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PHENOMENA IDENTIFICATION RANKING TABLES FOR ACCIDENT TOLERANT FUEL DESIGNS APPLICABLE TO SEVERE ACCIDENT CONDITIONS

Final Report

Work Performed under the Auspices of the
United States Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, D.C. 20555

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) is preparing to accept anticipated licensing applications for the commercial use of accident tolerant fuel (ATF) in commercial nuclear power plants in the United States. It is the objective of the NRC to evaluate the effects of ATF designs on severe accident behavior, and to determine potential changes to the NRC severe accident analysis computer codes that would simulate plant conditions using ATFs commensurate with the accuracy in accident analyses involving conventional fuels. This report documents the development of Phenomena Identification and Ranking Tables (PIRTs) for near-term ATFs under severe accident conditions in light water reactors (LWRs). The PIRTs were developed by a panel of experts for various near-term ATF design concepts (i.e., FeCrAl cladding, zirconium alloy cladding coated with chromium, and Cr_2O_3 dopants in uranium dioxide fuels) in addition to the impacts from fuel enrichment and burnup. Panel members also considered the severe accident implications of the longer-term ATF concepts. The main figures-of-merit considered in this ranking process are the amount of fission products released into the containment and the quantity of combustible gases generated during an accident. Special focus is given to whether existing severe accident codes and models would be sufficient as applied to LWRs employing these fuels, and whether additional experimental studies or model development would be warranted.

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EXECUTIVE SUMMARY

The United States (U.S.) Nuclear Regulatory Commission (NRC) is preparing to accept anticipated licensing applications for the commercial use of accident tolerant fuel (ATF) in U.S. commercial nuclear power plants. Several fuel vendors, in coordination with the U.S. Department of Energy (DOE), have announced plans to develop and seek approval for various fuel designs with enhanced accident tolerance (e.g., fuels with longer coping times during loss of cooling conditions). Vendors have also expressed an interest in increasing fuel burnup above the licensed limit (which varies by vendor, but roughly corresponds to 62 GWd/MTU peak rod average), as well as increasing enrichment beyond 5 percent.

The NRC has broadly categorized the ATF design concepts as near-term and longer-term. The near-term ATF concepts are those for which the existing data, models, and methods might be applicable with some modification and limited additional data. Longer-term ATF concepts are those for which substantial new data, models, and methods need to be developed.

The near-term ATF designs with respect to the time of commercial deployment include:

- Advanced stainless steel (FeCrAl) cladding,
- Conventional zirconium-alloy cladding coated with chromium (Cr), and
- Cr_2O_3[1] dopant in uranium dioxide fuels.

On the other hand, the longer-term ATF concepts being considered include:

- Silicon carbide cladding,
- High-density silicide fuels,
- High-density nitride fuels, and
- Metallic fuels (specifically uranium-Zr with the zirconium content near 50 wt%).

These ATF designs represent evolutions and deviations from the zirconium-alloy clad, uranium dioxide fuel forms.

In addition, the behavior of high-burnup (HBU) and high-assay low-enriched fuel (HALEU) with less than 20 wt% U-235 is also of interest to the NRC. In the present context, HALEU is viewed as increased fuel enrichment for use in LWRs, not (as envisioned) for non-LWRs. The LWR industry is anticipating fuel enrichments as high as 10 percent, although realistically it is expected that plants may request more modest enrichment increases in order to achieve the desired burnup and fuel cycle targets.

References [1] and [2] include a review of the various ATF design concepts, including the available literature related to the impacts from enrichment and fuel burnup during design-basis and severe accidents. This literature review aimed to identify fuel/cladding behaviors, degradation, and radiological release and transport phenomena that can potentially be impacted by the ATF design, fuel enrichment, and burnup under severe accident conditions. References [1] and [2] note that the available literature is much more complete with respect to ATF design characteristics than to the behavior of ATFs, fuel enrichment, and burnup under severe accident

[1] The present evaluation of ATF concepts does not include the Cr_2O_3 and Al_2O_3 doped or additive ATF pellet technologies that are also being used in some operating plants (mostly in Europe).

conditions. References [1] and [2] serve as the basis for the development of the Phenomena Identification and Ranking Tables (PIRTs) for near-term ATFs documented here. Note that where available and for completeness, References [1] and [2] include some information on other advanced fuel concepts that are under development (e.g., Tristructural-isotropic [TRISO]). However, these fuels are not within the scope of the PIRT process in the present report.

An objective of the NRC is to evaluate the effects of ATF designs on severe accident behavior, and to determine potential changes to its severe accident analysis code (i.e., MELCOR) to better model those effects. Such changes should result in simulation capabilities for ATFs commensurate in accuracy with accident analyses involving conventional fuels. The objective of this report is to document the development of the PIRTs for near-term ATF concepts under severe accident conditions in LWRs. The PIRTs were developed for various near-term ATF design concepts, including the effects of fuel enrichment and burnup. A panel of experts representing the U.S. and the international research and nuclear industry community participated in the development of the PIRTs for near-term ATFs. The panel members were also requested to provide their input relating to longer-term ATF issues, for future consideration by the NRC (see Appendix A). Particular points of interest in the case of each fuel design are summarized below.

FeCrAl-Clad UO_2 Fuel

Of the near-term ATFs considered in this study, FeCrAl-clad fuel exhibits the most differences under severe accident conditions compared to conventional fuels, because the cladding material is entirely replaced with an advanced stainless steel. The most salient differences noted by the panel are:

- There is limited knowledge of how FeCrAl will degrade at high temperatures, especially when it is oxidized. This affects the entire severe accident progression and, in turn, a number of downstream issues. Aspects include what eutectic reactions among materials occur; whether the cladding metal (or its oxides) wet the fuel versus draining down and leaving bare pellets; and whether foaming of the cladding may occur.
- Oxidation kinetics and behavior are different for FeCrAl. The expectation is a lower initial oxidation rate (prior to the melting of the cladding), but there is the potential for significant oxidation after the cladding melts. This process impacts the accident progression due to a lower hydrogen-to-steam ratio and initial oxidation heat input. However, it also leaves potentially more oxidation to occur in the late in-vessel phase, as well as at the start of the molten core concrete interaction (MCCI) in the ex-vessel phase of accidents.
- There is a significant amount of aluminum in a core with FeCrAl cladding. However, there is not a significant increase in the mass of chromium in the core compared with conventional fuel and cores (i.e., for conventional fuel the in-core stainless steel is 18 wt% chromium). There is a possibility that the chromium and aluminum could affect speciation and transport, but these aspects have not been established.
- Under conditions of high steam partial pressures, it is possible that vapor-phase hydroxides and oxyhydroxides of chromium and aluminum could form. These possibilities have not been established. Vapor-phase hydroxides and oxyhydroxides of chromium and aluminum may be important in the consideration of fission product behavior when the degradation takes place with high steam partial pressures. These

vapor species will not be abundant in a depressurized accident scenario. However, the potential for forming hydroxides and oxyhydroxides of chromium and aluminum can result in increased amounts of nonradioactive aerosols released into the containment, which in turn impact aerosol agglomeration, deposition, and potentially the clogging of any filters.

- Tellurium is not expected to be sequestered/delayed in release by FeCrAl cladding as has been considered possible for zirconium-alloy cladding. In fact, uncertainties remain related to how tellurium is sequestered or delayed by zirconium. Furthermore, there is no experimental evidence that FeCrAl cladding would make tellurium release more or less extensive than for zirconium-alloy cladding.

Cr-Coated Zirconium-Alloy-Clad UO_2 Fuel

The chromium-coated zirconium-alloy-clad fuels do not have as many differences with conventional fuels as were observed for FeCrAl. The importance of the differences is substantially diminished because the bulk cladding remains conventional zirconium-alloy, and the chromium is confined to a thin coating on the outside surface. Moreover, chromium is already present in conventional core structural materials, hence the added mass of chromium constitutes a small difference in the degree of effect following fuel degradation, rather than a significant qualitative difference. Among the main points noted by the panel for chromium-coated, zirconium-alloy-clad fuels are:

- Some thermophysical properties of the fuel are slightly less well known than those for conventional fuel designs. This holds true for thermal conductivity in particular because the thermal resistance at the coating interface is uncertain and probably process dependent.
- Initial oxidation is expected to be lower (prior to exceeding the Cr-Zr eutectic temperature), resulting from the presence of the chromium coating, as intended. This coating not only affects the oxidation behavior of this ATF, it also indirectly affects other phenomena such as fuel degradation/relocation behavior via how it potentially shifts the time frame during which rapid oxidation heat input is important. Note that cladding burst during a LOCA will allow oxidation of the interior surface of the cladding.

Cr-Doped UO_2 Fuel

The small amount of dopant added to the fuel primarily affects the grain size of the UO_2 and thus reduces the release of volatile radionuclides to the gap during normal operation, and the release of volatile fission products (e.g., iodine, cesium, etc.) during the early stages of a severe accident. In addition, it was concluded that there is no significant effect on hydrogen generation, and that there is virtually no effect on the later stages of a core meltdown accident. The panel therefore concluded that Cr-doped UO_2 fuel would not behave very differently from conventional fuels, with respect to the vast majority of severe accident phenomena. The only notable difference was fewer available experimental data (i.e., state-of-knowledge) regarding fission product speciation and chemistry. The limited available information does suggest that there might be a significant effect on oxygen potential in the fuel, from even small quantities of chromia added to the fuel.

High Burnup Fuel

A number of differences were noted that would affect severe accident behavior in comparison with lower burnup fuels. Among the points noted by the panel are:

- Some thermophysical properties of both fuel and cladding, particularly thermal conductivity and specific heat, may be affected by the higher burnup and irradiation.
- Fission product chemistry may be affected by the presence of more lower valence atoms in the fuel matrix as a result of higher burnup (i.e., greater degree of transmutation by fission of the uranium atoms). The larger magnitude of ruthenium release observed in some tests performed in pure steam atmosphere (i.e., without the presence of air) could be attributed to high burnup; however, this observation is not fully understood at present.
- Oxygen potential is expected to increase as a result of the larger concentration of lower valence atoms in the fuel, possibly affecting the rate of oxidation at the cladding inner surface. However, the oxygen potential in the fuel is buffered by the Mo/MoO_2 couple. Molybdenum inventories in high burnup fuels may be sufficient to prevent significant excursions of oxygen potential as burnup progresses.
- Gap inventories of fission products would be larger in an absolute sense, even if it is unclear whether they would be different when expressed as a fraction of the total core inventory of fission products.
- The fuel is expected to have a different amount of fragmentation or sintering at a higher burnup, which can in turn affect how the fuel behaves during core degradation and relocation, as well as phenomena such as temperature-induced creep rupture (of reactor coolant structures for high-pressure accident conditions) that can be affected by core debris particle size.
- The rate of release of volatile fission products has also been observed to be greater for high burnup fuels. Note that the timing of release of volatile fission products is of importance to the retention and transport inside the reactor coolant system. Nonetheless, it is not clear that this effect would significantly impact accident consequences since the volatile fission products are largely released from the fuel in early stages of a severe accident.
- Increased cladding embrittlement might be possible at a higher burnup, thus affecting the likelihood of core coolability upon reflood. However, this has not been closely examined to date.

High Burnup/ High Assay Low Enriched Uranium Fuel

The panel did not note significant qualitative differences in severe accident behavior applicable to HBU/HALEU fuels when compared with just HBU fuels, other than the fact that there may be a greater likelihood of recriticality in some accidents (e.g., accidents involving reflood following core damage using un-borated water).

Longer-Term ATF Concepts

PIRTs were not developed and used to assess the severe accident implications of the longer-term ATF concepts. While outside the scope of the PIRT process, the panel members were requested to provide their perspectives on the severe accident implications of the longer-term ATF concepts. Their contributions are documented in Appendix A.

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LIST OF ACRONYMS

ATF	Accident Tolerant Fuel
BDBA	Beyond Design Basis Accident (i.e., Severe Accident)
BISON	Fuel Analysis Computer code
BWR	Boiling Water Reactor
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas
CR	Control Rod
DBA	Design Basis Accident
DOE	Department of Energy
ECCS	Emergency Core Coolant System
EPRI	Electric Power Research Institute
FeCrAl	Iron-Chromium-Aluminum Alloy
FOM	Figure-of-Merit
FP	Fission Product
GNF	General Electric Nuclear Fuels
GWd/MTU	Gigawatt-day per metric tonne of heavy metal (burn-up unit)
HALEU	High Assay Low Enriched Uranium
HBU	High Burnup
IRSN	Institut de Radioproection et de Sûreté Nucléaire
JAEA	Japan Atomic Energy Agency
LEU	Low Enriched Uranium
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
MELCOR	Computer code name (not an acronym)
MCCI	Molten Core Concrete Interaction
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Development and Cooperation
ORNL	Oak Ridge National Laboratory
PCI	Pellet-Clad Interaction
PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor
QUENCH	Tests facility at the Karlsruhe Institute of Technology
RCS	Reactor Cooling System
RIA	Reactivity-Initiated Accident
RPV	Reactor Pressure Vessel

SA	Severe Accident
SBO	Station Blackout
SCC	Stress Corrosion Cracking
SiC	Silicon Carbide
SNL	Sandia National Laboratories
SOARCA	State-of-the-Art Reactor Consequence Analysis
SOK	State of Knowledge
TMI	Three Mile Island
TRISO	Tristructural-isotropic
UC	Uranium Carbide
UN	Uranium Nitride
UO_2	Uranium Dioxide
UW	University of Wisconsin
VERCORS	Radionuclide release test facility in France

1. INTRODUCTION

1.1 Background

The United States (U.S.) Nuclear Regulatory Commission (NRC) is preparing to accept anticipated licensing applications for the commercial use of accident-tolerant fuel (ATF) in U.S. commercial nuclear power reactors. Several fuel vendors in coordination with the U.S. Department of Energy (DOE), have announced plans to develop and seek approval for various fuel designs with enhanced accident tolerance (i.e., fuels with longer coping times during loss of cooling conditions). Vendors have also expressed interest in increasing fuel burnup above the licensed limit (which varies by vendor, but roughly corresponds to 62 GWd/MTU rod average), as well as increasing enrichment beyond 5 percent.

NRC needs to evaluate the effects of the various ATF designs on severe accident behavior and determine potential changes to its severe accident analysis code (i.e., MELCOR). Such changes would result in simulation capabilities for ATFs commensurate in accuracy with accident analyses involving conventional fuels.

The NRC has broadly categorized the ATF design concepts as near-term and longer-term. The near-term ATF concepts are those for which the existing data, models, and methods might be applicable with some modification and limited additional data. Longer-term ATF concepts are those for which substantial new data, models, and methods need to be developed.

The near-term ATF designs with respect to the time of commercial deployment include:

- Advanced stainless steel (FeCrAl) cladding,
- Cr_2O_3 and Al_2O_3 dopants in uranium dioxide fuels, and
- Conventional zirconium alloy cladding coated with chromium.

These ATF design proposals involve, for the most part, changes to the cladding. Such changes are manifestly significant because of the important role cladding plays in fuel degradation under severe accident conditions.

On the other hand, the longer-term ATF concepts under consideration include:

- Silicon carbide cladding
- High-density silicide fuels
- High-density nitride fuels
- Metallic fuels (specifically, uranium-zirconium alloys with zirconium content near 50 percent)

These ATF designs represent evolutions and deviations from the zirconium-alloy-clad uranium dioxide fuel forms.

In addition, the behavior of high burnup (HBU) and high assay, low-enriched fuel (HALEU) with less than 20 wt% U-235 is also of interest to the NRC. In the present context, HALEU is to be viewed as increased fuel enrichment for use in light water reactors (LWRs) and not as envisioned for non-LWRs. The LWR industry is anticipating fuel enrichments as high as 10 percent (but in reality, it is expected that plants may request more modest enrichment increases in order to achieve the desired burnup and fuel cycle targets).

The introduction of ATF, high-burnup, and/or increased enrichment for UO_2 fuels and ATF may impact the progression of severe accidents, release, and transport of radionuclides, with implications on safety and regulatory requirements. Phenomena such as cladding-steam interactions, cladding failure, fuel microstructure, eutectic formations, and release mechanisms (among others) need to be assessed. In addition, fuel burnup and enrichment may also impact severe accident progression and radiological releases through changes such as decay heat load, isotopic inventories, fuel/cladding thermo-mechanical properties, and fuel microstructure.

1.2 Objectives

An objective of the NRC is to evaluate the effects of ATF designs on severe accident behavior, and to determine potential changes to its severe accident analysis code (i.e., MELCOR) to better model those effects. Such changes should result in simulation capabilities for ATFs commensurate in accuracy with accident analyses involving conventional fuels.

The objective of this report is to develop PIRTs by an international panel of experts to assess modifications of existing computer codes and gaps in the available data needed to evaluate 'near term' ATF performance under severe accident conditions.

The panel was tasked with assessing the effect of near-term ATFs on reactor safety in particular; the potential for the release of radioactive material from the reactor coolant system; and the quantity of combustible gases produced that represent a challenge to containment integrity during severe accidents in both pressurized water reactor (PWR) and boiling water reactor (BWR) plants. In addition, the panel recommendations related to phenomenological and modeling considerations for some of the longer-term ATF concepts are also documented in Appendix A to this report.

The experts on the panel who participated in the formulation and ranking of the PIRTs represent the U.S. and the international research and nuclear industry community (see Appendix B). The five panel meetings were held virtually and included a number of observers from the U.S. nuclear industry and national laboratories (see Appendix C).

2. THE PHENOMENA IDENTIFICATION AND RANKING TABLE PROCESS

The objective of the Phenomena Identification and Ranking Table (PIRT) as defined by the NRC is to delineate relevant phenomena and solicit expert opinions on fuel degradation and failure phenomena, for select ATF concepts under severe accident conditions.

This effort includes identifying phenomena important to safety (i.e., through appropriately defined figures-of-merits); assessing of their importance; and the level of knowledge associated with each phenomenon.

The NRC anticipates that the PIRTs can be used by different entities for distinct purposes that include the following:

- Nuclear power reactor licensees/applicants, to develop and support their safety case,
- NRC to inform regulatory and computer code (e.g., MELCOR) development requirements, and
- DOE to prioritize its research activities.

This section details the PIRT process and steps for various ATF concepts.

2.1 Step 1 – Review of Available Literature

Reference [1] provides the first step—a survey of the available literature for various ATF concepts including impacts from fuel enrichment and burnup. The initial draft of Reference [1] was subjected to a peer review and subsequently revised to address the reviewers' comments, before being distributed to the PIRT panel. A few minor changes were implemented based on the comments by the panelists before embarking on the ranking process.

2.2 Step 2 – Plant and ATF Types

ATF design concepts are for both PWRs and BWRs. The primary focus of the present PIRT process is on the near-term ATFs, namely:

- Advanced stainless steel (FeCrAl) cladding,
- Conventional zirconium-alloy-cladding coated with chromium, and
- Cr_2O_3 dopant in uranium dioxide fuels.

The PIRT also considered high burnup high assay conventional fuels in addition to considering the impact of burnup and fuel enrichment on ATF fuels. In addition, the panel was requested to offer comments on the severe accident implications of the longer-term ATF design concepts. These comments are documented in Appendix A of the present report.

2.3 Step 3 – Accident Scenarios

To account for the impacts from accident-specific conditions, the PIRTs consider both high- and low-pressure accident scenarios, in addition to accounting for other accident-specific aspects that can influence the importance of relevant phenomena (e.g., reflood following the loss of core geometry).

The high-pressure accident scenario under consideration is a station blackout (SBO) with and without depressurization, whether it was intentional (e.g., operator action) or a consequence of accident progression (e.g., creep-rupture from an excessive heatup of the reactor coolant system pressure boundary at high-system pressure).

The low-pressure accident scenario includes a variant of an SBO or a large break loss-of-coolant accident (LOCA), and the failure of the emergency core coolant system (ECCS).

2.4 Step 4 – Phenomenological Issue Decomposition Process

The phenomenological issue decomposition processes related to radioactive material released from the reactor coolant system to the containment considers the various phases of severe accidents in both PWRs and BWRs. These include the in-vessel phase (i.e., prior to the breach of the reactor pressure vessel [RPV] lower head), as well as the ex-vessel phase (i.e., phenomenological processes following the RPV lower head failure). It is important to point out that the identification of phenomena excludes containment-specific processes.

The PIRTs document the issue decomposition process, listing PWR-specific, BWR-specific, and phenomena equally applicable to both plant types. Similarly, any distinctions for the ATF concept, burnup, and enrichment are also identified and discussed. Note that the panel also considered the same phenomenological issues and ranked conventional fuel designs as a reference, in order to provide a better perspective for then ranking the ATF, HBU, and HALEU fuel and cladding designs.

2.5 Step 5 – Applicable Figures-of-Merit

The following figures-of-merit (FOMs) are utilized to assess the importance of relevant phenomena:

- FOM-1: Source Term (i.e., the release of radionuclides into the containment atmosphere only)
- FOM-2: Generation rate and quantity of combustible gases (e.g., hydrogen)

To determine the importance ranking entails applying the assigned rank relative to one or both FOMs.

2.6 Importance Ranking

The importance/significance of identified phenomena relative to one or both FOMs is ranked using the following qualitative ranking scheme:

- High (H): Phenomenon has a significant impact on FOM.
- Medium (M): Phenomenon has a moderate impact on FOM.
- Low (L): Phenomenon has an insignificant impact on FOM.

A phenomenon or process can be important because it enhances or suppresses the figure-of-merit significantly. The importance rank assigned to each phenomenon differentiates between the inherent importance of a phenomenon and the equally important question of how ATF characteristics significantly change that phenomenon relative to conventional fuels. Discussions of the relevant technical bases (e.g., results of specific calculations, test data, etc.) support the assignment. In some instances an intermediate ranking such as “M/H” is used, when a more refined scheme was found to be necessary, or to obtain consensus among the experts.

2.7 State-of-Knowledge Ranks

The state-of-knowledge (SOK) relates to the identified phenomena that include analytic studies, any available separate effects, and/or integral experimental data. The SOK rankings are based on the following:

- High (H): Experimental data or validated simulations are either available or can be generated.
- Medium (M): Extrapolations of closely related experimental data are possible, or approximate models are either available or can be generated.
- Low (L): Applicable experimental data and/or approximate models are not available.

In areas where results of specific studies (i.e., analyses and/or experiments) are used to support the assignment of rankings, they are cited. Similar to the assignment of rankings to importance, in some instances an intermediate ranking such as “M/H” is used when a three-level scheme was found to be necessary or to obtain consensus among the experts.

2.8 Technical Adequacy of Severe Accident Models

The PIRTs do not include a column for ranks associated with technical adequacy. This is captured as part of the importance and state-of-knowledge rankings and the supporting rationales.

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3. PHENOMENA IDENTIFICATION AND RANKING TABLES

This section documents the results of the PIRT process, including the listing of relevant phenomena applicable to severe accidents and radiological releases and transport for various near-term ATF concepts.

3.1 Identification of Relevant Phenomena

The starting point for the PIRT process was the panel deliberation on the initial "strawman" list of phenomena applicable to severe accidents and radiological releases and transport documented in Reference [1]. This process resulted in the adaptation of a revised list that was subsequently used to develop importance and state-of-knowledge rankings by the panel. It is noted that the panel agreed to deliberate on various phenomena with the aim of arriving at a consensus, when possible. The panel also agreed that any potential disagreements would be documented as a dissenting opinion.

3.2 Rankings and Rationales

This section documents rankings and the associated technical basis (i.e., the rationale) for assigning ranks for the importance of each phenomenon related to severe accident combustible gas generation and fission product releases; as well as the transport-associated state-of-knowledge (i.e., experimental data and/or theoretical studies).

To provide a better perspective for the importance and state-of-knowledge ranks developed for various ATF design concepts, each phenomenon was also ranked for conventional fuels with zirconium-alloy cladding as the reference fuel/cladding design. These rankings also considered any differences between PWRs and BWRs and varying accident conditions.

The following is the key to the rankings for each fuel/cladding design:

- **H:** High importance/state-of-knowledge
- **M:** Medium importance/state-of-knowledge
- **L:** Low importance/state-of-knowledge
- **NA:** Not applicable or of negligible importance.

In some cases, a phenomenon is ranked separately for different sub-issues, time frames, components, and so forth; as a result of explicit distinctions by the panel or from a combination of categories in the PIRT.

3.2.1 General Remarks

Conventional Fuel/Cladding Designs

Conventional fuels are uranium dioxide fuel encased in zirconium-alloy cladding, which are currently used for operating LWRs designed and supplied by various fuel vendors. In the operating BWRs, the channel boxes also use the same zirconium-alloy materials used for the fuel cladding.

FeCrAl Cladding

This ATF design consists of UO_2 fuel encased in an advanced stainless steel chromium-aluminum alloy (FeCrAl) cladding [1-2]. The FeCrAl material is only considered to be fuel cladding, while other in-core structures—most notably the BWR channel boxes—will currently continue to use the same zirconium-alloy material.

FeCrAl is designed to provide an excellent mechanical behavior at high-temperature severe accident conditions. Exceptional steam oxidation resistance has been observed for FeCrAl at a temperature near its melting point [1-2], resulting in a reduction in heat and hydrogen generation during accident conditions. Therefore, compared to Zr-alloy cladding, the FeCrAl cladding designs are expected to increase coping time, enhance the ability to maintain a coolable geometry, enhance fission product retention, reduce coolant oxidation reaction kinetics with coolant, and increase allowable peak cladding temperatures during operational and accident conditions [1-2].

The high-temperature oxidation resistance of FeCrAl alloys relies on the formation of a protective alumina scale, which may be challenged during fast transients [1-2]. Even considering that the temperature at which a significant interaction between oxidized cladding and UO_2 fuel starts, it is likely that oxidation would be lower than that of conventional cladding. High-temperature oxidation resistance also comes with a penalty of the higher neutron absorption of steel, although the higher absorption may be mitigated by the use of thinner cladding and an increased fuel pellet radius, or increased fuel enrichment [1-2]. Other potential benefits and penalties associated with FeCrAl ATF are summarized in References [1-2] and will not be repeated here.

Cr-coated Zirconium-Alloy Cladding

A thin protective Cr coating on the surface of zirconium-based alloys is being considered for use in LWRs. The thin coating is expected to have a small effect on the thermal-mechanical behavior of Zr-based cladding, while it has the potential to enhance its corrosion resistance and high-temperature steam oxidation resistance during severe accidents. Therefore, the thickness of the coating plays a major role in maintaining the zirconium substrate properties and behavior as long as the cladding temperature is lower than the Cr-Zr eutectic temperature. In general, for coatings with a thickness less than 20 μm, the neutronic impact on the fuel cycle cost or cycle length is expected to be small and can be easily compensated for by very slight design modifications.

Similar to the FeCrAl cladding designs, the Cr-coated zirconium-alloy is only considered as fuel cladding. Yet at the present time, the other in-core structures—most notably the BWR channel boxes—will continue using the same zirconium-alloy material.

Cr-doped UO_2 Fuels

Modification of UO_2 fuel by Cr_2O_3 and Al_2O_3-Cr_2O_3 dopants is mainly for improving resistance to stress corrosion cracking (SCC) of the cladding caused by stress on the cladding from the pellet-clad interaction (PCI); and also for improved fission gas retention [1-2]. Some fuel vendors have chosen chromia (Cr_2O_3) as the relevant dopant for obtaining the desired fuel large-grain microstructure and enhanced visco-plastic behavior [1-2]. However, the chromia content is small (e.g., ~0.16 wt%). For a given optimized chromia content level, large grains favorably increase Cr_2O_3-doped fuel visco-plasticity. These features provide a lower stress-resistance capability of Cr_2O_3-doped fuel compared to the conventional UO_2 fuel. This fuel is characterized by a homogenous large-grain microstructure (i.e., 50-60 μm) expected to provide fuel performance benefits (e.g., dimensional stability, improved behavior in case of water/steam ingress, superior PCI and SCC-PCI resistance, a higher fission gas retention capability, etc.). Furthermore, the crystalline growth is expected to enhance the fuel matrix densification. A large database is available with a maximum rod burnup of approximately 75 GWd/MTU, as summarized in References [1] and [2].

Another vendor has developed UO_2 fuel containing Al_2O_3 and Cr_2O_3. The additives are expected to facilitate densification and diffusion during sintering, which results in about a 0.5 percent higher density within a shorter sintering time and with grains about five times larger than standard UO_2 fuel [1-2]. Measurements reportedly show that Al_2O_3 enhances the grain size with a pronounced effect of Cr_2O_3. The properties of the Al_2O_3-Cr_2O_3-doped pellets are very similar to pellets doped only with Cr_2O_3; Al_2O_3 can be viewed as a way to lower the total amount of dopant [1-2]. The present evaluation of ATF concepts does not include the Cr_2O_3 and Al_2O_3 doped or additive ATF pellet technologies, which are also being used in some operating plants (mostly in Europe). The doped fuel designs are expected to use cladding materials that are zirconium-alloy-based.

<u>HBU and HBU/HALEU Fuel/Cladding Designs</u>

The HBU fuel designs are considered as conventional fuel that will be operated to relatively high burnups. On the other hand, HBU/HALEU denotes designs of higher than conventional enrichment (i.e., greater than 5 percent but less than 10 percent), which again will be operated to relatively high burnups.

The current burnup upper limit is about 62 GWd/MTU (peak rod average). For HBU and HBU/HALEU, the burnup upper limits are as high as 80 GWd/MTU. However, for the present PIRT purposes specific designs with a burnup upper limit no greater than 80 GWd/MTU were considered.

It is noted that the panel considered that HBU/HALEU fuels may increase the importance of transient overpower for design basis accidents, but this is not an issue that can impact severe accidents. Therefore, the initial condition of the fuel is not considered to be impacted.

The following notes and/or assumptions are applicable to the phenomena evaluation and the ranking of importance and state-of-knowledge by the panel for the HBU/HALEU PIRT:

1. The nature of the dominant sequences leading to severe accidents for HBU/HALEU fuel is similar to conventional fuels. In particular, it is assumed that high assay fuels are not likely to increase the likelihood and importance of reactivity insertion accidents. For reactivity insertion accidents leading to high deposition of energy in the fuel, the high burnup fuel with embrittled cladding could lead to initial core configurations in which the cladding has failed and fuel may be dispersed. This is not a consideration in the present assessment.

2. Even though high assay fuels could potentially increase the likelihood of recriticality, the nature of the recriticality is a transient and slow process that would not approach prompt criticality. The recriticality would involve quasi-steady conditions for some period of time in which feedback mechanisms limit the increase in the power level, and fuel migration is expected to eventually lead to a subcritical state.

3. With these assumptions, the differences between the behavior of high burnup conventional fuel and high burnup HALEU fuel may not be substantial, except as noted in the present evaluation.

4. There is no basis for considering whether or not the HBU and/or HBU/HALEU designs incorporate any of the ATF characteristics (such as FeCrAl cladding or Cr-doped fuel).

5. On the differences between HBU and the HBU/HALEU designs as defined above, the panel's opinion is that except in a few cases, there is no specific impact from enrichment. Therefore, the rankings assigned to HBU also apply to HBU/HALEU.

6. The HALEU designs will likely use increased amounts of poison to counteract the effect on core reactivity from the higher enrichment. The extra poison could be in the form of components (e.g., poison rods) of a conventional design. Boron coating on fuel surfaces and gadolinium (Gd) in different parts of the core that may be present in the HALEU fuels would mitigate any reactivity implications of higher enrichments. Therefore, the panel concluded that with respect to the phenomena of interest, in most cases the HALEU designs tend not to behave significantly different from conventional designs. Even though it is quite likely that HBU and HBU/HALEU designs could be developed employing near-term ATF fuel concepts (e.g., FeCrAl cladding, Cr-doped fuel, etc.). At this time, there are no available ATF design concepts that would provide a basis for assessment.

3.2.2 Ranking of Importance and State-of-Knowledge for Each Fuel/Cladding Design

In this section, the consolidated ranking of importance and state-of-knowledge for the identified phenomena applicable to each fuel/cladding design is followed by the panel's rationale for supporting the PIRT assignments.

The importance and state-of-knowledge ranks for each of the PIRT-listed issues for conventional fuels, ATFs, high burnup and HALEU are listed in Tables 3.1 to 3.38. Tables 3.39 through 3.42 summarize the PIRT rankings for each ATF (and HBU/HALEU) compared with conventional (low-enriched) fuel cladding designs.

Mass Densities – Fuel, Cladding, Channel Boxes, and Control Rods

Table 3.1 Importance and State-of-Knowledge Ranks for Mass Densities (Fuel, Cladding, Channel Boxes, and Control Rods)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
Fuel/Cladding	L	H	L	H
Channel box/CR[2]	L	H	L	H
FeCrAl Cladding				
Fuel/Cladding	L	H	L	H
Channel box/CR	L	H	L	H
Cr-coated Zr-alloy Cladding				
Fuel/Cladding	L	H	L	H
Channel box/CR	L	H	L	H
Cr-doped UO_2 Fuel				
Fuel/Cladding	L	H	L	H
Channel box/CR	L	H	L	H
HBU Fuel				
Fuel/Cladding	L	H	L	H
Channel box/CR	L	H	L	H
HBU/HALEU Fuel				
Fuel/Cladding	L	H	L	H
Channel box/CR	L	H	L	H

<u>*Rationales:*</u>

Conventional Fuels

This property is well known for conventional fuels, and the panel agreed that densities of intact components/pure materials are not of significant importance.

FeCrAl Cladding

High importance ought to imply either a first-order impact on the FOM or a potential cliff-edge effect. It is noted that the results from a code calculation are unlikely to be impacted by incorrect mass density, as long as the total mass of the components is correct. There was agreement that the importance of mass density data would be low for all components. With the fuel remaining as a conventional UO_2,, the state-of-knowledge is high.

[2] CR: Control Rod

Cr-coated Zirconium-alloy Cladding

The densities of the constituent materials are well known in this case. The importance of the phenomenon should be minimally changed from conventional zirconium-alloy cladding, given that the thickness of the cladding chromium layer is very small. Therefore, it should not qualitatively affect its behavior in this regard.

Cr-doped UO_2 Fuel

The amount of chromia additive (or alternative dopants) is too small to have an effect on most of the physical properties of the fuel [1-2]. Uncertainties in these densities are small, and do not have a significant impact on the figures-of-merit. Furthermore, studies for conventional fuels and cladding have not shown any significant dependencies on mass densities. Therefore, the panel considered both the importance and state-of-knowledge rankings for conventional fuels to be also applicable to the Cr-doped UO_2 fuels.

HBU Fuel

The panel agreed that there are no specific impacts from burnup, and the rankings applicable to various fuel types remain unchanged.

HBU/HALEU Fuel

The panel agreed that there are no specific impacts from burnup or enrichment, and the rankings applicable to various fuel types remain unchanged.

Mass Density – In- and Ex-Vessel Molten Configurations/Melts

Table 3.2 Importance and State-of-Knowledge Ranks Mass Density (In- and Ex-Vessel Molten Configurations/Melts)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	L	M	L	M
FeCrAl Cladding				
	L	L	L	L
Cr-coated Zr-alloy Cladding				
	L	M	L	M
Cr-doped UO_2 Fuel				
	L	M	L	M
HBU Fuel				
	L	M	L	M
HBU/HALEU Fuel				
	L	M	L	M

<u>Rationales</u>:

Conventional Fuels

The panel reached consensus that the importance of melt density to fission product releases and hydrogen generation would be low. There is some experimental data available, hence the state-of-knowledge is ranked as medium.

FeCrAl Cladding

Most panelists agreed that the importance of melt density would be low. Not only from the low influence of density on the FOM, but because most of the volatile fission product releases would have been complete by the time the core reaches a molten configuration. Both fission product releases and steam oxidation to form hydrogen depend on the availability of a free surface exposed to the atmosphere. The surface-to-volume ratio of the melt is also low in comparison with the intact components. Although for FeCrAl cladding the release of volatile radionuclides may not be complete at the time the molten pool of fuel and cladding form, and there could be an impact on stratification, the panel agreed that the overall importance of mass density on both FOMs is low. The panel also concluded that the state-of-knowledge for density of melts resulting from FeCrAl clad fuels is low. The fact that there is the potential for stratification of a molten corium (i.e., with different molten material constituents) impacting the melt densities; and phenomena such as retention/revaporization in the RCS (as discussed later) could influence FOM-1. Nonetheless, the panel agreed that the overall importance of mass density on both FOMs is low. The panel also concluded that the state-of-knowledge for the density of melts resulting from FeCrAl-clad fuels is low.

Cr-Coated Zirconium-Alloy Cladding

The consensus was that melt density should not be significantly impacted by the relatively small amount of added chromium in the core.

Cr-doped UO_2 Fuel

The amount of chromia additive is too small to affect the rankings for the molten configurations. Therefore, the panel considered both the importance and state-of-knowledge rankings for conventional fuels to be also applicable to Cr-doped UO_2 fuel.

HBU Fuel

The panel agreed that there are no specific impacts from burnup, and the rankings as applicable to various fuel types remain unchanged.

HBU/HALEU Fuel

The panel agreed that there are no specific impacts from burnup or enrichment, and the rankings as applicable to various fuel types remain unchanged.

Thermal Conductivities – Fuel, Cladding, Channel Boxes, and Control Rods; Melt

Table 3.3 Importance and State-of-Knowledge Ranks for Thermal Conductivities (Fuel, Cladding, Channel Boxes, and Control Rods; Melt)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
Fuel	M	H	M	H
Cladding	M	H	M	H
Channel Boxes & CR	L	H	L	H
Melt	L	M	L	M
FeCrAl Cladding				
Fuel	M	H	M	H
Cladding	M	M	M	M
Channel Boxes & CR	L	H	L	H
Melt	L	M	L	M
Cr-coated Zr-alloy Cladding				
Fuel	M	H	M	H
Cladding	M	M	M	M
Channel Boxes & CR	L	H	L	H
Melt	L	M	L	M
Cr-doped UO_2 Fuel				
Fuel	M	H	M	H
Cladding	M	H	M	H
Channel Boxes & CR	L	H	L	H
Melt	L	M	L	M
HBU Fuel				
Fuel	M	M	M	M
Cladding	M	M	M	M
Channel Boxes & CR	L	H	L	H
Melt	L	M	L	M
HBU/HALEU Fuel				
Fuel	M	M	M	M
Cladding	M	M	M	M
Channel Boxes &	L	H	L	H

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
CR				
Melt	L	M	L	M

Rationales:

Conventional Fuels

The panel consensus was that thermal conductivities are of medium importance for both fuel and cladding, and of low importance for channel boxes and control rods/blades; while the state-of-knowledge is high for all conventional intact components. However, melt properties are of low importance, and there is only medium state-of-knowledge, because there are no means for reliably calculating the thermal conductivity of melts.

FeCrAl Cladding

It is noted that there is a potential for thermal conductivity to have some influence on the fuel temperature distribution and therefore, the timing of core degradation and fuel-cladding interactions. However, given the low degree of uncertainty associated with the fuel thermal-conductivity, it is unlikely that it would influence either figures-of-merits, and that the expected temperature gradients would be relatively low. The state-of-knowledge is medium for the FeCrAl cladding and high for zirconium-alloy channel boxes and control rods/blades.

Cr-coated Zirconium-alloy Cladding

The importance remains the same as for conventional fuels. There is uncertainty as a result of the thermal resistance between the coating and the bulk cladding (probably dependent on the method by which the coating is applied), which reduces the state-of-knowledge ranking from high to medium in this case.

Cr-doped UO_2 Fuel

Only the property of the fuel is affected and the amount of dopant is too small to affect the rankings relative to those of conventional fuel. Therefore, the panel considered both the importance and state-of-knowledge rankings for conventional fuels to be also applicable to the Cr-doped UO_2 fuel.

HBU Fuel

There are a number of factors that impact the change in fuel thermal conductivity with burnup resulting from the changing character and number of lattice vacancies; as well as the growing porosity of the fuel as fission gases accumulate on the fuel grain faces and edges. Other factors include the dissolved solid fission products, solid precipitated fission products, radiation damage, porosity and gas bubbles, deviation from stoichiometry, and pellet cracking. Cladding thermal conductivity also changes with burnup because of the ongoing annealing. Work at Karlsruhe [3] shows a large difference between the thermal conductivities of fresh versus modestly irradiated fuel (to about ~25

GWd/MTU). But the difference between the conductivities of highly burned and very highly burned fuel (up to ~100 GWd/MTU) is small. At high temperatures some of the decrease of thermal conductivity due to burnup is recovered due to temperature dependence [4]. The effect of burnup on thermal conductivity is not considered to be important during severe accidents. Therefore, the HBU rankings are considered identical to those of conventional designs, except for the state-of-knowledge rankings for fuel and for cladding, which are reduced by one level (i.e., from H to M).

HBU/HALEU Fuel

There are a number of factors that impact the fuel thermal conductivity, a function of burnup discussed above. Concerning HALEU, there may be some impact from the composition of the burnable material (e.g., it may change the transport of fission products through the fuel pellet). However, a large impact on rankings relative to those listed above for HBU, is not expected.

Specific Heats – Fuel, Cladding, Channel Boxes, and Control Rods; Melt

Table 3.4 Importance and State-of-Knowledge Ranks for Specific Heats (Fuel, Cladding, Channel Boxes, and Control Rods; Melt)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State of Knowledge	Importance	State of Knowledge
Conventional Fuels				
Fuel	M	H	M	H
Cladding	M	H	M	H
Channel Boxes	L	H	H	H
CR	L	H	M	H
Melt	M	H	M	H
FeCrAl Cladding				
Fuel	M	H	M	H
Cladding	M	H	M	H
Channel Boxes[3]	L	H	H	H
CR	L	H	M	H
Melt	M	H	M	H
Cr-coated Zr-alloy Cladding				
Fuel	M	H	M	H
Cladding	M	H	M	H
Channel Boxes	L	H	H	H
CR	L	H	M	H
Melt	M	H	M	H
Cr-doped UO_2 Fuel				
Fuel	M	H	M	H
Cladding	M	H	M	H
Channel Boxes	L	H	H	H
CR	L	H	M	H
Melt	M	H	M	H
HBU Fuel				
Fuel	M	M	M	H
Cladding	M	H	M	H
Channel Boxes	L	H	H	H
CR	L	H	M	H
Melt	M	H	M	H
HBU/HALEU Fuel				
Fuel	M	M	M	H

[3] It is assumed that channel boxes use conventional material.

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State of Knowledge	Importance	State of Knowledge
Cladding	M	H	M	H
Channel Boxes	L	H	H	H
CR	L	H	M	H
Melt	M	H	M	H

Rationales:

Conventional Fuels

It is recognized that the specific heat and enthalpy are related by thermodynamic relations through the Gibbs free energy. Enthalpy used in the severe accident computer codes is simply the integral of specific heat over a temperature interval (i.e., the specific heat is not used directly). Specific heat does not affect free-energy so much because the effect of specific heat on enthalpy is countered by a negative term involving the temperature times an integral over temperature of specific heat divided by temperature. In most cases, an adequately accurate estimation of heat capacity of complex mixtures arising during severe core degradation can be made by weighted averaging of the heat capacities of the pure constituents of the mixture. Therefore, specific heat is well known (i.e., high state-of-knowledge) for both the intact components and mixtures/amalgamations, including melts.

Under severe reactor accident conditions in BWRs, channel box and control rod materials near the center of a core can be at a higher temperature than the fuel near the periphery of the core. Furthermore, because of the radiation heat transfer as well as convective heat transfer, the core structural materials may become hot enough to participate in the degradation process. Nonetheless, the importance of the channel boxes and control rods/blades on fission product release and combustible gas generation is downgraded by the fact that temperatures of those components are generally lower than the fuel and cladding material.

FeCrAl Cladding

Similar arguments apply to the same importance and state-of-knowledge rankings as for the conventional fuels discussed above. The properties of thin alumina layers on metal alloys have been studied and there is reasonable information available. However, the properties of the oxidized and more complicated layers that develop as oxidation progresses are not well known. In this case, knowledge of the composition of the oxide layer is important. Alumina and chromia form a continuous solid solution, but the involvement of iron complicates the situation because of the tendency to form ferrous aluminate and spinels. Nonetheless, this consideration did not alter the overall state-of-knowledge ranking assigned by the panel. Note that because it is assumed that the channel boxes and control rod/blade material will remain zirconium-alloy, the importance and state-of-knowledge for those components all remain the same as for conventional fuels.

Cr-coated Zirconium-alloy Cladding

Rankings and rationales are the same as for conventional fuels, relatively unaffected by the small amount of chromium added to the cladding. Therefore, the panel considered both the importance and state-of-knowledge rankings for conventional fuels that also apply to Cr-doped UO_2 fuel.

Cr-doped UO_2 Fuel

The rankings and rationales are the same as those for conventional fuels, and they are relatively unaffected by the small amount of chromia dopant added to fuel.

HBU Fuel

It is recognized that at high burnup more gas is present in the fuel matrix. On the other hand, for a typical calculation the specific heat of un-irradiated UO_2 would probably be accurate enough, and such data are available [1]. These considerations and others somewhat similar to those for thermal conductivity led to assigning consensus HBU rankings that are almost exactly the same as those given to the conventional design; the only difference being the reduction of state-of-knowledge rank by one level. At least one panelist, however, felt some concern that the assignment of rankings to phenomena that are influenced by the formation of the high-burnup rim structure including specific heat, thermal conductivity, and possibly other phenomena may give readers a misleading idea that rim structure formation, and other aspects of HBU, are better understood than they really are.

HBU/HALEU Fuel

The panel did not consider the higher enrichment to affect the rankings assigned to HBU (as discussed above). The panel agreed to retain the same rankings for HBU/HALEU as for HBU.

Melting Points – Fuel, Cladding, Channel Boxes, and Control Rods

Table 3.5 Importance and State-of-Knowledge Ranks for Melting Points (Fuel, Cladding, Channel Boxes, and Control Rods)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
Fuel	L	H	L	H
Cladding	H	H	H	H
Channel Boxes	M	H	H	H
CR	H	H	M	H
FeCrAl Cladding				
Fuel	L	H	L	H
Cladding	H	L	H	L
Channel Boxes	M	H	H	H
CR[4]	H	H	M	H
Cr-coated Zr-alloy Cladding[5]				
Fuel	L	H	L	H
Cladding	H	M	H	M/L
Channel Boxes	M	H	H	H
CR	H	H	M	H
Cr-doped UO_2 Fuel				
Fuel	L	H	L	H
Cladding	H	H	H	H
Channel Boxes	M	H	H	H
CR	H	H	M	H
HBU Fuel				
Fuel	L	M	L	M
Cladding	H	H	H	H
Channel Boxes	M	H	H	H
CR	H	H	M	H
HBU/HALEU Fuel				
Fuel	L	M	L	M
Cladding	H	H	H	H
Channel Boxes	M	H	H	H
CR	H	H	M	H

[4] Relates to recriticality
[5] Pure substances

Rationales:

Conventional Fuels

Fuel slumping may occur before the fuel reaches its melting temperature, and melting points are important in determining the conditions for the relocation of partially molten fuel and cladding material. However, with the qualifier provided that these are for pure materials, there was agreement that the importance is low in most cases. The importance of a clad melting point was ranked high. The melting point of the control rods also has a high importance to fission product release because of the significant impact from control rod materials (i.e., as observed in the Phébus tests for boron-carbide control rods) on the chemical form of iodine as well as the influence of control rod degradation on the release of nonradioactive aerosols. On the other hand, the importance of the melting point of control rod material on hydrogen generation is considered to be medium. The importance of the melting point of BWR channel boxes to fission product release is also medium, but the importance to hydrogen generation is high because of the potential for failing channel boxes to open cross flows and expose steam- or oxygen-starved parts of the degraded core.

FeCrAl Cladding

Similar arguments apply to FeCrAl and to conventional fuels in most respects. The knowledge of the liquidus and solidus of alumina-chromia solid solutions is excellent. However, knowledge of the composition of the oxide and whether ferrous oxides are present in the mixture is inadequate. Therefore, given that one can develop a good knowledge of composition, it is possible to estimate phase relationships in the oxide and where liquefaction begins; thus reducing the state-of-knowledge ranking for FeCrAl cladding to low.

Cr-coated Zirconium-alloy Cladding

Similar to the case with FeCrAl cladding, we know relatively little about the composition of the oxidized cladding material; so the state-of-knowledge for the cladding is downgraded relative to conventional fuels (although not as much as for FeCrAl, the cladding only has a thin coating layer of new material instead of being entirely made from new material). France is planning experiments to extensively investigate the equilibria in the Zr-Cr and the Zr-Cr-O systems at high temperatures, so as to better estimate the degradation criteria of the cladding. Fuel rankings and rationales are the same as for conventional fuels. For channel boxes and control rods/blades, rankings are the same as for FeCrAl.

Cr-doped UO_2 Fuel

The small amount of dopant does have any impact on the melting point of the fuel. The other components of the reactor (e.g., channel boxes) are not affected by the fuel composition. Therefore, the importance and state-of-knowledge rankings for both figures-of-merits as considered for conventional fuels are also applicable to Cr-doped UO_2 fuel.

HBU Fuel

In principle, burnup affects the melting point of fuel by introducing ‘impurities’ into the fuel in the form of condensed phase fission products. There have been several attempts to measure the impacts of fission products on fuel melting. These studies found that any depression of the melting point caused by fission products was within the uncertainty of the temperature measurement. There may be more recent investigations that have had the precision to measure the melting point depression. To be sure, the effect is not nearly as profound as the interaction of molten cladding with the fuel. The panel assigned HBU rankings that are the same as those given to the conventional designs for all components except fuel, while the two state-of-knowledge rankings for fuel are reduced by one level.

HBU/HALEU Fuel

It was agreed that HBU/HALEU rankings are the same as HBU rankings.

Heats of Fusion – Fuel, Cladding, Channel Boxes, and Control Rods

Table 3.6 Importance and State-of-Knowledge Ranks for Heats of Fusion (Fuel, Cladding, Channel Boxes, and Control Rods)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
Fuel	L	H	M	H
Cladding	L	H	M	H
Channel Boxes	L	H	M	H
CR	L	H	L	H
FeCrAl Cladding				
Fuel	L	H	M	H
Cladding	L	H	M	H
Channel Boxes	L	H	M	H
CR	L	H	L	H
Cr-coated Zr-alloy Cladding				
Fuel	L	H	M	H
Cladding	L	H	M	H
Channel Boxes	L	H	M	H
CR	L	H	L	H
Cr-doped UO_2 Fuel				
Fuel	L	H	M	H
Cladding	L	H	M	H
Channel Boxes	L	H	M	H
CR	L	H	L	H
HBU Fuel				
Fuel	L	H	M	H
Cladding	L	H	M	H
Channel Boxes	L	H	M	H
CR	L	H	L	H
HBU/HALEU Fuel				
Fuel	L	H	M	H
Cladding	L	H	M	H
Channel Boxes	L	H	M	H
CR	L	H	L	H

Rationales:

Conventional Fuels

Heat of fusion affects core temperatures and the timing of relocation and affects both figures-of-merits. During the candling process molten materials drain down the fuel rods. There is an internal convection in the moving drops of fuel that enhances mass transport to the free surface. There is intense, exothermic oxidation that takes place in the moving droplets because they are hotter than the surrounding fuel. This results in hydrogen production by oxidation with steam. Furthermore, there is the potential for the additional releases of fission products during the slumping and relocation process, which are usually attributed to enhanced mass transport to the free surface by convection in the flowing material. However, the importance should not be overstated, since most of the fission products and hydrogen will have been released by the time a molten configuration is reached (even though later phenomena could affect the reactor coolant system conditions that can influence the retention/revaporization of fission products), and hence the release to the containment. Therefore, importance is rated low with respect to fission product releases, but medium with respect to hydrogen generation because of the potential impact on heat balance during oxidation and core degradation. The state-of-knowledge is high for the pure material.

FeCrAl Cladding

The same rankings and rationales as for conventional fuels.

Cr-coated Zirconium-alloy Cladding

The same rankings and rationales as for conventional fuels.

Cr-doped UO_2 Fuel

The same rankings and rationales as for conventional fuels.

HBU Fuel

The same rankings and rationales as for conventional fuels.

HBU/HALEU Fuel

The panel agreed that HBU/HALEU rankings are the same as those for the HBU (i.e., conventional fuels).

Coefficients of Thermal Expansion (Volumetric) – Fuel, Cladding, Channel Boxes, and Control Rods; Melt

Table 3.7 Importance and State-of-Knowledge Ranks for Coefficients of Thermal Expansion (Volumetric) (Fuel, Cladding, Channel Boxes, and Control Rods; Melt)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
Fuel	L	H	L	H
Cladding	L	H	L	H
Channel Boxes	L	H	L	H
CR	L	H	L	H
Melt	L	M	L	M
FeCrAl Cladding				
Fuel[6]	L	H	L	H
Cladding	L	H	L	H
Channel Boxes	L	H	L	H
CR	L	H	L	H
Melt	L	M	L	M
Cr-coated Zr-alloy Cladding				
Fuel	L	H	L	H
Cladding	L	H	L	H
Channel Boxes	L	H	L	H
CR	L	H	L	H
Melt	L	M	L	M
Cr-doped UO_2 Fuel				
Fuel	L	H	L	H
Cladding	L	H	L	H
Channel Boxes	L	H	L	H
CR	L	H	L	H
Melt	L	M	L	M
HBU Fuel				
Fuel	L	H	L	H
Cladding	L	H	L	H
Channel Boxes	L	H	L	H
CR	L	H	L	H
Melt	L	M	L	M

[6] For molten material, the importance can be higher (M or H); but it is a secondary effect.

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
HBU/HALEU Fuel				
Fuel	L	H	L	H
Cladding	L	H	L	H
Channel Boxes	L	H	L	H
CR	L	H	L	H
Melt	L	M	L	M

Rationales:

Conventional Fuels

The importance of the coefficient of thermal expansion is generally low. It might be important for estimating stress resulting from differential thermal expansion, but the severe accident codes do not account for differential thermal expansion effects. Furthermore, this is basically equivalent to temperature-dependent mass density (see the rankings for “Density”).

FeCrAl Cladding

The same rankings and rationales as for conventional fuels. Therefore, no discernable differences exist from conventional fuels, and the same rankings and rationales discussed for conventional fuels are applicable.

Cr-coated Zirconium-alloy Cladding

The same rankings and rationales as for conventional fuels. Therefore, no discernable differences exist from conventional fuels, and the same rankings and rationales discussed for conventional fuels are applicable.

Cr-doped UO_2 Fuel

Thermal expansion does not have a direct impact on fission product releases or on hydrogen production. Therefore, no discernable differences exist from conventional fuels, and the same rankings and rationales discussed for conventional fuels are applicable.

HBU Fuel

Discussions about HBU led to the global (all-designs) rank for melt, appearing above; along with the remark that rankings for the thermal expansion coefficient should generally agree with those for density. It was also mentioned that melt thermal expansion may be important, and the available models for melt thermal expansion are fairly reliable. A consensus was reached that the rankings for HBU is the same as those for the conventional designs.

HBU/HALEU Fuel

There was agreement that the rankings for HBU/HALEU are the same as those for HBU and the conventional fuel designs.

Surface Emissivity – Cladding and Melt

Table 3.8 Importance and State-of-Knowledge Ranks for Surface Emissivity (Cladding and Melt)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
Cladding	L	M	M	M
Melt	M	L	M	L
FeCrAl Cladding				
Cladding	L	M	M	M
Melt	M	L	M	L
Cr-coated Zr-alloy Cladding				
Cladding	L	M	M	M
Melt	M	L	M	L
Cr-doped UO_2 Fuel				
Cladding	L	M	M	M
Melt	M	L	M	L
HBU Fuel				
Cladding	L	M	M	M
Melt	M	L	M	L
HBU/HALEU Fuel				
Cladding	L	M	M	M
Melt	M	L	M	L

Rationales:

Conventional Fuels

Radiation heat transfer in the core is of some importance during severe accidents. It is noted that even though there is medium to high state-of-knowledge on emissivity for components other than the melt, the values entered for emissivity in severe accident codes are crudely parametric and do not necessarily bear much relation to those measured experimentally. It is noted that the representation of grey gas through which radiant energy must pass is uncertain. This gas consists of steam, dissociated steam, and hydrogen; it will also contain large particulate material that scatters radiant energy. View factors are as important as emissivity values. The most important elements of radiative heat transfer from a severe accident perspective are rod-to-rod in the early time frame (including during the period when the majority of volatile fission products are released); and then rod-to-molten pool or rod-to-crust in the later time frames. Importance is generally low to medium. This parameter is a candidate for treatment via an uncertainty analysis, because of the limitations in the radiation heat transfer models in the severe accident codes.

FeCrAl Cladding

The same rankings and rationales discussed for conventional fuels are applicable.

Cr-coated Zirconium-alloy Cladding

The same rankings and rationales discussed for conventional fuels are applicable.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge for this property are the same as those for conventional fuels. The panel agreed to assign the same ranks as those for conventional fuels.

HBU Fuel

It was agreed that the rankings for HBU are the same as those for the conventional fuel designs.

HBU/HALEU Fuel

It was agreed that the rankings for HBU/HALEU are the same as those for the conventional fuel designs.

Viscosity – ATF-Specific Molten Mixtures

Table 3.9 Importance and State-of-Knowledge Ranks for Viscosity (ATF-Specific Molten Mixtures)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
In-Vessel Melt	M	M	H	M
Ex-Vessel Melt	M	M	H	M
FeCrAl Cladding				
In-Vessel Melt	M	L	H	L
Ex-Vessel Melt	M	L	H	L
Cr-coated Zr-alloy Cladding				
In-Vessel Melt	M	M	H	M
Ex-Vessel Melt	M	M	H	M
Cr-doped UO_2 Fuel				
In-Vessel Melt	M	M	H	M
Ex-Vessel Melt	M	M	H	M
HBU Fuel				
In-Vessel Melt	M	M	H	M
Ex-Vessel Melt	M	M	H	M
HBU/HALEU Fuel				
In-Vessel Melt	M	M	H	M
Ex-Vessel Melt	M	M	H	M

Rationales:

Conventional Fuels

Viscosity affects slumping behavior in-vessel and debris spreading ex-vessel (medium or high importance). It is noted that viscosity is not present as a parameter in MELCOR for the in-vessel phase, even though viscosity is an input to the MAAP candling model and also may be used in other codes. Viscosity is a parameter used in the ex-vessel spreading models. It is expected that most of the in-vessel fission product releases will have ended by the time core slumping behavior occurs, which is the reason for ranking the importance as medium rather than high for releasing fission products to the containment. However, some references suggest that the impact from ex-vessel debris spreading could be significant for hydrogen generation. This resulted in a higher importance ranking for hydrogen production.

The viscosities of most molten core materials are rather small. Even molten UO_2 has a viscosity of about 50 centipoises. Molten structural metals have viscosities on the order of a few centipoises. The problem of viscosity is when the fluid is a solid-liquid mixture, as it is in core melt accidents. Viscosities increase with solid content, but the increase is also affected by the shape and size distribution of the solids, as well as by their concentration. Typically, it has been observed that the increase in viscosity with solid content is slow, until a threshold concentration is reached. At concentrations above this threshold, the viscosity increases rapidly. The mixture is in fact non-Newtonian in its rheology. The threshold is particularly dependent on the shape and size of suspended solid particles. It may be possible to estimate the solid fraction, but there is very little information on the shape and size distribution of the solids. Therefore, a medium rank is attributed to the state-of-knowledge.

FeCrAl Cladding

Similar rankings and rationales apply to conventional fuels, except that the state-of-knowledge of viscosities for melts resulting from FeCrAl fuel is low. The presence of more metal in the corium implies a longer duration of ex-vessel oxidation. However, this consideration did not alter the already medium-to-high importance rankings assigned by the panel.

Cr-coated Zirconium-alloy Cladding

The importance of this property is similar to that for conventional fuels and FeCrAl-clad fuel. The panel agreed that the state-of-knowledge may not be as high as it is for conventional fuels, but higher than it is for FeCrAl, thus ranking it as medium.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge for this property are the same as those for conventional fuels. The panel agreed to assign the same rankings as those for conventional fuels.

HBU Fuel

Consensus was reached that rankings for HBU fuel are the same as those for the conventional fuel designs; while also recognizing that compared with conventional fuel designs, not as much information is available for fission product releases during the in-vessel phase of severe accidents.

HBU/HALEU Fuel

It was agreed that the rankings for HBU/HALEU are the same as those for the conventional designs, as discussed above for HBU.

Surface Tension – ATF-Specific Molten Mixtures

Table 3.10 Importance and State-of-Knowledge Ranks for Surface Tension (ATF-Specific Molten Mixtures)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
In-Vessel Melt	L	M	L	M
Ex-Vessel Melt	M	M	H	M
FeCrAl Cladding				
In-Vessel Melt	L/M	L/M	L	L/M
Ex-Vessel Melt	M	L	H	M
Cr-coated Zr-alloy Cladding				
In-Vessel Melt	L	M	L	M
Ex-Vessel Melt	M	M	H	M
Cr-doped UO_2 Fuel				
In-Vessel Melt	L	M	L	M
Ex-Vessel Melt	M	M	H	M
HBU Fuel				
In-Vessel Melt	L	M	L	M
Ex-Vessel Melt	M	M	H	M
HBU/HALEU Fuel				
In-Vessel Melt	L	M	L	M
Ex-Vessel Melt	M	M	H	M

Rationales:

Conventional Fuels

Even though MELCOR does not use surface tension as an input to the models, this property can be important to assessing fuel-cladding material interactions and in- and ex-vessel melt progression. This is particularly important inasmuch as it affects wettability of fuel by the cladding (i.e., partially oxidized molten Zircaloy is expected to wet the fuel, but molten steel beads and would wet the fuel little or not at all if cladding is not oxidized). Surface tension affects the size of the molten droplets and also appears in Weber number correlations for depressurization and steam explosion modeling. A basis exists for at least estimating the surface tension, and current codes appear to correctly capture the big picture of surface tension impacts; hence, the state-of-knowledge is ranked as medium.

FeCrAl Cladding

Similar rankings and rationales apply as for conventional fuels. However, the implications for fission product releases are possibly higher (low/medium instead of low), resulting from the differences in the wettability of fuel by molten steel versus zirconium-alloy. The surface tensions of molten structural metals are similar, hence, differences owing to the use of FeCrAl cladding are expected to be small. Relevant data regarding surface tension of various metals as a function of temperature are provided in Reference [5] and for molten ionic oxides in Reference [6]. However, the state-of-knowledge is somewhat lower (i.e., low/medium as opposed to uniformly medium for conventional fuels).

Cr-coated Zirconium-alloy Cladding

The importance of this property is similar to that for conventional fuel and FeCrAl-cladding fuel. The panel agreed that the state-of-knowledge should be higher than for FeCrAl, ranking it as medium.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge for this property are the same as those for conventional fuels. The panel agreed to assign the same rankings as those for conventional fuels.

HBU Fuel

The panel agreed that the rankings for HBU are the same as those for the conventional fuels.

HBU/HALEU Fuel

The panel agreed that the rankings for HBU/HALEU are the same as those for the conventional fuels.

Phase Equilibria, Eutectic Formation Temperatures, Solid and Liquid Fractions, Fuel Solubility in Molten Cladding – ATF-Specific Molten Mixtures

Table 3.11 Importance and State-of-Knowledge Ranks for Phase Equilibria, Eutectic Formation Temperatures, Solid and Liquid Fractions, Fuel Solubility in Molten Cladding (ATF-Specific Molten Mixtures)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	H	M
FeCrAl Cladding				
	H	L	H	L
Cr-coated Zr-alloy Cladding				
	H	M	H	M
Cr-doped UO_2 Fuel				
	H	M	H	M
HBU Fuel				
	H	M	H	M
HBU/HALEU Fuel				
	H	M	H	M

Rationales:

Conventional Fuels

These phenomena clearly determine the pace and nature of core degradation/ relocation, as well as debris spreading ex-vessel. They are therefore ranked as high in importance. Both fission product releases and hydrogen generation would be impacted by cladding failure and post-failure oxidation. There is a well-developed ability to calculate detailed phase relationships among materials such as those that arise in severe reactor accidents. However, calculations of phase relationships are computationally intensive and usually incompatible with the needs of systems-level reactor accident analysis computer codes. The codes use approximations and correlations as necessary to represent the salient features of phase relations where they are of crucial importance. The panel ranked the state-of-knowledge as medium because of uncertainties, particularly for the UO_2/Zr-alloy monotectic.

FeCrAl Cladding

Of particular interest for this ATF is the relatively low-temperature interaction between stainless steel and its oxides and UO_2. Even though some data are available regarding these properties for FeCrAl, it was agreed that the state-of-knowledge ranking is low rather than medium (as in the case of conventional fuels), in order to emphasize the priority for further attention.

Cr-coated Zirconium-alloy Cladding

The importance of these issues remains the same as for conventional fuels. The state-of-knowledge is perhaps impacted by uncertainty regarding the behavior of the added small amount of chromium. Considering the relatively small amount of thin Cr-coating, the overall state-of-knowledge remains unchanged and is ranked as medium.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge rankings are the same as those for the conventional fuels. The panel agreed to assign the same rankings as those for the conventional fuels.

HBU Fuel

One panelist noted that altered fuel composition attributable to HBU does probably affect phase equilibria, but such changes are inconsequential in the present context. It was agreed that the rankings for HBU are the same as those for the conventional designs.

HBU/HALEU Fuel

It was agreed that the rankings for HBU/HALEU are the same as those for the conventional designs.

Heats of Solution or Mixing for Formation of Intermetallic Compounds – In- and Ex-Vessel Melts

Table 3.12 Importance and State-of-Knowledge Ranks for Heats of Solution or Mixing for Formation of Intermetallic Compounds (In- and Ex-Vessel Melts)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
In-Vessel/formation	L	M	M	M
In-Vessel/Solution	L	M	M	M
Ex-Vessel/formation	L	M	M	M
Ex-Vessel/Solution	L	M	M	M
FeCrAl Cladding				
In-Vessel/formation	L	M	L[7]	M
In-Vessel/Solution	L	M	L	M
Ex-Vessel/formation	L	M	L	M
Ex-Vessel/Solution	L	M	L	M
Cr-coated Zr-alloy Cladding				
In-Vessel/formation	L	M	M	M
In-Vessel/Solution	L	M	M	M
Ex-Vessel/formation	L	M	M	M
Ex-Vessel/Solution	L	M	M	M
Cr-doped UO_2 Fuel				
In-Vessel/formation	L	M	M	M
In-Vessel/Solution	L	M	M	M
Ex-Vessel/formation	L	M	M	M
Ex-Vessel/Solution	L	M	M	M
HBU Fuel				
In-Vessel/formation	L	M	M	M
In-Vessel/Solution	L	M	M	M
Ex-Vessel/formation	L	M	M	M
Ex-Vessel/Solution	L	M	M	M
HBU/HALEU Fuel				
In-Vessel/formation	L	M	M	M
In-Vessel/Solution	L	M	M	M
Ex-Vessel/formation	L	M	M	M
Ex-Vessel/Solution	L	M	M	M

[7] There is less reason to believe that this is as important for FeCrAl as it is for Cr-coated zirconium-alloy fuels.

Rationales:

Conventional Fuels

There are a few circumstances where the heat of solution is important. The heat of solution for stainless steel in zirconium is one. The heat of solution released during dissolution is very large — sometimes much larger than the heat of fusion of either metal. The dissolution of basic oxides into silicon dioxide during ex-vessel core debris interactions is another case. For most other circumstances, heats of solution are similar in magnitude to uncertainties in the enthalpies of formation of high temperature species. Their neglect does not greatly affect predictions of either the course of an accident or the figures-of-merit adopted for this study. Eutectic reactions between materials are frequently discussed because they lead to liquefaction at temperatures that are below melting points of the pure reactants. Phase equilibria have an important effect on fission releases because these equilibria lead to the partitioning of important fission products between the metal phase and the oxide phase of core debris. Particularly noteworthy is the partitioning of tellurium and ruthenium from the fuel into the metallic phase of core debris. Therefore, the overall importance of heats of solution is ranked low with respect to fission product releases and medium with respect to hydrogen generation. It is noted that the codes do not really take account of these properties. The state-of-knowledge is ranked medium because some data are available for these properties for conventional fuel material.

FeCrAl Cladding

The importance of this phenomenon for FeCrAl-clad fuels is lower relative to conventional fuels (i.e., uniformly low importance), because heats of solution are much smaller compared with zirconium-alloy-clad fuels. One can calculate the heat of a solution of iron into molten FeCrAl accurately. Existing data for binary combinations show that the heat of a solution of iron in chromium can be on the order of a kilocalorie. Aluminum into iron is exothermic and is important because of the same factors that lead to the formation of Laves phases and other inter-metallics in the system. The ternary interaction will not be as significant as the binaries because of the substantial dilution and the small activity coefficient of aluminum in molten iron-chromium-aluminum (on the order of 0.005). The most important element for this ATF may be the heat of a solution for the interaction of metallic melt with the lower head affecting the mode and timing of lower head failure. The state-of-knowledge is ranked medium because there are some relevant data available in the literature on phase equilibria in iron alloys, both because some of the heats of solution and mixing involving FeCrAl are not well known, and also because models are not available.

Cr-coated Zirconium-alloy Cladding

Because the cladding remains conventional zirconium-alloy with a relatively thin coating added, the rankings and rationales were assigned the same as for conventional fuels rather than for FeCrAl-clad fuel.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel thus agreed to assign the same rankings as those for the conventional fuels.

HBU Fuel

It was agreed that the ranks for both HBU are the same as those assigned for the conventional fuel/cladding designs.

HBU/HALEU Fuel

It was agreed that the ranks for HBU/HALEU are the same as those assigned for the conventional fuel/cladding designs.

Surface Roughness – Cladding

Table 3.13 Importance and State-of-Knowledge Ranks for Surface Roughness (Cladding)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	L	H	L	H
FeCrAl Cladding				
	L	H	L	H
Cr-coated Zr-alloy Cladding				
	L	H	L	H
Cr-doped UO_2 Fuel				
	L	H	L	H
HBU Fuel				
	L	H	L	H
HBU/HALEU Fuel				
	L	H	L	H

Rationales:

Conventional Fuels

It was noted that this property is more relevant to design and design-basis accident analysis, even though some critical heat flux correlations include surface roughness as a parameter. In addition, there is not a complete understanding of how the surface roughness changes as a function of the extent of cladding oxidation. It is also strongly dependent on the fabrication process. In the context of severe accidents, assignments of low importance and high state-of-knowledge was the general consensus among the panelists.

FeCrAl Cladding

Same rankings and rationales as for conventional fuels.

Cr-coated Zirconium-alloy Cladding

Same rankings and rationales as for conventional fuels.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for conventional fuels.

HBU Fuel

There was a consensus that the rankings for HBU are the same as those of the conventional fuel/cladding designs.

HBU/HALEU Fuel

It was agreed that the rankings for HBU/HALEU are the same as those of the conventional fuel/cladding designs.

Foaming Potential – Fuel, Cladding

Table 3.14 Importance and State-of-Knowledge Ranks for Foaming Potential (Fuel, Cladding)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
Fuel	M	H	M	H
Cladding[8]	L	H	L	H
FeCrAl Cladding				
Fuel	M	H	M	H
Cladding	H	M	H	M
Cr-coated Zr-alloy Cladding				
Fuel	M	H	M	H
Cladding	L	H	L	H
Cr-doped UO_2 Fuel				
Fuel	M	H	M	H
Cladding	L	H	L	H
HBU Fuel				
Fuel[9]	M	M	M	M
Cladding	M/H	L	M	L
HBU/HALEU Fuel				
Fuel	M	M	M	M
Cladding	M/H	L	M	L

<u>Rationales</u>:

Conventional Fuels

There is no recognized potential for significant fuel foaming in the case of zirconium-alloy-clad fuels (high state-of-knowledge). Phébus PF tests showed limited and transient fuel foaming. If there were significant foaming of fuel, it could affect both how the release of volatile fission products occurs and also the generation of hydrogen in response to potential flow blockages in the core (medium importance). It is noted that blockages need not be complete to divert

[8] Not really applicable as there is no evidence that zirconium-alloy cladding foams. Phébus PF tests showed very limited foaming, and that foaming is a transient process. However, this is not currently considered in severe accident computer codes.

[9] Fuel foaming is considered more applicable to HBU, because there is a greater potential for more extensive foaming at high burnup.

some fraction of the flow from the region where the blockage occurs to other unblocked regions of the core. Foaming of zirconium-alloy cladding has not been observed (low importance, high state-of-knowledge).

FeCrAl Cladding

The issue of fuel foaming remains unchanged from that of conventional fuels. Foaming of steels during melting has been observed in post-test examinations, and it may be anticipated that this will occur for FeCrAl cladding under severe accident conditions. Experimental data on that phenomenon are limited, and understanding the reasons for any foaming is tentative (possibly due to the presence of carbon), resulting in a medium state-of-knowledge ranking. Models for foaming are not currently present in codes such as MELCOR. The importance of foaming would include an increased surface area for steel oxidation, potential local blockages of gases, and aerosol flow through the core. Hence, foaming has a high importance to both figures-of-merits.

Cr-coated Zirconium-alloy Cladding

The thin coating of chromium on the cladding does not alter the assessment or rankings described above for conventional fuels.

Cr-doped UO_2 Fuel

The small added fraction of chromia in the fuel does not alter the assessment or rankings described above for conventional fuels.

HBU Fuel

Fuel foaming is driven by fission gas, and at higher burnup the fission gas content increases. Unpublished tests at the European Joint Research Center (JRC) demonstrated large fuel foaming at high burnup. (The JRC tests did not measure fission product releases and hence, do not bear on the state-of-knowledge for fission product release.) (As stated above for conventional fuels, Phébus tests that did not show any evidence of foaming were performed with fuel burnup of 30 to 40 GWd/MT, and not for HBU.) At least medium importance is appropriate for fission product release, and foaming could also affect hydrogen generation due to blockages, as noted previously for conventional fuels. Rankings of high on importance were ruled out by the consideration that only a portion of the core attains the highest burnups (i.e., ranked low on Importance for both figures-of-merits). The state-of-knowledge rankings are somewhat lower than those for conventional fuel designs (i.e., medium as opposed to high). Foaming of the cladding is of potentially increased importance because of the higher concentration of hydrogen in the cladding at high burnup (medium to high). However, there are very few test data to quantify the magnitude of the impact (i.e., low state-of-knowledge).

HBU/HALEU Fuel

The panel agreed that the rankings for HBU/HALEU should be the same as those for HBU fuel.

Fuel Wetability by Molten Cladding – Fuel, Cladding

All assigned rankings for this phenomenon were identical to those for surface tension and the associated rationales.

Oxidation Kinetics (including possible pressure-dependence and influence of high-temperature forms of cladding degradation) – Cladding, Channel Boxes[10], and Control Rod or Blade Materials

Table 3.15 Importance and State-of-Knowledge Ranks for Oxidation Kinetics (including possible pressure-dependence and influence of high-temperature forms of cladding degradation) (Cladding, Channel Boxes, and Control Rod or Blade Materials)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State of Knowledge	Importance	State of Knowledge
Conventional Fuels				
Cladding	H	H	H	H
FeCrAl Cladding				
Cladding	H	L	H	L
Cr-coated Zr-alloy Cladding				
Cladding	H	M	H	M
Cr-doped UO_2 Fuel				
Cladding	H	H	H	H
HBU Fuel				
Cladding	H	M	H	M
HBU/HALEU Fuel				
Cladding	H	M	H	M

<u>Rationales</u>:

Conventional Fuels

The chemical kinetics of conventional cladding oxidation is known to be insensitive to ambient pressure. Of course, when the rate of cladding oxidation is limited by mass transport of the oxidant (steam) to the reactive surfaces, pressure is a consideration because it affects convection. Oxidation is of high importance to hydrogen generation by definition, and the effect on fuel temperature by the addition of oxidation heat also makes it of high importance to

[10] When compared with the conventional fuel designs, it is assumed that the material for channel boxes will remain unchanged.

fission product release. There exists a good state of understanding of oxidation kinetics for conventional fuels.

FeCrAl Cladding

The oxidation of metals protected by thin layers of aluminum oxide is well understood. Very thick oxide layers that can rupture due to stresses in the epitaxially grown oxide are not as well understood. It is known that above the melting point, the alumina oxide layer is not passivating. It is usually thought that the rate of molten metal oxidation is limited by the rate of oxidant mass transport to the surface. An issue peculiar to reactor accidents is that steam can be present at pressures of over 100 bar. This high-steam partial pressure can lead to the formation of vapor phase aluminum oxides, so that the passivating layer of aluminum oxide can evaporate. The normal parabolic kinetics can then evolve into paralinear oxidation kinetics. There do not seem to be data to verify this possibility for FeCrAl oxidation in very high pressure steam. The phenomenon is ranked low for state-of-knowledge because while there is significant knowledge for the oxidation of FeCrAl up to near the melting point, there does not exist sufficient experimental data for oxidation at and above the melting point; particularly regarding very significant oxidation when the cladding fails and permits steam to access the unoxidized metal. It is possible that models would need to be augmented in some ways to properly treat FeCrAl oxidation.

Cr-coated Zirconium-alloy Cladding

There are data on the steam oxidation of chromium for temperature pertinent to design basis accident considerations. Issues of very thick oxide layers after prolonged exposure to oxidant do not arise in the case of the proposed fuel cladding simply because the chromium layer is very thin. There is much data on the oxidation of chromium in air and in oxygen. It is routinely observed that the kinetics of oxidation evolve to become paralinear because of the evaporation of chromium trioxide. Oxidation of chromium in very high-pressure steam might follow paralinear kinetics because the passivating layer of chromium oxide evaporates due to the formation of vapor phase hydroxides of chromium. There is a lack of data on the oxidation of chromium-coated zirconium-alloy at high temperatures (i.e., at temperatures exceeding the eutectic temperature of Zr-Cr). Therefore, even though the importance of this issue remains high, the state-of-knowledge is downgraded from high to medium.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same rankings as those for the conventional fuels.

HBU Fuel

The oxygen potential is changed by the transmutation of U into atoms of lower valence, leading to a burnup-dependent increase in the amount of oxidation of the cladding on the inside cladding surface. The details are not documented, but it is known that Industry is considering adjustments of zirconium-alloy composition for purposes of HBU designs. The expected adjustments include less tin and more niobium. Due to lesser knowledge about these issues, relative to the conventional design the state-of-knowledge rankings are reduced by one level (to medium), while importance rankings remain the same as those for the conventional designs (high).

HBU/HALEU Fuel

The panel agreed that HBU and HBU/HALEU should be ranked the same.

Oxidation Kinetics (including influence of cladding failure mode) (Late In-Vessel[11] and Ex-Vessel) – Melt

Table 3.16 Importance and State-of-Knowledge Ranks for Oxidation Kinetics (including influence of cladding failure mode) (Late In-Vessel and Ex-Vessel) (Melt)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
In-Vessel	M	M	M	M
Ex-Vessel	M/H	M	M/H	M
FeCrAl Cladding				
In-Vessel	M	M	M	M
Ex-Vessel	M/H	M	M/H	M
Cr-coated Zr-alloy Cladding				
In-Vessel	M	M	M	M
Ex-Vessel	M/H	M	M/H	M
Cr-doped UO_2 Fuel				
In-Vessel	M	M	M	M
Ex-Vessel	M/H	M	M/H	M
HBU Fuel				
In-Vessel	M	M	M	M
Ex-Vessel	M/H	M	M/H	M
HBU/HALEU Fuel				
In-Vessel	M	M	M	M
Ex-Vessel	M/H	M	M/H	M

Rationales:

Conventional Fuels

Much of the fission product releases early in an accident, such as a station blackout scenario, deposit on surfaces of the flow pathway through the reactor coolant system. Continued oxidation will produce chemical heat that can cause these deposited fission products to revaporize and to continue their transport into the containment. There is also a tendency to think that stages of a reactor accident take place throughout the core at the same time. Three Mile Island (TMI) showed that core melting can take place along the centerline of the core, while fuel assemblies displaced from this are largely unaffected, but they will degrade later in the accident. In-vessel melt oxidation may have an indirect effect on the release of volatile fission products and hydrogen. This impact can be

[11] "Late In-Vessel" is not to be confused with the NUREG-1465 definition for fission product releases.

considered as indirect or/and less significant, as most of the volatile fission products will have been released early in-vessel (overall medium importance). The importance rises slightly for ex-vessel melts (to medium/high), since the mechanisms are different (i.e., sparging through the melt, and mass transfer-controlled release), and since SOARCA [4] has shown the importance of late releases of elements such as cerium and lanthanum. Even though there is some understanding of melt oxidation, nonetheless, there are many uncertainties, and codes have some limitations (e.g., inability to model oxidation of melts while in motion), resulting in the assignment of an overall medium ranking for state-of-knowledge.

FeCrAl Cladding

There may be a higher content of unoxidized metal due to FeCrAl cladding, permitting more fission products and hydrogen and carbon monoxide releases. However, the differences are expected to be small in magnitude, and the same rankings as for conventional fuels were agreed to by the panel. Some panelists believe the state-of-knowledge is low in view of the limitations of current models, but the majority believe that existing models should be adequate. Assigning a low ranking might give the incorrect impression that priority should be given to further investigation of FeCrAl ex-vessel melt oxidation phenomena. It was therefore agreed that a rank of medium (as for conventional fuels) for state-of-knowledge is appropriate.

Cr-coated Zirconium-alloy Cladding

The oxidation behavior of the melt should not be significantly affected by the relatively small amount of added chromium from the coating. Hence, the rankings and rationales remain the same as those for conventional fuels.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same rankings as those for conventional fuels.

HBU Fuel

It was agreed that the rankings for HBU are the same as those for the conventional fuel designs.

HBU/HALEU Fuel

It was agreed that the rankings for HBU/HALEU are also the same as those for conventional fuel designs.

Gap Inventories/Pressure and Release at Cladding Failure – Fuel

Table 3.17 Importance and State-of-Knowledge Ranks for Gap Inventories/Pressure and Release at Cladding Failure (Fuel)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	L	M	N/A	
FeCrAl Cladding				
	L	L	N/A	
Cr-coated Zr-alloy Cladding				
	L	M	N/A	
Cr-doped UO_2 Fuel				
	L	M	N/A	
HBU Fuel[12]				
	L/M	M	N/A	
HBU/HALEU Fuel				
	L/M	M	N/A	

Rationales:

Conventional Fuel

The state-of-knowledge for this phenomenon is ranked medium, which reflects the crude/approximate (upper bound) way in which it is handled in severe accident computer codes (e.g., MELCOR). Its importance to severe accidents is low, as this phenomenon is more pertinent to design-basis accidents and success criteria evaluation.

FeCrAl Cladding

The state-of-knowledge in the case of FeCrAl-clad fuel is lower than for conventional fuels, because there is not a good understanding of how the cladding inner surface may interact with fission products. It is possible that the aluminum oxide on the inner surface of the cladding could trap fission products differently than for Zr-alloy, but such differences may be minimized in severe

[12] The impact on severe accident source terms of possible HBU fuel fragmentation under LOCA conditions, related to increased release of noble gases, and the potential for formation of particulates and their transport into the reactor coolant system, is considered insignificant. In the discussions, both short-term and long-term consequences were discussed, including for release by leaching during long-term recovery.

accident situations due to revaporization. It was also noted that in sodium-cooled fast reactors with stainless steel cladding, interactions between the cladding and the fission products have been observed that could impact this determination. The importance for severe accidents is low, for the same reasons as for conventional fuels.

Cr-coated Zirconium-alloy Cladding

Under the assumption that only the outer surface of the cladding receives a chromium coating, the situation is expected to be the same as for conventional fuels, and therefore, the rankings remain the same.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

At high burnup there will be larger gap inventories, which can be calculated by the existing methods. Hence, a small increase in importance ranking is indicated relative to the conventional design, while the same ranking remains for the state-of-knowledge as for the conventional fuels. It was argued that a gap inventory as a fraction of the total inventory would be unchanged relative to the fraction traditionally used to characterize conventional fuels; but the total inventory will be larger with an HBU.

HBU/HALEU Fuel

The panel agreed that the HBU/HALEU should be ranked the same as for HBU.

Fission Product Speciation and Chemistry – Fuel, Cladding

Table 3.18 Importance and State-of-Knowledge Ranks for Fission Product Speciation and Chemistry (Fuel, Cladding)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	N/A	
FeCrAl Cladding				
	H	M/L	N/A	
Cr-coated Zr-alloy Cladding				
	H	M	N/A	
Cr-doped UO_2 Fuel				
	H	M/L	N/A	
HBU Fuel				
	H	L/M	N/A	
HBU/HALEU Fuel				
	H	L/M	N/A	

Rationales:

Conventional Fuels

Speciation is highly important for estimating fission product transport within the reactor coolant system and into the containment. It is not relevant to a combustible gas generation. The panel assigned a medium rank to the state-of-knowledge.

FeCrAl Cladding

The combustible gas generation is not affected by fission product speciation and chemistry. This phenomenon is dictated to a large extent by the fuel, which remains conventional UO_2. Differences with FeCrAl cladding are probably relatively small, but this is not known with certainty. Oxides and hydroxides of aluminum and chromium that can interact differently with fission product species (including effects on the fraction of iodine in molecular versus particulate forms), and the aluminum oxide layer on the inner surface of the cladding may modify the evolution of stoichiometry of fuel during irradiation, as compared with conventional fuel. Moreover, differences in FeCrAl oxidation affecting the hydrogen/steam mixtures in the reactor coolant system may indirectly induce feedback and thus affect the fission product speciation. The state-of-knowledge ranking is lower than that for conventional fuels (low/medium); even though some information on speciation in the presence of aluminum and chromium can be estimated by thermodynamic calculations. However, there is a need for improvement in the state-of-knowledge in this area. (MELCOR uses frozen fission product inventories that do not evolve from one class [containing chemical species of similar characteristics] to another over time, so any important dynamic aspects of speciation due to FeCrAl in this regard cannot currently be represented by MELCOR).

Cr-coated Zirconium-alloy Cladding

The combustible gas generation is not affected by fission product speciation and chemistry. The issue remains of high importance for Cr-coated Zr-alloy cladding fuel. The relatively thin coating of chromium does not affect the situation enough to reduce the state-of-knowledge from medium, as it is for conventional fuels.

Cr-doped UO_2 Fuel

The combustible gas generation is not affected by fission product speciation and chemistry. The importance ranking is the same as for the conventional fuels. Some studies suggest that the presence of chromia in the fuel can affect volatile fission product chemistry and speciation, in a similar manner that they are affected by gadolinia in the fuel. It is believed that chromia in the fuel can affect speciation by separating fission products into categories that form chromium precipitates and ones that remain in the fuel matrix. However, the panel was divided on whether the relatively small amount of chromium dopant added to the fuel would have a significant effect. Experiments are planned in France to examine fission product release in the presence of chromia. At present, the consensus ranking for the state-of-knowledge was somewhat lower than that for conventional fuels (i.e., low/medium).

HBU Fuel

The combustible gas generation is not affected by fission product speciation and chemistry. High burnup results in more metallic fission products in the matrix and hence a higher oxygen potential. This in turn changes the chemistry of the released metallic fission products (e.g., Mo, Ru, Pd, etc.) to more oxidic and more volatile forms. One panelist stated that even in the absence of air, the ruthenium release was measured by VERCORS RT6 [7-8] to be high (~30%) under a steam atmosphere for HBU (70 GWd/MTU). Another panelist discussed still higher ruthenium releases (~60 %) from HBU fuels under conditions of air intrusion. Reference [9] reports the results of some studies on ruthenium chemistry during severe accidents. The panel considered that the uncertainties exist about the behavior of ruthenium, which led to a lowering of the rank for the state-of-the art (relative to the rankings of the conventional designs), while the importance rank (high) stands.

HBU/HALEU Fuel

The panel agreed that HBU/HALEU should be ranked the same as HBU.

Fission Product Release from Fuel during Core Heatup and Melting – Fuel, Cladding

Table 3.19 Importance and State-of-Knowledge Ranks for Fission Product Release from Fuel during Core Heatup and Melting (Fuel, Cladding)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	H/M	N/A	
FeCrAl Cladding				
	H	M	N/A	
Cr-coated Zr-alloy Cladding				
	H	H/M	N/A	
Cr-doped UO_2 Fuel				
	H	H/M	N/A	
HBU Fuel				
	H	M/L	N/A	
HBU/HALEU Fuel				
	H	H/M	N/A	

Rationales:

Conventional Fuels

Combustible gas generation is not affected by release of fission products from the fuel. However, fission product release from the fuel is of first-order (high) importance during core heatup and degradation. Not everything related to this phenomenon is known with certainty, but current empirical models appear to be adequate (i.e., medium/high ranks for state-of-knowledge).

FeCrAl Cladding

Significant differences resulting from the use of FeCrAl cladding as compared with conventional fuels are not expected, as the fuel pellet material remains conventional UO_2. Although FeCrAl-clad fuel is expected to behave in most respects similarly to conventional fuels, a main difference is expected to be the effect of oxygen potential on fission product releases resulting from differences in oxidation behavior in the presence of FeCrAl. The state-of-knowledge for this ATF is somewhat less than for conventional fuels (i.e., medium as opposed to medium/high).

Cr-coated Zirconium-alloy Cladding

There is currently no reason to believe that fission product releases would behave differently for this ATF than for conventional fuels, Hence, the rankings are identical.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same rankings as those for the conventional fuels.

HBU Fuel

For HBU, a consensus was reached that relative to the conventional fuel designs, the importance ranking remains the same (high); while the state-of-knowledge rank is lowered to M/L, thus reflecting various uncertainties. Many issues contributing to these uncertainties were discussed by the panel. For instance, HBU entails a faster release at low temperatures, likely converging to releases similar to those that also occur with low-burnup fuels if the temperatures are high. In typical calculations, the low-temperature regions tend to be for such a short time that it makes little difference to the results. Yet differences could arise in accidents conducive to longer periods at lower temperatures (e.g., accidents involving the spent fuel pool, transportation accidents, etc.). Steam- or air-limited scenarios could also keep temperatures lower longer. There is not a sufficient data base for high-pressure situations. At a higher burnup, the stronger link between the microstructure and the release lowers the state-of-knowledge because models based purely on diffusive release are considered inadequate. The argument should not be overemphasized; any reasonable model will predict the final release of all the iodine and cesium. However, for releases of non-volatiles, doubts remain. The significantly increased and poorly understood Ru releases for HBU, even without air, was noted earlier.

HBU/HALEU Fuel

The panel agreed that the rankings of the conventional design, not the HBU design, are applicable to HBU/HALEU designs. The rationale provided by one panelist is that the higher enrichment alters the role of plutonium fission, causing speciation and releases from HALEU fuel at a high burnup that more nearly resembles conventional fuels burnup rather than conventional fuel enrichment at high burnup.

Cladding Interactions Affecting Speciation and Chemisorption (including tellurium retention) – Fuel, Cladding

Table 3.20 Importance and State-of-Knowledge Ranks for Cladding Interactions Affecting Speciation and Chemisorption (including tellurium retention) (Fuel, Cladding)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	M	M	N/A	
FeCrAl Cladding				
	H	M	N/A	
Cr-coated Zr-alloy Cladding				
	M	M	N/A	
Cr-doped UO_2 Fuel				
	M	M	N/A	
HBU Fuel				
	M	M	N/A	
HBU/HALEU Fuel				
	M	M	N/A	

<u>Rationales</u>:

Conventional Fuels

Cladding interactions with fission products in the case of conventional fuel material are understood reasonably well, and the phenomena can be modeled. zirconium-alloy cladding is known to sequester tellurium, thus delaying its release to the reactor coolant system and/or containment. The importance of tellurium transport is increased by the fact that it decays into iodine. The idea of tellurium sequestration was initiated by findings of limited tellurium releases in out-of-pile tests at Oak Ridge National Laboratory (ORNL). It was subsequently decided that sequestration would be by the reaction of tellurium with the tin alloying agent. As cladding oxidation progresses, tellurium would be released as SnTe. There does not appear to be any observation of SnTe in fission product release experiments. Data based on experiments in which SnTe vapor was passed over stainless steel coupons, showed that it promptly reacted with the nickel in these coupons. The flow pathway in the Phébus experiments was largely a nickel alloy and sufficiently cool that had SnTe passed over this alloy, it would have reacted and probably been retained. Nonetheless, considerable Te entered the Phébus containment model of the tests. While the phenomenon of tellurium sequestration (or lack thereof) is well known [10], the existing models treat this process poorly.

There were varying opinions among the panelists, but eventually the panel agreed to rank the state-of-knowledge as medium.

FeCrAl Cladding

Compared with conventional fuels, the main difference is that FeCrAl cladding will not sequester tellurium in the same manner as zirconium-alloy will. Because of this effect on tellurium releases, the majority of panelists considered the importance to be high. Two panelists argued that the importance would be better characterized as medium, because this phenomenon acts as a modifier to the real first-order effect for the Booth diffusion-based releases from the fuel. Cladding interactions are somewhat less understood for this ATF than for conventional fuels, although the consensus state-of-knowledge was still ranked as medium.

Cr-coated Zirconium-alloy Cladding

The situation is not significantly affected by the relatively thin layer of chromium on the outside of the zirconium-alloy cladding. The rankings thus remain the same as for conventional fuels.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same rankings as those for the conventional fuels.

HBU Fuel

A consensus was reached that rankings for HBU are the same as those for the conventional design. One panelist accepted the consensus and felt that the importance and state-of-knowledge rankings may need to be lower, in this context, and he mentioned the behavior of tin telluride. Another panelist reiterated a remark about ways that for HBU applications, zirconium-alloy may be fabricated with less tin content.

HBU/HALEU Fuel

A consensus was reached that rankings for HBU/HALEU are also the same as those for the conventional design.

Fission Product Retention and Revaporization in the Reactor Coolant System (RCS) – Primary Systems

Table 3.21 Importance and State-of-Knowledge Ranks for Fission Product Retention and Revaporization in the Reactor Coolant System (RCS) (Primary Systems)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	N/A	
FeCrAl Cladding				
	H	M/L	N/A	
Cr-coated Zr-alloy Cladding				
	H	M	N/A	
Cr-doped UO_2 Fuel				
	H	M	N/A	
HBU Fuel				
	H	M/L	N/A	
HBU/HALEU Fuel				
	H	M/L	N/A	

Rationales:

Conventional Fuels

The main focus of interest is on revaporization since retention is a function of RCS surfaces and is not expected to be significantly impacted by cladding properties. Revaporization of fission products from the RCS is a dominant contributor to offsite releases, as shown in the SOARCA study [7, 11-13] and in observations of the Fukushima-Daiichi accident. These justify an importance rank of high for fission product releases. Of course, this phenomenon is not relevant to combustible gas generation. Revaporization and deposition behavior is a mass transport process, and the state-of-knowledge for conventional fuels is reasonably good. Improvements to the existing revaporization models can be guided by additional experimental data related to speciation, surface chemistry, and adhesion. These factors resulted in a final ranking of medium for the state-of-knowledge.

FeCrAl Cladding

Compared with conventional fuels, the focus here is on fission product transport behavior of any new species that might be formed in a system with FeCrAl cladding. Examples include those resulting from chromium and aluminum vaporization. Because of the uncertainty surrounding new species, the state-of-knowledge is slightly lower than for conventional fuels (i.e., low/medium instead of medium).

Cr-coated Zirconium-alloy Cladding

There may be potential revaporization of chromium hydroxide in this case, which could affect iodine chemistry in the primary circuit. Nevertheless, the overall state-of-knowledge was ranked the same as for conventional fuels (medium).

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The phenomena are of importance for the reactor coolant surfaces and are not affected by the fuel design. The panel agreed to assign the same rankings as those for the conventional fuels.

HBU Fuel

For HBU, a consensus was reached that, relative to the conventional design, the Importance rank also remains the same (high) while the state-of-knowledge rank is lowered from medium to medium/low due to various uncertainties. Contributing to the uncertainties, many issues were discussed some of which are certainly not peculiar to HBU. These included the possibility of forming more volatile Mo for HBU (and Cr-coat); the main reason being the difference in chemistry compared to conventional fuels. The understanding and modeling of formation and deposition of aerosols was also discussed, with attention to the behavior of cesium compounds (especially how and whether they re-vaporize after deposition inside steam generators depends on whether the Cs is in the form of CsOH, CsI, or Cs_2MoO_4). The SOARCA uncertainty study for Sequoyah [7] occasionally predicted a lot of Cs released early that would circulate through and deposit Cs in the steam generators, and this might or might not be resuspended and released if there were early containment failure. Available codes are considered adequate although they make the dubious assumption of uniform deposition onto large heat structure surfaces. A curious limitation to the accuracy that might ever be attained arises in the different ways duplex oxides form on stainless steel, depending on whether the oxidant is steam versus hot water. These discussions are also equally applicable to conventional fuels. With HBU one could get substantially higher releases of Ru whose heatup effects have not been examined closely. A number of tests with Ru were made in the results are available [14], hence, there is a rather reasonable knowledge that can be used for the modeling of ruthenium transport in the reactor coolant system.

HBU/HALEU Fuel

The panel agreed that the ranks for HBU/HALEU are the same as those of HBU.

Ex-Vessel Release during MCCI of Semi-Volatile Fission Products during MCCI – Melt

Table 3.22 Importance and State-of-Knowledge Ranks for Ex-Vessel Release during MCCI of Semi-Volatile Fission Products during MCCI (Melt)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	N/A	
FeCrAl Cladding				
	H	M	N/A	
Cr-coated Zr-alloy Cladding				
	H	M	N/A	
Cr-doped UO_2 Fuel				
	H	M	N/A	
HBU Fuel				
	H	M	N/A	
HBU/HALEU Fuel				
	H	M	N/A	

<u>Rationales</u>:

Conventional Fuels

The release of fission products during MCCI is of significant importance to the overall source term, even though the in-vessel release of volatile radionuclides is dominant. Additional factors which could possibly increase its importance would be the different particle size distribution resulting from MCCI, and the fact that a portion of ex-vessel release might occur after containment failure (depending on the accident scenario). Existing codes rely on a Gibbs free energy minimization and the available thermochemical data, where the need for additional thermochemical data and experimental validation studies may be warranted. Hence even though there are some data available, the overall state-of-knowledge is ranked as medium.

FeCrAl Cladding

Differences during MCCI between FeCrAl-clad fuels and conventional fuels are expected to be minor. The lack of zirconium content might decrease the release of volatile radionuclides during the early oxidation phase of MCCI due to lower oxidation heat. Essentially most of the volatile fission products (except for Te) would be expected to have been released during the in-vessel phase of accidents, and their releases are not of much significance during MCCI. Codes such as VANESA (i.e., used in MELCOR) are capable of modeling any iron and chromium present in the melt, and the addition of aluminum to the model would not pose any difficulties. VANESA actually considers the formation of aluminum hydroxide vapors since there is aluminum oxide in concrete. Hydroxide make no notable contributions to the vapor in comparison to the H_2, H_2O, CO, and CO_2 coming from the decomposition of concrete and reaction with the molten core debris. The knowledge is limited for release of fission products from melts containing FeCrAl, nonetheless, the panel ranked the state-of-knowledge the same as that for conventional fuels.

Cr-coated Zirconium-alloy Cladding

The relatively small amount of added chromium from the coating should not make a significant difference to the release of fission products during MCCI, hence the same rankings as those for conventional fuels were agreed to by the panel.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel reached a consensus that the ranks for HBU are the same as those for the conventional designs.

HBU/HALEU Fuel

The panel reached a consensus that the ranks for HBU/HALEU are the same as those for the conventional designs.

Tritium Release and Transport – Fuel, Melt

Table 3.23 Importance and State-of-Knowledge Ranks for Tritium Release and Transport (Fuel, Melt)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	L	M	N/A	
FeCrAl Cladding				
	L	M	N/A	
Cr-coated Zr-alloy Cladding				
	L	M	N/A	
Cr-doped UO_2 Fuel				
	L	M	N/A	
HBU Fuel				
	L	M	N/A	
HBU/HALEU Fuel				
	L	M	N/A	

Rationales:

Conventional Fuels

The phenomenon is ranked as low in importance since the severe accident source term is dominated by fission products rather than tritium. This is more of an operational and design basis accident than a severe accident issue, and severe accident codes do not currently model the release of tritium. The state-of-knowledge is considered as medium.

FeCrAl Cladding

Even if there is reason to believe tritium release might be somewhat higher for FeCrAl-clad fuels, this fact does not alter the overall evaluation of importance from that arrived at for conventional fuels (low), and the state-of-knowledge also remains medium.

Cr-coated Zirconium-alloy Cladding

Rankings and rationales are the same as for conventional fuels.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel agreed that the ranks for HBU are the same as those of the conventional fuel designs.

HBU/HALEU Fuel

The panel reached a consensus that ranks for HBU/HALEU are also the same as those of the conventional fuel designs.

Fission Product Capture in Water Pools – Cavity/Suppression Pool

Table 3.24 Importance and State-of-Knowledge Ranks for Fission Product Capture in Water Pools (Cavity/Suppression Pool)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	N/A	
FeCrAl Cladding				
	H	M	N/A	
Cr-coated Zr-alloy Cladding				
	H	M	N/A	
Cr-doped UO_2 Fuel				
	H	M	N/A	
HBU Fuel				
	H	M	N/A	
HBU/HALEU Fuel				
	H	M	N/A	

Rationales:

Conventional Fuels

Because of the potential effect of reducing the magnitude of release to the containment, it was agreed that the importance of this phenomenon is high. Qualifying this benefit is the potential for scrubbed fission products to leak from containment and constitute an aqueous release to the environment or under some circumstances could lead to a resuspension of scrubbed fission products. Consistent with the breadth of uncertainties in severe accident analysis, SPARC-based models are sufficient for regulatory applications. Because of uncertainties associated with bubble size, rise velocity, aerosol size distribution, aerosol deposition mechanisms, circulation within bubbles, bubble eccentricity and thermal stratification of pools, the state-of-knowledge is ranked as medium. There is still some current work on development and validation of pool scrubbing models (e.g., the European Research Projects PASSAM [15-16] or IPRESCA [17]).

FeCrAl Cladding

There is little reason to believe that the importance of this phenomenon or the applicability of SPARC-based models would differ significantly for FeCrAl as compared to conventional fuels. The same importance and state-of-knowledge rankings and rationale as discussed for conventional fuels are also applicable to FeCrAl.

Cr-coated Zirconium-alloy Cladding

The panel agreed that the rankings and rationales are the same as those for the conventional fuels.

Cr-doped UO_2 Fuel

Fission product capture in water pools is not affected by fuel design; hence the importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel reached a consensus that the ranks for HBU are the same as those for the conventional fuels.

HBU/HALEU Fuel

The panel agreed that the ranks for HBU/HALEU are also the same as those for the conventional fuels.

Release of Nonradioactive Aerosols – Fuel, Cladding, Control Rods, and Structural Materials

Table 3.25 Importance and State-of-Knowledge Ranks for Release of Nonradioactive Aerosols (Fuel, Cladding, Control Rods, and Structural Materials)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	N/A	
FeCrAl Cladding				
	H	M	N/A	
Cr-coated Zr-alloy Cladding				
	H	M	N/A	
Cr-doped UO_2 Fuel				
	H	M	N/A	
HBU Fuel				
	H	M	N/A	
HBU/HALEU Fuel				
	H	M	N/A	

<u>Rationales</u>:

Conventional Fuels

The quantity of inert aerosols released during core degradation has the potential to increase fission product deposition via increased agglomeration. On the other hand, higher amounts of these aerosols could have detrimental effects on source term due to clogging of filters or cooling coils in the containment. Most panelists agreed that the importance would be high for conventional fuels. There is some test data applicable to release and impact of nonradioactive aerosols to accident source terms, and the existing models in the codes for aerosol agglomeration and distribution of particle size are considered adequate (i.e., medium state-of-knowledge).

FeCrAl Cladding

Panelists generally ranked the importance of this phenomenon medium to high, with the final consensus being high since it is possibly determinative of the in-vessel component of release rather than being merely an indirect effect. In FeCrAl-clad versus zirconium-alloy-clad fuels, the primary difference is expected to be increased quantities of inert aerosols resulting from release of chromium and aluminum hydroxides in-vessel. Al_2O_3 and Cr_2O_3 are refractory compounds. Under ordinary circumstances they are much less volatile than tin. The concern is that under the conditions of core degradation while the vessel is still pressurized (i.e., high steam partial pressure), these can vaporize as vapor-phase hydroxides. These vapor-phase hydroxides are considered in the modeling of ex-vessel release but seldom do they make much of a contribution since the steam partial pressures in the containment are never high enough to produce high partial pressures of vapor phase hydroxides. Therefore, it is unlikely that the release fractions of tin, aluminum and chromium will be commensurate. Formation of the vapor phase hydroxides as contributions to the non-radioactive aerosol is simply a possibility that needs to be examined. The panel agreed that the ex-vessel phase of severe accidents is expected to behave similarly to conventional fuels with respect to inert aerosol generation. Additional effects include differences in particle size as well as potential formation of cesium chromate (which affects iodine behavior). There are some data on this phenomenon, even though there is an uncertainty regarding fission product speciation in the presence of chromium and aluminum, it was agreed to rank the overall state-of-knowledge the same as that for conventional fuels.

Cr-coated Zirconium-alloy Cladding

In view of the relatively small quantities of chromium added as a result of the coating, the issue of nonradioactive aerosol release remains basically unchanged as compared to conventional fuels, hence the rankings remain identical to those for conventional fuels.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

It was agreed that ranks for HBU are the same as those for the conventional fuel designs. It was mentioned again the change in tin and niobium content of zirconium-alloy as formulated for HBU. However, since most of the non-radioactive aerosols (in the absence of Ag-In-Cd control rods) originate as fission product (e.g., non-radioactive cesium): trace constituents of cladding alloy are not a very significant contributor even though aerosols originating as structural material are not insignificant.

HBU/HALEU Fuel

A consensus was reached that the ranks for HBU/HALEU are also the same as those for the conventional fuel designs.

Formation of Hexavalent Chromium (effects on Cesium retention, Iodine speciation, etc.) – Cladding

Table 3.26 Importance and State-of-Knowledge Ranks for Formation of Hexavalent Chromium (effects on Cesium retention, Iodine speciation, etc.) (Cladding)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	M	M/L	N/A	
FeCrAl Cladding				
	H	M/L	N/A	
Cr-coated Zr-alloy Cladding				
	H/M	M/L	N/A	
Cr-doped UO_2 Fuel				
	M	M/L	N/A	
HBU Fuel				
	M	M/L	N/A	
HBU/HALEU Fuel				
	M	M/L	N/A	

<u>Rationales</u>:

Conventional Fuels

The strongly oxidizing nature of hexavalent chromium could have a number of significant impacts on fission product transport, including the oxidation of iodide into elemental form. Even in the absence of chromium in the cladding material, there is already substantial quantity of stainless steel from structural materials in core with conventional fuels. The Phébus tests did not show significant releases of chromium which would have been revealed if there had been oxidation of chromium to form volatile CrO_3. Current thinking is that oxygen potentials were simply too low for any significant oxidation to the hexavalent state. Air must be present to get sufficiently high oxygen potentials for copious generation of chromium trioxide. Air will be present late in an unterminated accident when core debris has penetrated the lower head of the reactor pressure vessel. It is the same issue for conventional fuels and for ATF. The knowledge of the phenomenon is conjectural and incomplete. More research in this area would be desirable if it could be shown through additional code calculations that this could indeed have a significant impact on fission product release. The panel agreed to rank the state-of-knowledge as medium/low in view of these factors.

FeCrAl Cladding

Increased chromium content in the core due to the FeCrAl cladding somewhat increases the relevance of the phenomenon for this ATF (i.e., high instead of medium importance). The chromium in FeCrAl cladding may potentially form hexavalent chromium either in situations with air ingress, or where reduced oxidation is expected to sufficiently increase the oxygen potential of the gas mixture in the reactor coolant system (i.e., less hydrogen relative to steam). However, there is already substantial quantity of stainless steel in the structural materials in core with conventional fuels, hence it is not clear that the difference would be significant when using FeCrAl cladding. The panel agreed to rank the state-of-knowledge as medium/low, for the same reasons as discussed for conventional fuels.

Cr-coated Zirconium-alloy Cladding

The importance of this issue for chromium-coated zirconium-alloy-clad fuels is intermediate between that of conventional fuels and FeCrAl-clad fuel, due to the smaller quantity of chromium involved (possibly in stoichiometric excess of cesium and iodine) (medium/high, relative to medium for conventional fuels and high for FeCrAl). The generic issue carries the same uncertainties and there is a need for more experimental work, hence the state-of-knowledge is also ranked as medium/low.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

A consensus was reached that the ranks for HBU are the same as those for the conventional fuel designs.

HBU/HALEU Fuel

A consensus was reached that the ranks for HBU/HALEU are also the same as those for the conventional fuel designs.

Relocation Phenomena – Fuel, Cladding, Control Rods or Blades, Channel Box Materials

Table 3.27 Importance and State-of-Knowledge Ranks for Relocation Phenomena (Fuel, Cladding, Control Rods or Blades, Channel Box Materials)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	H	M
FeCrAl Cladding				
	H	L	H	L
Cr-coated Zr-alloy Cladding				
	H	M	H	M
Cr-doped UO_2 Fuel				
	H	M	H	M
HBU Fuel				
	H	M/L	H	M/L
HBU/HALEU Fuel				
	H	M/L	H	M/L

Rationales:

Conventional Fuels

Fuel/cladding relocation behavior directly impacts both figures-of-merits, hence it is of high importance (e.g., fission product release and hydrogen generation could be very different depending on whether core degradation results in denuded pellet stacks versus a debris bed versus molten pool, the extent to which core blockages form, etc.). The relocation process for zirconium-alloy convention fuels consists of:

- The rapid oxidation of zirconium alloys creates a refractory shell so that even when unoxidized zirconium melts it does not flow down the rod. It is held in place by the oxide shell. Because of this rigid shell, tertiary creep of zirconium cladding prior to melting is not an issue.
- The molten metal does wet and dissolve the fuel along the grain boundaries. The fuel-molten cladding system involves a monotectic with significant fuel solubility in the molten cladding.
- This attack on the fuel leads liquefaction of the fuel. Eventually the molten Zr-U-O mixture accumulates to the point it can flow down the ZrO_2 'shell'. The melt can accumulate temporarily at dislocations along the fuel rod bundle such

as rod spacers. Eventually accumulates temporarily at the bottom the bundles where it can attack core structures.

- It is known from experiments that through much of this process the pellet stack remains intact. It is held in place by sintering between the pellets or at the high temperatures of severe accidents or residual solidified melt.

There is much uncertainty regarding the prediction of fission product release and oxidation after the core loses its intact geometry, and particularly after molten pool formation, resulting in an overall medium evaluation for state-of-knowledge.

FeCrAl Cladding

The importance of relocation phenomena in the case of FeCrAl-clad fuels remains high. In contrast to conventional fuels, there are three essential differences between zirconium-alloy cladding and FeCrAl cladding, namely:

- FeCrAl melts at a much lower temperature than do any of the zirconium-alloys used for cladding.
- There will not be a thick oxide on FeCrAl when it melts. Tertiary creep of the cladding prior to melting is an issue.
- There is no evidence that at the melting point of FeCrAl or even at higher temperatures there is significant solubility of reactor fuel in the melt. Oxygen is not very soluble in this molten metal – a clear contrast with molten zirconium. On the other hand, both chromium and aluminum are highly electropositive. The chemical activity of aluminum in the alloy should be quite low. It might be possible for chromium to reduce uranium sufficiently to attack along grain boundaries of the fuel.

Therefore, absence of experimental data on the relocation behavior of molten FeCrAl cladding contributes to a low ranking for state-of-knowledge for this phenomenon.

Cr-coated Zirconium-alloy Cladding

Fuel/cladding relocation behavior was believed by most panelists to be similar to that of conventional fuels, even though a view was expressed that the chromium coating might have a marginal effect on delaying relocation. Nevertheless, the importance remains high, and the state-of-knowledge is best characterized as being in the medium regime.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

There is the potential for more fragmented fuel (in the absence of reflood) for HBU and for more likelihood of debris slumping with consequences for coolability inside the lower head. The Phébus fission product (FP) tests [17] (lower than 35 GWd/MTU burnup) showed that the relocation process likely took place as a coherent movement of a solid-liquid mixture, growing by dissolution of surrounding fuels, until forming a large molten pool in the lower part of the test section. For higher burnup, the state-of-knowledge is low regarding the mechanisms of fuel relocation. For HBU, solid slumping could be anticipated. Consensus HBU rankings are the same as those given to the conventional fuel designs except for the reduction of the rank for state-of-knowledge from medium to medium/low. It was stated that HBU fuel will be more fragmented than conventional fuels (without consideration of reflooding); this will affect slumping; the sintering of the pellets may (or may not) be less complete.

HBU/HALEU Fuel

The same rankings as those of HBU are considered applicable to HBU/HALEU.

Aspects of In-Vessel Coolability: Effects and Efficiency of Reflooding (e.g., hydrogen and steam production, debris and melts coolability) – Degraded Core Configurations

Table 3.28 Importance and State-of-Knowledge Ranks for Aspects of In-Vessel Coolability: Effects and Efficiency of Reflooding (e.g., hydrogen and steam production, debris and melts coolability) (Degraded Core Configurations)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	H	M
FeCrAl Cladding				
	H	L	H	L
Cr-coated Zr-alloy Cladding				
	H	M	H	M
Cr-doped UO_2 Fuel				
	H	M	H	M
HBU Fuel				
	H	L	H	L
HBU/HALEU Fuel				
	H	L	H	L

<u>Rationales</u>:

Conventional Fuels

Core damage arrest would directly impact both figures-of-merits, so the importance of this phenomenon is high. The injection of coolant onto a degrading core would prompt a very large increase in flow through the reactor coolant system which could lead to resuspension of the deposited radioactive aerosols. There is major uncertainty regarding the conditions that would result in core coolability once core degradation has begun, even in the case of conventional fuels. Current models for coolability in the severe accident codes use parameters such as debris particle size, and debris bed porosity to predict debris cooling following water injection once the intact core geometry has been lost. In view of these factors, the state-of-knowledge is ranked medium.

FeCrAl Cladding

The state of current understanding of degraded core coolability is lower for this ATF than for conventional fuels. One difference is that whereas zirconium-alloy tends to become embrittled, FeCrAl is not expected to be embrittled though this needs experimental confirmation. FeCrAl can suffer dispersion hardening from intermetallic formation that leads to embrittlement. In addition, differences in molten pool stratification (i.e., molten material constituents with different densities) for FeCrAl could impact coolability. In view of these larger uncertainties, the state-of-knowledge ranking is reduced from medium to low.

Cr-coated Zirconium-alloy Cladding

The presence of the chromium coating would most likely not make a significant difference to coolability of degraded core configurations, and the available data and knowledge for conventional zirconium-alloy-clad fuels should be largely applicable. Therefore, the importance of the issue remains high and the state-of-knowledge medium.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The potential for embrittlement has not been examined for HBU fuel. It is expected that HBU fuels have a higher potential for fragmentation. Especially the behavior of higher Pu content needs to be explored. The panel reached a consensus that rankings for HBU are the same as those for conventional fuel designs except for the reduction of the state-of-knowledge rank from medium to low. Since there will be more Pu for HBU fuels, and not as much information is available under HBU conditions, a low ranking is assigned to the of the state-of-knowledge.

HBU/HALEU Fuel

The same rankings as for HBU are also considered applicable to the HBU/HALEU fuel as those of HBU. It was noted by the panel that there is greater potential for recriticality for HBU/HALEU fuels due to the higher enrichment, which can enhance/speed-up degradation of fuel and fission product release. It may also reduce the potential for core damage arrest. For instance, there is the potential that some part of the core may go critical while the rest of core remains sub-critical, and the part that goes critical may be mostly un-degraded, with result that the re-criticality is immediately responsible for much of the fission product release from fuel. Recriticality may also alter the potential for coolability of a degraded core condition.

Monotectic and Early Melt Formation – Fuel and Cladding, and Channel Box Materials

Table 3.29 Importance and State-of-Knowledge Ranks for Monotectic and Early Melt Formation (Fuel and Cladding, and Channel Box Materials)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M/H	H	M/H
FeCrAl Cladding				
	H	L	H	L
Cr-coated Zr-alloy Cladding				
	H	M/H	H	M/H
Cr-doped UO_2 Fuel				
	H	M/H	H	M/H
HBU Fuel				
	H	M	H	M
HBU/HALEU Fuel				
	H	M	H	M

<u>Rationales</u>:

Conventional Fuels

Even though this topic is closely related to Relocation Phenomena, it was agreed to retain it as a separate item in the PIRT. But because of its close association with that topic, similar rankings should apply (high importance, medium/high state-of-knowledge). This topic has been studied for many years, and the state-of-knowledge is considered adequate even if some residual uncertainties remain.

Note: It was decided by the panel that this phenomenon as it may apply to the control rods/blades to be addressed implicitly as part of the next item on recriticality.

FeCrAl Cladding

Because of close association of this topic with that of Relocation Phenomena, the same rankings should apply (high importance, low state-of-knowledge). Of special importance in the case of FeCrAl-clad fuels is whether the molten cladding drains away, or whether it interacts significantly with the UO_2 in a way similar to zirconium-alloy cladding. This behavior may depend to some extent on the thickness of the molten cladding layer. Similarly, earlier work by Lambertson and Mueller [18] indicate a eutectic between UO_2 and Al_2O_3 at temperature of 1930 °C. On the other hand, Reference [19] reported a contradictory conclusion mentioning that the UO_2-Al_2O_3 system has a eutectic behavior. There is no new data that would enable to reach a definitive conclusion. However, there is some experimental evidence based on the Japanese tests in support of the formation of a eutectic between aluminum oxide and UO_2 [20], implying that oxidized cladding could wet the fuel even if unoxidized FeCrAl may not. This needs to be confirmed by microstructure analyses in the interaction zone. Due to the lower melting temperature of FeCrAl cladding, it is possible that the cladding would melt while a significant portion of the volatile fission product release is still ongoing, whereas with zirconium-alloy cladding most of the release of volatile fission products would likely have completed before melting of cladding has occurred.

Cr-coated Zirconium-alloy Cladding

The addition of a thin chromium coating on the outer surface of the cladding should not significantly affect monotectic formation, especially in view of the fact that the chromium will likely have oxidized by that time. Therefore, the rankings remain the same as those for conventional fuels.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel agreed that HBU rankings are the same as those for the conventional fuels except for the reduction of the state-of-knowledge from medium/high to medium due to the absence of any available data for HBU fuels.

HBU/HALEU Fuel

The panel agreed that the HBU/HALEU rankings are also the same as those for the conventional fuels except for the reduction of the state-of-knowledge from medium/high to medium due to the absence of any available data as noted for HBU fuels.

Recriticality (including high-temperature control rod relocation/reflood) – Fuel, Cladding, Channel Boxes, and Control Rods or Degraded Core Configurations or Melt

Table 3.30 Importance and State-of-Knowledge Ranks for Recriticality (including high-temperature control rod relocation/reflood) (Fuel, Cladding, Channel Boxes, and Control Rods or Degraded Core Configurations or Melt)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	M	M/L	M	M/L
FeCrAl Cladding				
	M	L	M	L
Cr-coated Zr-alloy Cladding				
	M	M/L	M	M/L
Cr-doped UO_2 Fuel				
	M	M/L	M	M/L
HBU Fuel				
	M	M/L	M	M/L
HBU/HALEU Fuel				
	M	M/L	M	M/L

Rationales:

Conventional Fuel

Even though considered unlikely, the recriticality, if it were to occur, is viewed as a slow critical/subcritical process rather than a super prompt-critical event. The immediate effect could increase the release of fission products and hydrogen as a result of the temperature excursion. However, the impact is limited due to strong negative feedback mechanisms and the small likelihood of a sustained condition. Thus, the importance is ranked as medium. There is significant uncertainty related to the potential for recriticality and its impact on fission product release; hence the state-of-knowledge is evaluated as low/medium.

FeCrAl Cladding

Due to the poor current state-of-knowledge concerning FeCrAl degradation (discussed in other topics), a rank of low was agreed by the panel. The potential impacts of FeCrAl-clad fuels in comparison with conventional fuels include greater separation between fuel rod and control rod materials; greater permeability of debris beds affecting the degree of moderation; and the fact that FeCrAl is more parasitic to neutrons (i.e., higher neutron absorption cross section) than zirconium-alloy.

Cr-coated Zirconium-alloy Cladding

Here again because of the small amount of Cr coating, the same rankings and rationales as for conventional fuels are considered applicable.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The same ranks as those for conventional fuels are applicable for HBU.

HBU/HALEU Fuel

The same ranks as those for conventional fuels are applicable for HBU/HALEU fuels. It is noted that there is a greater likelihood of recriticality with HALEU as discussed for the In-vessel Coolability topic earlier. However, given a recritical event, the impact on fission product release remains the same as conventional fuels.

Molten Pool Behavior in the Lower Head, Including Stratification, Element Partitioning, Natural Convection, Overlying Water, Oxidation, and Crust Effects – Late Phase In-Vessel Melt Behavior

Table 3.31 Importance and State-of-Knowledge Ranks for Molten Pool Behavior in the Lower Head, Including Stratification, Element Partitioning, Natural Convection, Overlying Water, Oxidation, and Crust Effects (Late Phase In-Vessel Melt Behavior)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	L	M	L	M
FeCrAl Cladding				
	L	M	L	M
Cr-coated Zr-alloy Cladding				
	L	M	L	M
Cr-doped UO_2 Fuel				
	L	M	L	M[13]
HBU Fuel				
	L	M	L	M
HBU/HALEU Fuel				
	L	M	L	M

<u>Rationales</u>:

Conventional Fuels

The importance of molten pool behavior in the lower head is low because it comes into play at a time past when most of the volatile fission products and hydrogen will have been released, as well as due to the low surface-to-volume ratio of the molten pool. However, the magnitude of heat transfer from the melt to the reactor pressure vessel lower head could influence the timing of the lower head failure, the beginning of ex-vessel release, as well as the timing and conditions affecting revaporization from various reactor coolant system structural surfaces. The understanding of this molten pool behavior, including the separation into oxidic and metallic phases for conventional fuels is limited, and current code models are simple, resulting in medium state-of-knowledge. The molten pool issues are being re-evaluated in light of findings from the damaged

[13] The present PIRTs and phenomena evaluations exclude Cr-doped UO_2 fuel with Cr-coated cladding which may be considered for other ATF concepts.

reactors at the Fukushima installation. Note that the phenomenon of actual lower head thermomechanical failure, which this topic directly impacts, is addressed in other topics.

FeCrAl Cladding

While the phenomenon is understood to some extent for melts originating from conventional fuels, there are few data available on fission product release from melts with metallic contents[14] and the representations of molten pool behavior are based on idealized configurations and assumptions. In addition, if the molten FeCrAl cladding drain from the fuel rods, it could result in a metallic melt cascading into the lower head early in the accident. As a result, there would not have been an opportunity for fission products to partition from fuel into the molten cladding, and therefore, the metal melt would not have a significant internal heat source. Because the melting of cladding is piecemeal throughout the core, a molten pool might not be sustained in the lower head. Without more information on what the FeCrAl cladding does when molten, it is difficult to even 'speculate'. A crucial issue is whether the pellet stacks remain intact when denuded of clad. If not, what comes into the lower head may well be very different than what is envisaged for fuels with conventional clad. Other issues that can impact the molten pool behavior due to the change of cladding material would likely not have a strong effect, and that current models are considered sufficient, though there are other unresolved questions concerning how much the focusing effect (i.e., the potential for the formation of a thin, high conductivity metallic layer on top of the molten pool in the lower head) would be decreased by the higher iron content in the melt. Nevertheless, these factors result in the same medium classification for state-of-knowledge that was arrived at in the case of conventional fuels.

Cr-coated Zirconium-alloy Cladding

Here again because of the small amount of Cr coating, the same importance and state-of-knowledge rankings and rationales as for conventional fuels are considered applicable.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

There is no discernable impact from fuel burnup on the molten pool behavior inside the lower head. Hence, the same rankings as those of conventional fuels are considered applicable.

[14] The great majority of the metal content of the debris is due to structural steel in the core, and the additional amount resulting from FeCrAl cladding may not be as significant as compared with conventional cladding.

HBU/HALEU Fuel

It was agreed that the same rankings as those of conventional fuels are considered applicable for HBU/HALEU fuels.

Low-pressure RPV Thermomechanical Failure or Melting Through, Rupture Location (including potential effects of eutectic and intermetallic interactions with vessel wall) – RPV

Table 3.32 Importance and State-of-Knowledge Ranks for Low-pressure RPV Thermomechanical Failure or Melting Through, Rupture Location (including potential effects of eutectic and intermetallic interactions with vessel wall) (RPV)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	L	M	L	M
FeCrAl Cladding				
	L	L	L	L
Cr-coated Zr-alloy Cladding				
	L	M	L	M
Cr-doped UO_2 Fuel				
	L	M	L	M
HBU Fuel				
	L	M	L	M
HBU/HALEU Fuel				
	L	M	L	M

<u>Rationales</u>:

Conventional Fuels

The impact of the mode and conditions of lower vessel failure on the two FOMs is low, in view of the event occurring after most of the hydrogen and volatile fission product release has completed. There are some data on corium stratification and morphology from the OECD MASCA lower head program applicable to Zr-clad fuels, and there are models for molten pool natural convection and interactions with the lower head that appear to be, for the most part, adequate, even though uncertainties exist. Hence, the state-of-knowledge is ranked as medium. Note that this topic considers non-energetic melt pours only (high pressure melt ejection issue is addressed later).

FeCrAl Cladding

The importance of these phenomena on the figures-of-merits remains low, as for conventional fuels. The state-of-knowledge associated with reactor lower head vessel failure mechanisms has greater uncertainty in the case of FeCrAl-clad fuels, although some panelists believe it would behave similarly to conventional fuels under low-pressure conditions. It is unclear whether the cladding material in this case plays a significant role, other than an impact on the focusing effect due to likely higher metallic content in the melt. The panel agreed to rank the state-of-knowledge lower than for conventional fuels (i.e., low instead of medium).

Cr-coated Zirconium-alloy Cladding

Phenomena involved with lower head failure are not likely to be significantly impacted by a relatively small mass of chromium coating, and therefore the rankings remain identical to those for the conventional fuels.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel agreed that there is no impact from burnup and hence the rankings for HBU are the same as those for the conventional fuels.

HBU/HALEU Fuel

The panel agreed that there is no impact from burnup and enrichment and hence the rankings for HBU/HALEU are also the same as those for the conventional fuels.

Melt and Debris Ejection, High- and Low-pressure Scenarios – Late-phase In-Vessel Melts and Debris in RPV, Hot-Leg, Steam Generator Tubes

Table 3.33 Importance and State-of-Knowledge Ranks for Melt and Debris Ejection, High- and Low-pressure Scenarios (Late-phase In-Vessel Melts and Debris in RPV, Hot-Leg, Steam Generator Tubes)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	H	M	H	M
FeCrAl Cladding				
	H	M	H	M
Cr-coated Zr-alloy Cladding				
	H	M	H	M
Cr-doped UO_2 Fuel				
	H	M	H	M
HBU Fuel				
	H	M	H	M
HBU/HALEU Fuel				
	H	M	H	M

<u>Rationales</u>:

Conventional Fuels

High-pressure melt ejection could result in a significant burst of oxidation due to the high surface-to-volume ratio associated with the core debris during this event, and it may also increase the release of some less volatile fission products (i.e., high importance for source term to the containment). The melt ejection process adds a brief but intense burst of particulates into the containment atmosphere and initiates combustion of accumulated hydrogen during in-vessel core degradation. The high concentration of the dispersed debris drops out of the atmosphere relatively quickly. Current models in severe accident codes are mostly parametric in nature and the state-of-knowledge is ranked as medium.

FeCrAl Cladding

No significant changes are expected with FeCrAl-clad fuels in comparison with conventional Zr-alloy cladding fuels, but there are limited data on this phenomenon for FeCrAl. It is possible that reduced in-vessel oxidation that is expected with FeCrAl cladding may alter the likelihood of high pressure at vessel failure (e.g., due to potential effect on reduction in heat-up of reactor coolant system structures and the resulting likelihood of creep-rupture prior to vessel failure).

Cr-coated Zirconium-alloy Cladding

No discernible impact from the small amount of Cr coating is expected; therefore, the same rankings and rationales as those for conventional fuels are applicable.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel discussed the potential impact on temperature-induced creep rupture (e.g., steam generator tubes for PWRs). The code calculations have shown a large sensitivity to debris particle size and porosity, and any potential for more fragmentation of HBU fuel that may require different values for these parameters and thus to bifurcating predictions on induced rupture behavior. (Such ruptures would be decisive for the pressure boundary conditions that determine the character of the melt ejection process.) After some deliberation, it was agreed that HBU rankings should remain the same as those for the conventional fuels. However, it was noted that there is a need for sensitivity studies to assess the impact of different particle sizes on the reactor coolant system structural heatup and failure location which may be more relevant for HBU fuel conditions.

HBU/HALEU Fuel

The panel agreed that HBU/HALEU rankings are also the same as those for the conventional fuels.

Melt and Debris Interactions with Water in the Lower Head, Melt Fragmentation, Melt and Debris Coolability – Late-phase In-Vessel Melts and Degraded Core Configurations Interacting with Water in the Vessel/Lower Plenum

Table 3.34 Importance and State-of-Knowledge Ranks for Melt and Debris Interactions with Water in the Lower Head, Melt Fragmentation, Melt and Debris Coolability (Late-phase In-Vessel Melts and Degraded Core Configurations Interacting with Water in the Vessel/Lower Plenum)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	M	M	H	M
FeCrAl Cladding				
	M	M	H	M
Cr-coated Zr-alloy Cladding				
	M	M	H	M
Cr-doped UO_2 Fuel				
	M	M	H	M
HBU Fuel				
	M	M	H	M
HBU/HALEU Fuel				
	M	M	H	M

Rationales:

Conventional Fuels

The important effect for FOM-1 of core debris interaction with water in the lower head is the flow of gas through the reactor coolant system. A sudden burst of steam flow will resuspend previously deposited aerosols and enhance mass transport of revaporizing fission products deposited on flow pathways in the reactor coolant system. The impact on fission product release was ranked as medium importance. For FOM2, the sudden burst of steam will enhance the availability of oxidant for hydrogen production in parts of the core that have not yet relocated (i.e., high importance). Existing models in the codes are parametric (e.g., assumed debris particle size, debris porosity, etc.); hence, the state-of-knowledge is ranked medium.

FeCrAl Cladding

The importance with respect to hydrogen generation may be increased with FeCrAl, although it is already of high importance. Energetics of steam production in the lower plenum are probably reduced with FeCrAl in comparison with conventional fuels. Additional evidence and research is required for FeCrAl in this area, and the existing models in the codes are highly parametric hence, the state-of-knowledge is ranked medium.

Cr-coated Zirconium-alloy Cladding

The addition of a small amount of coating to the cladding does not have any impact; therefore, the same rankings and rationales as for conventional fuels are considered applicable.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

It was agreed that HBU rankings are the same as those assigned to the conventional fuels.

HBU/HALEU Fuel

It was agreed that HBU/HALEU rankings are also the same as those assigned to the conventional fuels. It was noted that late-phase debris/water interactions can potentially result in higher likelihood of recriticality for HALEU; however, recriticality was not considered to have any impact on the assigned rankings.

Solid Debris Particle Size and Porosity (In-vessel prior to molten pool formation)* or *(In- and ex-vessel) – Fuel and Cladding

Table 3.35 Importance and State-of-Knowledge Ranks for Solid Debris Particle Size and Porosity (In-vessel prior to molten pool formation) or (In- and ex-vessel) (Fuel and Cladding)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	M/L	M	M/H	M
FeCrAl Cladding				
	M/L	M/L	M/H	M/L
Cr-coated Zr-alloy Cladding				
	M	M	M/H	M
Cr-doped UO_2 Fuel				
	M/L	M	M/H	M
HBU Fuel				
	M/L	M/L	M/H	M/L
HBU/HALEU Fuel				
	M/L	M/L	M/H	M/L

Rationales:

Conventional Fuels

Debris particle size and porosity have the potential to influence both figures-of-merits, but the effects on fission product release should not be very large (low/medium), particularly in consideration of the fact that most fission products would have been released by the time a debris bed can be formed. Based on the results of the MELCOR/MAAP crosswalk [21], and the fact that debris particle size controls the rate of steam production, the importance with respect to hydrogen generation may be larger (medium/high). It is also noted that the debris bed would likely not be in this solidified state for a long period (lower importance), and if it does, then there would be no impact. There is scant technical basis for assessing fission product release from fuel in a degraded configuration. The panel ranked the overall state-of-knowledge as medium.

FeCrAl Cladding

The technical basis for assessing fission product release from degraded fuel is somewhat less for FeCrAl ATF as compared with conventional fuels (i.e., the state-of-knowledge rank is assigned as medium/low instead of medium). Previously acknowledged differences in fuel degradation for FeCrAl-clad fuels as compared with conventional fuels (including changes to how pellet stacks collapse) would imply differences in this category as well. There is the potential for greater debris bed permeability for FeCrAl-clad fuels, as well as differences in particle sizes (impacted by the possibility of fuel pellets relocating whole upon cladding failure) and slumping behavior. Delayed release of fission products for FeCrAl fuels affects the importance ranking as compared with conventional fuels, although the importance with respect to release of fission products to the containment remained ranked as medium/low.

Cr-coated Zirconium-alloy Cladding

The chromium coating may hold up the release of certain fission products such as cesium and iodine, which might call for a different importance ranking with respect to FOM-1. In addition, delayed heatup from oxidation for chromium-clad fuels may contribute to these properties. Nevertheless, the majority of the panel agreed that the importance would remain medium with respect to FOM-1 and medium/high with respect to FOM2. The state-of-knowledge is about the same as for conventional fuel (medium).

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel agreed that the rankings for HBU remain the same as those for the conventional fuels except for the reduction of the state-of-knowledge from medium to medium/low.

HBU/HALEU Fuel

The panel agreed that the rankings for HBU/HALEU are the same as those HBU fuels.

Ex-Vessel Melt and Debris Interactions with Water, Fragmentation – Melt and Debris

Table 3.36 Importance and State-of-Knowledge Ranks for Ex-Vessel Melt and Debris Interactions with Water, Fragmentation (Melt and Debris)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
Pre-existing Water	M	M	H	M
Water Overlaying Melt	H/M	M	M	M
FeCrAl Cladding				
Pre-existing Water	M	M	H	M
Water Overlaying Melt	H/M	M	M	M
Cr-coated Zr-alloy Cladding				
Pre-existing Water	M	M	H	M
Water Overlaying Melt	H/M	M	M	M
Cr-doped UO_2 Fuel				
Pre-existing Water	M	M	H	M
Water Overlaying Melt	H/M	M	M	M
HBU Fuel				
Pre-existing Water	M	M	H	M
Water Overlaying Melt	H/M	M	M	M
HBU/HALEU Fuel				
Pre-existing Water	M	M	H	M
Water Overlaying Melt	H/M	M	M	M

<u>Rationales</u>:

Conventional Fuels

Persistent release of tellurium (a volatile fission product) is observed during tests of core debris interactions with concrete. The tellurium partitions into metallic phase of core debris during in-vessel core degradation and sparged from the metallic phase by gaseous products of concrete degradation during the ex-vessel stage of the accident. When water is poured on core debris interacting with concrete it is nearly always observed that a solidified crust of material forms even when a water pool cannot be sustained over the core debris. This crust attenuates aerosol production substantially. If a water pool can be sustained aerosol production is very substantially reduced. The release of other (than Te) volatile fission products is low during the ex-vessel phase of severe accidents in comparison to the in-vessel phase, reducing the importance of this phenomenon to medium. Hydrogen production due to the interaction of core debris with an overlying water pool pales in comparison to the hydrogen production and carbon monoxide production going on as core debris degrades concrete. The overall state-of-knowledge is ranked as medium.

FeCrAl Cladding

The use of FeCrAl cladding in not expected to make a large difference with regard to this phenomenon; hence, the importance and state-of-knowledge remain ranked the same as those for conventional fuels.

Cr-coated Zirconium-alloy Cladding

The small amount of Cr coating is not expected to affect the importance and state-of-knowledge rankings and rationales as compared to the conventional fuels.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel agreed that the rankings for importance and state-of-knowledge for HBU remain the same as those for the conventional fuels.

HBU/HALEU Fuel

The panel agreed that the rankings for importance and state-of-knowledge for HBU/HALEU also remain the same as those for the conventional fuels.

MCCI Behavior: Stratification, Crust Formation, Concrete Erosion, Short-term Cooling Mechanisms (water ingress, eruption related to MCCI gas release) – Cavity

Table 3.37 Importance and State-of-Knowledge Ranks for MCCI Behavior: Stratification, Crust Formation, Concrete Erosion, Short-term Cooling Mechanisms (water ingress, eruption related to MCCI gas release) (Cavity)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	M	M	M/H	M
FeCrAl Cladding				
	M	M	H	M
Cr-coated Zr-alloy Cladding				
	M	M	M/H	M
Cr-doped UO_2 Fuel				
	M	M	M/H	M
HBU Fuel				
	M	M	M/H	M
HBU/HALEU Fuel				
	M	M	M/H	M

<u>Rationales</u>:

Conventional Fuels

The cladding metal is a minor component of the corium melt as compared with the available structural steel in the early phase of MCCI; however, additional iron does not affect fission product release, even though it has a pronounced influence on hydrogen generation, especially over the long-term phase of MCCI. The MCCI processes have a moderate effect (i.e., medium importance) on fission product release and an important (i.e., medium/high) impact on hydrogen generation in comparison with the in-vessel phenomena. However, issues such as the low evidence of MCCI at Fukushima need to be resolved.

FeCrAl Cladding

Changes in MCCI behavior as a result of FeCrAl cladding in comparison with zirconium-alloy are limited. The increased metallic content of the melt could make a difference to hydrogen generation (i.e., high importance), and in the duration of the oxidation phase of MCCI, but the difference is minimized over the long-term as more rebar is incorporated into the debris. In addition, the FPT-2 experiment [22] showed the evidence for strontium trapped in the iron oxide as a result of interaction between UO_2 and iron oxide, implying that corium containing

FeCrAl cladding could possibly impact strontium chemistry. Silicates have a more significant influence on Ba and Sr releases. CORCON/VANESA are considered generally adequate for modeling the MCCI processes during for severe accidents, and the codes are likely to remain adequate for melts containing FeCrAl-cladding.

Cr-coated Zirconium-alloy Cladding

Most of the metal in the melt by this time comes from structural metal and rebar in the ablated concrete, and the relatively small amount of chromium from the coating does not make any difference. Therefore, the same rankings for importance and state-of-knowledge as for conventional fuels remain applicable.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel concluded that the same rankings for importance and state-of-knowledge as for conventional fuels remain applicable to HBU fuels.

HBU/HALEU Fuel

The panel concluded that the same rankings for importance and state-of-knowledge as for conventional fuels also remain applicable to HBU/HALEU fuels.

Long-term Corium Behavior and Arrested Progression – Cavity

Table 3.38 Importance and State-of-Knowledge Ranks for Long-term Corium Behavior and Arrested Progression (Cavity)

Topic	FOM-1 Source term to containment		FOM-2 Combustible gas production	
	Importance	State-of-Knowledge	Importance	State-of-Knowledge
Conventional Fuels				
	L	L/M	L	L/M
FeCrAl Cladding				
	L	L/M	L	L/M
Cr-coated Zr-alloy Cladding				
	L	L/M	L	L/M
Cr-doped UO_2 Fuel				
	L	L/M	L	L/M
HBU Fuel				
	L	L/M	L	L/M
HBU/HALEU Fuel				
	L	L/M	L	L/M

Rationales:

Conventional Fuels

No significant hydrogen generation is expected during late phase of MCCI, and release of the most important fission products will have long since completed (i.e., low importance). Basemat melt-through is of no significance/importance to the figures-of-merit as currently defined (i.e., source term to the containment). The technical knowledge for the long-term behavior of corium during the ex-vessel phase is limited, and it is mainly extrapolating models into a time frame where they have not been validated (i.e., low/medium rank for state-of-knowledge). In particular, current models do not address the attack on concrete by solidified debris or the sintered crust that forms after basemat melt-through.

FeCrAl Cladding

There are likely no significant differences between melts composed of fuels with FeCrAl and zirconium-alloy cladding that would impact the MCCI simulation models; even though the corium may be higher in metallic content initially, that difference diminishes over the long-term as addition of concrete rebar to the melt becomes the dominant contributor.

Cr-coated Zirconium-alloy Cladding

The panel agreed to the same rankings and rationales as for conventional fuels. This because the relatively small mass of chromium coating is a relatively tiny contributor to the total metal mass incorporated into the corium melt during MCCI.

Cr-doped UO_2 Fuel

The importance and state-of-knowledge are the same as those for the conventional fuels. The panel agreed to assign the same ranks as those for the conventional fuels.

HBU Fuel

The panel concluded that the rankings for HBU remain the same as those for the aforementioned conventional and ATF designs.

HBU/HALEU Fuel

The panel concluded that the rankings for HBU/HALEU also remain the same as those for conventional and ATF designs.

Table 3.39 PIRT Evaluation for Conventional and FeCrAl-Clad Fuels

Phenomenon	Sub-Topic	Conventional Fuels				FeCrAl-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Mass Densities	Fuel	L	H	L	H	L	H	L	H	No.
	Cladding	L	H	L	H	L	H	L	H	Yes.
	CB/CR	L	H	L	H	L	H	L	H	No.
	Melt	L	M	L	M	L	**L**	L	**L**	Yes.
Thermal Conductivities	Fuel	M	H	M	H	M	H	M	H	No.
	Cladding	M	H	M	H	M	**M**	M	**M**	Yes.
	CB/CR	L	H	L	H	L	H	L	H	No.
	Melt	L	M	L	M	L	M	L	M	Yes.
Specific Heats	Fuel	M	H	M	H	M	H	M	H	No.
	Cladding	M	H	M	H	M	H	M	H	Yes.
	CB	L	H	H	H	L	H	H	H	No.
	CR	L	H	M	H	L	H	M	H	No.
	Melt	M	H	M	H	M	H	M	H	Yes.
Melting Points	Fuel	L	H	L	H	L	H	L	H	No.
	Cladding	H	H	H	H	H	**L**	H	**L**	Yes.
	CB	M	H	H	H	M	H	H	H	No.
	CR	H	H	M	H	H	H	M	H	No.
Heats of Fusion	Fuel	L	H	M	H	L	H	M	H	No.

Phenomenon	Sub-Topic	Conventional Fuels				FeCrAl-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
	Cladding	L	H	M	H	L	H	M	H	Yes.
	CB	L	H	M	H	L	H	M	H	No.
	CR	L	H	L	H	L	H	L	H	No.
Coefficients of Thermal Expansion (Volumetric)	Fuel	L	H	L	H	L	H	L	H	No.
	Cladding	L	H	L	H	L	**H**	L	**H**	No.
	CB	L	H	L	H	L	H	L	H	No.
	CR	L	H	L	H	L	H	L	H	No.
Surface Emissivity	Cladding	L	M	M	M	L	M	M	M	No.
	Melt	M	L	M	L	M	L	M	L	No.
Viscosity	In-V Melt	M	M	H	M	M	**L**	H	**L**	Possibly.
	Ex-V Melt	M	M	H	M	M	**L**	H	**L**	Possibly.
Surface Tension	In-V Melt	L	M	L	M	**L/M**	**L/M**	L	**L/M**	Minor.
	Ex-V Melt	M	M	H	M	M	**L**	H	M	Minor.
Phase Equilibria, Eutectic Formation Temperatures, etc.	Molten mixtures	H	M	H	M	H	**L**	H	**L**	Yes.
Heats of Solution or Mixing for Formation of Intermetallic Compounds		L	M	M	M	L	M	M	M	No.
Surface roughness		L	H	L	H	L	H	L	H	Possibly.
Foaming potential	Fuel	M	H	M	H	M	H	M	H	No.
	Cladding	L	H	L	H	**H**	**M**	**H**	**M**	Yes.

Phenomenon	Sub-Topic	Conventional Fuels				FeCrAl-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Fuel Wetability by Molten Cladding		All ranks are assigned the same as surface tension.								Yes.
Oxidation Kinetics (including possible pressure-dependence and influence of high-temperature forms of cladding degradation)	Cladding	H	H	H	H	H	**L**	H	**L**	Yes.
	In-V Melt	M	M	M	M	M	M	M	M	Yes.
	Ex-V Melt	M/H	M	M/H	M	M/H	M	M/H	M	Yes.
Gap Inventories/Pressure and Release at Cladding Failure		L	M	N/A		L	**L**	N/A		Minor.
Fission Product Speciation and Chemistry		H	M	N/A		H	**M/L**	N/A		Yes.
Fission Product Release from Fuel during Core Heatup and Melting		H	H/M	N/A		H	**M**	N/A		Minor.
Cladding Interactions Affecting Speciation and Chemisorption (including tellurium retention)		M	M	N/A		**H**	M	N/A		Yes.
Fission Product Retention and Revaporization in the RCS		H	M	N/A		H	**M/L**	N/A		Minor.
Ex-Vessel Release during MCCI of Semi-Volatile Fission Products during MCCI		H	M	N/A		H	M	N/A		Minor.
Tritium Release and Transport		L	M	N/A		L	M	N/A		Minor.
Fission Product Capture in Water Pools		H	M	N/A		H	M	N/A		No.
Release of Nonradioactive Aerosols		H	M	N/A		H	M	N/A		Yes (in-vessel).
Formation of Hexavalent Chromium (effects on Cesium retention, Iodine		M	M/L	N/A		**H**	M/L	N/A		Yes.

Phenomenon	Sub-Topic	Conventional Fuels				FeCrAl-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
speciation, etc.)										
Relocation Phenomena		H	M	H	M	H	**L**	H	**L**	Yes.
Aspects of in-vessel coolability: effects and efficiency of reflooding, e.g., hydrogen and steam production, debris and melts coolability		H	M	H	M	H	**L**	H	**L**	Yes.
Monotectic and Early Melt Formation		H	M/H	H	M/H	H	**L**	H	**L**	Yes.
Recriticality (including high-temperature control rod relocation/reflood)		M	M/L	M	M/L	M	**L**	M	**L**	Minor.
Molten Pool Behavior in the Lower Head, Including Stratification, Element Partitioning, Natural Convection, Overlying Water, Oxidation, and Crust Effects		L	M	L	M	L	M	L	M	No.
Low-pressure RPV Thermomechanical Failure or Melting Through, rupture location (including potential effects of eutectic and intermetallic interactions with vessel wall)		L	M	L	M	L	**L**	L	**L**	Possible/minor.
Melt and debris ejection, high- and low-pressure scenarios		H	M	H	M	H	M	H	M	No.
Melt and debris interactions with water in the lower head, melt fragmentation, melt and debris coolability		M	M	H	M	M	M	H	M	Yes.

Phenomenon	Sub-Topic	Conventional Fuels				FeCrAl-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Solid Debris Particle Size and Porosity		M/L	M	M/H	M	**M/L**	**M/L**	M/H	**M/L**	Yes.
Ex-Vessel Melt and Debris Interactions with Water, Fragmentation	Pre-existing Water	M	M	H	M	M	M	H	M	No.
	Water Overlying Melt	H/M	M	M	M	H/M	M	M	M	No.
MCCI behavior: stratification, crust formation, concrete erosion, short-term cooling mechanisms (water ingress, eruption related to MCCI gas release)		M	M	M/H	M	M	M	**H**	M	Minor.
Long-term Corium Behavior and Arrested Progression		L	L/M	L	L/M	L	L/M	L	L/M	No.

Table 3.40 PIRT Evaluation for Conventional and Cr-Coated Zr-Alloy-Clad Fuels

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Coated Zr-Alloy-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Mass Densities	Fuel	L	H	L	H	L	H	L	H	No.
	Cladding	L	H	L	H	L	H	L	H	Minor.
	CB/CR	L	H	L	H	L	H	L	H	No.
	Melt	L	M	L	M	L	M	L	M	Minor.
Thermal Conductivities	Fuel	M	H	M	H	M	H	M	H	No.
	Cladding	M	H	M	H	M	**M**	M	**M**	Minor.
	CB/CR	L	H	L	H	L	H	L	H	No.
	Melt	L	M	L	M	L	M	L	M	Minor.
Specific Heats	Fuel	M	H	M	H	M	H	M	H	No.
	Cladding	M	H	M	H	M	H	M	H	Minor.
	CB	L	H	H	H	L	H	H	H	No.
	CR	L	H	M	H	L	H	M	H	No.
	Melt	M	H	M	H	M	H	M	H	Minor.
Melting Points	Fuel	L	H	L	H	L	H	L	H	No.
	Cladding	H	H	H	H	H	**M**	H	**M/L**	Minor.
	CB	M	H	H	H	M	H	H	H	No.
	CR	H	H	M	H	H	H	M	H	No.
Heats of Fusion	Fuel	L	H	M	H	L	H	M	H	No.
	Cladding	L	H	M	H	L	H	M	H	Minor.

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Coated Zr-Alloy-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
	CB	L	H	M	H	L	H	M	H	No.
	CR	L	H	L	H	L	H	L	H	No.
Coefficients of Thermal Expansion (Volumetric)	Fuel	L	H	L	H	L	H	L	H	No.
	Cladding	L	H	L	H	L	H	L	H	Minor.
	CB	L	H	L	H	L	H	L	H	No.
	CR	L	H	L	H	L	H	L	H	No.
Surface Emissivity	Cladding	L	M	M	M	L	M	M	M	No.
	Melt	M	L	M	L	M	L	M	L	No.
Viscosity	In-V Melt	M	M	H	M	M	M	H	M	Yes.
	Ex-V Melt	M	M	H	M	M	M	H	M	Yes.
Surface Tension	In-V Melt	L	M	L	M	L	M	L	M	Yes.
	Ex-V Melt	M	M	H	M	M	M	H	M	Yes.
Phase Equilibria, Eutectic Formation Temperatures, etc.	Molten mixtures	H	M	H	M	H	M	H	M	Minor.
Heats of Solution or Mixing for Formation of Intermetallic compounds		L	M	M	M	L	M	M	M	No.
Surface roughness		L	H	L	H	L	H	L	H	No.
Foaming potential	Fuel	M	H	M	H	M	H	M	H	No.
	Cladding	L	H	L	H	L	H	L	H	Minor.
Fuel Wetability by Molten Cladding		All ranks are assigned the same as surface tension.								Yes.

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Coated Zr-Alloy-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Oxidation Kinetics (including possible pressure-dependence and influence of high-temperature forms of cladding degradation)	Cladding	H	H	H	H	H	**M**	H	**M**	Yes.
	In-V Melt	M	M	M	M	M	M	M	M	Yes.
	Ex-V Melt	M/H	M	M/H	M	M/H	M	M/H	M	Yes.
Gap Inventories/Pressure and Release at Cladding Failure		L	M	N/A		L	M	N/A		No.
Fission Product Speciation and Chemistry		H	M	N/A		H	M	N/A		Minor.
Fission Product Release from Fuel during Core Heatup and Melting		H	H/M	N/A		H	H/M	N/A		No.
Cladding Interactions Affecting Speciation and Chemisorption (including tellurium retention)		M	M	N/A		M	M	N/A		No.
Fission Product Retention and Revaporization in the RCS		H	M	N/A		H	M	N/A		Yes.
Ex-Vessel Release during MCCI of Semi-Volatile Fission Products during MCCI		H	M	N/A		H	M	N/A		Minor.
Tritium Release and Transport		L	M	N/A		L	M	N/A		No.
Fission Product Capture in Water Pools		H	M	N/A		H	M	N/A		No.
Release of Nonradioactive Aerosols		H	M	N/A		H	M	N/A		Minor.
Formation of Hexavalent Chromium (effects on Cesium retention, Iodine speciation, etc.)		M	M/L	N/A		**H/M**	**M/L**	N/A		Yes.

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Coated Zr-Alloy-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Relocation Phenomena		H	M	H	M	H	M	H	M	Minor.
Aspects of in-vessel coolability: effects and efficiency of reflooding, e.g., hydrogen and steam production, debris and melts coolability		H	M	H	M	H	M	H	M	No.
Monotectic and Early Melt Formation		H	M/H	H	M/H	H	M/H	H	M/H	Minor.
Recriticality (including high-temperature control rod relocation/reflood)		M	M/L	M	M/L	M	M/L	M	M/L	No.
Molten Pool Behavior in the Lower Head, Including Stratification, Element Partitioning, Natural Convection, Overlying Water, Oxidation, and Crust Effects		L	M	L	M	L	M	L	M	No.
Low-pressure RPV Thermomechanical Failure or Melting Through, rupture location (including potential effects of eutectic and intermetallic interactions with vessel wall)		L	M	L	M	L	M	L	M	No.
Melt and debris ejection, high- and low-pressure scenarios		H	M	H	M	H	M	H	M	No.
Melt and debris interactions with water in the lower head, melt fragmentation, melt and debris coolability		M	M	H	M	M	M	H	M	No.
Solid Debris Particle Size and Porosity		M/L	M	M/H	M	**M**	M	M/H	M	Yes.
Ex-Vessel Melt and Debris Interactions with Water, Fragmentation	Pre-existing Water	M	M	H	M	M	M	H	M	No.
	Water Overlying Melt	H/M	M	M	M	H/M	M	M	M	No.

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Coated Zr-Alloy-Clad Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
MCCI behavior: stratification, crust formation, concrete erosion, short-term cooling mechanisms (water ingress, eruption related to MCCI gas release)		M	M	M/H	M	M	M	M/H	M	No.
Long-term Corium Behavior and Arrested Progression		L	L/M	L	L/M	L	L/M	L	L/M	No.

Table 3.41 PIRT Evaluation for Conventional and Cr-Doped UO_2 Fuels

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Doped UO_2 Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Mass Densities	Fuel	L	H	L	H	L	H	L	H	Negligible.
	Cladding	L	H	L	H	L	H	L	H	No.
	CB/CR	L	H	L	H	L	H	L	H	No.
	Melt	L	M	L	M	L	M	L	M	Negligible.
Thermal Conductivities	Fuel	M	H	M	H	M	H	M	H	Negligible.
	Cladding	M	H	M	H	M	H	M	H	No.
	CB/CR	L	H	L	H	L	H	L	H	No.
	Melt	L	M	L	M	L	M	L	M	Negligible.
Specific Heats	Fuel	M	H	M	H	M	H	M	H	Negligible.
	Cladding	M	H	M	H	M	H	M	H	No.
	CB	L	H	H	H	L	H	H	H	No.
	CR	L	H	M	H	L	H	M	H	No.
	Melt	M	H	M	H	M	H	M	H	Negligible.
Melting Points	Fuel	L	H	L	H	L	H	L	H	Negligible.
	Cladding	H	H	H	H	H	H	H	H	No.
	CB	M	H	H	H	M	H	H	H	No.
	CR	H	H	M	H	H	H	M	H	No.
Heats of Fusion	Fuel	L	H	M	H	L	H	M	H	Negligible.

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Doped UO_2 Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
	Cladding	L	H	M	H	L	H	M	H	No.
	CB	L	H	M	H	L	H	M	H	No.
	CR	L	H	L	H	L	H	L	H	No.
Coefficients of Thermal Expansion (Volumetric)	Fuel	L	H	L	H	L	H	L	H	Negligible.
	Cladding	L	H	L	H	L	H	L	H	No.
	CB	L	H	L	H	L	H	L	H	No.
	CR	L	H	L	H	L	H	L	H	No.
Surface Emissivity	Cladding	L	M	M	M	L	M	M	M	No.
	Melt	M	L	M	L	M	L	M	L	No.
Viscosity	In-V Melt	M	M	H	M	M	M	H	M	No.
	Ex-V Melt	M	M	H	M	M	M	H	M	No.
Surface Tension	In-V Melt	L	M	L	M	L	M	L	M	No.
	Ex-V Melt	M	M	H	M	M	M	H	M	No.
Phase Equilibria, Eutectic Formation Temperatures, etc.	Molten mixtures	H	M	H	M	H	M	H	M	Negligible.
Heats of Solution or Mixing for Formation of Intermetallic Compounds		L	M	M	M	L	M	M	M	No.

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Doped UO_2 Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Surface roughness		L	H	L	H	L	H	L	H	No.
Foaming potential	Fuel	M	H	M	H	M	H	M	H	Negligible.
	Cladding	L	H	L	H	L	H	L	H	No.
Fuel Wetability by Molten Cladding		All ranks are assigned the same as surface tension.								No.
Oxidation Kinetics (including possible pressure-dependence and influence of high-temperature forms of cladding degradation)	Cladding	H	H	H	H	H	M	H	M	No.
	In-V Melt	M	M	M	M	M	M	M	M	No.
	Ex-V Melt	M/H	M	M/H	M	M/H	M	M/H	M	No.
Gap Inventories/Pressure and Release at Cladding Failure		L	M	N/A		L	M	N/A		No.
Fission Product Speciation and Chemistry		H	M	N/A		H	**M/L**	N/A		Possible.
Fission Product Release from Fuel during Core Heatup and Melting		H	H/M	N/A		H	H/M	N/A		No.
Cladding Interactions Affecting Speciation and Chemisorption (including tellurium retention)		M	M	N/A		M	M	N/A		No.
Fission Product Retention and Revaporization in the RCS		H	M	N/A		H	M	N/A		No.
Ex-Vessel Release during MCCI of Semi-Volatile Fission Products during MCCI		H	M	N/A		H	M	N/A		No.
Tritium Release and Transport		L	M	N/A		L	M	N/A		No.

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Doped UO_2 Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Fission Product Capture in Water Pools		H	M	N/A		H	M	N/A		No.
Release of Nonradioactive Aerosols		H	M	N/A		H	M	N/A		Negligible.
Formation of Hexavalent Chromium (effects on Cesium retention, Iodine speciation, etc.)		M	M/L	N/A		M	M/L	N/A		Negligible.
Relocation Phenomena		H	M	H	M	H	M	H	M	No.
Aspects of in-vessel coolability: effects and efficiency of reflooding, e.g., hydrogen and steam production, debris and melts coolability		H	M	H	M	H	M	H	M	No.
Monotectic and Early Melt Formation		H	M/H	H	M/H	H	M/H	H	M/H	Negligible.
Recriticality (including high-temperature control rod relocation/reflood)		M	M/L	M	M/L	M	M/L	M	M/L	No.
Molten Pool Behavior in the Lower Head, Including Stratification, Element Partitioning, Natural Convection, Overlying Water, Oxidation, and Crust Effects		L	M	L	M	L	M	L	M	No.
Low-pressure RPV Thermomechanical Failure or Melting Through, rupture location (including potential effects of eutectic and intermetallic interactions with vessel wall)		L	M	L	M	L	M	L	M	No.
Melt and debris ejection, high- and low-pressure scenarios		H	M	H	M	H	M	H	M	No.

Phenomenon	Sub-Topic	Conventional Fuels				Cr-Doped UO_2 Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Melt and debris interactions with water in the lower head, melt fragmentation, melt and debris coolability		M	M	H	M	M	M	H	M	No.
Solid Debris Particle Size and Porosity		M/L	M	M/H	M	M/L	M	M/H	M	No.
Ex-Vessel Melt and Debris Interactions with Water, Fragmentation	Pre-existing Water	M	M	H	M	M	M	H	M	No.
	Water Overlying Melt	H/M	M	M	M	H/M	M	M	M	No.
MCCI behavior: stratification, crust formation, concrete erosion, short-term cooling mechanisms (water ingress, eruption related to MCCI gas release)		M	M	M/H	M	M	M	M/H	M	No.
Long-term Corium Behavior and Arrested Progression		L	L/M	L	L/M	L	L/M	L	L/M	No.

Table 3.42 PIRT Evaluation for Conventional, HBU and HBU/HALEU Fuels

Phenomenon	Sub-Topic	Conventional Fuels				HBU and HBU/HALEU Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Mass Densities	Fuel	L	H	L	H	L	H	L	H	No.
	Cladding	L	H	L	H	L	H	L	H	No.
	CB/CR	L	H	L	H	L	H	L	H	No.
	Melt	L	M	L	M	L	M	L	M	No.
Thermal Conductivities	Fuel	M	H	M	H	M	**M**	M	**M**	Yes.
	Cladding	M	H	M	H	M	**M**	M	**M**	Yes.
	CB/CR	L	H	L	H	L	H	L	H	No.
	Melt	L	M	L	M	L	M	L	M	No.
Specific Heats	Fuel	M	H	M	H	M	**M**	M	H	Yes.
	Cladding	M	H	M	H	M	H	M	H	No.
	CB	L	H	H	H	L	H	H	H	No.
	CR	L	H	M	H	L	H	M	H	No.
	In-V Melt	M	H	M	H	M	H	M	H	No.
	Ex-V Melt	M	H	M	H	M	H	M	H	No.
Melting Points	Fuel	L	H	L	H	L	**M**	L	**M**	Yes.
	Cladding	H	H	H	H	H	H	H	H	No.
	CB	M	H	H	H	M	H	H	H	No.
	CR	H	H	M	H	H	H	M	H	No.

Phenomenon	Sub-Topic	Conventional Fuels FOM-1 (FP) Imp	Conventional Fuels FOM-1 (FP) SOK	Conventional Fuels FOM-2 (H_2) Imp	Conventional Fuels FOM-2 (H_2) SOK	HBU and HBU/HALEU Fuels (Differences with conventional fuels noted) FOM-1 (FP) Imp	HBU and HBU/HALEU Fuels FOM-1 (FP) SOK	HBU and HBU/HALEU Fuels FOM-2 (H_2) Imp	HBU and HBU/HALEU Fuels FOM-2 (H_2) SOK	Difference from Conventional Fuels?
Heats of Fusion	Fuel	L	H	M	H	L	H	M	H	No.
	Cladding	L	H	M	H	L	H	M	H	No.
	CB	L	H	M	H	L	H	M	H	No.
	CR	L	H	L	H	L	H	L	H	No.
Coefficients of Thermal Expansion (Volumetric)	Fuel	L	H	L	H	L	H	L	H	No.
	Cladding	L	H	L	H	L	H	L	H	No.
	CB	L	H	L	H	L	H	L	H	No.
	CR	L	H	L	H	L	H	L	H	No.
Surface Emissivity	Cladding	L	M	M	M	L	M	M	M	No.
	Melt	M	L	M	L	M	L	M	L	No.
Viscosity	In-V Melt	M	M	H	M	M	M	H	M	Yes.
	Ex-V Melt	M	M	H	M	M	M	H	M	No.
Surface Tension	In-V Melt	L	M	L	M	L	M	L	M	No.
	Ex-V Melt	M	M	H	M	M	M	H	M	No.
Phase Equilibria, Eutectic Formation Temperatures, etc.		H	M	H	M	H	M	H	M	No.
Heats of Solution or Mixing for Formation of Intermetallic Compounds	In-V formation In-V solution Ex-V formation	L	M	M	M	L	M	M	M	No.

Phenomenon	Sub-Topic	Conventional Fuels				HBU and HBU/HALEU Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
	Ex-V solution									
Surface roughness		L	H	L	H	L	H	L	H	No.
Foaming potential	Fuel	M	H	M	H	M	**M**	M	**M**	Yes.
	Cladding	L	H	L	H	**M/H**	**L**	**M**	**L**	Yes.
Fuel Wetability by Molten Cladding		All ranks are assigned the same as surface tension.								No.
Oxidation Kinetics (including possible pressure-dependence and influence of high-temperature forms of cladding degradation)		H	H	H	H	H	**M**	H	**M**	Yes.
Oxidation Kinetics (including influence of cladding failure mode) - late In-Vessel and Ex-Vessel melts	In-V Melt	M	M	M	M	M	M	M	M	No.
	Ex-V Melt	M/H	M	M/H	M	M/H	M	M/H	M	No.
Gap Inventories/Pressure and Release at Cladding Failure		L	M	N/A		**L/M**	M	N/A		Yes.
						The potential for fuel fragmentation for HBU under LOCA conditions that can result in some increase in noble gases, etc. (due to particulate release to the RCS) is insignificant in terms of its impact on severe accident source term.				
Fission Product Speciation and Chemistry		H	M	N/A		H	**L/M**	N/A		Yes.

Phenomenon	Sub-Topic	Conventional Fuels				HBU and HBU/HALEU Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Fission Product Release from Fuel during Core Heatup and Melting		H	H/M	N/A		H	**M/L**	N/A		Yes.
						H	H/M	N/A		No.
						Here the lower row applies to HBU/HALEU fuels.				
Cladding Interactions Affecting Speciation and Chemisorption (including tellurium retention)		M	M	N/A		M	M	N/A		No.
Fission Product Retention and Revaporization in the RCS		H	M	N/A		H	**M/L**	N/A		Yes.
Ex-Vessel Release during MCCI of Semi-Volatile Fission Products during MCCI		H	M	N/A		H	M	N/A		No.
Tritium Release and Transport		L	M	N/A		L	M	N/A		No.
Fission Product Capture in Water Pools		H	M	N/A		H	M	N/A		No.
Release of Nonradioactive Aerosols		H	M	N/A		H	M	N/A		No.
Formation of Hexavalent Chromium (effects on Cesium retention, Iodine speciation, etc.)		M	M/L	N/A		M	M/L	N/A		No.
Relocation Phenomena		H	M	H	M	H	**M/L**	H	**M/L**	Yes.

Phenomenon	Sub-Topic	Conventional Fuels				HBU and HBU/HALEU Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Aspects of in-vessel coolability: effects and efficiency of reflooding, e.g., hydrogen and steam production, debris, and melts coolability		H	M	H	M	H	**L**	H	**L**	Yes.
						For HBU/HALEU there is greater potential for recriticality, and the potential for core damage arrest may be reduced.				
Monotectic and Early Melt Formation		H	M/H	H	M/H	H	**M**	H	**M**	Yes.
Recriticality (including high-temperature control rod relocation/reflood)		M	M/L	M	M/L	M	M/L	M	M/L	No..
						For HBU/HALEU, see previous issue related to reflood above.				
Molten Pool Behavior in the Lower Head, Including Stratification, Element Partitioning, Natural Convection, Overlying Water, Oxidation, and Crust Effects		L	M	L	M	L	M	L	M	No.
Low-pressure RPV Thermomechanical Failure or Melting Through, rupture location (including potential effects of eutectic and intermetallic interactions with vessel wall)		L	M	L	M	L	M	L	M	No.
Melt and debris ejection, high- and low-pressure scenarios		H	M	H	M	H	M	H	M	No.

Phenomenon	Sub-Topic	Conventional Fuels				HBU and HBU/HALEU Fuels (Differences with conventional fuels noted)				Difference from Conventional Fuels?
		FOM-1 (FP)		FOM-2 (H_2)		FOM-1 (FP)		FOM-2 (H_2)		
		Imp	SOK	Imp	SOK	Imp	SOK	Imp	SOK	
Melt and debris interactions with water in the lower head, melt fragmentation, melt and debris coolability		M	M	H	M	M	M	H	M	No.
						The potential for any recriticality is not significant.				
Solid Debris Particle Size and Porosity		M/L	M	M/H	M	M/L	**M/L**	M/H	**M/L**	Yes.
Ex-Vessel Melt and Debris Interactions with Water, Fragmentation	Pre-existing Water	M	M	H	M	M	M	H	M	No.
	Water Overlying Melt	H/M	M	M	M	H/M	M	M	M	No.
MCCI behavior: stratification, crust formation, concrete erosion, short-term cooling mechanisms (water ingress, eruption related to MCCI gas release)		M	M	M/H	M	M	M	M/H	M	No.
Long-term Corium Behavior and Arrested Progression		L	L/M	L	L/M	L	L/M	L	L/M	No.

4. SUMMARY AND CONCLUSIONS

This report documented the development of PIRTs for near-term ATFs under severe accident conditions in LWRs. The PIRT process followed a structured approach as discussed in Section 2 of this report.

The starting point for the PIRT process was the panel deliberation on the initial "strawman" list of phenomena applicable to severe accidents and radiological releases and transport documented in Reference [1]. This process resulted in the adaptation of a revised list that was subsequently used to develop importance and state-of-knowledge rankings by the panel.

The identified phenomena were evaluated by the panel for various near-term ATF design concepts (i.e., FeCrAl cladding, zirconium-alloy cladding coated with chromium, and Cr_2O_3 dopants in uranium dioxide fuels), including the impacts from fuel enrichment and burnup. Importance and state-of-knowledge ranks were also developed for conventional fuels, as a reference and for comparison to the near-term ATFs. In addition, the panel considered and provided comments on the severe accident implications of the longer-term ATF concepts; however, the development of PIRTs for longer-term ATFs was outside the scope of the present effort.

The main figures-of-merit considered in this ranking process were the amount of fission products released into the containment and the quantity of combustible gases generated during an accident. The panel also considered whether existing severe accident codes and models would be sufficient for application to LWRs employing these near-term ATFs, and whether additional experimental studies or model development would be warranted.

Particular points of note in the case of each fuel design are summarized below.

<u>FeCrAl-Clad UO_2 Fuel:</u>

Of the near-term ATFs considered in this study, FeCrAl-clad fuel exhibits the most differences under severe accident conditions compared to conventional fuels, because the cladding material is entirely replaced with an advanced stainless steel. The most salient differences noted by the panel are:

- There is limited knowledge of how FeCrAl will degrade at high temperatures, especially when it is oxidized. This affects the entire severe accident progression and, in turn, a number of downstream issues. Aspects include what eutectic reactions among materials occur, whether the cladding metal (or its oxides) wet the fuel versus draining down and leaving bare pellets, and whether foaming of the cladding may occur.
- Oxidation kinetics and behavior are different for FeCrAl. The expectation is a lower initial oxidation rate (prior to the melting of the cladding), along with the potential for significant oxidation after the cladding melts. This process impacts the accident progression due to a lower hydrogen-to-steam ratio and initial oxidation heat input. However, it also leaves potentially more oxidation to occur in later phases of the accident (e.g., at the start of the molten core concrete interaction (MCCI) in the ex-vessel phase of accidents).
- There is a significant amount of aluminum in a core with FeCrAl cladding. However, there is not a significant increase in the mass of chromium in the core compared with

conventional fuel and cores (i.e., for conventional fuel the in-core stainless steel is 18 wt% chromium). There is a possibility that the chromium and aluminum could affect speciation and transport, but these aspects have not been established.

- Under conditions of high steam partial pressures, it is possible that vapor-phase hydroxides and oxyhydroxides of chromium and aluminum could form. These possibilities have not been established. Vapor-phase hydroxides and oxyhydroxides of chromium and aluminum may be important in the consideration of fission product behavior when the degradation takes place with high steam partial pressures. These vapor species will not be abundant in a depressurized accident scenario. However, the potential for forming hydroxides and oxyhydroxides of chromium and aluminum can result in increased amounts of nonradioactive aerosols released into the containment, which in turn impact aerosol agglomeration, deposition, and potentially the clogging of any filters.
- Tellurium is not expected to be sequestered/delayed in release by FeCrAl cladding as has been considered possible for zirconium-alloy cladding. In fact, uncertainties remain related to how tellurium is sequestered or delayed by zirconium. Furthermore, there is no experimental evidence that FeCrAl cladding would make tellurium release more or less extensive than for zirconium-alloy cladding.

Cr-Coated Zirconium-Alloy-Clad UO_2 Fuel:

The chromium-coated zirconium-alloy-clad fuels do not have as many differences with conventional fuels as were observed for FeCrAl. The importance of the differences is substantially diminished because the bulk cladding remains conventional zirconium-alloy and the chromium is confined to a thin coating on the outside surface. Moreover, chromium is already present in conventional core structural materials, so the added mass of chromium constitutes a small difference in the degree of effect following fuel degradation, rather than a significant qualitative difference. The main points noted by the panel for chromium-coated, zirconium-alloy-clad fuels are:

- Some thermophysical properties of the fuel are slightly less well known than those for conventional fuel designs. This holds true for thermal conductivity in particular, because the thermal resistance at the coating interface is uncertain and probably process dependent.
- Initial oxidation is expected to be lower (prior to exceeding the Cr-Zr eutectic temperature), resulting from the presence of the chromium coating, as intended. This coating not only affects the oxidation behavior of this ATF, it also indirectly affects other phenomena such as fuel degradation/relocation behavior via how it potentially shifts the time frame during which rapid oxidation heat input is important. Note that cladding burst during a LOCA will allow oxidation of the interior surface of the cladding.

Cr-Doped UO_2 Fuel:

The small amount of dopant added to the fuel primarily affects the grain size of the UO_2 and thus reduces the release of volatile radionuclides to the gap during normal operation and the release of volatile fission products (e.g., iodine, cesium, etc.) during the early stages of a severe accident. In addition, it was concluded that there is no significant effect on hydrogen generation, and that there is virtually no effect on the later stages of a core meltdown accident. The panel therefore concluded that Cr-doped UO_2 fuel would not behave very differently from conventional fuels, with respect to the vast majority of severe accident issues. The only notable difference was fewer available experimental data (i.e., state-of-knowledge) regarding fission product speciation and chemistry. The limited information available does suggest that there might be a significant effect on oxygen potential in the fuel, from even small quantities of chromia added to the fuel.

High Burnup Fuel:

A number of differences were noted that would affect severe accident behavior in comparison with lower burnup fuels. Among the points noted by the panel are:

- Some thermophysical properties of both fuel and cladding, particularly thermal conductivity and specific heat, may be affected by the higher burnup and irradiation.
- Fission product chemistry may be affected by the presence of more lower valence atoms in the fuel matrix as a result of higher burnup (i.e., greater degree of transmutation by fission of the uranium atoms). An important aspect of this is the larger magnitude of release of ruthenium that has been observed in some tests performed in a pure steam atmosphere (i.e., without the presence of air), although this observation is not fully understood at present.
- Oxygen potential is expected to increase as a result of the larger concentration of lower valence atoms in the fuel, possibly affecting the rate of oxidation at the cladding inner surface. However, the oxygen potential in the fuel is buffered by the Mo/MoO_2 couple. Molybdenum inventories in high burnup fuels may be sufficient to prevent significant excursions of oxygen potential as burnup progresses.
- Gap inventories of fission products would be larger in an absolute sense, even if it is unclear whether they would be larger when expressed as a fraction of the total inventory.
- The fuel is expected to have a different amount of fragmentation or sintering at a higher burnup, which can in turn affect how the fuel behaves during core degradation and relocation, as well as phenomena such as temperature-induced creep rupture that can be affected by core debris particle size.
- The rate of release of volatile fission products has also been observed to be greater for high burnup fuels [20]. Note that the timing of release of volatile fission products is of importance to the retention and transport inside the reactor coolant system. Nonetheless, it is not clear that this effect would significantly impact accident consequences since the volatile fission products are largely released from the fuel in early stages of a severe accident.
- Increased cladding embrittlement might be possible at a higher burnup, thus affecting the likelihood of core coolability upon reflood. However, this has not been closely examined to date.

High Burnup/High Assay Fuel:

The panel did not note significant qualitative differences in severe accident behavior applicable to HBU/HALEU fuels when compared with just HBU fuels, other than the fact that there may be a greater likelihood of recriticality in some accidents (e.g., accidents involving reflood following core damage using un-borated water).

Longer-Term ATF Concepts:

PIRTs were not developed and used to assess the severe accident implications of the longer-term ATF concepts. However, the panel was requested to provide high-level opinions on the severe accident implications of these ATF concepts. These comments are documented in Appendix A of this report.

5. REFERENCES

1. M. Khatib-Rahbar, A. Krall, Z. Yuan, and M. Zavisca, “Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases,” Energy Research, Inc., ERI/NRC 20-209 (October 2020).
2. M. Khatib-Rahbar, A. Krall, Z. Yuan, and M. Zavisca, “Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases,” Energy Research, Inc., ERI/NRC 21-203, NUREG/CR- 7282 (April 2021).
3. C. Ronchi, M. Scheindlin, D. Staicu, and M. Kinoshita, “Effect of Burn-up on the Thermal Conductivity of Uranium Dioxide up to 100 GWd/MTU,” Journal of Nuclear Materials, 327, 58 (2004).
4. D. Staicu, V. V. Rondinella, C. T. Walker, D. Papaioannou, and M. Kinoshita, “Effect of Burn-up on the Thermal Conductivity of Uranium–Gadolinium Dioxide up to 100 GWd/tHM,” Journal of Nuclear Materials, 453, 259 (2014).
5. F.D. Richardson, “*Physical Chemistry of Melts in Metallurgy*”, Volume 2, page 427, Academic Press Inc. (1974).
6. C. E. Sims, Editor “Electric Furnace Steelmaking”, Volume II: Theory and Fundamentals, page 203 (2002).
7. “State-of-the-Art Reactor Consequence Analysis (SOARCA) Project, Sequoyah Integrated Deterministic and Uncertainty Analyses,” U.S. Nuclear Regulatory Commission, NUREG/CR-7245 (October 2019).
8. Y. Pontillon and G. Ducros, “Behaviour of Fission Products under Severe PWR Accident Conditions, The VERCORS Experimental Programme Part 2: Release and Transport of Fission Gases and Volatile Fission Products,” Nuclear Engineering and Design, 240,1853 (2010).
9. T. Kärkelä, C. Mun, G. Ducros, N. Vér, M-N. Ohnet, I. Kajan, S. Souvi, ““Recent Findings on Ruthenium Chemistry in a Severe Accident,” International Iodine Workshop Summary Report, NEA/CSNI/R(2016)5, pp. 132-146 (2016).
10. J. McFarlane, and J. C. Leblanc, “Fission-Product Tellurium and Cesium Telluride Chemistry Revisited, AECL-11333, COG-95-276-I (1996).
11. “State-of-the-Art Reactor Consequence Analyses Project, Volume 1: Peach Bottom Integrated Analysis,” U.S. Nuclear Regulatory Commission, NUREG/CR-7110, Vol. 1, Rev. 1 (May 2013).
12. “State-of-the-Art Reactor Consequence Analyses Project, Volume 2: Surry Integrated Analysis,” U.S. Nuclear Regulatory Commission, NUREG/CR-7110, Vol. 2, Rev. 1 (August 2013).
13. “State-of-the-Art Reactor Consequence Analyses Project, Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station,” U.S. Nuclear Regulatory Commission, NUREG/CR-7155 (May 2016).
14. M. N. Ohnet, O. Leroy, A.S. Mamede, “Ruthenium behavior in the reactor cooling system in case of PWR severe accident,” Journal of Radioanalytical and Nuclear Chemistry, 316, 161 (2018).

15. T. Albiol, et al., "Main Results of the European PASSAM project on severe accident source term mitigation," Annals of Nuclear Energy, 116, pp. 42-56 (2018).

16. T. Albiol, L. E. Herranz, E. Riera, C. Dalibart, T. Lind, A. Del Corno, T. Kärakela, N. Losch, B. Azambre, "Main Results of the European PASSAM Project on Severe Accident Source Term Mitigation. Proceedings of the European Review Meeting on Severe Accident Research, Warsaw, Poland (May 16-18, 2017).

17. Sanjeev Gupta, Luis Herranz, J.-P. Van Dorssaelere, "Integration of Pool Scrubbing Research to enhance Source Term Calculations (IPRESCA)," Presentation at the 8th ERMSAR Conference (2017).

18. W. A. Lambertson and M.H. Mueller, "Uranium Oxide Phase Equilibrium Systems: I, UO_2-Al_2O_3," J. American Ceramic Society, Volume 36, 329-331 (1953).

19. P. P. Budnikov, S. G. Tresvyatskii, and V. I. Kushakovsky, "Binary Phase Diagrams for UO_2-Al_2O_3, UO_2-BeO and UO_2-MgO," Proceeding of the Second Conference on Peaceful Uses of Atomic Energy, Genève (1) P/2193 USSR 124-131 (1958).

20. K. Sakamoto, A. Ouchi, and S. Yamashita. Preliminary Study on Behavior of FeCrAl-ODS/UO2 Fuels under Severe Accident Conditions. Fukushima Research Conference, 10-12 July, Fukushima, Japan (2019).

21. "Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk: Phase 1 Study," EPRI 3002004449 (November 2014).

22. M. Barrachin, Fuel and Fission Product Behaviour in Early Phases of a Severe Accident. Part I: Experimental Results of the PHÉBUS FPT2 test, Journal of Nuclear Materials Volume 453, 340 (2014).

APPENDIX A: LONGER-TERM ATF CONCEPTS

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Comments by Marc Barrachin

SiC cladding behaviour in severe accidents

This document is largely inspired by the work internally performed at IRSN in 2012 [0]. Silicon carbide SiC is often considered as an alternative material to metals or metal alloys for fuel rod cladding. The main objective is to reduce the oxidation of the material by steam and the associated hydriding. This mainly concerns nominal operation (primary hydriding) and design basis accidents, in particular of the LOCA type (secondary hydriding). In the latter case, the objective would be to make the cladding less fragile and to gain margins with respect to safety criteria. Westinghouse studied a solution of this type [1]. The mechanical strength of a solid SiC ceramic part being problematic, the solution studied is that of a part comprising, in addition to the solid SiC, a layer of SiC/SiC composite based on fibers in order to ensure the mechanical strength. We only examine here the issues related to severe accidents, mainly from bibliography and thermodynamic calculations of SiC oxidation. As is often the case for carbides, SiC is not stable at high temperatures. Discrepancies in eutectic in the rich-Si part and peritectic temperatures of SiC decomposition as well as in C solubility in Si liquid could not be explained at that time. Gröbner et al. [2] pointed out that the best agreement between experimental data imposes a peritectic temperature of about 3100 K with C mole fraction in the liquid phase equal to ≈18%, and an eutectic temperature of 1686 K with a very low C solubility equal to 0.02 at%.

Since 1951, it has been observed that water vapor accelerates the oxidation of silicon carbide compared to dry air or oxygen. It is explained either by the volatilization of silica, or by the partial transformation of glassy, amorphous, protective silica into porous, non-protective cristobalite (crystallization, devitrification).

Numerous works have been carried out by NASA:

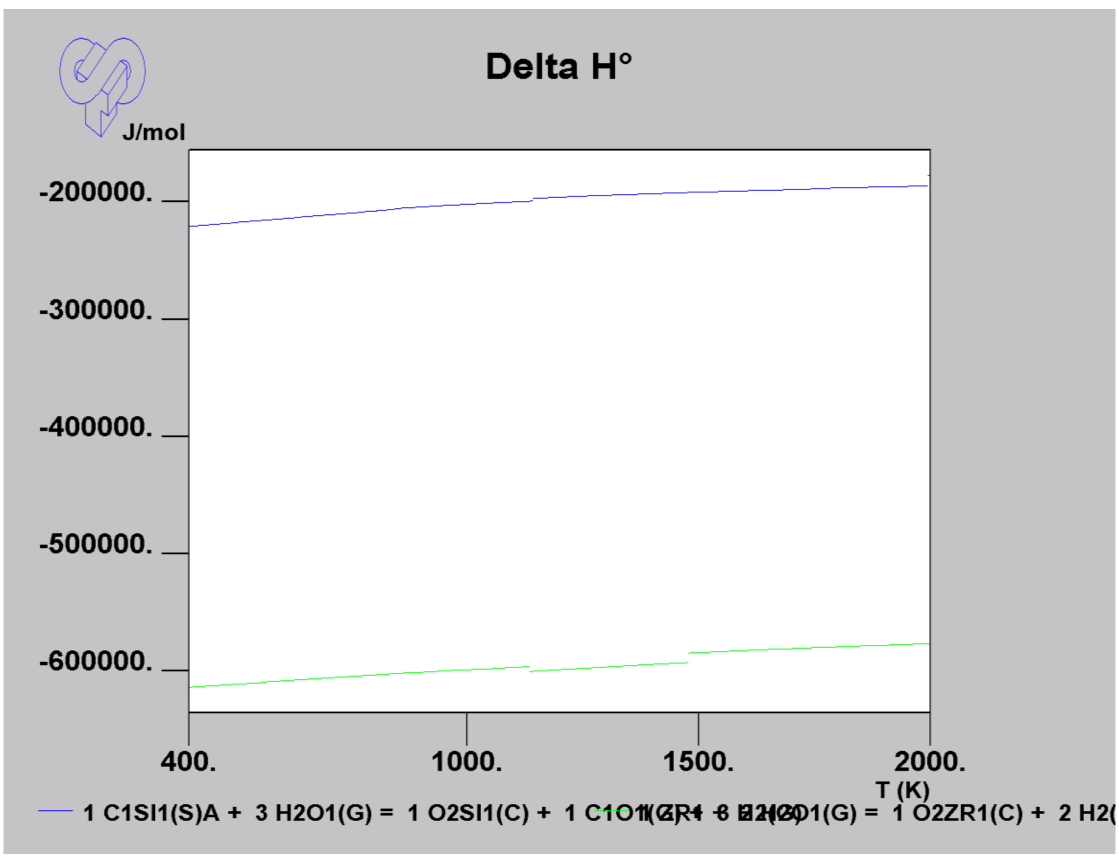
- Dry O_2, 1 bar, 0.4 cm/s, 1200-1400°C: parabolic kinetics (protective silica layer); reaction is $SiC + 3/2\ O_2 = SiO_2 + CO$ (mass gain 20 g/mol of SiC)
- Mixture H_2O, 0.1 bar + O_2, 0.9 bar, 0.4 cm/s, 1200-1400°C : idem
- Mixture H_2O, 0.5 bar + O_2, 0.5 bar, 1.4 cm/s, 1200-1400°C: the thickness of the protective silica layer does not exceed few microns and the kinetics become linear (loss of mass by evaporation into CO and Si hydroxides); the reaction is for example $SiC + 5\ H_2O = Si(OH)_4 + CO + 3\ H_2$ (loss of mass 40 g/mol of SiC) [3]. At 1300°C, the identified species is $Si(OH)_4$ [4]. At 1400°C, $SiO(OH)_2$ also contributes [5-6].
- Tests with saturation in Si hydroxides, inhibiting volatilization, by prior passage over a quartz tube, $H_2O + O_2$ mixtures, 1 bar total, 1100-1400°C: kinetics accelerated by a factor of 10 compared to dry O_2, and increasing with the partial pressure of water vapor [7].

Tests at higher steam pressure were performed by ORNL [8]:

- H_2O mixture, 1.5 bar + air, 8.5 bars, 3 cm/min, 1200°C: Distinct SiO_2 scale structures were observed on the SiC; thick, porous nonprotective cristobalite scales formed above a thin nearly dense vitreous SiO_2 layer, which remained constant in thickness with time as the crystalline SiO_2 continued to grow**.** The thickness of the *vitreous* silica layer does not exceed microns and the kinetics become linear (breakaway: formation of a thicker layer of porous cristobalite). It is found that the high water-vapor pressure accelerated the formation of SiO_2, the rate of its crystallization, and the amount of porosity formed in the resulting oxide. These processes resulted in measurements of SiC recession rates that are higher than what could be explained by models that relate parabolic oxidation-rate constants to water-vapor pressures or linear recession rates to gas velocity based on SiO_2 volatility.

A NASA paper [9] attempts to synthesize these different results: at high gas velocity, the silica volatilization is the dominant process whereas at low gas velocity, the breakaway is the dominant one. A first conclusion is that representative tests in water vapour partial pressure and velocity should be available. For the large breach LOCA, the primary pressure quickly equalizes with the pressure in the vessel; the NRC Reactor Safety Course (Figure 4.1-10 in [10]) gives an example of a calculation where the vessel pressure is 5 bars absolute for a breach in the RCV discharge line. For the intermediate breach LOCA, the NRC PIRT LOCA [14] gives a Westinghouse calculation as an example where the pressure is 35 bars for a 3-inch breach (Figure 7 p I-54 in Appendix I of [11]).

Oxidation tests have been performed by MIT at 1140 and 1200°C [12] at atmospheric pressure. Comparative tests were performed on Zry. SiC weight loss two orders of magnitude less than Zry-4 weight gain. Other tests were performed at ORNL, studying higher pressure conditions [13] up to 20 bars. Also, in these tests, material recession is much less than for Zrys. Data should be produced at higher pressure (~40 bars) and up to 1204°C if the lack of data in these particular conditions are still missing.



From around 1700°C, as for all molten oxides, the molten or at least softened silica no longer protects the SiC efficiently, it no longer offers any resistance to oxygen diffusion. As for steel and boron carbide, this is a "cliff effect" known in the literature as "catastrophic oxidation by molten oxides" [14]. In the models, it is therefore necessary to calculate an instantaneous oxidation of SiC above the melting temperature of the silica, the oxidation being limited only by the addition of water vapor (hydrogen/CO blanketing). It is above this temperature that the oxidation of SiC will produce more combustible gases and heat of oxidation than zirconium. There is a risk of saturation of recombiners, and risk of CO + H_2 explosion in the containment (PWR).

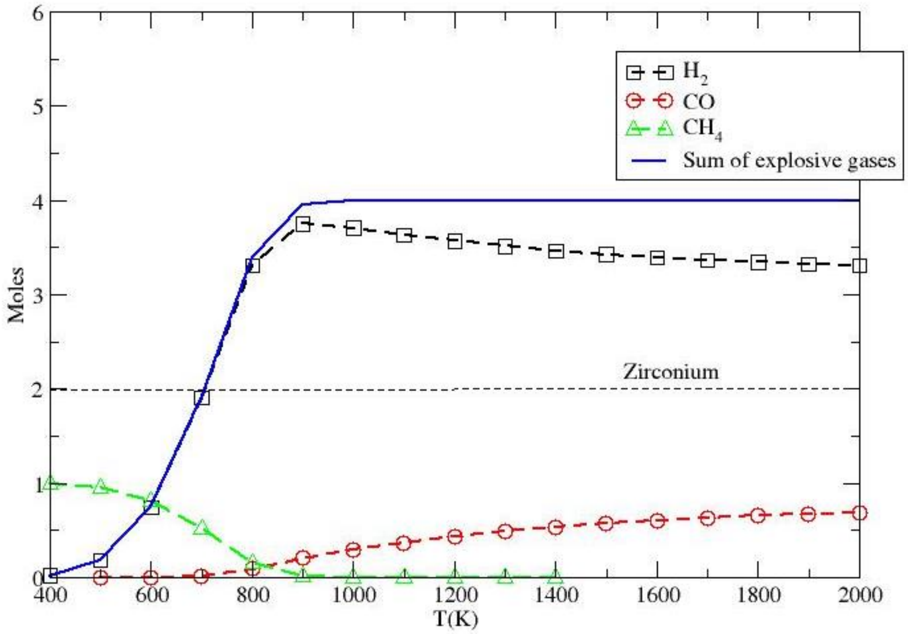
Thermodynamic calculations were performed with the NUCLEA thermodynamic database [15] and the NucleatoolBox [16|. The calculation conditions were as follows:

- Excess water vapor (10 moles of H_2O for 1 mole of SiC),
- Atmospheric pressure,
- Temperature from 400 to 2000 K.

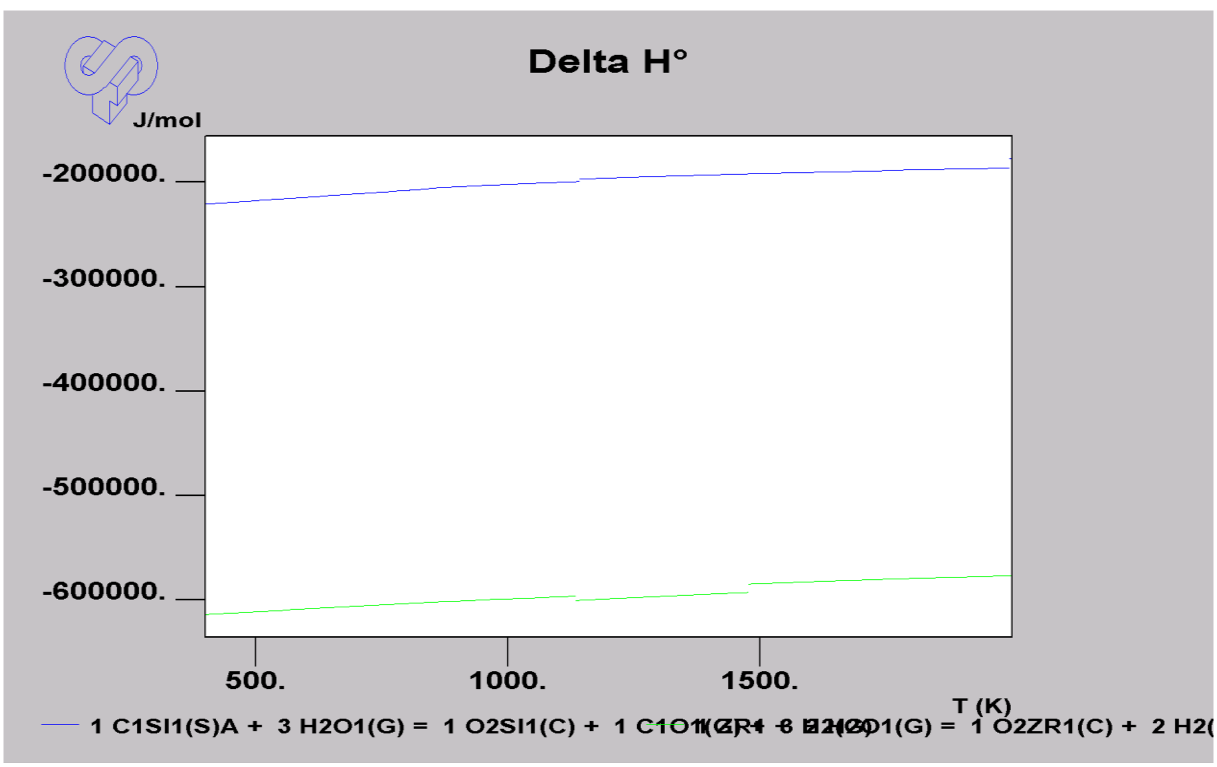
The oxidation products formed are on the one hand silica SiO_2, solid until 2100 K then liquid beyond, on the other hand combustible and explosive gases CO and H_2. Depending on the temperature level, CO_2 is also formed at the expense of CO and other hydrogenated compounds such as OH. The following graph shows the results for the explosive gases H_2 and CO. The formation of CH_4 is present at "low" temperatures, i.e., below 1000 K.

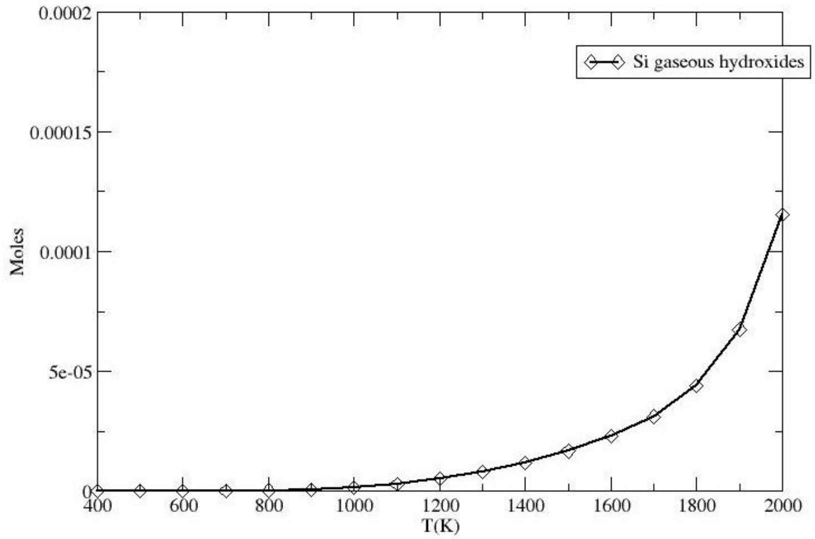
The following graph summarizes the calculation results for CO and H_2. For comparison, the amount of hydrogen formed by the oxidation of 1 mole of Zr is shown. From 900 K, the number of explosive moles of CO + H_2 stabilizes at 4, whereas it is 2 for Zr. A refined comparison with Zr must take into account the dimensions of the cladding and the density of the materials. The table below, using the data of for SiC [17], allows to evaluate the number of moles of SiC or Zr per meter of cladding and to make the necessary corrections. It allows to confirm that at thermodynamic equilibrium, the number of moles of explosive gas resulting from the oxidation of SiC per unit length will be more than twice the number of moles of explosive gas resulting from the oxidation of zirconium.

The enthalpy of reaction $SiC + 3 H_2O(g) \rightarrow SiO_2 + CO + 3H_2$. calculated from Coach [3| is much less negative (then less exothermic) than that for oxidation of Zr ($Zr + 2 H_2O(g) \rightarrow ZrO_2 + 2H_2$).



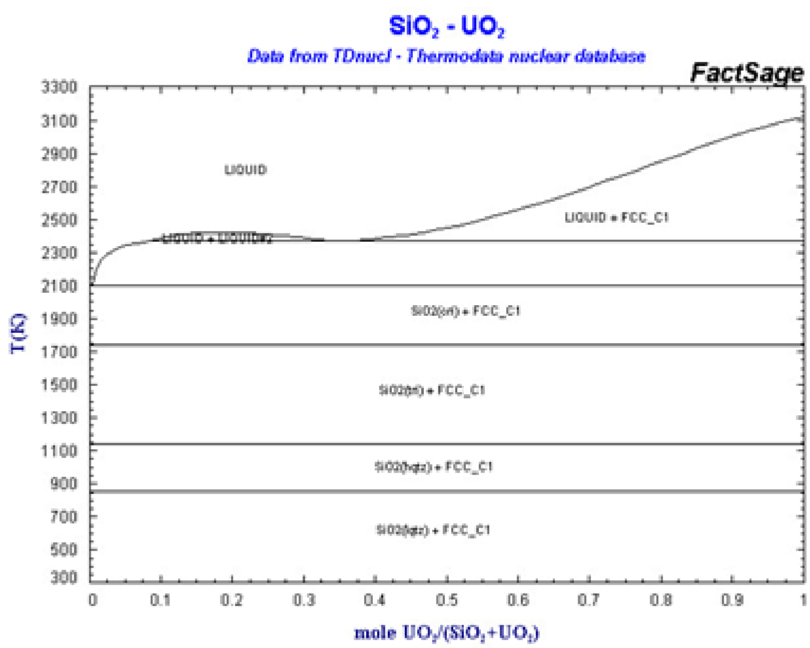
	External diameter *D* (cm)	Internal diameter *d* (cm)	D^2-d^2	Density ρ(kg/dm^3)	$\rho(D^2$-$d^2)$ (g/m)	Molar mass *M* (g/mol)	$\rho(D^2$-$d^2)/M$ (mol/m)
SiC	1.090 [17]	0.958 [17]	0.270	3.1 [17]	84	40	2.1
Zr	0.950	0.836	0.204	6.5	132	91	1.5





Regarding the hydroxide production, the calculation shows that the related gases begin to be released at rather high temperature. They are expected to modify the chemistry in the primary circuit, in particular the Cs one by potentially forming silicates.

The UO_2-SiO_2 phase diagram, based on Lungu's measurements [18-21] indicates that from 2100 K, liquid phase is able to be formed. Recent measurement from Bechta et al. [22] do not drastically change the phase diagram shape. Then liquid SiO_2 is likely to interact with UO_2 to form liquid mixtures at temperatures well below that of fuel melting. In the other direction, the phase diagram in the silica-rich region is of monotectic type, indicating that urania and silica are likely poorly miscible. It could have some consequences in terms of degradation and corium morphology with a separation between the silica-rich and urania-rich phases. It could favour the occurrence of solid fuel slumping degradation scenario.



The comparison with zirconium-based alloys is as follows:

- The number of moles of explosive gases (hydrogen + carbon monoxide) is about twice as high per mole, and three times as high per meter of cladding, as for zirconium,
- The oxidation kinetics will probably be lower below the melting temperature of silica, above which a "cliff effect" occurs making it higher.
- The heat of reaction is lower per mole, than for Zr.
- The formed can interact with uranium dioxide to form liquid phases from 2100 K.

References

[0] G. Hache, B. Clément, M. Barrachin, Safety of some candidate fuel cladding(& structural) materials, OECD/CSNI/WGFS meeting, September 12-13th, 2012, Paris, France

[1] L. Hallstadius, S. Johnson, E. Lahoda, Cladding for High Performance Fuel, Progress in Nuclear Energy 57 (2012) 71-76

[2] J. Grobner, H.L. Lukas, F. Aldinger, Thermodynamic Calculation of the Ternary System Al-Si-C, Calphad 20(2) (1996) 247-254

[3] E.J. Opila, R.E. Hann, Paralinear oxidation of CVD SiC in water vapour, Journal of the American Ceramic Society 80 (1997) 197-205

[4] E.J. Opila et al., Mass spectrometric identification of Si-O-H(g) species from the reaction of silica with water vapor at atmospheric pressure, Journal of the American Ceramic Society 80 (1997) 1009-1012

[5] N. Jacobson et al., Interactions of water vapour with oxides at elevated temperatures, Journal of Physics and Chemistry of Solids 66 (2005) 471-478

[6] N.S. Jacobson et al., Thermodynamics of gas phase species in the Si-O-H system, Journal of Chemical Thermodynamics 37 (2005), 1130-1137

[7] E.J. Opila, Variation of the oxidation rate of silicon carbide with water-vapor pressure, Journal of the American Ceramic Society 82 (1999) 625-636

[8] K.L. More et al., Observations of accelerated silicon carbide recession by oxidation at high water-vapor pressures, Journal of the American Ceramic Society 83 (2000) 211-213

[9] P.F. Tortorelli, K.L. More, Effects of high water-vapor pressure on oxidation of silicon carbide at 1200°C, Journal of the American Ceramic Society 86 (2003) 8 1249-1255

[10] ERI Consulting et al., Perspectives on Reactor Safety, NUREG/CR-6042 Rev.2, § 4.1.2 (1993)

[11] B.E. Boyack et al., Phenomenon Identification and Ranking Tables (PIRTs) for LOCAs in Pressurized and BWRs containing High Burnup fuel, NUREG/CR-6744, December 2001, Appendix I, p I-40 to I-72

[12] A.P. Ramsey, Silicon Carbide Oxidation in High Temperature Steam, MIT, PhD Thesis, 2009.

[13] B.A. Pint, High Temperature and Pressure Steam-H_2 Interaction with Candidate Advanced LWR Fuel Claddings, Oak Ridge National Laboratory, Report ORNL/TM-2012/254 (2012)

[14] High Temperature Corrosion, P. Kofstad. Elsevier Applied Science Publishers Ltd., London New York, 1988.

[15] E. Fischer, NUCLEA Thermodynamic Database, Version 2020, Institut de Radioprotection et Sûreté Nucléaire (France)

[16] B. Piar, NucleaToolbox, Version 2020, Institut de Radioprotection et Sûreté Nucléaire (France)

[17] S. Higushi et al., A study on thermo-mechanical behaviour of LWR fuel rod with SiC cladding, Topfuel 2009, Paris, September 6-10, 1236-1243

[18] S. Lungu, Equilibrium diagram and masses from the molten SiO_2-UO_2 system, Revue Roumaine de Physique 7 (1962) 419-425 @article{lun-66-etu,

[19] S. Lungu, Etude des courbes de liquidus et des propriétés thermodynamiques des systèmes SiO_2-ThO_2 et SiO_2-ThO_2-UO_2, Journal of Nuclear Materials 19 (1966) 155-159

[20] S. Lungu, Some properties of UO_2-SiO_2 melts, Journal of Nuclear Materials 35 (1970) 35-40

[21] S. Lungu, I.L. Beleuta, D. Apostol, O. Radulescu, Some theoretical aspects in the spinodal decomposition and crystallization of silicate melts. Experimental results for the UO_2-SiO_2 system, Journal of Nuclear Materials 48 (1973) 165-171

[22] S.V. Bechta, Investigation of binary oxidic systems: UO_2-SiO_2 system, Alexandrov Research Institute of Technology, NITI, Russia, ISTC Report (2006)

Comments by Richard Denning

These comments are limited to two advanced fuel concepts: uranium silicide and uranium nitride fuels, with either FeCrAl or chromium-coated Zircaloy cladding. I have not been associated with research on these concepts and the comments are based on reviewing research in the recent literature. The most important observation is that the behavior of these two fuel types under severe accident conditions is substantially different from the behavior of conventional fuels. In contrast to the near-term concepts for which modest modifications to MELCOR modeling would be sufficient to make judgments about safety implications without additional experimental validation, that is not the case for these fuels within the context of the figures of merit of interest: fission product release and hydrogen production. Furthermore, the experiments performed to date, of which I am aware, provide an insufficient basis for modeling. The most advanced modeling effort is being performed for the BISON code. Although these activities provide important input to understanding accident behavior, a substantial experimental activity under prototypic conditions would be required to support MELCOR model development for application to risk-informed decision making.

Both of these fuel types would deteriorate rapidly under attack by high temperature steam. Thus, modeling of cladding integrity as a function of time in the accident is critical. With a loss of external cooling, the high thermal conductivity advantage of these fuels is lost, the temperature of the fuel will rise and volatile fission products will be released to the fuel gap (if any) and gas plenum of the rod but not to the reactor coolant system until there is cladding failure. Cladding failure could be accelerated in time by swelling of the fuel, contact with the clad and fuel-clad mechanical interaction. Once the cladding has failed, high temperature steam can attack the structure of the fuel, leading to fuel oxidation, hydrogen production, disintegration of the fuel matrix, and release of fission products to the reactor coolant system. For the analysis of these fuels, it will be necessary to develop a number of models that do not currently exist in MELCOR. Phenomena associated with eutectic formation, candling, fuel slumping, fission product release from fuel as a function of temperature, and hydrogen production will have limited applicability to the degradation of these fuels. New models must be developed for the prediction of the rate and consequences of fuel degradation under steam attack, hydrogen production and the associated release of radionuclides (and whether their chemical forms are impacted by the different fuel constituents). At a minimum, prototypic experiments will be required on high-temperature steam exposure to pre-irradiated fuel samples to determine the associated release fractions of radionuclides. It is less clear that full primary system experiments as in Phebus tests would be required. Consideration must be given to the potential for the aerosolization of fuel particles, which could result in the dispersion of lower volatility radionuclides with higher biological effectiveness, although I think it is unlikely that such aerosols would be small enough to be respirable.

Some references of interest:

K.A.Gamble et al., "BISON Capability and Validation for U3Si2, Cr2O3-doped U)2, FeCrAl, and Cr-coated Zircaloy ATF Concepts," INL/CON-20-57530, April 2020.

K.E.Metzger et al., "Model of U3Si2 Fuel System Using Bison Fuel Code," INL/CON-13-30445, 2014.

J.T.White et al., “Thermophysical Properties of U3Si2 to 1773K,” LA-UR-14-29472, Journal of Nuclear Materials, 2016.

E.S.Wood et al., “High Temperature Steam Oxidation Dynamics of U3Si2 with Alloying Additions: Al, Cr, Y,” Journal of Nuclear Materials, 2020.

A.T.Nelson et al., “Overview of Properties and Performance of Uranium-Silicide Compounds for Light Water Reactor Applications,” Trans of ANS, Vol. 110, June 2014.

J.M.Harp et al., “Uranium Silicide Pellet Fabrication by Powder Metallurgy for Accident Tolerant Fuel Evaluation and Irradiation,” Journal of Nuclear Materials, 2015.

NEA, “State-of-the-Art Report on Light Water Reactor Accident-Tolerant Fuels,” NEA-7317, 2018.

Comments by Jeff Gabor

The longer-term ATF concepts being considered by industry include:

- Silicon carbide cladding,
- High-density silicide fuels,
- High-density nitride fuels, and
- Metallic fuels (specifically, uranium-zirconium alloys with zirconium content near 50 weight percent).

Specific areas investigated related to the long-term ATF concepts include:

- For each longer-term ATF, please comment on availability of any test data that can help focus modeling of fuel/cladding behavior under severe accident conditions.
- For each longer-term ATF, list any potentially unique in-vessel and ex-vessel fuel/cladding degradation and melt progression issues that may alter our current modeling framework.
- For each longer-term ATF, list potentially unique in-vessel and ex-vessel radionuclide release and transport issues that need to be addressed.
- For each longer-term ATF, list any potential physiochemical processes and materials interactions issues that may merit additional testing and research.

Silicon carbide cladding

1. Given the increased strength at high temperature and the specific control rod material used, the potential may exist where the control rod material relocates prior to the clad and fuel.
2. A delay in clad rupture due to increased strength at high temperature could delay the fission product release to the containment. Early transport and deposition of fission products in the BWR suppression pool could be impacted by this cladding material.
3. Information provided in ERI/NRC 20-209 indicates potential for inner SiO_2 surface layer to absorb cesium. This could result in an increase of gaseous iodine released to the containment early.
4. Impact of possible dissolution of SiC clad material into coolant over time via physio-chemical processes (i.e., formation of silicic acid which is swept away from clad surface via coolant flow reducing clad thickness and possibly structural integrity over time). Will need experimental data for characterization of this process for both borated water (PWR) and steam (BWR).

High-density silicide fuel

1. Higher thermal conductivity could impact fuel/clad relocation and in-core melt progression.

High-density nitride fuels

1. Higher thermal conductivity could impact fuel/clad relocation and in-core melt progression.

Metallic fuels (specifically U-ZR alloys)

1. Potential for increased Zr oxidation and increased hydrogen generation.
2. Low melting point will impact in-core melt progression and timing (particularly for extended loss of cooling events such as ELAP).

3. Impact of fretting on fuel geometry (and possibly structural integrity) under typical BWR/PWR core flow conditions.

Comments by Randall Gauntt

The main emphasis of this PIRT has been focused on emerging, higher readiness-level ATF concepts that are evolutionary in design relative to the well-studied UO_2-Zircaloy fuel design that is nearly ubiquitous to commercial LWR use. Another class of ATF fuels are being considering where the fuel properties are significant deviations from current UO_2 designs, in the goal of further advancing thermal and mechanical performance and to further increase safety margins between normal operation and the performance envelope under accident conditions. Some aspects of these extended designs will be discussed, with a key emphasis on the applicability of current safety codes (e.g. MELCOR) to the safety assessment of these fuels under DBA and BDBA accident conditions.

High Density Advanced Silicide and Nitride Fuel Forms

Two such fuel concepts consider the use of either uranium nitride or uranium silicide forms, which are of interest because of their comparatively higher physical density and higher thermal conductivity. These properties act to limit peak fuel centerline temperature for a given fuel rod power density with greater net heat removal from the fuel, thereby slowing fuel heatup and potentially extending coping times for signature accidents such as SBO and ELAP. As long as the geometry of the fuel is intact and the properties such as enthalpy and thermal conductivity are known, the accident codes will be able to predict the response of the fuel types up to the point where temperature limits for the fuel design, such as melting points (which can be significantly power than UO_2) and the “rod” geometry changes.

Computer codes can be fuel-form agnostic to a point relative to traditional UO_2-Zr fuel performance, where the codes can simply replace UO_2 and Zr properties with those of advanced fuel thermal properties – as far as the codes are concerned, fuel is fuel and clad is clad. However, current LWR computer code architecture is based on the assumption that fuel materials remain as solid or melt/mixture form and that the fuel material (assumed to behave as UO_2) does not participate in oxidation with steam. The advanced nitride and silicide fuel forms may not fit this view since the fuel materials can react with steam to produce volatile non-solid physical state as well as hydrogen and other gases arising from chemical reactions. While experiments can shed light on the fuel behaviors and provide the basis for the creation of performance models, it can remain a significant technological effort to extend the computer data base to track the fuel oxidation products as the fuel degrades. Additionally, the code data base must be expanded to provide all model state variables necessary to predict performance and track mass as the fuel form reacts chemically and degrades throughout the accident transient. In order for the codes to be able to predict fuel degradation and source term release, these architectural extensions must be implemented and the interactions and transport behavior of the new added fields must be added. It is likely that current models for fission product release from fuel that are based on the Booth diffusion approximation could need significant modification to predict source term evolution, especially if the fuel form expands or sublimes. The latter of course requires extension of tracked material fields in MELCOR. Degraded fuel material that is mobilized as a vapor or particulate form need new database variables to calculate transport and need new interaction models to predict downstream deposition.

Metallic Fuel Forms

Another class of advanced fuel forms include metallic forms of uranium and zirconium. Such forms are not completely novel in that they have been examined for fast reactor (sodium) applications as well as maritime reactor applications. Many of the properties of metallic fuel forms are already included in MELCOR, and the nominal performance of such designs up to the melting point of the fuel is not a great stretch for current safety codes. Fuel forms may be rod-like and may have Zr cladding as the barrier between the coolant (here assumed to be water) and the underlying fuel material (assumed to be uranium with some weight percent e.g. ~50% weight), and under accident conditions where fuel is exposed to steam, oxidation of both cladding and fuel can occur. The inner U-Zr fuel may be molten before the cladding melts owing the phase equilibrium of the U-Zr system. This fuel form would seem to more compliant for existing code (MELCOR) architecture, where existing models and database parameters simply need updated properties. One would expect some significant differences in fission product release in this fuel form relative to UO2 fuel, requiring some reassessment of the applicability of a diffusional approach for fission product transport as well as modeling heatup and oxidation of both fuel and cladding.

Silicon Carbide

Silicon carbide has been examined as possible application as a fuel cladding or as a channel box material, and manufacturing of silicon carbide has made advances in these proposed applications. Additionally uranium carbide has also been examined as a potential fuel form. Like the previously discussed advanced fuel forms, the safety codes such as MAAP and MELCOR have significant existing and applicably code database and architecture to model fuel performance provided substitute properties are available to the existing code database variables. However, it is anticipated that some fairly significant code architecture extensions would be needed to allow for oxidation of these silicide forms in order to populate new material fields required for modeling oxidation behavior.

Comments by Richard Hobbins

SiC Cladding

K. Terrani [1] states that SiC/SiC composites are deemed the ideal ATF cladding material. Oxidation in high pressure, high temperature steam produces a protective layer of SiO_2 resulting in a much slower rate of oxidation than that of Zircaloy. Tests of SiC/SiC composite tubes showed that the tubes remained coolable by quenching for exposures in steam up to three days at 1600 ℃ and on the order of hours at 1700 °C and 1800 °C [2]. However, the SiO_2 layer undergoes slow (mass transport limited) volatilization in steam [3], so Terrani calls for additional separate effects and integral tests to examine the behavior of SiC/SiC cladding technologies under BDBA scenarios [1].

There are issues with SiC/SiC composites during normal LWR operation that deserve additional consideration according to Terrani [1]. Hydrothermal corrosion of the protective layer of SiO_2 has been observed that can be controlled by hydrogen water chemistry in PWRs, but in BWRs, even when using hydrogen water chemistry, selective migration of hydrogen to the gas phase may occur as boiling takes place in fuel bundles, resulting in a detrimental increase in oxygen activity in the liquid phase. The potential for microcracking during normal irradiation in LWRs may open up pathways for release of radionuclides to the reactor coolant; however, the outer monolithic SiC layer may serve as an additional barrier to radionuclide release.

The much slower oxidation of SiC/SiC composite cladding results in much reduced production of combustible gases, H_2 and CO, compared with Zircaloy. However, the database on the behavior of SiC/SiC cladding under BDBA conditions is sparse and requires additional testing.

References:

1. K. Terrani, Accident tolerant fuel cladding development: promise, status, and challenges", J. Nucl. Mater. 501, 13-30 (2018).

2. V.A. Avincola, et al., "Oxidation at high temperatures in steam atmosphere and quench of silicon carbide composites for nuclear application" Nucl. Eng. Des. 295, 468-478 (2015).

3. E.J. Opila, et al., Paralinear oxidation of CVD SiC in water vapor, J. Am. Ceram. Soc 80, 197-205 (1997).

U_3Si_2 Fuel

Microstructural degradation is observed following one hour of exposure of monolithic pellets of U_3Si_2 in flowing steam at 350 ℃ and pulverization occurs on the timescale of minutes when temperatures are increased above 400 °C [1]. This mechanism is accelerated in flowing Ar-6%H_2 at the same temperatures [1]. These degradation mechanisms for U_3Si_2 pellets in flowing steam do not bode well for this fuel when cladding failure occurs under BDBA conditions in severe accident scenarios for LWRs.

I suspect degradation of U_3Si_2 in steam will be exacerbated in irradiated fuel and with H_2 from Zircaloy cladding oxidation. Experiments with irradiated U_3Si_2 fuel are required to more fully investigate fuel degradation, fission product release, and combustible gas production as a function of H_2/H_2O ratio and temperature in flowing steam under BDBA conditions.

Reference:

1. E. Sooby Wood, et al., U_3Si_2 behavior in H_2O: Part I, flowing steam and the effect of hydrogen, J. Nucl. Mater. 501, 404-412 (2018).

UN Fuel

UN fuel pellets exposed to a flowing mixture of steam and Ar (0.5 atm steam) at 500 ℃ were reduced to a fine oxidic powder within 1 hour [1]. These experimental results suggest that both fission product release and hydrogen production could be unacceptable for irradiated UN fuel after cladding failure under BDBA conditions. Clearly, any proponent of this fuel for use as an ATF in an LWR will need to provide evidence of the behavior of irradiated UN fuel under BDBA conditions.

Reference:

1. M. Jolkkonen, et al., Uranium nitride fuels in superheated steam, J. Nucl. Science and Technol. 54, 513-519 (2017).

U-50wt%Zr Fuel

This fuel (72 at.%Zr) will oxidize in steam after cladding failure and produce copious amounts of hydrogen, which would appear to be counterproductive for use as an ATF in LWRs. Any proponent of this fuel concept will need to provide experimental evidence of the performance of irradiated fuel rods, for both hydrogen production and fission product release, in flowing steam under BDBA conditions.

Comments by Luis E. Herranz

The comments below are based on open literature the author has had access to, not on research done on any of them. As in the PIRT done on short-term ATFs (STATF), the figures of merit to focus on are fission product and hydrogen release (state of knowledge and importance).

Generally speaking the information available is not enough for a deep assessment of these long-term ATFs under severe accident conditions, as most of their characterization so far has addressed less extreme conditions than what foreseen in severe accidents. Nonetheless, there are some studies that have attempted to explore what the (LTATF) might be; all those studies are lacking some elements that might become crucial for the accident unfolding. One of those elements is the potential for eutectics formation, which effect might or might not be substantial. So, the applicability of those studies would be to the very early stages of the accident, to say the most.

Given the lack of info (on my side) concerning potential eutectic formation, applicability of MELCOR models (fitted to the current LWR database) would require “ad-hoc” development and validation. In addition, few data on performance on fuel components at very high temperature makes highly uncertain any estimate at T> 1700ºC. Therefore, to properly update MELCOR to reach the same maturity level as it does have with conventional fuels, further data and model development would be needed.

The specific paragraphs below refer to: Silicon carbide cladding, high-density silicide fuels. Literature has been read concerning other LTATFs, but the information contained does not allow any discussion as for severe accident behavior.

- **Silicide fuels**

Even though some analyses have already done on accident scenarios (PWR LOCA and SBO or SBO –Hess et al., 2019), the major difference modeled with respect to the conventional fuel is thermal conductivity (kU3Si2) and no considerations is made on in-reactor materials interactions. Those exploratory materials do not bring large differences in terms of FP releases, but they do for H2 releases (associated with the use of SiC). Hence, the analysis looks just partial and, besides, a noticeable scatter in kU3Si2 has been also reported (White et al. 2015) at temperatures below 1773 K. Some mechanistic attempts to model FP release during normal operation at low burnup fuel have been done, but even those are said to be highly uncertain (Gamble et al., 2019).

Some analysis out there (Wu et al., 2015) show comparative studies under accident conditions; however, the code used is unknown to me and some of their hypotheses are not supported (like no interaction of fuel rod with structural and/or control rod material). This said, it brings up the matter of CO production from SiC cladding, which should be addressed when dealing with SiC based claddings.

- **Silicon carbide**

The major difference with respect to conventional fuel is, of course, the lower oxidation rate (even orders of magnitude) what would mean a drastically different production of H2 (Avincola et al, 2015). Not much has been found concerning FP interaction with cladding, what might affect retention of some radio-nucleids. As for severe accident conditions it has been indicated that SiO2 formed in the oxidation process might tie up Cs and affect to the fraction of gaseous iodine travelling through RCS and eventually reaching containment (higher).

Reference documents:

U_3Si_2 Fuel

- Yang et al., JNM 542 (2020) 152517.

- Wood et al., JNM 533 (2020) 152072.

- Sweet et al. , JNM 539 (2020) 152263

- Blanchard et al., Project Number 11-3041

- White et al., 464 (2015).

- Watkins et al., JNM 518 (2019)30-40.

- Harrison et al., Corrosion Science 174 (2020) 108822.

SiC-SiC Composites

- Deck et al., JNM 406 (2015) 667-681.

- Koyanagi et al., JNM 540 (2020) 152375.

- Braun et al., JNM 487 (2017) 380-395.

- Avincola et al., NED 295 (2015) 466-478.

(U,Pu)C Fuel

- Mazandier et al., JNM 406 (2010) 277-284.

Microcell UO_2 Pellets

- Kim et al., NET 50 (2018).

Cross-references

- Zhon et al, ANE 119 (2018) 66-86.

- Wu et al., ANE 80 (2015) 1-13.

Comments by Didier Jacquemain

SiC cladding (composite of SiC fibers and SiC matrix (SiCf/SiC))

Different modes of fabrication can be used for SiC cladding which may result in different thermal and mechanical behavior [1,2]. We consider here a SiC multi-layered structure with a layer with sufficient hermeticity (e.g. fabricated by chemical vapor deposition (CVD)-SiC layer or by nano-filtration and transient eutectic phase (NITE) or including a metal liner).

Main technical advantages of SiC as a cladding material: neutron transparency, high temperature strength (e.g. stiffness and competitive fatigue behavior), resistance to oxidation in high temperature steam, inherent radiation resistance, lack of progressive irradiation growth, added margins for DBA and DEC (may maintain a coolable geometry at higher temperature): expectations are that coolability can be maintained for a few hours at 1700°C to 1800°C and for about three days at 1600°C as reported in [1,2].

Economic advantages: possible use with smaller U enrichment and higher fuel burn-ups, allows for increased fuel cycle duration, etc.

Plans are to use SiC both for cladding and core structures (e.g. BWR channel boxes).

Main issues associated with the use of SiC cladding [1,3]:

- Chemical compatibility with the coolant under normal conditions: hydrothermal corrosion at the interface material between SiC fiber and matrix, SiC dissolution in water (recession: loss of mass and thickness) dependent on water chemistry
- Possible early formation of micro-cracks that can affect leak-tightness and reduce capability of SiC ceramic composites for fission gas and FP retention
- Possible bowing of rods (can be mitigated using spacer grids)
- Resistance of SiCf/SiC cladding to PCMI failure has to be assessed because the material is susceptible of brittle failure (PCMI inner pressure tests to be done)

Low thermal conductivity and multi-layered structure of SiC result in larger interfacial thermal resistance and reduced radial heat transfers and possibly (1) in differential swelling of irradiated SiC generated by the high temperature side of SiCf/SiC which may cause high tensile stress and early multi-cracking (2) in elevated centerline fuel temperature. The fuel could reach its melting temperature faster with an increase in the fission gas release affecting inner gap pressure.

Related to these issues [1,3]:

- A lot of research has been done to address the chemical compatibility issue: development of material with superior resistance, optimization of water chemistry to limit recession
- A lot of experimentation was done to investigate SiC behavior under irradiation and post-irradiation annealing. These would tend to conclude that irradiation effects will have a limited impact on high temperature behavior. A lot of effort were done for the development of material with superior neutron irradiation tolerance

- Significant research was also done to address the hermeticity issue (where tritium and FGR release during normal and transient conditions may be issues) with development of joining technologies
- Significant research was done related to the mitigation of the PCMI risk but need to be pursued

Globally, research has mostly focused till now on material enhancement for performance under normal and transient conditions. Research on behavior during accidents remains to date limited.

For DBA, new cladding failure criteria, new licensing guidelines need to be developed considering specific failure modes of SiC/SiC [3]. Thermo-mechanical behavior related to PCMI and under RIA need to be further investigated.

For SA, mechanisms for SiC oxidation and SiO_2 volatilization at high temperature (up to 1 800°C) are rather well known, see e.g. [4]. Oxidation and volatilization become significant when reaching 1700- 1800°C with formation of bubbles on the cladding surface. No data appear to be available above 1800°C and it would be of interest to get further insights on high temperature SiC cladding failure mechanisms. Both are needed to appreciate H_2, CO formation at high temperature and assess main FP release upon clad failure.

In many areas, knowledge developed for conventional cladding and fuel may not be applicable, examples include:

- Critical phenomena leading to the loss of coolable geometry with SiCf/SiC claddings
- Clad dislocation criteria, material relocation phenomena and debris formation and accumulation (with a highly brittle material)
- Interactions between SiC and other structure materials if any (e.g. stainless steel, control rods or blades materials) and resulting relocation phenomena
- Interactions between SiC and UO_2 in steam at high temperature
- generation of CO and CH_4 and effect on FP chemistry and transport (e.g. iodine)
- generation of silicate compounds that may affect FP chemistry and transport (e.g. formation of Cs silicates as seen for the Three Mile Island and the Fukushima Daiichi accidents)

Knowledge may need to be developed in most of the above areas, particularly on the first three as the description of evolution of in-core degraded configurations at high temperature (above 1800°C) should be adapted for the SiC cladding and knowledge on high temperature behavior above 1800°C is limited. This will largely influence both H_2, CO formation and FP release. Note that one of the OECD/NEA Quench-ATF project test may be performed to investigate the high temperature behavior of SiC claddings.

For the last three areas, fundamental knowledge on properties and characteristics of SiC composites, see for instance [5], on formation of phases such as UC, USi_2, U_3Si_2 by SiC/UO_2 interaction [6], on formation of USix along with CO, of a liquid phase between 1850 and 1950 K by SiC/UO_2 interaction [7] on the (U, C, O) phase diagram (looking e.g. at CO/CO_2 formation by interaction between UO_2 and carbon for the case of TRISO particles [8]) can be used to develop some modeling of the key high temperature chemical interactions for the SiC/UO_2 system. Also, formation of cesium silicates and interactions of carbon compounds with iodine at lower temperatures have been studied and existing chemical models can possibly be used to assess the effect of SiC degradation materials on transport of main fission products.

Higher density fuels

Developments in this area are related to the replacement of Zr cladding with other materials with reactivity penalties.

Higher density fuels are less water/steam tolerant than UO_2 and research work is being made to increase water/steam tolerance of such fuels and they are intended to be used with new accident-tolerant claddings which are expected to have a much higher resistance to leakage up to higher temperature.

Fundamentally, regarding behavior under severe accident conditions in LWRs, little is known and a major concern will be the fuel oxidation with steam when reaching clad failure and its eventual dissociation (nitride) or melting (silicide) when reaching higher temperature (due e.g. to exothermic oxidation). Use of such fuels can only be envisaged with a cladding that remains highly protective up to high temperature.

Nitride fuel UN or (U, Pu)N

A large variety of fabrication methods exist for these fuels [1,3]. The need to have low O and C content for material resistance to oxidation is highlighted. High ^{15}N enrichment is needed to avoid formation of 14-chain by products and reduce neutronic penalties.

The fission gas release is expected to be lower than for UO_2.

PCMI requires specific attention due to high swelling rates.

The compatibility with the cladding, e.g. SiC or FeCrAl, needs to be addressed.

Lack of in-reactor experience as well as of knowledge of off-normal behavior is highlighted. Knowledge is usually deduced from carbide and oxide fuel systems in fast breeder reactors. PCMI may occur fast due to accelerated swelling at high temperature.

In case of cladding failure, nitride fuels react with steam producing hydrogen with an accelerated kinetic above 523 K. it also produces compounds such a NH_3.

Fuel dissociation was observed in (U, Pu)N fuel releasing Pu at high temperature in He ($T > 2000$ K) and producing N_2 which could lead to rod over-pressurization.

Thus, regarding SA behavior in LWRs, a major concern will be the fuel oxidation when reaching clad failure and its eventual dissociation when reaching higher temperature.

Note that regarding the modeling of these fuels, Transuranus has been adapted for (U,Pu) and thermodynamic and thermophysical properties of actinide nitrides are rather extensively characterized [9].

Silicide fuel U_3Si_2 or 2 phases UN-U_3Si_2 or U_3Si-U_3Si_2

No significant issue is foreseen on fabrication while work to establish an industrial process is still needed, e.g. testing still needs to be done on “industrial” pellets [1].

Silicide fuels have high thermal conductivity and high U density, swelling rate and fission gas release are expected to be less than for UO_2. Fuel centerline temperature is expected to be lower and this would allow use of SiC-SiC composite advanced cladding with silicide fuels.

Acquiring irradiation behavior at intermediate burn-up is needed (e.g. on swelling, fracturation, fission gas release, creep, thermal conductivity).

In case of cladding failure, little knowledge exists on interaction of U_3Si_2 pellet with water and steam. Fuel oxidation is expected to be more exothermic (though reaction enthalpies need to be developed). Long term stability of fuel pellets while operating with fuel rod with breached cladding needs to be investigated.

Due to much higher thermal conductivity, larger margin to melting could be expected compared to UO_2. However, due to lower melting (1665°C) than UO_2, it is hard to assess precisely margin to melt and for severe accident conditions, the fuel is expected to melt at much lower temperature than UO_2.

Additional protection and metrics may be required to operate with U_3Si_2 [3]. The effect of molten fuel interaction with the cladding needs to be assessed to be sure that a coolable geometry can be maintained.

Metallic fuels U-Zr, U-Pu-Zr

Metallic fuels are suitable for Na-cooled reactors but their applicability to LWRs has hardly been discussed [1].

Fuel pins fabrication is established.

Large irradiation swelling is expected. The gap between fuel and cladding inner wall may be filled with a thermal bond material (material to be investigated for LWR, Na for Na-cooled reactors).

Larger margin to melting compared to UO_2 despite a lower melting point (1600°C for U-50%wZr)) than UO_2 due to much higher thermal conductivity.

When irradiated, reacts easily with water and steam and oxidation results in disintegration into pieces and hydrogen production. The contact with water/steam should be prevented for LWR applications.

No report exists on accident conditions with U-50 wt%Zr. A major concern is the H_2 generation and related risks.

From the viewpoint of safety, it is difficult to consider the use of metallic fuels in LWRs irrespective of the cladding material. When reaching severe accident conditions, major concerns are the metal oxidation with large H_2 production and fuel melting.

References

[1] State-of-the Art Report on Light Water Reactor Accident-Tolerant Fuels, NEA report N° 7317, 2018.

[2] Silicon Carbide and its Composites for Nuclear Applications: a Historical Overview, Yutai Katoh, Lance Snead, Journal of Nuclear Materials Diamond Anniversary Review Series (2019).

[3] Technical opinion paper on Applicability of Nuclear Fuel Safety Criteria to Accident Tolerant Fuel Designs, OECD/NEA report, under publication.

[4] Oxidation Kinetics of SiC from 1400°C to 1800°C, Journal of Nuclear Material, Vol. 529, Feb. 2020, Hai Vu Pham, Yuki Nagae, Masaki Kurata, David Bottomley, Kenichiro Furomoto.

[5] Properties and Characteristics of SiC/SiC Composites, Vol. 7, Comprehensive Nuclear Material, 2nd Edition, Rudy Konings and Roger E. Stoller, Editors, Elsevier, 2020.

[6] Chemical reactivity of CVC and CVD SiC with UO_2 at high temperature, Journal of Nuclear Material, Vol. 460, pp. 52, 2015, Chinthaka M. Silva, Yutai Katoh, Stewart L. Voit, Lance L. Snead.

[7] Chemical compatibility between UO_2 fuel and SiC cladding for LWRs. Application to ATF (Accident-Tolerant Fuels), Journal of Nuclear Material, Vol. 487, pp. 380, 2017, James Braun, Christine Guéneau, Thierry Alpettaz, Cédric Sauder, Emmanuelle Brackx, Renaud Domenger, Stéphane Gossé, Fanny Balbaud-Célérier.

[8] High temperature interaction between UO2 and carbon, Application to TRISO particles for very high temperature reactor, Journal of Engineering for Gas Turbine and Power, Vol. 132, pp. 1, 2009, S. Gossé, C. Gueneau, T. Alpettaz, S. Chatain, C. Chatillon.

[9] Thermodynamic and thermophysical properties of Actinide Nitrides, Vol. 7, Comprehensive Nuclear Material, 2nd Edition, Rudy Konings and Roger E. Stoller, Editors, Elsevier, 2020.

Comments by Yu Maruyama

The responses to three out of the following four requests (requests 1 through 3) are described below only for SiC_{fibre}/SiC_{matrix} composite with SiC coating cladding (SiC cladding).

Request 1: For each longer-term ATF, please comment on availability of any test data that can help focus modeling of fuel/cladding behavior under severe accident conditions.

Request 2: For each longer-term ATF, list any potentially unique in-vessel and ex-vessel fuel/cladding degradation and melt progression issues that may alter our current modeling framework.

Request 3: For each longer-term ATF, list potentially unique in-vessel and ex-vessel radionuclide release and transport issues that need to be addressed.

Request 4: For each longer-term ATF, list any potential physiochemical processes and materials interactions issues that may merit additional testing and research.

Request 1:

- Limited test data was found for SiC cladding behavior under severe accident conditions. Although several tests are available for high-temperature behavior and oxidation of SiC cladding in air and steam atmospheres (e.g. [1][2]), it is considered that temperatures of SiC cladding in those tests (< ~2,000 ^{o}C) are not high enough for severe accidents.

[1] V. A. Avinola, et al., "Oxidation at High Temperatures in steam atmosphere and Quench of Silicon Carbide Composites for Nuclear Application", Nucl. Eng. Des., 295, 468-478, 2015.

[2] K. Furumoto, et al., "Out of Pile Test with SiC Cladding Simulating LOCA Conditions", Proc. TopFuel 2018, Prague, Czech Republic, September 30 - October 4, 2018.

Request 2:

- Higher temperature (> ~1,800 ^{o}C) oxidation on the outer surface of SiC cladding to form SiO_2 layer and vaporization of SiO_2 under steam atmosphere or melting of SiO_2, resulting in appearance of new SiC surface
- Formation of H_2 and CO (and CO_2, CH_4) in higher temperature oxidation of SiC cladding
- Formation of SiO_2 on the inner surface of SiC cladding and its melting and interaction with UO_2 fuels
- Structural integrity of SiC cladding against seismic loads
- Influence of sublimation of SiC at higher temperatures (~2,700 ^{o}C) in case that SiC cladding is not covered with SiO_2 layer
- Material interaction (phase diagram) for Si-C-U-O(-Fe) system, influencing on core degradation and melt progression (Fe coming from core support plate, lower head, and so on)
- MCCI with Si/SiO_2 rich molten core (influence on viscosity of molten core)

Request 3:

- Influence of low thermal conductivity of SiC cladding on fuel temperature, resulting in increase of FP amount in fuel/cladding gap
- Influence of SiO2 on Cs chemistry and release (formation of silicates)
- Chemical form of iodine affected by SiO2 due to decrease of Cs amount to form CsI
- Variation in pH of water phase (e.g. suppression pool) in case that sufficient amount of CO2 is formed by the SiC oxidation at higher temperature
- Release of semi-volatile FPs during MCCI with Si/SiO2 rich molten core

Comments by James Metcalf

This discussion of longer-term Accident Tolerant Fuel (ATF) designs focuses on UO_2 as the fuel. This reviewer has no background with respect to the higher density nitride and silicide types or metallic types being proposed.

Advanced fuel designs using non-metallic cladding are being considered for LWR application. The non-metallic cladding design approach that seems most likely to succeed in development is that using SiC. This approach has similarities and ties to the TRISO fuel configuration used for certain HTGRs.

An international patent application [1] was filed in July 2006 for a "Multi-Layered Ceramic Tube for Fuel Containment Barrier and Other Applications in Nuclear and Fossil Power Plants" using SiC. Those submitting the application had initially proposed alumina composite cladding, but it was found to be too permeable. The purposed multi-layer configuration allows for an impermeable inner SiC "monolithic" layer, a strong, more ductile central composite layer with carbon fiber reinforcement, and an outer "protective" layer. This three-layer (or "Triplex") configuration appears to be emerging as the "standard" SiC cladding approach.

The principal advantage of such a design is substantial resistance to oxidation up to a cladding temperature of 1600-1700 °C as compared to the approximate 1450 °C threshold for runaway oxidation for zirconium-based claddings. This oxidation resistance may have a significant impact on the acceptance limits for Emergency Core Cooling System performance, potentially justifying 10CFR50.46 peak clad temperature (PCT) limits greater than the current 2200 °F (1204 °C). The high-temperature strength of the SiC Triplex cladding adds to that justification. It is also expected that the SiC Triplex cladding will be less susceptible to neutron irradiation embrittlement than zirconium-based cladding [2], and (according to [1]) may permit burnups as great as 100 GWD/MTU (although currently 80 GWD/MTU is viewed more realistically). Finally, the SiC Triplex cladding may be less susceptible to damage from localized overheating should limited film boiling occur.

All of these factors lead to the potential for the SiC-clad fuel to be licensed to operate at conditions exceeding those currently permitted for fuel using zirconium-based cladding. On the other hand, the possibility of thermal shock cracking during the reflood phase of emergency core cooling [3] may limit that potential although recent tests of 60 mm tube samples have shown resistance to that kind of damage [4]. Also, the lower thermal conductivity of the SiC Triplex (as compared to zirconium-based cladding) may lead to increased fuel centerline temperatures which will need to be accommodated in the fuel pellet selection (high-density/high-conductivity or annular) and overall licensing limits.

The central question is the severe accident behavior of such fuels. Part of the answer to that question depends on the margin established as part of the fuel licensing. For example, the superior high-temperature strength of the SiC Triplex cladding may permit substantially greater burn-up and pin pressure, but how much of that increased capability will be retained as margin? The same question can be asked about oxidation resistance and the associated 10CFR50.46 PCT. The severe accident behavior of the SiC-clad fuel may be influenced by the way in which the greater capabilities of the SiC cladding are treated in the licensed fuel design. If the fuel licensing limits are relaxed (e.g., greater burn-up being allowed because higher pin pressures can be tolerated by the stronger SiC cladding), then there may not be additional margin for severe accidents. It is difficult to assess the severe accident behavior of the SiC-clad fuel, particularly in the early, in-vessel stages of the accident, without knowing the final, licensed fuel design details.

The increase in capability associated with emergency core cooling performance for SiC Triplex may not be matched to the same degree by increased capability in severe accident behavior. For example, considerable design challenges exist with respect to end plugs for the SiC Triplex rods. A number of leak-tight end-plug joining techniques are being explored [2], but the problem is not a simple one. For example, "In SiC/SiC-based fuel clads, the end-plug joints will likely be the weakest link for possible irradiation creep-induced failure" [5] although a yttrium aluminosilicate glass brazing technique shows promise in terms of strength and leak-tightness [6]. If a satisfactory end-plug design is developed for normal operations, incidents of moderate frequency, and emergency core cooling performance, that design may still represent an inherent weakness for severe accidents as cladding temperature and rod pressures exceed design values. Several abrupt end-plug failure mechanisms are discussed in the literature, and multiple, systemic end-plug failures occurring over a relatively short period of time late in the core heat-up could result in a more rapid activity release from the fuel than the "progressive" release observed with zirconium-based cladding. Moreover, if the SiC Triplex cladding maintains its geometry and basic integrity to a much greater temperature than zirconium-based claddings but control rod materials and designs are not as forgiving, it may be possible that restoration of core cooling and a recovered core could lead to a return to fission power should control materials have relocated from the core region.

In general, the ex-vessel behavior for SiC Triplex cladding would not be expected to differ greatly from that for current fuel with zirconium-based cladding. If anything, the MCCI and hydrogen generation for the SiC Triplex cladding would be expected to be less aggressive than that for current fuel [7].

References:

1. International Patent Application Published under the Patent Cooperation Treaty (PCT) "Multi-Layered Ceramic Tube for Fuel Containment Barrier and Other Applications in Nuclear and Fossil Power Plants", International Application Number PCT/US2005/019789, International Publication Number WO 2006/076039 A2, July 2006.
2. Kim, W., Kim, D., and Park, J.Y., Fabrication and Material Issues for the Application of SiC Composites to LWR Fuel Cladding, Nuclear Engineering and Technology Volume 45, Issue 4, August 2013, Pages 565-572.
3. Lee, Y., McKrell, T.J., and Kazimi, M.S., Thermal Shock Fracture of Silicon Carbide and its Application to LWR Fuel Cladding Performance during Reflood, Nuclear Engineering and Technology, Volume 45, Issue 6, November 2013, Pages 811-820.
4. Lorrette, C., Guilbert, T., Bourlet, F., Sauder, C., Briottet, L., Palancher, H., Bischoff, J., and Pouillier E., Quench Behavior of SiC/SiC Cladding after a High Temperature Ramp under Steam Conditions, 2017 Water Reactor Fuel Performance Meeting, September 2017.

5. Katoh, Y. and Terrani, K.A., Systematic Technology Evaluation Program for SiC/SiC Composite-based Accident-Tolerant LWR Fuel Cladding and Core Structures: Revision 2015, ORNL/TM-2015/454, August 2015.
6. Grosse, M., Avincola, V., Jianu, A., Ahmad, S., and Steinbruck, M., Research on Promising Cladding Materials for Accident Tolerant Fuels at KIT, Accident Tolerant Fuel Concepts for Light Water Reactors, Proceedings of a Technical Meeting held at Oak Ridge National Laboratories, United States of America, IAEA-TECDOC-1797, October 2014.
7. Farmer, M.T., A Scoping Analysis of the Impact of SiC Cladding on Late-Phase Accident Progression Involving Core–Concrete Interaction, ANL/NE-15/39, November 2014.

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APPENDIX B: PANEL MEMBERS

Mohsen Khatib-Rahbar received his Bachelor of Chemical Engineering degree from the University of Minnesota in 1974 and his PhD degree in Nuclear Science and Engineering, School of Applied and Engineering Physics, Cornell University, in 1978. Dr. Khatib-Rahbar specializes in fast and thermal reactor safety, including severe accidents, thermal-hydraulics, reactor dynamics, numerical methods, and probabilistic risk/safety analysis (PRA/PSA). Before forming Energy Research, Inc. in 1989, he was a Senior (visiting) Technical Advisor at the NRC/RES (during 1988 to early 1989). From 1978 to 1988 he managed programs dealing with fast reactor code development, accident analysis, LWR severe accidents, source term research, and PRA, at Brookhaven National Laboratory. He has served on numerous NRC peer review panels including the MELCOR, SCDAP/RELAP5, CONTAIN, FRAPCON/FRAPTRAN, and the source terms for high burn-up and MOX fuels (chairman) panels. Dr. Khatib-Rahbar served as a consultant to the European Commission (EC) from 2005 to 2010, and was a member of the peer review panel for the EC-SARNET program, and the EC advanced reactors research program selection committee.

He has been an invited lecturer at universities (University of Michigan, Cornell University, and University of Maryland), research centers (Centro Atómico Bariloche [Argentina], GRS-Germany, IRSN (workshop at Ecole Normale Supérieure), IAEA (Russia, China, Argentina, and Hungary), Japan, Taiwan, South Korea, and several regulatory organizations (NRC, Swiss Federal Nuclear Safety Inspectorate [ENSI], Swedish Radiation Safety Authority, the Finish Radiation and Nuclear Safety Authority [STUK], and the Consejo de Seguridad Nuclear [CSN]). He has served as a consultant to numerous organizations including IAEA, European Space Agency (ESA), ENSI, U.S. Department of Energy (DOE), Los Alamos National Laboratory, Sandia National Laboratories, Pacific Northwest National Laboratory, among others.

He has also directed and participated in numerous PRAs/PSAs for various reactors (LWRs and VVERs), including the draft NUREG-1150 study for Zion, regulatory PSA studies for all Swiss nuclear power plants (conducted for ENSI), the Level-2 PSAs for all Spanish nuclear power plants (conducted for CSN), Novonoronezh-5 [VVER] PRA in Russia, the Level-2 PSA for GKN units 1 & 2 (conducted for the Germany utility EnBW), and the on-going NRC integrated site PRA, among others. He has published more than 450 government reports, archival articles, and conference papers related to the safety of liquid metal fast reactors and thermal reactors.

He has been directing and managing numerous research and technical support contracts with various NRC offices, DOE, NASA, IAEA, and several European organizations.

Marc Barrachin graduated from the Paris XI University with an MSc in Solid State Physics in 1989. He then joined the Office National des Etudes et Recherches Aerospatiales (ONERA) as researcher in the field of thermodynamics of metallic alloys and defended his PhD at the University of Paris XI in 1990. He joined CEA in 1993 and IRSN in 1996 in the Severe Accident Unit to be deeply involved in the development and qualification of the NUCLEA thermodynamic database and the modelling of material interactions in severe accident conditions for the ASTEC code. In 1998 he was detached at the European Commission's Joint Research Centre in Karlsruhe (Germany) to participate to the post-irradiation examinations of the Phébus FP Program and their interpretation. Until today his research interests and expertise are nuclear fuels and structural materials, with particular emphasis on high-temperature chemistry and thermodynamics. He is currently the deputy head of the Major Accident Department at IRSN. He has published over 60 technical articles in referred journals, presented over 100 conference papers, and contributed to 3 books.

Richard Denning received his Bachelor of Engineering Physics degree from Cornell University in 1963 and PhD degree in Nuclear Engineering Sciences from the University of Florida in 1967. He subsequently joined Battelle, Columbus Laboratories where he held various research positions leading to Senior Research Leader. At Battelle, much of his research was in support of the NRC related to the demonstration of emergency core cooling performance and the modeling of severe accident behavior. Rich coordinated the Level 2 PRA activities in WASH-1400, the first comprehensive examination of nuclear power plant risk. He was a consultant to the NRC's TMI Special Inquiry Group. He participated as a technical expert in the areas of fission product behavior, severe accident loads and containment performance for the NUREG-1150 uncertainty analysis. Dr. Denning authored two chapters of the PRA Procedures Guide, NUREG/CR-2300. The Source Term Code Package (predecessor to MELCOR) was developed under his direction.

Subsequent to the Chernobyl accident, he had responsibility for the management of safety hardware upgrades in a program led by Pacific Northwest National Laboratories for the DOE to improve the safety of former Soviet Union reactors (Russia, Ukraine, Lithuania and Armenia). In 1999 he joined the faculty of the nuclear engineering program at The Ohio State University. In addition to teaching and performing research, he twice served as interim chair of OSU's nuclear program and director of the OSU reactor laboratory. He retired from OSU in 2014.

As a private consultant he has performed research associated with a variety of advanced reactor designs including sodium-cooled fast reactors, gas-cooled high temperature reactors, small modular LWRs, molten salt reactors, and Advanced Test Reactor upgrades. Dr. Denning has served on the DOE's Advisory Committee on Nuclear Facility Safety (now defunct) and the NRC's Advisory Committee on Reactor Safeguards. He is a fellow of the American Nuclear Society.

In his lengthy career, Dr. denning has participated in a number of Phenomena Identification and Ranking Table (PIRT) activities including Molten Salt Reactor PIRT activities for Brookhaven National Laboratory (NRC), Ohio State University (DOE) and Oak Ridge National Laboratory (DOE); and Sodium Cooled Fast Reactors (accident scenario PIRT and source term PIRT) for Argonne National Laboratory (DOE) in addition to the Accident Tolerant Fuel PIRT for ERI (NRC).

Jeff Gabor is a Technical Fellow with JENSEN HUGHES. He has 39 years of experience in severe accident analysis and PRA. Mr. Gabor attended the University of Cincinnati and graduated in 1979 with a Bachelor of Science degree in Nuclear Engineering and a Master of Science degree in Mechanical Engineering in 1980. Recent technical work has included providing EPRI with support for the Containment Performance and Release Reduction (CPRR) rulemaking, co-development of NEI 13-02 for implementation of the Mark I and II hardened vent order, author of the severe accident water addition (SAWA) and water management (SAWM) strategies, update of the EPRI Technical Basis Report (TBR) following the events at Fukushima and IAEA SAMG course instructor. Mr. Gabor served on the peer review team for the NRC's State-of-the-Art Reactor Consequence Analyses (SOARCA).

Randall Gauntt was manager of the Severe Accident Analysis Department at Sandia National Laboratories where he led a team of engineers and software developers that developed the MELCOR severe accident analysis code and the MACCS source term assessment code. His team additionally provided expert applications for the U.S. Nuclear Regulatory Commission. Dr Gauntt has led numerous research studies into severe accident progression such as the State of the Art Reactor Consequence Analysis (SOARCA) and revisions proposed to the NUREG-1465 source terms for high burnup and MOX fuels. Dr. Gauntt also provided in-country support to Japan and the U.S.DOE during the immediate post-accident Fukushima recovery and acted as Program lead role for the OECD-NEA BSAF-2 project, producing a landmark report on the collective analyses of the Fukushima Daiichi accidents.

He has also provided severe accident training for the USNRC as well as the IAEA, most recently with the IAEA Severe Accident Management Guideline training and numerous IAEA international missions on reactor safety and severe accidents. He also supported and authored sections of the IAEA Summary report on the accidents at Fukushima.

Richard Hobbins received an A.B. degree in chemistry from Princeton University in 1960 and a Ph.D. in physical metallurgy from the University of Delaware in 1969. His senior thesis was an experimental determination of the heats of fusion of several alloy systems and his Ph.D. dissertation was the measurement of the self-diffusion of iron in ferrous sulfide single crystals.

Dr. Hobbins has been engaged in fuel development involving fuel irradiation testing and post-irradiation examination (PIE) on a variety of fuels over his career of 24 years at the Idaho National Laboratory and 28 years as an independent consultant based in Jackson Hole, Wyoming. Initially, he worked on understanding the behavior of uranium aluminide dispersion fuel plates, including fuel particle-matrix interaction, fission gas behavior, fuel plate swelling, and cladding oxidation, that resulted in increasing the allowable burnup (fission density) in the Advanced Test Reactor (ATR) by 25%. Studies of the behavior of LWR fuels tested under postulated design-basis accident conditions (LOCA, RIA, Power-Cooling Mismatch) in the Power Burst Facility (PBF) provided understanding of fuel rod damage mechanisms, including cladding ballooning and rupture, cladding oxidation, hydriding and embrittlement, rod fragmentation upon quenching, UO2-Zircaloy chemical interaction, UO2 restructuring, and fission gas bubble agglomeration. Following the TMI-2 accident, testing of LWR fuel rod bundles (one containing control rods) in PBF under severe accident conditions, including measurements of fission product and aerosol release and transport, generated understanding of fuel bundle failure mechanisms, melt progression, molten pool and crust formation, debris bed formation (with and without reflood), and source terms. Further understanding of severe accident phenomena resulted from carefully detailed PIE carried out at INL, directed by Dr. Hobbins, of many samples from the damaged core of the TMI-2 reactor. Dr. Hobbins next became involved with the development of TRISO fuels for high temperature gas-cooled reactors, where irradiation testing was performed at HFIR at ORNL and continues in the ATR, with PIE at both ORNL and INL. Dr. Hobbins also participates in the ongoing development of U-10%Mo monolithic fuel for conversion of high-power research reactors from HEU to LEU fuel with irradiations in ATR and BR2 (in Belgium) and PIE at INL and BR2.

Dr. Hobbins has been a member of numerous expert review committees for DOE and NRC, including the NUREG/CR-1150 Expert Review Panel on In-Vessel Core-Melt Progression Issues and the Kouts panels providing NRC with “Review on Research on Uncertainties in Estimates of Source Terms from Severe Accidents in Nuclear Power Plants”, NUREG/CR-4883.

Dr. Hobbins currently serves as a member of the Technical Coordination Team of the Advanced Gas Reactor Fuel Development and Qualification Program and is Chairman of the US High-Performance Research Reactor Fuel Qualification Independent Technical Review Committee.

Dr. Hobbins has been active in teaching having served as a thesis advisor for two MS degree students, one at the University of Idaho and the second at the University of Washington and one Ph.D. student at MIT. He was a lecturer for 10 years in the Nuclear Power Reactor Safety Course at MIT.

Dr. Hobbins has given some 15 invited lectures, including to professional technical societies in the U.S. and abroad and at six US universities.

Dr. Hobbins is an author or co-author of 143 technical publications including the book, "Transport and Removal of Aerosols in Nuclear Power Plants Following Severe Accidents", R. Sher and R. R. Hobbins, American Nuclear Society, 2011.

Luis E. Herranz has led the research group on Nuclear Safety in CIEMAT since 1999. He graduated in Quantum Chemistry (1986) and did a master on Nuclear Engineering at the Instituto de Estudios de la Energía in 1987. He defended his PhD thesis on Thermal-hydraulics of passive systems in 1996 and in 2016 he became Research Professor on Nuclear Safety, the highest category in Spanish research scale. Among his fields of interest are: Severe Accidents, and Thermal-hydraulics of advanced nuclear systems, Thermo-mechanics of nuclear fuels and safety and power cycles of Generation IV reactors. As a result of his research he has published more than 100 papers in refereed journals and has made more than 200 contributions to international conferences and congresses.

At present, after more than 30 years of professional career, he has been Chairman of the OECD-NEA Working Group on Analysis and Management of Accidents (WGAMA) since 2015 and after several years coordinating the Sub-Technical Area on Source Term within SNETP/NUGENIA, he became Leader of the entire Technical Area 2 on Severe Accidents of SNETP/NUGENIA in 2018. In addition, he is a member of several expert groups of OECD/NEA on Fuel Safety (WGFS) and Reactor Fuel Performance (EGFRP).

Along his career, Dr. Herranz has collaborated with universities and national and international graduate, masters and post-graduate courses. He was Co-Director of the Master on Nuclear Engineering and Applications (MINA) for 10 years, organized by UAM/CIEMAT, Professor at the Department of Mechanical Engineering of UPCO (ICAI) on Heat Transfer and on Energy Technologies for another 10 years. Presently, he is a lecturer in a number of national and international masters and post-graduate courses related to nuclear energy and energy technologies.

Didier Jacquemain received an Engineer degree from the "Ecole Supérieure de Chimie Industrielle de Lyon", Lyon, France in 1988 and a Ph.D. degree in Physical Chemistry from the Weizmann Institute of Science, Rehovot, Israel, in 1992. His research for this degree program involved self-assembling monolayer structures as model systems for the study of crystal nucleation, growth and dissolution. In 1993, he joined the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN), technical support organization of the French Nuclear Safety Authority, where he mostly worked in the Severe Accident Division contributing to the development of the severe accident code ASTEC and to international experimental research programs in the severe accident field. His principal research interests were in fission product, cladding and corium behavior. He also supervised research laboratories in charge of the experimentation on severe accident (in-pile Phebus FP and out-of-pile testing). He then became the deputy head of the Severe Accident Division and then the program manager of the Nuclear Safety Division for severe accident and ageing research. He was also promoted as an Expert then as a Senior Expert in severe accidents. Until the end of 2019, he was the vice-chair of the Nuclear Energy Agency (NEA) Working group on Accident Analysis and Management and the chair of some NEA joint safety research projects related to severe accident and the Fukushima-Daiichi accident analysis. At the beginning of 2020, after 26 years at IRSN, he joined the Division of Nuclear Safety Technology and Regulation of the NEA where he supports the Working Group on Fuel Safety and NEA joint safety research projects. He is the author of about 40 technical publications and coordinated the publication of a book on Nuclear Power Reactor Core Melt Accidents published by EDP Sciences in 2015.

Yu Maruyama joined the Japan Atomic Energy Research Institute (JAERI) (currently Japan Atomic Energy Agency [JAEA]) in 1986. Since then, he has been actively involved in the experimental and analytical studies on severe accidents and thermal-hydraulics of light water reactors (LWRs), including fuel/coolant interaction, debris coolability, molten core/concrete interaction (MCCI), fission product transport behavior and source term. He was at Sandia National Laboratories as a visiting scientist for 18 months in 1990 and participated in MCCI related research activities. He summarized outputs from his studies on severe accidents and received his Ph.D. degree in Engineering from the University of Tsukuba, Japan in 2003.

His current position is Deputy Director General of Nuclear Safety Research Center in Sector of Nuclear Safety Research and Emergency Preparedness of JAEA. The Sector of Nuclear Safety Research and Emergency Preparedness is a technical supporting organization of the Nuclear Regulation Authority (NRA) of Japan. He is a member of two committees of the NRA, the Reactor Safety Examination Committee and the Committee on Accident Analysis of Fukushima Daiichi Nuclear Power Station. In addition, he has participated in, as a member, Working Group on Risk Assessment (WGRISK) of OECD/NEA and several joint projects of OECD/NEA associated with LWR severe accidents and the accident at the Fukushima Daiichi Nuclear Power Station. Specifically, he acted as the chair in the Management Board of the OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) project Phase 2, and is taking a role of the Project Manager for the OECD/NEA Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima Daiichi Nuclear Power Station (ARC-F) project.

James Metcalf received his Bachelor of Science degree in mechanical engineering from Bucknell University in 1967. He then joined Grumman Aircraft working in their Aircraft Integration (flight test) Group. In early 1969, he left Grumman to work in the Design Division of the Portsmouth Naval Shipyard taking graduate courses in naval architecture at the University of New Hampshire (UNH). He transferred to the Nuclear Power Division, Fluid Systems Group as a production engineer on two new-construction nuclear submarines and a fluid decontamination barge. In late 1970 he joined the UNH Engineering Design and Analysis Laboratory (EDAL) and became Project Engineer for the UNH saturation-diving habitat EDALHab used in Project FLARE (FLorida Atlantic Research Expedition). He also studied welding, passing the U.S. Navy test in three positions. In early 1972 he left EDAL after accumulating 30 graduate credits in naval architecture and mechanical engineering. He then joined the Research Vessel *Atlantic Twin* as engineer.

In early 1973, he joined Stone & Webster Engineering Corporation (SWEC) as a fluid systems engineer with responsibility for Emergency Core Cooling, Containment Spray, and Hydrogen Control Systems for the Wisconsin Utilities Project. He later joined the Safety Engineering and Analysis (SEA) staff group with responsibility for various containment thermal-hydraulic analyses. He chaired two subcommittees of the industry's Mark II Containment Owner's Group regarding newly-identified containment hydrodynamic loads. He became supervisor of the SEA group, participated in the NRC's Containment Loads Working Group, and served on the NUREG-1150 Containment Loading Expert Panel.

In the mid-1980's, SWEC became active in source term research. SWEC brought together representatives of three staff groups: SEA, the Chemistry Group, and the Radiation Protection Group to form the Source Term Project with Mr. Metcalf as Lead Engineer. He became a member of the DOE-Advanced Reactor Severe Accident Program (ARSAP) Source Term Expert Group. He led the first "pilot" application of the NUREG-1465 source term (the Design Certification of the ABB-CE System 80+) while the document was still in draft.

He left SWEC in late 1994 to join the start-up Polestar Applied Technology, Inc. He directed ten full applications of the NUREG-1465/RG 1.183 source term and provided peer review and/or support for more than a dozen others. He was a key player in developing a Maximum Hypothetical Accident source term for the ATR. He also consulted with Westinghouse and PBMR Ltd on the Pebble Bed Modular Reactor dose analysis methodology. He was one of two Principal Investigators for an Advanced Light Water Reactor (ALWR) Program emergency planning study (EPRI TR-113509, "Technical Aspects of ALWR Emergency Planning") including preparation of the appendix on the ingestion pathway.

He left Polestar in late 2008 to join AREVA. He was promoted to Technical Consultant, the highest technical level within AREVA, and became a member of their College of Experts. He remained with AREVA until his retirement in 2013 working primarily on the U.S. EPR™. Upon retirement, he formed Bison Nuclear, Inc. as a vehicle for part-time support of the U.S. nuclear industry. His clients have included BWXT, Southern Company, Exelon, and Energy Research, Inc. Mr. Metcalf is a licensed professional engineer in the Commonwealth of Massachusetts.

Dana A. Powers received his Bachelor of Science degree in chemistry from the California Institute of Technology in 1970. He received a Ph.D. degree in Chemistry, Chemical Engineering and Economics in 1975 from the California Institute of Technology. His research for this degree program included magnetic properties of basic iron compounds, catalyst characterization and the rational pricing of innovative products. In 1974, Powers joined Sandia National Laboratories where he worked in the Chemical Metallurgy Division. His principal research interests were in high temperature and aggressive chemical processes. In 1981, he became the supervisor of the Reactor Safety Research Division and conducted analytic and experimental studies of severe reactor accident phenomena in fast reactors and light water reactors. These studies included examinations of core debris interactions with concrete, sodium interactions with structural materials, fission product chemistry under reactor accident conditions, aerosol physics, and high temperature melt interactions with coolants. In 1991, Powers became the acting Manager of the Nuclear Safety Department at Sandia that was involved in the study of fission reactor accident risks and the development of plasma-facing components for fusion reactors. Powers has also worked on the Systems Engineering for recovery and processing of defense nuclear wastes and has developed computer models for predicting worker risks in Department of Energy nuclear facilities. Powers conducted analyses of accident phenomena for safety analyses of the Cassini and New Horizons deep space probes. Dr. Powers was promoted to Senior Scientist at Sandia in 1997. Dr. Powers is the author of 126 technical publications. He is a Fellow of the American Nuclear Society. Dr. Powers retired from Sandia National Laboratories in 2015.

From 1988 to 1991, Dr. Powers served as a member of the Department of Energy's Advisory Committee on Nuclear Facility Safety (ACNFS). In 1994, he was appointed to the Advisory Committee on Reactor Safeguards (ACRS) for the U.S. Nuclear Regulatory Commission. He was Vice Chairman of the ACRS in 1997 and 1998. He was elected Chairman in 1999 and 2000. In 2001, Dr. Powers received the Distinguished Service Award from the US Nuclear Regulatory Commission. He completed his service to ACRS in 2018. Dr. Powers served on committees for the National Research Council involved with the safety of Department of Energy facilities and the nuclear safety of reactors in the former Soviet Union. He has been an instructor for courses on reactor safety and accident management held by the International Atomic Energy Agency in Brazil, Slovenia, China, Korea, South Africa and Pakistan. Dr. Powers received the Theos J. (Tommy) Thompson Award for Nuclear Safety in 2007 from the American Nuclear Society "in recognition of outstanding contributions to the field of nuclear reactor safety".

In 2015, Dr. Powers was elected to the National Academy of Engineering.

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APPENDIX C: OBSERVERS

Subject matter experts from industry observed the panel discussions and provided additional resources as needed in response to specific requests from the panel members pertaining to the ATF design concepts, research and development activities, as well as other related subjects. However, these experts did not deliberate or contribute to the PIRTs, the evaluation of importance and state-of knowledge rankings.

These observers and their affiliations are listed below.

Fran Bolger (EPRI)

Bret Boman (Framatome)

Myles Connor (GNF)

Aladar Csontos (EPRI)

Rob Daum (EPRI)

Ricardo Davis-Zapata (GNF)

Russ Fawcett (GNF)

Kent Halac (GNF)

Luke Hallman (Westinghouse)

Colby Jenson (INL)

Ben Holtzman (NEI)

Zeses Karoutas (Westinghouse)

David Luxat (SNL)

Dave Mitchell (Westinghouse)

Matthew Nudi (EPRI)

Gary Peters (Framatome)

Jesse Phillips (SNL)

Jeff Reed (Framatome)

Raymond Schneider (Westinghouse)

James Scobel (Westinghouse)

Fred Smith (EPRI)

Paul Whiteman (Framatome)

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