



Preliminary assessment of accident-tolerant fuels on LWR performance during normal operation and under DB and BDB accident conditions[☆]



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ABSTRACT

Following the severe accidents at the Japanese Fukushima Daiichi Nuclear Power Station in 2011, the US Department of Energy initiated research and development on the enhancement of the accident tolerance of light water reactors by the development of fuels/cladding that, in comparison with the standard UO₂/Zircaloy (Zr) system, can tolerate loss of active cooling in the core for a considerably longer time period while maintaining or improving the fuel performance during normal operations. Analyses are presented that illustrate the impact of these new candidate fuel/cladding materials on the fuel performance at normal operating conditions and on the reactor system under DB and BDB accident conditions.

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1. Background

Following the disastrous severe accidents at the Japanese Fukushima Daiichi Nuclear Power Station, caused by the 2011 Tohoku earthquake and tsunami, enhancing the accident tolerance of light water reactors (LWRs) became a hot topic in the US Congress. As a result, the Consolidated Appropriations Act of 2012 directed the Nuclear Energy branch of the US Department of Energy (DOE) to do the following:

- Give “priority to developing enhanced fuels and cladding for light water reactors to improve safety in the event of accidents in the reactor or spent fuel pools”.
- Give “special technical emphasis and funding priority ... to activities aimed at the development and near-term qualification of meltdown-resistant, accident-tolerant nuclear fuels that would enhance the safety of present and future generations of light water reactors”.
- Report “to the Committee ... on its plan for development of meltdown-resistant fuels leading to reactor testing and utilization by 2020.” A draft report was completed but not issued, pending Congressional approval.

At the beginning of FY 2013, The DOE Fuel Cycle Research and Development (FCRD) Advanced Fuels Campaign (AFC) was

reorganized to include research, development, and demonstration (RD&D) on Accident-Tolerant Fuels (ATF) with the goal of inserting a lead test assembly (LTA) or lead test rod (LTR) into a commercial reactor by 2022 [1].

By definition, ATFs are fuels/cladding that, in comparison with the standard UO₂/Zircaloy (Zr) system, can tolerate loss of active cooling in the core for a considerably longer time period while maintaining or improving the fuel performance during normal operations. There are three major potential approaches for the development of ATFs:

- Improved fuel properties.
- Improved cladding properties to maintain core coolability and retain fission products.
- Improved reaction kinetics with steam to minimize enthalpy input and hydrogen generation.

There are constraints on new fuel/cladding designs, such as:

- Backward compatibility (must be qualified in an existing reactor),
- Economics.
- Minimal impact on the fuel cycle.
- Minimal impact on the plant operations.
- Significant impact (improvement) on plant safety (for the entire spectrum of design basis [DB] accidents and beyond design basis [BDB] accidents).

During FY 2012, significant new capabilities [2] were established by DOE's FCRD Program at the Oak Ridge National Laboratory to examine alternative candidates for fuel and cladding:

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- High-temperature furnace (up to 1700 °C, with a steam and hydrogen atmosphere).
- High-pressure furnace (up to 1400 °C, with a steam and hydrogen atmosphere at 2 MPa pressures).
- Thermogravimetry facilities (up to 1500 °C, with a steam and hydrogen atmosphere).
- Loss-of-coolant accident (LOCA) integral test furnace (up to 1204 °C, with a steam and hydrogen atmosphere, with quench capabilities).

The LOCA station currently can test unirradiated fuel/cladding, and in FY 2013 a comparable facility will be established in the hot cells in the Irradiated Fuel Examination Laboratory (IFEL) at ORNL to examine irradiated fuel/clad segments in an integral manner. These complementary facilities have been established to examine the behavior of advanced fuel/cladding materials under severe accident conditions.

During FY 2012, several candidate cladding materials (such as 310 SS, FeCrAl, and SiC) were demonstrated at ORNL to have significantly lower oxidation kinetics and hydrogen generation [3]. Also, a new fuel form (a fully ceramic micro-encapsulated [FCM] matrix with TRISO fuel particles, the ceramic being silicon carbide [SiC]) was shown to have superior structural integrity with minimal swelling under irradiation conditions [4].

The purpose of this paper is to illustrate the impact of these new candidate fuel/cladding materials on the fuel performance at nor-

mal operating conditions and on the reactor system under DB and BDB accident conditions. Section 2 provides an overview of the thermo-physical properties and oxidation kinetics of these materials. Section 3 addresses the fuel performance during normal operations. Section 4 focuses on two DB accidents, and Section 5 presents the results of BDB accident simulations.

2. Thermo-physical properties and oxidation kinetics of candidate materials

2.1. Thermo-physical properties

The fuel material selected for this study is the fully ceramic micro-encapsulated (FCM) fuel. This concept was originally developed for use in high-temperature gas-cooled reactors in which the TRISO particles would be dispersed within a graphite matrix but has since been adapted for use in LWR fuel elements using a SiC matrix. The SiC matrix improves irradiation stability, provides another effective barrier to fission product release, performs with greater reliability under the anticipated operating and long-term storage conditions, and provides greater resistance towards proliferation [4,5]. In this study, thermal properties of the FCM fuel are assumed to be the same as those for SiC [6–8] since actual data for the FCM are not currently available. This is a reasonable assumption since the FCM fuel pellet is largely SiC.

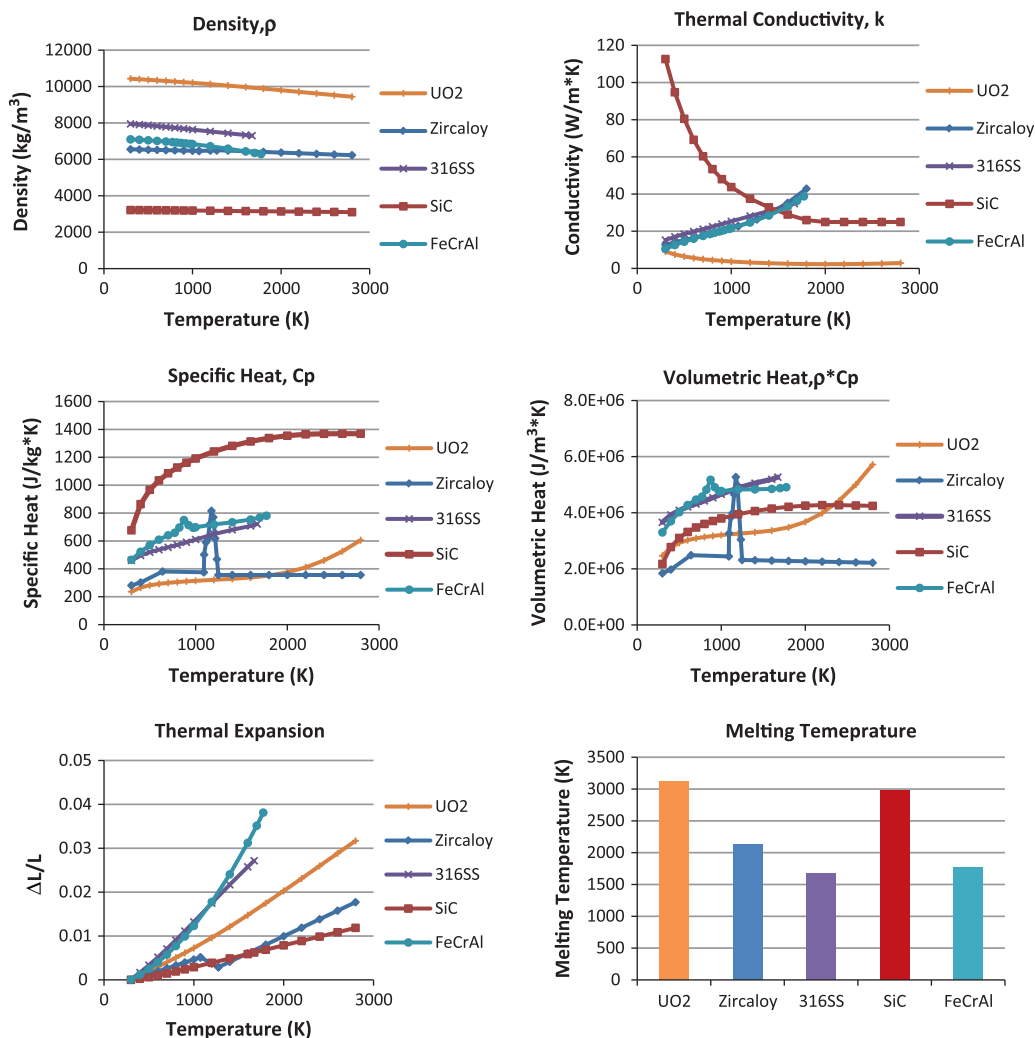


Fig. 2.1. Material thermo-physical properties.

The cladding materials to be examined are 316SS, SiC, and FeCrAl. 316SS contains Fe with 17% Cr, 12% Ni, 2.5% Mo, and 0.03% C [9,10]. FeCrAl consists of $\geq 69.52\%$ Fe, 20.5–23.5% Cr, 5.8% Al, $\leq 0.08\%$ C, $\leq 0.4\%$ Mn, and $\leq 0.7\%$ Si [10,11].

The thermal properties are presented in Fig. 2.1 for the selected fuel and cladding materials, as well as conventional UO_2 fuel [12,13] and Zircaloy cladding [12,13].

Thermal conductivity values shown do not consider the effects of neutron irradiation. It is observed that the thermal conductivity of SiC is reduced by $>30\%$ after irradiation. The FCM fuel (SiC) has much higher thermal conductivity than the UO_2 fuel, which means the FCM fuel will have a much lower centerline temperature and smaller radial temperature gradient within the fuel pellet at full power compared to the UO_2 fuel. Volumetric heat capacity of the FCM fuel is slightly higher than that of the UO_2 fuel, although its specific heat capacity is much higher than that of the UO_2 fuel. Additionally, SiC has lower thermal expansion and volumetric swelling (during irradiation) than UO_2 .

Zircaloy has a sharp spike in specific heat capacity because it undergoes a phase change (alpha to beta) from 1090 to 1248 K. FeCrAl also demonstrates a peak (at ~ 800 K) in the specific heat capacity due to a magnetic phase transition. SiC has a higher specific heat capacity than metal materials, while its volumetric heat capacity values are lower than those of 316SS and FeCrAl. Note that volumetric heat capacity (ρ^*C_p) and thermal conductivity play a key role in the heat-up of the fuel and cladding during a LOCA. The fuel and cladding with higher volumetric heat capacity will have a slower and lower temperature increase in fuel and cladding (i.e., slower transient thermal response). The lower the thermal conductivity, the higher the initial component temperatures and thus the higher the initial stored energy. However, we have found that initial stored energy has almost no effect on the heat-up of the fuel and cladding in a boiling water reactor (BWR) LOCA, though it shows some effect in a pressurized water reactor (PWR) LOCA [14]. Hence, it is expected that the FCM fuel with the FeCrAl or 316SS cladding will perform best in heat-up during a LOCA.

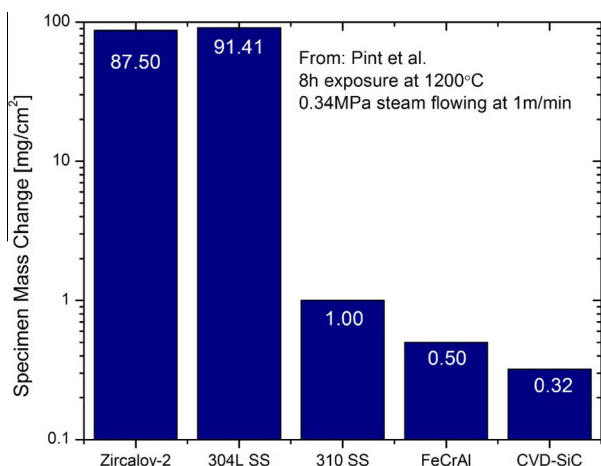


Fig. 2.2. Mass change at 1200 °C in steam at 8 h duration.

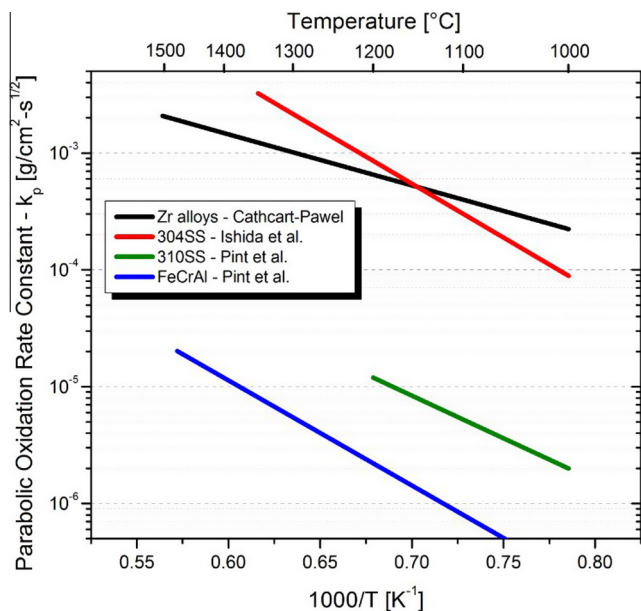


Fig. 2.3. Material oxidation kinetics.

2.2. Cladding oxidation kinetics

The oxidation of these candidate cladding materials is extensively discussed in an accompanying paper [3] in this journal. A summary of the studies is illustrated in Figs. 2.2 and 2.3. As shown, these advanced iron-based alloys and SiC materials offer significant improvements over zirconium alloys and conventional stainless steels with respect to oxidation kinetics.

3. Fuel Behavior at normal operating conditions

The FRAPCON-3 fuel performance code [12] is the primary audit code for the US Nuclear Regulatory Commission (NRC); it has been developed and is maintained for the NRC by the Pacific Northwest National Laboratory (PNNL). The code models nuclear fuel pin performance under normal operating conditions in a LWR environment.

For these simulations, the thermo-physical properties of SiC were incorporated into the MATPRO routines within the FRAPCON-3 code (thus allowing the use of FCM pellets). The FCM (fuel) properties required by FRAPCON-3 were (1) the specific heat, (2) enthalpy, (3) Young's modulus, (4) surface emissivity, (5) Poisson's ratio, (6) thermal conductivity, (7) thermal expansion, and (8) densification and swelling under irradiation conditions. Besides modifying the appropriate MATPRO routines, the information flow and logic within the code were modified to allow the modeling of fuel pins loaded with an alternate fuel form (ergo, FCM pellets). For these code modifications, no changes (or additions) were made to the cladding materials or user cladding options.

The modified FRAPCON-3 code [15] was exercised on a case that had recently been analyzed (an oxide fuel pin that was irradiated in an LWR for two cycles). These simulations (one using an oxide fuel and the second FCM compacts) use the same rod operating powers (average linear heat generation rates [LHGR]), as shown in Fig. 3.1, and axial rod power shape [not shown] and the same initial fuel pin and fuel physical dimensions. Also, since there is negligible TRISO particle failure below 1600 °C, it is assumed that there is negligible fission gas release from the FCM compact; and since there is significantly lower thermal stresses within the FCM compact due to lower compact temperatures and radial temperature gradients, it is also assumed that the compact does not fracture and "relocate" as is currently modeled for oxide fuel. Both these assumptions are reflected in these FRAPCON-3 simulations, and both assumptions impact the fuel-clad gap and the gap conductance.

These simulations employ a Zircaloy cladding primarily because the metallic cladding thermal conductivities are very similar (see Section 2.1) and the creep models for Zircaloy are well established. There are no similar models available for the other metallic claddings under normal LWR operating conditions (pressures, temperatures, and fluences). Thus these simulations primarily focus on the fuel behavior.

For these simulations, the oxide fuel operates at much higher temperatures (see Fig. 3.2, solid blue curve) primarily due to the much lower oxide thermal conductivity. Note in Fig. 3.2 that the cladding temperature (bottom curve, green) is the same for both simulations (as it should be since the power generation and fluid conditions are the same in both cases). The centerline temperatures for the FCM compact (solid red curve) are 300–550 °C lower than the oxide fuel for the same reactor operating conditions—even though the oxide fuel surface temperature is lower than the FCM compact (due to higher thermal expansion, swelling, relocation, swifter gap closure, as shown in Figs. 3.2 and 3.3). The low radial temperature gradient in the FCM compacts justifies the assumption that the compact will not fracture and relocate.

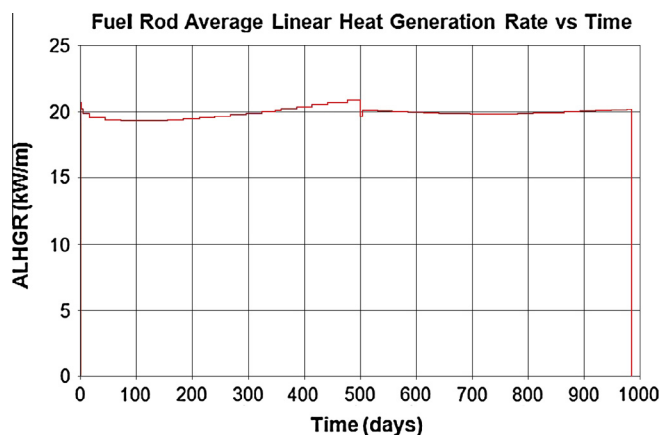


Fig. 3.1. Fuel rod average linear heat generation rate versus time.

The pressure in the fuel pins is higher for the oxide rod due to the lower fuel pin void volume—because of the higher oxide fuel thermal expansion and swelling which closes the radial fuel/clad gap (Fig. 3.3a at about 350 days) and the higher fission gas release. Note the rate of clad creep-down onto the fuel is approximately the same through ~350 days (Fig. 3.3a and b). Swelling is much lower and saturates early in the irradiation in the FCM pellet; and gap closure occurs late (at ~900 days, Fig. 3.3b) in the irradiation, primarily due to clad creep-down.

As expected, in comparing a fuel (FCM) with higher thermal conductivity and better structural characteristics (thermal conductivity that is higher than that of oxide fuel, low and stable swelling, low thermal expansion, high mechanical strength) versus oxide fuel (low thermal conductivity, high swelling, etc.), the thermal performance of the FCM yields much lower centerline temperatures and temperature gradients within the fuel. Even operating at normal LWR powers, the temperatures in the FCM compacts (pellets) are well below temperatures (by at least a 1000 °C) at which TRISO fuel would fail and release fission products; thus, the assumption that there is no fission gas release from the FCM fuel form is reasonable. Also, the lower radial thermal gradients in the FCM fuel (Fig. 3.2) yield lower thermal stresses in the FCM pellet (unlike the UO₂ pellet), which will result in much less pellet cracking, less fission gas release, and less “fines” generation that could be released upon a cladding breach. The integrity of the FCM pellet (with surrogate TRISO fuel) has been demonstrated in High Flux Isotope Reactor (HFIR) test irradiations [4].

The assumption that the FCM fuel can operate at the same power levels and cycle lengths (Fig. 3.1) is questionable since at least half the oxide fuel mass in equivalent pellets has been replaced with inert SiC and the TRISO particles would probably need to use higher ²³⁵U enrichments to achieve comparable LHGRs. These are core design issues that are not addressed in these analyses.

4. DB accident analyses

4.1. Introduction

For this paper only the Reactivity Insertion Accident (RIA) and the LOCA DB accidents will be discussed.

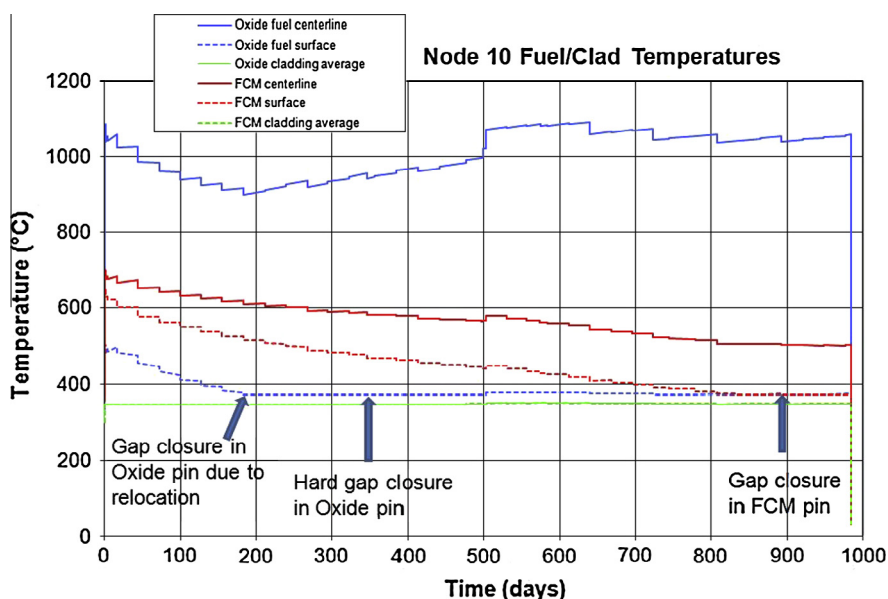


Fig. 3.2. Fuel/clad temperatures.

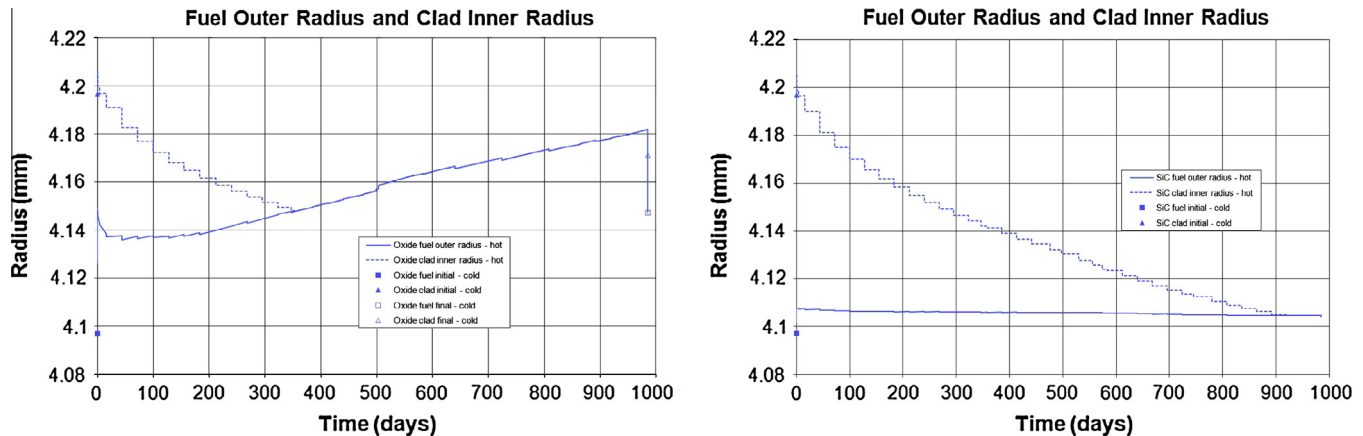


Fig. 3.3. (a) Oxide fuel/clad radii (gap) behavior and (b) FCM fuel/clad radii (gap) behavior.

4.2. RIA DB accident

With respect to the RIA DB accident in current regulations, the RIA is initiated by a sudden removal of a control element (rod in a PWR, blade in a BWR) from the core with the reactor at power with UO_2 fuel and zirconium alloy cladding. Past research has focused on the determination of limits on the deposited energy (J/g) in the fuel due to the reactivity insertion. (These limits are functions of the zirconium alloy cladding oxidation and hydride content.) Simply stated, the core must remain in a coolable geometry without dispersal of fuel into the coolant, and the limits are the maximum powers for which the coolable geometries (and no fuel dispersion) are maintained.

These “limits” historically have been determined experimentally (for example, in the US Transient Reactor Test Facility (TREAT), the Japanese Nuclear Safety Research Reactor (NSRR) and the French CABRI reactor); new fuel/cladding systems would need these RI “limits” to be determined experimentally (ergo, domestically in TREAT).

These RIA limits essentially determine the allowable control rod worths in the core design.

The DOE FCRD program is currently funding RIA analyses at Brookhaven National Laboratory (BNL) for reactor systems utilizing advanced ATF (specifically the FCM fuel form). Preliminary analyses [16], based on a AP1000 reactor design, indicates that FCM cores behave very similarly to that of an oxide core. That is, the FCM ejected rod worth (\$0.517) is very close to that of the oxide ejected rod worth (\$0.523), and the power curve (versus time) is also very close, as shown in Fig. 4.1.

Given the BNL simulations, the power deposition (Fig. 4.1) within the FCM fuel is very similar to that of the oxide fuel with respect to magnitude and time. The fuel rod simulations for this transient, however, have not been completed; but with the higher thermal conductivity and higher volumetric heat capacity of the FCM fuel, it is expected that the transient thermal response of the FCM fuel will be slower with lower peak temperatures than that of the oxide fuel.

4.3. LOCA DB accident

4.3.1. Introduction

The LOCA DB Accident in current regulations (10CFR50.46 Appendix K) focuses on a break in the primary coolant piping of a reactor at full power with subsequent scram with UO_2 fuel and zirconium alloy cladding. For this DB accident, the following criteria must be met:

- Must avoid cladding temperatures $>1204^\circ\text{C}$ 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.

These requirements essentially determine the design, capacity, and response of the emergency core cooling system (ECCS) for a reactor core utilizing UO_2 fuel and zirconium alloy cladding. These limits/restrictions could/would change for a different fuel/cladding system (e.g., ATFs); in all probability, experiments would be needed to determine the new limits that would be applicable for an ATF core.

4.3.2. LOCA DB analyses

All reactor conditions being equal (scram time, break, etc.), the heat-up of the fuel and cladding during a LOCA is mainly determined by their volumetric heat capacity and thermal conductivity. The heat-up of the fuel cladding will be accelerated when the exothermic metal–water reaction is initiated. Therefore, the peak cladding temperature (PCT) during any LOCA scenario should be limited to 1204°C (2200°F) to prevent an autocatalytic excursion (e.g., for zirconium alloy clad fuel).

As discussed in Section 2, the FCM fuel and the FeCrAl (or 316SS) cladding have a great potential to mitigate the heat-up during a LOCA since they have high volumetric heat capacities and much lower rates of oxidation. In this study, we will focus on the performance of the FCM fuel and the FeCrAl cladding during a

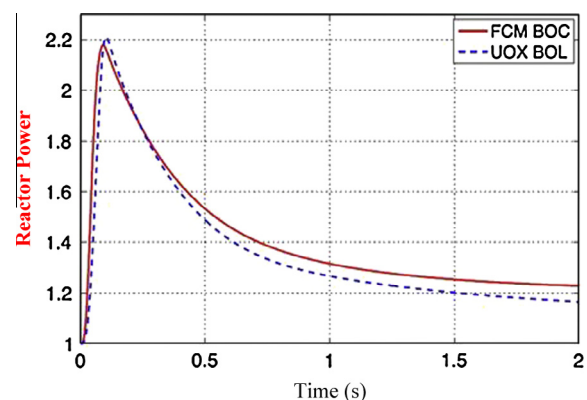


Fig. 4.1. AP-1000 RIA power curve using an oxide or FCM core.

large break (LB) LOCA (in comparison with the response of a UO_2 –Zircaloy core).

A GE BWR4 plant with a Mark I containment operating at 1155 MWe power (see Fig. 4.2) will be employed as the bases for these analyses.

The TRACE code [17] is the NRC's primary thermal hydraulics tool. TRACE contains 1-D and 3-D resolution of the nuclear power plant's primary and secondary systems. The code includes detailed modeling of the system thermal-hydraulics including two-phase flow.

A TRACE model of the GE BWR4 is employed to perform this study. The TRACE model includes all of the major flow paths and system components to perform large and small break LOCA simulations, as shown in Fig. 4.3. Both recirculation loops are modeled explicitly. The reactor pressure vessel (RPV) is modeled using the TRACE VESSEL component with axial levels, radial rings, and azimuthal sectors representing the downcomer, lower plenum, core, upper plenum, and upper head regions. The GE14 fuel bundles are modeled using TRACE CHAN components. Also modeled are the drywell and suppression chamber components with a CONTAN

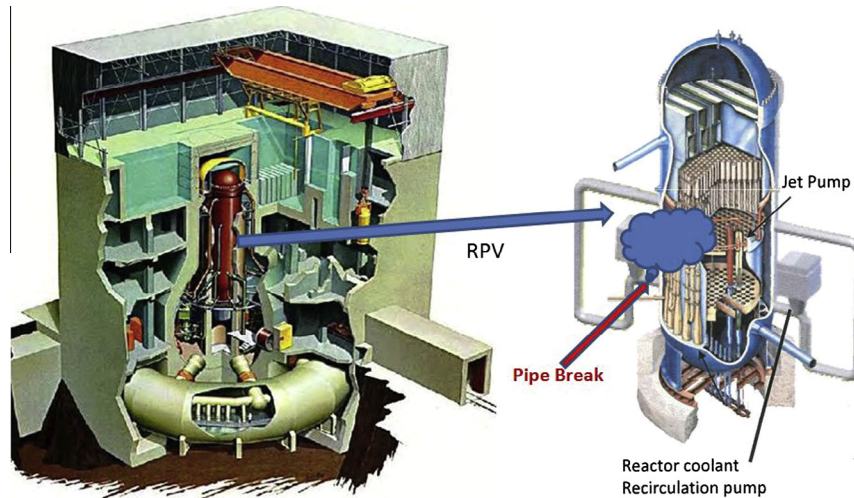


Fig. 4.2. BWR with a DB LOCA pipe break occurring in the reactor coolant recirculation pump suction line.

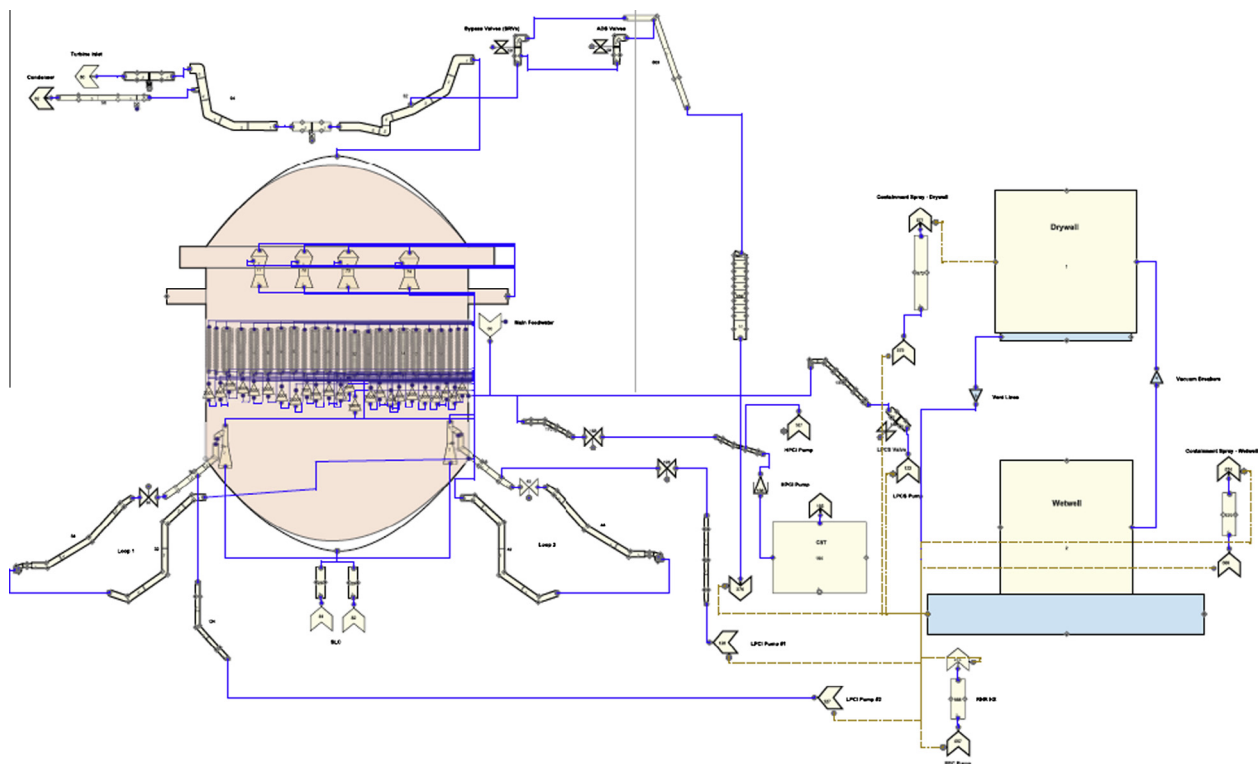


Fig. 4.3. GE BWR4 TRACE model.

component. The steam lines are modeled out to the turbine control valve. The feedwater system is modeled with a FILL component set up to control the feedwater flow rate to maintain the desired downcomer level in the RPV. The ECCSs are also modeled using FILL components and include the High-Pressure Coolant Injection (HPCI), Low-Pressure Coolant Injection (LPCI), and Low-Pressure Core Spray (LPCS) injection systems.

The LBLOCA is a double-ended guillotine recirculation suction line break (0.362 m^2). A loss of offsite power is assumed to occur at the beginning of the transient so that the feedwater stops flowing into the RPV and the two recirculation pumps are tripped at 0.0 s. It is also assumed that the reactor is scrammed at 0.0 s. Only two LPCIs are assumed to be available (HPCI, LPCS, and the automatic depressurization system [ADS] are all unavailable). Three scenarios of ECCS injection were simulated, as summarized in Table 4.1. Scenario 1 is a normal LBLOCA case in which LPCI injection is actuated as designed. Scenario 2 assumes that there is delay in LPCI injection. Case 2 is executed in order to have an extended heat-up of the fuel cladding (approaching the 1204°C PCT for the UO_2 -Zircaloy core). Scenario 3 assumes all ECCS injection fails, so the metal (clad)-water oxidation does occur in this scenario and PCTs do exceed 1204°C , and thus this scenario is a BDBA.

LBLOCA simulations were performed for three combinations of fuel and cladding materials: (1) UO_2 -Zircaloy; (2) UO_2 -FeCrAl; and (3) FCM-FeCrAl. The first one is the reference case in which the fuel and cladding materials are conventional UO_2 and Zircaloy. In the second case only the fuel cladding is changed to FeCrAl. The third case is that the fuel and cladding materials are replaced with FCM and FeCrAl. Note that the replacement of the fuel and cladding materials is only made to the hottest fuel bundle. It means that all other fuel bundles still use the same UO_2 fuel and the same Zircaloy cladding. Hence, the changes have no impact on core thermal-hydraulic conditions during the LBLOCA.

The GE14 fuel used in the TRACE model is composed of 78 full-length fuel rods, 14 part-length rods, and two large central water rods in a 10×10 lattice array. Design parameters of the GE14 fuel are summarized in Table 4.2.

The sequences of events calculated for the three LBLOCA scenarios are presented in Table 4.3.

At time 0.0 s, the break causes a blowdown of the coolant from the RPV into the drywell, resulting in a rapid depressurization of the RPV, and pressurization in the containment. The downcomer collapsed level starts to drop at the very beginning of the transient. As the RPV pressure decreases, the saturation temperature decreases to the coolant temperature, and flashing begins. The core flow stops at 20 s into the transient. When the LPCI starts injection

Table 4.3

LBLOCA sequences of events.

Event	Scenario 1 (s)	Scenario 2 (s)	Scenario 3 ^a (s)
Initiation of break	0.0	0.0	0.0
Reactor scrams	0.0	0.0	0.0
Feedwater flow tripped	0.0	0.0	0.0
MSIVs start to close	0.0	0.0	0.0
Recirculation pumps tripped	0.0	0.0	0.0
Pressurization of the dry well activates the ECCS	0.1	0.1	0.1
LPCI injection starts	91	319	0.0
Peak cladding temperature reached, or 1204°C reached	128	347	347 ^a
Calculation ends	300	500	500

^a Note: In the Scenario 3 case, the 1204°C limit is exceeded in the UO_2 -Zircaloy simulation at ~ 350 s, and clearly after 350 s, this is beyond a DB accident.

at 91 s in Scenario 1, the core flow resumes and starts to significantly oscillate when the water level in the downcomer increases above the inlet of the jet pumps. The PCT declines in the first 20 s, due to the core power decay and an increase in the core flow from coolant flashing.

The PCT starts to increase after the core flow stops and reaches the maximum value at 128 s, as shown in Fig. 4.4. For this scenario, the core quenches at ~ 180 s. The maximum PCT reached in the fuel and cladding of FCM-FeCrAl is about 75 K lower than the conventional fuel. Note that there is only a small difference in the maximum PCT between UO_2 -FeCrAl and FCM-FeCrAl because the high volumetric heat capacity of FeCrAl plays a dominant role in the heat-up of the fuel cladding.

In Scenario 2, Fig. 4.5, it is assumed there is a delay in the LPCI injection (in order to show an extended cladding heat-up). The difference in the maximum PCT increases to 224 K between UO_2 -Zircaloy and FCM-FeCrAl.

Scenario 3 assumes that all ECCSs fail during the LBLOCA, and clearly without coolant injection this scenario will develop into a BDB accident. The PCT and the Zircaloy cladding oxidation are shown in Fig. 4.6. This scenario further demonstrates that the Fe-CrAl cladding has superior performance in terms of heat-up as compared with the Zircaloy cladding. It should be noted that Fe-CrAl also has better performance in the metal-water oxidation kinetics than Zircaloy. When Zircaloy reaches a sufficiently high temperature ($\sim 900^\circ\text{C}$) in a steam environment, the exothermic metal-water reaction is initiated, as shown in Fig. 4.6. However, FeCrAl oxidation is almost negligible until the cladding temperature goes beyond 1650°C . (For this case, the simulation was terminated at 500 s, at which point the cladding temperature is approaching 1204°C .)

Table 4.1

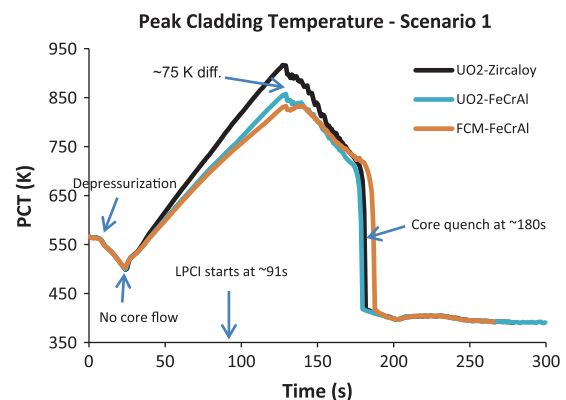
ECCS injection scenarios.

Scenario	ECCS injection	Comments
1	LPCI injection starts at 91 s	LPCI is actuated as designed
2	LPCI injection starts at 319 s	LPCI is actuated with a delay
3	All ECCS systems fail	Metal-water reaction starts

Table 4.2

GE14 fuel and cladding dimensional design parameters.

Parameters	Values
Cladding thickness (mm)	0.6604
Gap thickness (mm)	0.0889
Fuel pellet diameter	8.7633
Fuel rod pitch (mm)	1.2954
Active fuel length (m)	3.81
Active fuel length (Part-length rods) (m)	2.4384

**Fig. 4.4.** PCT – Scenario 1 LPCI injection at 91 s.

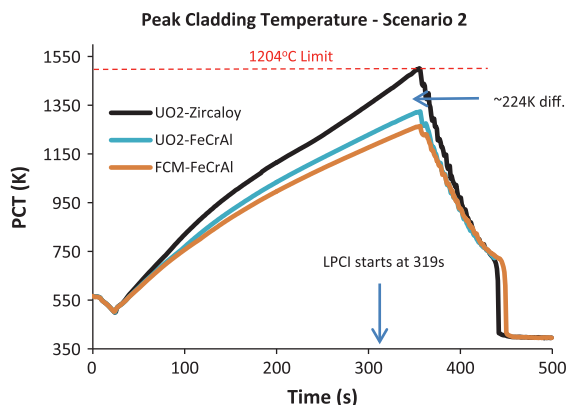


Fig. 4.5. PCT – Scenario 2 LPCI injection at 319 s.

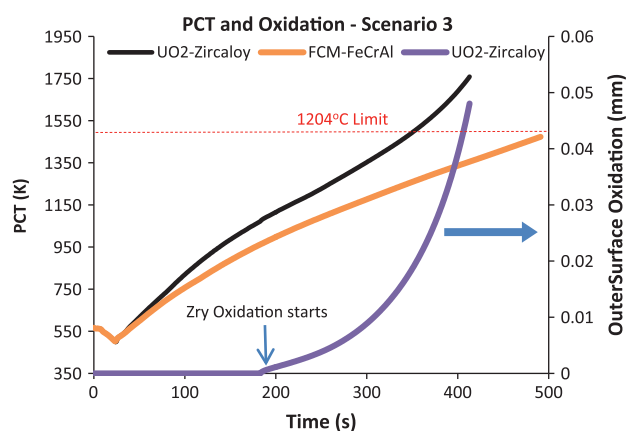


Fig. 4.6. PCT and oxidation – Scenario 3 all ECCSs fail.

5. BDB accident analyses

5.1. Overview

Scoping simulations were performed using the MELCOR severe accident code [18] to investigate the influence of advanced materials on the BDB accident progression and to identify any existing code limitations. Short-term and long-term station blackout (STSBO, LTSBO) accidents were simulated for Peach Bottom (PB), a GE BWR/4 with a Mark-I containment.

The MELCOR code is the primary code the NRC uses to model and analyze the progression of severe accidents. It has been developed and maintained for the NRC by the Sandia National Laboratories (SNL). MELCOR is a systems-level severe accident code which includes the major phenomena of the system thermal-hydraulics, fuel heat-up, cladding oxidation, radionuclide release and transport, fuel melting and relocation, etc.

The analyses used the Peach Bottom (a GE BWR4/Mark I) MELCOR 1.8.5 model, which is described by Francis [19]. The model has a long development history with its origination in the late 1980's at BNL followed by additional development at ORNL and Dycoda, LLC. The core is modeled with five rings and eight axial segments. Excluding the core, the RPV is modeled with six control volumes. The reactor power before SCRAM was assumed to be 3,293 MWth. The model contains the key equipment important in modeling severe accidents (reactor core isolation cooling [RCIC] and high-pressure coolant injection [HPCI] systems, safety relief valves [SRVs], water storage tanks, etc.). Although the model contains a detailed representation of the reactor building, the current study focused on the impact of ATFs on the in-vessel accident progression.

5.2. Simulation setup

The fuel and cladding properties were modified in MELCOR to reflect the candidate ATF materials FCM and FeCrAl, respectively. Material melting, oxidation, oxide properties, and relocation characteristics of the materials must be specified in MELCOR. However, some characteristics and material properties for these advanced materials have not yet been determined. The following describes how the material properties in MELCOR were modified to represent the advanced materials and the modeling limitations encountered.

MELCOR 1.8.5 assumes “UO₂” for fuel and “Zircaloy” for the cladding and channel boxes. Therefore, the material properties specified for “UO₂” and “Zircaloy” were modified to simulate different fuel and cladding materials. However, in doing so, all components such as the channel boxes that are made of the “Zircaloy” material also had their material properties changed. The inability to decouple the material property specifications for the cladding and channel box was a key modeling limitation encountered in this study.

The new fuel and cladding were assumed to have the same geometry, dimensions, and flow characteristics as the nominal UO₂–Zr based core. To accomplish this, the material masses were scaled by their densities to retain the same material volumes. The core flow paths and flow resistances were also not modified. The radionuclide composition, distribution, and release physics were not modified from the base model. Other model parameters not discussed, such as heat transfer correlations, were not modified. Modification of these parameters may be performed in future studies.

The properties implemented are summarized in Table 5.1 and Fig. 5.1. Due to the lack of material property information for FCM, the fuel material properties were modeled as those of SiC. To simulate the FeCrAl material, many of the material properties for 304 stainless steel (304SS) were substituted for Zr. 304SS and FeCrAl have similar material properties. The properties implemented for the cladding (and channel box) are given in Table 5.1 and Fig. 5.1. Some sensitivity coefficients related to cladding failure and relocation were also modified to those for 304SS. The fuel rod collapse temperature (SC1132) was set to 2800 K. The dissolution rate for the cladding material (SC1010) was set to those of steel and steel oxide (see user manual [18]). The holdup of molten material by an oxide shell was set to occur at a minimum oxide thickness of 0.001 m and for oxide temperatures below 1700 K (SC1131, defaults for steel [18]).

The “Zircaloy” material oxidation rate in MELCOR is governed by parabolic kinetics with consideration of gaseous diffusion limits and accounts for oxidation by both H₂O and O₂. Oxidation of the fuel material is not modeled in MELCOR.

The oxidation characteristics of FeCrAl were not available when these simulations were performed (only that the oxidation mass depletion and kinetics were roughly two orders lower than that of Zircaloy [3], see Figs. 2.2 and 2.3). Therefore, for this study, the oxidation rate equations for the FeCrAl material were modified to reflect 1% of the values of Zircaloy. In addition, a simulation where oxidation was effectively turned off (0%) and a simulation using the default UO₂ and Zircaloy properties were conducted to bound the range of possible cladding oxidation behavior. Table 5.2 provides

Table 5.1
Material melting properties modified in MELCOR.

Property	“FCM”	“FeCrAl”	“FeCrAl” oxide
Melting temperature (K)	2818.0	1672.0	1870.0
Latent heat of fusion (kJ/kg)	1000	268	598

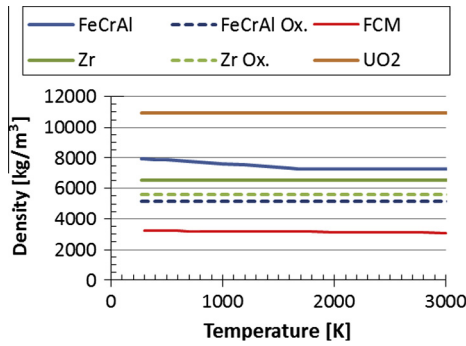


Fig. 5.1a. Density.

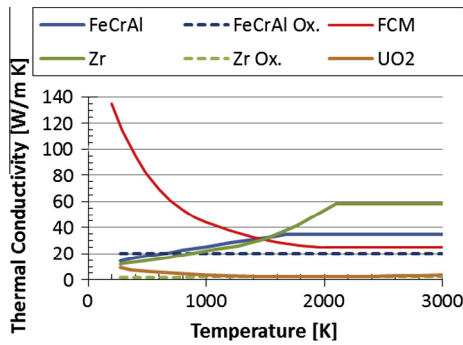


Fig. 5.1b. Thermal conductivity.

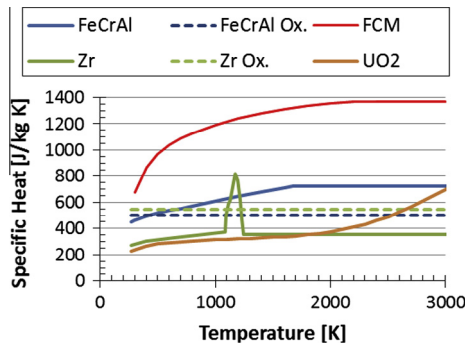


Fig. 5.1c. Specific heat.

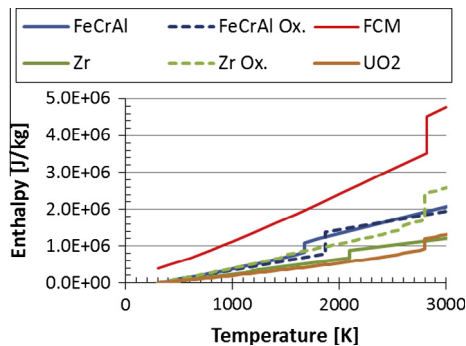


Fig. 5.1d. Enthalpy.

(SC1003) is specified for H₂O and is multiplied by two internally in the code for O₂.

A major code limitation was encountered in that the heat of reaction during oxidation is hardcoded in MELCOR (Clad-H₂O: 5.797×10^6 J/kg clad at 298.15 K, Clad-O₂: 1.2065×10^7 J/kg clad at 298.15 K). The heat of reaction of Zircaloy is approximately 10 times that of stainless steels. As shown later, the heat released due to cladding oxidation can be a major contributor to the overall heat load.

$$\frac{d(W^2)}{dt} = K(T) = A_1 \exp\left(\frac{B_1}{T}\right) \quad \text{if } T < T_{\text{tran},1}, \quad (5.1)$$

$$\frac{d(W^2)}{dt} = K(T) = A_2 \exp\left(\frac{B_2}{T}\right) \quad \text{if } T \geq T_{\text{tran},2}, \quad (5.2)$$

$$\frac{dW}{dt} = D \frac{k_c P_{\text{ox}}}{T_f}, \quad (5.3)$$

where A_1 is the low-temperature-range constant coefficient, A_2 is high-temperature-range constant coefficient, B_1 is low-temperature-range exponential constant, B_2 is high-temperature-range exponential constant, D is user-defined sensitivity parameter, $K(T)$ is oxidation rate constant, k_c is mass transfer coefficient, P_{ox} is partial pressure of oxidant, t is time, T is temperature (absolute); T_f is gas film temperature, $T_{\text{tran},1}$ is upper-temperature boundary for low-temperature range, $T_{\text{tran},2}$ is lower-temperature boundary for high-temperature range, and W is the mass of metal oxidized per unit surface area.

The scenarios investigated include a STSBO and a LTSBO. The STSBO assumes the reactor shuts down (SCRAM) and all power is lost at time zero and all injection is lost at time zero. The LTSBO assumes the reactor SCRAMs and AC power is lost at time zero and DC power (batteries) is lost at 8 h. The RCIC and HPCI are available for operation during the LTSBO up until the loss of battery power. For these scenarios, the ability to inject water via the control rod drive (CRD) pumps and low-pressure ECCS pumps (no AC power) and diesel-driven pumps is also assumed to be lost.

An additional scenario, a mitigated STSBO (MSTSBO), is analyzed in which operators take action to inject water through the CRD pumps at a rate of 110 GPM (25 m³/h). This provides a steady flow of high-pressure water, injected into the bottom of the RPV.

5.3. Simulation results

5.3.1. STSBO

With the standard UO₂-Zr system, as expected, rapid cladding oxidation occurs after the core is mostly uncovered (see reactor water level in Fig. 5.2 and core PCT in Fig. 5.3). A large amount of hydrogen is generated (Fig. 5.4), releasing a substantial amount of heat between 1 and 2.5 h after reactor SCRAM (Fig. 5.6). The relocating core melt dries out the lower head approximately 4 h after SCRAM (Fig. 5.2).

The progression of the UO₂-FeCrAl case and the UO₂-Zr case with oxidation prohibited closely align (Fig. 5.2). After 10 h, the UO₂-FeCrAl case generates approximately half of the hydrogen as the UO₂-Zr case (Fig. 5.4). The relocating core melt dries out the lower head approximately 8.5 h after SCRAM. (Note: there are additional uncertainties involved with these new core materials since core degradation mechanisms [materials interactions, eutectic formation, failure temperatures, etc.] have not been experimentally determined.)

The FCM-FeCrAl case also closely aligns with the UO₂-FeCrAl case. However, the core stays mostly intact and relocation to the lower plenum is reduced. This is due to the high “melting” temperature specified for the FCM material. As noted previously,

the values used in the oxidation rate equations and gaseous diffusion equation, Eqs. (5.1)–(5.3). The default values in MELCOR for stainless steel are provided for comparison. The parameter D

Table 5.2

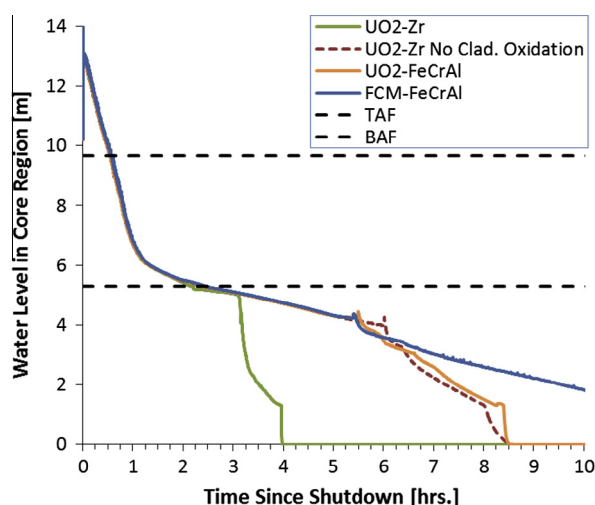
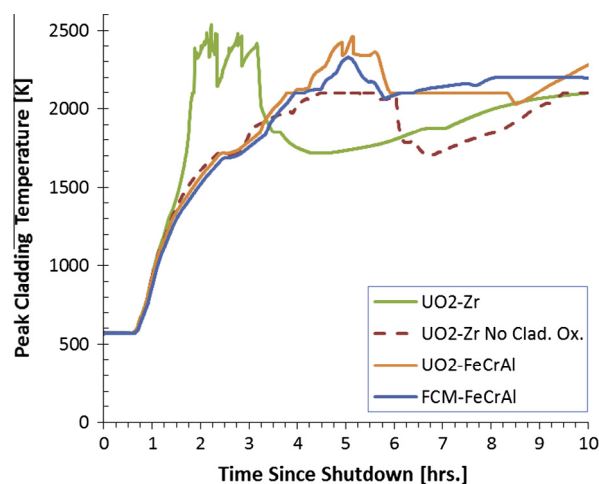
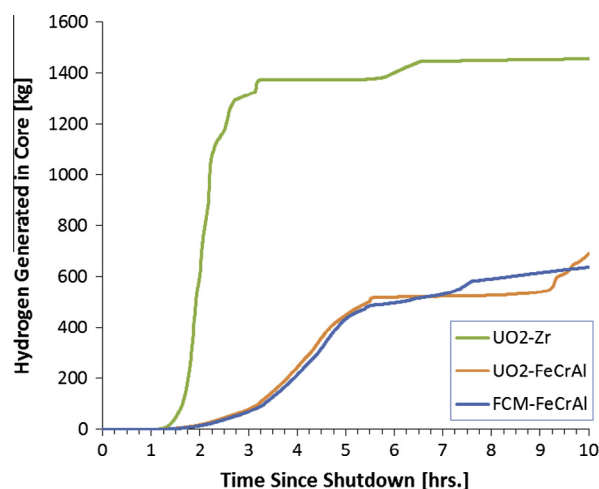
Oxidation rate equation values implemented in MELCOR.

Reacting species	Parameter	Zr	“FeCrAl” ($0.01 \times \text{Zr}$)	SS	Units
O_2	A_1	50.4	0.504	0.0	$\text{kg}^2/\text{m}^4 \text{ s}$
	B_1	14630.0	14630.0	0.0	K
	$T_{\text{tran},1}$	10000.0	10000.0	1333.0	K
	A_2	0.0	0.0	0.0	$\text{kg}^2/\text{m}^4 \text{ s}$
	B_2	0.0	0.0	0.0	K
	$T_{\text{tran},2}$	10000.0	10000.0	1550.0	K
H_2O	A_1	29.6	0.296	2.42E9	$\text{kg}^2/\text{m}^4 \text{ s}$
	B_1	16820.0	16820.0	42400.0	K
	$T_{\text{tran},1}$	1853.0	1853.0	1333.0	K
	A_2	87.9	0.879	2.42E9	$\text{kg}^2/\text{m}^4 \text{ s}$
	B_2	16,610	16610.0	42400.0	K
	$T_{\text{tran},2}$	1873.0	1873.0	1550.0	K
H_2O	D	0.00548	0.00504	0.00504	$\text{kg K}/\text{Pa m}^3$

the relocation characteristics and modeling of the FCM material should be further investigated.

The steam dryer temperature and main steam line temperature, upstream of the main steam isolation valve (MSIV), are given in Figs. 5.5 and 5.6, respectively. While the water level drops, the fuel superheats the gas flow passing through the core and into the steam separators, steam dryers, main steam line, and finally out the SRVs. In the UO_2 -Zr system, the rapid release of energy from cladding oxidation further accelerates the heat-up of the upper structures. However, the UO_2 fuel and Zr cladding begin to relocate into the bottom head, sending additional steam upwards. The additional steam flow limits the temperature rise in the upper structures. Soon afterwards, the bottom head fails and the melt is relocated. However, for the other cases where there is low/no cladding oxidation and the core remains intact longer, the upper structures continue to heat up. Note that another limitation in MELCOR and the current Peach Bottom model is the inability to melt and relocate upper structures and core shroud. Thus, even though temperatures are predicted well above their structural failure limits and are in the range of the material melting point (1675–1725 K), the structures are modeled as remaining intact.

Table 5.3 summarizes the differences in the predicted accident progression timing between the nominal UO_2 -Zr case and the UO_2 -FeCrAl case. The advanced cladding slows the accident progression, providing a few extra hours for accident response measures.

**Fig. 5.2.** Core water level, STBSO.**Fig. 5.3.** Peak intact cladding temp., STBSO.**Fig. 5.4.** In-core hydrogen gen., STBSO.

5.3.2. LTSBO

The LTSBO scenario results are generally consistent with the STBSO scenario results, except the time scales are extended. The first 12 h of the accident (SCRAM, RCIC operation, boildown) are similar between the cases.

Around 12 h, cladding oxidation in the UO_2 -Zr case accelerates with the decrease in the reactor water level (Fig. 5.8), generating hydrogen (Fig. 5.10) and causing the excursion in cladding

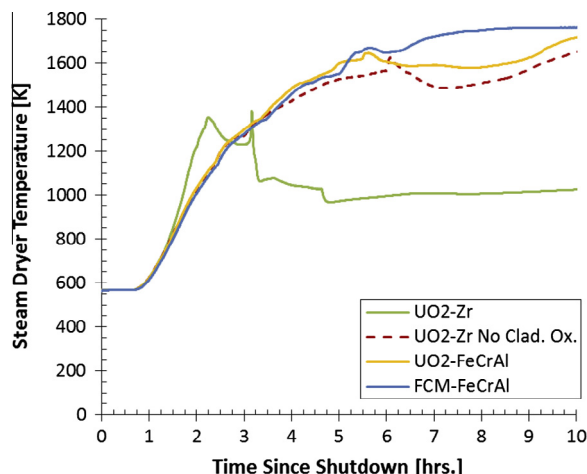


Fig. 5.5. Steam dryer temperature, STSBO.

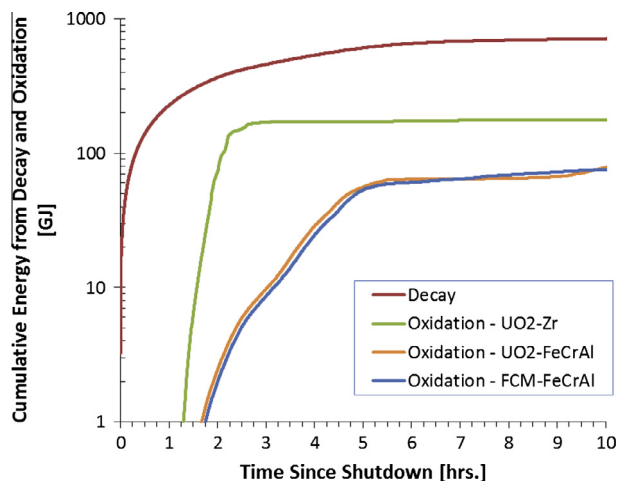


Fig. 5.6. Energy source terms, STSBO.

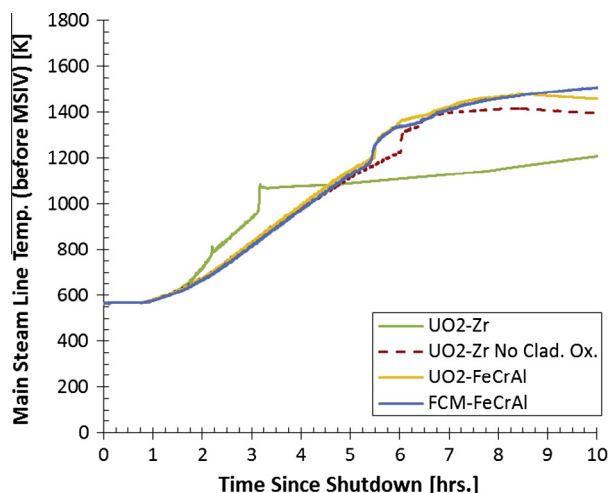


Fig. 5.7. Main steam line temp., STSBO.

Table 5.3
STSBO event timing comparison.

Event	UO ₂ -Zr case time (h)	UO ₂ -FeCrAl case time (h)	Difference Δ time (h)
First cladding gap release	1.23	1.26	0.03
Time to 50 kg H ₂ generated	1.52	2.59	1.07
First channel box failure	1.87	4.22	2.35
First cladding relocation	2.07	4.93	2.86
Lower head dries out	3.98	8.52	4.54
Lower head failure	4.64	12.14	7.50

case where Zr oxidation was prohibited). By reducing the cladding oxidation, the time to first cladding relocation is delayed 6.1 h until approximately 20.3 h after shutdown (for the UO₂-FeCrAl case).

The UO₂-FeCrAl and FCM-FeCrAl cases became unstable during core relocation, and the simulations were terminated in MELCOR. However, the UO₂-Zr case where cladding oxidation is prohibited was extended through the 24 h simulation duration. The UO₂-FeCrAl case likely closely follows the UO₂-Zr case without cladding oxidation.

The steam dryer temperature response (Fig. 5.11) is similar to that predicted in the STSBO scenario. The UO₂-Zr case results in a rapid heat-up of the steam dryer during cladding oxidation; however, the temperature is reduced as the melt relocates into the lower head and then ex-vessel. In contrast, the cases using advanced materials have a more gradual and long-term heat-up of the steam dryers, reaching very high temperatures. The integrity of the upper internals at such high temperatures over such long durations is questionable.

Table 5.4 summarizes the differences in the predicted accident progression timing between the nominal UO₂-Zr case and the UO₂-FeCrAl case. The advanced cladding slows the accident progression, providing several extra hours for accident response measures.

5.3.3. Mitigated STSBO (MSTBO) with water injection via CRD results

In this scenario the STSBO is successfully mitigated by operators recovering water injection capability through the control rod drive (CRD) pumps. In these simulations the CRD pumps inject water from the condensate storage tanks at the rate of 110 gpm (0.42 m³ per min), which is the nominal PB CRD pump capacity after SCRAM. The core slowly refills and remains partially exposed for many hours (Fig. 5.12). Only the UO₂-Zr and UO₂-FeCrAl cases were analyzed.

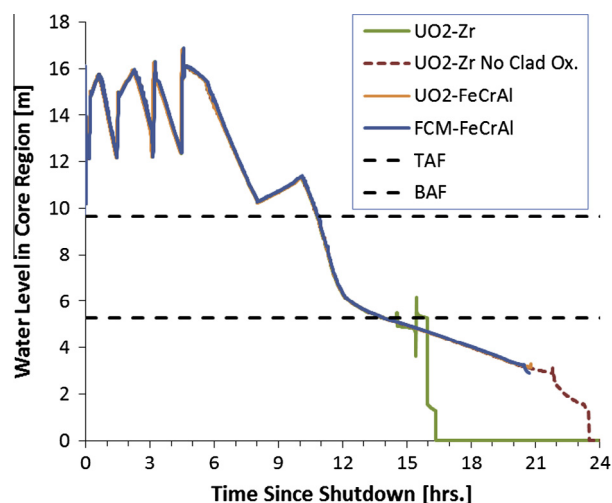


Fig. 5.8. Core water level, LTSBO.

temperature, as shown in Fig. 5.9. The first cladding failure and relocation occur at 14.1 h. In contrast, the cladding temperature for the cases with advanced materials increases at a slower rate due to the reduced cladding oxidation (or lack of for the UO₂-Zr

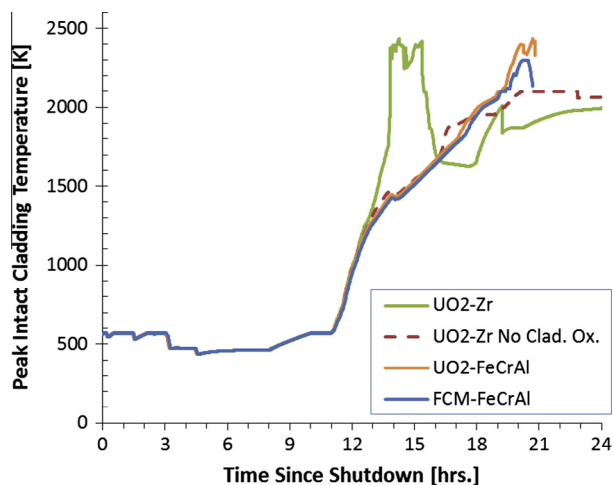


Fig. 5.9. Peak intact cladding temp., LTSBO.

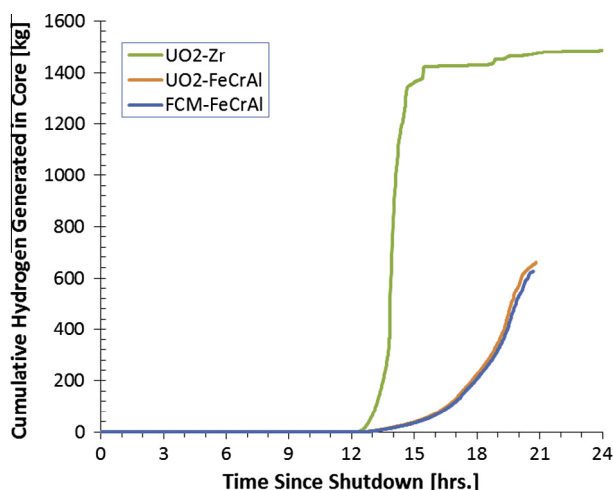


Fig. 5.10. In-core hydrogen gen., LTSBO.

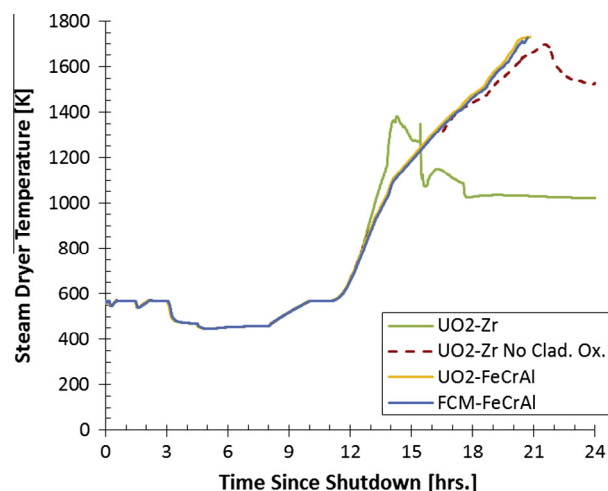


Fig. 5.11. Steam dryer temp., LTSBO.

The UO₂-Zr case reaches a peak cladding temperature of 1720 K (Fig. 5.13), while the UO₂-FeCrAl case reaches a peak temperature

Table 5.4

LTSBO event timing comparison.

Event	UO ₂ -Zr case time (h)	UO ₂ -FeCrAl case time (h)	Difference Δ time (h)
First cladding gap release	11.83	12.54	0.71
Time to 50 kg H ₂ generated	12.89	15.42	2.53
First channel box failure	13.83	19.38	5.55
First cladding relocation	14.12	20.25	6.13
Lower head dries out	16.38	23.52 ^a	7.14 ^a
Lower head failure	17.55	24.48 ^a	6.93 ^a

^a Estimated based on the results of the UO₂-Zr case where Zr oxidation is prohibited.

of 1455 K. Coincidentally, the UO₂-FeCrAl case remains below the regulatory peak cladding temperature limit of 2200°F (1477.7 K).

The UO₂-Zr case produces 583 kg of hydrogen (Fig. 5.14) over the time interval of 1.75–8 h after SCRAM, while the UO₂-FeCrAl case only produces 31 kg.

In the UO₂-Zr case, the steam dryer reaches temperatures (Fig. 5.15) similar to those predicted in the STSBO and LTSBO scenarios. However, for the UO₂-FeCrAl case, the predicted steam dryer temperature is much lower than that predicted to occur in the STSBO and LTSBO scenarios.

5.3.4. Simulation limitations

A few modeling limitations were encountered in MELCOR while performing this work. First, for a BWR simulation, the channel box and fuel cladding are both hardcoded to use the same material in MELCOR. Thus, the material specification and properties cannot be altered independently for these components. Second, the heat of reaction (that of Zircaloy) during oxidation is hardcoded in MELCOR. Another code limitation is that oxidation of the fuel is not modeled by MELCOR. While this is of limited interest for UO₂, future fuel materials may be more susceptible to oxidation. These limitations restrict a user's ability to evaluate advanced fuel and cladding materials that are outside the scope of the traditional UO₂, Zr, and steel system.

As new fuels and claddings are developed to be more robust, the other structures within the RPV may face additional challenges and need to be considered for evaluation. Currently, a limitation in MELCOR is the inability for the upper internal structures and the core shroud in BWRs to melt and relocate. The simulation results presented here suggest the upper internals may be challenged before the fuel for cases with advanced fuels/cladding.

This study focused on the in-vessel impact of new fuels and cladding on the accident progression with respect to accident

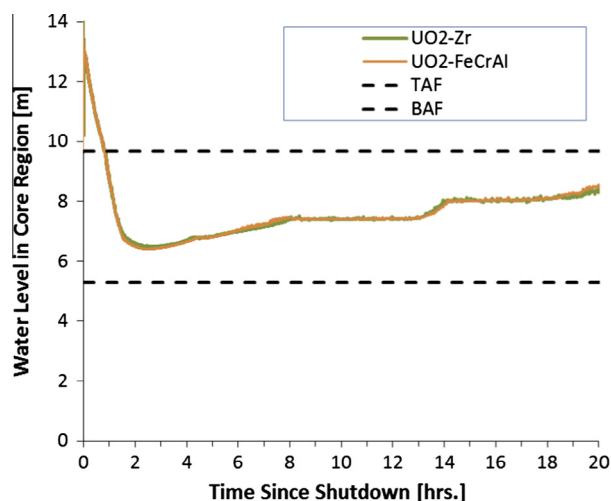


Fig. 5.12. Core water level, MSTBSO.

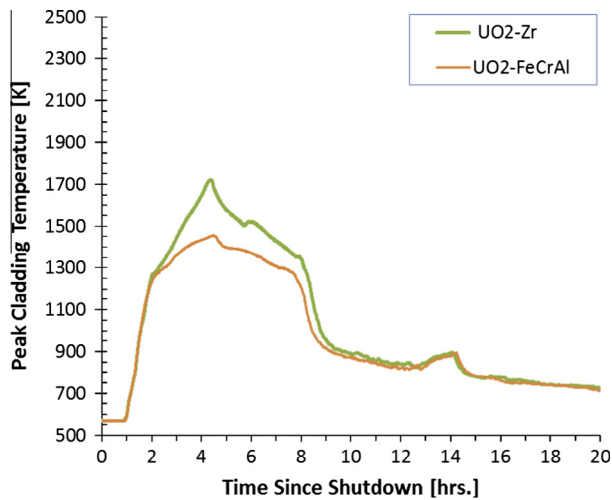


Fig. 5.13. Peak intact cladding temp., MSTSB0.

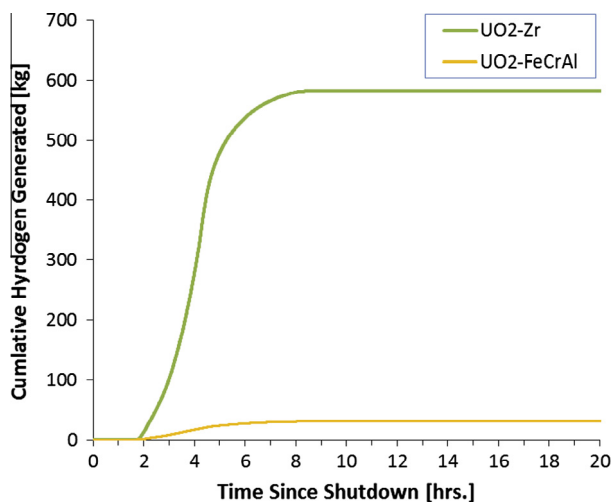


Fig. 5.14. In-core hydrogen gen., MSTSB0.

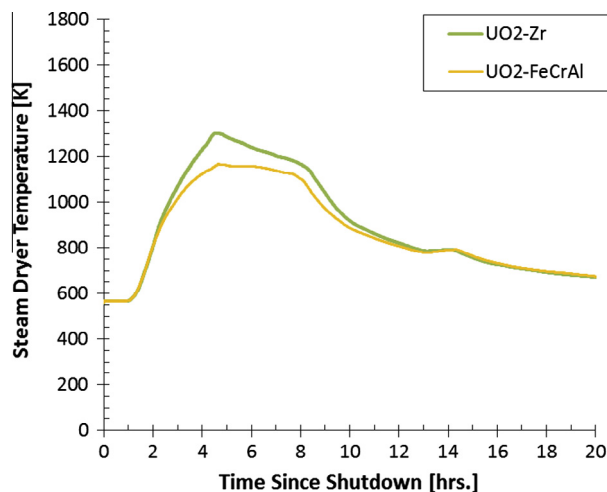


Fig. 5.15. Steam dryer temp., MSTSB0.

vanced fuels and claddings on the ex-vessel phase of an accident. Note that the possible additional material, from melting and relocation of the upper structures, may also impact the ex-vessel progression. Finally, the fission product release characteristics for the advanced fuels will need to be developed and integrated into systems tools to analyze their risk reduction potential. This could be a key benefit of the FCM fuel.

6. Conclusions

Under normal reactor operations, as expected, a fuel (such as FCM) with higher thermal conductivity and better structural stability operates at much lower temperatures (and temperature gradients) than standard oxide fuel (with its low thermal conductivity, high swelling, etc.). However, the assumption that the FCM fuel can operate at the same power levels (LHGRs) and cycle lengths is obviously questionable since ^{235}U mass has been reduced (higher ^{235}U enrichments probably would be required). Core design issues were not addressed in these analyses.

The ATF LOCA studies indicate that peak cladding temperatures could be reduced (75 °C) simply by replacing the Zircaloy cladding with a cladding (such as FeCrAl) with a higher volumetric heat capacity, even using UO_2 as the fuel. An additional, although small, advantage (lower PCT) could also be gained by employing a fuel such as FCM. For the ATF LOCA studies in which the ECCS injection is delayed or eliminated and thus clad oxidation occurs, peak cladding temperatures could be significantly reduced by as much as 225 °C or the time to the peak temperature delayed by ~150 s by replacing the Zircaloy cladding with an oxidation-resistant cladding (such as FeCrAl). With respect to BDB accidents, higher-melting/lower-hydrogen-producing core components WILL NOT preclude a severe accident (SA). There are no “silver bullets”; if core cooling cannot be restored, the SA will continue. The time for boil off and when the core becomes uncovered is primarily dependent upon the fuel decay heat and not the sensible heat of the fuel at the start of the accident. Assuming the decay heat from ATFs is comparable to the traditional $\text{UO}_2\text{-Zr}$ system, changing the cladding and fuel has a minor impact on the early stages of a BDBA. It is not until near the start of cladding oxidation where the benefits of ATFs are realized. The use of ATFs results in an increased margin of time for accident response and mitigation measures. This additional time, to restore core cooling and mitigate the accident, is on the order of an hour to a few hours. Hydrogen generation, which can exacerbate the accident, is also slowed by the use of ATFs. This study highlighted some limitations with the MELCOR code with respects to modeling ATFs; these limitations need to be addressed for subsequent ATF MELCOR analyses.

As for our current commercial nuclear plants, a significant number of experiments (materials interaction and degraded core) conducted over the past 30 years yielded the current state of knowledge of the progression of BDB accidents in LWRs with $\text{UO}_2/\text{Zircaloy}$ cores. In general, the initiation of core degradation in current cores starts with the failure (via eutectic interactions) of the control elements, which then propagates into the fuel assemblies, accelerating the fuel/cladding failure. It is imperative with the study or introduction of ATFs into nuclear cores that the influence of all the core components on the core degradation process be considered (e.g., the stainless steel control blade with B_4C absorber in a BWR and other RPV stainless steel components).

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progression timing, hydrogen production, and component temperatures. Future analysis should also encompass the impact of ad-

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