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

L. J. Ott

Reactor and Nuclear Systems Division
Oak Ridge National Laboratory

Presented at the OECD/NEA Workshop on Accident Tolerant Fuels for LWRs

NEA Headquarters
Issy-les-Moulineaux, France

10-12 December 2012
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
Areas For Consideration: **all** must be considered when developing new fuel/cladding system

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

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

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

Current UO$_2$/Zr system: easy to manufacture, existing infrastructure (cost and qualification are big factors in new systems)

Preliminary Simulation Focus

Cladding integrity issues
Operation

• Normal 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

(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

- $\text{UO}_2$/Zr
- FCM/Zr

- FCM thermal expansion $< \text{Oxide}$
- FCM volumetric swelling during irradiation $<< \text{oxide}$
- FCM does not fracture and relocate
- Clad creep is $\sim$ the same
- Essentially no FGR in FCM fuel
- FCM thermal conductivity $> \text{Oxide}$

- 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

Node 10 Fuel/Clad Temperatures

- Oxide fuel centerline
- Oxide fuel surface
- Oxide cladding average
- FCM centerline
- FCM surface
- FCM cladding average

Gap closure in Oxide pin due to relocation
Hard gap closure in Oxide pin
Gap closure in FCM pin

Temperature (°C)

Time (days)
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)
• 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

FCM ejected worth is $0.517, UOX ejected worth is $0.523
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 $> 1204^\circ$ 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

RPV

Pipe Break

Jet Pump

Reactor coolant Recirculation pump
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

![Graph showing peak cladding temperature over time for different scenarios.]

- LPCI starts at 90 s
- ∼75 K difference
- Core quench
- No core flow
- Depressurization

Note: clad temps not high enough for significant Zry oxidation OR clad ballooning and burst

2) LPCI injection at 319 secs

![Graph showing peak cladding temperature over time for different scenarios.]

- PCT ∼1230° C
- ∼185 K diff.
- ∼60 K diff.

Core quench

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

- UO₂ fuel and Zry cladding
- UO₂ 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

- Core-heatup, clad oxidation, \( H_2 \) generation
- \( H_2 \) deflagration (if any)

Industry SA Code

NRC Sponsored SA Code
MELCOR Model of the PB GE BWR4 with a Mark I Containment
MELCOR Model of the PB RPV and Core
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
STSBO: 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**: H₂ required (estimated) for Fukushima explosions (Units 1, 3, and 4)
  - Approximately 100-300 kg

- **H₂ generated in STSBO simulations**
  - Approximately 55% less generated after 10 hrs.
  - Time delay to 100 kg H₂ compared to UO₂-Zr:
    - 94 min. for UO₂-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.

**Note:** as cladding sections fail they are removed from consideration for PCT Cladding Failure & Relocation

**Zr-H₂O Reaction**

**Decay Heat**

**Oxidation - UO₂-Zr**
**Oxidation - UO₂-FeCrAl**
**Oxidation - FCM-FeCrAl**
**Decay**
STSBO: Core Position vs Time

UO₂-Zr

0 hr.  1 hr.  2 hr.  3 hr.  4 hr.  5 hr.  6 hr.  7 hr.

0 hr.  1 hr.  2 hr.  3 hr.  4 hr.  5 hr.  6 hr.  7 hr.

UO₂-FeCrAl
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

• $\text{H}_2$ generation occurs much slower
  – Several additional hours before reaching flammable concentrations
Higher Melting / Lower H₂ 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, 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 H₂ 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 B₄C 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 ??