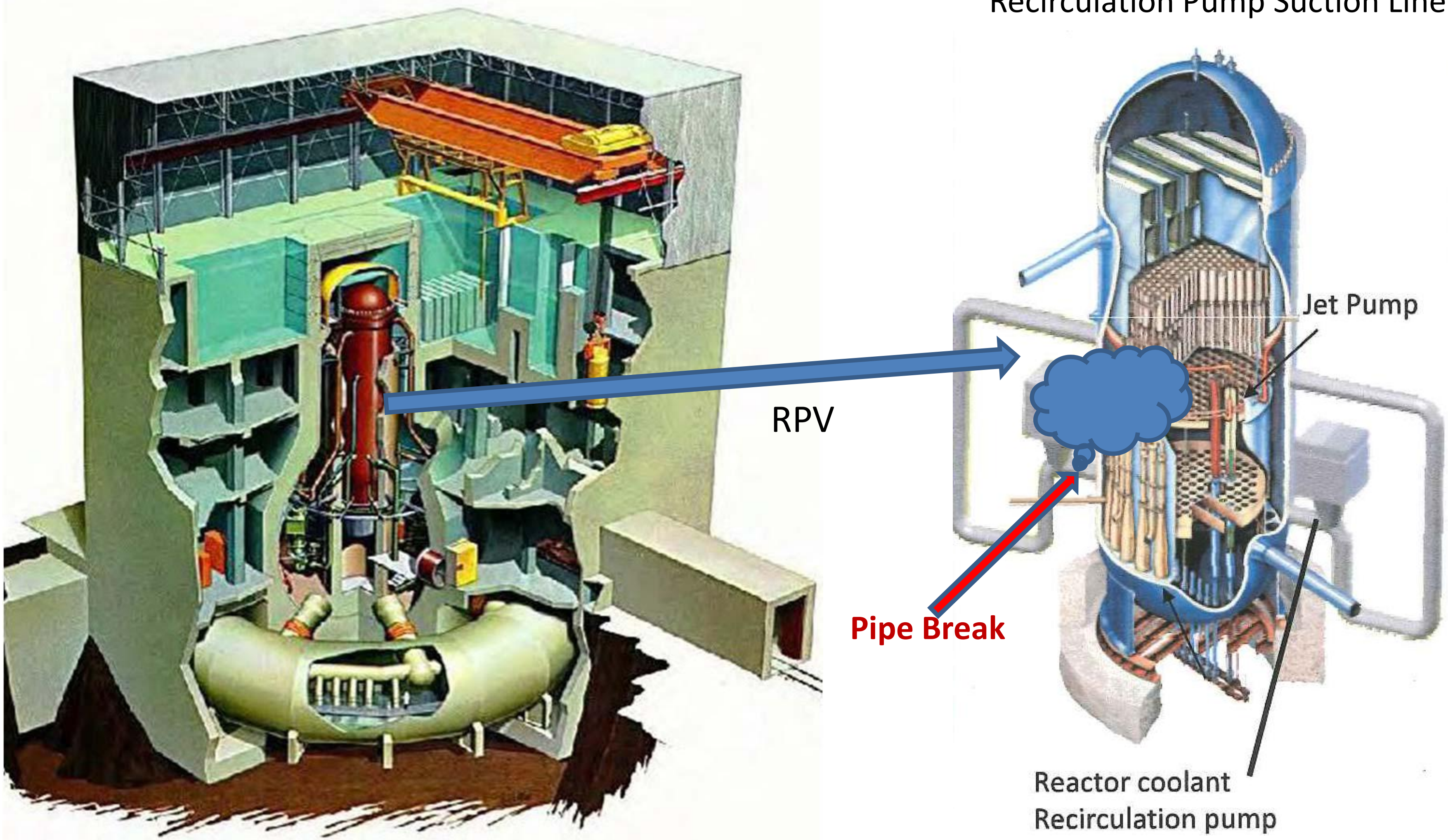
DBAs: LOCA

- 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 > 1204° 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 UO_2 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

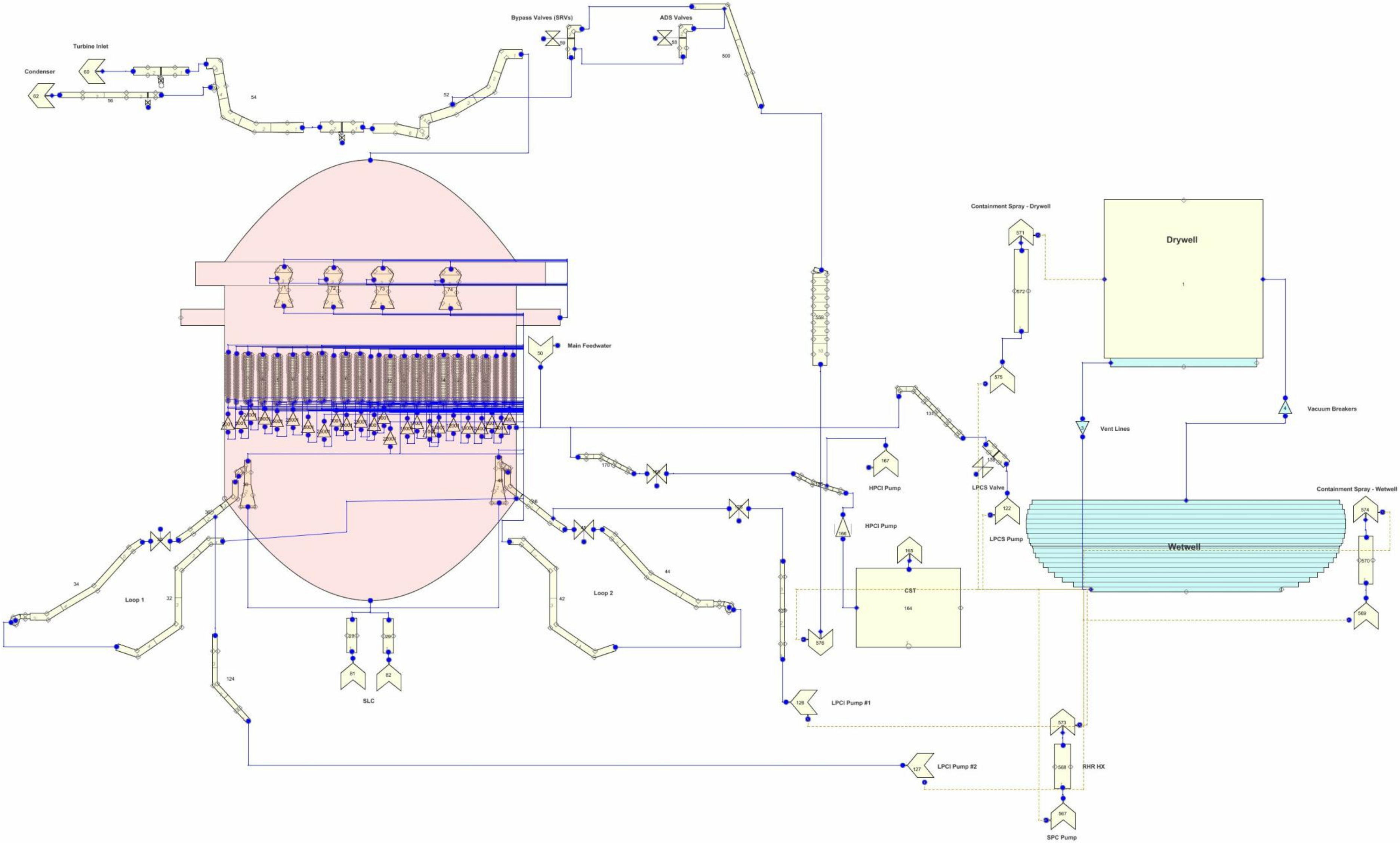
Simulation of a Large Break Loss of Coolant Accident (LBLOCA) in a BWR

GE BWR4 with a Mark I Containment
Basis: Browns Ferry NPP (1155 MWe)

Design Basis Accident: Break Assumed in Reactor Coolant Recirculation Pump Suction Line

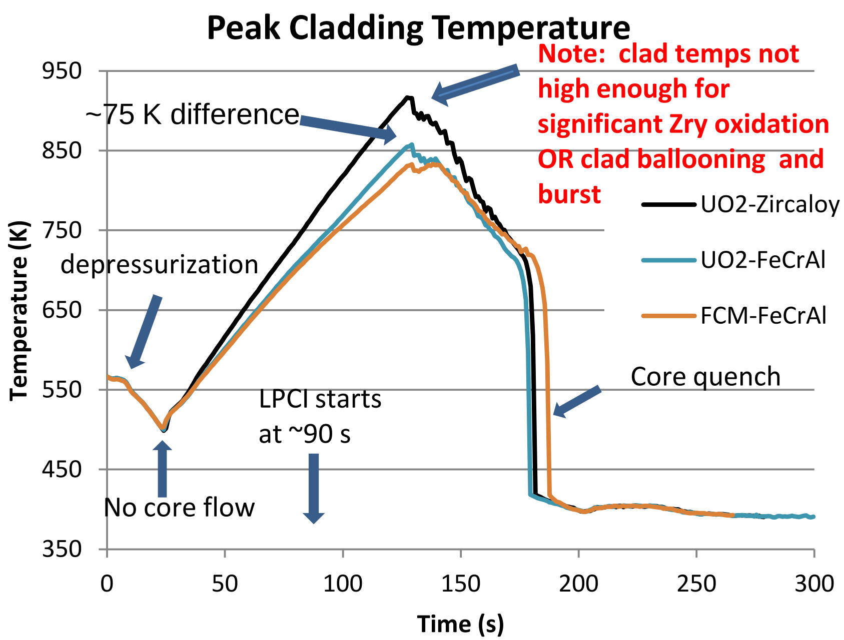


TRACE Model of the BF GE BWR4 with a Mark I Containment

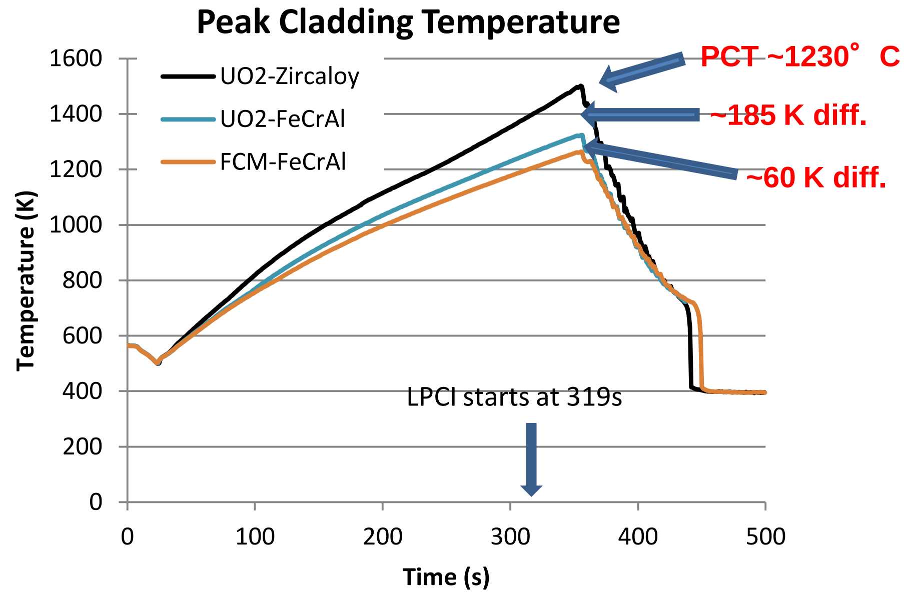


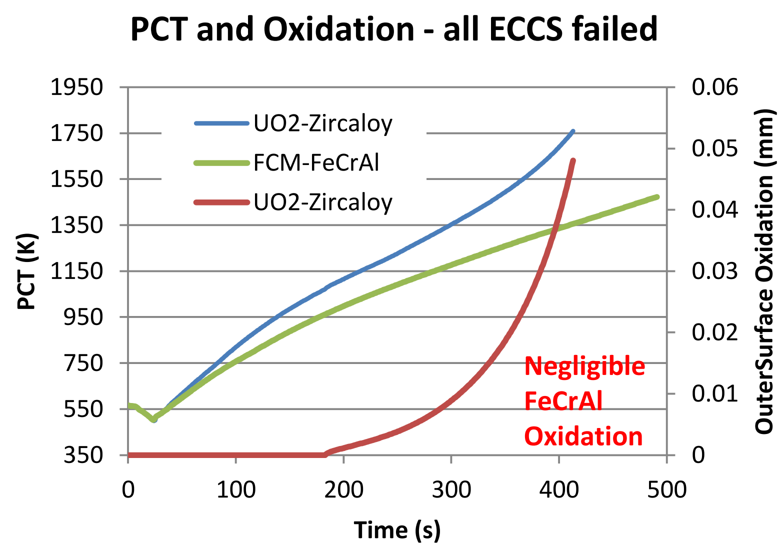
BF LBLOCA Simulation Preliminary Results

Main Cases: 1) LPCI injection at 90 secs



2) LPCI injection at 319 secs





Study Focus:

The effect on the predicted peak cladding temperature (PCT) and clad oxidation for cores consisting of :

- UO_2 fuel and Zry cladding
- UO_2 fuel and FeCrAl cladding
- FCM fuel and FeCrAl cladding

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

Station Blackout Involves Failure of AC Electrical Power

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

Short-Term
Station Blackout

Immediate Loss of
Water Makeup

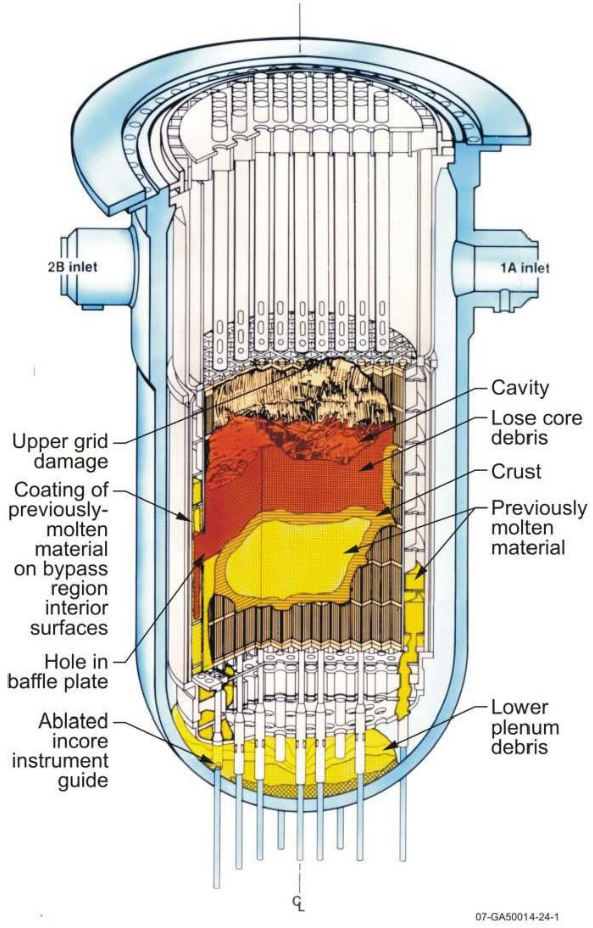
Fukushima Dai-ichi Unit 1

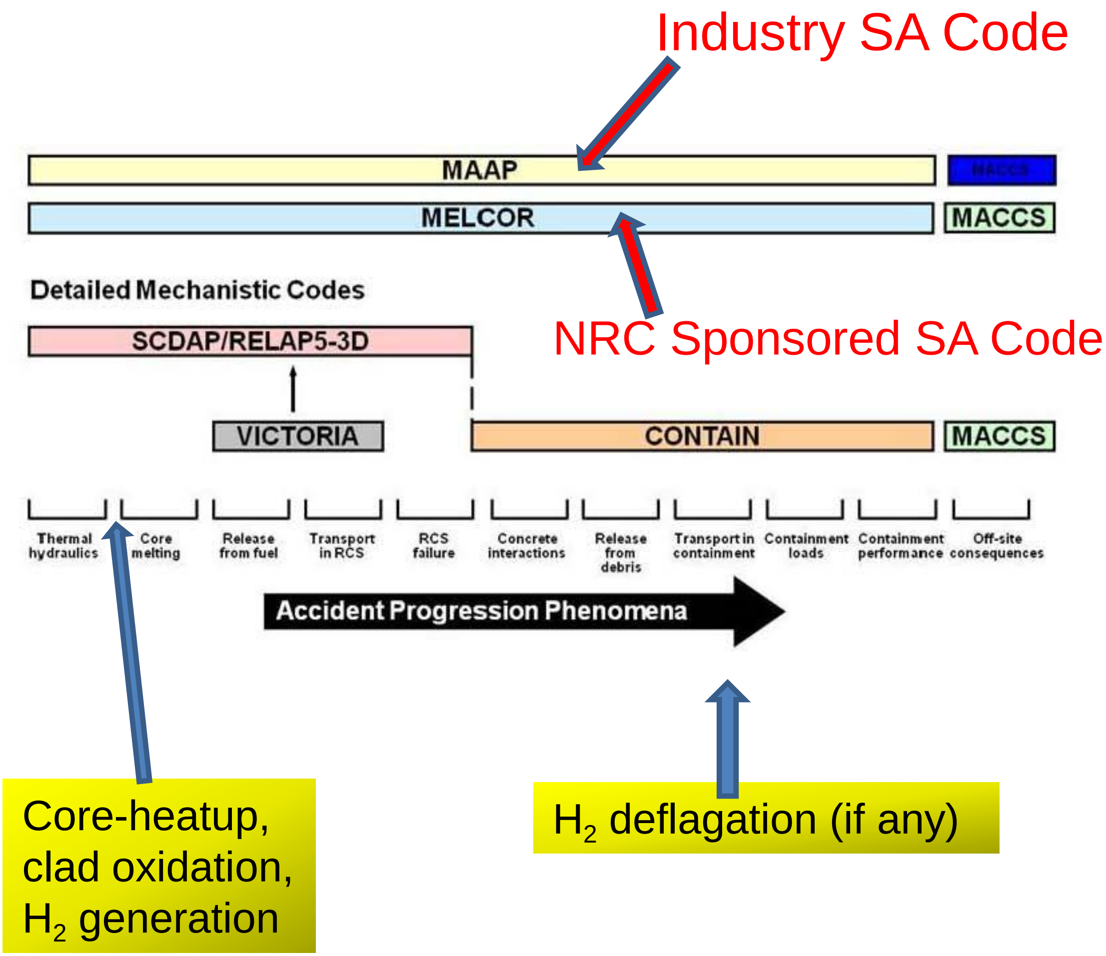
Long-Term
Station Blackout

Loss of Water Makeup
Following Battery
Exhaustion

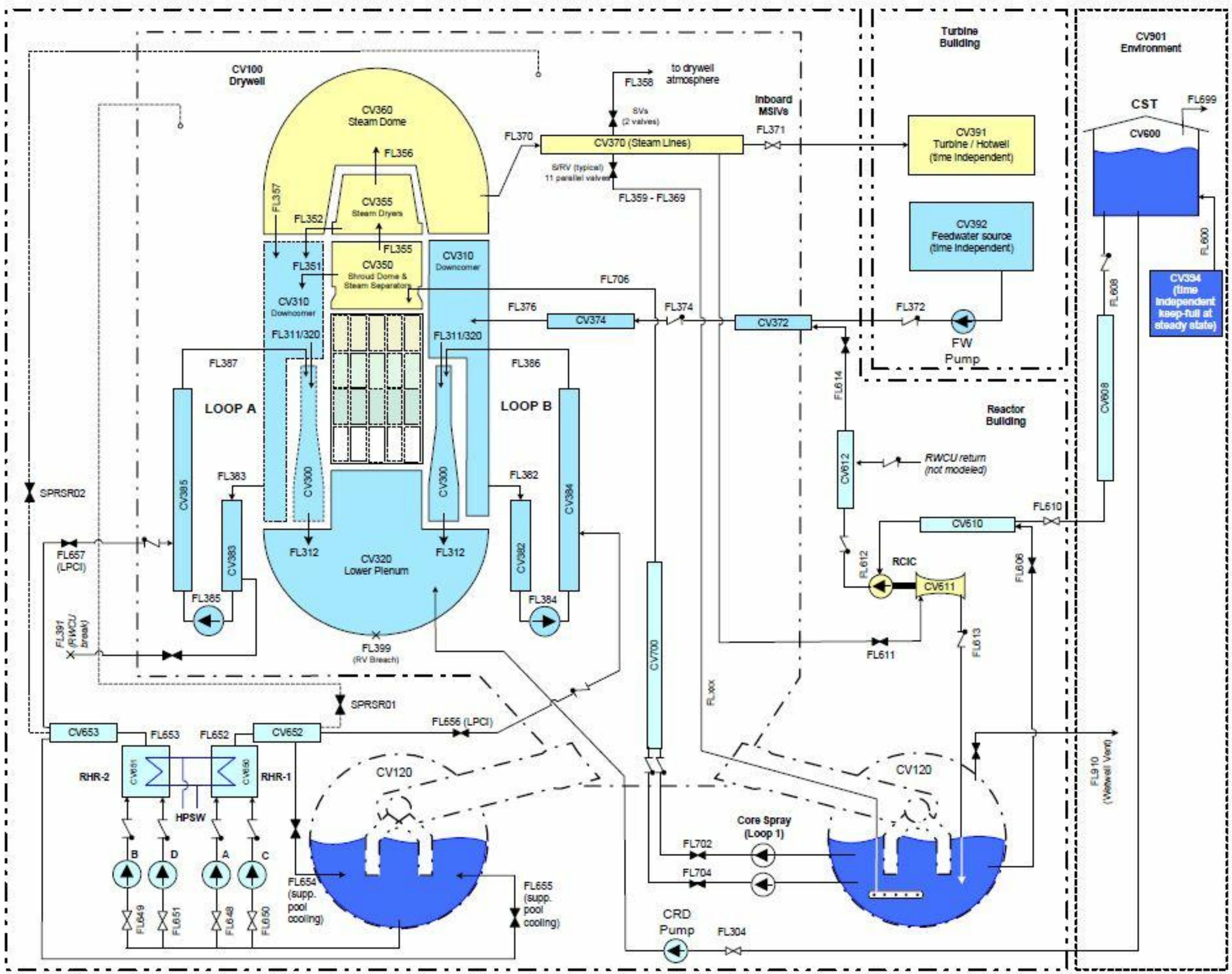
Fukushima Dai-ichi Units 2 and 3

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

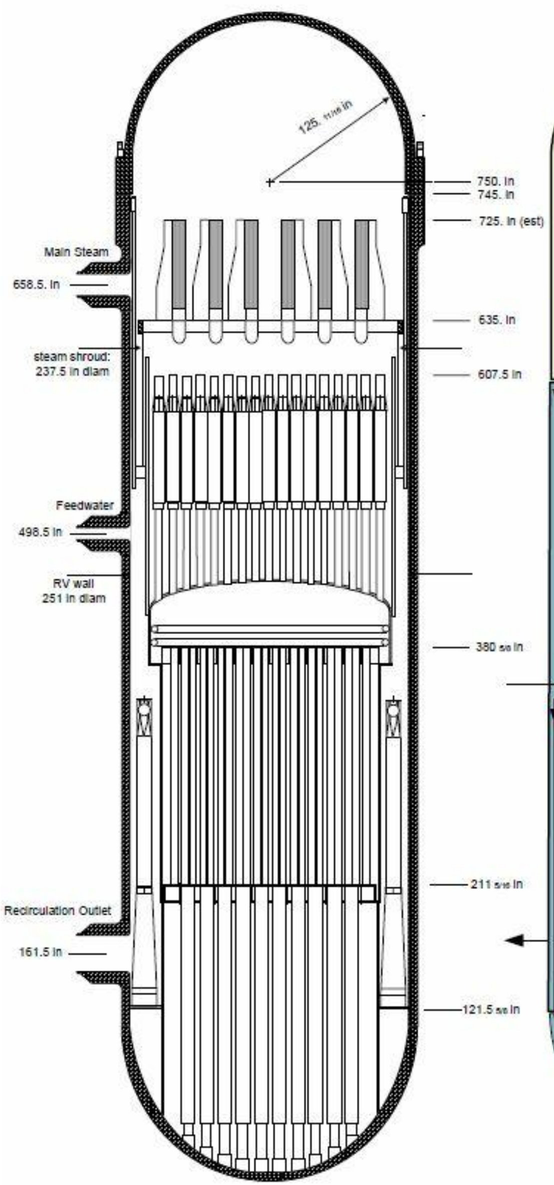


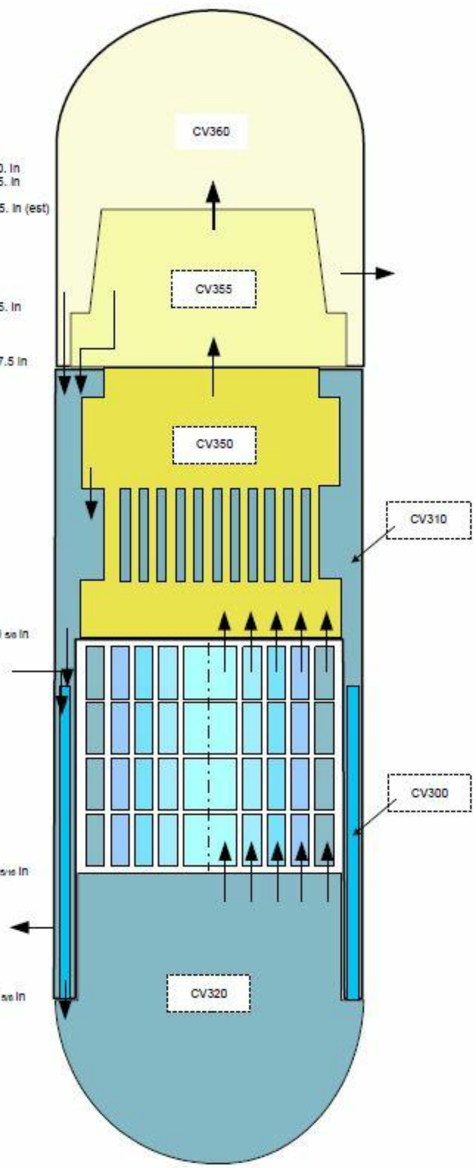


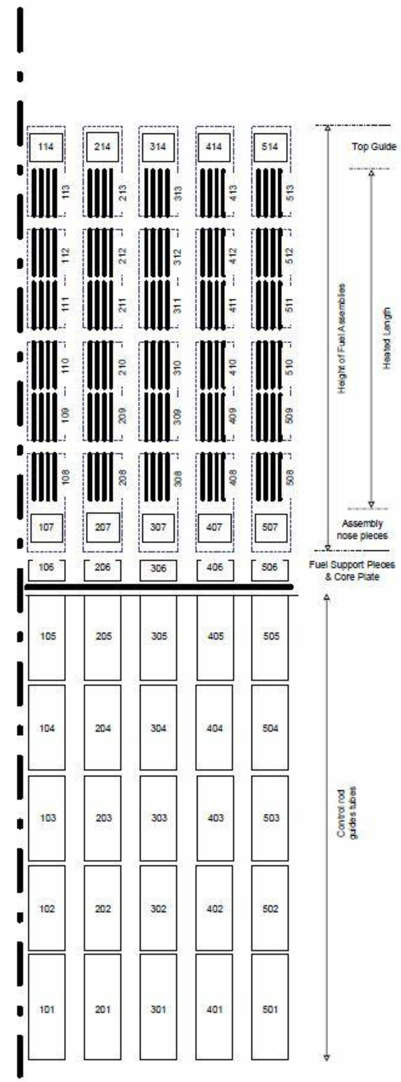
MELCOR Model of the PB GE BWR4 with a Mark I Containment



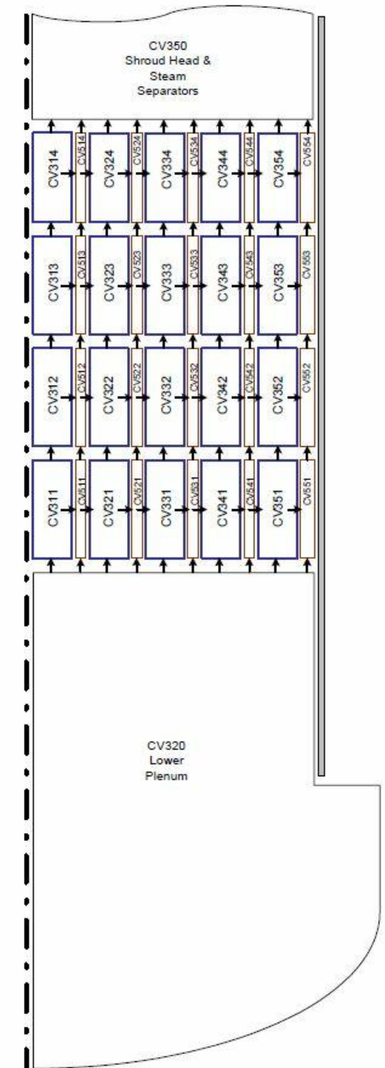
MELCOR Model of the PB RPV and Core







BWR/4 Core Nodalizaation



CVH Nodalization accompanying COR Nodalization

MELCOR Simulation(s) of Beyond Design Basis Accidents

Beyond Design Basis Scenarios:

- STSBO – short term station blackout
  - Loss of all power at time **0** (at SCRAM)
  - No water injection into RPV
- LTSBO – long term station blackout
  - Loss of all power after **8 hrs.**
  - <8hrs: water injection into RPV
  - >8hrs: no water injection into RPV

Basis: Peach Bottom NPP

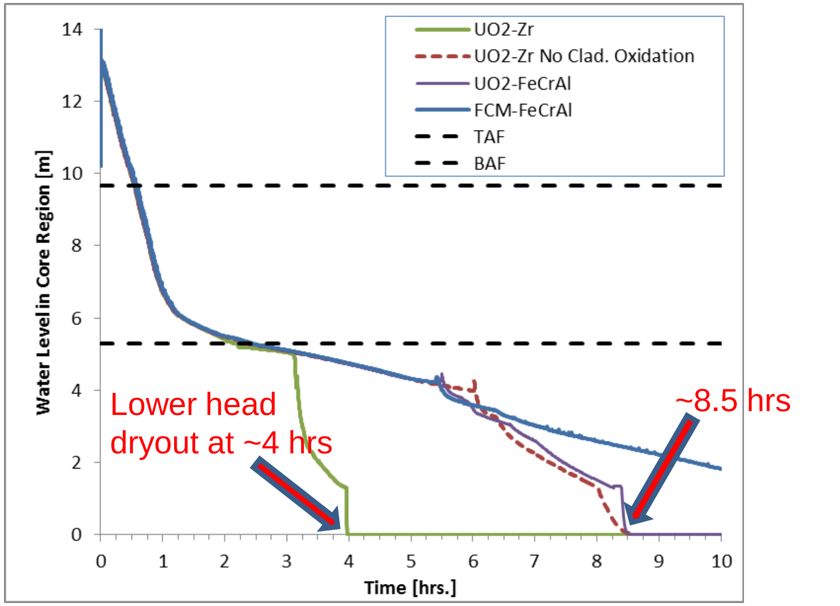
- GE BWR4 with a Mark I Containment
- Similar to Fukushima Daiichi Unit 1 accident (STSBO)

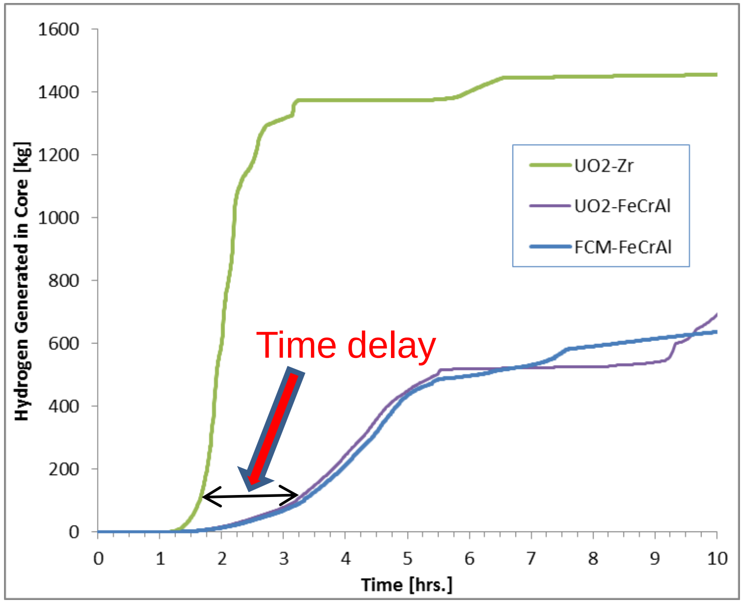
Study Focus:

- *The effect on the predicted peak cladding temperature (PCT) and clad oxidation for cores consisting of :*
  - UO_2 fuel and Zry cladding
  - UO_2 fuel and FeCrAl cladding
  - FCM fuel and FeCrAl cladding

<u>STSBO</u>: General Accident Progression

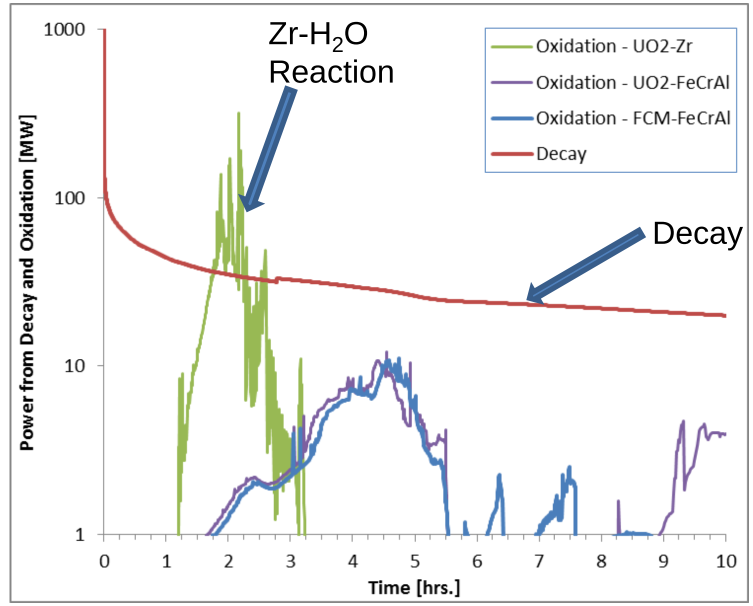
- Water Level
  - Boil down is similar (time to BAF is delayed ~12 min.)
  - FeCrAl core degradation and relocation occurs much slower
  - FeCrAl case similar to ideal case where there is no cladding oxidation
  - Due to material property modeling FCM core remains intact (up to 10+ hrs.)
- NOTE: H_2 required (estimated) for Fukushima explosions (Units 1, 3, and 4)
  - Approximately 100-300 kg
- H_2 generated in STSBO simulations
  - Approximately 55% less generated after 10 hrs.
  - Time delay to 100 kg H_2 compared to UO_2-Zr:
    - 94 min. for UO_2-FeCrAl
    - 101 min. for FCM-FeCrAl

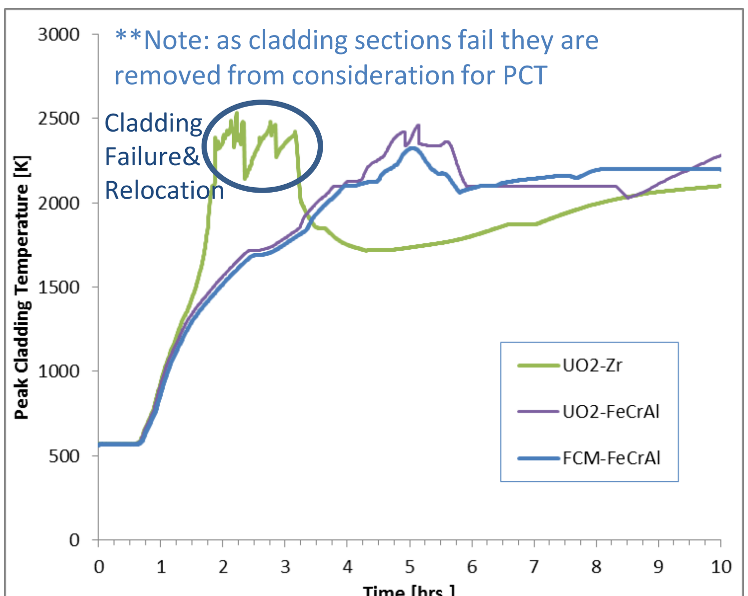


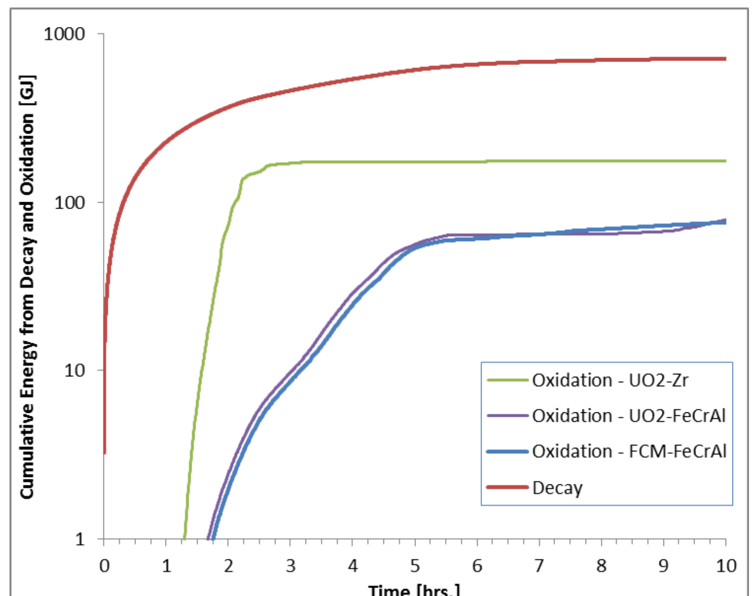


STSBO: General Accident Progression

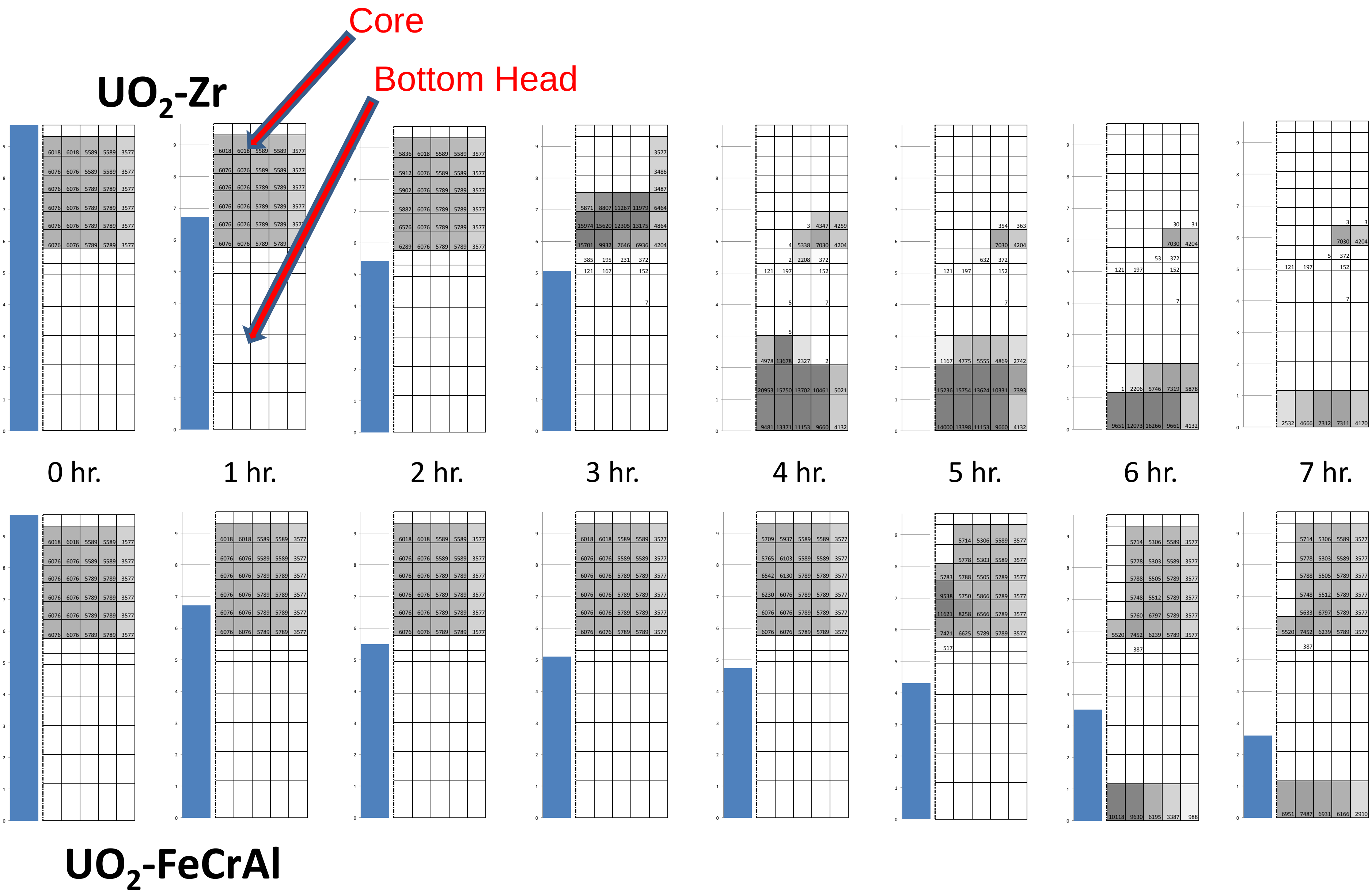
- Lower reaction rate of FeCrAl decreases energy source term
- As currently modeled, the FeCrAl heat of oxidation is the same as for Zr (i.e. much greater than it actually is)
  - (Lowering it would further delay timing)
- Core Peak Cladding Temp. increases much slower
  - Delay to 1204°C is approximately 15-21 min.





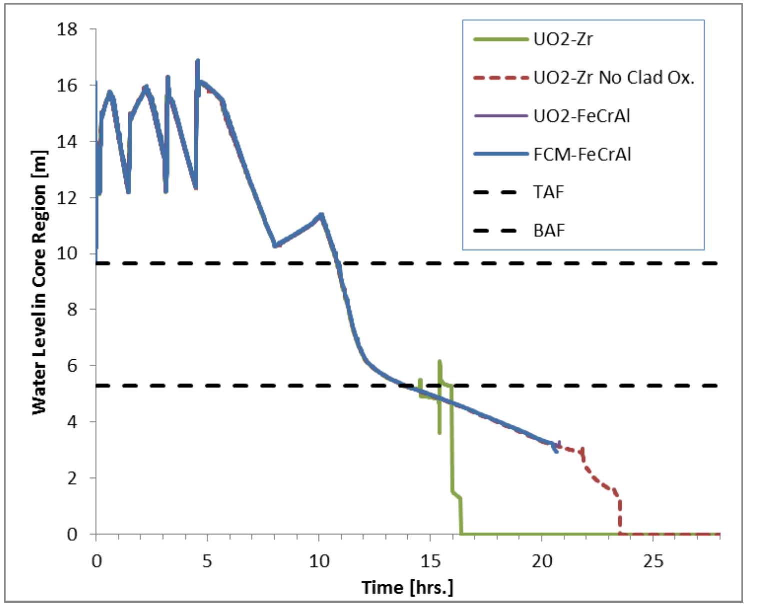


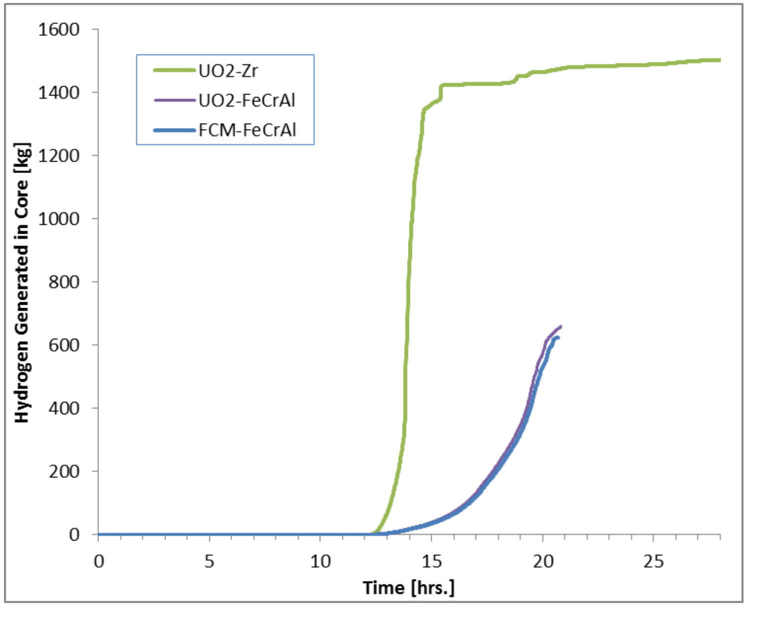
STSBO: Core Position vs Time



LTSBO: General Accident Progression

- Water injection via RCIC or HPCI available until battery depletion 8 hrs after SCRAM
- Water Level
  - Boil down is similar between cases
  - FeCrAl cases behaves like ideal case where there is no oxidation
  - Lower head dry out would be delayed by approximately 7 hours
- H_2 generation occurs much slower
  - Several additional hours before reaching flammable concentrations





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 H_2 generation had been drastically reduced, maybe 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 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 fuels
- Broad range of accident testing needed to understand fuel behavior under accident conditions
  - Currently fuel 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
- These preliminary evaluations are currently being documented for a journal article

Questions ??